BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION

In re: Nuclear Cost Recovery Clause

DOCKET NO. 110009
Submitted for filing: March 1, 2011

REDACTED

**DIRECT TESTIMONY
OF JON FRANKE**

**ON BEHALF OF
PROGRESS ENERGY FLORIDA**

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IN RE: NUCLEAR COST RECOVERY CLAUSE

BY PROGRESS ENERGY FLORIDA

FPSC DOCKET NO. 110009

DIRECT TESTIMONY OF JON FRANKE

I. INTRODUCTION AND QUALIFICATIONS

Q. Please state your name and business address.

A. My name is Jon Franke. My business address is Crystal River Nuclear Plant, 15760 West Power Line Street, Crystal River, Florida 34428.

Q. By whom are you employed and in what capacity?

A. I am employed by Progress Energy Florida, Inc. ("PEF" or the "Company") in the Nuclear Generation Group and serve as Vice President – Crystal River Nuclear Plant.

Q. What are your responsibilities as the Vice President at the Crystal River Nuclear Plant?

A. As Vice President – Crystal River Nuclear Plant, I am responsible for the safe operation of the nuclear generating station. The Plant General Manager, Engineering Manager and Training sections report to me either directly or indirectly. Additionally, I have responsibilities in oversight of major project activities at the station. Through my management team I have about 420 employees that perform the daily work required to operate and maintain the station and provide engineering, training, and other support to the station.

Q. Please summarize your educational background and work experience.

A. I have a Bachelor's degree in Mechanical Engineering from the United States Naval Academy at Annapolis. I have a graduate degree in the same field from the University of Maryland and a Masters of Business Administration from the University of North Carolina at Wilmington.

I have over 20 years of experience in nuclear operations. I received training by the U.S. Navy as a nuclear officer and oversaw the operation and maintenance of a nuclear aircraft carrier propulsion plant during my service. Following my service in the Navy, I was hired by Carolina Power and Light and have been with the company through the formation of Progress Energy. My early assignments involved engineering and operations, including oversight of the daily operation of the Brunswick nuclear plant as a Nuclear Regulatory Commission ("NRC") licensed Senior Reactor Operator. I was the Engineering Manager of that station for three years prior to assignment to Crystal River as the Plant General Manager in 2002. Almost two years ago, in April of 2009, I was promoted to my current position.

II. PURPOSE AND SUMMARY OF TESTIMONY

Q. What is the purpose of your direct testimony?

A. My direct testimony supports the Company's request for cost recovery pursuant to the nuclear cost recovery rule for costs incurred in 2010 for the Crystal River 3 ("CR3") Extended Power Uprate ("EPU") project ("CR3 Uprate") and the Company's request for a prudence determination of the costs incurred for the EPU project in 2010.

I will also provide testimony regarding PEF's 2010 project management, contracting, and oversight controls policies and procedures that are designed to manage project costs and maintain the project schedule and explain why they are reasonable and prudent.

In addition, pursuant to Commission Order No. PSC-11-0095-FOF-EI, issued on February 2, 2011, my testimony will also support a request for a determination of prudence of PEF's CR3 Uprate 2009 costs and 2009 project management, contracting and oversight controls policies and procedures. In this Order, the Florida Public Service Commission ("FPSC" or the "Commission") deferred the determination of the prudence of PEF's 2009 CR3 Uprate costs and 2009 project management, contracting and oversight controls policies and procedures to the 2011 nuclear cost recovery clause ("NCRC") proceeding to investigate the management of the License Amendment Request ("LAR") development process and ascertain the impacts, if any, that it had on actual 2009 CR3 Uprate costs.

Q. Do you have any exhibits to your testimony?

A. Yes. I am sponsoring the following exhibits:

- Exhibit No.___ (JF-1), Work Authorization No. 84 between PEF and AREVA;
- Exhibit No. ___ (JF-2), Review Standard for Extended Power Uprates (RS-001);
- Exhibit No. ___ (JF-3), Excerpts from 2010 Commission Staff Audit Report applicable to the CR3 Uprate Project;

- Exhibit No. ___ (JF-4), EPU Expert Panel November 6, 2009 Management Debrief;
- Exhibit No. ___ (JF-5), Breakdown of 2009 Project Management and License Application Cost;
- Exhibit No. ___ (JF-6), Change Order 23 to Work Authorization No. 84; and
- Exhibit No. ___ (JF-7), Index of 2010 Revised and New Project Management Policies and Procedures.

These exhibits were prepared by me or the Company under my direction and control, or they are documents regularly used by the Company in the normal course of business, and they are true and correct.

I am also sponsoring the cost portions of Schedules T-4, T-4A, T-6, T-6A through T-7B, and Appendix B (2009) and Appendix D (2010) of the Nuclear Filing Requirements ("NFRs") for the 2009 and 2010 CR3 Uprate project costs respectively, which are included as part of the exhibits to Will Garrett's testimony. Schedule T-4 reflects Capacity Cost Recovery Clause ("CCRC") recoverable Operations and Maintenance ("O&M") expenditures for the 2009 and 2010 period. Schedules T-4A reflects CCRC recoverable O&M expenditure variance explanations for the 2009 and 2010 period. Schedule T-6.3 reflects the construction expenditures for the project by category. Schedules T-6A.3 reflect descriptions of the major cost categories of the expenditures and Schedules T-6B.3 reflect explanations for the significant variances between these expenditures and previously filed estimates for 2009 and 2010. Schedules T-7 are lists of the contracts executed in excess of $1.0 million for 2009 and 2010. Schedules T-7A reflect details pertaining to the contracts executed in excess of $1.0 million for 2009 and 2010. Schedules T-7B reflect contracts executed in excess of $250,000, but less than $1.0 million for 2009 and 2010. All of these schedules are true and accurate.

Q. Please summarize your testimony.

A. PEF requests a prudence determination and approval of the recovery of its 2009 and 2010 actual CR3 Uprate project costs. The Company's 2009 actual CR3 Uprate costs were determined to be reasonable and PEF was permitted to recover them as a result of the Commission's order in the 2010 NCRC proceeding. The Commission deferred, however, the determination of the prudence of PEF's 2009 actual CR3 Uprate project costs to the 2011 NCRC proceeding to address issues that were raised with respect to PEF's management of the LAR development process in 2009.

These LAR development costs represent a small fraction of the total CR3 Uprate project costs in 2009. The bulk of those costs were incurred for the work that was performed during the CR3 refueling outage in 2009 and for engineering analyses that supported the LAR. These costs were necessary for CR3 Uprate project work and they were reasonably and prudently incurred. In fact, there was no issue with respect to the reasonableness and prudence of these costs in the 2009 NCRC proceeding. As a result, the Commission should find the balance of the 2009 CR3 Uprate project costs for the Balance of Plant ("BOP") phase work and the engineering analyses for the LAR, including related project management costs, prudent pursuant to the nuclear cost recovery rule.

The LAR development costs in 2009 were also prudent. Problems with the quality and completeness of the initial draft LAR document were corrected by AREVA at no additional cost to PEF and its customers. Further, the cost for more engineering and analytical work in support of the revised LAR document, for example, to meet evolving NRC expectations for EPU LARs, would have been the same or higher if it was performed <u>before</u> rather than <u>after</u> the draft LAR document was reviewed by the expert panel in June-July 2009. The need for more engineering analyses and a new LAR format template due to evolving NRC expectations coincided with the expert panel reviews. As a result, devoting more management and other resources to this work before the expert panel reviews would have resulted in the same or likely an increase in the costs to PEF and its customers for the preparation of the draft LAR document. In summary, PEF and its customers paid no more than they should have paid to complete the EPU LAR draft document consistent with the evolving NRC expectations for EPU LAR submittals.

PEF also incurred CR3 Uprate project costs in 2010 in preparation for phase 3, the EPU phase of the project, during the Company's next re-fueling outage for CR3. The majority of these costs were incurred for necessary engineering analyses for the engineering change packages for the phase 3 work, for long lead item payments, and for related licensing and project management work. PEF took appropriate steps under its project management, contracting, and oversight policies and procedures to ensure that the 2010 CR3 Uprate project costs were reasonable and prudent, and that all of these costs were necessary for completion of the CR3 Uprate project. Accordingly, the Commission should approve PEF's 2010 CR3 Uprate project costs as reasonable and prudent pursuant to the nuclear cost recovery rule.

III. STATUS OF CR3 UPRATE PROJECT.

Q. Please explain the status of the CR3 Uprate project.

A. The CR3 Uprate project is a three-phase project involving the engineering, design, equipment procurement, and equipment installation necessary to generate an additional, estimated 180 MWe of efficient nuclear power at the Company's existing nuclear unit. The work necessary for this project was divided into three phases to be performed during separate, planned re-fueling outages at CR3. The first phase of the work was successfully completed during the 2007 CR3 refueling outage and it was brought online in January, 2008, providing PEF and its customers with an additional 12 MWe of nuclear energy generation.

The second phase of the work, primarily BOP work, was performed during the 2009 CR3 refueling outage and was successfully installed on schedule and at projected costs for the R16 outage with no major issues with the BOP work. When CR3 returns to service the BOP phase work will yield an additional 4 MWe nuclear energy production and supports the final EPU phase to be installed in R17.

PEF is currently performing the engineering and design analyses and is identifying and procuring the material and equipment necessary to complete the third and final phase of the CR3 Uprate. This is called the EPU work phase because, upon completion of the EPU work and NRC approval of the LAR for the power uprate, the Company will be able to increase the power generated at CR3 by an additional 164 MWe. The EPU work will be completed during the next refueling outage for CR3. The next refueling outage is scheduled for April 2013 (because of the extended outage of the CR3 unit). PEF expects the EPU phase of the CR3 Uprate project to be successfully completed in 2013. When phase 3 is complete, the CR3 Uprate will, in total, provide the Company with an estimated, gross additional 180 MWe nuclear energy production.

IV. COSTS INCURRED IN 2009 FOR THE CR3 UPRATE PROJECT.

A. COMMISSION DECISION IN DOCKET NO. 100009-EI.

Q. What did the Commission determine regarding PEF's 2009 CR3 Uprate Project costs and project management?

A. In Order No. PSC-11-0095-FOF-EI in Docket No. 100009-EI, the Commission determined that PEF's 2009 CR3 Uprate costs and 2009 project management, contracting, and oversight control policies and procedures were reasonable and allowed PEF to recover its actual 2009 CR3 Uprate costs. The Commission, however, decided to defer a decision on the prudence of the CR3 Uprate 2009 costs and 2009 project management, contracting, and oversight control policies and procedures to the 2011 NCRC proceeding.

Q. Did the Commission make any finding with respect to the prudence of PEF's 2009 CR3 Uprate project costs in its Order?

A. No. The Commission stated that it was deferring the prudence of PEF's 2009 CR3 Uprate costs and 2009 project management, contracting, and oversight controls to the 2011 NCRC proceeding to allow the parties "the opportunity to fully investigate and present the facts and circumstances surrounding the management of the CR3 Uprate LAR development process." The Commission expressly stated that the reason it was providing the parties this opportunity was to "ascertain the impacts [the CR3 Uprate LAR development process] had on actual 2009 costs, if any." See Order No. PSC-11-0095-FOF-EI, Docket No. 100009-EI, p. 39. The Commission expressed that there was insufficient information in the record in Docket No. 100009-EI to determine whether the Company's 2009 CR3 Uprate costs were negatively affected by PEF's management actions concerning the LAR development. Id. The Commission did not, however, make any finding that any of PEF's actual 2009 CR3 Uprate costs were imprudently incurred. Instead, the Commission deferred the consideration of this limited issue into this year's NCRC docket.

Q. What was the issue with respect to PEF's management of the CR3 Uprate LAR development process?

A. The issue was PEF's management of the preparation of the draft LAR document submittal in 2009. To obtain NRC approval for the CR3 EPU, PEF must submit a LAR to the NRC for review and approval. The LAR document includes a substantial technical report outlining the impact on and changes to plant systems and associated analyses, a set of proposed changes to the plant's technical specifications (detailed operational controls), and various other technical reports. This EPU Technical Report (Attachment 5 of the LAR document) is supported by extensive engineering studies and analyses that are used to develop the information in the Technical Report or included in other parts of the LAR document. Neither the Commission nor any of the parties raised any issues with the engineering work on the studies and analyses that were necessary to obtain the information required for the LAR document. In fact, the only issue last year that was deferred for consideration this year was PEF's management of the preparation of the LAR document itself.

Q. Who was the contractor who prepared the LAR document?

A. PEF contracted with AREVA to perform engineering work and draft portions of the LAR document in Work Authorization 84 in 2007 after obtaining the need determination from the Commission for the CR3 Uprate. See Exhibit No. ___ (JF-1) to my testimony. PEF contracted with AREVA to perform this work because AREVA is the successor to Babcock & Wilcox ("B&W"), the original Nuclear Steam Supply System ("NSSS") vendor of the CR3 plant. AREVA owns licenses to perform certain of the critical analyses as well as other technology rights and is the most experienced and knowledgeable vendor with respect to B&W plants like CR3. AREVA, therefore, was a necessary vendor for the power uprate work at CR3. Other portions of the LAR are the responsibility of PEF.

Q. What was the issue with AREVA's work on the draft LAR document?

A. AREVA commenced work on the necessary engineering analyses and studies to support the LAR document development in 2007 and prepared draft documents for PEF review in 2008 and 2009. PEF and AREVA provided the draft LAR document to an expert panel that PEF established to review the draft application.

The expert panel reviewed the draft document and found that the level of detail included in the draft likely would not meet acceptance review approval at the NRC. The expert panel reached this conclusion because the draft LAR document included poor quality work, incomplete work, and work that did not meet the evolving NRC standards for LAR draft submittals. PEF documented these short-comings in PEF's corrective action system, investigated the circumstances and contributing causes, and prepared an adverse conditions report as required by our Quality Assurance and Project Controls Programs. This report concluded that PEF did not devote adequate management resources to the management of the LAR draft document to ensure that a quality draft was completed as scheduled. PEF's management of the preparation of this draft is the LAR development issue that the Commission deferred to this proceeding.

Q. What questions did the Commission identify with respect to the deferral of the LAR development issue to the 2011 NCRC proceeding?

A. The Commission acknowledged that PEF's management, contracting, and oversight controls policies and procedures on the CR3 Uprate project -- which included the expert panel review of the draft LAR document -- provided the mechanisms for PEF to identify the issues with respect to the draft LAR document and correct them. The Commission, therefore, did not question the quality of the Company's project management and oversight controls mechanisms. Instead, the Commission questioned the timing of PEF's implementation of these mechanisms with respect to PEF's and AREVA's work on the draft LAR document. Order No. PSC-11-0095-FOF-EI, Docket No.

100009-EI, pp. 17-18. Specifically, the Commission questioned whether PEF had a process or the resources to timely redirect work under the AREVA contract due to evolving NRC expectations to potentially avoid performing superseded and, thus, unnecessary work on the draft LAR document. Id. at p. 18. The Commission further questioned whether the need for additional resources was apparent at any time before the expert panel review of the draft LAR document and whether PEF engaged in any mitigation strategies to augment the actual level of resources devoted to the development of the draft LAR document. Id. In summary, the Commission questions: (1) whether PEF's costs for the draft LAR document could have been reduced if PEF devoted additional management resources to the draft LAR document work prior to the first expert panel review; and (2) whether PEF incurred management or other costs for work on the draft LAR document that was unnecessary because of evolving NRC expectations. As a result of these two questions, the Commission concluded that it could not determine from the record last year whether customer costs were negatively affected by PEF's management of the development of the draft LAR document. Id.

B. 2009 CR3 UPRATE DRAFT LAR DOCUMENT DEVELOPMENT PROCESS AND COSTS.

Q. Addressing the first of the two remaining Commission questions, would PEF have reduced its costs for the draft LAR document if PEF had devoted additional resources to manage work on the draft LAR document prior to the first expert panel review?

A. No, because the timing of the additional management resources did not negatively impact customer costs for the draft LAR document. If PEF devoted the same additional management resources to the AREVA and PEF work on the draft LAR document prior to the first expert panel review in June-July 2009 that PEF devoted after this review, the costs to customers for the draft LAR document would have been the same or higher than they were. As a result, PEF's customers were not adversely impacted by PEF's management of the draft LAR document development prior to the first expert panel review.

Q. Why would the cost to customers for the draft LAR document have been the same or higher then they ended up being had PEF dedicated more management resources to the project prior to the first expert panel review in June-July 2009?

A. As I will explain later in more detail, the summary answer to this question is that (1) the problems with the quality and completeness of the initial AREVA draft LAR document were corrected by AREVA at no additional cost to PEF, and (2) the cost for more engineering work in the original draft, and more engineering analyses and a new LAR document template to meet evolving NRC expectations for EPU LAR submittal applications, would have been the same or higher if it was performed <u>before</u> rather than <u>after</u> the draft LAR document was submitted to the expert panel for review, because the need for more engineering analyses and a new LAR document template due to evolving NRC expectations coincided with the expert panel reviews.

Q. What resources were devoted to the draft LAR document preparation?

A. PEF and AREVA devoted experienced licensing and engineering staff to the draft LAR document preparation. PEF and AREVA LAR licensing and engineering staff were experienced nuclear professionals with prior licensing and engineering experience on nuclear projects. PEF and AREVA's licensing and engineering staff had prior experience on similar, technical issues on other nuclear projects they had worked on that were reviewed or approved consistent with NRC standards.

PEF also used the NRC's express guidelines for EPU projects to develop the CR3 Uprate draft LAR document. These guidelines were published by the NRC in 2003 explicitly for EPU projects. The guidelines are called the "Review Standard for Extended Power Uprates (RS-001)." These guidelines outline the NRC's expectations for an EPU project. <u>See</u> Exhibit No. ___ (JF-2) to my testimony.

In addition, the first plant to submit an EPU LAR based on RS-001 was the R.E. Ginna nuclear power plant in 2005. Ginna received a Safety Evaluation report ("SE") from the NRC in 2006 indicating the NRC's approval of the Ginna LAR document submittal. The NRC indicated to PEF that the Ginna LAR document provided a good model for future EPU LAR document submittals. As a result of the NRC staff's tacit approval of the Ginna LAR as a model EPU LAR and the NRC's ultimate approval of the Ginna EPU LAR submittal, PEF used the Ginna LAR document as a model for the CR3 EPU LAR document. AREVA was provided both the RS-001 and Ginna LAR document as a guide for the development of the CR3 EPU LAR inputs.

As a result, PEF was reasonably assured that the quality of its draft LAR document resources at PEF and AREVA were adequate. These resources included licensing and engineering staff at both PEF and AREVA that had prior experience on nuclear projects subject to review or approval under NRC standards. These resources were also provided the existing NRC and industry guidelines at that time for the preparation of EPU LAR documents.

Q. Did PEF take other steps to manage the work on the CR3 EPU draft LAR document?

A. Yes. The management of the AREVA and PEF work on the CR3 EPU draft LAR document was subject to the same project management and oversight controls and procedures that applied to all PEF major capital projects like the CR3 Uprate project. These are the same project management and oversight controls and procedures that I described in my testimony in the 2009 NCRC proceeding that were determined by the Commission to be reasonable and prudent in Order No. PSC-09-0783-FOF-EI in Docket No. 090009-EI. See Order No. PSC-09-0783-FOF-EI, p. 24 ("Therefore, we find that during 2008, PEF's project management, contracting, and oversight controls were reasonable and prudent for the CR3 Uprate project."). I provided testimony updating these same project management, contracting, and oversight controls in my testimony in the 2010 NCRC proceeding and I provide similar testimony below in this NCRC proceeding.

These project management, contracting, and oversight controls include mechanisms and procedures for the management of vendor work like AREVA on the CR3 Uprate project, and they provide for the auditing of that management and work as a matter of nuclear oversight by personnel independent from the Uprate project managers. As a result, PEF did not just accept the draft LAR document or its own management of that work. Both were subject to further review in accordance with project management, contracting, and oversight mechanisms PEF established for the CR3 Uprate project.

This further review included the expert panel review followed by the investigation of the expert panel findings that produced the adverse condition report and recommendations. As I explained in my rebuttal testimony last year, PEF established a team of industry experts, including outside experts, to critically review the draft LAR for completeness, correctness, clarity and conformance with industry best practices at the time of their review. They were specifically directed to review and critique the draft LAR document. PEF's decision to have an expert panel review the LAR document was consistent with best industry practices and, therefore, prudent project management.

Similarly, the subsequent, independent adverse conditions report regarding the quality of PEF's management of the work on the draft LAR document also reflected critical oversight and prudent project management, as I also explained last year. These independent external and internal critical reviews were and are necessary to any prudent project management process. The fact that PEF had them in place and that they were implemented on the CR3 Uprate project with respect to the LAR document development process demonstrates PEF's prudent project management, contracting, and oversight controls over this aspect of the work on the Uprate project in particular, and on the CR3 Uprate project in general. This review process ensured that the LAR document work was reviewed, that any work that was not up to par was corrected, and that a LAR document that was sufficient and consistent with the standards at the time of the review for LAR document submittals was prepared. The Commission audit staff agreed in the 2010 NCRC proceeding that the expert panel performed an important role in insuring a complete and thorough LAR submittal to the NRC. See Exhibit No. ___ (JF-3), 2010 Staff Audit Report, p. 40. The Commission further agreed that this process represented a prudent project management process. Order No. PSC-11-0095-FOF-EI, pp. 17-18.

Q. Did the expert panel include members with experience on other EPU projects?

A. Yes. All members of the expert panel that conducted the first review of the draft LAR document in June-July 2009 and in subsequent time periods extending into early 2010 were experienced nuclear professionals with prior NRC EPU or similar NRC experience. One of the two Progress Energy members of the expert panel reviews of the CR3 Uprate EPU draft LAR document had worked on the EPU at the Progress Energy Brunswick nuclear power plant. The other Progress Energy member of the expert panel was previous to his employment with Progress Energy the licensing lead on the Waterford nuclear power plant EPU. The others had contributed to other EPUs or other similar projects.

Q. Would placing those Progress Energy members of the expert panel reviews on the CR3 Uprate project to help manage the draft LAR development work

prior to the first expert panel review have resulted in lower costs to customers?

A. No. As I testified earlier, if these expert panel members were employed by PEF to manage the draft LAR document prior to the expert panel review --- or if PEF employed the additional resources PEF added after the expert panel review to manage the draft LAR document prior to the expert panel review --- the costs to PEF's customers would have been the same or higher than they were. Changing the quality or quantity of the management resources on the draft LAR document prior to the expert panel review would not have reduced the customer costs for the LAR document. In fact, adding to the quality or quantity of the management of the development of the draft LAR document prior to the expert panel reviews likely would have increased the costs to customers of the ultimate LAR document that was prepared and ready for NRC submittal consistent with the known NRC standards for EPU LAR submittals at the time of the expert panel reviews.

Q. Why would increasing the quality or quantity of the LAR management resources for the work on the draft LAR document prior to the expert panel reviews have likely resulted in the same or increased costs for the draft LAR document?

A. There are several reasons, all related to the nature of the expert panel and adverse condition report recommendations for improving the draft LAR document that was provided to the expert panel to review. First, one issue with the initial draft LAR document was the quality and completeness of the work consistent with the RS-001 and Ginna model guidelines at the time this draft was prepared. In essence, the expert panel determined that the draft LAR document lacked the level of detail or quality necessary to meet these guidelines in certain sections of the draft LAR document. The expert panel also determined that certain sections were incomplete in that they did not incorporate sufficient engineering information or analysis when compared to the pre-existing guidelines that AREVA and PEF used for the preparation of the EPU draft LAR document. This did not mean that this engineering analysis had not been done or the engineering information was not developed by AREVA. Rather, it meant that AREVA had not included this engineering information or analyses in the draft LAR document consistent with the RS-001 and Ginna EPU LAR submittal model guidelines.

As I testified in last-year's proceeding, the correction of these issues with the draft LAR document by AREVA was performed at AREVA's cost. PEF paid AREVA no additional funds to re-do or re-write unchanged LAR document sections to improve the quality or add additional engineering information or analysis that had already been performed for the LAR document consistent with the RS-001 and Ginna EPU LAR submittal model guidelines. As I further testified last year, PEF, therefore, addressed the expert panel and internal adverse condition report recommendations concerning the quality and completeness of the draft LAR document at no additional cost to customers. Because these issues were corrected at no additional cost to customers anyway, adding more and more experienced licensing and engineering staff to manage the draft LAR document work prior to the expert panel review would not have resulted in any change in the cost to customers. Adding these additional resources to manage the draft LAR document work prior to the expert panel review may have led to the correction of these issues earlier, but having these corrections earlier than June-July 2009 would not have saved PEF's customers any money. Said another way, if PEF had deployed the expert panel the first day work began to review each page of the draft LAR document as soon as it was completed, PEF's customers would not have saved one penny over what they are being charged now. Granted, this hypothetical process would have likely avoided AREVA having to do re-work on the LAR document to correct issues, but as I have made clear, all costs of the correction work that AREVA had to do did not cause any delay costs and that work was performed solely at AREVA's cost.

Q. How do you know PEF did not pay AREVA twice for the same work on the quality and completeness of the draft LAR document?

A. AREVA was paid a flat fee of [redacted] to write the draft LAR document sections reviewed by the expert panel in June-July 2009 using the RS-001 and Ginna LAR draft submittal document as guides. These payments are identified at line items 8.28, 8.28 revised, and Note 2 in the "Deliverable Section" on page 4 of Work Authorization No. 84 between PEF and AREVA for design and engineering work to support the CR3 Uprate project, including the work to support the LAR document and draft the LAR document. These line items demonstrate that AREVA was paid [redacted] for LAR inputs and draft comment responses and that AREVA was paid another [redacted] when the LAR document was ready for submittal to the NRC. See Exhibit No. ___ (JF-1) to my testimony. That is all AREVA will be paid for the initial draft LAR document work. After the expert panel issued its report and recommendations, AREVA corrected their quality and completeness issues and re-wrote the LAR document sections at AREVA's own cost.

As I testified last year, PEF met with AREVA prior to AREVA submitting each invoice under Work Authorization No. 84 and related changes orders. That is why the costs for work to re-write portions of the LAR document do not show up in subsequent AREVA invoices to PEF. AREVA did, however, correct the portions of the draft LAR document that did not meet the quality or completeness standards of the RS-001 and Ginna LAR model guidelines. Subsequent expert panel reviews confirmed that these corrections were made. See, e.g., Exhibit No. ___ (JF-4) to my testimony. PEF, however, paid no additional compensation for that work. The Commission agreed that, in reviewing the record in the NCRC proceeding last year, it found no instance where work performed by AREVA under the original contract or change orders was shown to be unneeded. See Order No. PSC-11-0095-FOF-EI, Docket No. 100009-EI, p. 18.

Q. **You testified there were other reasons adding additional management resources to the LAR document development prior to the expert panel review would have resulted in the same or higher costs to customers, what are they?**

A. One of the expert panel criticisms was that PEF and AREVA had not done enough work on the draft LAR document at the time it was submitted to the expert panel for review in June-July 2009. The expert panel concluded that the draft LAR document did not provide sufficient design and engineering detail for the EPU equipment and modifications to plant operations for the NRC to accept the LAR document for review and approval. This required additional design and engineering detail that had not yet been performed. In sum, then, the expert panel found that PEF had not incurred the costs and performed the work necessary to that point to prepare a draft LAR document that was capable of NRC acceptance review.

PEF accepted the expert panel comments and recommendations and invested the money in the design, engineering, and analysis work necessary to complete the design and engineering work for the EPU modifications described in the draft LAR document to improve the LAR submittal to the NRC. Subsequent expert panel reviews confirmed that PEF adequately addressed these expert panel comments and recommendations to prepare a LAR document that was acceptable for NRC acceptance review. See, e.g., Exhibit No. ___ (JF-4) to my testimony. The Commission audit staff further noted last year that PEF had to expend resources to strengthen the EPU LAR submittal to meet the NRC's acceptance review. See Exhibit No. ___ (JF-3), 2010 Staff Audit Report, p. 59.

The fact that PEF had to expend additional costs on additional resources for the design, engineering, and procurement work to enhance the EPU draft LAR document demonstrates that these costs would have been incurred in any event to produce the same end LAR document work product. In other words, the same work required to produce a draft LAR document that met NRC submittal requirements at the time the expert panel concluded they were met in early 2010 would have to be done and the costs for that work incurred no matter when that work took place. The timing of the costs for this additional, required design and engineering work recommended by the expert panel was different but the work and costs required were essentially the same.

Q. Did the difference in the timing of these costs cause customers to bear more costs than they otherwise would have?

A. No, because the timing difference was a matter of months -- from early 2009 to mid-to-late 2009 and early 2010 -- and the costs for engineering work that might have been performed at the beginning compared to the costs of the same work that was performed at the end of this relatively short period of time did not change. In other words, PEF moved quickly to perform the additional, required engineering work in response to the recommendations of the expert panel and, as a result, there were no delays in performing this additional work that resulted in additional costs to customers.

Q. In what way could adding additional management resources to the LAR document development prior to the expert panel review have resulted in higher costs to customers?

A. Another expert panel recommendation was that the draft LAR document needed to conform to evolving NRC expectations for LAR document submittals. These evolving NRC expectations resulted in additional engineering work as well as the development of a new template for the LAR document. In fact, PEF and AREVA developed an updated template for its LAR document, after the expert panel review in June-July 2009, consistent with the evolving NRC standards to replace the Ginna model as a guideline for LAR document submittals to the NRC.

If PEF had added more and more experienced licensing and engineering staff to the management of work to develop the draft LAR document prior to the expert panel review, then, the cost to perform the additional engineering work and develop a new LAR submittal template to meet evolving NRC expectations would have been higher for PEF's customers. The reason is that any additional management resources employed on the draft LAR document work <u>before</u> the first expert panel review would have followed the same LAR submittal format for the scope of work for the draft LAR document that PEF followed. The RS-001 and Ginna LAR submittal were the only guidelines available at that time. The generalized need for additional engineering work for and work on a new template for the LAR document submittal due to evolving NRC expectations was not apparent at that time. As a result, had PEF prematurely added more and/or more experienced licensing and engineering staff to manage the draft LAR document work prior to the first expert panel review PEF would have incurred more costs for these additional management resources, but PEF could not and would not have improved the draft LAR document submittal to meet evolving NRC expectations for EPU LAR submittals that were not known at that time.

Q. When did PEF first become aware of evolving NRC expectations for EPU LAR submittals that indicated PEF needed additional engineering work and a new template for its EPU LAR submittal document?

A. PEF became aware that the Ginna EPU LAR submittal may no longer be an acceptable model for LAR document submittals in mid to late 2009, after reviewing the NRC's Requests for Additional Information ("RAIs") to Point

Beach regarding its EPU LAR document submittal to the NRC. Point Beach submitted its EPU LAR document to the NRC for acceptance review in April 2009. The NRC commenced RAIs to Point Beach regarding its EPU LAR document submittal in late April to early May 2009, and these RAIs continued in number and complexity over the course of the summer and into the fall of 2009. In October of 2009 the NRC issued a non-acceptance to Point Beach stating that Point Beach should supplement its LAR application or withdraw it.

Q. Why was the Point Beach EPU LAR submittal an indication of evolving NRC requirements for EPU LAR application submittals to the NRC?

A. Point Beach is a Westinghouse Pressurized Water Reactor ("PWR"). CR3 is also a PWR. The Point Beach PWR is a similar PWR design to the Ginna nuclear power plant. As a result, the Point Beach LAR submittal was similar to the Ginna LAR submittal that had been accepted and approved by the NRC and that PEF was using as a model for its CR3 EPU LAR submittal with the tacit approval of the NRC staff.

Point Beach submitted its EPU LAR document to the NRC for acceptance review and, by the end of April to early May 2009, Point Beach began receiving numerous, detailed RAIs from the NRC. During the period from mid to late 2009, it became increasingly clear from reviewing the NRC RAIs and the Point Beach RAI responses that NRC expectations for EPU LAR submittals had changed. The scope, extent, and nature of the Point Beach RAIs suggested a more stringent NRC review of EPU LAR submittals and more stringent requirements for the design and engineering analyses to be performed for or submitted in the EPU LAR document.

The level of detailed engineering analysis and information required by the Point Beach EPU LAR NRC RAIs demonstrated that the level of detail included in the Ginna LAR document submittal was inadequate for NRC acceptance review and, therefore, that the Ginna LAR document was no longer an acceptable model for EPU LAR submittals. This view was confirmed when the NRC wrote Point Beach in October 2009 indicating that it was not accepting the Point Beach LAR submittal --- which was based on the same Ginna LAR submittal that PEF was using as a model --- and that Point Beach could either supplement its LAR submittal or withdraw it.

Q. Did the Point Beach EPU LAR review experience at the NRC mean that PEF incurred costs that were unnecessary for its draft EPU LAR submittal document?

A. No. All PEF and AREVA work on the draft EPU LAR document for the CR3 Uprate prior to the Point Beach LAR RAIs and RAI responses was necessary to meet the known LAR document submittal requirements at that time. PEF and AREVA were using the RS-001 and Ginna LAR submittal document that had been approved by the NRC and indicated by NRC staff as a useful guide for EPU LAR submittals. Prior to the Point Beach EPU LAR RAIs and RAI responses, then, PEF and AREVA work on the CR3 EPU LAR draft document was based on the known and knowable LAR submittal requirements at that time.

It is important to remember too that one of the expert panel's criticisms of the PEF and AREVA EPU LAR document work was that PEF did not have sufficient resources devoted to this work prior to the expert panel review. PEF likewise reached a similar conclusion that more licensing and engineering staff was needed for the LAR submittal work following a routine Nuclear Oversight Organization audit in early 2009. This audit was performed in accordance with PEF's project management and oversight policies and procedures. This audit recommended employing additional licensing and engineering staff for the CR3 EPU LAR submittal. In response to this audit recommendation, PEF revised the LAR submittal schedule to provide for a later September 2009 target date for LAR submittal to the NRC to give PEF more time to develop the LAR submittal. PEF also firmed up the expert panel review commencing in the summer of 2009 to determine where the draft LAR document stood in terms of the engineering analysis and work that was being done and that needed to be done for the EPU LAR submittal. As a result, PEF did not have excess resources devoted to the EPU LAR document work before PEF became aware of the evolving NRC requirements for such submittal documents.

Finally, the most significant impact of the evolving and more stringent NRC requirements for the LAR submittal following the Point Beach RAIs and RAI responses was the identification of more – not less – engineering and other, related work that needed to be performed for the LAR submittal document to meet NRC acceptance review. PEF had to add licensing and engineering resources and incur more costs to meet the evolving NRC standards for EPU LAR documents. If these additional LAR submittal requirements were known earlier --

which was not the case -- the same costs would have been incurred to meet these requirements earlier that were incurred to meet them after they were known to PEF.

Q. Did PEF's management of the LAR document development process in 2009 actually benefit PEF's customers?

A. Yes. PEF took a reasonable and prudent approach to the need for additional LAR application licensing and engineering support in 2009 when it determined that such additional resources were necessary. PEF planned to employ its existing oversight plan involving the expert panel review of the draft EPU LAR document to identify where additional licensing and engineering resources were necessary to support the LAR document development process. The most efficient way to add additional resources to support the development of the LAR submittal was to first evaluate the draft AREVA work product and determine what resources were needed and where the resources were needed. The expert panel reviews provided PEF with an existing independent mechanism to timely review the draft EPU LAR document and determine where to most efficiently add licensing and engineering resources.

The timing of the expert panel review also provided PEF the opportunity to efficiently incorporate the evolving NRC standards for EPU LAR submittals into the CR3 draft EPU LAR submittal. The expert panel reviews coincided with the development of the additional NRC requirements for EPU LARs as a result of the Point Beach EPU LAR NRC RAIs and RAI responses from mid 2009 to late October 2009. As a result, the expert panel reviews provided a timely mechanism to incorporate these evolving NRC standards into the expert panel recommendations for the CR3 EPU LAR submittal application. This is in fact what the expert panel did when they made their recommendations to PEF to improve the CR3 EPU LAR draft document.

As a result, the LAR development work in 2009 for the CR3 EPU LAR draft document was valuable. The addition of licensing and engineering resources to assist with the draft LAR submittal work and incorporate evolving NRC standards for such applications as recommended by the expert panel reviews ultimately provided a necessary and efficient mechanism to add resources and meet the changing EPU LAR submittal requirements at that time. This saved PEF's customers money. PEF's customers paid no more than what they should have paid for the EPU LAR draft document work by AREVA and PEF in 2009.

Q. Should PEF have been aware of the evolving NRC standards prior to the NRC review of the Point Beach EPU LAR submittal?

A. No. As I previously testified, the first indication of a general increase in the NRC requirements for EPU LAR submittals was the Point Beach NRC RAIs and RAI responses because this LAR submittal was also for a PWR plant and the EPU LAR document was similar to the Ginna LAR submittal document that PEF was using as a guide for its EPU LAR document. There was one earlier relevant EPU LAR submittal to the NRC, but this submittal was for a Boiling Water Reactor ("BWR"), not a PWR plant, and it involved a more limited set of LAR issues. Thus, this earlier EPU LAR submittal did not indicate a more general increase in the requirements for EPU LAR submittals to the NRC. Nevertheless, PEF followed this EPU LAR submittal at the NRC and incorporated applicable changes in its EPU draft LAR work where such changes were indicated.

Q. Can you explain what this prior EPU LAR submittal was and how PEF responded to it with its own EPU draft LAR document?

A. Yes. In late June 2008 the Monticello nuclear power plant withdrew its EPU LAR submittal in response to a NRC letter indicating that the EPU LAR was inadequate with respect to three, limited EPU issues. As I indicated earlier, the Monticello nuclear power plant is a BWR, which is a very different design from the PWR plants like CR3. Also, Monticello had followed a different format for its EPU LAR submittal. Monticello used a series of General Electric Topical Reports on EPU as the basis for its EPU LAR. These reports were very different from the Ginna EPU LAR document that PEF was using at the time as its model application and contained far less information than the Ginna EPU LAR document. These circumstances made the Monticello EPU LAR submittal distinguishable from the CR3 EPU LAR draft document and, thus, of limited value to PEF as PEF worked with AREVA on the draft CR3 EPU LAR document.

As a result of these differences, the narrow issues identified by the NRC for determining the Monticello EPU LAR submittal inadequate were of limited application to the CR3 EPU LAR submittal document. For example, the NRC questioned the Steam Dryer Integrity in the Monticello EPU LAR submittal. PWRs like CR3 do not have Steam Dryers. Nevertheless, based on the NRC's expressed interest in this area, PEF did determine if it had any applicability to the

CR3 EPU LAR and, as a result, PEF established a more thorough vibration monitoring plan for its EPU LAR than PEF originally planned.

Also, the NRC had questioned the instrumentation setpoints for the Monticello BWR EPU in the LAR submittal document. This issue did not impact the CR3 EPU LAR document because no instrumentation setpoints were being changed at that time.

The third and final EPU LAR issue identified by the NRC with respect to the Monticello EPU LAR was the Environmental Qualification ("EQ"). EQ is a standard design requirement for all nuclear power plants. It basically requires the licensee to demonstrate through testing that components required to mitigate accidents can withstand a normal operating environment and the environment during related accidents. Previously, all utility EPU LAR submittals planned to complete their EQ work after LAR submittal but prior to EPU implementation. Starting with the Monticello EPU LAR submittal, the NRC required that EQ impacts be completed prior to LAR submittal and incorporated into the EPU LAR submittal. As a result of the NRC response to the EQ analysis in the Monticello EPU LAR submittal, and confirmation of the changing expectation through direct communications with the NRC, PEF moved up its EQ work activities and included the EQ analysis in its draft LAR document. The expert panel reviews raised no concerns with PEF's EQ analysis in the draft EPU LAR document for CR3.

As I have demonstrated here, PEF responded to the limited NRC issues identified with the very different Monticello EPU and EPU LAR submittal when a response was in any way indicated for the CR3 EPU LAR submittal. There was no indication from this limited NRC review of the Monticello EPU LAR, however, that the NRC was going to generally expand the requirements for all EPU LAR submittals to include substantially more and different engineering information and analysis in the EPU LAR document.

Q. Did PEF have appropriate processes in place to identify and timely redirect EPU LAR licensing and engineering work if the redirection of that work was necessary?

A. Yes. PEF did have a proper process in place to timely redirect work if redirection of the EPU LAR work was in fact necessary. This process included (i) AREVA and PEF participation in the industry working group that developed the EPU guidance, (ii) AREVA and PEF involvement in the review of other EPU LAR submittals, (iii) PEF participation with other utility EPU licensing managers in discussions regarding EPU LAR submittal requirements, and (iv) PEF involvement in following and participating in NRC public reviews of EPU LAR submittals. PEF's decision to re-direct EQ work for its EPU LAR submittal following information gleaned from this process regarding the Monticello EPU LAR submittal that I just described demonstrates that PEF's process to redirect EPU LAR licensing and engineering efforts worked when the information indicating the need to re-direct work existed.

Q. Did the issues with AREVA's work on the draft LAR document have an impact on the CR3 Uprate project?

A. Yes, but no impact that cost PEF's customers more money than they would otherwise have paid. When PEF had to add additional licensing and engineering resources for the draft LAR work PEF did miss its targeted September 2009 EPU LAR submittal date. Missing this target for the EPU LAR submittal to the NRC, however, did not result in any additional cost to ratepayers and it did not delay EPU LAR approval beyond the planned implementation of the CR3 power uprate at that time.

The September 2009 EPU LAR submittal deadline was an aggressive target date for the CR3 EPU LAR submittal. This target date provided approximately twenty-four months to complete the EPU LAR submittal review process prior to the then-planned November 2011 refueling outage for the phase 3 EPU work. The NRC licensing process performance indicators is based on fourteen months from the date the EPU LAR is submitted to the NRC for approval to obtain a SE report from the NRC approving the EPU LAR submittal. PEF therefore had approximately ten months of float in the EPU LAR schedule based on the initial September 2009 target EPU LAR submittal date. As a result, even if the November 2011 refueling outage at CR3 was not extended, PEF had until August 2010 to submit its EPU LAR document for NRC approval in time for the original planned completion of the phase 3 EPU work and the power uprate. The Commission audit staff agreed that PEF had substantial float in its initial EPU LAR submittal schedule. See Exhibit No. ___ (JF-3), 2010 Staff Audit Report, p. 38.

Although PEF had established a September 2009 target for submittal of the EPU LAR document to the NRC, the priority for PEF engineering resources in

2009 was correctly focused on the R16 outage work in the fall of 2009. That was why the original EPU LAR submittal date was June 2009 (before the R16 outage), then shifted to September 2009 (right before the R16 outage), before being moved to March 2010 (after the scheduled R16 outage). These shifts in the targeted EPU LAR submittal date provided the additional time necessary to address the expert panel and adverse conditions report recommendations while ensuring that the additional licensing and engineering work required to comply with these recommendations did not interfere with the engineering work for the R16 outage. Because PEF had substantial float in its EPU LAR schedule, PEF was able to perform the additional licensing and engineering work recommended by the expert panel and internal audit for the EPU LAR document without adversely impacting the timing for submittal of the EPU LAR to the NRC for approval. The delay in preparation of the EPU LAR document for submittal to the NRC from June 2009 to March 2010 therefore did not result in any additional ratepayer costs.

Q. What project management costs were incurred in 2009 for the AREVA draft LAR document development management and oversight?

A. PEF incurred $110,261 in project management costs for direct oversight of the EPU LAR document development process from January to June 2009, as referenced in the footnotes in Exhibit No. __ (JF-5) to my testimony.

Q. Did Commission Order No. PSC-11-0095-FOF-EI identify a broader range of potential cost impacts due to the draft LAR document development process?

A. Yes. The Commission noted that potential cost impacts, if any, due to PEF's management of the LAR document development process range from $0 to over $40 million, with the higher range based solely on intervenor arguments -- not the evidence -- in the 2010 NCRC proceeding, as I explain in more detail below. Order No. PSC-11-0095-FOF-EI, Docket No. 100009-EI, p. 39. The Commission further determined that all of PEF's 2009 CR3 Uprate costs were reasonable, but deferred finding that they were prudent to the 2011 NCRC proceeding. The total CR3 Uprate 2009 costs included capital costs of $118,140,493, operation and maintenance ("O&M") expenses of $821,773, carrying charges of $14,351,595, and a base revenue requirement of $396,018. Thus the vast majority of the CR3 Uprate 2009 costs have nothing to do with the CR3 Uprate LAR development process that was identified as an issue for the 2011 NCRC proceeding by the Commission. The Commission recognized this, noting "that beyond those items identified in the issues above [the LAR development process issues], no other concerns where [sic] identified with the 2009 final costs and final true-up amount for the CR3 Uprate project." Order No. PSC-11-0095-FOF-EI, Docket No. 100009-EI, p. 39. In fact, as I testified earlier, the actual 2009 PEF management costs for the development of the draft EPU LAR document is a very small fraction of the potential cost range identified by the Commission based on the parties' arguments in the 2010 NCRC proceeding and an even smaller percentage of the total 2009 CR3 Uprate costs.

Q. Where did the identified range of potential cost impacts come from in Order No. PSC-11-0095-FOF-EI?

A. The low end of the range, or $0, represents PEF's testimony and document evidence in the 2010 NCRC proceeding that the issues with the draft EPU LAR document that led to the expert panel and adverse conditions report recommendations resulted in no additional costs to PEF's customers. For the reasons I explained earlier, PEF's customers paid no more than they should have paid for the development of the EPU LAR document for submittal to the NRC.

The high end of the range, $40 million, was first identified in the Office of Public Counsel's ("OPC") post-hearing brief in the 2010 NCRC proceeding. It simply represents the total of all Licensing Application and Project Management costs incurred in 2009 for the entire CR3 Uprate project. PEF obviously did not spend all of its 2009 Licensing Application and Project Management costs on the management of the EPU LAR draft document by AREVA.

PEF incurred [redacted] for EPU LAR document development work by AREVA and a total of $132,059 for PEF internal project management oversight and associated PE Company labor costs for the LAR document development work from January to June of 2009. To illustrate this point, I have provided a line-by-line breakdown of the $40 million in License Application and Project Management actual costs incurred in 2009 in Exhibit No. ____ (JF-5) to my testimony. As this exhibit demonstrates, the vast majority of the 2009 CR3 Uprate License Application and Project Management costs were not incurred in connection with the EPU LAR document development. Most of the $20,016,839 in License Application costs in 2009 for the CR3 Uprate project was incurred for the extensive engineering work under WA 84 including the fuels analysis, safety analysis, and system and program reviews. Most of the $21,154,156 in 2009 CR3

Uprate Project Management costs were incurred for the management oversight of the BOP phase engineering work that was completed during the R16 CR3 refueling outage in 2009.

Q. Can you please explain how you calculated the costs for the 2009 AREVA and PEF LAR document development work in Exhibit No. ___ (JF-5)?

A. Yes. The NSSS contract WA 84 with AREVA was approved on February 27, 2007. WA 84 provided the basis for all engineering and accident safety analysis work required to support the EPU LAR submittal and the work required to draft the EPU LAR document. WA 84 is attached as Exhibit No. ___ (JF-1) to my testimony. As I testified earlier, line items 8.28, 8.28 revised, and Note 2 totaling [redacted] show the fixed price amount PEF contracted with AREVA to draft sections of the initial EPU LAR document. Two additional items are added to the 2009 EPU LAR development costs. The first work item was change order number 23 to WA 84, executed in December 2009 (Rev 1) in the amount of [redacted]. See Exhibit No. ___ (JF-6) to my testimony. Of this total change order amount, [redacted] was incurred in 2009 for LAR document development activities (the remaining change order amount was incurred in 2010). See Exhibit No. ___(JF-5). The second work item is the inclusion of internal Company labor for the management oversight and work on the LAR document development in 2009. As I indicated previously, for January to June 2009, PEF costs for the management oversight and associated PEF labor costs for the draft LAR document development process was $132,059.

Q. Why was Change Order Number 23 necessary for the LAR development work?

A. This change order was for the work necessary to re-write the original EPU LAR document in 2009 to comply with the revised EPU LAR template to meet evolving industry standards and NRC expectations. Change Order Number 23 expressly states the LAR re-write effort was to re-write sections of the LAR document to comply with the revised template and other new scope activities. It was not payment to AREVA to re-write poorly drafted LAR sections identified in the June-July 2009 expert panel review of the AREVA draft LAR document. Indeed, Change Order Number 23 further expressly states that the expert panel "comment incorporation is considered part of the original scope of activities and is not included in this scope of work." See Exhibit No. ___ (JF-6) to my testimony. Under this Change Order, AREVA was entitled to more compensation for more work to conform the LAR document to meet the additional EPU LAR document requirements based on evolving industry standards and NRC expectations. These evolving industry and NRC expectations for EPU LAR documents required more detailed engineering analyses and documentation of that analyses to be included in the EPU LAR document than was the previous industry standard.

C. CR3 UPRATE 2009 CAPITAL AND OTHER COSTS.

Q. What were the other 2009 CR3 Uprate project costs?

A. PEF incurred costs in 2009 related to the last two phases -- the BOP and EPU phases -- of the CR3 Uprate project. In fact, PEF completed the work necessary to install the BOP phase work during the R16 refueling outage in 2009. The total capital expenditures for 2009, gross of joint owner billing and exclusive of carrying cost, were $118,140,493. These costs were incurred for (i) license application costs, (ii) project management costs, (iii) permitting costs, (iv) on-site construction facility costs, (v) power block engineering, procurement and related construction costs, and (vi) non-power block engineering, procurement, and related construction costs. Schedule T-6A attached as Exhibit No. ___ (WG-1) to Mr. Garrett's testimony provides further details regarding these costs.

PEF also incurred O&M expenses in 2009 for the CR3 Uprate project in the amount of $821,773. These expenses were incurred for Administrative and General ("A&G") expenses and other, related project overhead costs in 2009. Schedules T-4 and T-4A attached as Exhibit No. ___ (WG-1) to Mr. Garrett's testimony provides further details regarding these costs.

Q. You mentioned that PEF completed the installation of the BOP phase work in 2009, did this work account for a significant portion of PEF's 2009 CR3 Uprate project costs?

A. Yes. The Company incurred $71,243,000 for Power Block Engineering, Procurement, and related construction cost items. Most of the costs incurred in this category in 2009 were associated with the R16 outage scope of work for the BOP phase of the Uprate project, which included:

- Installation of 4 Moisture Separator Reheaters
- Installation of 2 Secondary Cooling Heat Exchangers
- Installation of 2 Moisture Separator Reheater Shell Side Drain Heat Exchangers
- Installation of 4 Turbine Bypass Valves and Mufflers

- Modification of the Turbine Generator Electrical Output Bus Duct Cooling System
- Installation of 2 Condensate Heaters
- Replacement of the Turbine Generator Exciter
- Turbine Generator Electrical Stator Rewind
- Rescaled Integrated Control System
- Installation of a fiber optic "backbone" to interface with new turbine monitoring equipment
- Installation of 2 Secondary Cooling Pumps and Motors
- Installation of a Turbine Lube Oil Cooler
- Installation of Heater Drain Valves
- Plant computer updates
- Facilities

PEF's 2009 Power Block Engineering and Procurement costs were necessary for the timely completion of the CR3 Uprate BOP phase work during the 2009 R16 refueling outage. PEF also incurred On-Site Construction Facilities costs of $1,203,995 for the labor costs associated with mobilizing and maintaining temporary facilities to house the extra personnel needed to implement the BOP phase of the CR3 Uprate project during the R16 refueling outage in 2009. Finally, PEF incurred Project Management costs related to the management of the BOP phase work performed during the R16 outage in 2009.

As I testified earlier, the BOP phase work was performed during the 2009 R16 refueling outage on schedule and on budget. These costs were reviewed by the parties and Commission audit staff in the 2010 NCRC proceeding and no issues with respect to these costs were identified. Commission audit staff reviewed and verified that the project remained on schedule with minor variances and confirmed that no major issues were identified during the work. <u>See</u> Exhibit No. ___ (JF-3), 2010 Staff Audit Report, p. 37. The Commission staff auditors further confirmed that the BOP phase work during the R16 outage was completed as scheduled and at projected costs for the R16 outage. <u>Id.</u>

Q. What were the Company's 2009 Project Management costs for the CR3 Uprate project?

A. The Company incurred Project Management costs of $21,154,156. As reflected on Exhibit No. __ (JF-5), the Company's Project Management costs included the following Project Management activities performed for the CR3 Uprate project, but primarily for the BOP phase work that was completed in 2009:

(1) project administration, including project instructions, staffing, roles and responsibilities, and interface with accounting, finance, and senior management;

(2) contract administration, including status and review of project requisitions, purchase orders, and invoices, contract compliance, and contract expense reviews;

(3) project controls, including schedule maintenance and milestones, cost estimation, tracking and reporting, risk management, and work scope control;

(4) project management, including project plans, project governance and oversight, task plans, task monitoring plans, lessons learned, and task item completions;

(5) project training, including the uprate project training program, training of personnel in accordance with the training program, and maintaining training records; and

(6) management of CR3 Uprate licensing work.

Each activity was conducted under the Company's project management and oversight controls policies and procedures. Such costs were necessary to ensure that the BOP scope of work was adequate to achieve the uprate project objectives, that the engineering and construction labor, material, and equipment provided by

PEF or outside vendors for the BOP phase of the project was available when needed at a reasonable cost, and that the project schedule was maintained.

Q. Did PEF incur other CR3 Uprate project costs in 2009?

A. Yes. PEF incurred permitting costs of $882,003 for permitting needs for 2009. These costs were necessary for the permitting activities to support the BOP phase construction work in 2009. PEF also incurred permitting costs to develop the environmental report associated with the EPU LAR and to obtain environmental permits for facilities and other construction activities.

PEF also incurred Non-Power Block Engineering, Procurement and related construction costs in the amount of $3,640,540 in 2009. These costs were associated with the studies the Company completed on the effects of the increased heat at the Point of Discharge ("POD"). These costs were necessary for the project because PEF will not be able to complete the full uprate without analyzing and accommodating the higher water temperature in the discharge canal.

Q. Were there variances between the actual 2009 CR3 Uprate project capital expenditures and PEF's actual/estimated costs for 2009?

A. Yes. As I explained in the 2010 NCRC proceeding, there were variances in actual 2009 costs compared to PEF's actual/estimated capital expenditures in 2009. These variances were primarily driven by the additional EPU LAR preparation costs that I previously described and permitting activity costs, but they were partially off-set by Non-Power Block Engineering work, as described in more detail below. To begin with though, there was a change in the assignment of costs on the CR3 Uprate project that impacted the variances in 2009 actual and actual/estimated project costs. As I explained last year, at the time of the Actual/Estimated filing in 2010, the assigning of costs into the filing categories was based on general assumptions that were determined to be the most appropriate guidelines to assign costs to the categories at that time. As the project has matured and a more detailed task structure has been implemented, the Company established a new and more accurate method for assigning costs to the various categories. This change did not affect the total project cost or the total capital expenditure variance, but did affect variances within individual categories, particularly in Project Management, Power Block Engineering, and On-Site Construction Facilities. With this background, the variances include:

License Application:

The 2009 License Application capital expenditures on the T-6 schedule were $20,016,839 with a total estimate of $16,277,263, resulting in a variance of $3,739,576. This variance is attributable to additional, more detailed information for the EPU LAR document, the necessary acceleration of engineering work scope to create the information, and the creation and implementation of a revised template for the EPU LAR document, for all the reasons I explained earlier in my testimony.

Project Management:

Project Management capital expenditures were $21,154,156. The original estimate was $39,666,137, resulting in a variance of ($18,511,981). This variance is primarily driven by the new method for assigning costs to cost categories that I described above.

Permitting:

Permitting capital expenditures were $882,003. The original estimate was $151,463, resulting in a variance of $730,540. The variance was primarily due to the need for environmental permits to support the project and temporary facilities that were not originally anticipated in the projected facilities plan.

On-Site Construction Facilities:

On-Site Construction Facilities capital expenditures were $1,203,955. The original estimate was $4,223,713, resulting in a variance of ($3,019,758). This variance is primarily driven by actuals only capturing the labor to manage facilities work due to the change in method for assigning costs to the categories as described above. All costs to mobilize, rent, and maintain the temporary facilities needed to house the additional personnel for the EPU Phase 2 implementation that were estimated for this category are being appropriately captured in the Power Block Engineering category.

Power Block Engineering:

Power Block Engineering capital expenditures were $71,243,000. The original estimate was $52,560,048, resulting in a variance of $18,682,952. This variance is also primarily driven by the new method for assigning costs to categories that I explained above.

Non-Power Block Engineering, Procurement, etc.:

Non-Power Block Engineering, Procurement, etc. capital expenditures were $3,640,540. The original estimate was $4,658,928, resulting in a variance of ($1,018,388). This variance is primarily driven by scope and schedule changes associated with POD/Cooling Tower work. As the engineering evaluation of the New Forced Draft Cooling Tower progressed, the location of the tower was changed. The new location relieved the project of relocating a warehouse, thus reducing the project cost for 2009. Also in 2009, the recirculation line work that was scheduled to start was put on hold for further evaluation and rescheduled.

Q. Were all of PEF's 2009 CR3 Uprate project costs reasonably and prudently incurred?

A. Yes. For all the reasons I have provided, PEF reasonably and prudently incurred the 2009 CR3 Uprate project costs. These costs were necessary for completion of the BOP phase work for the CR3 Uprate project in 2009 and the continuation of work for the EPU phase during the Company's next planned refueling outage for CR3. The Commission, in fact, reviewed the Company's 2009 CR3 Uprate project costs and determined that they were reasonably incurred in Order No. PSC-11-0095-FOF-EI. The Commission also determined, based on the record evidence in the 2010 NCRC proceeding, that no concerns other than the management of the LAR development process in 2009 were identified for PEF's 2009 CR3 Uprate project costs. Order No. PSC-11-0095-FOF-EI, Docket No. 100009-EI, p. 39. As I have explained earlier, PEF's management of the development of the EPU LAR document by AREVA in 2009 did not result in any unnecessary or duplicative costs and, therefore, did not result in additional costs to PEF's customers that should not have been incurred. All of PEF's 2009 CR3 Uprate project costs were reasonably and prudently incurred.

V. ACTUAL COSTS INCURRED IN 2010 FOR THE CR3 UPRATE PROJECT.

Q. What costs did PEF incur for the CR3 Uprate project in 2010?

A. PEF incurred construction costs related to the last phase of the CR3 Uprate project in 2010. The total capital expenditures for 2010, gross of joint owner billing and exclusive of carrying cost, were $45,544,492. These costs cover (i) license application, (ii) project management, (iii) permitting, (iv) on-site construction facilities, (v) power block engineering, procurement and related construction, and (vi) non-power block engineering, procurement, and related construction. Schedule T-6 in Exhibit No. ___ (WG-3) to Mr. Garrett's testimony further details these costs.

Q. Please describe the total License Application costs incurred and explain why the Company incurred them.

A. The License Application costs reflected on the T-6.3 Schedule were $3.3M. These costs were incurred for continued work on the EPU LAR submittal document into 2010 to address evolving industry and NRC expectations for EPU LAR submittals as I explained earlier. These activities included fuels analysis, safety analysis and system and program reviews.

Q. Please describe the total Project Management costs incurred and explain why the Company incurred them.

A. The Company incurred Project Management costs of $5.2M. The Company's Project Management costs include the following Project Management activities for the EPU phase of the CR3 Uprate project:

(1) project administration, including project instructions, staffing, roles and responsibilities, and interface with accounting, finance, and senior management;
(2) contract administration, including status and review of project requisitions, purchase orders, and invoices, contract compliance, and contract expense reviews;
(3) project controls, including schedule maintenance and milestones, cost estimation, tracking and reporting, risk management, and work scope control;
(4) project management, including project plans, project governance and oversight, task plans, task monitoring plans, lessons learned, and task item completions; and
(5) overall management of CR3 Uprate licensing work.

Each activity was conducted under the Company's project management and oversight control policies and procedures.

Q. Please describe the total Permitting costs incurred and explain why the Company incurred them.

A. Permitting costs incurred were ($10,607) for permitting needs for 2010. This credit in actual costs incurred is due to an adjustment to reclassify costs from environmental permitting to licensing.

Q. Please describe the total On-Site Construction Facilities costs incurred and explain why the Company incurred them.

A. On-Site Construction Facilities costs incurred were $164,692. This represents the labor costs associated with demobilizing and maintaining temporary facilities to house the extra personnel needed to implement Phase 3 of the CR3 Uprate project.

Q. Please describe the total costs incurred for the Power Block Engineering, Procurement and related construction cost items and explain why the Company incurred them.

A. The Company incurred $32.7M for Power Block Engineering, Procurement, and related construction cost items. The majority of the costs incurred in this category in 2010 were associated with the preparation of design changes for the phase 3 scope and for procurement of long lead time equipment. Over thirty Engineering Change (EC) packages were initiated and progressed to different levels in support of the Phase 3 scope. These 2010 Power Block Engineering and Procurement costs were necessary for the implementation of the CR3 Uprate work during the next refueling outage.

Q. Please describe the total costs incurred for the Non-Power Block Engineering, Procurement and related construction cost items and explain why the Company incurred them.

A. These costs total $4.2M. The majority of the costs incurred in this category in 2010 were associated with the POD portion of the EPU Project. The major contributors to these costs were payments for Helper Cooling Tower (HCT) design, a discharge canal cooling study, and delivered HCT equipment. These costs are necessary for the project because PEF will not be able to complete the full uprate without mitigation of the higher water temperature in the discharge canal.

Q. How did actual capital expenditures for January 2010 through December 2010 compare to PEF's actual/estimated costs for 2010?

A. PEF's actual capital expenditures for the CR3 Uprate project in 2010 were lower than PEF's actual/estimated costs for 2010 by $20.8M. This variance is primarily due to the deferral of funds for Uprate project work from 2010 to 2011 and 2012. These deferrals were required as a result of rescheduling milestone payments and the extended outage at CR3. I will explain the reasons for the major (more than $1.0 million) variances below:

License Application:

The 2010 License Application capital expenditures on the T-6 schedule were $3.3M with a total estimate of $1.6M, resulting in a variance of $1.7M. This variance is primarily attributable to invoice timing on carry-over AREVA work-scope under Change Order No. 23. This additional, more detailed engineering information was needed for the EPU LAR draft document to meet evolving NRC requirements for EPU LAR submittals, which is a continuation of the cost variance for the 2009 License Application cost I described earlier. It is also noted that some AREVA Change Orders were inappropriately charged to License Application and a re-classification to move these charges to Engineering will be made in the first quarter of 2011.

Project Management:

Project Management capital expenditures were $5.2M. The original estimate was $9.7M, resulting in a variance of ($4.6M). This variance is due to the reallocation of project management resources during 2010 because of the extended CR3 outage and the associated delay of the R17 refueling outage.

Power Block Engineering:

Power Block Engineering, Procurement and related construction costs capital expenditures were $32.7M. The original estimate was $43.0M, resulting in a variance of ($10.3M). This variance is due to the deferral of contract milestone payments from 2010 to 2011 because of the extended CR3 outage and the associated delay of the R17 refueling outage.

Non-Power Block Engineering:

Non-Power Block Engineering capital expenditures were $4.2M. The original estimate was $11.3M, resulting in a variance of ($7.0M). This variance is primarily driven by scope and schedule changes associated with POD/Cooling Tower work, which is a result of the extended CR3 R16 outage and the associated delay of the R17 refueling outage, as well as pending and emerging environmental regulations which would impact the fossil units at Crystal River. The POD project has been placed on hold until such time that the impact of these changes can be appropriately assessed.

Q. Did PEF incur O&M costs in 2010 for the CR3 Uprate project?

A. Yes. PEF incurred necessary O&M costs to support the continuation of the CR3 Uprate project work in 2010. These O&M costs are identified and included in Schedule T-4 in Exhibit No. ___ (WG-3) to Mr. Garrett's testimony.

Q. How did actual O&M expenditures for January 2010 through December 2010 compare with PEF's actual/estimated O&M expenditures for 2010?

A. Schedule T-4A, Line 15, on Exhibit No. ___ (WG-3) to Mr. Garrett's testimony shows that total O&M costs were $1,000,181, or $345,037 less than estimated. By cost category, major cost variances between PEF's actual/estimated and actual 2010 CR3 Uprate O&M costs are as follows:

Legal: O&M expenditures for Legal were $281,116 or $139,871, less than projected. This variance was due to lower than anticipated outside legal counsel services.

Nuclear Generation: O&M expenditures for Nuclear Generation were $538,893 or $236,025 less than projected. This variance was primarily attributable to identification of less obsolete inventory than expected.

Q. Were all of PEF's 2010 CR3 Uprate project costs reasonably and prudently incurred?

A. Yes. PEF reasonably and prudently incurred the 2010 CR3 Uprate project costs. These costs were necessary for the continuation of work for the EPU phase during the Company's next planned refueling outage for CR3, as I have just described.

All of PEF's 2010 CR3 Uprate project costs were reasonably and prudently incurred.

VI. ALL COSTS INCLUDED FOR THE CR3 UPRATE ARE "SEPARATE AND APART FROM" THOSE COSTS NECESSARY TO RELIABLY OPERATE CR3 DURING ITS REMAINING LIFE

Q. Are the CR3 Uprate project costs included in this NCRC docket for recovery separate and apart from those that the Company would have incurred to operate CR3 during the extended life of the plant?

A. Yes, PEF has only included for recovery in this proceeding those costs that were incurred solely for the CR3 Uprate project. In other words, the Company only included project costs that would not have been incurred but for the CR3 Uprate project.

VII. PROJECT MANAGEMENT AND COST CONTROL OVERSIGHT.

Q. Does the Company have project management and cost control oversight policies and procedures that it utilizes for its capital projects?

A. Yes. The Company has several project management and cost oversight control policies and procedures that it employs for all of its capital projects fleet-wide. Each year the Company will review and revise its policies and procedures as necessary. These policies and procedures, existing, new, and as revised, that are applicable to the 2009 CR3 Uprate project were produced to the Commission Staff auditors in Data Request One ("DR -1") for 2010 in Bates range 11PMA-DR1CR3-9A-000001 through 000330 and 11PMA-DR1Levy-12A-000001 through 002553 and 11PMA-DR1Levy-12B-000001 through 000080 (a few of the CR3 procedures listed were inadvertently omitted from this production and will be produced in a supplemental DR production). See Index of 2010 Revised and New Project Management Policies and Procedures attached hereto as Exhibit No. ___ (JF-7) to my testimony. These are the same Company-wide capital project policies and procedures that are applicable to the Levy Nuclear Project ("LNP") and that have been approved as reasonable and prudent three straight years for the LNP and in the 2008 NCRC proceeding, Docket No. 090009-EI, for the CR3 Uprate project.

Q. Can you please provide an overview of the Company's 2009 and 2010 project management and cost oversight policies and procedures?

A. Yes. The CR3 Uprate project is being undertaken by the Company consistent with its Project Management Manual, which the Company has used to manage capital projects since early in this decade. Additionally, because the CR3 Uprate project is a major capital project for the Company, the project must comply with the Company's Major Capital Projects – Integrated Project Plan ("IPP") procedure that was issued in 2009. The CR3 Uprate is also being undertaken by the Company consistent with the project standards established and implemented by Progress Energy's Project Management Center of Excellence organization ("PMCoE"). These standards are based on principles from the internationally recognized Project Management Institute Project Management Body of Knowledge and establish a standardized project management approach that spans tools, templates and processes; training and qualification programs; and adoption of best practices. The CR3 Uprate project was also approved in accordance with the Company's Project Evaluation and Authorization Process. This evaluation and project authorization process has been in place at the Company for many years. The CR3 Uprate project is subject to the Progress Energy Project Governance Policy, which also has been in place for many years.

Along with these general procedures the Company utilizes several specific project management and cost oversight procedures as listed on Exhibit No. __ (JF-7) to my testimony. These procedures are reviewed on a continuous basis for changing business conditions and to incorporate improvements, lessons learned, and clarifications.

Q. Have PEF's project management and cost oversight controls substantially changed between 2009 and 2010?

A. No, however the Company continuously reviews and revises policies and procedures based on changing conditions, lessons learned, and best industry practices.

Q. Does the Company have any policies or procedures in place to assess and mitigate project risks?

A. Yes. The Company routinely assesses various project risks and assigns each risk with a probability of occurrence and level of importance in terms of effect on project schedule and cost using its CR3 Uprate Risk Register. The risk register facilitates monitoring and controlling risk by providing a tool to document risk probability, impact, response plans, ownership, triggers, and expected monetary value. It also provides the ability to document risk mitigation opportunities for the project.

Q. Are employees involved in the CR3 Uprate Project trained in the Company's project management and cost control policies and procedures?

A. Yes, they are. PEF's project management team for the CR3 Uprate project has been trained in these Company policies. There are also formal Project Manager qualification requirements for projects of various sizes as well as for other roles within the Project Team (Designated Representative, Field Lead, etc.).

Q. What polices and procedures does the Company utilize to ensure that its selection and management of outside vendors is reasonable and prudent?

A. First, a requisition is created in the Passport Contracts module for the purchase of services. The requisition is reviewed by the appropriate Contract Specialist in Corporate Services, or field personnel on the CR3 Uprate project, to ensure sufficient data has been provided to process the contract requisition. The Contract Specialist prepares the appropriate contract document from pre-approved contract templates in accordance with the requirements stated on the contract requisition.

The contract requisition then goes through the bidding or finalization process. Once the contract is ready to be executed, it is approved online by the appropriate levels of the approval matrix pursuant to the Approval Level Policy and a contract is created. Contract invoices are received by the CR3 Uprate project managers. The invoices are validated by the project managers and

Payment Authorizations approving payment of the contract invoices are entered and approved in the Contracts module of the Passport system.

When selecting vendors for the CR3 Uprate project, PEF utilizes bidding procedures through a Request For Proposal ("RFP") process when possible for the particular services or materials needed to ensure that the chosen vendors provide the best value for PEF's customers. When an RFP cannot be used, PEF ensures that the contracts with the sole source vendors contain reasonable and prudent contract terms with adequate pricing provisions (including fixed price and/or firm price, escalated according to indexes, where possible). When deciding to use a sole source vendor, PEF must provide a sole source justification for not doing an RFP for the particular work.

Q. Does the Company verify that the Company's project management and cost control policies and procedures are followed?

A. Yes, it does. PEF uses internal audits to verify that its program management and oversight controls are being implemented and are effective in practice. During the first quarter of 2009, an audit was conducted to review financial controls related to the Nuclear Cost Recovery Rule for the CR3 Uprate project. These processes were found effective. On July 2, 2009, an audit was completed regarding the effectiveness of project management and cost management for the CR3 Uprate project. Areas needing improvement were risk management, earned value analysis and KPI reporting, and improvements have since been made consistent with the audit recommendations. The Financial Controls Internal Auditing Program, financial status reporting, and information and process management were found effective. As a result of the audit, observations and recommendations were provided for improvement. The Company implemented the recommended action plans, and action items are completed.

For 2010, the Company conducted the Florida Nuclear Plant Cost Recovery Rule Compliance Monitoring Review on March 12, 2010. These processes were found effective.

Q. Are the Company's project management and cost control policies and procedures on the CR3 Uprate project reasonable and prudent?

A. Yes, they are. These project management policies and procedures reflect the collective experience and knowledge of the Company across the fleet. These policies and procedures have also been tested by the Company on other capital projects. Any lessons learned from those projects have been incorporated in the current policies and procedures. Moreover, in 2009 the project management policies and procedures in place successfully recognized issues with project resources and realigned work on the EPU LAR submittal exactly as intended. We believe, therefore, that our project management policies and procedures are consistent with best practices for capital project management in the industry and are reasonable and prudent.

VIII. <u>CONCLUSION.</u>

Q. Will the CR3 Uprate project be successfully completed at a reasonable and prudent cost to the Company and its customers?

A. Yes. As I explained above, we are well on the way to successfully completing the CR3 Uprate project and achieving the power uprate benefits, albeit on a longer schedule than originally anticipated due to the extended CR3 outage. There is no indication that the CR3 Uprate project cannot be successfully completed and NRC approval of the EPU LAR obtained at a reasonable cost to PEF and its customers.

Q.. Does this conclude your testimony?

A. Yes, it does.

EXHIBIT JF-1
REDACTED IN ITS ENTIRETY

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW STANDARD FOR EXTENDED POWER UPRATES

APPROVED BY: /RA/
L. Marsh, Director
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Office of Nuclear Reactor Regulation

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RS-001, Revision 0
DECEMBER 2003

11PMA-DR1CR3-2S-000001

OFFICE OF NUCLEAR REACTOR REGULATION

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RS-001, Revision 0
DECEMBER 2003

11PMA-DR1CR3-2S-000002

RS-001, "Review Standard for Extended Power Uprates"

RS-001 CHANGE HISTORY			
Date	**Description of Changes**	**Method Used to Announce & Distribute**	**Training**
12/2002	Initial issuance for interim use and public comment	• *Federal Register* • Power Uprate Web site • ADAMS	None
12/2003	Issuance of RS-001, Revision 0 ● Revised the Purpose section to add paragraphs 3 thru 5 to reflect changes resulting from public comments ● Reformatted matrices in Section 2 and SE inserts in Section 3 to reflect NRR reorganization ● Moved the guidance for independent calculations from the individual matrices to Item (6) in Section 2.1 ● Revised the matrix for Mechanical and Civil Engineering to add a note to highlight experience with dryer failures at Quad Cities 2 and identify focus of staff review in relation to this experience ● Revised the matrix for Reactor Systems to: • delete the reference to the ISCOR computer code and spectrum of breaks analyzed in the note on BWR reviews • delete the bullet regarding hot leg streaming • deleted the note regarding overfill analyses for SGTR • delete reference to Item II.K.3.5 of NUREG-0737 in the note on LOCA reviews • combine and reformat the notes on ATWS reviews ● Added two notes to the matrix in Section 2 for Health Physics to identify obsolete guidance ● Revised the regulatory evaluation sections of the SE inserts for Health Physics in Section 3 to add the statement that the NRC also considers the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary ● Revised the regulatory evaluation sections of the SE inserts for Human Performance in Section 3 to add a reference to GL 82-33 ● Revised the conclusion sections of the SE inserts for Power Ascension and Testing Plan in Section 3 to make them consistent with the wording in proposed SRP Section 14.2.1 ● Made miscellaneous editorial changes ● Revised RS-001 to incorporate OGC comments, which included using the term "design bases" in lieu of "licensing bases" and modifying the regulatory evaluation sections of the template safety evaluations to incorporate additional wording from pertinent regulations such as the General Design Criteria.	• *Federal Register* • Power Uprate Web site • ADAMS	Training sessions for staff in DLPM, DRIP, DIPM, DSSA & DE

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11PMA-DR1CR3-2S-000004

PURPOSE

The purpose of this review standard is to provide guidance for the Nuclear Regulatory Commission (NRC) staff's review of extended power uprate (EPU) applications to enhance consistency, quality, and completeness of reviews.

This review standard also informs licensees of the guidance documents the staff uses when reviewing EPU applications. These documents provide acceptance criteria for the areas of review. This should allow licensees to prepare EPU applications that are complete with respect to the areas that are within the staff's scope of review. To further improve the efficiency of the staff's review of EPU applications, licensees are encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants.

Use of this review standard should not undermine the NRC's longstanding topical report review and approval process. If a licensee references an NRC-approved topical report for an area covered by this review standard, the staff will review the application only to ensure that the licensee is applying the topical report under conditions for which the topical report was approved, using appropriate plant-specific inputs.

The staff will review plants against their design bases. Licensees are encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should help the staff identify areas where the criteria and/or guidance in the review standard does not apply to the plant under review. The staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard.

In addition to this review standard, the NRC maintains a Web site on power uprates at http://www.nrc.gov/reactors/operating/licensing/power-uprates.html. Some of the material on this Web site includes:

- the status of completed, ongoing, and expected power uprate reviews
- general guidance related to power uprates
- references to publicly available correspondence related to reviews of recently completed power uprates (including licensees' responses to NRC staff requests for additional information, as well as NRC staff safety evaluations)

11PMA-DR1CR3-2S-000005

BACKGROUND

Facility operating licenses and technical specifications specify the maximum power level at which commercial nuclear power plants may be operated. NRC approval is required for any changes to facility operating licenses or technical specifications. The process for making changes to facility operating licenses and technical specifications is governed by Title 10 of the *Code of Federal Regulations*, Part 50.

The process of increasing the licensed power level at a commercial nuclear power plant is called a "power uprate." Power uprates are categorized based on the magnitude of the power increase and the methods used to achieve the increase. Measurement uncertainty recapture power uprates result in power level increases that are less than 2 percent and are achieved by implementing enhanced techniques for calculating reactor power. Stretch power uprates typically result in power level increases that are up to 7 percent and do not generally involve major plant modifications. EPUs result in power level increases that are greater than stretch power uprates and usually require significant modifications to major plant equipment. The NRC has approved EPUs for increases as high as 20 percent. This review standard is applicable to EPUs.

This review standard establishes standardized review guidance and acceptance criteria for the staff's reviews of EPU applications to enhance the consistency, quality, and completeness of reviews. It serves as a tool for the staff's use when processing EPU applications in that it provides detailed references to various NRC documents containing information related to the specific areas of review.

This review standard also informs licensees of the guidance documents the staff will use when reviewing EPU applications. This will help licensees prepare EPU applications that address those topics necessary for a complete application. By addressing the areas in the review standard, a licensee could prepare and submit a more complete application and thus minimize the staff's need for requests for additional information (RAIs). This would improve the efficiency of the staff's reviews.

The development of this review standard included an evaluation of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), to determine the applicability and adequacy of the various SRP sections to the review of EPU applications and development/revision of guidance, as necessary. During this evaluation, the staff considered the versions of the SRP sections identified in the matrices in Section 2 of this review standard. To determine the need for guidance beyond that in the SRP, the staff reviewed: (1) safety evaluations for previously approved power uprates, (2) previously approved topical reports for EPUs, (3) various reports related to lessons learned from the Maine Yankee experience (e.g., Report of the Maine Yankee Lessons Learned Task Group, dated December 1996), and (4) generic communications. The staff also considered feedback from internal and external stakeholders. In addition, the staff reviewed RAIs issued for recent EPU applications to ensure that the review standard adequately addresses areas where repeat RAIs have been issued.

The staff reviewed NRC procedural guidance documents to identify those applicable to processing EPU applications. The review of these documents also included consideration of the recommendations in various reports related to the Maine Yankee experience and the feedback received from internal and external stakeholders.

Figure 1 provides a graphical representation of the development of the review standard.

11PMA-DR1CR3-2S-000006

GUIDANCE

This review standard provides guidance for

- processing EPU applications (Section 1)
- performing technical reviews (Section 2)
- preparing safety evaluations to document the reviews (Section 3)

This review standard also includes a reference to the NRC's Inspection Manual, which provides guidance for conducting inspections related to the implementation of power uprates (Section 4).

11PMA-DR1CR3-2S-000007

SECTION 1

PROCEDURAL GUIDANCE

11PMA-DR1CR3-2S-000008

1.1 Processing Extended Power Uprate Applications

The process flow chart (Figure 1.1-1) identifies each step involved in processing an EPU application (for which a hearing is not requested). (If a hearing is requested, the Project Manager will provide support to the Office of the General Counsel and arrange for staff to be available to support the hearing process.) The flow chart also identifies the responsible individual/organization and applicable procedures for completing each step. The staff should use the flow chart and referenced guidance documents when processing EPU applications.

Processing an EPU application involves, but is not limited to:

- performing an acceptance review
- issuing a *Federal Register* notice (without making a proposed no significant hazards consideration determination)
- performing a detailed technical review
- conducting ACRS briefings
- issuing draft and final environmental assessments
- making proprietary determinations, as necessary

The cognizant licensing Project Manager is responsible for coordinating the staff's review and ensuring that it is conducted in accordance with the process defined herein.

11PMA-DR1CR3-2S-000009

Figure 1.1-1 EPU Process Flow Chart

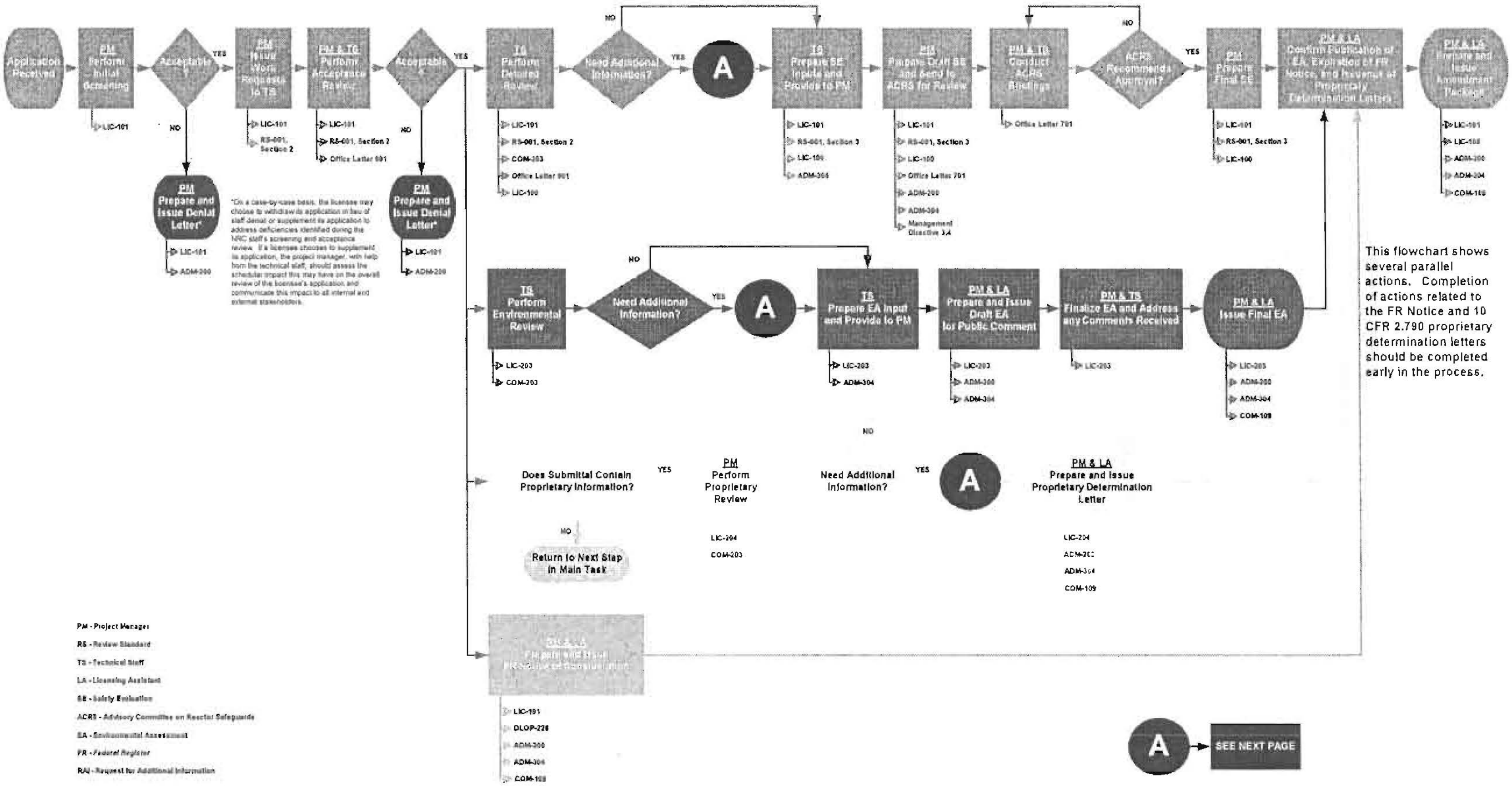

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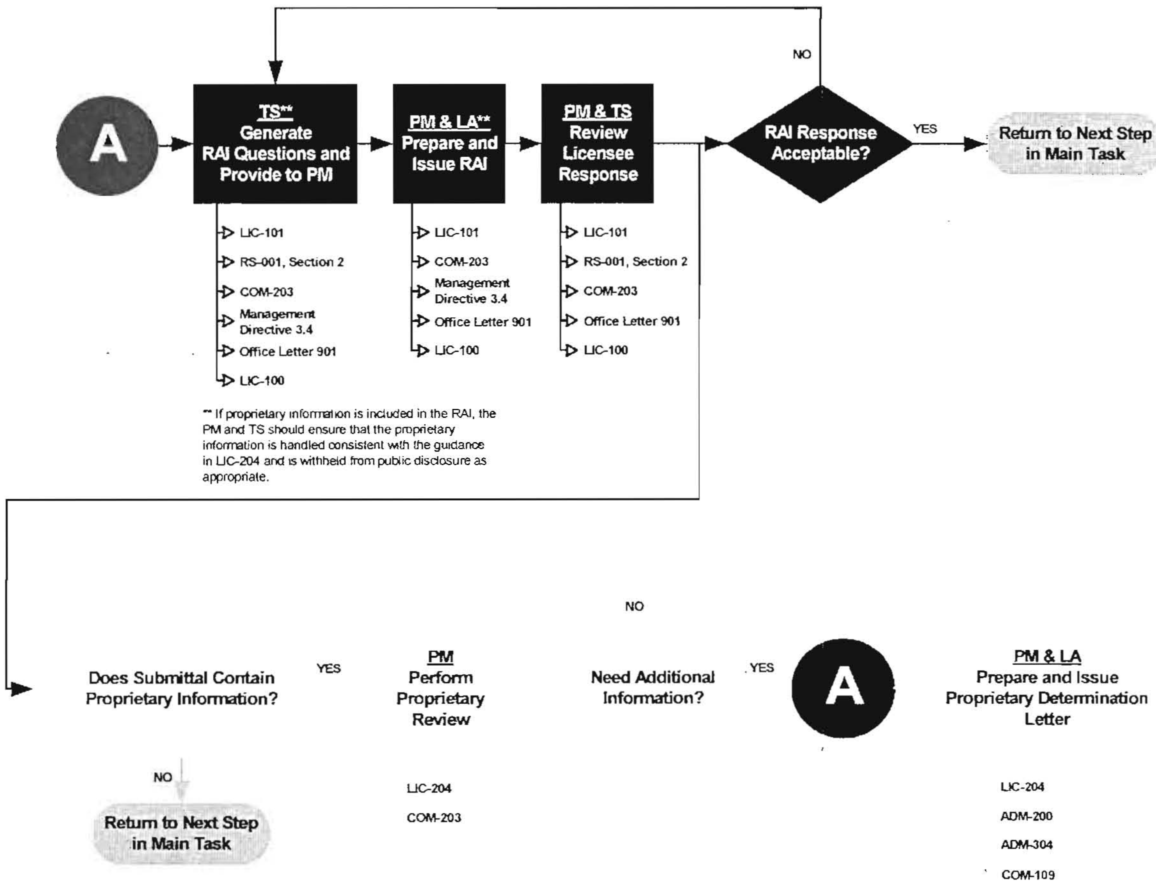
Figure 1.1-1 EPU Process Flow Chart

continued



PM - Project Manager

RS - Review Standard

TS - Technical Staff

LA - Licensing Assistant

SE - Safety Evaluation

ACRS - Advisory Committee on Reactor Safeguards

EA - Environmental Assessment

FR - *Federal Register*

RAI - Request for Additional Information

11PMA-DR1CR3-2S-000011

SECTION 2

TECHNICAL REVIEW GUIDANCE

11PMA-DR1CR3-2S-000012

2.1 Reviewing Extended Power Uprate Applications

This section defines the scope of technical review for EPU applications and identifies the guidance to be used when performing technical reviews of such applications.

Matrices 1 thru 11 of this section identify: (1) the technical areas to be reviewed, (2) the technical branches within the Office of Nuclear Reactor Regulation (NRR) responsible for the primary and secondary reviews, and (3) the applicable guidance documents to be used for performing the reviews. Acceptance criteria for the reviews are included in the referenced guidance documents.

The review of an EPU application involves the following three steps:

Step 1. Initial Screening

Upon receipt of an EPU application, the Project Manager will conduct an initial screening of the application for completeness and acceptability consistent with the guidance in NRR Office Instruction LIC-101, "License Amendment Review Procedures." This review is conducted to ensure that the application meets the minimum requirements described in 10 CFR 50.4, 10 CFR 50.90, 10 CFR 50.91, and 10 CFR 50.92. The Project Manager will distribute the application to the technical staff and proceed with the acceptance review if the application meets the minimum requirements.

Step 2. Acceptance Review

The Project Manager will review the EPU application to ensure that it adequately identifies the design basis of the plant for the items in the "Areas of Review" column in the matrices. The Project Manager should coordinate this effort with the acceptance review conducted by the reviewers with the primary review responsibility (discussed below).

Reviewers with primary review responsibility should follow the instructions below for completing the acceptance review.

(1) Based on the information provided in the EPU application, annotate the items in the "Areas of Review" column in the matrices to indicate (a) applicability of the items to the plant under review, (b) any additional areas of review that are affected by the EPU (as identified in the EPU application), and (c) any beyond-scope items that are included in the EPU application. (Licensees are also encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should avoid potential delays and improve the efficiency of the staff's review.)

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(2) Conduct an acceptance review to confirm that the licensee has addressed the applicable areas identified in the "Areas of Review" column of the matrices (as modified based on instruction (1) above). Review the information provided by the licensee for each area of review that is affected by the EPU to confirm that the regulatory requirements and design basis are adequately characterized and addressed with respect to the proposed EPU.

(3) Use the "Acceptance Review" column of the matrices as a checklist to document whether the licensee has addressed the areas of review in sufficient detail to allow the staff to proceed with its detailed technical review. Any negative comments in this column may lead to the NRC staff's denial of the application, or in substantial schedule delays.

(4) Before proceeding with the detailed technical review, provide the plant Project Manager a copy of the matrix completed as a result of instruction (3) above.

Step 3. Detailed Technical Review

(1) Compare the guidance in the documents referenced in the "SRP Section Number" and "Other Guidance" columns of the matrices to the design basis of the plant as described in the EPU application for each item in the "Areas of Review" column. Use the "Focus of SRP Usage" column to identify the applicable portions of the SRP sections identified. If the design basis of the plant that is identified in the EPU application is different from the guidance provided in the documents referenced in the matrices, consult with the Project Manager regarding the differences and compliance of the information in the EPU application with applicable regulations. Revise the matrices, as appropriate, based on the results of the review.

(2) If the areas of review for the plant are determined to be different from the areas identified in the matrices, obtain oral concurrence from the branch chief of the primary review branch for the differences. This should be done for additions to as well as deletions from the list of items in the "Areas of Review" column.

(3) Provide the revised matrices to the Project Manager. (Licensees are also encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should avoid potential delays and improve the efficiency of the staff's review.)

(4) Conduct a detailed review of the application consistent with the guidance provided in the documents listed in the "SRP Section Number" and "Other Guidance" columns (as modified to suit the design basis of the plant). Use the "Focus of SRP Usage" column to identify the applicable portions of the SRP sections identified.

(5) Coordinate with the technical branches identified in the "Secondary Review Branch(es)" column to ensure that all important aspects of each technical area are adequately covered during the review.

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(6) Perform audits and/or independent calculations as deemed necessary and appropriate to support review of the licensee's application. In determining the need for performing audits and/or independent calculations, consider the following:

- confidence of the NRC staff in the models and/or methods used by the licensee
- confidence of the NRC staff in the analysis results
- familiarity of the NRC staff with the models and/or methods used by the licensee
- prior use of the models and/or methods for similar plant designs and operating conditions and the NRC staff's experience related to such use
- NRC staff experience with the impact of proposed changes on analysis results
- available margin versus level of uncertainty in analysis results
- efficiency gains that may result from performing audits and/or independent calculations

Any issues identified as a result of independent calculations should be resolved with the licensee. If necessary, the licensee should be requested to update and resubmit any affected analyses. It should be noted that the NRC staff's approval of the application is to be based on the licensee's docketed information.

(7) Document the results of the detailed technical review in accordance with the guidance in Section 3.1 of this review standard.

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MATRIX 1

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Materials and Chemical Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Vessel Material Surveillance Program	All EPUs	EMCB	SRXB	5.3.1 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR Part 50, App. H 10 CFR 50.60	RG 1.190	2.1.1	2.1.1	
Pressure-Temperature Limits and Upper-Shelf Energy	All EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR Part 50, App. G 10 CFR 50.60	RG 1.161 RG 1.190 RG 1.99	2.1.2	2.1.2	
Pressurized Thermal Shock	PWR EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50.61	RG 1.190 RG 1.154		2.1.3	
Reactor Internal and Core Support Materials	All EPUs	EMCB	SRXB	4.5.2 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a	Note 1*	2.1.3	2.1.4	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Coolant Pressure Boundary Materials	All EPUs	EMCB	EMEB SRXB	5.2.3 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR Part 50, App. G	RG 1.190 GL 97-01 IN 00-17s1 BL 01-01 BL 02-01 BL 02-02 Note 2* Note 3*	2.1.4	2.1.5	
				4.5.1 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-14				
				5.2.4 Draft Rev. 2 April 1996	10 CFR 50.55a				
				5.3.1 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR Part 50, App. G				
				5.3.3 Draft Rev. 2 April 1996					
				6.1.1 Draft Rev. 2 April 1996					
Leak-Before-Break	PWR EPUs	EMCB		3.6.3 Draft Aug. 1987	GDC-4	NUREG 1061 Vol. 3 Nov. 1984		2.1.6	
Protective Coating Systems (Paints) - Organic Materials	All EPUs	EMCB		6.1.2 Draft Rev. 3 April 1996	10 CFR Part 50, App. B RG 1.54		2.1.5	2.1.7	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Effect of EPU on Flow-Accelerated Corrosion	All EPUs	EMCB				Note 4*	2.1.6	2.1.8	
Steam Generator Tube Inservice Inspection	PWR EPUs	EMCB		5.4.2.2 Draft Rev. 2 April 1996	10 CFR 50.55a	Plant TSs RG 1.121 GL 95-03 BL 88-02 GL 95-05 Note 5*		2.1.9	
Steam Generator Blowdown System	PWR EPUs	EMCB		10.4.8 Draft Rev. 3 April 1996	GDC-14			2.1.10	
Chemical and Volume Control System (Including Boron Recovery System)	PWR EPUs	EMCB	SPLB SRXB	9.3.4 Draft Rev. 3 April 1996	GDC-14 GDC-29			2.1.11	
Reactor Water Cleanup System	BWR EPUs	EMCB		5.4.8 Draft Rev. 3 April 1996	GDC-14 GDC-60 GDC-61		2.1.7		

Notes:

1. In addition to the SRP, guidance on the neutron irradiation-related threshold for inspection for irradiation-assisted stress-corrosion cracking for BWRs is in BWRVIP-26 and for PWRs in BAW-2248 for E>1 MeV and in WCAP-14577 for E>0.1 MeV. For intergranular stress-corrosion cracking and stress-corrosion cracking in BWRs, review criteria and review guidance is contained in BWRVIP reports and associated staff safety evaluations. For thermal and neutron embrittlement of cast austenitic stainless steel, stress-corrosion cracking, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs.

2. For thermal aging of cast austenitic stainless steel, review guidance and criteria is contained in the May 19, 2000, letter from C. Grimes to D. Walters, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components."

3. For intergranular stress corrosion cracking in BWR piping, review criteria and review guidance is contained in BWRVIP reports, NUREG-0313, Revision 2, GL 88-01, Supplement 1 to GL-88-01, and associated safety evaluations.

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4. Criteria and review guidance needed to review EPU applications in the area of flow-accelerated corrosion is contained in Electric Power Research Institute (EPRI) Report NSAC-202L-R2, "Recommendations for Effective an Flow-Accelerated Corrosion Program," dated April 1999. This EPRI document is copyrighted. EPRI has provided copies of this document to EMCB for use by NRC staff. Copying of this document, however, is not allowed.

5. Also see the plant-specific license amendments approving alternate repair criteria and redefining inspection boundaries.

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LIST OF ACRONYMS FOR MATRIX 1

BL = bulletin
BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
GL = generic letter
PWR = pressurized-water reactor
RG = regulatory guide
SPLB = Plant Systems Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

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MATRIX 2

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Mechanical and Civil Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Pipe Rupture Locations and Associated Dynamic Effects	All EPUs	EMEB		3.6.2 Draft Rev. 2 April 1996	GDC-4		2.2.1	2.2.1	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Pressure-Retaining Components and Component Supports	All EPUs	EMEB		3.9.1 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-14 GDC-15		2.2.2	2.2.2	
				3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-15	IN 95-016 IN 02-026			
				3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-14 GDC-15	IN 96-049 GL 96-06			
				5.2.1.1 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1	RG 1.84 RG 1.147 DG 1.1089 DG 1.1090 DG 1091			

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Pressure Vessel Internals and Core Supports	All EPUs	EMEB		3.9.1 Draft Rev. 3 April 1996	GDC-1 GDC-2		2.2.3	2.2.3	
				3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4	IN 95-016 IN 02-026			
				3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4	IN 96-049 GL 96-06			
				3.9.5 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-10	IN 02-026 Note 1*			

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Safety-Related Valves and Pumps	All EPUs	EMEB		3.9.3 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a(f)	IN 96-049 GL 96-06	2.2.4	2.2.4	
				3.9.6 Draft Rev. 3 April 1996	GDC-1 GDC-37 GDC-40 GDC-43 GDC-46 GDC-54 10 CFR 50.55a(f)	GL 89-10 GL 95-07 GL 96-05 IN 97-090 IN 96-048s1 IN 96-048 IN 96-003 RIS 00-003 RIS 01-015 RG 1.147 RG 1.175 DG 1089 DG 1091			
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	All EPUs	EMEB	EEIB	3.10 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-30 10 CFR Part 100, App. A 10 CFR Part 50, App. B USI A-46		2.2.5	2.2.5	

Notes:

1. As indicated in IN 2002-26 and Supplement 1 to IN 2002-26, the steam dryers and other plant components recently failed at Quad Cities Units 1 and 2 during operation under extended power uprate (EPU) conditions. The failures occurred as a result of high-cycle fatigue caused by increased flow-induced vibrations at EPU conditions. The staff's review of the reactor internals as part of EPU requests will cover detailed analyses of flow-induced vibration and acoustically-induced vibration (where applicable) on reactor internal components such as steam dryers and separators, and the jet pump sensing lines that are affected by the increased steam and feedwater flow for EPU conditions. In addition, the staff is evaluating the need to address potential adverse effects on other plant components from the increased steam and feedwater flow under EPU conditions.

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LIST OF ACRONYMS FOR MATRIX 2

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
DG = draft guide
EEIB = Electrical & Instrumentation & Controls Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
GL = generic letter
IN = information notice
PWR = pressurized-water reactor
RG = regulatory guide
RIS = regulatory issue summary
SRP = Standard Review Plan

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MATRIX 3

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Electrical Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Environmental Qualification of Electrical Equipment	All EPUs	EEIB		3.11 Draft Rev. 3 April 1996	10 CFR 50.49		2.3.1	2.3.1	
Offsite Power System	All EPUs	EEIB		8.1 Draft Rev. 3 April 1996	GDC-17	BTP PSB-1 Draft Rev. 3 April 1996 BTP ICSB-11 Draft Rev. 3 April 1996	2.3.2	2.3.2	
				8.2 Draft Rev. 4 April 1996	GDC-17				
				8.2, App. A Draft Rev. 4 April 1996	GDC-17				
AC Onsite Power System	All EPUs	EEIB		8.1 Draft Rev. 3 April 1996	GDC-17		2.3.3	2.3.3	
				8.3.1 Draft Rev. 3 April 1996	GDC-17				

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
DC Onsite Power System	All EPUs	EEIB		8.1 Draft Rev. 3 April 1996	GDC-17 10 CFR 50.63		2.3.4	2.3.4	
				8.3.2 Draft Rev. 3 April 1996	GDC-17 10 CFR 50.63				
Station Blackout	All EPUs	EEIB	SPLB SRXB	8.1 Draft Rev. 3 April 1996	10 CFR 50.63	Note 1*	2.3.5	2.3.5	
				8.2, App. B Draft Rev. 4 April 1996	10 CFR 50.63				

1. The review of station blackout includes the effects of the EPU on systems relied upon for core cooling in the station blackout coping analysis (e.g., condensate storage tank inventory, controls and power supplies for relief valves, residual heat removing system) to ensure that the effects are accounted for in the analysis.

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LIST OF ACRONYMS FOR MATRIX 3

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EEIB = Electrical & Instrumentation & Controls Branch
EPUs = extended power uprates
GDC = General Design Criterion
PWR = pressurized-water reactor
SRP = Standard Review Plan
BTP = branch technical position
AC = alternating current
DC = direct current

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MATRIX 4

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Instrumentation and Controls

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Trip System	All EPUs	EEIB		7.2 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-4 GDC-13 GDC-19 GDC-20 GDC-21 GDC-22 GDC-23 GDC-24		2.4.1	2.4.1	
Engineered Safety Features Systems	All EPUs	EEIB		7.3 Rev. 4 June 1997			2.4.1	2.4.1	
Safety Shutdown Systems	All EPUs	EEIB		7.4 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-4 GDC-13 GDC-19 GDC-24		2.4.1	2.4.1	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Control Systems	All EPUs	EEIB		7.7 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-13 GDC-19 GDC-24		2.4.1	2.4.1	
Diverse I&C Systems	All EPUs	EEIB		7.8 Rev. 4 June 1997			2.4.1	2.4.1	
General guidance for use of other SRP Sections related to I&C	All EPUs	EEIB		7.0 Rev. 4 June 1997					

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LIST OF ACRONYMS FOR MATRIX 4

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EEIB = Electrical & Instrumentation & Controls Branch
EPUs = extended power uprates
GDC = General Design Criterion
I&C = instrumentation and controls
PWR = pressurized-water reactor
SRP = Standard Review Plan

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MATRIX 5

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Plant Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Flood Protection	EPUs that result in significant increases in fluid volumes of tanks and vessels	SPLB		3.4.1 Rev. 2 July 1981	GDC-2		2.5.1.1.1	2.5.1.1.1	
Equipment and Floor Drainage System	EPUs that result in increases in fluid volumes or in installation of larger capacity pumps or piping systems	SPLB		9.3.3 Rev. 2 July 1981	GDC-2 GDC-4		2.5.1.1.2	2.5.1.1.2	
Circulating Water System	EPUs that result in increases in fluid volumes associated with the circulating water system or in installation of larger capacity pumps or piping systems	SPLB		10.4.5 Rev. 2 July 1981	GDC-4		2.5.1.1.3	2.5.1.1.3	
Internally Generated Missiles (Outside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.1 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	
Internally Generated Missiles (Inside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.2 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Turbine Generator	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.2 Rev. 2 July 1981	GDC-4		2.5.1.2.2	2.5.1.2.2	
Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	EPUs that affect environmental conditions, habitability of the control room, or access to areas important to safe control of postaccident operations	SPLB	EMCB EMEB	3.6.1 Rev. 1 July 1981	GDC-4		2.5.1.3	2.5.1.3	
Fire Protection Program	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.5.1 Rev. 3 July 1981	10 CFR 50.48 10 CFR Part 50, App. R GDC-3 GDC-5	Note 1*	2.5.1.4	2.5.1.4	
Pressurizer Relief Tank	PWR EPUs that affect pressurizer discharge to the PRT	SPLB	EMEB	5.4.11 Rev. 2 July 1981	GDC-2 GDC-4			2.5.2	
Fission Product Control Systems and Structures	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	6.5.3 Rev. 2 July 1981	GDC-41		2.5.2.1	2.5.3.1	
Main Condenser Evacuation System	EPUs for which the main condenser evacuation system is modified	SPLB		10.4.2 Rev. 2 July 1981	GDC-60 GDC-64		2.5.2.2	2.5.3.2	
Turbine Gland Sealing System	EPUs for which the turbine gland sealing system is modified	SPLB		10.4.3 Rev. 2 July 1981	GDC-60 GDC-64		2.5.2.3	2.5.3.3	
Main Steam Isolation Valve Leakage Control System	BWR EPU that affect the amount of valve leakage that is assumed and resultant dose consequences.	SPLB		6.7 Rev. 2 July 1981	GDC-54		2.5.2.4		

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Spent Fuel Pool Cooling and Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	9.1.3 Rev. 1 July 1981	GDC-5 GDC-44 GDC-61	Note 2*	2.5.3.1	2.5.4.1	
Station Service Water System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.1 Rev. 4 June 1985	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.3.2	2.5.4.2	
Reactor Auxiliary Cooling Water Systems	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.2 Rev. 3 June 1986	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.3.3	2.5.4.3	
Ultimate Heat Sink	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.5 Rev. 2 July 1981	GDC-5 GDC-44		2.5.3.4	2.5.4.4	
Auxiliary Feedwater System	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.9 Rev. 2 July 1981	GDC-4 GDC-5 GDC-19 GDC-34 GDC-44			2.5.4.5	
Main Steam Supply System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.3 Rev. 3 April 1984	GDC-4 GDC-5 GDC-34		2.5.4.1	2.5.5.1	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Main Condenser	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.1 Rev. 2 July 1981	GDC-60		2.5.4.2	2.5.5.2	
Turbine Bypass System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.4 Rev. 2 July 1981	GDC-4 GDC-34		2.5.4.3	2.5.5.3	
Condensate and Feedwater System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.7 Rev. 3 April 1984	GDC-4 GDC-5 GDC-44		2.5.4.4	2.5.5.4	
Gaseous Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of gaseous waste	SPLB	IEPB	11.3 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-3 GDC-60 GDC-61 10 CFR Part 50, App. I		2.5.5.1	2.5.6.1	
Liquid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of liquid waste	SPLB	IEPB	11.2 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-61 10 CFR Part 50, App. I		2.5.5.2	2.5.6.2	
Solid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of solid waste	SPLB	IEPB	11.4 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-63 GDC-64 10 CFR Part 71		2.5.5.3	2.5.6.3	
Emergency Diesel Engine Fuel Oil Storage and Transfer System	EPUs that result in higher EDG electrical demands	SPLB		9.5.4 Rev. 2 July 1981	GDC-4 GDC-5 GDC-17		2.5.6.1	2.5.7.1	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Light Load Handling System (Related to Refueling)	EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.1.4 Rev. 2 July 1981	GDC-61 GDC-62	[redacted]	2.5.6.2	2.5.7.2	

Notes:
1. Supplemental guidance for review of fire protection is provided in Attachment 1 to this matrix.

2. Supplemental guidance for review of spent fuel pool cooling is provided in Attachment 2 to this matrix.

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LIST OF ACRONYMS FOR MATRIX 5

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
GL = generic letter
IEPB = Emergency Preparedness and Plant Support Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan

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ATTACHMENT 1 TO MATRIX 5

Supplemental Fire Protection Review Criteria

Plant Systems

This attachment provides guidance for the review of the fire protection information to be provided in an application for a power uprate. Power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's application should confirm that these elements are not impacted by the extended power uprate. This confirmation should be reflected in the staff's safety evaluation. If the licensee indicates that there is an impact on these elements, the staff should review the licensee's assessment of the impact using this attachment.

The systems relied upon to achieve and maintain safe shutdown following a fire may be affected by the power uprate due to the increase in decay heat generation following a plant trip. For fire events where the licensee is relying on one full train of the redundant systems normally used for safe shutdown, the analysis of the impact of the power uprate on the important plant process parameters performed for other plant transients (such as a loss of offsite power or a loss of main feedwater) will typically bound the impact of a fire event. In this case, a specific analysis for fire events may not be necessary. However, where licensees rely on less than full capability systems for fire events (e.g., partial automatic depressurization system capability for reduced capability makeup pump), the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the integrity of the reactor pressure vessel or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability. The staff should verify that the capability of the alternative/dedicated or backup systems relied upon for post-fire safe shutdown is sufficient to achieve and maintain safe shutdown considering the impact of the power uprate.

The plant's post-fire safe shutdown procedures may also be impacted by the power uprate. For example, the allowable time to perform necessary operator actions may decrease as a result of the power uprate. In this case, the flow rates needed for systems required to achieve and maintain safe shutdown may need to be increased. The licensee should identify the impact of the power uprate on the plant's post-fire safe shutdown procedures.

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ATTACHMENT 2 TO MATRIX 5

Supplemental Spent Fuel Pool Cooling Review Criteria

Plant Systems

1. BACKGROUND

All operating nuclear power plants were licensed to certain design criteria regarding the adequacy of spent fuel pool (SFP) cooling capability. The most common criterion is that contained in General Design Criterion (GDC)-61 of Appendix A to 10 CFR Part 50. This criterion specifies, in part, that the fuel storage system (1) be designed with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal and (2) be designed to prevent a significant reduction in coolant inventory under accident conditions. Earlier licensing criteria are generally consistent with GDC-61. However, later guidance contained in Section 9.1.3 of the Standard Review Plan applied GDC-44 to the SFP cooling system. GDC-44 requires, in part, that a licensee provide a cooling system that is capable of accomplishing its safety function with or without offsite sources of power, assuming a single failure. To satisfy these criteria, each licensee should demonstrate that there is adequate SFP cooling capacity and should also demonstrate the ability to supply adequate make-up water in the event of total loss of SFP cooling.

A significant design-basis challenge to the SFP cooling system is imposed by a planned evolution (fuel transfer from the reactor vessel). Emergency offloads are not considered credible because fuel transfers may be accomplished only after plant cooldown, reactor disassembly, and refueling cavity flooding, which are time-consuming, manual processes. As a result, the staff will review factors that increase heat load (e.g., power increases, decay-time reductions, or storage capacity increases) and other operational factors that reduce heat load (e.g., longer decay times or transfer of fewer fuel assemblies to the SFP) or that increase heat removal capability (e.g., scheduling offloads for periods of reduced ultimate heat sink temperature or optimizing cooling system performance) to ensure that the licensee has demonstrated the adequacy of the SFP cooling system.

This guidance supercedes the guidance of paragraphs III.1.d. and III.1.h. of Standard Review Plan Section 9.1.3.

2. ACCEPTANCE CRITERIA

The adequacy of cooling may be evaluated against the capability to complete normal, planned activities, including fuel handling, without a degradation in safety and the ability to maintain defense-in-depth against a significant reduction in coolant inventory under accident conditions. With respect to fuel handling, which is a manual process, SFP temperatures affect safety through operating environment and visibility. At SFP temperatures below 140°F, (1) the fuel handling building ventilation is typically adequate to maintain a suitable operating environment, (2) evaporation from the SFP surface is at a sufficiently low rate to preclude fogging, and (3) the SFP temperature is within the design range of the cleanup system demineralizes to maintain water clarity. Defense-in-depth is provided by:

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(1) alarms to notify operators of a loss of cooling;
(2) the capability of the SFP cooling system to maintain or reestablish, within a reasonable time, forced cooling following a single failure of an active component;
(3) the ability of the cooling system to maintain the SFP temperature below the design temperature of the SFP structure and liner following a single-active failure or a design-basis event (e.g., a seismic event) within the current design basis of the facility; and
(4) the availability of two reliable sources of makeup water, one having sufficient capacity to make up for evaporation following a total loss of forced cooling. Only one source need have this capacity because the heat load and boil-off rate decrease rapidly with time from the peak value such that a much lower makeup rate would be effective in extending the recovery time.

The reliability of the systems relied upon to meet these guidelines should be maintained consistent with the plant's current design basis.

3. REVIEW PROCEDURES

3.1. Adequate SFP Cooling Capacity

The licensee demonstrates adequate SFP cooling capacity by either performing a bounding evaluation or committing to a method of performing outage-specific evaluations.

3.1.1. Bounding Calculation

Two scenarios are analyzed: (1) full cooling capability and (2) a single failure of an active cooling system component.

3.1.1.1. Full Cooling System Capability Evaluation

Analysis conditions:

(1) decay heat load is calculated based on bounding estimates of offload size, decay time, power history, and inventory of previously discharged assemblies
(2) heat removal capability is based on bounding estimates of ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance (e.g., fouling and tube plugging margin)
(3) alternate heat removal paths (e.g., evaporative cooling) should be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate)
(4) actual bulk SFP temperature should remain below 140 °F - calculated SFP temperatures up to approximately 150 °F are acceptable when justified by conservative methods or assumptions
(5) with appropriate administrative controls to verify that analysis inputs bound actual conditions, a set of bounding analyses may be prepared by the licensee to support operational flexibility.

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3.1.1.2. Single-Active Failure Evaluation

Analysis conditions:

(1) decay heat load is calculated based on a bounding estimate of offload size, decay time, power history, and inventory of previously discharged assemblies
(2) heat removal capability is based on a bounding estimate of ultimate heat sink temperature, heat exchanger performance (e.g., fouling and tube plugging margin), and cooling system flow rates assuming the limiting single failure with regard to heat removal capability
(3) alternate heat removal paths (e.g., evaporative cooling) should be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate)
(4) calculated bulk SFP temperature should remain below the design temperature of the SFP structure and liner, and calculated peak storage cell temperature should remain below the storage rack design temperature
(5) for plants where a single failure results in a complete loss of forced cooling, the licensee's analysis should demonstrate that the loss of cooling would be identified and forced cooling would be restored before the bounding decay heat load would cause the SFP temperature to reach its design limit
(6) with appropriate administrative controls to verify that analysis inputs bound actual conditions, a set of bounding analyses may be prepared by the licensee to support operational flexibility.

3.1.2. Cycle-Specific Calculation:

The licensee can choose to define a method to calculate operational limits prior to every offload using the anticipated actual conditions at the time of the offload.

Cycle-specific analysis conditions:

(1) define the method to calculate decay heat load based on decay time, power history, and inventory of previous fuel discharges
(2) define the method to calculate cooling system heat removal capacity based on ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance parameters
(3) define the method for calculating alternate heat removal capability (e.g., evaporative cooling) and provide validation of the method
(4) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will remain below 140 °F with full cooling capability
(5) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will be maintained below the SFP structure design temperature assuming a single failure affecting the forced cooling system (this may be a heat-balance analysis if cooling is degraded or a heatup-rate analysis if forced cooling is completely lost and subsequently recovered using redundant components)

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(6) describe administrative controls that will be implemented each offload to ensure the cycle-specific analysis inputs and results bound actual conditions prior to fuel movement

3.2. <u>Adequate Make-Up Supply</u>

(1) Following a loss-of-SFP cooling event, the licensee should be able to provide two sources of make-up water prior to the occurrence of boiling in the pool. To determine the time to boil, the initial pool temperature is the peak temperature from a planned offload, assuming the worst single-active failure occurred.

(2) At least one make-up source should have a capacity that is equal to or greater than the calculated boil-off rate so that the SFP level can be maintained. Only one source need have this capacity because the heat load and boil-off rate decrease rapidly with time from the peak value such that a much lower makeup rate would be effective in extending the recovery time.

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MATRIX 6

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Containment Review Considerations

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
PWR Dry Containments, Including Subatmospheric Containments	EPUs for PWR plants with dry containments (including subatmospheric containments) except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.A Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	
Ice Condenser Containments	EPUs for PWR plants with ice condenser containments except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.B Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Pressure-Suppression Type BWR Containments	EPUs for BWR plants with pressure-suppression containments except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.C Rev. 6 Aug. 1984	GDC-4 GDC-13 GDC-16 GDC-50 GDC-64		2.6.1		
Subcompartment Analysis	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.2 Rev. 2 July 1981	GDC-4 GDC-50		2.6.2	2.6.2	
Mass and Energy Release Analysis for Postulated Loss-of-Coolant	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.3 Rev. 1 July 1981	GDC-50 10 CFR Part 50, App. K		2.6.3.1	2.6.3.1	
Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.4 Rev. 1 July 1981	GDC-50			2.6.3.2	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Combustible Gas Control In Containment	EPUs that impact hydrogen release assumptions	SPSB		6.2.5 Rev. 2 July 1981	10 CFR 50.44 10 CFR 50.46 GDC-5 GDC-41 GDC-42 GDC-43		2.6.4	2.6.4	
Containment Heat Removal	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.2 Rev. 4 Oct. 1985	GDC-38	DG-1107	2.6.5	2.6.5	
Secondary Containment Functional Design	EPUs that affect the pressure and temperature response, or draw-down time of the secondary containment	SPSB		6.2.3 Rev. 2 July 1981	GDC-4 GDC-16		2.6.6		
Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPSB	SRXB	6.2.1 Rev. 2 July 1981 6.2.1.5 Rev. 2 July 1981	10 CFR 50.46 10 CFR Part 50, App. K			2.6.6	

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LIST OF ACRONYMS FOR MATRIX 6

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
DG = draft guide
EPUs = extended power uprates
GDC = General Design Criterion
PWR = pressurized-water reactor
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

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MATRIX 7

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Habitability, Filtration, and Ventilation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Control Room Habitability System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.4 Draft Rev. 3 April 1996	GDC-4 GDC-19	Note 1* Note 2*	2.7.1	2.7.1	
ESF Atmosphere Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.5.1 Rev. 2 July 1981	GDC-19 GDC-41 GDC-61 GDC-64		2.7.2	2.7.2	
Control Room Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.1 Rev. 2 July 1981	GDC-4 GDC-19 GDC-60		2.7.3	2.7.3	
Spent Fuel Pool Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.2 Rev. 2 July 1981	GDC-60 GDC-61		2.7.4	2.7.4	
Auxiliary and Radwaste Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.3 Rev. 2 July 1981	GDC-60		2.7.5	2.7.5	
Turbine Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.4 Rev. 2 July 1981	GDC-60		2.7.5	2.7.5	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
ESF Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB	[redacted]	9.4.5 Rev. 2 July 1981	GDC-4 GDC-17 GDC-60	[redacted]	2.7.6	2.7.6	

Notes:

1. Under SRP Section 6.4, Section II, "Acceptançe Criteria," the discussion for Item C related to GDC-19 should be supplemented with "and providing a suitably controlled environment for the control room operators and the equipment located therein."

2. Under SRP Section 6.4, Section II, Item 2, "Ventilation System Criteria," the discussion related to review of the control room area ventilation system under SRP Section 9.4.1 should be retained.

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LIST OF ACRONYMS FOR MATRIX 7

BWR = boiling-water reactor
EPUs = extended power uprates
ESF = engineered safety feature
GDC = General Design Criterion
PWR = pressurized-water reactor
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan

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MATRIX 8

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Reactor Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Fuel System Design	All EPUs	SRXB	[redacted]	4.2 Draft Rev. 3 April 1996	10 CFR 50.46 GDC-10 GDC-27 GDC-35	Note 1* Note 2*	2.8.1	2.8.1	
Nuclear Design	All EPUs	SRXB	[redacted]	4.3 Draft Rev. 3 April 1996	GDC-10 GDC-11 GDC-12 GDC-13 GDC-20 GDC-25 GDC-26 GDC-27 GDC-28	RG 1.190 GSI 170 IN 97-085	2.8.2	2.8.2	
Thermal and Hydraulic Design	All EPUs	SRXB	[redacted]	4.4 Draft Rev. 2 April 1996	GDC-10 GDC-12	Note 3*	2.8.3	2.8.3	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Functional Design of Control Rod Drive System	All EPUs	SRXB	SPLB	4.6 Draft Rev. 2 April 1996	GDC-4 GDC-23 GDC-25 GDC-26 GDC-27 GDC-28 GDC-29 10 CFR 50.62(c)(3)		2.8.4.1	2.8.4.1	
Overpressure Protection during Power Operation	All EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31	Note 4*	2.8.4.2	2.8.4.2	
Overpressure Protection during Low Temperature Operation	PWR EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31			2.8.4.3	
Reactor Core Isolation Cooling System	BWR EPUs	SRXB		5.4.6 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-29 GDC-33 GDC-34 GDC-54 10 CFR 50.63		2.8.4.3		
Residual Heat Removal System	All EPUs	SRXB		5.4.7 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-19 GDC-34	Note 5*	2.8.4.4	2.8.4.4	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Emergency Core Cooling System	All EPUs	SRXB		6.3 Draft Rev. 3 April 1996	GDC-4 GDC-27 GDC-35 10 CFR 50.46 10 CFR Part 50, App. K	Note 6*	2.8.5.6.2	2.8.5.6.3	
Standby Liquid Control System	BWR EPUs	SRXB	EMCB SPLB	9.3.5 Draft Rev. 3 April 1996	GDC-26 GDC-27 10 CFR 50.62(c)(4)	Note 10*	2.8.4.5		
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	All EPUs	SRXB		15.1.1-4 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26	Note 7*	2.8.5.1	2.8.5.1.1	
Steam System Piping Failures Inside and Outside of Containment	PWR EPUs	SRXB		15.1.5 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.8.5.1.2	
Loss of External Load; Turbine Trip, Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	All EPUs	SRXB		15.2.1-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.1	2.8.5.2.1	
Loss of Nonemergency AC Power to the Station Auxiliaries	All EPUs	SRXB		15.2.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.2	2.8.5.2.2	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Loss of Normal Feedwater Flow	All EPUs	SRXB	EEIB	15.2.7 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.3	2.8.5.2.3	
Feedwater System Pipe Breaks Inside and Outside Containment	PWR EPUs	SRXB	EEIB	15.2.8 Draft Rev. 2 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.8.5.2.4	
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All EPUs	SRXB		15.3.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.3.1	2.8.5.3.1	
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All EPUs	SRXB		15.3.3-4 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31	Note 7*	2.8.5.3.2	2.8.5.3.2	
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	All EPUs	SRXB		15.4.1 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.8.5.4.1	2.8.5.4.1	
Uncontrolled Control Rod Assembly Withdrawal at Power	All EPUs	SRXB		15.4.2 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.8.5.4.2	2.8.5.4.2	
Control Rod Misoperation (System Malfunction or Operator Error)	PWR EPUs	SRXB		15.4.3 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*		2.8.5.4.3	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	All EPUs	SRXB		15.4.4-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26 GDC-28	Note 7*	2.8.5.4.3	2.8.5.4.4	
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	PWR EPUs	SRXB		15.4.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*		2.8.5.4.5	
Spectrum of Rod Ejection Accidents	PWR EPUs	SRXB		15.4.8 Draft Rev. 3 April 1996	GDC-28	Note 7*		2.8.5.4.6	
Spectrum of Rod Drop Accidents	BWR EPUs	SRXB		15.4.9 Draft Rev. 3 April 1996	GDC-28	Note 7*	2.8.5.4.4		
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	All EPUs	SRXB		15.5.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7* Note 8*	2.8.5.5	2.8.5.5	
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	All EPUs	SRXB		15.6.1 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.6.1	2.8.5.6.1	
Steam Generator Tube Rupture	PWR EPUs	SRXB		15.6.3 Draft Rev. 3 April 1996	Note 7*	Note 7*		2.8.5.6.2	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Loss-of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	All EPUs	SRXB		15.6.5 Draft Rev. 3 April 1996	GDC-35 10 CFR 50.46	Note 7* Note 9*	2.8.5.6.2	2.8.5.6.3	
Anticipated Transient Without Scram	All EPUs	SRXB				Note 7* Note 10*	2.8.5.7	2.8.5.7	
New Fuel Storage	EPU applications that request approval for new fuel design.	SRXB		9.1.1 Draft Rev. 3 April 1996	GDC-62		2.8.6.1	2.8.6.1	
Spent Fuel Storage	EPU applications that request approval for new fuel design.	SRXB		9.1.2 Draft Rev. 4 April 1996	GDC-4 GDC-62		2.8.6.2	2.8.6.2	

Notes:

1. When mixed cores (i.e., fuels of different designs) are used, the review covers the licensee's evaluation of the effects of mixed cores on design-basis accident and transient analyses.

2. The current acceptance criteria for fuel damage for reactivity insertion accidents (RIAs) need revision per Research Information Letter No. 174, "Interim Assessment of Criteria for Analyzing Reactivity Accidents at High Burnup." The Office of Nuclear Regulatory Research is conducting confirmatory research on RIAs and the Office of Nuclear Reactor Regulation is discussing the issue of fuel damage criteria with the nuclear power industry as part of the industry's proposal to increase future fuel burnup limits. In the interim, current methods for assessing fuel damage in RIAs are considered acceptable based on the NRC staff's understanding of actual fuel performance, as shown in three-dimensional kinetic calculations which indicate acceptably low fuel cladding enthalpy.

3. The review also covers core design changes and any effects on radial and bundle power distribution, including any changes in critical heat flux ratio and critical power ratio. The review will also confirm the adequacy of the flow-based average power range monitor flux trip and safety limit minimum critical power ratio at the uprated conditions.

4. The review also covers the determination of allowable power levels with inoperable main steam safety valves.

5. The review also covers the total time necessary to reach the shutdown cooling initiation temperature.

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6. The review for BWRs will cover the justification for changes in calculated peak cladding temperature (PCT) for the design-basis case and the upper-bound case and any impact of the changes in PCTs on the use of the design methods for the power uprate.

7. The review:
   - confirms that the licensee used NRC-approved codes and methods for the plant-specific application and the licensee's use of the codes and methods complies with any limitations, restrictions, and conditions specified in the approving safety evaluation.
   - confirms that all changes of reactor protection system trip delays are correctly addressed and accounted for in the analyses.
   - (for PWRs) confirms that steam generator plugging and asymmetry limits are accounted for in the analyses.
   - (for PWRs) covers the licensee's evaluation of the effects of Westinghouse Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4, and NSAL 02-5. These NSALs document problems with water level setpoint uncertainties in Westinghouse-designed steam generators. The review is conducted to ensure that the effects of the identified problems have been accounted for in steam generator water level setpoints used in LOCA, non-LOCA, and ATWS analyses.

8. For the inadvertent operation of emergency core cooling system and chemical and volume control system malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.

9. The review also verifies that:
   - Licensee and vendor processes ensure LOCA analysis input values for PCT-sensitive parameters bound the as-operated plant values for those parameters
   - (For PWRs) The models and procedures continue to comply with 10 CFR 50.46 during the switchover from the refueling water storage tank to the containment sump (i.e., the core remains adequately cool during any flow reduction or interruption that may occur during switchover).
   - (For PWRs) Large-break LOCA analyses account for boric acid buildup during long-term core cooling and that the predicted time to initiate hot leg injection is consistent with the times in the operating procedures.
   - (For BWRs) The licensee's comparison of parameters used in the LOCA analysis with actual core design parameters provide the needed justification to confirm the applicability of the generic LOCA methodology.

10. The ATWS review is conducted to ensure that the plant meets the 10 CFR 50.62 requirements:

   - For PWR plants with both a diverse scram system (DSS) and ATWS mitigation system actuation circuitry (AMSAC), the staff will not review ATWS for EPUs.
   - For PWR plants where a DSS is not specifically required by 10 CFR 50.62, a review is conducted to verify that the consequences of an ATWS are acceptable. The acceptance criteria is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient and the primary system relief capacity.
   - For BWR plants, the review is conducted to ensure that the licensee has appropriately accounted for changes in analyses due to the uprated power level and confirm that required equipment, such as the standby liquid control system (SLCS) pumps, can deliver required flowrates. The review will also cover the SLCS relief valve margin. In addition, a review is conducted to ensure that SLCS flow can be injected at the assumed time without lifting bypass relief valves during the limiting ATWS.

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LIST OF ACRONYMS FOR MATRIX 8

BWR = boilling-water reactor
CFR = *Code of Federal Regulations*
EMCB = Materials and Chemical Engineering Branch
EPUs = extended power uprates
GDC = general design criterion
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SRP = standard review plan
SRXB = Reactor Systems Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
EMCB = Materials & Chemical Engineering Branch
LOCA = loss-of-coolant accident
ATWS = anticipated transients without scram
ASME = American Society of Mechanical Engineers
AMSAC = ATWS Mitigation System Actuation Circuitry
DSS = Diverse Scram System

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MATRIX 9

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Source Terms and Radiological Consequences Analyses

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Source Terms for Input into Radwaste Management Systems Analyses	All EPUs	SPSB		11.1 Draft Rev. 3 April 1996	10 CFR Part 20 10 CFR Part 50, App. I GDC-60		2.9.1	2.9.1	
Radiological Consequence Analyses Using Alternative Source Terms	EPUs that utilize alternative source term	SPSB	EEIB EMCB EMEB IEPB SPLB SRXB	15.0.1 Rev. 0 July 2000	10 CFR 50.67 GDC-19 10 CFR 50.49 10 CFR Part 51 10 CFR Part 50, App. E NUREG-0737		2.9.2	2.9.2	
Radiological Consequences of Main Steamline Failures Outside Containment for a PWR	PWR EPUs that do not utilize alternative source term whose main steamline break analyses result in fuel failure	SPSB	SRXB	15.1.5, App. A Draft Rev. 3 April 1996	10 CFR Part 100	Notes 4, 5, 6, 7, 27*		2.9.2	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiological Consequences of Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	EPUs that do not utilize alternative source term whose reactor coolant pump rotor seizure or reactor coolant pump shaft break results in fuel failure	SPSB	SRXB	15.3.3-4 Draft Rev. 3 April 1996	10 CFR Part 100	Notes 5, 8, 9, 27*		2.9.3	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Control Rod Ejection Accident	PWR EPUs that do not utilize alternative source term whose rod ejection accident results in fuel failure or melting	SPSB	SRXB	15.4.8, App. A Draft Rev. 2 April 1996	10 CFR Part 100	Notes 4, 21, 22, 27*		2.9.4	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Control Rod Drop Accident	BWR EPUs that do not utilize alternative source term whose control rod drop accident results in fuel failure or melting	SPSB	SRXB	15.4.9, App. A Draft Rev. 3 April 1996	10 CFR Part 100	Notes 9, 10, 27*	2.9.2		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	EPUs that do not utilize alternative source term whose failure of small lines carrying primary coolant outside containment result in fuel failure	SPSB		15.6.2 Draft Rev. 3 April 1996	GDC-55 10 CFR Part 100		2.9.3	2.9.5	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiological Consequences of Steam Generator Tube Failure	PWR EPUs that do not utilize alternative source term whose steam generator tube failure results in fuel failure	SPSB	SRXB	15.6.3 Draft Rev. 3 April 1996	10 CFR Part 100	Notes 4, 13, 14, 15, 27*		2.9.6	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Main Steamline Failure Outside Containment for a BWR	BWR EPUs that do not utilize alternative source term whose main steam line failure outside containment results in fuel failure	SPSB	SRXB	15.6.4 Draft Rev. 3 April 1996	10 CFR Part 100	Note 27*	2.9.4		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident Including Containment Leakage Contribution	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. A Draft Rev. 2 April 1996	10 CFR Part 100	Notes 4, 23, 24, 25, 26, 27*	2.9.5	2.9.7	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from ESF Components Outside Containment	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. B Draft Rev. 2 April 1996	10 CFR Part 100	Notes 11, 27*	2.9.5	2.9.7	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from Main Steam Isolation Valves	BWR EPUs that do not utilize alternative source term	SPSB		15.6.5, App. D Draft Rev. 2 April 1996	10 CFR Part 100	Notes 9, 12, 27*	2.9.5		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Fuel Handling Accidents	EPUs that do not utilize alternative source term	SPSB	SPLB	15.7.4 Draft Rev. 2 April 1996	10 CFR Part 100 GDC-61	Notes 4, 5, 18, 19, 20, 27*	2.9.6	2.9.8	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Spent Fuel Cask Drop Accidents	EPUs that do not utilize alternative source term	SPSB	EMEB SPLB	15.7.5 Draft Rev. 3 April 1996	10 CFR Part 100 GDC-61	Notes, 5, 16, 17, 8, 18, 27*	2.9.7	2.9.9	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

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Notes:

1. In addition to SRP Section 15.6.5, Appendices A, B, and D, dose consequences in the control room are determined from design-basis accidents as part of the review for SRP Sections 15.0.1; 15.1.5, Appendix A; 15.3.3-4, 15.4.8, Appendix A; 15.4.9, Appendix A; 15.6.2, 15.6.3, 15.6.4, 15.7.4, and 15.7.5.

2. Regulatory Guide 1.95 was canceled. Relevant guidance from Regulatory Guide 1.95 was incorporated into Regulatory Guide 1.78, Revision 1 in January 2002. Therefore, Regulatory Guide 1.95 should not be used.

3. Table 6.4-1, attached to SRP Section 6.4 and referred to in Item 7, "Independent Analyses," of the "Review Procedures" Section of SRP Section 6.4 may not be used.

4. Acceptable dose conversion factors may be taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Environmental Protection Agency, 1988; and Table III.1 of Federal Guidance Report 12, " External Exposure to Radionuclides in Air, Water, and Soil," Environmental Protection Agency, 1993.

5. NUREG-1465 should not be used.

6. For the review of the main steamline failure accident, review of facilities licensed with, or applying for, alternative repair criteria (ARC) should use SRP Section 15.1.5, Appendix A, in conjunction with the guidance in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

7. For facilities that implement ARC, the primary-to-secondary leak rate in the faulted generator should be assumed to be the maximum accident-induced leakage derived from the repair criteria and burst correlations. The leak rate limiting condition for operation specified in the technical specifications is equally apportioned among the unaffected steam generators.

8. Guidance for the radiological consequences analyses review with respect to acceptable modeling of the radioactivity transport is given in SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," for applicants that use the traditional source term, based on TID-14844.

9. References to specific computer codes (e.g., SARA, TACT, Pipe Model) are not necessary since other computer codes/methods may be used.

10. In the second paragraph of Section III, "Review Procedure," it is stated that the control rod drop accident is expected to result in radiological consequences less than 10 percent of the 10 CFR Part 100 guideline values, even with conservative assumptions. The value of 10 percent should be replaced with 25 percent.

11. In Section III, "Review Procedures," the guidance in the fourth paragraph, which deals with passive failures, should not be used.

12. The last paragraph on page 15.6.5-4 refers to a "code" developed by J. E. Cline and Associates, Inc. This is identified as Reference 5 in the paragraph. The word "code" should be changed to "model" because the staff does not have the computer code. In addition, the correct reference to the work by J. E. Cline and Associates, Inc., is 4.

13. Item 4 of the "Review Interfaces" section should be deleted. SPSB review of the steam generator tube rupture accidents for their contribution to plant risk is not currently used in the design-basis accident review for radiological consequences.

14. The reference to Figure 3.4-1 of the Nuclear Steam Supply System vendor Standard Technical Specification in Item 6.(a) of Section III, "Review Procedures," does not apply. In addition, the primary coolant iodine concentration discussed in this Item is the 48-hour maximum value.

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15. In Item 6.(b) of Section III, "Review Procedures," the multiplier of 500 used for estimating the increase in iodine release rate is reduced to 335 as a result of the staff's review of iodine release rate data collected by Adams and Atwood.

16. The reference to SRP Section 9.1.4 in Item 2.c of the "Review Interfaces" section should be changed to SRP Section 9.1.5.

17. The reference to Regulatory Guide 1.25, which was deleted in 1996, should be retained, with exceptions as noted below in Note 18.

18. The following exceptions to Regulatory Guide 1.25 are provided. These exceptions are based on the staff's review of NUREG/CR-6703.

The fraction of the core inventory assumed to be in the gap for the various nuclides are given in the table below. The release fractions from the table are used in conjunction with the calculated fission product inventory and the maximum core radial peaking factor. These release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU, provided that the maximum linear heat generation rate will not exceed 6.3 kW/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations using NRC-approved methodologies may be considered on a case-by-case basis.

NON-LOCA FRACTION OF FISSION PRODUCT INVENTORY IN GAP	
GROUP	**FRACTION**
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Iodines	0.05

19. References to the Standard Technical Specifications should be replaced with references to the plant-specific technical specifications or technical requirements manual (TRM).

20. Technical Specification Task Force (TSTF) Traveler TSTF-51 proposed to add the term "recently," as it applies to irradiated fuel, to the applicability section of certain technical specifications. The proposed change is intended to remove certain technical specifications requirements for operability of ESF systems (e.g., secondary containment isolation and filtration systems) during refueling. The associated technical specifications bases define "recently," as it applies to irradiated fuel, as the minimum decay time used in supporting radiological consequences analyses of fuel handling accidents. Radiological consequences analyses for these applicants should generally assume a 2-hour release directly to the environment, without holdup or mitigation by ESF systems and no credit for containment closure. Additionally, licensees adding the term "recently" must make a commitment for a single normal or contingency method to promptly close primary or secondary containment penetrations. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The review of this commitment and the prompt methods should be coordinated with IORB, SPLB, and IEPB.

21. In the last sentence of Item 2 of the "Review Interfaces" section, the reference to the number of fuel pins experiencing departure from nucleate boiling (DNB) should be deleted. The reference to fuel clad melting should be used and is therefore retained.

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22. In Item 2 of the "Review Procedures" section, the references to the "number of fuel pins reaching DNB" should be deleted and replaced with "the number of fuel pins with cladding failure." In addition, the use of a conservative value of 10 percent for fuel cladding failure in the calculation of the radiological consequences of the rod ejection accident is acceptable.

23. In Item 1 of the "Areas of Review" section, the use of the word "established" is incorrect. The word "established" should be replaced with the word "assessed."

24. In Item 1 of the "Acceptance Criteria" section, the following text in the last line should be deleted: "3.0 Sv (300 rem) to the thyroid and 0.25 Sv (25 rem) to the whole body."

25. In Item 1 of the "Review Procedures" section, the following should be added after the first sentence:

> Appendix K to 10 CFR Part 50 defines conservative analysis assumptions for evaluation of ECCS performance during design-basis LOCAs. Appendix K requires the licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. Appendix K allows for an assumed power level less than 1.02 times the licensed power level but not less than the licensed power level, provided the alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

26. In Item 2 of the "Review Procedures" section, the following statements should be deleted:

> "A check is made of the LOCA [loss-of-coolant accident] assumptions listed in Chapter 15 of the SAR to verify that the primary containment leakage rate has been assumed to remain constant over the course of the accident for a BWR and to remain constant at one half of the initial leak rate after 24 hours for a PWR."
>
> "The leakage rate used should correspond to that given in the technical specification."

The above statements should be replaced with the following:

> "A check is made of the LOCA assumptions listed in Chapter 15 of the SAR to verify acceptable primary containment leakage assumptions. The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leakage rate may be reduced after the first 24 hours to 50 percent of the TS leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the TS leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition, as defined by the TSs."

27. The staff has drafted updated guidance on performing design-basis radiological analyses in draft Regulatory Guide DG-1113, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," issued for public comment January 2002. The resulting final regulatory guide may be used for guidance on review of design-basis accident non-alternative source term radiological analyses after the date of issuance of the final regulatory guide.

28. In Section II, "Acceptance Criteria," the discussion for Item C related to GDC-19 should be supplemented with

> "and providing a suitably controlled environment for the control room operators and the equipment located therein."

29. In Section II, Item 2, "Ventilation System Criteria," the discussion related to review of the control room area ventilation system under SRP Section 9.4.1 should be retained.

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LIST OF ACRONYMS FOR MATRIX 9

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EEIB = Electrical & Instrumentation & Controls Branch
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
IEPB = Emergency Preparedness and Plant Support Branch
PWR = pressurized-water reactor
IROB = Reactor Operations Branch
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

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MATRIX 10

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Health Physics

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiation Sources	All EPUs	IEPB		12.2 Draft Rev. 3 April 1996	10 CFR Part 20		2.10.1	2.10.1	
Radiation Protection Design Features	All EPUs	IEPB		12.3-4 Draft Rev. 3 April 1996	10 CFR Part 20 GDC-19	Note 1*	2.10.1	2.10.1	
Operational Radiation Protection Program	All EPUs	IEPB		12.5 Draft Rev. 3 April 1996	10 CFR Part 20	Note 2* Note 3*	2.10.1	2.10.1	

Notes:

1. Regulatory Guide 8.12, "Criticality Accident Alarm Systems" has been withdrawn and should not be used.
2. Regulatory Guide 8.3, "Film Badge Performance Criteria" has been withdrawn and should not be used.
3. Regulatory Guide 8.14, "Personnel Neutron Dosimeters".has been withdrawn and should not be used.

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LIST OF ACRONYMS FOR MATRIX 10

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EPUs = extended power uprates
GDC = General Design Criterion
IEPB = Emergency Preparedness and Plant Support Branch
PWR = pressurized-water reactor
SRP = Standard Review Plan

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MATRIX 11

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Human Performance

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Operator Training	All EPUs	IROB		13.2.1* Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	
Training for Non-Licensed Plant Staff	All EPUs	IROB		13.2.2* Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	
Operating and Emergency Operating Procedures	All EPUs	IROB	SPLB SPSB SRXB	13.5.2.1* Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	

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Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Human Factors Engineering	All EPUs	IROB		18.0** Draft Rev. 0 April 1996	Specific review questions are provided in the template safety evaluations.		2.11	2.11	

*The staff is currently finalizing SRP Sections 13.2.1, 13.2.2, and 13.5.2.1. While these SRP Sections are being finalized, the staff will continue to use the versions issued in December 2002 for interim use and public comment. Once finalized, the staff will use the new versions of these SRP Sections.

**The staff received significant comment on draft SRP Chapter 18.0 that was issued in December 2002 for interim use and public comment. The staff is working on finalizing this SRP. However, due to the significance of the comments received, the staff will use Draft SRP Chapter 18.0, Revision 0, dated April 1996.

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LIST OF ACRONYMS FOR MATRIX 11

BWR = boiling-water reactor
EPUs = extended power uprates
IROB = Reactor Operations Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabilistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

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MATRIX 12

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Power Ascension and Testing Plan

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Power Ascension and Testing	All EPUs	IEPB	EEIB EMCB EMEB IROB SPLB SPSB SRXB	14.2.1* Draft Rev. 0 Dec. 2002	Entire Section		2.12	2.12	

*The staff is currently finalizing SRP Section 14.2.1. While this SRP Section is being finalized, the staff will continue to use the version issued for interim use and public comment in December 2002. Once finalized, the staff will use the new version.

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LIST OF ACRONYMS FOR MATRIX 12

BWR = boiling-water reactor
EEIB = Electrical & Instrumentation & Controls Branch
EMCB = Materials and Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
IEPB = Emergency Preparedness and Plant Support Branch
IROB = Reactor Operations Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

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MATRIX 13

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Risk Evaluation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Risk Evaluation	All EPUs	SPSB	[redacted]	[redacted]	[redacted]	Note 1* RG 1.174 RIS 2001-02	2.13	2.13	

Notes:
1. The staff's review is based on Attachment 1 to this matrix. Attachment 1 invokes SRP Chapter 19, Appendix D, if special circumstances are identified during the review.

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LIST OF ACRONYMS FOR MATRIX 13

BWR = boiling-water reactor
EPUs = extended power uprates
PWR = pressurized-water reactor
RG = regulatory guide
RIS = regulatory issue summary
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan

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ATTACHMENT 1 TO MATRIX 13

Supplemental Risk Evaluation Review Guidance

Risk Evaluation

1. INTRODUCTION

In addition to ensuring that a license amendment request complies with the U.S. Nuclear Regulatory Commission's (NRC's) regulations and other requirements, it is also the staff's responsibility to consider the risk aspects of a license amendment request (cf. COMSAJ-97-08 and RIS 2001-02). The use of risk information is clear when the licensee or the NRC designates the submittal as a "risk-informed" license application. Guidance is also provided to the staff in Appendix D of Chapter 19 of the Standard Review Plan (SRP) (Reference 1) as to the "special circumstances" under which a detailed risk review may be necessary, even for license applications that are not designated as being risk-informed. This process is also described in Regulatory Issue Summary (RIS) 2001-02 (Reference 2). Special circumstances are defined in the above guidance as "conditions or situations that would raise questions about whether there is adequate protection, and that could rebut the normal presumption of adequate protection from compliance with existing requirements. In such situations, undue risk may exist even when all regulatory requirements are satisfied."

Though power uprates are not submitted as risk-informed license applications, it is recognized that there are potential risk increases associated with implementing a power uprate due to the increased heat loads at higher powers and the resulting reductions in the times available to perform specific accident response actions. In addition, there can be impacts on the equipment loads and the potential for an increase in the frequency of reactor scrams due to these increased loads and tighter operating margins. For small power uprates (i.e., those referred to as measurement uncertainty recapture power uprates and stretch power uprates), the risk increases are expected to be exceedingly small. However, notwithstanding any plant modifications that could reduce risk, some increase in risk is expected for larger power uprates. Depending on the type of plant-specific modifications necessary to implement the larger power uprates, these power uprates have the potential for significantly increasing plant risks, especially if they significantly impact initiating event frequencies, component reliabilities, system success criteria, and/or operator response times. Further, large power uprate requests are specifically identified in Appendix D to SRP Chapter 19 as an example of the type of situation that might create "special circumstances" since they could "involve changes for which the synergistic or cumulative effects could significantly impact risk." Therefore, the Probabilistic Safety Assessment Branch (SPSB) Safety Program Section formally reviews all license application submittals for extended power uprates.

As of December 2002, the SPSB Safety Program Section staff had performed risk reviews of eight extended power uprate license applications involving twelve units. All but one of these applications were for boiling water reactors (BWRs) of various design vintages, including: five BWR-3/Mark-I units (Monticello, Dresden 2 and 3, and Quad Cities 1 and 2), five BWR-4/Mark-I units (Hatch 1 and 2, Duane Arnold, and Brunswick 1 and 2), and one BWR-6/Mark III unit (Clinton). The one pressurized water reactor (PWR) extended power uprate license application

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was for a Combustion Engineering (CE) plant with a large dry containment (Arkansas Nuclear One - Unit 2). The extended power uprates have been as high as 20 percent of original licensed thermal power.

The staff, recognizing the need to address the potential risk increase associated with extended power uprates, stated in a 1996 position paper (Reference 3) that licensees should conduct risk evaluations for extended power uprate license applications. Specifically, the paper states that it is appropriate for each applicant to assess the effect of the proposed power uprate on the results of its independent plant examination (IPE)/probabilistic risk assessment (PRA) and that this assessment should cover the potential impacts on initiating event frequencies, success criteria, component failure rates, and the time available for operator actions and equipment restoration. The paper also states that these inputs and assumptions are examples of the appropriate areas of the IPE/PRA for review and expects that applicants will address any other areas that the applicants determine also may be affected by power uprate. Finally, the paper states that the staff will request that each applicant report the effects of the proposed uprate on its core damage frequency and frequencies of large magnitude radioactive release and indicates that this process may be as simple as reporting that the applicant's review of its IPE/PRA found that none of the items previously discussed are changed as a result of the uprate; but it may be as complex as reevaluating the logic model to obtain new dominant cutsets that reflect the significant changes in multiple IPE/PRA assumptions and inputs.

In September 1998, the staff proposed guidelines for the staff's risk review of power uprates (Reference 4). These guidelines, as well as the guidance in Appendix D of SRP Chapter 19, have formed the basis and focus for the current risk reviews of power uprate license applications. The lessons learned from past power uprate reviews have been integrated into the development of this guidance and in establishing the staff's expectations for future reviews of extended power uprate license applications.

This guidance is provided to aid the staff in conducting the risk review of a licensee's application for an extended power uprate, leading up to a determination regarding the potential for the existence of "special circumstances," as defined by Appendix D of Chapter 19 of the SRP. Specific guidance is provided for the scope of the review, the risk information needed to perform the review, the staff review guidance to use in determining the acceptability of the license application and in determining if special circumstances may exist that would warrant invoking the special circumstances notification and review process of Appendix D to SRP Chapter 19, and the review process and documentation requirements for this risk review.

2. SCOPE OF REVIEW

Consistent with SRP Chapter 19 and Regulatory Guide (RG) 1.174 (Reference 5), the licensee's risk analyses used to support a license application and the level of detail of the staff review of those analyses, should be commensurate with the role that the risk results play in the utility's and staff's decisionmaking processes and should be commensurate with the degree of rigor needed to provide a valid technical basis for the staff's decision. As for extended power uprates, the licensees do not request the relaxation of any deterministic requirements for their proposed power uprates and the staff's approval is primarily based on the licensee meeting the current deterministic engineering requirements.

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Thus, the purpose of the staff's risk review is to determine if there are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the deterministic requirements in the regulations. Such issues could represent the "special circumstances" that would call for a more detailed risk review to determine the acceptability of the extended power uprate license application. These reviews can involve an extensive level of effort depending upon the required plant modifications to implement the extended power uprate, the plant-specific features and/or vulnerabilities, and the quality of the licensee's supporting analyses. These reviews need to address the risk impacts to core damage frequency (CDF) and large early release frequency (LERF) due to internal events, external events, and shutdown operations. In addition, these reviews need to address the quality of the licensee's analyses that are used to support the license application, including addressing any issues or weaknesses that may have been raised in the previous staff reviews of the licensee's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE) or by an industry peer review. Further, if the licensee's results indicate a significant risk impact or if there are significant questions regarding the licensee's supporting analyses, a site audit of these areas may be deemed appropriate. A site audit might also be performed to resolve PRA quality questions by auditing the licensee's PRA-related procedures and processes and reviewing their evaluations and resolutions of previous PRA reviews, including the IPE, IPEEE, and industry peer review findings.

If special circumstances are identified, additional information and analyses beyond those identified in this guidance may be needed for the staff to be able to determine the acceptability of the license application. This may result in the licensee and/or staff obtaining more detailed information to support performing detailed quantitative analyses (e.g., perform seismic PRA instead of reliance on seismic margins analysis or perform shutdown PRA instead of reliance on shutdown outage risk management guidance) to determine the acceptability of the license application. This guidance does not address these review details, which should be mainly focused on the issue(s) creating the circumstances and other considerations as directed by NRC management per the process described in Appendix D of SRP Chapter 19.

3. RISK INFORMATION NEEDED FOR REVIEW

The guidance in this section addresses the information needed by the staff to evaluate the acceptability of the risks and to determine if the potential for special circumstances exist.

3.1 Internal Events Risk Information

The licensee needs to address the risk impacts to the internal events analyses associated with implementing the extended power uprate. Specifically, the licensee needs to address the impacts of the extended power uprate on initiating event modeling and frequencies, component and system reliability and response times, operator response times and associated error probabilities, and functional and system-level success criteria, as well as the overall impact of internal events on CDF and LERF. The discussion of the impacts due to the extended power uprate should include an explanation of why the impacts occur and, where applicable, the quantification of these impacts (e.g., the reduction in operator response timing and revised operator error probabilities).

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In addition, if there are any impacts on the PRA results from any other areas that either are affected by the power uprate or are being implemented in parallel with the power uprate (e.g., emergency operating procedure changes, changes in maintenance activities or approach, turbine trip setpoint changes, improved turbine bypass capability, condensate/feedwater modifications or operational changes, main transformer modifications, increased burnup, and longer cycles), then the potential impact of these changes also needs to be addressed. For example, if there is a plant modification associated with the uprate that may affect an initiating event (e.g., addition of automatic recirculation system runback on feedwater pump trip), then the initiating event (e.g., loss of feedwater) may need to be explicitly modeled to account for new potential impacts (e.g., spurious runback at full power or failure to runback upon feedwater pump trip). If generic or plant-specific data are used to derive the initiating event frequency, instead of using an explicit model, then the applicability of the data to the new operating conditions will need to be justified. Further, note that the new operating conditions may also impact the top-level, functional plant response (i.e., event tree) modeling. This may then necessitate revising the modeling of and inputs to the best estimate thermal-hydraulic code used to support the development of functional and/or system-level success criteria. The licensee's submittal would also need to describe these modeling, supporting analyses, and success criteria impacts.

The licensee also needs to address the scope, level of detail, and quality of their PRA and other relied upon evaluations (e.g., thermal-hydraulic analyses) used to support their determination that the plant risk is acceptable. The licensee should describe how they ensure that the PRA adequately models the as-built, as-operated plant and that the analyses supporting the extended power uprate adequately reflect how the plant will be operated and configured for the extended power uprate plant conditions. This discussion should specifically address any vulnerabilities, weaknesses, or review findings identified in the IPE, the staff safety evaluation reports or contractor technical evaluation reports on the IPE, and/or any independent/industry peer review findings that could impact the PRA results and conclusions pertinent to this application. The licensee's information needs to be sufficient for the staff to conclude that their PRA and other relied upon evaluations adequately reflect the as-built, as-operated plant for the specific extended power uprate license application.

It is expected that, if a peer review has been performed on the PRA, the licensee will present the overall findings of the review (by element) and discuss any elements that were rated low (e.g., less than a 3 on a scale of 1 to 4) and any findings and observations that could potentially impact the licensee's proposed extended power uprate. To address these findings and observations, the licensee may need to perform sensitivity calculations that address the specifically identified weaknesses (e.g., removing credit for equipment repair and recovery). In addition, if the licensee's IPE/PRA took credit for modifications or improvements that had not been implemented, then the licensee needs to explicitly address these conditions. For these areas, the licensee needs to indicate if the improvements have been implemented in accordance with the assumptions and conditions identified in the IPE/PRA. If they have not been implemented, then the licensee needs to provide either a qualitative or quantitative justification for the acceptability of the existing situations for the post-uprate plant conditions.

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In addition, some licensees have performed their evaluations of the risk impacts of the extended power uprate prior to having fully determined the plant modifications that will be implemented. In these situations, the licensee needs to justify that their evaluations properly address the potential risk impacts due to the extended power uprate. If there are some modifications that are proposed that may not be implemented (i.e., the final decision of making the modification has not been made or the licensee may wait to see how the equipment performs at uprated power conditions before deciding if a change is needed), then a sensitivity calculation of the risk impacts assuming these modifications are not implemented should be performed. If the design of a modification has not been established at the time of the risk evaluation, then the licensee needs to justify that the assumed design features and resulting failure probabilities bound the proposed modification. Again, a sensitivity calculation may be used to show the impact of different design modifications and/or failure probabilities. If multiple sensitivity calculations are performed to address the above situations, then there should be at least a combination sensitivity calculation performed that combines the adverse impacts of the individual sensitivity calculations.

If the estimated change in CDF and/or LERF, or base CDF and/or LERF, exceeds the RG 1.174 guidelines, including the results of any sensitivity calculations, the licensee should provide a more detailed justification to support the acceptability of implementing the extended power uprate. The licensee's information needs to be sufficient for the staff to conclude that the risk impact from internal events is acceptable and does not create special circumstances.

3.2 External Events Risk Information

The licensee needs to address the risk impacts from external events associated with implementing an extended power uprate. Based on previous reviews, the main issues have involved the analyses and assumptions that date back to the original IPEEE in which credit was taken for plant modifications that had not yet been performed (e.g., taking credit for fixing low-capacity seismic outliers or re-routing cables to eliminate them from certain rooms). Another issue that has been identified is related to the licensee's use of non-PRA type methods in performing their analyses (e.g., margins or vulnerability type analyses). To resolve some of these issues, licensees have had to provide additional information, including performing additional analyses or simplified risk calculations, to show that the risks associated with these outliers or vulnerabilities are acceptable under both current and uprated power conditions. In addition, the staff has performed some simplified calculations, based on licensees' seismic margins analysis results, to provide a quantitative seismic risk perspective.

If the licensee has a PRA for some external events, the licensee should describe the risk impacts for these external events associated with implementing the extended power uprate and demonstrate that the calculated risk contribution is acceptable. However, if the licensee does not have a PRA for some external events, such as if a margins-type analysis was performed as part of their IPEEE, they should describe how the extended power uprate affects these external events analysis results and conclusions.

The licensee also needs to address the scope, level of detail, and quality of their external events PRA and/or other relied upon evaluations (e.g., seismic margins analysis) used to support their determination that the risk is acceptable. The licensee should describe how they ensure that the analyses adequately represent the as-built, as-operated plant and that the analyses supporting the extended power uprate adequately reflects how the plant will be

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operated and configured for the extended power uprate plant conditions. Further, if vulnerabilities, outliers, anomalies, or weaknesses were identified in their IPEEE, the associated IPEEE staff safety evaluation reports, IPEEE contractor technical evaluation reports, or industry peer reviews or if the licensee took credit for plant modifications that had not been implemented when the analysis was conducted (e.g., seismic A-46 modifications), the licensee should identify these conditions, how they have resolved these conditions for the extended power uprate, and demonstrate, either quantitatively or qualitatively, that the risk associated with these external events are acceptable. This may involve performing additional analyses or simplified risk calculations that address the specifically identified weaknesses or evaluates the risk implications of the existing conditions (e.g., removing credit for seismic modifications not implemented). The licensee's information needs to be sufficient for the staff to conclude that their external events analyses adequately reflect the as-built, as-operated plant for the specific extended power uprate license application.

If the estimated risk contributions exceed the RG 1.174 guidelines, including the consideration of the existence of a potential vulnerability that is identified in a margins-type analysis or if new potential vulnerabilities are introduced by the extended power uprate, the licensee should provide a more detailed justification to support the acceptability of implementing the extended power uprate. The licensee's information needs to be sufficient for the staff to conclude that the risk from external events is acceptable and does not create special circumstances.

3.3 Shutdown Operations Risk Information

The licensee needs to address the risk impacts on shutdown operations associated with implementing the extended power uprate and describe the plant's shutdown risk management philosophies, processes, and controls that are relied upon to ensure that the risk impacts of the extended power uprate on shutdown operations is not significant. Based on previous reviews, an extended power uprate typically impacts shutdown operations due to the greater decay heat under these conditions, which causes longer times to reach shutdown, longer times before alternative decay heat removal systems can be used, shorter times to boiling, and shorter times for operator responses.

If the licensee has a shutdown PRA, the licensee should describe the risk impacts associated with implementing the extended power uprate and demonstrate that the calculated risk contribution is acceptable. The licensee should specifically address any changes in initiating event frequencies, component reliability, success criteria, and operator actions that are caused by the extended power uprate. However, most licensees do not have a shutdown PRA. If the licensee does not have a shutdown PRA, they should discuss how the extended power uprate affects shutdown risks, how they manage and control these risks, and address any critical or time-limited conditions to demonstrate that these risks are not significant and are properly managed and controlled at the extended power uprate conditions.

The licensee also needs to address the scope, level of detail, and quality of their shutdown PRA and/or other relied upon evaluations (e.g., outage risk management guidance) used to support their determination that the risk impacts associated with extended power uprate are acceptable. The licensee should describe how they ensure that their approach and/or analyses adequately represent the as-built, as-operated plant and that it reflects how the plant will be operated and configured for the extended power uprate plant conditions. The licensee's information needs to be sufficient for the staff to conclude that their analysis of shutdown operations adequately

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reflects the as-built, as-operated plant for the specific extended power uprate license application.

If the estimated risk contributions exceed the RG 1.174 guidelines, including the consideration of potential vulnerabilities, weaknesses, or limitations in the licensee's shutdown risk management approach or if new potential vulnerabilities are introduced by the extended power uprate, the licensee should provide a more detailed justification to support the acceptability of implementing the extended power uprate. The licensee's information needs to be sufficient for the staff to conclude that the risk impact of the extended power uprate for shutdown operations is acceptable.

4. REVIEW GUIDANCE

Consistent with the current guidance, the appropriate starting points for determining if the potential for special circumstances exists are the acceptance guidelines provided in RG 1.174. This evaluation should address the risks from internal events, external events, and shutdown operations. However, since the review is primarily directed towards determining if adequate protection is challenged, the focus should be primarily on the base risk evaluations (i.e., CDF, LERF, and no potential vulnerabilities identified from a margins-type analysis) as opposed to the change in risk evaluations (i.e., ΔCDF and ΔLERF). While the primary focus is the base risk evaluation, it is still important to assess the change in risk to understand the magnitude of the risk increase associated with the extended power uprate. Large base risk values or large changes in risk values that surpass the RG 1.174 acceptance guidelines should warrant additional staff scrutiny of the analyses, results, and quality of the licensee's analyses. This would be a factor in determining the need to conduct a site audit of the licensee's PRA and/or their PRA management procedures and processes. If the staff determines that the base risk values are significantly beyond the RG 1.174 acceptance guidelines, then this should invoke the special circumstances process of Appendix D of SRP Chapter 19.

To determine that the analyses used in support of the license application are of sufficient quality, scope, and level of detail, the staff should evaluate the information provided by the licensee using the guidance provided in RG 1.174, as well as consider the staff's previous reviews on the licensee's IPE and IPEEE submittals and the conclusions and findings of any industry or independent peer reviews. The staff needs to be assured that the relied upon analyses adequately reflect the as-built, as-operated plant.

All licensees have at least a Level I internal events PRA, but most licensees do not have a fully integrated PRA that addresses internal events, external events, and shutdown operations. Further, the analyses that are performed for many external events and shutdown operations either are not quantitative in nature or are screening/vulnerability-type analyses that are not performed to the same level of depth and rigor as the internal events analyses. Therefore, the staff may need to rely on some general figures of merit or simplistic calculations to provide a more comprehensive perspective of the potential risks associated with a licensee's extended power uprate application.

For example, in addressing the risk impacts for shutdown operations in the absence of a licensee's shutdown PRA, the review staff should refer to SECY 97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," in which the staff provides estimates of shutdown risk for various interpretations of the industry guidance. The risk estimates cited in SECY 97-168 were not meant to bound plant operations,

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but were intended to be examples of reasonable interpretations of industry guidance. Depending on the specific licensee's approach to managing shutdown risks, an estimate of the magnitude of the risk for shutdown operations can be determined using SECY 97-168. An example of this review approach is provided as Attachment 2 to Matrix 13 of RS-001.

As a further example, in addressing the risk impacts related to seismic events for situations in which the licensee has performed a seismic margins analysis instead of a seismic PRA, the review staff may need to perform a simplistic calculation to determine the magnitude of the seismic risk. An approximation method is provided in a paper by Robert P. Kennedy entitled *"Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations,"* (Reference 6) that uses the plant's high confidence of a low probability of failure (HCLPF) value that is determined by the licensee's seismic margins analysis and the site's seismic hazard curve that is based on NUREG-1488 (Reference 7) to derive an approximation of the magnitude of the risk associated with seismic events. An example of this calculation is provided as Attachment 3 to Matrix 13 of RS-001.

The results of these simplistic approaches should not be used as the sole basis for determining the acceptability of a license application, but rather should be used to gain perspective into the risks associated with these events/operations, insights into the licensee's management of these risks, and a focus for areas that may warrant further review or may indicate the potential for special circumstances. If these results indicate the potential for significantly exceeding the RG 1.174 acceptance guidelines (i.e., indicating the potential existence of special circumstances), then the staff should pursue these risk aspects further with the licensee and seek more information and analyses to more accurately define these risk contributors. If the licensee cannot or will not be able to provide the additional information or analyses in a timely fashion, then the staff should progress in its review of the risk information and notify management of this potential for special circumstances.

If issues are identified that could rebut the presumption of adequate protection (i.e., special circumstances), the process delineated in Appendix D of Chapter 19 of the SRP should be implemented. This process is also described in Regulatory Issue Summary (RIS) 2001-02, "Guidance on Risk-Informed Decisionmaking in License Amendment Reviews," and includes informing/engaging the licensee and NRC management regarding the risk concern, obtaining management approval to request additional risk information, and evaluating this risk information to determine if there is reasonable assurance of adequate protection. If NRC management agrees with the staff that special circumstances appear to exist, there is also direction to notify the Commission of this decision. The rationale that led to the expansion of the depth of the review, as well as the findings of the associated review, should be documented in the staff's safety evaluation.

5. RISK REVIEW PROCESS AND DOCUMENTATION

The SPSB Safety Program Section staff should document their review activities associated with extended power uprate license applications through the issuance of a safety evaluation, which, upon management approval, is subsequently transmitted to the responsible project manager to incorporate into the NRC safety evaluation report on the license application. The review activities leading up to the development of the staff safety evaluation are described in this section.

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In initiating the risk review, the staff should first perform an "acceptance review" of the information provided by the licensee. The acceptance review should ensure that the licensee's submittal contains enough information for the reviewer to evaluate the application in accordance with Section 3 of this guidance. The information provided by the licensee needs to be sufficient for the staff to be able to make a determination regarding special circumstances, based on the guidance described in Section 4. If the licensee's information, provided in accordance with Section 3 of this guidance, combined with any staff independent and/or simplified calculations, performed in accordance with Section 4 of this guidance, indicates that the overall plant risks are well below the acceptance guidelines of RG 1.174 and that there are no special circumstances, the staff need not develop a detailed safety evaluation. Instead, the staff may provide an abbreviated safety evaluation that documents that the licensee's submittal, combined with any staff independent and/or simplified calculations, has adequately addressed the risks associated with the extended power uprate and that these risks have been shown to be acceptably small.

If the staff identifies any issues with the licensee's submittal or needs to clarify any information provided by the licensee, then the staff should pursue these areas initially through the issuance of requests for additional information (RAIs). Some issues, such as a lack of information about expected risk contributors or differences between the supporting analyses and the actual plant operations, may be resolved through RAIs or by conducting a site audit of the licensee's pertinent documentation and/or processes, without needing to invoke the process for special circumstances. If issues are identified that could indicate the potential for special circumstances, then these issues should be elevated to management as early as possible during the staff review since such a determination may invoke a detailed review process and mean that the project schedules and staff-hour estimates will need to be revised.

Through the staff reviews, a number of issues may be identified with respect to specific aspects of the risk analyses used to support a licensee's application for an extended power uprate. The main issues that have been identified have involved the change in risk calculation when bounding or conservative values are used in the base risk model and the reliance on external events analyses and assumptions that date back to the original IPEEE (e.g., taking credit for fixing low-capacity seismic outliers or re-routing cables to eliminate them from certain rooms). In some of these cases, the licensee has had to provide additional information, including performing additional analyses or simplified calculations, to make the relied upon analyses more reflective of the actual plant conditions and to show that the associated risks are acceptable under both current and uprated power conditions. However, being a non-risk-informed submittal review, the staff focus is primarily on determining if there are any conditions associated with implementing the extended power uprate that would significantly alter the current practices of the licensees or create new vulnerabilities, such that issues are raised that could rebut the presumption of adequate protection provided by meeting the deterministic requirements and regulations. If these circumstances arise, the staff should seek to perform a more in-depth review to determine the appropriateness of accepting the extended power uprate license application or if there would be grounds warranting denial of the licensee's application for an extended power uprate. However, if the identified issues do not raise adequate protection questions, the issues should be documented in the safety evaluation and clearly explained as why they do not rise to this level of concern.

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The staff safety evaluation should address the staff's findings and conclusions for each of the major review areas (i.e., internal events, external events, and shutdown operations), including the quality of the licensee's analyses supporting these areas (i.e, PRA, margins-type analyses, vulnerability assessments, etc.), and if any issues were identified that could potentially create special circumstances. The results of any detailed review called for in response to a determination of special circumstances should also be documented in the safety evaluation. In performing the review, the staff may also identify issues related to the licensee's supporting analyses that do not affect the determination regarding special circumstances for the extended power uprate license application. These issues should be identified within the staff safety evaluation, with an explanation as to why they do not impact the extended power uprate license application.

In addition to the primary task of performing the risk review, the Safety Program Section staff may be requested by other NRC technical review branches to provide risk analyses and/or insights to support the evaluations of potential impacts that are identified in these other branches' review areas. The results associated with these requested evaluations should be integrated directly within the safety evaluations of the technical branch(es) that requested the support. Thus, there should not be a separate input from the SPSB Safety Program Section in these requested support areas, unless it impacts the staff risk review findings.

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6. REFERENCES

1. U.S. Nuclear Regulatory Commission, *Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance*, NUREG-0800, Standard Review Plan Chapter 19.0, Revision 1, December 2002.

2. U.S. Nuclear Regulatory Commission, *Guidance on Risk-Informed Decisionmaking in License Amendment Reviews*, Regulatory Issue Summary 2001-02, January 18, 2001.

3. Letter from Dennis M. Crutchfield (NRC) to G. L. Sozzi (GENE), *Staff Position Concerning General Electric Boiling Water Reactor Extended Power Uprate Program*, February 8, 1996.

4. Memorandum from Richard J. Barrett (NRC) to Gary M. Holahan (NRC), *Proposal for a Guideline on Risk-Informed Staff Review of Power Uprate*, September 22, 1998.

5. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, July 1998.

6. Kennedy, R.P., *Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations*, Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August 1999.

7. U.S. Nuclear Regulatory Commission, *Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains*, NUREG-1488, April 1994.

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ATTACHMENT 2 TO MATRIX 13

Example Staff Review of Shutdown Risk Based on SECY 97-168

Risk Evaluation

In SECY 97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," the staff provided two estimates of pressurized water reactor (PWR) shutdown risk, which credited equipment required by technical specification (TS) and equipment recommended to be available based on guidance from generic letter (GL) 88-17 and NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." These two "voluntary action cases" represent different interpretations of NUMARC 91-06 and GL 88-17. These two cases were not meant to bound plant operations, but were intended to be examples of reasonable interpretations of industry guidance. These two cases cover cold shutdown operations and refueling operations until the refueling cavity is flooded. Reduced inventory operations are a subset of this condition.

The high core damage frequency (CDF) voluntary action case represents a minimal level of implementation of both guidance documents in terms of the amount of extra equipment and additional sources of water being made available. For PWRs, the higher CDF voluntary action case includes the equipment credited by TS, based on Westinghouse standard TS, plus one emergency core cooling system (ECCS) pump, gravity feed, and an "available" containment. An "available" containment is defined as one that can be closed by remote or local manual actions before containment conditions become intolerable. The high case had a CDF estimate of 8E-5/year.

The low CDF voluntary action case represents a more in-depth implementation of both guidance documents. The lower CDF case adds an additional emergency diesel generator (EDG) or equivalent power source, a second ECCS pump, containment spray pumps to supplement the residual heat removal (RHR) pumps, and an enhanced recirculation capability. The low case had a CDF estimate of 2E-6/year

Based on the licensee's shutdown cooling control procedures, the operators should have a high pressure safety injection (HPSI) flow path available at all times unless the reactor vessel is defueled. During reduced inventory operations, the licensee maintains a second flow path in addition to the HPSI flow path. However, based on conversations with the licensee (as documented in meeting summaries, notes to file, etc.), the second flow path may be a small charging pump that may not have the capability to keep the core covered following a loss of inventory event that includes a loss of both the RHR flow path, which is the normal means of decay heat removal, and the HPSI flow path.

Concerning the licensee's containment closure capability, the outage risk management guidelines (ORMGs) allow for a containment breach that cannot be closed prior to the estimated time to boiling. However, the licensee maintains that such a breach would not be incorporated into the outage schedule and, based on discussions with the licensee (as documented in meeting summaries, notes to file, etc.), such breaches would be unanticipated and/or inadvertent. The small increase in decay heat due to the proposed extended power

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uprate (EPU) will reduce the time available for operator actions, such as to achieve containment closure. However, even for the containment breach that takes the longest to close (i.e., the equipment hatch), the licensee has demonstrated that such a breach can be closed within 5 minutes to 15 minutes, and the estimated time to boiling would be greater than 18 minutes for EPU conditions. For the pre-EPU conditions, the time to boil is estimated at over 20 minutes. Therefore, the operator's ability to inject before core damage and close containment before boiling should not be significantly changed, since (1) there is margin between the longest time needed to close containment and the time to boiling, (2) the operators regularly calculate the time to boiling, and (3) the licensee maintains the availability of the core exit thermocouples to monitor reactor coolant system (RCS) temperature until preparations for vessel head removal.

Based on the staff's review of the licensee's shutdown mitigation capability provided by the licensee's responses to the staff's requests for additional information, the licensee's shutdown mitigation capability appears to be closer to the high CDF voluntary action case.

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ATTACHMENT 3 TO MATRIX 13

Example Staff Review of Seismic Risk Using Simplified Calculations

Risk Evaluation

The safety evaluation report (SER) on the licensee's individual plant examination of external events (IPEEE) indicated, based on the staff's screening review, that the licensee's process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. Therefore, as set forth in that SER, the staff concluded that the licensee had met the intent of Supplement 4 to Generic Letter (GL) 88-20. For the IPEEE seismic analysis, the licensee's plant is categorized as a 0.3g focused-scope plant, per NUREG-1407. The licensee performed the seismic evaluation using the Electric Power Research Institute (EPRI) seismic margins analysis (SMA) methodology, as described in EPRI NP-6041-SL.

As a consequence of using the EPRI SMA methodology, the licensee did not quantify a seismic core damage frequency (CDF). However, the licensee states in their supplemental information for the extended power uprate (EPU) license amendment that the conclusions and results of the SMA were judged to be unaffected by the EPU. Further, they state that the EPU has no impact on the seismic qualifications of the systems, structures, and components. Specifically, the EPU results in additional thermal energy stored in the reactor pressure vessel (RPV), but the additional blowdown loads on the RPV and containment given a coincident seismic event are judged not to alter the results of the SMA.

The SER on the IPEEE indicates that the licensee had implemented a number of improvements during the resolution of unreviewed safety issue (USI) A-46 and that a number of additional improvements were still under consideration. The licensee indicated that any necessary design changes to address these items would be completed in conjunction with the approved schedule for resolution of the USI A-46 outliers. In particular, the SER states that the licensee was developing a concept for providing a seismically-qualified/verified make-up path for a particular accident scenario. The licensee's IPEEE SMA took credit for this plant modification and related operational changes needed to implement the seismically-qualified/verified make-up feature. However, these plant modifications had not been implemented at the time of the original EPU license amendment submittal. Thus, it appears that the IPEEE SMA does not accurately represent the as-built, as-operated plant. Therefore, the staff requested that the licensee augment their IPEEE SMA by performing some simplified seismic risk evaluations of the current and EPU plant configurations for the outlier scenario (i.e., non-seismically qualified make-up source). In addition, the staff performed an independent simplistic calculation to estimate the magnitude of the seismic risk associated with the identified outlier condition.

Although the IPEEE indicates that it is a 0.3g focused-scope SMA, this scenario involves equipment with a high confidence of a low probability of failure (HCLPF) value that is much lower than 0.3g. The scenario involves a seismic event that involves the failure of the non-seismically-qualified makeup source, which has a HCLPF value of 0.15g peak ground acceleration (PGA). The licensee's results indicate that the current, pre-uprate plant and the

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EPU plant CDF values for this scenario are both about 1E-5/year, with a change in risk due to the uprate of about 1E-8/year.

The staff used the approximation method provided in a paper by Robert P. Kennedy entitled *"Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations."* This approach uses the plant's HCLPF value that is determined by the licensee's SMA and the site's seismic hazard curve that is based on NUREG-1488 to derive an approximation of the magnitude of the risk associated with seismic events. The staff's independent simplistic calculation used a plant HCLPF value of 0.15g PGA, since that is the HCLPF of the non-seismically-qualified makeup source, and the recommended logarithmic standard deviation of 0.4. Using these values, the seismic CDF for the outlier scenario is estimated to be approximately 1.7E-5/year. The seismic risk associated with the remainder of the plant having a HCLPF at 0.3g PGA using the same approach is about 3.1E-6/year. Thus, based on the staff's approximation, the total seismic CDF is estimated to be about 2E-5/year.

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SECTION 3

DOCUMENTATION OF REVIEW

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3.1 Documenting Reviews of Extended Power Uprate Applications

This section includes two template safety evaluations for use in generating plant-specific safety evaluations: one for boiling-water reactor (BWR) plants and one for pressurized-water reactor (PWR) plants. These template safety evaluations were developed consistent with NRR Office Instruction LIC-101, "License Amendment Review Procedures."

When preparing plant-specific safety evaluations, Project Managers have the lead for completing Sections 1.0, 3.0, 4.0, 6.0, 7.0, 8.0, and 9.0 of the template safety evaluation. Reviewers with primary review responsibility identified in the matrices in Section 2.1 of this review standard have the lead for completing the subsections of Section 2.0 of the template safety evaluations that correspond to the areas within their branch's primary review responsibility. Reviewers with primary review responsibility also have the lead for completing Section 5.0 of the template safety evaluation. Project Managers are responsible for preparing and finalizing the plant-specific safety evaluation, including consolidating the inputs received from other branches.

When preparing plant-specific safety evaluations, follow the instructions below.

(1) Use the applicable template safety evaluation in Section 3.2 (for BWRs) or Section 3.3 (for PWRs) of this review standard.

(2) Replace the information within the brackets with applicable plant-specific information.

(3) Based on the results of the technical review performed in accordance with Section 2.1 of this review standard, for each technical area of the template safety evaluation where the design basis of the plant has been identified as different from the guidance provided in the documents referenced in the "SRP Section Number" and "Other Guidance" columns of the matrices, modify the "Regulatory Evaluation" and "Conclusion" sections to be consistent with the design basis of the plant. [Note: This is most likely to occur with respect to the General Design Criteria (10 CFR Part 50, Appendix A), which may need to be replaced by plant-specific principal design criteria (PDC). The PDC are usually based on proposed draft GDC.] Ensure that the changes are written consistent with the format and content of the template safety evaluation.

(4) Based on the results of the technical review performed in accordance with Section 2.1 of this review standard, if additional technical areas beyond those identified in the matrices in Section 2.1 of this review standard are necessary, address the additional technical areas under the "Additional Review Areas" subsection of the appropriate section of the safety evaluation. Provide a regulatory evaluation, technical evaluation, and conclusion for each of the additional technical areas. Ensure the additional sections are written consistent with the format and content of the template safety evaluation.

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(5) Based on the results of the technical review performed in accordance with Section 2.1 of this review standard, if a technical area is determined to not be applicable or necessary for the plant under review, keep that section's heading in the safety evaluation, delete the "Regulatory Evaluation" and "Conclusion" sections for that area, and discuss the reasons why a review of that particular technical area is not needed.

(6) Summarize the technical review and findings in the appropriate "Technical Evaluation" section of the safety evaluation.

(7) Discuss independent calculations performed to support the review in the appropriate "Technical Evaluation" section of the safety evaluation.

(8) Review the "Conclusion" sections of the safety evaluation and modify them, as necessary, to reflect the conclusions reached as a result of the review. If a "Conclusion" section summarizes more than one technical evaluation, include an intermediate conclusion in each technical evaluation (e.g., see Section 2.2.2 of Insert 2 for RS-001 Section 3.2 - BWR Template Safety Evaluation).

(9) Identify areas for consideration by the NRC's inspection staff in the "Recommended Areas for Inspection" section of the safety evaluation. Each area identified should include a rationale. The identified areas are not intended to be inspection requirements, but are provided to give the inspectors insight into important bases for approving the EPU.

(10) Generate a detailed table of contents for the final plant-specific safety evaluation. The detailed table of contents should include a listing of all areas addressed within each insert.

(11) Modify, as necessary, the acronym list that is attached to the template safety evaluation to ensure that it accurately reflects the acronyms defined in the plant-specific safety evaluation.

It may be necessary to modify the license to include license conditions to capture certain future licensee actions discussed in the EPU application. These actions are typically included as commitments in the EPU application and may include things such as plant modifications, analyses, and updates to licensee-controlled documents. In addition, in cases where a licensee proposes to implement the EPU in multiple stages, it may be appropriate to modify the license to include license conditions to limit plant operation to lower than the full EPU power level pending completion of certain actions. To determine if such actions are appropriate for inclusion in the license as license conditions, refer to the guidance in NRR Office Instruction LIC-101, "License Amendment Review Procedures."

For EPUs to be implemented in one stage, the PM should consider including conditions in the implementation section of the amendment to appropriately capture near-term licensee actions meeting the threshold for inclusion in the license as license conditions.

For EPUs to be implemented in multiple stages, the PM should consider including conditions in the license to appropriately capture longer-term licensee actions meeting the threshold for inclusion in the license as license conditions. Including these actions in the license is appropriate due to the licensee's extended schedule for implementing the EPU.

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SECTION 3.2 of RS-001

TEMPLATE SAFETY EVALUATION

for

BOILING-WATER REACTOR EXTENDED POWER UPRATE

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Attachment: List of Acronyms

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. ______ TO FACILITY OPERATING LICENSE NO. **[XXX-XX]**

[NAME OF LICENSEE]

[NAME OF FACILITY]

DOCKET NO. 50-**[XXX]**

1.0 INTRODUCTION

1.1 Application

By application dated **[ ]**, as supplemented by letter**[s]** dated **[ ]**, the **[Name of Licensee]** (the licensee) requested changes to the Facility Operating License and Technical Specifications (TSs) for the **[Plant Name]**. The supplemental letter[s] dated **[ ]**, provided additional clarifying information that did not expand the scope of the initial application and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on **[date] (XX FR XXXX)**.

The proposed changes would increase the maximum steady-state reactor core power level from **[current licensed power level]** megawatts thermal (MWt) to **[power level proposed by the licensee]** MWt, which is an increase of approximately **[##]** percent. The proposed increase in power level is considered an extended power uprate (EPU).

1.2 Background

[Plant Name] is a boiling-water reactor (BWR) plant of the BWR/**[#]** design with a Mark-**[#]** containment. **[Plant Name]** has the following special features/unique designs:

[Insert any special features/unique designs]

The NRC originally licensed **[Plant Name]** on **[date]** for operation at **[original licensed power level]** MWt. **[By Amendment No. [###] dated [ ], the NRC granted a power uprate to [Plant Name] of [##] percent, allowing the plant to be operated at [current licensed power level] MWt.] Therefore, the proposed EPU would result in an increase of approximately [##] percent over the original licensed power level [and [##] percent over the current licensed power level] for [Plant Name].]**

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1.3 Licensee's Approach

The licensee's application for the proposed EPU follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, "Review Standard for Extended Power Uprates," to the extent that the review standard is consistent with the design basis of the plant. Where differences exist between the plant-specific design basis and RS-001, the licensee described the differences and provided evaluations consistent with the design basis of the plant. The licensee also used **[Identify topical reports or other documents used by the licensee for guidance related to the scope of the proposed EPU; NRC staff approvals, ranges of applicability, any limitations/restrictions associated with the documents; and consistency of the licensee's application with the ranges of applicability and limitations/restrictions. The discussion in this section is to cover topical reports and other documents referenced for the overall power uprate process. It is not intended to cover topical reports and other documents for specific methods of analyses. Topical reports and other documents referenced for specific methods of analyses are to be covered in the applicable technical evaluation section of this safety evaluation]**.

Insert this sentence if the licensee is planning to implement the EPU in one stage.
[The licensee plans to implement the EPU in one step. The licensee plans to make the modifications necessary to implement the EPU during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt starting in Cycle [##].]

Insert this paragraph if the licensee is planning to implement the EPU in stages:
[The licensee plans to implement the EPU in [#] steps of [## and ##] percent. The licensee plans to make modifications necessary to implement the first step during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt during Cycle [##]. The remainder of the modifications will be completed during the refueling outage in [season year (e.g., fall 2003)], with subsequent operation at [##] MWt starting in Cycle [##].]

1.4 Plant Modifications

The licensee has determined that several plant modifications are necessary to implement the proposed EPU. The following is a list of these modifications and the licensee's proposed schedule for completing them.

[Provide a list of plant modifications.]

The NRC staff's evaluation of the licensee's proposed plant modifications is provided in Section 2.0 of this safety evaluation.

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1.5 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed EPU on design-basis analyses. The NRC staff evaluated the licensee's application and supplements. The NRC staff also evaluated **[Include additional review items, as necessary (e.g., audits of certain information at the plant and vendor sites, and independent analyses), for areas where such analyses were deemed appropriate by the NRC staff]**.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the affects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed EPU conditions. Details of the NRC staff's review are provided in Section 2.0 of this safety evaluation.

Audits of analyses supporting the EPU were conducted in relation to the following topics:

[Provide a list of areas for which audits were performed.]

The results of the audits are discussed in section 2.0 of this safety evaluation.

Independent NRC staff calculations were performed in relation to the following topics:

[Provide a list of areas for which independent NRC staff calculations were performed.]

The results of the calculations are discussed in section 2.0 of this safety evaluation.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

SEE INSERT 1 FOR SECTION 3.2 OF RS-001

2.2 Mechanical and Civil Engineering

SEE INSERT 2 FOR SECTION 3.2 OF RS-001

2.3 Electrical Engineering

SEE INSERT 3 FOR SECTION 3.2 OF RS-001

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2.4 Instrumentation and Controls

SEE INSERT 4 FOR SECTION 3.2 OF RS-001

2.5 Plant Systems

SEE INSERT 5 FOR SECTION 3.2 OF RS-001

2.6 Containment Review Considerations

SEE INSERT 6 FOR SECTION 3.2 OF RS-001

2.7 Habitability, Filtration, and Ventilation

SEE INSERT 7 FOR SECTION 3.2 OF RS-001

2.8 Reactor Systems

SEE INSERT 8 FOR SECTION 3.2 OF RS-001

2.9 Source Terms and Radiological Consequences Analyses

SEE INSERT 9 FOR SECTION 3.2 OF RS-001

2.10 Health Physics

SEE INSERT 10 FOR SECTION 3.2 OF RS-001

2.11 Human Performance

SEE INSERT 11 FOR SECTION 3.2 OF RS-001

2.12 Power Ascension and Testing Plan

SEE INSERT 12 FOR SECTION 3.2 OF RS-001

2.13 Risk Evaluation

SEE INSERT 13 FOR SECTION 3.2 OF RS-001

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the EPU, the licensee proposed the following changes to the Facility Operating License and TSs for **[Plant Name]**.

[Provide a list of license and TSs changes (including license conditions) and an NRC staff evaluation of each.]

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4.0 REGULATORY COMMITMENTS

Insert the following sentence if the licensee has not made any regulatory commitments in support of the EPU.
The licensee has made no regulatory commitments in its application for the EPU.

Insert the following if the licensee has made regulatory commitments in support of the EPU.
The licensee has made the following regulatory commitment(s):

[Provide a summary of each regulatory commitment made by the licensee.]

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 RECOMMENDED AREAS FOR INSPECTION

As described above, the NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC staff's review has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications necessary to implement the proposed EPU, and new conditions of operation necessary for the proposed EPU. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the EPU.

[Provide list of recommended areas for inspection.]

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the **[Name of State]** State official was notified of the proposed issuance of the amendment. The State official had **[no]** comments. **[If comments were received, address them here.]**

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on **[Date] (FR)**. The draft Environmental Assessment provided a 30-day opportunity for public comment. *If no comments were received, use the following sentence:* **[No comments were received on the draft Environmental Assessment.]** *If comments were received, use the following sentence*: **[The NRC staff received comments which were addressed in the final environmental assessment.]** The final Environmental Assessment was published in the *Federal Register* on **[Date] (FR)**. Accordingly, based upon the environmental

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assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003.

2. **[Insert additional references as necessary]**

Attachment: List of Acronyms

Principal Contributors:

Date:

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LIST OF ACRONYMS

AAC	alternate ac sources
ac	alternating current
ALARA	as low as reasonably achievable
ARAVS	auxiliary and radwaste area ventilation system
ARI	alternate rod insertion
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B&PV	boiler and pressure vessel
BL	bulletin
BOP	balance-of-plant
BTP	branch technical position
BWR	boiling-water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CDF	core damage frequency
CFR	*Code of Federal Reguations*
CFS	condensate and feedwater system
CRAVS	control room area ventilation system
CRDA	control rod drop accident
CRDM	control rod drive mechanism
CRDS	control rod drive system
CUF	cumulative usage factor
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
dc	direct current
DG	draft guide
EAB	exclusion area boundary
ECCS	emergency core cooling system

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EFDS	equipment and floor drainage system
EPG	emergency procedure guideline
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESFVS	engineered safety feature ventilation system
FAC	flow-accelerated corrosion
FHA	fuel handling accident
FPP	fire protection program
GDC	general design criterion (or criteria)
GL	generic letter
I&C	instrumentation and controls
IN	information notice
IPE	individual plant examination
IPEEE	individual plant examination of external events
LERF	large early release frequency
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low population zone
MC	main condenser
MCES	main condenser evacuation system
MOV	motor-operated valve
MSIV	main steam isolation valve
MSIVLCS	main steam isolation valve leakage control system
MSLB	main steamline break

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MSSS	main steam supply system
MWt	megawatts thermal
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
O&M	operations and maintenance
P-T	pressure-temperature
PWSCC	primary water stress-corrosion cracking
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RS	review standard
RWCS	reactor water cleanup system
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	Safety Analysis Report
SBO	station blackout
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SGTS	standby gas treatment system
SLCS	standby liquid control system
SRP	Standard Review Plan
SSCs	structures, systems, and components
SSE	safe-shutdown earthquake

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SWMS	solid waste management system
SWS	service water system
TAVS	turbine area ventilation system
TBS	turbine bypass system
TCV	turbine control valve
TEDE	total effective dose equivalent
TS	technical specification
UHS	ultimate heat sink

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INSERT 1

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

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2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The NRC staff's review primarily focused on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with GDC-14 and GDC-31 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the reactor vessel material surveillance program.

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2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of P-T limits covered the P-T limits methodology and the calculations for the number of effective full power years specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the P-T limits for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the P-T limits. The NRC staff further concludes that the licensee has demonstrated the validity of the proposed P-T limits for operation under the proposed EPU conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the proposed P-T limits.

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2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS)). The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2 and Boiling Water Reactor Vessel and Internals Project (BWRVIP)-26.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a with respect to material specifications, welding controls, and inspection following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to reactor internal and core support materials.

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2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (3) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (4) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking (PWSCC) of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice (IN) 00-17, Bulletin (BL) 01-01, BL 02-01, and BL 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC-1, GDC-4, GDC-14, GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RCPB materials.

11PMA-DR1CR3-2S-000109

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff's review covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for guidance on application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coatings systems.

11PMA-DR1CR3-2S-000110

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. The NRC staff has reviewed the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusions

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. The NRC staff further concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

11PMA-DR1CR3-2S-000111

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The reactor water cleanup system (RWCS) provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCS comprise the RCPB. The NRC staff's review of the RWCS included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant's TSs in these areas under the proposed EPU conditions. The NRC's acceptance criteria for the RWCS are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the RWCS and concludes that the licensee has adequately addressed changes in impurity levels and pressure and their effects on the RWCS. The NRC staff further concludes that the licensee has demonstrated that the RWCS will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC-14, GDC-60, and GDC-61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RWCS.

11PMA-DR1CR3-2S-000112

[2.1.8 <u>Additional Review Areas</u> (Materials and Chemical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000113

INSERT 2

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000114

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducted a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The NRC staff's review covered (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented inservice inspection (ISI) programs or the use of special protective devices such as pipe-whip restraints, (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review focused on the effects that the proposed EPU may have on items (1) thru (4) above. The NRC's acceptance criteria are based on GDC-4, which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture. Specific review criteria are contained in SRP Section 3.6.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that SSCs important to safety will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

11PMA-DR1CR3-2S-000115

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code), Section III, Division 1, and GDCs 1, 2, 4, 14, and 15. The NRC staff's review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covered (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (4) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (5) GDC-15, insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

[Insert technical evaluation for nuclear steam supply system (NSSS) piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the NSSS piping, components, and supports are adequate under the proposed EPU conditions."]

Balance-of-Plant Piping, Components, and Supports

[Insert technical evaluation for balance-of-plant piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the balance-of-plant piping, components, and supports are adequate under the proposed EPU conditions."]

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the reactor vessel and supports are adequate under the proposed EPU conditions."]

11PMA-DR1CR3-2S-000116

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism. Include an intermediate conclusion in the form of "Because [summarize reasons], the control rod drive mechanism is adequate under the proposed EPU conditions."]

Recirculation Pumps and Supports

[Insert technical evaluation for reactor coolant pumps and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the recirculation pumps and supports are adequate under the proposed EPU conditions."]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, the NRC staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

11PMA-DR1CR3-2S-000117

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The NRC staff's review covered (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on the reactor internals and core supports. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, and GDC-10 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

11PMA-DR1CR3-2S-000118

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's staff's review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of Section XI of the ASME B&PV Code and the ASME Operations and Maintenance (O&M) Code, as applicable. The NRC staff's review focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covered any impacts that the proposed EPU may have on the licensee's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. The NRC staff also evaluated the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-37, GDC-40, GDC-43, and GDC-46, insofar as they require that the emergency core cooling system (ECCS), the containment heat removal system, the containment atomospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessments related to the functional performance of safety-related valves and pumps and concludes that the licensee has adequately addressed the effects of the proposed EPU on safety-related pumps and valves. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related, power-operated valves. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the requirements of GDC-1, GDC-37, GDC-40, GDC-43, GDC-46, GDC-54, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to safety-related valves and pumps.

11PMA-DR1CR3-2S-000119

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; (3) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (4) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (5) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (6) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed EPU on this equipment and (2) demonstrated that the equipment will continue to meet the requirements of GDCs 1, 2, 4, 14, and 30; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

11PMA-DR1CR3-2S-000120

[2.2.6 Additional Review Areas (Mechanical and Civil Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000121

INSERT 3

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000122

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The NRC staff's review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The NRC staff's review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EQ of electrical equipment and concludes that the licensee has adequately addressed the effects of the proposed EPU on the environmental conditions for and the qualification of electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

11PMA-DR1CR3-2S-000123

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on GDC-17. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The NRC staff further concludes that the impact of the proposed EPU on grid stability is insignificant. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

11PMA-DR1CR3-2S-000124

2.3.3 AC Onsite Power System

Regulatory Evaluation

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ac onsite power system.

11PMA-DR1CR3-2S-000125

2.3.4 DC Onsite Power System

Regulatory Evaluation

The direct current (dc) onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covered the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the dc onsite power system.

11PMA-DR1CR3-2S-000126

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the LOOP concurrent with a turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources" (AACs). The NRC staff's review focused on the impact of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed EPU on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SBO.

11PMA-DR1CR3-2S-000127

[2.3.6 Additional Review Areas (Electrical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000128

INSERT 4

FOR

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11PMA-DR1CR3-2S-000129

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducted a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

11PMA-DR1CR3-2S-000130

[2.4.2 Additional Review Areas (Instrumentation and Controls)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000131

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FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000132

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

The NRC staff conducted a review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NRC staff's review covered flooding of SSCs important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focused on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. The NRC's acceptance criteria for flood protection are based on GDC-2. Specific review criteria are contained in SRP Section 3.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the proposed changes in fluid volumes in tanks and vessels for the proposed EPU. The NRC staff concludes that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of GDC-2 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to flood protection.

11PMA-DR1CR3-2S-000133

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The EFDS is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system. The NRC staff's review of the EFDS included the collection and disposal of liquid effluents outside containment.
The NRC staff's review focused on any changes in fluid volumes or pump capacities that are necessary for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations. The NRC's acceptance criteria for the EFDS are based on GDCs 2 and 4 insofar as they require the EFDS to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures). Specific review criteria are contained in SRP Section 9.3.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EFDS and concludes that the licensee has adequately accounted for the plant changes resulting in increased water volumes and larger capacity pumps or piping systems. The NRC staff concludes that the EFDS has sufficient capacity to (1) handle the additional expected leakage resulting from the plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensure that contaminated fluids are not transferred to noncontaminated drainage systems. Based on this, the NRC staff concludes that the EFDS will continue to meet the requirements of GDCs 2 and 4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EFDS.

11PMA-DR1CR3-2S-000134

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The NRC staff's review of the CWS focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed to accommodate the proposed EPU. The NRC's acceptance criteria for the CWS are based on GDC-4 for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs. Specific review criteria are contained in SRP Section 10.4.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the modifications to the CWS and concludes that the licensee has adequately evaluated these modifications. The NRC staff concludes that, consistent with the requirements of GDC-4, the increased volumes of fluid leakage that could potentially result from these modifications would not result in the failure of safety-related SSCs following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CWS.

11PMA-DR1CR3-2S-000135

2.5.1.2 Missile Protection

2.5.1.2.1. Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NRC staff's review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The NRC staff's review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviewed the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NRC staff's review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. The NRC's acceptance criteria for the protection of SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on GDC-4. Specific review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to internally generated missiles.

11PMA-DR1CR3-2S-000136

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The NRC staff's review of the turbine generator focused on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on GDC-4, and relates to protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. Specific review criteria are contained in SRP Section 10.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the turbine generator and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The NRC staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine generator.

11PMA-DR1CR3-2S-000137

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducted a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The NRC staff's review of pipe failures included high and moderate energy fluid system piping located outside of containment. The NRC staff's review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of postaccident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on GDC-4, which requires, in part, that SSCs important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. Specific review criteria are contained in SRP Section 3.6.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the changes that are necessary for the proposed EPU and the licensee's proposed operation of the plant, and concludes that SSCs important to safety will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

11PMA-DR1CR3-2S-000138

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's fire-related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and GDCs 3 and 5 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to fire protection.

11PMA-DR1CR3-2S-000139

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

Regulatory Evaluation

The NRC staff's review for fission product control systems and structures covered the basis for developing the mathematical model for DBLOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The NRC staff's review primarily focused on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria are based on GDC-41, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on fission product control systems and structures. The NRC staff concludes that the licensee has adequately accounted for the increase in fission products and changes in expected environmental conditions that would result from the proposed EPU. The NRC staff further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the fission product control systems and structures will continue to meet the requirements of GDC-41. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fission product control systems and structures.

11PMA-DR1CR3-2S-000140

2.5.2.2 Main Condenser Evacuation System

Regulatory Evaluation

The main condenser evacuation system (MCES) generally consists of two subsystems: (1) the "hogging" or startup system which initially establishes main condenser vacuum and (2) the system which maintains condenser vacuum once it has been established. The NRC staff's review focused on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the MCES are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the MCES and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The NRC also concludes that the MCES will continue meet the requirements of GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MCES.

11PMA-DR1CR3-2S-000141

2.5.2.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment. The NRC staff reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the turbine gland sealing system and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine gland sealing system.

11PMA-DR1CR3-2S-000142

2.5.2.4 Main Steam Isolation Valve Leakage Control System

Regulatory Evaluation

Redundant quick-acting isolation valves are provided on each main steamline. The leakage control system is designed to reduce the amount of direct, untreated leakage from the main steam isolation valves (MSIVs) when isolation of the primary system and containment is required. The NRC staff's review of the MSIV leakage control system focused on the effects of the proposed EPU on the amount of leakage assumed to occur. The NRC's acceptance criteria for the MSIV leakage control system are based on GDC-54, insofar as it requires that piping systems penetrating containment be provided with leakage detection and isolation capabilities. Specific review criteria are contained in SRP Section 6.7.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the MSIV leakage control system and finds that the licensee has adequately accounted for the effects of the proposed EPU on the assumed leakage through the MSIVs. The NRC staff further concludes that the leakage control system will continue to reliably detect and isolate the leakage, as required by GDC-54. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSIV leakage control system.

11PMA-DR1CR3-2S-000143

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review for the proposed EPU focused on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions. The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and (3) GDC-61, insofar as it requires that fuel storage systems be designed with RHR capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions. Specific review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the spent fuel pool cooling and cleanup system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. Based on this review, the NRC staff concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of GDCs 5, 44, and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the spent fuel pool cooling and cleanup system.

11PMA-DR1CR3-2S-000144

2.5.3.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to non-safety-related auxiliary components that are used for normal plant operation. The NRC staff's review covered the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with the LOOP). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.1, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the station SWS and concludes that the licensee has adequately accounted for the increased heat loads on system performance that would result from the proposed EPU. The NRC staff concludes that the station SWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the station SWS will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the station SWS.

11PMA-DR1CR3-2S-000145

2.5.3.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

The NRC staff's review covered reactor auxiliary cooling water systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and mitigating the consequences of accident conditions, or (2) preventing the occurrence of an accident. These systems include closed-loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS. The NRC staff's review covered the capability of the auxiliary cooling water systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. Emphasis was placed on the cooling water systems for safety-related components (e.g., ECCS equipment, ventilation equipment, and reactor shutdown equipment). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.2, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the reactor auxiliary cooling water systems and concludes that the licensee has adequately accounted for the increased heat loads from the proposed EPU on system performance. The NRC staff concludes that the reactor auxiliary cooling water systems will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the reactor auxiliary cooling water systems will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the reactor auxiliary cooling water systems.

11PMA-DR1CR3-2S-000146

2.5.3.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The NRC staff's review focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review included evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The NRC's acceptance criteria for the UHS are based on (1) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety; and (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the information that was provided by the licensee for addressing the effects that the proposed EPU would have on the UHS safety function, including the licensee's validation of the design-basis UHS temperature limit based on post-licensing data. Based on the information that was provided, the NRC staff concludes that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of GDCs 5 and 44 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the UHS.

11PMA-DR1CR3-2S-000147

2.5.4 Balance-of-Plant Systems

2.5.4.1. Main Steam

Regulatory Evaluation

The main steam supply system (MSSS) transports steam from the NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. The NRC staff's review focused on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC's acceptance criteria for the MSSS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including the effects missiles, pipe whip, and jet impingement forces associated with pipe breaks; and (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MSSS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MSSS. The NRC staff concludes that the MSSS will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. The NRC staff further concludes that the MSSS will continue to meet the requirements of GDCs 4 and 5. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSSS.

11PMA-DR1CR3-2S-000148

2.5.4.2 Main Condenser

Regulatory Evaluation

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system (TBS). For BWRs without an MSIV leakage control system, the MC system may also serve an accident mitigation function to act as a holdup volume for the plateout of fission products leaking through the MSIVs following core damage. The NRC staff's review focused on the effects of the proposed EPU on the steam bypass capability with respect to load rejection assumptions, and on the ability of the MC system to withstand the blowdown effects of steam from the TBS. The NRC's acceptance criteria for the MC system are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MC system and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MC system. The NRC staff concludes that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the TBS and thereby continue to meet GDC-60 with respect to controlling releases of radioactive effluents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MC system.

11PMA-DR1CR3-2S-000149

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The TBS is designed to discharge a stated percentage of rated main steam flow directly to the MC system, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure. For a BWR without an MSIV leakage control system, the TBS could also provide an accident mitigation function. A TBS, along with the MSSS and MC system, may be credited for mitigating the effects of MSIV leakage during a LOCA by the holdup and plateout of fission products. The NRC staff's review for the TBS focused on the effects that the proposed EPU have on load rejection capability, analysis of postulated system piping failures, and the consequences of inadvertent TBS operation. The NRC's acceptance criteria for the TBS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including pipe breaks or malfunctions of the TBS), and (2) GDC-34, insofar as it requires that a RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the TBS. The NRC staff concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the TBS. The NRC staff concludes that the TBS will continue to mitigate the effects of MSIV leakage during a LOCA and provide a means for shutting down the plant during normal operations. The NRC staff further concludes that TBS failures will not adversely affect essential SSCs. Based on this, the NRC staff concludes that the TBS will continue to meet GDCs 4 and 34. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the TBS.

11PMA-DR1CR3-2S-000150

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at a particular temperature, pressure, and flow rate to the reactor. The only part of the CFS classified as safety-related is the feedwater piping from the NSSS up to and including the outermost containment isolation valve. The NRC staff's review focused on how the proposed EPU affects previous analyses and considerations with respect to the capability of the CFS to supply adequate feedwater during plant operation and shutdown, and isolate components, subsystems, and piping in order to preserve the system's safety function. The NRC's acceptance criteria for the CFS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including possible fluid flow instabilities (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that the system be provided with suitable isolation capabilities to assure the safety function can be accomplished with electric power available from only the onsite system or only the offsite system, assuming a single failure. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the CFS. The NRC staff concludes that the CFS will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. The NRC staff further concludes that the CFS will continue to meet the requirements of GDCs 4, 5, and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CFS.

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2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

Regulatory Evaluation

The gaseous waste management systems involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of the condenser air removal system; the gland seal exhaust and the mechanical vacuum pump operation exhaust; and the building ventilation system exhausts. The NRC staff's review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exists. The NRC's acceptance criteria for gaseous waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion. Specific review criteria are contained in SRP Section 11.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the gaseous waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC staff finds that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the gaseous waste management systems will continue to meet the requirements of 10 CFR 20.1302; GDCs 3, 60, and 61; and 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the gaseous waste management systems.

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2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for liquid waste management systems focused on the effects that the proposed EPU may have on previous analyses and considerations related to the liquid waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion. Specific review criteria are contained in SRP Section 11.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the liquid waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management systems to control releases of radioactive materials. The NRC staff finds that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the liquid waste management systems will continue to meet the requirements of 10 CFR 20.1302; GDCs 60 and 61; and 10 CFR Part 50, Appendix I, Sections II.A and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the liquid waste management systems.

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2.5.5.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for the solid waste management systems (SWMS) focused on the effects that the proposed EPU may have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the SWMS. The NRC's acceptance criteria for the SWMS are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels, (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SWMS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the SWMS to process the waste. The NRC staff finds that the SWMS will continue to meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the SWMS will continue to meet the requirements of 10 CFR 20.1302, GDCs 60, 63, and 64, and 10 CFR Part 71. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SWMS.

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2.5.6 Additional Considerations

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure. The NRC staff's review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the amount of required fuel oil for the emergency diesel generators and concludes that the licensee has adequately accounted for the effects of the increased electrical demand on fuel oil consumption. The NRC staff concludes that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of GDCs 4, 5, and 17. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel oil storage and transfer system.

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2.5.6.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks. The NRC staff's review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review focused on the effects of the new fuel on system performance and related analyses. The NRC's acceptance criteria for the LLHS are based on (1) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC-62, insofar as it requires that criticality be prevented. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that the licensee has adequately incorporated the effects of the new fuel in the analyses. Based on this review, the NRC staff further concludes that the LLHS will continue to meet the requirements of GDCs 61 and 62 for radioactivity releases and prevention of criticality accidents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LLHS.

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[2.5.7 Additional Review Areas (Plant Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

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2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The NRC staff's review for the primary containment functional design covered (1) the temperature and pressure conditions in the drywell and wetwell due to a spectrum of postulated LOCAs, (2) the differential pressure across the operating deck for a spectrum of LOCAs (Mark II containments only), (3) suppression pool dynamic effects during a LOCA or following the actuation of one or more RCS safety/relief valves, (4) the consequences of a LOCA occurring within the containment (wetwell), (5) the capability of the containment to withstand the effects of steam bypassing the suppression pool, (6) the suppression pool temperature limit during RCS safety/relief valve operation, and (7) the analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; (2) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (3) GDC-50, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety; and (5) GDC-64, insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.C.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of GDCs 4, 13,

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16, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to primary containment functional design.

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2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects, and (2) GDC-50, insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization resulting from the increased mass and energy release. The NRC staff concludes that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. Based on this, the NRC staff concludes that the plant will continue to meet GDCs 4 and 50 for the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to subcompartment analyses.

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2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss of Coolant

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) GDC-50, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the NRC staff finds that the mass and energy release analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated LOCA.

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2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covered (1) the production and accumulation of combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NRC staff's review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-41, insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; (4) GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection; and (5) GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing. **[*Include the following sentence for BWRs with Mark III containments:* Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with a Mark III type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.]** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities consistent with the requirements in 10 CFR 50.44 and GDCs 5, 41, 42, and 43 as discussed above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

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2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NRC staff's review in this area focused on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers. The NRC's acceptance criteria for containment heat removal are based on GDC-38, insofar as it requires that a containment heat removal system be provided, and that its function shall be to rapidly reduce the containment pressure and temperature following a LOCA and maintain them at acceptably low levels. Specific review criteria are contained in SRP Section 6.2.2, as supplemented by Draft Guide (DG) 1107.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed EPU. The NRC staff finds that the systems will continue to meet GDC-38 with respect to rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment heat removal systems.

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2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems of dual containment plants are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NRC staff's review covered (1) analyses of the pressure and temperature response of the secondary containment following accidents within the primary and secondary containments; (2) analyses of the effects of openings in the secondary containment on the capability of the depressurization and filtration system to establish a negative pressure in a prescribed time; (3) analyses of any primary containment leakage paths that bypass the secondary containment; (4) analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment; and (5) the acceptability of the mass and energy release data used in the analysis. The NRC staff's review primarily focused on the effects that the proposed EPU may have on the pressure and temperature response and drawdown time of the secondary containment, and the impact this may have on offsite dose. The NRC's acceptance criteria for secondary containment functional design are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and be protected from dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures; and (2) GDC-16, insofar as it requires that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. Specific review criteria are contained in SRP Section 6.2.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The NRC staff concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU and further concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the secondary containment and associated systems will continue to meet the requirements of GDCs 4 and 16. Therefore, the NRC staff finds the proposed EPU acceptable with respect to secondary containment functional design.

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[2.6.7 Additional Review Areas (Containment Review Considerations)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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INSERT 7

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

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2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staff's review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU. The NRC staff further concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of GDCs 4 and 19. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

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2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in postaccident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., standby gas treatment systems and emergency or postaccident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; (3) GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of GDCs 19, 41, 61, and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

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2.7.3 Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, the NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of GDCs 4, 19, and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

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2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel handling accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC's acceptance criteria for the SFPAVS are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents, and (2) GDC-61, insofar as it requires that systems which contain radioactivity be designed with appropriate confinement and containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the SFPAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's capability to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the NRC staff concludes that the SFPAVS will continue to meet the requirements of GDCs 60 and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SFPAVS.

11PMA-DR1CR3-2S-000171

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system (ARAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the ARAVS and TAVS are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ARAVS and TAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the auxiliary and radwaste equipment areas and in the turbine area, permit personnel access, control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the ARAVS and TAVS will continue to meet the requirements of GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ARAVS and the TAVS.

11PMA-DR1CR3-2S-000172

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review for the ESFVS focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the ESFVS will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESFVS will continue to meet the requirements of GDCs 4, 17 and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

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[2.7.7 Additional Review Areas (Habitability, Filtration, and Ventilation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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INSERT 8

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000175

2.8 Reactor Systems

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, GDC-27, and GDC-35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel system design.

11PMA-DR1CR3-2S-000176

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

11PMA-DR1CR3-2S-000177

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the nuclear design.

11PMA-DR1CR3-2S-000178

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is **[equivalent to or a justified extrapolation from]** proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed EPU on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs 10 and 12 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to thermal and hydraulic design.

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2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; and (8) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient cooling exists to ensure the system's design bases will continue to

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be followed upon implementation of the proposed EPU. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

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2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steamlines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee has (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet GDCs 15 and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

11PMA-DR1CR3-2S-000182

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function; (3) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; (4) GDC-33, insofar as it requires that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided so the fuel design limits are not exceeded; (5) GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded; (6) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (7) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and a station blackout event and the ability of the system to provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of GDCs 4, 5, 29, 33, 34 and 54, and 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RCIC system.

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2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-34, which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of GDCs 4, 5, and 34 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

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2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) GDC-26, insofar as it requires that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor subcritical in the cold condition; (2) GDC-27, insofar as it requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions; and (3) 10 CFR 50.62(c)(4), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control, and **[DEPENDING ON CONSTRUCTION PERMIT DATE OR ORIGINAL DESIGN]** that the system initiate automatically. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the SLCS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU. Based on this, the NRC staff concludes that the SLCS will continue to meet the requirements of GDCs 26 and 27, and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SLCS.

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2.8.5 Accident and Transient Analyses

2.8.5.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Main Steam Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; (3) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and (4) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

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2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

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2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

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2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

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2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

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2.8.5.3.2 Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor recirculation pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor system components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (3) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, and 31 following implementation of the

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proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

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2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design necessary for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

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2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal at power event and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power.

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2.8.5.4.3 Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences; (3) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during AOOs; (4) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (5) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the increase in core flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

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2.8.5.4.4 Spectrum of Rod Drop Accidents

Regulatory Evaluation

The NRC staff evaluated the consequences of a control rod drop accident in the area of reactor physics. The NRC staff's review covered the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the rod drop accident and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the rod drop accident.

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2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory.

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2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.

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2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; (4) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 4, 27, 35, and 10 CFR 50.46 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LOCA.

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2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- each BWR have an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- each BWR have a standby liquid control system (SLCS) with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel. The system initiation must be automatic.

- each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed EPU, and (3) operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), insofar as they apply to the plant design. In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The NRC staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses. *Insert the following sentence if the licensee relied upon generic vendor analyses* **[The NRC staff reviewed the licensee's justification of the applicability of generic vendor analyses to its plant and the operating conditions for the proposed EPU.]** Review guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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Conclusion

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on ATWS. The NRC staff concludes that the licensee has demonstrated that ARI, SLCS, and recirculation pump trip systems have been installed and that they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to ATWS.

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2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on GDC-62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of GDC-62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the new fuel storage.

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2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covered the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (2) GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. The NRC staff also concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, the NRC staff concludes that the spent fuel storage facilities will continue to meet the requirements of GDCs 4 and 62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to spent fuel storage.

11PMA-DR1CR3-2S-000203

[2.8.7 Additional Review Areas (Reactor Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000204

INSERT 9

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000205

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's **[Updated Safety Analysis Report *or* Updated Final Safety Analysis Report]** related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

11PMA-DR1CR3-2S-000206

NOTE: Use Sections 2.9.2 and 2.9.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

NOTE: There are two cases that may be encountered here: (1) a licensee may be implementing an alternative source term for the first time, or (2) a licensee may have already fully implemented an alternative source term and is revising the previously approved dose analyses that use alternative source term methodologies. The second paragraph for each heading is only needed for a first-time implementation of an alternative source term (either partial or full implementations). Several accidents may have been analyzed - see corresponding SRP sections for further regulatory evaluation text (to be modified), as needed.

Regulatory Evaluation

The NRC staff reviewed the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod drop accident (CRDA), and main steamline break (MSLB). The NRC staff's review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident. Specific review criteria are contained in SRP Section 15.0.1.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff reviewed the implementation of alternative source terms. The NRC's acceptance criteria for implementation of alternative source terms are based on (1) 10 CFR 50.67, insofar as it sets standards for the implementation of an alternative source term in current operating nuclear power plants; (2) 10 CFR 50.49, insofar as it requires qualification of safety-related equipment, as defined in that section, including and based on integrated radiation dose during normal and accident conditions; (3) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident; (4) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, insofar as it requires a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency; and (5) plant-specific licensing commitments made in response to NUREG-0737 (Items II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2, and III.D.3.4). Specific review criteria are contained in SRP Sections 15.0.1.

11PMA-DR1CR3-2S-000207

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has adequately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC-19, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff has reviewed the alternative source term methodology used by the licensee in evaluating the effects of the proposed EPU and concludes that changes continue to provide a sufficient margin of safety with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and parameter inputs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the implementation of an alternative source term.

11PMA-DR1CR3-2S-000208

[2.9.3 Additional Review Areas (Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as *necessary*]

11PMA-DR1CR3-2S-000209

NOTE: Use Sections 2.9.2 - 2.9.8 below if the licensee's radiological consequences analyses are not based on an alternative source term (i.e., if the analyses are based on a traditional source term (i.e., TID-14844)

2.9.2 Radiological Consequences of Control Rod Drop Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a control rod drop accident (CRDA). The NRC staff's review included an examination of (1) the plant's response to the accident, (2) the release of fission products from the core to the environment via the turbine and condensers as a result of the accident, (3) and the calculation of radiological doses at the exclusion area boundary (EAB) and low population zone (LPZ) outer boundary, and in the control room due to the releases from the accident. The NRC's acceptance criteria for the radiological consequences of a control rod drop accident are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.4.9.A, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a control rod drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated control rod drop accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a control rod drop accident.

11PMA-DR1CR3-2S-000210

2.9.3 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analysis of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review included (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident, and (2) GDC-55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs will remain acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are substantially below the exposure guideline values of 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary.

11PMA-DR1CR3-2S-000211

2.9.4 Radiological Consequences of Main Steamline Failure Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an MSLB accident outside the containment to ensure that radioactive releases due to such an event are adequately limited by the TS limit on primary coolant activity. The NRC staff's review included two cases for the reactor coolant iodine concentration: (1) an MSLB with a preaccident iodine spike and (2) an MSLB with the maximum equilibrium concentration for continued full-power operation. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.6.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for an MSLB with the primary coolant at the maximum equilibrium concentration for continued full-power operation. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to a postulated failure of an MSLB outside containment.

11PMA-DR1CR3-2S-000212

2.9.5 Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a design-basis LOCA. This review included a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also included (1) the contribution to the dose due to leakage from the main steam isolation valves (MSIVs); (2) the methodology and results of calculations of the radiological consequences resulting from containment and ESF components and MSIV leakage following a hypothetical LOCA; and (3) an assessment of the containment with respect to the assumptions and the input parameters for the dose calculations. The NRC's calculations were based on pertinent information in the **[Updated Safety Analysis Report *or* Updated Final Safety Analysis Report]** and considers the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 6.4 and Appendices A, B, and D of SRP Section 15.6.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis LOCA and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis LOCA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 and the calculated doses in the control room meet the requirements of GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a design-basis LOCA.

11PMA-DR1CR3-2S-000213

2.9.6 Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a postulated FHA. The purpose of this review was to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review included (1) the sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of FHAs and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of FHAs.

11PMA-DR1CR3-2S-000214

2.9.7 Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review was conducted to verify various design and operational aspects of the system. The NRC staff's review included (1) determining a need for a design-basis radiological analysis sequence of events; (2) models and assumptions used by the licensee for the calculation of the radiological doses; (3) comparing calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitability; and (4) examining the relationship of the operational modes of the standby gas treatment system (SGTS) to the time sequence of the accident in order to give proper credit, in a dual containment design where the fuel building atmosphere may be exhausted through the SGTS. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a spent fuel cask drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to spent fuel cask drop accidents.

11PMA-DR1CR3-2S-000215

[2.9.8 Additional Review Areas (Source Terms and Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000216

INSERT 10

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11PMA-DR1CR3-2S-000217

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered the effects of the proposed EPU on nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC staff also considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 and GDC-19. Specific review criteria are contained in SRP Sections 12.2, 12.3,12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. The NRC staff further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

11PMA-DR1CR3-2S-000218

[2.10.2 Additional Review Areas (Health Physics)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000219

INSERT 11

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SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000220

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implemented the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area. The licensee has addressed these questions in its application. Following are the NRC staff's questions, the licensee's responses, and the NRC staff's evaluation of the responses.

1. Changes in Emergency and Abnormal Operating Procedures

   Describe how the proposed EPU will change the plant emergency and abnormal operating procedures. (SRP Section 13.5.2.1)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

2. Changes to Operator Actions Sensitive to Power Uprate

   Describe any new operator actions needed as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU. (SRP Section 18.0)

   (i.e., Identify and describe operator actions that will involve additional response time or will have reduced time available. Your response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the acceptability of these changes).

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

11PMA-DR1CR3-2S-000221

3. Changes to Control Room Controls, Displays and Alarms

   Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators will be tested to determine they could use the instruments reliably. (SRP Section 18.0)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

4. Changes on the Safety Parameter Display System

   Describe any changes to the safety parameter display system resulting from the proposed EPU. How will the operators know of the changes? (SRP Section 18.0)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

5. Changes to the Operator Training Program and the Control Room Simulator

   Describe any changes to the operator training program and the plant referenced control room simulator resulting from the proposed EPU, and provide the implementation schedule for making the changes. (SRP Sections 13.2.1 and 13.2.2)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that the licensee has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of GDC-19, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the human factors aspects of the required system changes.

11PMA-DR1CR3-2S-000222

[2.11.2 Additional Review Areas (Human Performance)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000223

INSERT 12

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11PMA-DR1CR3-2S-000224

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The NRC staff's review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The staff has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. The staff concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Further, the staff finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test program acceptable.

11PMA-DR1CR3-2S-000225

[2.12.2 Additional Review Areas (Power Ascension and Testing Plan)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000226

INSERT 13

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000227

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

Regulatory Evaluation

The licensee conducted a risk evaluation to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are present if any issue would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff's review covered the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff's review covered the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This included a review of the licensee's actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review. The NRC's risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 13 of RS-001 and its attachments.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the risk implications associated with the implementation of the proposed EPU and concludes that the licensee has adequately modeled and/or addressed the potential impacts associated with the implementation of the proposed EPU. The NRC staff further concludes that the results of the licensee's risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the NRC staff finds the risk implications of the proposed EPU acceptable.

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[2.13.2 Additional Review Areas (Risk Evaluation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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SECTION 3.3 of RS-001

TEMPLATE SAFETY EVALUATION

for

PRESSURIZED-WATER REACTOR EXTENDED POWER UPRATE

11PMA-DR1CR3-2S-000230

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Attachment: List of Acronyms

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. ____ TO FACILITY OPERATING LICENSE NO. **[XXX-XX]**

[NAME OF LICENSEE]

[NAME OF FACILITY]

DOCKET NO. 50-**[XXX]**

1.0 INTRODUCTION

1.1 Application

By application dated **[ ]**, as supplemented by letter**[s]** dated **[ ]**, the **[Name of Licensee]** (the licensee) requested changes to the Facility Operating License and Technical Specifications (TSs) for the **[Plant Name]**. The supplemental letter[s] dated **[ ]**, provided additional clarifying information that did not expand the scope of the initial application and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on **[date] (XX FR XXXX)**.

The proposed changes would increase the maximum steady-state reactor core power level from **[current licensed power level]** megawatts thermal (MWt) to **[power level proposed by the licensee]** MWt, which is an increase of approximately **[##]** percent. The proposed increase in power level is considered an extended power uprate (EPU).

1.2 Background

[Plant Name] is a pressurized-water reactor (PWR) plant of the **[Babcock & Wilcox (B&W), Combustion Engineering (CE), or Westinghouse 2-Loop, 3-Loop, or 4-Loop]** design with a **[######]** containment. **[Plant Name]** has the following special features/unique designs:

[Insert any special features/unique designs]

The NRC originally licensed **[Plant Name]** on **[date]** for operation at **[original licensed power level]** MWt. **[By Amendment No. [###] dated [ ], the NRC granted a power uprate to [Plant Name] of [##] percent, allowing the plant to be operated at [current licensed power level] MWt.] Therefore, the proposed EPU would result in an increase of approximately [##] percent over the original licensed power level [and [##] percent over the current licensed power level] for [Plant Name].]**

1.3 Licensee's Approach

The licensee's application for the proposed EPU follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, "Review Standard for Extended

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Power Uprates," to the extent that the review standard is consistent with the design basis of the plant. Where differences exist between the plant-specific design basis and RS-001, the licensee described the differences and provided evaluations consistent with the design basis of the plant. The licensee also used **[Identify topical reports or other documents used by the licensee for guidance related to the scope of the proposed EPU; NRC staff approvals, ranges of applicability, any limitations/restrictions associated with the documents; and consistency of the licensee's application with the ranges of applicability and limitations/restrictions. The discussion in this section is to cover topical reports and other documents referenced for the overall power uprate process. It is not intended to cover topical reports and other documents for specific methods of analyses. Topical reports and other documents referenced for specific methods of analyses are to be covered in the applicable technical evaluation section of this safety evaluation]**.

Insert this sentence if the licensee is planning to implement the EPU in one stage.
[The licensee plans to implement the EPU in one step. The licensee plans to make the modifications necessary to implement the EPU during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt starting in Cycle [##].]

Insert this paragraph if the licensee is planning to implement the EPU in stages:
[The licensee plans to implement the EPU in [#] steps of [## and ##] percent. The licensee plans to make modifications necessary to implement the first step during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt during Cycle [##]. The remainder of the modifications will be completed during the refueling outage in [season year (e.g., fall 2003)], with subsequent operation at [##] MWt starting in Cycle [##].]

1.4 Plant Modifications

The licensee has determined that several plant modifications are necessary to implement the proposed EPU. The following is a list of these modifications and the licensee's proposed schedule for completing them.

[Provide a list of plant modifications.]

The NRC staff's evaluation of the licensee's proposed plant modifications is provided in Section 2.0 of this safety evaluation.

1.5 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed EPU on design-basis analyses. The NRC staff evaluated the licensee's application and

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supplements. The NRC staff also evaluated **[Include additional review items, as necessary (e.g., audits of certain information at the plant and vendor sites, and independent analyses), for areas where such analyses were deemed appropriate by the NRC staff]**.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the affects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed EPU conditions. Details of the NRC staff's review are provided in Section 2.0 of this safety evaluation.

Audits of analyses supporting the EPU were conducted in relation to the following topics:

[Provide a list of areas for which audits were performed.]

The results of the audits are discussed in section 2.0 of this safety evaluation.

Independent NRC staff calculations were performed in relation to the following topics:

[Provide a list of areas for which independent NRC staff calculations were performed.]

The results of the calculations are discussed in section 2.0 of this safety evaluation.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

INSERT 1 FOR SECTION 3.3 OF RS-001

2.2 Mechanical and Civil Engineering

INSERT 2 FOR SECTION 3.3 OF RS-001

2.3 Electrical Engineering

INSERT 3 FOR SECTION 3.3 OF RS-001

2.4 Instrumentation and Controls

INSERT 4 FOR SECTION 3.3 OF RS-001

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2.5 Plant Systems

SEE INSERT 5 FOR SECTION 3.3 OF RS-001

2.6 Containment Review Considerations

SEE INSERT 6 FOR SECTION 3.3 OF RS-001

2.7 Habitability, Filtration, and Ventilation

SEE INSERT 7 FOR SECTION 3.3 OF RS-001

2.8 Reactor Systems

SEE INSERT 8 FOR SECTION 3.3 OF RS-001

2.9 Source Terms and Radiological Consequences Analyses

SEE INSERT 9 FOR SECTION 3.3 OF RS-001

2.10 Health Physics

SEE INSERT 10 FOR SECTION 3.3 OF RS-001

2.11 Human Performance

SEE INSERT 11 FOR SECTION 3.3 OF RS-001

2.12 Power Ascension and Testing Plan

SEE INSERT 12 FOR SECTION 3.3 OF RS-001

2.13 Risk Evaluation

SEE INSERT 13 FOR SECTION 3.3 OF RS-001

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the EPU, the licensee proposed the following changes to the Facility Operating License and TSs for **[Plant Name]**.

[Provide a list of license and TSs changes (including license conditions) and an NRC staff evaluation of each.]

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4.0 REGULATORY COMMITMENTS

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Insert the following sentence if the licensee has not made any regulatory commitments in support of the EPU.
The licensee has made no regulatory commitments in its application for the EPU.

Insert the following if the licensee has made regulatory commitments in support of the EPU.
The licensee has made the following regulatory commitment(s):

[Provide a summary of each regulatory commitment made by the licensee.]

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 RECOMMENDED AREAS FOR INSPECTION

As described above, the NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC staff's review has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications necessary to implement the proposed EPU, and new conditions of operation necessary for the proposed EPU. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the EPU.

[Provide list of recommended areas for inspection.]

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the **[Name of State]** State official was notified of the proposed issuance of the amendment. The State official had **[no]** comments. **[If comments were received, address them here.]**

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on **[Date] (FR   )**. The draft Environmental Assessment provided a 30-day opportunity for public comment. *If no comments were received, use the following sentence:* **[No comments were received on the draft Environmental Assessment.]** *If comments were received, use the following sentence*: **[The NRC staff received comments which were addressed in the final environmental assessment.]** The final Environmental Assessment was published in the *Federal Register* on **[Date] (FR   )**. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

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8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003.

2. **[Insert additional references as necessary]**

Attachment: List of Acronyms

Principal Contributors:

Date:

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LIST OF ACRONYMS

AAC	alternate ac sources
ac	alternating current
ALARA	as low as reasonably achievable
ARAVS	auxiliary and radwaste area ventilation system
ARI	alternate rod insertion
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B&PV	boiler and pressure vessel
B&W	Babcock and Wilcox
BL	bulletin
BOP	balance-of-plant
BRS	boron recovery system
BTP	branch technical position
CDF	core damage frequency
CE	Combustion Engineering
CFR	*Code of Federal Reguations*
CFS	condensate and feedwater system
CRAVS	control room area ventilation system
CRDM	control rod drive mechanism
CRDS	control rod drive system
CUF	cumulative usage factor
CVCS	chemical and volume control system
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
dc	direct current
DG	draft guide
DSS	diverse scram system

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EAB	exclusion area boundary
ECCS	emergency core cooling system
EFDS	equipment and floor drainage system
EPG	emergency procedure guideline
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESFVS	engineered safety feature ventilation system
FAC	flow-accelerated corrosion
FHA	fuel handling accident
FPP	fire protection program
GDC	general design criterion (or criteria)
GL	generic letter
I&C	instrumentation and controls
IN	information notice
IPE	individual plant examination
IPEEE	individual plant examination of external events
LERF	large early release frequency
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low population zone
MC	main condenser
MCES	main condenser evacuation system
MOV	motor-operated valve
MSLB	main steamline break

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MSSS	main steam supply system
MTC	moderator temperature coefficient
MWt	megawatts thermal
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
O&M	operations and maintenance
P-T	pressure-temperature
PRT	pressurizer relief tank
PWR	pressurized-water reactor
PWSCC	primary water stress-corrosion cracking
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
REA	rod ejection accident
RG	regulatory guide
RHR	residual heat removal
RS	review standard
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	Safety Analysis Report
SBO	station blackout
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SG	steam generator
SGBS	steam generator blowdown system
SGTR	steam generator tube rupture

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SRP	Standard Review Plan
SSCs	structures, systems, and components
SSE	safe-shutdown earthquake
SWMS	solid waste management system
SWS	service water system
TAVS	turbine area ventilation system
TBS	turbine bypass system
TCV	turbine control valve
TEDE	total effective dose equivalent
TS	technical specification
UHS	ultimate heat sink

11PMA-DR1CR3-2S-000242

INSERT 1

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000243

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The NRC staff's review primarily focused on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) General Design Criterion (GDC)-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with GDC-14 and GDC-31 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the reactor vessel material surveillance program.

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2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of P-T limits covered the P-T limits methodology and the calculations for the number of effective full power years specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the P-T limits for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the P-T limits. The NRC staff further concludes that the licensee has demonstrated the validity of the proposed P-T limits for operation under the proposed EPU conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the proposed P-T limits.

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2.1.3 Pressurized Thermal Shock

Regulatory Evaluation

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of the reactor vessel beltline materials to PTS events to assure that adequate fracture toughness is provided for supporting reactor operation. The NRC staff's review covered the PTS methodology and the calculations for the reference temperature, RT_{PTS}, at the expiration of the license, considering neutron embrittlement effects. The NRC's acceptance criteria for PTS are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (3) 10 CFR 50.61, insofar as it sets fracture toughness criteria for protection against PTS events. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the PTS for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on PTS. The NRC staff further concludes that the licensee has demonstrated that the plant will continue to meet the requirements of GDC-14, GDC-31, and 10 CFR 50.61 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to PTS.

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2.1.4 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS)). The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2, WCAP-14277, and BAW-2248.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to reactor internal and core support materials.

11PMA-DR1CR3-2S-000247

2.1.5 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (3) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (4) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking (PWSCC) of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice (IN) 00-17, Bulletin (BL) 01-01, BL 02-01, and BL 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC-1, GDC-4, GDC-14, GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RCPB materials.

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2.1.6 Leak-Before-Break

Regulatory Evaluation

Leak-before-break (LBB) analyses provide a means for eliminating from the design basis the dynamic effects of postulated pipe ruptures for a piping system. NRC approval of LBB for a plant permits the licensee to (1) remove protective hardware along the piping system (e.g., pipe whip restraints and jet impingement barriers) and (2) redesign pipe-connected components, their supports, and their internals. The NRC staff's review for LBB covered (a) direct pipe failure mechanisms (e.g., water hammer, creep damage, erosion, corrosion, fatigue, and environmental conditions); (b) indirect pipe failure mechanisms (e.g., seismic events, system overpressurizations, fires, flooding, missiles, and failures of SSCs in close proximity to the piping); and (c) deterministic fracture mechanics and leak detection methods. The NRC's acceptance criteria for LBB are based on GDC-4, insofar as it allows for exclusion of dynamic effects of postulated pipe ruptures from the design basis. Specific review criteria are contained in draft SRP Section 3.6.3 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the LBB analysis for the plant and concludes that the licensee has adequately addressed changes in primary system pressure and temperature and their effects on the LBB analyses. The NRC staff further concludes that the licensee has demonstrated that the LBB analyses will continue to be valid following implementation of the proposed EPU and that lines for which the licensee credits LBB will continue to meet the requirements of GDC-4. Therefore, the NRC staff finds the proposed EPU acceptable with respect to LBB.

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2.1.7 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff's review covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for guidance on application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coatings systems.

11PMA-DR1CR3-2S-000250

2.1.8 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. The NRC staff has reviewed the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusions

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. The NRC staff further concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

11PMA-DR1CR3-2S-000251

2.1.9 Steam Generator Tube Inservice Inspection

Regulatory Evaluation

Steam generator (SG) tubes constitute a large part of the RCPB. SG tube inservice inspection (ISI) provides a means for assessing the structural and leaktight integrity of the SG tubes through periodic inspection and testing of critical areas and features of the tubes. The NRC staff's review in this area covered the effects of changes in differential pressure, temperature, and flow rates resulting from the proposed EPU on plugging limits, potential degradation mechanisms (e.g., flow-induced vibration), plant-specific alternate repair criteria, and redefined inspection boundaries. The NRC's acceptance criteria for SG tube ISI are based on 10 CFR 50.55a requirements for periodic inspection and testing of the RCPB. Specific review criteria are contained in SRP Section 5.4.2.2 and other guidance provided in Matrix 1 of RS-001. Additional review guidance is contained in **[provide specific plant technical specification]** for SG surveillance requirements, Regulatory Guide 1.121 for SG tube plugging limits, GL 95-03 and Bulletin 88-02 for degradation mechanisms, NEI 97-06 for structural and leakage performance criteria, and **[provide topical reports approved for the plant],** all of which form the basis for alternate repair criteria or redefined inspection boundaries.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on SG tube integrity and concludes that the licensee has adequately assessed the continued acceptability of the plant's TSs under the proposed EPU conditions and has identified appropriate degradation management inspections to address the effects of changes in temperature, differential pressure, and flow rates on SG tube integrity. The NRC staff further concludes that the licensee has demonstrated that SG tube integrity will continue to be maintained and will continue to meet the performance criteria in NEI 97-06 and the requirements of 10 CFR 50.55a following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SG tube ISI.

11PMA-DR1CR3-2S-000252

2.1.10 Steam Generator Blowdown System

Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SG blowdown system (SGBS) provides a means for removing SG secondary-side impurities and thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected and design flows for all modes of operation. The NRC staff's review covered the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary side during normal operation, including anticipated operational occurrences (main condenser inleakage and primary-to-secondary leakage). The NRC's acceptance criteria for the SGBS are based on GDC-14, insofar as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the SGBS and concludes that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The NRC staff further concludes that the licensee has demonstrated that the SGBS will continue to be acceptable and will continue to meet the requirements of GDC-14 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SGBS.

11PMA-DR1CR3-2S-000253

2.1.11 Chemical and Volume Control System

Regulatory Evaluation

The chemical and volume control system (CVCS) and boron recovery system (BRS) provide means for (a) maintaining water inventory and quality in the RCS, (b) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (c) controlling the boron neutron absorber concentration in the reactor coolant, (d) controlling the primary water chemistry and reducing coolant radioactivity level, and (e) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the emergency core cooling system (ECCS) in the event of postulated accidents. The NRC staff reviewed the safety-related functional performance characteristics of CVCS components. The NRC's acceptance criteria are based on (1) GDC-14, insofar as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture, and (2) GDC-29, insofar as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of anticipate operational occurrences. Specific review criteria are contained in SRP Section 9.3.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the CVCS and BRS and concludes that the licensee has adequately addressed changes in the temperature of the reactor coolant and their effects on the CVCS and BRS. The NRC staff further concludes that the licensee has demonstrated that the CVCS and BRS will continue to be acceptable and will continue to meet the requirements of GDC-14 and GDC-29 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CVCS.

11PMA-DR1CR3-2S-000254

[2.1.12 Additional Review Areas (Materials and Chemical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000255

INSERT 2

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000256

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducted a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The NRC staff's review covered (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented ISI programs or the use of special protective devices such as pipe-whip restraints, (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review focused on the effects that the proposed EPU may have on items (1) thru (4) above. The NRC's acceptance criteria are based on GDC-4, which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture. Specific review criteria are contained in SRP Section 3.6.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that SSCs important to safety will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

11PMA-DR1CR3-2S-000257

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code), Section III, Division 1, and GDCs 1, 2, 4, 14, and 15. The NRC staff's review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covered (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC 1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (4) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (5) GDC 15, insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

[Insert technical evaluation for nuclear steam supply system (NSSS) piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the NSSS piping, components, and supports are adequate under the proposed EPU conditions."]

Balance-of-Plant Piping, Components, and Supports

[Insert technical evaluation for balance-of-plant piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the balance-of-plant piping, components, and supports are adequate under the proposed EPU conditions."]

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the reactor vessel and supports are adequate under the proposed EPU conditions."]

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Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism. Include an intermediate conclusion in the form of "Because [summarize reasons], the control rod drive mechanism is adequate under the proposed EPU conditions."]

Steam Generators and Supports

[Insert technical evaluation for SGs and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the SG and supports are adequate under the proposed EPU conditions."]

Reactor Coolant Pumps and Supports

[Insert technical evaluation for reactor coolant pumps and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the reactor coolant pumps and supports are adequate under the proposed EPU conditions."]

Pressurizer and Supports

[Insert technical evaluation for pressurizer and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the pressurizer and supports are adequate under the proposed EPU conditions."]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, the NRC staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

11PMA-DR1CR3-2S-000259

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The NRC staff's review covered (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on the reactor internals and core supports. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, and GDC-10 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

11PMA-DR1CR3-2S-000260

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's staff's review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of Section XI of the ASME B&PV Code and the ASME Operations and Maintenance (O&M) Code, as applicable. The NRC staff's review focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covered any impacts that the proposed EPU may have on the licensee's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. The NRC staff also evaluated the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 37, GDC 40, GDC 43, and GDC 46, insofar as they require that the ECCS, the containment heat removal system, the containment atomospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessments related to the functional performance of safety-related valves and pumps and concludes that the licensee has adequately addressed the effects of the proposed EPU on safety-related pumps and valves. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related power-operated valves. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the requirements of GDC-1, GDC-37, GDC-40, GDC-43, GDC-46, GDC-54, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to safety-related valves and pumps.

11PMA-DR1CR3-2S-000261

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; (3) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (4) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (5) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (6) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed EPU on this equipment and (2) demonstrated that the equipment will continue to meet the requirements of GDCs 1, 2, 4, 14, and 30; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

11PMA-DR1CR3-2S-000262

[2.2.6 Additional Review Areas (Mechanical and Civil Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000263

INSERT 3

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000264

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The NRC staff's review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The NRC staff's review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EQ of electrical equipment and concludes that the licensee has adequately addressed the effects of the proposed EPU on the environmental conditions for and the qualification of electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meets the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

11PMA-DR1CR3-2S-000265

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on GDC-17. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The NRC staff further concludes that the impact of the proposed EPU on grid stability is insignificant. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

11PMA-DR1CR3-2S-000266

2.3.3 AC Onsite Power System

Regulatory Evaluation

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ac onsite power system.

11PMA-DR1CR3-2S-000267

2.3.4 DC Onsite Power System

Regulatory Evaluation

The direct current (dc) onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covered the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the dc onsite power system.

11PMA-DR1CR3-2S-000268

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the LOOP concurrent with a turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources" (AACs). The NRC staff's review focused on the impact of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed EPU on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SBO.

11PMA-DR1CR3-2S-000269

[2.3.6 Additional Review Areas (Electrical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000270

INSERT 4

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000271

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducted a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

11PMA-DR1CR3-2S-000272

[2.4.2 Additional Review Areas (Instrumentation and Controls)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000273

INSERT 5

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000274

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

The NRC staff conducted a review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NRC staff's review covered flooding of SSCs important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focused on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. The NRC's acceptance criteria for flood protection are based on GDC-2. Specific review criteria are contained in SRP Section 3.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the proposed changes in fluid volumes in tanks and vessels for the proposed EPU. The NRC staff concludes that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of GDC-2 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to flood protection.

11PMA-DR1CR3-2S-000275

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The EFDS is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system. The NRC staff's review of the EFDS included the collection and disposal of liquid effluents outside containment.
The NRC staff's review focused on any changes in fluid volumes or pump capacities that are necessary for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations. The NRC's acceptance criteria for the EFDS are based on GDCs 2 and 4 insofar as they require the EFDS to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures). Specific review criteria are contained in SRP Section 9.3.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EFDS and concludes that the licensee has adequately accounted for the plant changes resulting in increased water volumes and larger capacity pumps or piping systems. The NRC staff concludes that the EFDS has sufficient capacity to (1) handle the additional expected leakage resulting from the plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensure that contaminated fluids are not transferred to noncontaminated drainage systems. Based on this, the NRC staff concludes that the EFDS will continue to meet the requirements of GDCs 2 and 4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EFDS.

11PMA-DR1CR3-2S-000276

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The NRC staff's review of the CWS focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed to accommodate the proposed EPU. The NRC's acceptance criteria for the CWS are based on GDC-4 for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs. Specific review criteria are contained in SRP Section 10.4.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the modifications to the CWS and concludes that the licensee has adequately evaluated these modifications. The NRC staff concludes that, consistent with the requirements of GDC-4, the increased volumes of fluid leakage that could potentially result from these modifications would not result in the failure of safety-related SSCs following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CWS.

11PMA-DR1CR3-2S-000277

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NRC staff's review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The NRC staff's review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviewed the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NRC staff's review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. The NRC's acceptance criteria for the protection SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on GDC-4. Specific review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to internally generated missiles.

11PMA-DR1CR3-2S-000278

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The NRC staff's review of the turbine generator focused on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on GDC-4, and relates to protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. Specific review criteria are contained in SRP Section 10.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the turbine generator and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The NRC staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine generator.

11PMA-DR1CR3-2S-000279

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducted a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The NRC staff's review of pipe failures included high and moderate energy fluid system piping located outside of containment. The NRC staff's review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of postaccident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on GDC-4, which requires, in part, that SSCs important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. Specific review criteria are contained in SRP Section 3.6.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the changes that are necessary for the proposed EPU and the licensee's proposed operation of the plant, and concludes that SSCs important to safety will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

11PMA-DR1CR3-2S-000280

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's fire-related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and GDCs 3 and 5 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to fire protection.

11PMA-DR1CR3-2S-000281

2.5.2 Pressurizer Relief Tank

Regulatory Evaluation

The pressurizer relief tank (PRT) is a pressure vessel provided to condense and cool the discharge from the pressurizer safety and relief valves. The tank is design with a capacity to absorb discharge fluid from the pressurizer relief valve during a specified step-load decrease. The PRT system is not safety-related and is not designed to accept a continuous discharge from the pressurizer. The NRC staff conducted a review of the PRT to ensure that operation of the tank is consistent with transient analyses of related systems at the proposed EPU level, and that failure or malfunction of the PRT system will not adversely affect safety-related SSCs. The NRC staff's review focused on any design changes related to the PRT and connected piping, and changes related to operational assumptions that are necessary in support of the proposed EPU that are not bounded by previous analyses. In general, the steam condensing capacity of the tank and the tank rupture disk relief capacity should be adequate, taking into consideration the capacity of the pressurizer power-operated relief and safety valves; the piping to the tank should be adequately sized; and systems inside containment should be adequately protected from the effects of high-energy line breaks and moderate-energy line cracks in the pressurizer relief system. The NRC's acceptance criteria for the PRT are based on: (1) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes; and (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles. Specific review criteria are contained in SRP Section 5.4.11.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the increase in pressurizer discharge to the PRT as a result of the proposed EPU and concludes that (1) the PRT will operate in a manner consistent with transient analyses of related systems and (2) safety-related SSCs will continue to be protected against failure of the PRT consistent with GDCs 2 and 4. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the PRT.

11PMA-DR1CR3-2S-000282

2.5.3 Fission Product Control

2.5.3.1 Fission Product Control Systems and Structures

Regulatory Evaluation

The NRC staff's review for fission product control systems and structures covered the basis for developing the mathematical model for DBLOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The NRC staff's review primarily focused on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria are based on GDC-41, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on fission product control systems and structures. The NRC staff concludes that the licensee has adequately accounted for the increase in fission products and changes in expected environmental conditions that would result from the proposed EPU. The NRC staff further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the fission product control systems and structures will continue to meet the requirements of GDC-41. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fission product control systems and structures.

11PMA-DR1CR3-2S-000283

2.5.3.2 Main Condenser Evacuation System

Regulatory Evaluation

The main condenser evacuation system (MCES) generally consists of two subsystems: (1) the "hogging" or startup system which initially establishes main condenser vacuum and (2) the system which maintains condenser vacuum once it has been established. The NRC staff's review focused on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the MCES are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the MCES and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The NRC also concludes that the MCES will continue meet the requirements of GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MCES.

11PMA-DR1CR3-2S-000284

2.5.3.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment. The NRC staff reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the turbine gland sealing system and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine gland sealing system.

11PMA-DR1CR3-2S-000285

2.5.4 Component Cooling and Decay Heat Removal

2.5.4.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review for the proposed EPU focused on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions. The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and (3) GDC-61, insofar as it requires that fuel storage systems be designed with RHR capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions. Specific review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the spent fuel pool cooling and cleanup system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. Based on this review, the NRC staff concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of GDCs 5, 44, and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the spent fuel pool cooling and cleanup system.

11PMA-DR1CR3-2S-000286

2.5.4.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to non-safety-related auxiliary components that are used for normal plant operation. The NRC staff's review covered the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with the LOOP). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.1, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the station SWS and concludes that the licensee has adequately accounted for the increased heat loads on system performance that would result from the proposed EPU. The NRC staff concludes that the station SWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the station SWS will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the station SWS.

11PMA-DR1CR3-2S-000287

2.5.4.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

The NRC staff's review covered reactor auxiliary cooling water systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and mitigating the consequences of accident conditions, or (2) preventing the occurrence of an accident. These systems include closed-loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS. The NRC staff's review covered the capability of the auxiliary cooling water systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. Emphasis was placed on the cooling water systems for safety-related components (e.g., ECCS equipment, ventilation equipment, and reactor shutdown equipment). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.2, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the reactor auxiliary cooling water systems and concludes that the licensee has adequately accounted for the increased heat loads from the proposed EPU on system performance. The NRC staff concludes that the reactor auxiliary cooling water systems will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the reactor auxiliary cooling water systems will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the reactor auxiliary cooling water systems.

11PMA-DR1CR3-2S-000288

2.5.4.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The NRC staff's review focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review included evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The NRC's acceptance criteria for the UHS are based on (1) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety; and (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the information that was provided by the licensee for addressing the effects that the proposed EPU would have on the UHS safety function, including the licensee's validation of the design-basis UHS temperature limit based on post-licensing data. Based on the information that was provided, the NRC staff concludes that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of GDCs 5 and 44 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the UHS.

11PMA-DR1CR3-2S-000289

2.5.4.5 Auxiliary Feedwater System

Regulatory Evaluation

In conjunction with a seismic Category I water source, the auxiliary feedwater system (AFWS) functions as an emergency system for the removal of heat from the primary system when the main feedwater system is not available. The AFWS may also be used to provide decay heat removal necessary for withstanding or coping with an SBO. The NRC staff's review for the proposed EPU focused on the system's continued ability to provide sufficient emergency feedwater flow at the expected conditions (e.g, steam generator pressure) to ensure adequate cooling with the increased decay heat. The NRC staff's review also considered the effects of the proposed EPU on the likelihood of creating fluid flow instabilities (e.g., water hammer) during normal plant operation, as well as during upset or accident conditions. The NRC's acceptance criteria for the AFWS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-19, insofar as it requires that equipment at appropriate locations outside the control room be provided with (a) the capability for prompt hot shutdown of the reactor, and (b) a potential capability for subsequent cold shutdown of the reactor; (4) GDC-34, insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure; and (5) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure. Specific review criteria are contained in SRP Section 10.4.9.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the AFWS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the AFWS to supply adequate water to the SGs to ensure adequate cooling of the core. The NRC staff finds that the AFWS will continue meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the AFWS will continue to meet the requirements of GDCs 4, 5, 19, 34, and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the AFWS.

11PMA-DR1CR3-2S-000290

2.5.5 Balance-of-Plant Systems

2.5.5.1 Main Steam

Regulatory Evaluation

The main steam supply system (MSSS) transports steam from the NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. The NRC staff's review focused on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC's acceptance criteria for the MSSS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-34, insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MSSS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MSSS. The NRC staff concludes that the MSSS will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. The NRC staff further concludes that the MSSS will continue to meet the requirements of GDCs 4 and 5. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSSS.

11PMA-DR1CR3-2S-000291

2.5.5.2 Main Condenser

Regulatory Evaluation

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system (TBS). The NRC staff's review focused on the effects of the proposed EPU on the steam bypass capability with respect to load rejection assumptions, and on the ability of the MC system to withstand the blowdown effects of steam from the TBS. The NRC's acceptance criteria for the MC system are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 10.4.1. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MC system and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MC system. The NRC staff concludes that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the TBS and thereby continue to meet GDC-60 for prevention of the consequences of failures in the system. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MC system.

11PMA-DR1CR3-2S-000292

2.5.5.3 Turbine Bypass

Regulatory Evaluation

The TBS is designed to discharge a stated percentage of rated main steam flow directly to the MC system, bypassing the turbine. This steam bypass enables the plant to take step load reductions up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control SG pressure. The NRC staff's review focused on the effects that EPU has on load rejection capability, analysis of postulated system piping failures, and on the consequences of inadvertent TBS operation. The NRC's acceptance criteria for the TBS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; and (2) GDC-34, insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the TBS. The NRC staff concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the system. The NRC staff concludes that the TBS will continue to provide a means for shutting down the plant during normal operations. The NRC staff further concludes that TBS failures will not adversely affect essential systems or components. Based on this, the NRC staff concludes that the TBS will continue to meet GDCs 4 and 34. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the TBS.

11PMA-DR1CR3-2S-000293

2.5.5.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at the appropriate temperature, pressure, and flow rate to the steam generators. The only part of the CFS classified as safety-related is the feedwater piping from the SGs up to and including the outermost containment isolation valve. The NRC staff's review focused on the effects of the proposed EPU on previous analyses and considerations with respect to the capability of the CFS to supply adequate feedwater during plant operation and shutdown, and to isolate components, subsystems, and piping in order to preserve the system's safety function. The NRC staff's review also considered the effects of the proposed EPU on the feedwater system, including the AFWS piping entering the SG, with regard to possible fluid flow instabilities (e.g., water hammer) during normal plant operation, as well as during upset or accident conditions. The NRC's acceptance criteria for the CFS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the CFS. The NRC staff concludes that the CFS will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. The NRC staff further concludes that the CFS will continue to meet the requirements of GDCs 4, 5, and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CFS.

11PMA-DR1CR3-2S-000294

2.5.6 Waste Management Systems

2.5.6.1 Gaseous Waste Management Systems

Regulatory Evaluation

Gaseous waste management systems involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of the condenser air removal system, the steam generator blowdown flash tank, and the containment purge exhausts; and the building ventilation system exhausts. The NRC staff's review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exist. The NRC's acceptance criteria for the gaseous waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50 Appendix I, Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion. Specific review criteria are contained in SRP Section 11.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the gaseous waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC staff finds that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the gaseous waste management systems will continue to meet the requirements of 10 CFR 20.1302, GDCs 3, 60, and 61, and 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the gaseous waste management systems.

11PMA-DR1CR3-2S-000295

2.5.6.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for liquid waste management systems focused on the effects that the proposed EPU may have on previous analyses and considerations related to the liquid waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion. Specific review criteria are contained in SRP Section 11.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the liquid waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management systems to control releases of radioactive materials. The NRC staff finds that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the liquid waste management systems will continue to meet the requirements of 10 CFR 20.1302, GDCs 60 and 61, and 10 CFR Part 50, Appendix I, Sections II.A and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the liquid waste management systems.

11PMA-DR1CR3-2S-000296

2.5.6.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for the solid waste management systems (SWMS) focused on the effects that the proposed EPU may have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the SWMS. The NRC's acceptance criteria for the SWMS are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels, (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SWMS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the SWMS to process the waste. The NRC staff finds that the SWMS will continue to meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the SWMS will continue to meet the requirements of 10 CFR 20.1302, GDCs 60, 63, and 64, and 10 CFR Part 71. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SWMS.

11PMA-DR1CR3-2S-000297

2.5.7 Additional Considerations

2.5.7.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure. The NRC staff's review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the amount of required fuel oil for the emergency diesel generators and concludes that the licensee has adequately accounted for the effects of the increased electrical demand on fuel oil consumption. The NRC staff concludes that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of GDCs 4, 5, and 17. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel oil storage and transfer system.

11PMA-DR1CR3-2S-000298

2.5.7.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks. The NRC staff's review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review focused on the effects of the new fuel on system performance and related analyses. The NRC's acceptance criteria for the LLHS are based on (1) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC-62, insofar as it requires that criticality be prevented. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that the licensee has adequately incorporated the effects of the new fuel in the analyses. Based on this review, the NRC staff further concludes that the LLHS will continue to meet the requirements of GDCs 61 and 62 for radioactivity releases and prevention of criticality accidents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LLHS.

11PMA-DR1CR3-2S-000299

[2.5.8 Additional Review Areas (Plant Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000300

INSERT 6

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000301

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident.

NOTE: Use the following paragraph in the regulatory evaluation and the conclusion section provided below for dry containments, including subatmospheric containments

The NRC staff's review covered the pressure and temperature conditions in the containment due to a spectrum of postulated LOCAs and secondary system line-breaks. The NRC's acceptance criteria for primary containment functional design are based on (1) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (2) GDC-50, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; (3) GDC-38, insofar as it requires that the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions; and (5) GDC-64, insofar as it requires that means be provided for monitoring the plant environs for radioactivity that may be released from normal operations and postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.A.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment pressure and temperature transient and concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that the containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will continue to meet the requirements of GDCs 13, 16, 38, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment functional design.

11PMA-DR1CR3-2S-000302

NOTE: Use the following paragraph in the regulatory evaluation and the conclusion section provided below for ice condenser containments

The NRC staff's review covered the pressure and temperature conditions in the containment due to a spectrum of LOCAs and secondary system line-breaks, the design of the ice condenser system, and the maximum allowable operating deck steam bypass area for a full spectrum of RCS pipe breaks. The NRC's acceptance criteria for primary containment functional design are based on (1) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (2) GDC-50, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; (3) GDC-38, insofar as it requires that the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions; and (5) GDC-64, insofar as it requires that means be provided for monitoring the plant environs for radioactivity that may be released from normal operations and postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.B.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment pressure and temperature transient and concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will continue to meet the requirements of GDCs 13, 16, 38, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment functional design.

11PMA-DR1CR3-2S-000303

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects, and (2) GDC-50, insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization resulting from the increased mass and energy release. The NRC staff concludes that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. Based on this, the NRC staff concludes that the plant will continue to meet GDCs 4 and 50 for the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to subcompartment analyses.

11PMA-DR1CR3-2S-000304

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss of Coolant

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) GDC-50, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the NRC staff finds that the mass and energy release analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated LOCA.

11PMA-DR1CR3-2S-000305

2.6.3.2 Mass and Energy Release Analysis for Secondary System Pipe Ruptures

Regulatory Evaluation

The NRC staff's review covered the energy sources that are available for release to the containment, the mass and energy release rate calculations, and the single-failure analyses performed for steam and feedwater line isolation provisions, which would limit the flow of steam or feedwater to the assumed pipe rupture. The NRC's acceptance criteria for mass and energy release analysis for secondary system pipe ruptures are based on GDC-50, insofar as it requires that the margin in the design of the containment structure reflect consideration of the effects of potential energy sources that have not been included in the determination of peak conditions, the experience and experimental data available for defining accident phenomena and containment response, and the conservatism of the model and the values of input parameters. Specific review criteria are contained in SRP Section 6.2.1.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the mass and energy release assessment performed by the licensee for postulated secondary system pipe ruptures and finds that the licensee has adequately addresses the effects of the proposed EPU. Based on this, the NRC staff concludes that the analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative (i.e., that the analysis includes sufficient margin). Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated secondary system pipe ruptures.

11PMA-DR1CR3-2S-000306

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covered (1) the production and accumulation of combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NRC staff's review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-41, insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; (4) GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection; and (5) GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing. **[*Include the following sentence for PWRs with ice condenser containments*: Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with an ice condenser type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.]** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities, consistent with the requirements in 10 CFR 50.44, 10 CFR 50.46, and GDCs 5, 41, 42, and 43 as discussed above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

11PMA-DR1CR3-2S-000307

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment sump. The NRC staff's review in this area focused on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers. The NRC's acceptance criteria for containment heat removal are based on GDC-38, insofar as it requires that the containment heat removal system be capable of rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels. Specific review criteria are contained in SRP Section 6.2.2 as supplemented by Draft Guide (DG) 1107.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed EPU. The NRC staff finds that the systems will continue to meet GDC-38 for rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment heat removal systems.

11PMA-DR1CR3-2S-000308

2.6.6 Pressure Analysis for ECCS Performance Capability

Regulatory Evaluation

Following a LOCA, the ECCS will supply water to the reactor vessel to reflood, and thereby cool the reactor core. The core flooding rate will increase with increasing containment pressure. The NRC staff reviewed analyses of the minimum containment pressure that could exist during the period of time until the core is reflooded to confirm the validity of the containment pressure used in ECCS performance capability studies. The NRC staff's review covered assumptions made regarding heat removal systems, structural heat sinks, and other heat removal processes that have the potential to reduce the pressure. The NRC's acceptance criteria for the pressure analysis for ECCS performance capability are based on 10 CFR 50.46, insofar as it requires the use of an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs or an ECCS evaluation model developed in conformance with 10 CFR Part 50, Appendix K. Specific review criteria are contained in SRP Section 6.2.1.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the impact that the proposed EPU would have on the minimum containment pressure analysis and concludes that the licensee has adequately addressed this area of review to ensure that the requirements in 10 CFR 50.46 regarding ECCS performance will continue to be met following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to minimum containment pressure for ECCS performance.

11PMA-DR1CR3-2S-000309

[2.6.7 Additional Review Areas (Containment Review Considerations)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000310

INSERT 7

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000311

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staff's review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU. The NRC staff further concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of GDCs 4 and 19. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

11PMA-DR1CR3-2S-000312

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in postaccident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., emergency or postaccident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for the ESF atmosphere cleanup systems are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC 41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; (3) GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC-64, insofar as it requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of GDCs 19, 41, 61, and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

11PMA-DR1CR3-2S-000313

2.7.3 Ventilation Systems

2.7.3.1 Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, the NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of GDCs 4, 19, and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

11PMA-DR1CR3-2S-000314

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel handling accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC's acceptance criteria for the SFPAVS are based on (1) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents, and (2) GDC-61, insofar as it requires that systems which contain radioactivity be designed with appropriate confinement and containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the SFPAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's capability to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the NRC staff concludes that the SFPAVS will continue to meet the requirements of GDCs 60 and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SFPAVS.

11PMA-DR1CR3-2S-000315

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system (ARAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the ARAVS and TAVS are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ARAVS and TAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the auxiliary and radwaste equipment areas and in the turbine area, permit personnel access, control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the ARAVS and TAVS will continue to meet the requirements of GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ARAVS and the TAVS.

11PMA-DR1CR3-2S-000316

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review for the ESFVS focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the ESFVS will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESFVS will continue to meet the requirements of GDCs 4, 17 and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

11PMA-DR1CR3-2S-000317

[2.7.7 Additional Review Areas (Habitability, Filtration, and Ventilation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000318

INSERT 8

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000319

2.8 Reactor Systems

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, GDC-27, and GDC-35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel system design.

11PMA-DR1CR3-2S-000320

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

11PMA-DR1CR3-2S-000321

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the nuclear design.

11PMA-DR1CR3-2S-000322

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is **[equivalent to or a justified extrapolation from]** proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed EPU on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs 10 and 12 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to thermal and hydraulic design.

11PMA-DR1CR3-2S-000323

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient cooling exists to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29 following implementation of the proposed EPU.

11PMA-DR1CR3-2S-000324

Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

11PMA-DR1CR3-2S-000325

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered pressurizer relief and safety valves and the piping from these valves to the quench tank and RCS relief and safety valves. The NRC's acceptance criteria are based on (1) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs and (2) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee has (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to provide adequate protection to meet GDC-15 and GDC-31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

11PMA-DR1CR3-2S-000326

2.8.4.3 Overpressure Protection During Low Temperature Operation

Regulatory Evaluation

Overpressure protection for the reactor coolant pressure boundary (RCPB) during low temperature operation of the plant is provided by pressure-relieving systems that function during the low temperature operation. The NRC staff's review covered relief valves with piping to the quench tank, the makeup and letdown system, and the residual heat removal (RHR) system which may be operating when the primary system is water solid. The NRC's acceptance criteria are based on (1) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during low temperature operation. The NRC staff concludes that the licensee has (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the low temperature overpressure protection features will continue to provide adequate protection to meet GDC-15 and GDC-31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during low temperature operation.

11PMA-DR1CR3-2S-000327

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-34, which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of GDCs 4, 5, and 34 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

11PMA-DR1CR3-2S-000328

2.8.5 Accident and Transient Analyses

2.8.5.1. Increase in Heat Removal by the Secondary System

2.8.5.1.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; (3) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and (4) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

11PMA-DR1CR3-2S-000329

2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment

Regulatory Evaluation

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions; (2) methods of thermal and hydraulic analyses; (3) the sequence of events; (4) assumed responses of the reactor coolant and auxiliary systems; (5) functional and operational characteristics of the reactor protection system; (6) operator actions; (7) core power excursion due to power demand created by excessive steam flow; (8) variables influencing neutronics; and (9) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC-35, insofar as it requires the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.1.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of steam system piping failure events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of a propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, 31, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to steam system piping failures.

11PMA-DR1CR3-2S-000330

2.8.5.2 Decrease in Heat Removal By the Secondary System

2.8.5.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

11PMA-DR1CR3-2S-000331

2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

11PMA-DR1CR3-2S-000332

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events. (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

11PMA-DR1CR3-2S-000333

2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment

Regulatory Evaluation

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break) or a RCS heatup (by reducing feedwater flow to the affected RCS). In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed response of the reactor coolant and auxiliary systems, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC-35, insofar as it requires the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.2.8 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of feedwater system pipe breaks and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, 31, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to feedwater system pipe breaks.

11PMA-DR1CR3-2S-000334

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

11PMA-DR1CR3-2S-000335

2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor system components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (3) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

11PMA-DR1CR3-2S-000336

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design necessary for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

11PMA-DR1CR3-2S-000337

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal at power event and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power.

11PMA-DR1CR3-2S-000338

2.8.5.4.3 Control Rod Misoperation

Regulatory Evaluation

The NRC staff's review covered the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations; (2) the sequence of events; (3) the analytical model used for analyses; (4) important inputs to the calculations; and (5) the results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to initiate automatically operation of systems and components important to safety under accident conditions; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of control rod misoperation events and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs will not be exceeded during normal or anticipate operational transients. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to control rod misoperation events.

11PMA-DR1CR3-2S-000339

2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to automatically initiate the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences; (3) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during AOOs; (4) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (5) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inactive loop startup event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

11PMA-DR1CR3-2S-000340

2.8.5.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

Regulatory Evaluation

Unborated water can be added to the RCS, via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator should stop this unplanned dilution before the shutdown margin is eliminated. The NRC staff's review covered (1) conditions at the time of the unplanned dilution, (2) causes, (3) initiating events, (4) the sequence of events, (5) the analytical model used for analyses, (6) the values of parameters used in the analytical model, and (7) results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; (2) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.4.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in boron concentration in the reactor coolant due to a CVCS malfunction and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in boron concentration in the reactor coolant due to a CVCS malfunction.

11PMA-DR1CR3-2S-000341

2.8.5.4.6 Spectrum of Rod Ejection Accidents

Regulatory Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The NRC staff evaluates the consequences of a control rod ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covered initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.8 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the rod ejection accident and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the rod ejection accident.

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2.8.5.5 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS and CVCS event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS and CVCS event.

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2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of Pressurizer Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. A reactor trip normally occurs due to low RCS pressure. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressurizer pressure relief valve event.

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2.8.5.6.2 Steam Generator Tube Rupture

Regulatory Evaluation

A steam generator tube rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and main steam safety or atmospheric relief valves. Reactor protection and ESFs are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR Part 100. The NRC staff's review covered (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analysis, (3) the sequence of events (assuming offsite power either available or unavilable), (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions consistent with the plant's emergency operating procedures (EOPs), and (7) the results of the accident analysis. A single failure of a mitigating system is assumed for this event. The NRC staff's review of the SGTR is focused on the thermal and hydraulic analysis for the SGTR in order to (1) determine whether 10 CFR Part 100 is satisfied with respect to radiological consequences, which are discussed in Section 2.7 of this safety evaluation and (2) confirm that the faulted SG does not experience an overfill. Preventing SG overfill is necessary in order to prevent the failure of main steam lines. Specific review criteria are contained in SRP Section 15.6.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analysis of the SGTR accident and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the faulted SG. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SGTR event.

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2.8.5.6.3 Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; (4) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 4, 27, 35, and 10 CFR 50.46 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LOCA.

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2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

Anticipated transients without scram (ATWS) is defined as an anticipated operational occurrence followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- each PWR must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent from the existing reactor trip system, and

- each PWR manufactured by Combustion Engineering (CE) or Babcock and Wilcox (B&W) must have a diverse scram system (DSS). This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

The NRC staff's review was conducted to ensure that (1) the above requirements are met, and (2) the setpoints for the ATWS mitigating system actuation circuitry (AMSAC) and DSS remain valid for the proposed EPU. In addition, for plants where a DSS is not specifically required by 10 CFR 50.62, the NRC staff verified that the consequences of an ATWS are acceptable. The acceptance criterion is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient (MTC) and the primary system relief capacity. The NRC staff reviewed (1) the limiting event determination, (2) the sequence of events, (3) the analytical model and its applicability, (4) the values of parameters used in the analytical model, and (5) the results of the analyses. *Insert the following sentence if the licensee relied upon generic vendor analyses* **[The NRC staff reviewed the licensee's justification of the applicability of generic vendor analyses to its plant and the operating conditions for the proposed EPU.]** Review guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on ATWS. The NRC staff concludes that the licensee has demonstrated that the AMSAC **[and DSS]** will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed EPU. **[*For plants not required to install DSS, use the following sentence:* The licensee has shown that the plant is not required by 10 CFR 50.62 to have a DSS. Additionally, the licensee has demonstrated, as explained above, that the peak primary**

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system pressure following an ATWS event will remain below the acceptance limit of 3200 psig.] Therefore, the NRC staff finds the proposed EPU acceptable with respect to ATWS.

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2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on GDC-62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of GDC-62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the new fuel storage.

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2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covered the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (2) GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. The NRC staff also concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, the NRC staff concludes that the spent fuel storage facilities will continue to meet the requirements of GDCs 4 and 62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to spent fuel storage.

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[2.8.7 Additional Review Areas (Reactor Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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INSERT 9

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

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2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's [**Updated Safety Analysis Report *or* Updated Final Safety Analysis Report**] related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

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NOTE: Use Sections 2.9.2 and 2.9.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.9.2. Radiological Consequences Analyses Using Alternative Source Terms

NOTE: There are two cases that may be encountered here: (1) a licensee may be implementing an alternative source term for the first time, or (2) a licensee may have already fully implemented an alternative source term and is revising the previously approved dose analyses that use alternative source term methodologies. The second paragraph for each heading is only needed for a first-time implementation of an alternative source term (either partial or full implementations). Several accidents may have been analyzed - see corresponding SRP sections for further regulatory evaluation text (to be modified), as needed.

Regulatory Evaluation

The NRC staff reviewed the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod ejection accident (REA), MSLB, SGTR, and locked-rotor accident. The NRC staff's review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternate source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident. Specific review criteria are contained in SRP Section 15.0.1.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff reviewed the implementation of alternative source terms. The NRC's acceptance criteria for implementation of an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for the implementation of an alternative source term in current operating nuclear power plants; (2) 10 CFR 50.49, insofar as it requires qualification of safety-related equipment, as defined in that section, including and based on integrated radiation dose during normal and accident conditions; (3) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident; (4) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, insofar as it requires a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency; and (5) plant-specific licensing commitments made in response to NUREG-0737 (Items II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2, and III.D.3.4). Specific review criteria are contained in SRP Sections 15.0.1.

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Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has adequately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC-19, as well as applicable acceptance criteria denoted in SRP 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff has reviewed the alternative source term methodology used by the licensee in evaluating the effects of the proposed EPU and concludes that changes continue to provide a sufficient margin of safety with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and parameter inputs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the implementation of an alternative source term.

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[2.9.3 Additional Review Areas (Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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NOTE: Use Sections 2.9.2 - 2.9.10 below if the licensee's radiological consequences analyses are not based on an alternative source term (i.e., if the analyses are based on traditional source term, based on TID-14844)

2.9.2. Radiological Consequences of Main Steamline Failures Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a main steamline break (MSLB) outside the containment. The NRC staff's review included (1) the sequence of events, models and assumptions used by the licensee for the calculation of the radiological doses; (2) evaluation of the TSs on the primary and secondary coolant iodine activities; and (3) determination of reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.1.5.A, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary meet the exposure guideline values specified in 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for the concurrent iodine spike. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of MSLB accidents outside the containment.

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2.9.3 Radiological Consequences of a Reactor Coolant Pump Locked-Rotor Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a reactor coolant pump locked-rotor accident. The review included (1) determination of a need for a radiological consequences analysis; and (2) the sequence of events, models and assumptions used by the licensee for the calculation of radiological doses. The NRC's acceptance criteria for the radiological consequences of a reactor coolant pump locked-rotor accident are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.3.3-15.3.4; and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised analyses for the radiological consequences of a reactor coolant pump locked rotor and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated locked-rotor accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are a small fraction of exposure guideline values specified in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a locked-rotor accident.

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2.9.4 Radiological Consequences of a Control Rod Ejection Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a control rod ejection accident. The NRC staff's review included the plant response to a control rod ejection accident and the calculation of radiological doses at the EAB and LPZ outer boundary and in the control room due to the releases resulting from a rod ejection accident. The purpose of the NRC staff's review was to (1) ensure that plant's procedures for recovery from a rod ejection accident and the plant's TSs are properly taken into account in computing the doses and (2) compare the calculated doses against the appropriate guidelines. The NRC's acceptance criteria for the radiological consequences of a control rod ejection accident are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.4.8.A, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a rod ejection accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated control rod ejection accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values specified in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a control rod ejection accident.

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2.9.5 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review included (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of the failure of small lines carrying primary coolant outside containment are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) GDC-55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are substantially below the exposure guideline values of 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary.

11PMA-DR1CR3-2S-000360

2.9.6 Radiological Consequences of Steam Generator Tube Rupture

Regulatory Evaluation

The NRC staff reviewed the analysis of the radiological consequences of a postulated steam generator tube rupture (SGTR). The NRC staff's review included (1) a review of the sequence of events and plant procedures for recovery from the accident to ensure that the most severe case of radioactive releases has been considered; (2) a review of the models and assumptions for the calculation of the radiological doses for the postulated accident; (3) an evaluation of the TSs on the primary and secondary coolant iodine activity concentration; and (4) an evaluation of the radiological consequences of an SGTR concurrent with a loss of offsite power and the most limiting single failure. The NRC staff's review included two cases for the reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an SGTR are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.6.3, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an SGTR and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of an SGTR accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for the concurrent iodine spike. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of an SGTR.

11PMA-DR1CR3-2S-000361

2.9.7 Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a design-basis LOCA. The review included a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also included (1) the methodology and results of calculations of the radiological consequences resulting from containment and ESF component leakage following a hypothetical LOCA; and (2) an assessment of the containment with respect to the assumptions and the values of input parameters for the dose calculations. The NRC staff's calculations are based on pertinent information in the **[Updated Safety Analysis Report *or* Updated Final Safety Analysis Report]** and considered the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 6.4 and Appendices A and B of SRP Section 15.6.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis LOCA and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis LOCA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 and the calculated doses in the control room meet the requirements of GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a design-basis LOCA.

11PMA-DR1CR3-2S-000362

2.9.8 Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a postulated FHA. The purpose of this review was to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review included (1) the sequence of events, models, and assumptions used by the licensee for the calculation of radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC 61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of FHAs and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated whole-body and thyroid doses at the EAB and the LPZ boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of FHAs.

11PMA-DR1CR3-2S-000363

2.9.9 Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review was conducted to verify various design and operations aspects of the system. The NRC staff's review included (1) determining a need for a design-basis radiological analysis; (2) sequence of events, models and assumptions used by the licensee for the calculation of the radiological doses; and (3) comparing the calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitablity. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a spent fuel cask drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident since the calculated whole-body and thyroid doses at the EAB and LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of spent fuel cask drop accidents.

11PMA-DR1CR3-2S-000364

[2.9.10 Additional Review Areas (Source Terms and Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000365

INSERT 10

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000366

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 and GDC-19. Specific review criteria are contained in SRP Sections 12.2, 12.3,12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. The NRC staff further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

11PMA-DR1CR3-2S-000367

[2.10.2 Additional Review Areas (Health Physics)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000368

INSERT 11

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000369

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implemented the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area. The licensee has addressed these questions in its application. Following are the NRC staff's questions, the licensee's responses, and the NRC staff's evaluation of the responses.

1. Changes in Emergency and Abnormal Operating Procedures

   Describe how the proposed EPU will change the plant emergency and abnormal operating procedures. (SRP Section 13.5.2.1)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

2. Changes to Operator Actions Sensitive to Power Uprate

   Describe any new operator actions needed as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU. (SRP Section 18.0)

   (i.e., Identify and describe operator actions that will involve additional response time or will have reduced time available. Your response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the acceptability of these changes).

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

11PMA-DR1CR3-2S-000370

3. Changes to Control Room Controls, Displays and Alarms

   Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators will be tested to determine they could use the instruments reliably. (SRP Section 18.0)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

4. Changes on the Safety Parameter Display System

   Describe any changes to the safety parameter display system resulting from the proposed EPU. How will the operators know of the changes? (SRP Section 18.0)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

5. Changes to the Operator Training Program and the Control Room Simulator

   Describe any changes to the operator training program and the plant referenced control room simulator resulting from the proposed EPU, and provide the implementation schedule for making the changes. (SRP Sections 13.2.1 and 13.2.2)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that the licensee has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of GDC-19, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the human factors aspects of the required system changes.

11PMA-DR1CR3-2S-000371

[2.11.2 Additional Review Areas (Human Performance)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000372

INSERT 12

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000373

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The NRC staff's review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The staff has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. The staff concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Further, the staff finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test program acceptable.

11PMA-DR1CR3-2S-000374

[2.12.2 Additional Review Areas (Power Ascension and Testing Plan)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000375

INSERT 13

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

11PMA-DR1CR3-2S-000376

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

Regulatory Evaluation

The licensee conducted a risk evaluation to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are present if any issue would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff's review covered the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff's review covered the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This included a review of the licensee's actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review. The NRC's risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 13 of RS-001 and its attachments.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the risk implications associated with the implementation of the proposed EPU and concludes that the licensee has adequately modeled and/or addressed the potential impacts associated with the implementation of the proposed EPU. The NRC staff further concludes that the results of the licensee's risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the NRC staff finds the risk implications of the proposed EPU acceptable.

11PMA-DR1CR3-2S-000377

[2.13.2 Additional Review Areas (Risk Evaluation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

11PMA-DR1CR3-2S-000378

SECTION 4

INSPECTION GUIDANCE

11PMA-DR1CR3-2S-000379

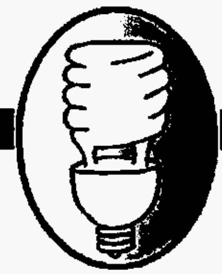
11PMA-DR1CR3-2S-000379

4.1 Inspection Requirements

Inspection Procedure (IP) 71004, "Power Uprates," describes the inspections necessary for power uprate related activities and provides guidance for the inspectors to use in conducting these inspections. In addition, the "Recommended Areas for Inspection" section of the final safety evaluation approving an EPU should be considered by inspectors when selecting a sample for implementing IP 71004. The recommendations in the final safety evaluation do not constitute inspection requirements, but are provided to give the inspectors insight into important bases the NRC staff used for approving the EPU.

11PMA-DR1CR3-2S-000380

JULY 2010



REVIEW OF

Progress Energy Florida's

Project Management Internal Controls

FOR

Nuclear Plant Uprate and Construction Projects

REDACTED

By Authority of
The State of Florida
Public Service Commission
Office of Auditing and Performance Analysis

REDACTED

Review of

Progress Energy Florida's Project Management Internal Controls for Nuclear Plant Uprate and Construction Projects

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Government Analyst II
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July 2010

By Authority of
The State of Florida
Public Service Commission
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PA-10-01-001

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3.0 Crystal River 3 Extended Power Uprate Project

3.1 EPU Key Project Developments

Progress Energy Florida completed Phase II of its three-phase EPU project during the fall 2009 refueling outage. This work included the installation and modification of the major balance of plant components necessary to support the additional MWe output. The company anticipates obtaining an additional 180 MWe output from the EPU project.

In addition to the EPU project, the company performed a steam generator replacement for CR3 during the outage. The generator work was independent and separate from the EPU project and did not directly impact the EPU project or scope. During the steam generator replacement, a delamination occurred in the concrete of the unit's containment vessel wall This event caused the outage to extend past the planned 85-day timeline. The containment vessel delamination issue was solely the result of the steam generation project and in no way connected to the EPU project or work. The unit is still off-line as a result of the delamination repairs.

The delamination did not hinder nor impact the work performed for the EPU The company was able to maintain its original EPU work schedule and complete all work identified for Phase II. However, the extended outage prevented the project team from completing certain testing requirement that can only be performed during start-up. The project team will perform this work at the appropriate time.

3.1.1 Significant Events

The EPU project team states that all work was completed as scheduled and within the allotted budget. During the outage, the project team monitored the work performed for each major component and tracked variances and delays in the schedule. The project team issued daily project updates that tracked the target and actual schedules for each component. The team used these reports to monitor its vendors and identify potential issues. Audit staff reviewed these management reports and verified that the project remained on schedule with minor variances and no major issues were identified during the work.

Once the delamination issue was identified and it became evident that the outage would extend past the planned timeline, the EPU project team made the decision to remain on its original schedule. The team determined that the resources to complete the work were in place and on schedule to finish in the allotted timeline.

The project team estimated a project cost range for the Phase II work and the team states that the costs for the Phase II work was within the range. However, the project team did exceed its original estimates for certain costs during Phase II. Specifically, the company made adjustments for additional resources, such as labor and scaffoldings. These expenditures were contracted through a time and materials contract and remained within the original estimated range.

During the Phase II implementation, PEF's project team monitored the labor and vendor invoices daily to verify that all costs were reasonable and in line with the work scope. Each manager was responsible for monitoring their work assignment to ensure resources were managed accordingly. The Project Controls group oversaw the total overall cost and schedule during the project timeline. This group was responsible for monitoring and booking the costs associated with the work performed by the vendors. Project Controls also ensured the costs were within the budgeted scope and verified all approvals for any changes in scope or costs.

The company has not experienced any problems it considers to have jeopardized the project or its viability However, there are two areas where the project team has re-evaluated its approach—the License Amendment Request Application process and the Low Pressure Turbine replacement. These two issues could have schedule and cost impacts on the project and are discussed below

License Amendment Request Application

The company must receive approval from the NRC to operate the unit at the higher output rate achieved by the EPU project. To initiate this request, the company must submit a License Amendment Request (LAR) with the NRC asking for an amendment to its current operating license. According to the company, the NRC review process should take approximately 14 months. Once approved, the company will be authorized to run the unit at the higher generation output.

It was the company's intent to submit the LAR application in September 2009 This would have provided the NRC approximately 24 months to complete the LAR application review process prior to the planned November 2011 Phase III outage. The company contracted with AREVA, the current owner of the Babcock & Wilcox, to complete its LAR along with an internal team made up from within its corporate licensing division to prepare the application.

The company stated that it prepared for the LAR application process by reviewing and monitoring prior LAR applications, particularly the Ginna Nuclear Plant application from 2005 [31] Also, the company utilized the resources of the Nuclear Energy Institute taskforce for uprates, which provided insight from other utilities completing similar projects. The company notes that this is the first Babcock & Wilcox plant of this type to undergo an EPU project of this scope. PEF created a template, using previous applications as its models, as the framework for completing the initial 116 sections of the LAR application.

The company states that the LAR application process is continually evolving as additional requests are reviewed by the NRC. In mid-2008, the NRC rejected the application for the Monticello Nuclear Plant uprate, specifically related to the NRC's expectation and depth of detail expected in a LAR application. This action caused concern through the industry, In addition, PEF states that the NRC's review and RAIs for the Point Beach Nuclear Plant application (submitted to the NRC in April 2009) signaled another significant shift in the depth of detail required in the applications by the agency

[31] Ginna was the most recent LAR approved by the NRC when the company initiated its licensing development.

Expert Panel

In June 2009 with the LAR work heading to completion, the company convened an "Expert Panel" to review the June 15, 2009 *Final Draft-CR3 EPU Licensing Report* and assess the application from "an NRC acceptance review perspective" and to verify the application contains "sufficient detail to allow NRC to independently conclude acceptability" [32] This panel consisted of both in-house nuclear operations staff and outside consultants versed in nuclear licensing. At the time of review, the company believed it was within three to four months of the LAR submittal. The company acknowledged that the draft "should have been of sufficient quality and content to support the scheduled submittal date of September 2009."[33] At the time, the company had completed 77 of the anticipated 116 sections.

The expert panel presented its analysis and results to company management on July 14, 2009 The panel concluded that the EPU draft would not pass the NRC acceptance review and that the company could not meet its fall 2009 submittal timeline. Specific issues identified in the review include:

> "Many CR-3 LR sections lack sufficient data";
>
> "Portions of many LR sections have been cut/paste from Ginna submittal without a thorough review" Specifically, the application contained Ginna-specific text that was not applicable to the CR3 unit;
>
> RAIs and Safety Evaluation issues raised by the NRC within the Ginna EPU application "were not considered or addressed in the CR-3 LR" application,
>
> "Quality was an issue in sections prepared both by AREVA and CR-3"[34]

In response to the Expert Panel's assessment, the company charged its Manager of Nuclear Regulation to conduct an Adverse Condition Investigation of the LAR process. The purpose of this investigation was to determine the root-cause of the issues identified by the Expert Panel. This investigation identified several reasons for the deficiencies identified by the panel. These include:

> "Inadequate performance monitoring by EPU supervision led to LAR content problems not being identified during the development and review process";
>
> "There was inconsistent, and in some cases insufficient understanding among the CR3 EPU team regarding the level of detail and content needed in the LAR to pass NRC acceptance review and receive NRC approval";

[32] PEF Response to Staff Data Request CR3 1 14, Bates-002043.
[33] PEF Response to Staff Data Request CR3 1.14, Bates 002076.
[34] PEF Response to Staff Data Request CR3 1.14, Bates 002041-002047.

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> Numerous organizational and management changes, and lack of clarity regarding roles and responsibilities adversely impacted organizational effectiveness and contributed to insufficient alignment between EPU Engineering and LAR activities."[35]

The company implemented a corrective action plan to resolve the issues identified by the panel and to strengthen the content of the application. The company hired outside consultants to assist with this restructuring. Specifically, the company determined that its original format template was not adequate in addressing the details necessary for the NRC review The company developed a new template, which required AREVA and the licensing group to restructure the existing application. The Expert Panel completed two additional reviews through January 2010 to monitor the changes incorporated into the LAR application.

AREVA Change Orders

PEF contracted with AREVA to complete "CR3 EPU LAR Re-write Activities"[36] for previously drafted sections of the application. In October 2009, PEF initiated a change order on the AREVA contact for [redacted][37] to perform a three-phase work scope that included re-writes of LAR sections to incorporate the revised template and revise specified portions of the application. In January 2010, the company increased this change order to [redacted]

Additionally in October 2009, the company initiated a separate contract change order to AREVA for [redacted] for additional LAR work. This work was a result of the Expert Panel evaluation and focused on finalizing engineering and design related topics. This contract amount was increased in January 2010 to [redacted][39]. As with the other change order, the increase was for the additional time it took to complete the engineering scope. In total, these two change orders added [redacted][40] to the company's LAR expenditures.

Audit staff recognizes the important role of the Expert Panel and its critical evaluation had in insuring a complete and thorough LAR submittal to the NRC. Given the panel's findings, there was a potential for significant delays in the LAR approval process had the company not commissioned this detailed evaluation. Additionally, the company devised an initial schedule that included a float, which allowed for the necessary time needed for restructuring and strengthening of the application without impacting the project timing. Appropriately, the company performed a root-cause analysis to assess the reasoning for the deviances in its application and developed an action plan to resolve any outstanding issues.

While audit staff acknowledges the importance and value in the self-assessment process used by company, the findings of its Adverse Conditions Investigation are concerning. This

[35] PEF Response to Staff Data Request CR3 1 14, Bates 002080-002081
[36] PEF Response to Staff Data Request CR3 1.22, Bates 000081
[37] Ibid. Bates 000080.
[38] PEF Response to Staff Data Request CR3 4.2, Bates 000001
[39] Ibid. Bates 000011
[40] Ibid. Bates 000021.

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internal PEF investigation notes a lack of understanding, experience, and oversight of the licensing preparation team.

The company points out that the regulatory review process is ever evolving and the NRC's expectations can differ based on the specifics of each application. PEF also believes that the NRC's expectations expanded during the time its licensing group developed its application, based on the NRC's handling of the Monticello and Point Beach EPU applications. The company states that this environment created an uncertainty and lack of expertise within the industry on LAR application. While this may be an accurate description of the evolution of the process, two of the four members of the expert panel were Progress Energy Carolina employees. This indicates that Progress Energy Corporation had the corporate knowledge to assess and evaluate an application. However, these needed resources were not deployed for the CR3 LAR work during the earlier stages of the process.

Audit staff believes the panel's findings were less about shifts in NRC expectations than project team knowledge and supervisory oversight. The company's internal findings clearly identify poor management oversight and lack of the very specific type of needed expertise among its staff as the critical reasons for the deficient draft application. While audit staff agrees that significant resources are necessary to complete the LAR application and the company's extensive efforts post-expert panel to revise its application may have be necessary to develop a sound application from the onset, significant resources were spent prior to develop the final draft. These resources may not have been appropriately supported by the company to allow for a successful outcome. As a result, avoidable-work may have been performed as corrective action work by AREVA and the additional efforts by PEF staff.

Low Pressure Turbine Replacement

As part of the EPU project, PEF contracted with Siemens for two $18m^2$ low pressure turbines. Originally, the company included installation of these turbines as part of its Phase II work scope. However, in mid-2009 the company determined that it would shift the installation of the low pressure turbines from Phase II until Phase III of the project. At the time, the company was still evaluating the impact of a major turbine failure at the D.C. Cook Nuclear Plant, which involved similar Siemens $18m^2$ turbines. This 2008 event and resulting fire caused significant damage to that facility resulting in a costly repair and extended outage.

While PEF was monitoring the results of the D.C. Cook event, the company continued with the order of these turbines. [redacted] certain quality tests on this equipment. One quality assessment required the turbines to successfully operate at 120 percent of maximum output. The company refers to this as the "spin test." Siemens performed the spin test in April 2009, and the turbines did not pass this test. The turbines experienced disk slippage between the final blade components and the turbine's main shaft. After a detailed evaluation, [redacted] PEF informed Siemens that the turbines [redacted]

In addition to concerns from the spin-test failure, PEF states that the D.C. Cook incident created an unwillingness by the Nuclear Electric Insurance Limited (NEIL)—the group that

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insures nuclear plants against a variety of risks[41]—to insure any newly-installed Siemens $18m^2$ turbine for its first two in-service fuel cycles. It was determined that the cause of the D.C. Cook failure was the $18m^2$ model's L0 blades. According to PEF, [redacted]

The turbines are a critical component to maximizing the additional MWe output from the EPU efforts. The contracted Siemens model—$18m^2$—allows for the maximum capture of steam, resulting in the largest MWe output. While the company states it anticipates resolving the current turbine issues and installing the Siemens $18m^2$ model, management is evaluating several replacement options as a precaution. These options are shown in **EXHIBIT 15.**

PEF CR3 Low Pressure Turbine Replacement Options and the Resulting MWe Output		
Option	MWe output Added by EPU	Final Unit Output
Option 1: Continue Operating CR3 with its current Alstom Turbines	16 MWe	916 MWe
Option 2: Install the contracted Siemens $18m^2$ as originally designed during Phase III*	180 MWe	1080 MWe
Option 3: Install the contracted Siemens $18m^2$ without the L0** blades during Phase III*	100 MWe	1000MWe
Option 4: Install Siemens' smaller $13.9m^2$ turbines in 2013 (additional time is needed to manufacturing the equipment)	172 MWe	1072 MWe

** The $18m^2$ must pass the spin test prior to installation.*
*** The L0 blades were determined to be the cause of the D.C. Cook failure. According to PEF,* [redacted]

EXHIBIT 15 *PEF Response to Staff Data Request CR3 3-8*

In addition to the turbine options being considered by the company, PEF states it is in settlement negotiations with Siemens [redacted] The company states if its moves forward with the current $18m^2$ turbines, it will require [redacted]. PEF states it is optimistic that the negotiations will result in a positive outcome for the company and anticipates finalizing its turbine decision in mid-2010.

[41] www.nmlneil.com

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3.1.2 Impact on Schedule and Cost

While there is no direct correlation between the work for the EPU project and the events leading to the delamination of the CR3 containment vessel, the completion of the EPU project will be delayed as a result of the delamination repair work necessary to bring the unit back online. The timeline for completing the necessary repairs is in flux. Originally, the company anticipated the unit to be operational in mid-2010; however, after further evaluating the repair scope, the company shifted its estimate for start-up to third quarter 2010.

This will require a shift in the refueling schedule 17 (R17). The final phase of the EPU project is currently scheduled to occur during the R17 outage. As of May 2010, the company anticipates the R17 to shift from fall 2011 to spring 2012. However, if additional delays arise in the delamination repair schedule, the R17 schedule could shift further out in time.

The company states the cost implications for the shift in R17 will not significantly impact the EPU project. Currently, the company does not anticipate any additional direct costs to the project other than costs associated with any cost escalations over time. However, the company does not have an estimate of the cost impact at this time. The total shift in schedule is anticipated at six to twelve months from the original November 2011 timeline.

While the company anticipates minimal cost-impact resulting from this schedule shift, audit staff recommends the Commission monitor for any additional EPC costs associated with the Phase III work. This schedule shift is a direct result of the delamination issue at CR3, and PEF and the NRC are investigating the root cause of this incident. Depending on the outcome of this investigation, additional EPC project costs related to the shift may need to be excluded from the NCRC docket and addressed separately.

Low Pressure Turbine

The company is currently assessing the overall impact of the Low Pressure Turbine installation on the project. The unresolved issues surrounding Siemens 18m^2 turbines resulted in a shift in installation from Phase II to Phase III. Because of this shift, there may be additional costs associated with the delivery and installation of the turbines during Phase III of the EPU Additionally, the shift in installation required the company to adjust certain engineering designs for the Phase II work. This redesign required an additional work authorization with AREVA, totaling [redacted]

The company states it is currently negotiating a settlement with Siemens and anticipates that [redacted] However, until the settlement is finalized, it remains to be seen whether the anticipated settlement [redacted] Staff recommends that the Commission monitor the results of this process to ensure that the company only request recovery of the appropriate costs and excludes any resulting from a possible vendor error

In addition, if the company chooses not to move forward with its current Siemens low pressure turbine selection, there will be a decrease in the final MWe output for the project. If this occurs, an evaluation would be necessary to assess the appropriate handling of the reduction in planned versus achieved MWe output. In effect, the uprate would then have cost more per additional MWe added, and adjustments may be warranted.

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License Amendment Request

The company has shifted its LAR submittal timeline from 2009 to mid-to-late 2010. The company originally incorporated a float into its original schedule, and with the impact of the delamination repairs on the R17 outage, the company has gained additional float in its submittal window Audit staff does not believe the delays resulting from the company's restructuring and revising its LAR application will ultimately impact the EPU schedule. The company states that the Phase III work will continue as schedule, even if there is a delay in the LAR approval. If the company completes the work prior to approval, however, the unit will not be able to operate at the higher capacity prior to the NRC's issuance of an amended license.

The company increased its spending on the LAR preparations in 2009 and 2010. This was a result of the expert panel's assessment that the final draft would not meet the expectations of the NRC. The company estimated its 2009 License Application capital expenditures at [redacted] However, the company spent an additional [redacted] on this effort. This was attributed, in part, to the additional work necessary to strengthen its LAR after the Expert Panel review. Of these additional costs, AREVA was paid [redacted] to re-write and restructure previously drafted sections within the LAR application.[42]. Additionally, [redacted] was paid to finalize the engineering requirements.

The company anticipates that through 2010, it will spend an [redacted] to complete its LAR efforts. PEF estimates that at completion, the LAR application process will cost approximately [redacted] This represents a [redacted] over its original 2007 estimate of [redacted][43] The company states the application is ready to submit to the NRC, but it does not anticipate filing the application until fall 2010.

Overall Project Cost

The overall anticipated final cost of the EPU project has increased during the course of the project. The company originally anticipated the project to cost $426.6 million, while the most recent estimate is $479.4 million, a 12 percent increase.[44] The project team documented and updated these costs within its 2009 IPP, and received senior management's approval for the additional expenditures. The company states the increases in costs include additions and modification to the engineering specifications and increases in labor and support costs.

3.2 EPU Project Controls and Oversight

3.2.1 Project Controls, Risk and Management Oversight Changes

As discussed in the context of the Levy plant, the company requires an *Integrated Project Plan* (IPP) for each major project implemented by the company For both the Levy and the Crystal River 3 Uprate, the IPP establishes the financial requirements necessary to complete the project along with the project scope, deliverables, and risks associated with the project. Senior management uses this document to assess the overall feasibility of the project and to track the overall financial commitment for the project.

[42] PEF responses to Staff Data Request CR3 4.2, Bates-000001
[43] PEF Response to Staff Data Request CR3 4 2, Bates-000021
[44] PEF Response to Staff Data Request CR3 1.18.

Integrated Project Plan Revisions

The initial *Business Analysis Package* (BAP) for the uprate project was completed in November 2006. It outlined the project's phases and a cost estimate of approximately $427 million. This was comprised of a base $250 million uprate work estimate; plus $89 million for transmission upgrades, and $88 million for cooling tower upgrades. This cost estimate also included studies that would allow for development of the plant specific project plan including schedule and specifications. In the BAP, PEF used modeling to develop sensitivity analyses of assumptions and to quantify potential outcomes of the risks being assessed. These model runs led to outputs of base case, worst case, and best case scenarios for various combinations of assumptions. For each scenario, PEF developed cost/benefit ratios, break-even year projections, and net present value analyses.

During 2008, PEF began to migrate major projects towards its new IPP for approval and control. The IPP process still includes the identification and assessment of key risks and risk management approaches, but provides senior management with more frequent and continuing opportunities to endorse or redirect the project. Like the BAP, the IPP documents assumptions, constraints and decisions to be made, defines approval requirements for funding, and provides a baseline for the progress measurement and project control [45]

The original IPP for the Crystal River 3 Extended Power Uprate project was initiated in March 2008, updated in March 2009 (Rev 1), and further updated in October 2009 (Rev 2). The changes made in October 2009 reflect the scope change related to Phase III work. With this revision, EPU project management requested an additional $52.8 Million (Financial View) cost between 2009 and 2011 The additional funding will be used on the following major items:[46]

2009

- Moved LP Turbine scope to 2011 – ($15.5M)
- Reduced Cooling Tower Scope for Recirculation Line & Warehouse– ($9.0M)
- Cross/NGG Fleet Support Charges - $1.7M
- R16 Engineering Cost Increase - $6.7M
- R17 Engineering Cost Increase - $1.5M
- Augmented Staff Needed Earlier than Resource Shares Available--$2.5M
- Atlantic Implementation - $2.7M

2010

- R17 Purchase Major Component - $8.7M
- R17 Engineering - $12.0M
- Cooling Tower Recirculation Line - $8.0M
- Staff Augmentation and Support - $4.8M

2011

- LP Turbine Scope Moved from 2009 - $18.0M
- R17 Major Components - $5.0M

[45] FPSC's August 2008 *Review of PEF's Project Management Internal Controls for Nuclear Plant Uprate and Construction Projects*, page 10.
[46] PEF Response to Staff Data Request 1.8, BATES 000012 – 000014.

Project Management Augmented Labor · $2.6M
R17 Temporary Facilities · $3.0M

Project approval and updates will occur at critical milestone intervals. Planned updates are recommended as of this IPP revision. There is a request for review by the Progress Energy Senior Management Committee at each key milestone to allow prudent senior management evaluation of project progress and control.

According to PEF, in addition to SMC communications, project information is disseminated in both formal and informal methods. Stakeholder management through effective communication is vital. Formal information consists of written documents that are used for project studies, design documents, procurements, work assignments, project issues, status reports, schedules, presentations, meeting agendas, and other Informal communications are generally verbal but may be written.[47]

Project Management Policies and Procedures

PEF has procedures in place that direct the oversight and control of the Crystal River 3 Uprate project. The company created or updated these procedures as the project progressed and developed over time. Additionally, the company developed (and is continuing to refine) standard procedures for project management, through its *Project Management Center of Excellence*. These procedures cover areas including the evaluation and authorization process, project management, and organization/administration. A list of the seventeen new or recently revised Project Management procedures may be found in **APPENDIX E**[48]

Oversight and Management Policies and Procedures for Contractors

There have been no changes made to the Vendor Oversight Plan that was addressed during last year's review.[49] The company's revised oversight and management procedures for contractors working on the CR3 EPU project are shown in **EXHIBIT 16**, and are discussed below [50]

New or Revised CR3 Contractor Oversight and Management Procedures		
Procedure Title	Procedure Number	Procedure Revision Number (Date)
Standards & Expectations for the Acquisition and Training of Non-Station Personnel	AI-525	Rev 25 (Nov-2009)
Vendor Quality Program for Critical Non-Safety Equipment	NGGM-PM-0020	Rev 6 (Sep-2009)

EXHIBIT 16

Source: Data Request 1.27

[47] PEF Response to Staff Data Request 1.8, BATES 000027
[48] PEF Response to Staff Data Request 1.9, BATES 0000013 – 0000014
[49] PEF Response to Staff Data Request 1.30, BATES 0000043.
[50] PEF Response to Staff Data Request 1.27, BATES 0000039.

The *Standards and Expectations for the Acquisition and Training of Non-Station Personnel* (AI-525)[51] procedure establishes standards and expectations for control of Non-Station Personnel at Crystal River Unit 3 (CR3). It provides guidance for obtaining, training, and monitoring Vendor/Contractor, Shared Resources and interface organizations at CR3, including Supplemental Manpower

The *Vendor Quality Program for Critical Non-Safety Equipment* (NGGM-PM-0020) procedure is intended to proactively schedule and determine quality assurance measures to be implemented prior to sending high risk critical equipment out for repair or acquiring new high risk critical equipment or major purchases. This helps ensure critical equipment and major purchases are repaired/purchased in a timely manner to support improved equipment reliability A guide for performing vendor surveillance is included with this procedure. This procedure applies to both the Levy Nuclear Project and the Crystal River 3 Uprate.[52]

Controls Implemented in 2009 or Planned for 2010

PEF updated Nuclear Projects Guidance Documents, Financial Controls, Project Controls, and made enhancements to their organization as a result of quality assurance assessments, internal audits, and external audits. In addition, PEF implemented the following controls:

***Integrated Change Form*[53] (ICF) register**
In August 2009, the company implemented this register to aid in tracking the impact of approved ICFs against established project and annual budgets as a function of reduced contingency values. The register is reviewed at subsequent Project Review Group meetings.

Daily Earned Value (Schedule Performance Index based on scheduled man-hours) was implemented during R16. Weekly Earned Value Reports are distributed to the Task Managers.

***Project Management Center of Excellence* (PMCoE)**
The project began the transition to the corporate *Project Management Center of Excellence* (PMCoE) standards and procedures in January 2009. The project's risk assessment process has been integrated through implementation of the PMCoE.[54]

<u>Nuclear Projects Guidance Documents</u>

Several new Nuclear Projects Guidance Documents were created and/or revised as a result of quality assurance assessments, internal audits, and external audits. The documents created or revised are listed in **EXHIBIT 17**

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[51] PEF Response to Staff Data Request 1.27, BATES 000001 – 000045
[52] PEF Response to Staff Data Request 1 12A, BATES 000866 – 000904
[53] PEF Response to Staff Data Request 1 35, BATES 000128.
[54] PEF Response to Staff Data Request 1 17, BATES 0000025

Created or Revised Nuclear Projects Guidance Documents 2009	
Title	**Procedure Number**
Information and Process Management	NPGD-002
Staffing Management Plan	**NPGD-003**
Financial Controls Internal Invoice Audit Process	NGPD-004
Financial Group Invoice Processing	**NGPD-006**
Financial End of Month Activities	NGPD-007
Roll up Cost Management Report	**NGPD-008**
Nuclear Projects Cash Flow Projections True-up	NGPD-009
Nuclear Projects Month-End Journal Entries	**NGPD-010**
Project Budget Preparation	NGPD-011
Time Entry Guidelines	**NGPD-012**

EXHIBIT 17 *Source: Data Request 1.35*

Financial Controls

Changes to financial controls were made in 2009 as a result of quality assurance assessments, internal audits, and external audits. These changes were:

> **Fleet standard financial controls for Major Projects** were established that included monthly reporting on Month to Date/Year to Date capital and O&M costs. Roll up cost management reports were distributed to senior management on the tenth business day of each month and Guidelines were established to formally establish the reporting process.
>
> **Monthly cash flow projection process** was begun. This consisted of reporting the updated Year End projection for each major project through the monthly Roll-up Major Projects Cost Management Report.
>
> **Month end activity working guideline** was created and approved by project controls management.
>
> **Earned Value Analysis** improvements were established so that reports were on the same frequency during the outage. A tracking methodology was also established to meet the reporting deadlines, and efficiencies were created with the interface with Guidant Timesheet that match the master contractor time sheet format.

Project Controls

Changes to project controls were made in 2009 as a result of quality assurance assessments, internal audits, and external audits. These changes were:

> Improvements to Nuclear Project Guidance Document NPGD-002 for Integrated Change Forms. A flow chart was added to simplify understanding of the process.

Trigger points for the initiation of contingencies were added in the project schedules for items that were reviewed as high risk during the risk assessments.

Improvements were made in estimating scope changes, budgetary refinements and project costs. An estimator was hired, and the company purchased Timberline estimating software which was installed, tested, and is now in use.

Establishment of a consolidated list of tools, equipment and consumables. This provided better accountability of purchased tools used during the Extended Power Uprate during the R16 outage.

Organization

The company states that it made enhancements to its project organization to better define roles and responsibilities. Organization structures for EPU were established with a project control center and discipline direct reports for engineering, planning, scheduling, and CAP EPU management will have personnel in the station outage control center for communication and tracking of activities that affect station resources. The documents created or revised were:

Nuclear Engineering & Services 2009-2011 Business Plan

Extended Power Uprate (EPU) Project Engineering Change (EC) Quality; Field Implementation and Readiness, and Procurement Control/Vendor Oversight

Life Cycle Management 20 Year Cost Report

Quick Hit Self Assessment Earned Value Analysis

EPU/SGR Tool and Material Inventory Control

Risk Assessment and Mitigation

Project risk evaluations were conducted for the 2009 CR3 Uprate in accordance with the Progress Energy PMCoE (*Project Management Center of Excellence*) program. The procedure provides guidance on project risk management, including execution of the risk management process and reporting metrics.[55] Both the Levy Nuclear Project and the Crystal River 3 Uprate employ this procedure.[56] The standard probability and impact scales used by all Progress Energy Florida projects are provided in **APPENDIX F.**

Risk Matrix

The process of communicating and consulting to/with key stakeholders on the status of the project relative to risk is facilitated through the use of a *Post Response Strategy Risk Matrix*. The risk matrix is a visual tool for indicating what degree of management involvement the project team requires to address the risk.

[55] PEF Response to Staff Data Request 1 5, BATES 000006 – 000007
[56] PEF Response to Staff Data Request 1 12B, BATES 000264 – 000280.

In the Risk Assessment section of the October 2009 IPP, six risks were shown in the *Post Response Strategy Risk Matrix.* These risks are shown in EXHIBIT 18. Risk items that were documented in the matrix were dispositional using one of three methods.

Mitigation planning (Risk Matrix-RED area)[57]
Develop Contingency plans (Risk Matrix-YELLOW area)[58]
Accept the risk (Risk Matrix-GREEN area)[59]

PEF Risk Matrix

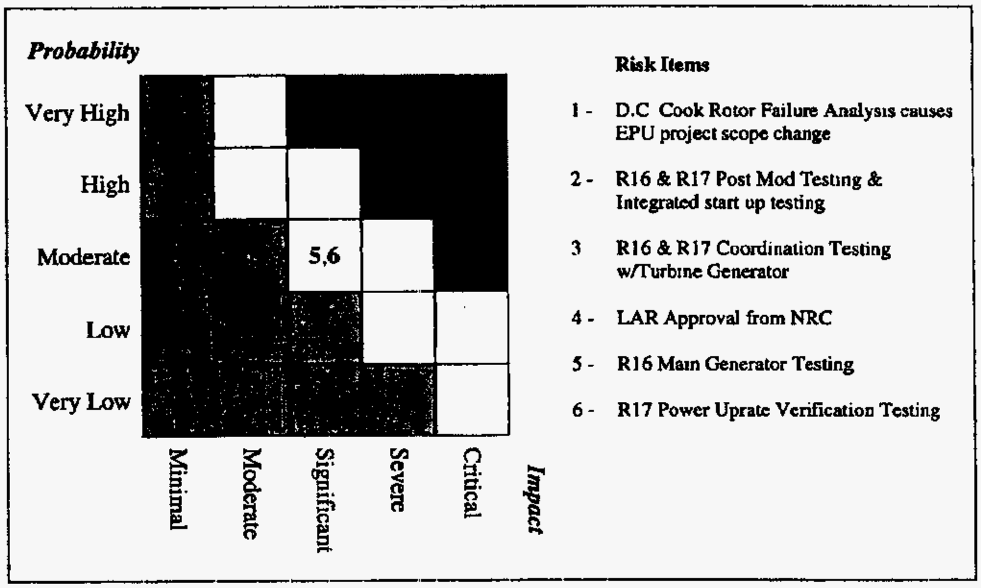

EXHIBIT 18 *Source: PEF Response to Data Request 1.8*

Risk Register

The December 2009[60] risk register identified twenty risks for the EPU project. Each risk contained a description, date of entry, date of last revision, risk status, response strategy (planned and/or action taken) and probability of occurrence. Of the twenty risks identified, seventeen had the status of "Closed [Risk Did Not Occur]", two had the status of "Open" and one involving the

[57] The Red area of the matrix shows items of "high risk" This designation indicates that the target is unachievable and that major disruption to the project is likely. This requires priority management attention to develop a different approach.
[58] The Yellow area of the matrix shows items of "moderate risk". This designation indicates that the achievement of the target is uncertain, and that some disruption to the project is likely. This may require additional management attention to consider a different approach.
[59] The Green area of the matrix shows items of "low risk" This designation indicates that achievement of the target is almost certain and that the item carries minimal impact to the project. Requires minimal management oversight to ensure that the risk remains low.
[60] PEF Response to Staff Data Request 1.5, BATES 000001 – 000002.

LP Turbines (R16 DC Cook Rotor Failure Analysis) had the status of "Triggered [Risk Occurred]".

The mitigation strategy that management deployed for the triggered risk was to defer the installation of the Low Pressure Turbines. Management stated that based on Industry Operating Experience associated with the $18m^2$ Low Pressure Turbine last stage blade failures at D.C. Cook in September of 2008 and the CR3 blade disc slippage during bunker spin testing in April 2009, the installation of the Low Pressure Turbine replacements at CR3 has been deferred. PEF required newly manufactured turbine parts after the spin test failure, which could not be delivered in 2009 PEF is currently negotiating with the turbine manufacturer and the insurance carrier regarding the Low Pressure Turbines, and evaluating its options, therefore, any impact on the project's cost and schedule is unknown at this point.

Changes to Management Oversight

EPU management created an implementation organization during the summer of 2009 The basis of the change was to ensure that there were task managers to oversee the field activities during R16 and to manage the engineering work associated with R17 The expected benefits of the changes are ensured personal accountability to meet the schedule and cost goals of the project.[61]

Senior/Executive managerial changes have occurred during 2009 The Director of Major Projects and supervisor of the EPU Project Manager left Progress Energy in March 2009. The Director of Major Projects reported to the Vice President – Nuclear Engineering and Support. This Vice President filled the vacant Director role until June 2009 A new Director of Nuclear Upgrades (formerly Major Projects) was hired in June 2009 This position manages four nuclear power facilities (Crystal River 3, Harris, Brunswick and Robinson). A new Project Manager for the Crystal River 3 EPU project was hired in early 2009.[62]

3.2.2 Internal Audits and Quality Assessments

Progress Energy's Audit Services Department (ASD) maintains an annual Construction Audit Strategy that solicits input from management, ranks potential audits based on risks, and establishes an annual audit plan. The 2009 CR3 Power Uprate audit focus areas used to rank audits based on risk are the same as those for the Levy Nuclear Plant discussed previously The three internal audits performed by Progress Energy's Audit Services Department for the EPU project during 2009 are shown in **EXHIBIT 19** FPSC Audit staff reviewed these audits and does not consider the findings to be of particular concern. In each case, the findings were satisfactorily resolved by PEF management.

[61] PEF Response to Staff Data Request 1.11, BATES 0000016.
[62] PEF Response to Staff Data Request 1 12, BATES 0000017

PEF Internal Audits Completed during 2009 for the EPU Project		
Audit Title	**Project Number**	**Opinion**
Florida Nuclear Plant Cost Recovery Rule Compliance	20013334 A916	Is effective
Crystal River Construction Logistics Support	20013334 A909	Is effective with many strengths noted
CR3 EPU and SGR Projects	20013334 A906 & A907	Needs Improvement

EXHIBIT 19 *Source: Data Request 1.31*

Planned 2010 Internal Audits

The internal audits planned for completion by Progress Energy's Audit Services Department in 2010 are shown in **EXHIBIT 20**.

PEF Internal Audits Planned for 2010 for the EPU Project		
Audit Title	**Project Number**	**Report Date**
Florida Plant Cost Recovery	20010800 A1016	TBD
Crystal River 3 Extended Power Uprate	20010800 A1003	TBD

EXHIBIT 20 *Source: Data Request 1.31*

Quality Assurance Reviews and Audits

The Crystal River 3 Nuclear Oversight (NOS) Department is charged with inspecting and monitoring the nuclear safety work performed at the Crystal River 3 unit. The quality assurance reviews and audits are accomplished through performance-based, real-time observations, technical reviews, and interviews with personnel. Findings, when identified, are based on best practices or minimum acceptable standards or requirements. Identification of a finding does not indicate unsatisfactory performance unless specifically stated.

The six quality assurance reviews and audits completed in 2009 are shown in **EXHIBIT 21.** FPSC Audit staff reviewed these audits and does not consider the findings to be of particular concern. In each case, the findings were satisfactorily resolved according to PEF There are currently no quality assurance reviews or audits planned for 2010

PEF Quality Assurance Reviews and Audits Performed for the EPU 2009		
Description	**Report Number**	**Completion Date**
Nuclear Projects Assessment	C-MP-09-01	July 22, 2009
Focused Review of EPU Project Modification Package Review	C-MP-FR-09-06	March 26, 2009
Focused Review of Nuclear Projects ALARA Work Plans	C-MP-FR-09-07	September 3, 2009
Focused Review of NP Material Acquisition and Contract Initiation and Administration	C-MP-FR-09-09	June 9, 2009
Focused Review of NP Document Control of Work Packages	C-MP-FR-09-10	August 12, 2009
Focused Review of NP Integrated Start-up and Test Activities	C-MP-FR-09-11	September 3, 2009

EXHIBIT 21 *Source: Data Request 1.32*

Quality Assurance - Contractors

The Quality Assurance group conducted several vendor oversight trips throughout 2009. All contracted manufacturing and Purchase Orders have a Vendor Oversight Plan. Vendor Oversight Plans are documented, and Vendor Oversight Checklists are created to record the acceptance or rejection of contractual acceptance criteria. Post Trip Reports are prepared by the Progress Energy employee with designated responsibility These reports capture observations and results, and any nonconformance and proposed resolutions. Non-Compliance Reports are written for identified deficiencies.

The Vendor Oversight Plan contains a Component Reliability Plan which includes the following applicable requirements:

Asset Management Policy
Zero Tolerance for Equipment Failure
Equipment Reliability Process Guideline
Vendor Quality Program for Critical Non-Safety Equipment

Vendor Oversight also involves Benchmarking and a Self-Assessment Plan.[63] Staff believes that the Vendor Oversight Plan is important, as evidenced by the disk slippage of the low pressure turbines during the "spin test". PEF's Quality Assurance group rejected the component because of the failure to meet contractual acceptance criteria.

[63] PEF Response to Staff Data Request 1 14, BATES 0000019

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3.3 EPU Contract Oversight and Management

PEF provided all RFPs issued and bid evaluations (both financial and technical) supporting the CR3 Uprate project contracts in excess of a $100,000 bid.[64] A listing of the 2009 EPU contracts is provided in **EXHIBIT 22.**

PEF Contracts Greater than $1 Million for the EPU Project as of December 31, 2009				
Company	Contract Number-Work Authorization	Description	Original Contract Amount ($000's)	Estimate of Final Value ($000's)
AREVA-NP	101659 WA 84	EPU NSSS Engineering, Fuel Engineering and LAR Support	[redacted]	[redacted]
Thermal Engineering International (TEI)	342253	Purchase of Four Moisture Separator Reheaters (MSRs)	[redacted]	[redacted]
AREVA-NP	101659 WA 93	EPU Balance of Plant and Turbine Bypass Valves	[redacted]	[redacted]
Siemens	145569 WA 50	CR3 Turbine Retrofit for EPU Including Supply of All Equipment and Installation	[redacted]	[redacted]
Yuba	355217	CR3 Feedwater Heater and SC Cooler Replacement	[redacted]	[redacted]
Barnhart Crane and Rigging Co.	384426	Heavy Hauling	[redacted]	[redacted]
MHF Logistical Solutions	47083-08	Radiation Waste Disposal	[redacted]	[redacted]
Mesa Associates, Inc.	221186-24	Civil Engineering POD Cooling Tower	[redacted]	[redacted]
Atlantic Group	3714 / 72&74	CR3 R16 EPU Implementation Labor and Support	[redacted]	[redacted]
Modspace	418171	Lease of Two-Story Trailer for EPU	[redacted]	[redacted]
Bartlett Nuclear	3707 / 43	EPU Portion of Health Physics / Decontamination for R16	[redacted]	[redacted]
Bettle Plastics	450189	Fiberglass Reinforced Piping for Helper Cooling Tower South	[redacted]	[redacted]
ITT	450795	Four Intake Pumps for HCTS	[redacted]	[redacted]
EvapTech	433059	CR3 Cooling Tower Construction	[redacted]	[redacted]

EXHIBIT 22 *Source: Exhibit WG-2, Schedule T-7, March 1, 2010 Testimony 100009-EI*

The AREVA contract, change order 23, increased the Work Authorization value by [redacted] on a time and materials basis for CR3 LAR re-write activities. [redacted]

[64] PEF Response to Staff Data Request 1.24, BATES 0000035.

REDACTED

[redacted] Change Order 25 increased the Work Authorization value by [redacted] on a time and materials basis for [redacted]

AREVA contract, change order 31, increased the Work Authorization value by [redacted] on a time and materials basis to support revisions to the design models due to the deferral of the LP turbine. This change order would not have been necessary if the [redacted] PEF is working with Siemens and NEIL to resolve the manufacturing issues, final costs and schedule.

Planned Contracts for 2010

Engineering design specifications of material are scheduled and are progressing for the remaining EPU work scope. After the engineering design specifications are issued, the procurement of material will begin. The company states it has used a competitive-bid RFP process for all its contracts and materials. The procurement of material is scheduled with end dates selected to support the pre-outage milestones established by outage and project management.

Long-lead items that have been identified to date[65] include:

Feed Water Booster Pump Motors
Condensate Pump Motors
Atmospheric Dump Valves
Safety Related Motor Operated Valves
Low Pressure and High Pressure Injection Components

The contracts planned for 2010 (R17)[66] are in their initial bid process. These contracts and their status are:

POD/HCTS Supporting Structures – vendor selection expected in early 2010
Booster Feed Pumps – RFP under development
Condensate Pumps – RFP under development
Atmospheric Dump Valves – RFP under development
Feed Pump / Main Impeller – specification under development
Main Feed Pump turbine re-rate – specification under development
Motor Operated Valves – specification under development
LPI Cross Tie – specification under development

As noted previously, PEF is continuing negotiations with Siemens and NEIL regarding the LP Turbine issue. Based on documentation reviewed by FPSC staff, the company appears to have followed its procurement procedures for initiating and implementing its EPU contracts. Staff recognizes that many remaining contracts for the EPU project will be initiated in 2010.

[65] PEF Response to Staff Data Request 1 19, BATES 0000028.
[66] PEF Response to Staff Data Request 1.21, BATES 0000030.

Contractor Selection and Management Policies and Procedures

PEF recently revised three procedures for the CR3 project which are shown in **APPENDIX G**. Plans, forms and checklists are incorporated throughout these procedures.

The *Corporate Contracting Process* (CNT-SUBS-00001) procedure provides instruction for the development, review, approval, issuance, revision and administration of contracts. This procedure provides the detailed requirements necessary for compliance with Progress Energy's *Procurement Policy* (MCP-SUBS-00002) regarding the contracting process.[67]

The *NGG Contract Initiation, Development and Administration* (MCP-NGGC-0001) procedure provides instruction for the initiation, development, and administration of contracts with the Nuclear Generation Group with certain exceptions. This procedure also pertains to the procurement for nuclear fuels, but does not replace appropriate corporate and plant fuel receipt/handling procedures.[68]

The *Contractor Safety* (SAF-SUBS-00041) procedure provides guidance for compliance with OSHA standards and Company Safety Policies.[69]

[67] PEF Response to Staff Data Request 1.26, BATES 000003.
[68] PEF Response to Staff Data Request 1.26, BATES 000059
[69] PEF Response to Staff Data Request 1.26, BATES 000050.

EXHIBIT JF-4
REDACTED IN ITS ENTIRETY

CR3 EPU Licensing Expenses Submitted on 2009 True-Up Filing T-6 Schedule 2009 REDACTED

Areva Non-Licensing Engineering Support	17,520,137.00		
Other Misc Non-Licensing Engineering Related Expenses Support	67,970.00		
PE Company Labor, Augmented Labor, Non-Licensing Expenses	445,525.00		
Subtotal of Engineering Related Expenses		*18,033,632.00*	*0.90*
Actual Expenses January-June 2009 (Pre-expert panel)			
Areva LAR Inputs (51-Doc)	[redacted]		
PE Company and Augmented Labor and Expenses Jan-Jun	21,798.00		
		[redacted]	
Actual Expenses July-December 2009 (Post-expert panel)			
Areva LAR Re-Write Activities CO-23[FN1.]	[redacted]		
PE Company and Augmented Labor and Expenses Jul-Dec	413,409.00		
Areva LAR Support Progress Payment	[redacted]		
Excel Services June-Dec			
NRC Charges	126,667.50		
Adjustment of 2008 Charges to Permitting	-106,077.00		
Subtotal of LAR Submittal Related Expenses		[redacted]	
Licensing Expenses on 2009 T-6 Schedule		**<u>20,016,838.88</u>**	**1.00**

CR3 EPU Project Management Expenses Submitted on 2009 True-Up Filing T-6 Schedule 2009

Power Block/Non-Power Block Engineering, Procurement, Etc., Permitting and Construction Facilities Related Project Management Expenses			18,298,585.00	
Licensing Application Related Project Management Expenses [FN1.]			2,855,571.00	
License Amendment Project 20072135 Related Project Management Expenses				
90% Related to Engineering Support	*2,572,649.80*	*0.90*		
10% Related to LAR Submittal	*282,921.20*	*0.10*		
Actual Project Management Expenses January-June 2009 (Pre-expert panel)	**117,236.42**	**0.04**		
Actual Project Management Expenses July-December 2009 (Post-expert panel)	**165,684.78**	**0.06**		
Project Management Expenses on 2009 T-6 Schedule			**21,154,156.00**	**1.00**

FN1. Areva invoices under LAR Re-Write Activities CO-23, in the amount of [redacted], were misallocated to Power Block Engineering in 2009. An adjustment will be made to correct this allocation in 2011. Based on the updated CO-23 spend in 2009 the project management percentage allocations indicated above would also be changed. Accordingly, for 2009, License Application Project Management Expenses were 2,855,571 and of that 85% or 2,419,589 was related to Engineering Support and 15% or 435,981 was related to LAR Submittal. Further, actual project management expenses January-June 2009 (pre-expert panel) were 110,261 and actual project management expenses July -December 2009 (post-expert panel) were 325,720.

EXHIBIT JF-6
REDACTED IN ITS ENTIRETY

INDEX OF CR3 UPRATE 2010 REVISED AND NEW PROJECT MANAGEMENT POLICIES AND PROCEDURES

Revised:

Procedure Number	Procedure Revision Number/Date	Procedure Title
PJM-SUBS-00005	Rev 1 (Aug 2010)	Project Cost & Financial Management
PJM-SUBS-00008	Rev 1 (Aug 2010)	Project Risk Management
PJM-SUBS-00011	Rev 1 (Jun 2010)	Project Environmental Health & Safety Management
PJM-SUBS-00016	Rev 1 (Apr 2010)	Project Manager Training Development Program
ADM-NGGC-0206	Rev 6 (Sep 2010)	Managing Fatigue and Working Hour Limits
NGGM-PM- 0018	Rev 6 (April 2010)	Project Management
NGGD-1000	Rev 3 (Aug 2010)	Fleet Alignment Manual
NGGM-IA-0047	Rev 1 (March 2010)	Interface Agreement
NGGM-PM-0007	Rev 19 (Sep 2010)	Quality Assurance Program Manual
NGGM-PM-0017	Rev 13 (Sep 2010)	Document Hierarchy
NGGM-PM-0020	Rev 7 (Aug 2010)	Vendor Quality Program for Critical Equipment & Major Purchase
NGGM-PM-0032	Rev 1 (Nov 2010)	Margin Management
PRO-NGGC-0200	Rev 14 (Nov 2010)	Procedure Use and Adherence
PRO-NGGC-0201	Rev 22 (Nov 2010)	NGG Procedure Writer's Guide
PRO-NGGC-0203	Rev 15 (Sept 2010)	Development and Approval of NGG Directives, Interface Agreements, and Program Manuals
PRO-NGGC-0204	Rev 19 (Sept 2010)	Procedure Review and Approval
CAP-NGGC-0200	Rev 33 (Aug 2010)	Condition Identification and Screening Process
CAP-NGGC-0201	Rev 14 (Aug 2010)	Self-Assessment/Benchmark Programs
CAP-NGGC-0202	Rev 18 (Dec 2010)	Operating Experience and Construction Experience Program
CAP-NGGC-0205	Rev 12 (Aug 2010)	Condition Evaluation and Corrective Action Process
CAP-NGGC-0206	Rev 5 (Jan 2010)	NGG Performance Assessment and Trending
CAP-NGGC-1000	Rev 3 (Nov 2010)	Conduct of Performance Improvement
EGR-NGGC-0011	Rev 16 (Dec 2010)	Engineering Rigor
EGR-NGGC-0021	Rev 3 (Aug 2010)	Common Conduct of Engineering Operations
EGR-NGGC-0157	Rev 5 (Dec 2010)	Engineering of Plant Digital Systems and Components
MCP-NGGC-0001	Rev 15 (Aug 2010)	NGG Contract Initiation, Development and Administration
MCP-NGGC-0002	Rev 18 (Aug 2010)	Purchasing of Materials for NGG

MCP-NGGC-0004	Rev 5 (Aug 2010)	Training of Contract Development Personnel
MCP-NGGC-0401	Rev 29 (Dec 2010)	Material Acquisition (Procurement, Receiving and Shipping)
NOS-NGGC-0200	Rev 2 (Oct 2010)	Supplier Qualification, Surveillance, and Audits/Surveys
NOS-NGGC-0600	Rev 2 (Jan 2010)	Nuclear Oversight Assessment, Vendor & Equipment Quality and Independent Review Personnel, Training and Development, Qualification, and Certification Program
NOS-NGGC-0601	Rev 2 (July 2010)	Certification of Quality Control Inspectors
NOS-NGGC-1000	Rev 10 (Jan 2011)	Nuclear Oversight Conduct of Operations
RDC-NGGC-0001	Rev 25 (Oct 2010)	NGG Standard Records Management Program
RDC-NGGC-0002	Rev 23 (Dec 2010)	Document Control Program
ACT-SUBS-00002	Rev 18 (June 2010)	Progress Energy Corporate Approval Level Policy
ACT-SUBS-00271	Rev 06 (Oct 2010)	Progress Energy Business Analysis Package
ADM-SUBS-00080	Rev 02 (Oct 2010)	Major Capital Projects – Integrated Project Plan
CNT-SUBS-00001	Rev 22 (Aug 2010)	Corporate Contracting Process
CNT-SUBS-00007	Rev 05 (Aug 2010)	Contract Compliance Program
MCP-SUBS-00010	Rev 15 (Jan 2010)	Corporate Purchasing Process
SAF-SUBS-00041	Rev 11 (Jan 2010)	Contractor Safety
ADM-NGGC-1001	Rev 01 (Oct 2010)	Conduct of NGG Human Resources
ADM-NGGC-0208	Rev 01 (Aug 2010)	NGG Fleet NGG Fleet New Employee Orientation Procedure
ADM-NGGC-0204	Rev 05 (Aug 2010)	Work Management (Scheduling)
ADM-NGGC-0203	Rev 16 (Aug 2010)	Preventative Maintenance and Surveillance Testing Administration
ADM-NGGC-0116	Rev 01 (Aug 2010)	Nuclear Planning
ADM-NGGC-0114	Rev 01 (Aug 2010)	Plant Health Process
ADM-NGGC-0113	Rev 02 (Oct 2010)	Performance Planning and Monitoring
ADM-NGGC-0112	Rev 04 (Aug 2010)	Reactor Coolant System Material Integrity Management Program
ADM-NGGC-0107	Rev 09 (July 2010)	Equipment Reliability Process Guideline
ADM-NGGC-0106	Rev 08 (Aug 2010)	Configuration Management Program Implementation
ADM-NGGC-0105	Rev 10 (Aug 2010)	ALARA Planning
ADM-NGGC-0104	Rev 39 (Dec 2010)	Work Implementation and Completion
ADM-NGGC-0102	Rev 06 (Aug 2010)	Long Range Planning (LRP) and Project Review Group (PRG)
ADM-NGGC-0101	Rev 21 (Aug 2010)	Maintenance Rule Program
ADM-NGGC-0003	Rev 08 (June 2010)	Conduct of Probabilistic Safety Assessment Unit Operations

New:

Procedure Number	Procedure Revision Number/Date	Procedure Title
PJM-SUBS-00002	Rev 0 (Jan 2010)	Project Integration Management
PJM-SUBS-00018	Rev 0 (Feb 2010)	Project Management Responsibilities for IT&T Scope
PJM-SUBS-00020	Rev 0 (Mar 2010)	Integrated Logistics Support Planning Project and Program Management Standard
PJM-SUBS-00030	Rev 0 (May 2010)	Achieving Excellence in Program Management
EPU-004	Rev 0 (Sep 2010)	Design Information Transmittal (DIT) Guidance
PJM-CDGX-00001	Rev 0 (Sept 2010)	Corporate Development & Improvement Group Contract Change Order Management
CON-NGPX-00001	Rev 0 (Aug 2010)	Execution of Large Construction Projects & Programs
CON-NGPX-00017	Rev 0 (July 2010)	Contracting and Supplier Strategy
ADM-CDGX-00001	Rev 0 (June 2010)	Corporate Development Group Human Performance Procedure