REDACTED

BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION ORIGINAL

In re: PEF's Petition for Determination of Need for Expansion of an Electrical Power Plant, for Exemption from Rule 25-22.082, F.A.C., and for Cost Recovery through the Fuel Clause	)	**Docket No.: 070052** -EI

PROGRESS ENERGY FLORIDA'S FIRST REQUEST FOR CONFIDENTIAL CLASSIFICATION

APPENDIX B

DOCUMENT NUMBER-DATE
04613 JUN-7 07
FPSC-COMMISSION CLERK

PEF DOCUMENTS RESPONSIVE TO CITIZENS' FIRST REQUEST TO PRODUCE DOCUMENTS TO PROGRESS ENERGY FLORIDA, INC. (No. 2)

Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10CFR50.90

April 25, 2007
3F0407-10

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – License Amendment Request #296, Revision 0, Measurement Uncertainty Recapture Uprate

References:

1. Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

2. Amendment No. 205 to the Crystal River Unit 3 Facility Operating License (TAC NO. MB5289), dated December 4, 2002

Dear Sir:

Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., hereby submits License Amendment Request (LAR) #296, Revision 0. The proposed amendment will revise the maximum power limit in the Crystal River, Unit 3 (CR-3) facility operating license, DPR-72, from 2568 MWt to 2609 MWt. The proposed uprate is characterized as a Measurement Uncertainty Recapture (MUR) using a Caldon Leading Edge Flowmeter (LEFM) CheckPlus™ system and other improvements to support an approximate 1.6 percent power increase.

Changes are requested to two definitions in Section 1.1 of the CR-3 Improved Technical Specifications (ITS), Effective Full Power Days and Rated Thermal Power, Section 5.6.2.20, Containment Leakage Rate Testing Program, and License Condition 2.C.(1). The power uprate will permit more economical operation of CR-3 and will not have a significant impact on the environment or the health and safety of the general public.

Regulatory Issue Summary (RIS) 2002-03, Reference 1, provided guidance for submittals requesting power uprates involving feedwater flow measurement uncertainty recapture. The RIS stated that licensee's applications which follow this guidance will require less review time and could be approved in six months or less. The guidance of RIS 2002-03 was used to ensure all areas of concern to the NRC staff were addressed in this submittal. Therefore, FPC requests approval of this request by November 15, 2007 with a 60 day implementation period in order to support implementation following Refuel Outage 15 which is scheduled for Fall 2007.

Amendment 205 to the CR-3 Operating License, Reference 2, implemented a previous increase in power from 2544 to 2568 MWt. At that time, evaluations were performed at 102% of 2568 MWt to demonstrate compliance with applicable regulations and the capability of systems and components to operate safely.

Feedwater instrumentation will be supplemented with an ultrasonic flowmeter system that has significantly decreased measurement uncertainty. Existing feedwater flow instrumentation will continue to be used for protective and control functions. The majority of existing CR-3 accident analyses were performed at 2619 MWt or higher, therefore, few analytical changes are needed to support this power uprate.

The proposed amendment has been reviewed and recommended for approval by the Plant Nuclear Safety Committee.

Regulatory commitments are identified in Attachment H.

If you have any questions regarding this submittal, please contact Mr. Paul Infanger, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/par

Attachments:
- A. Description of the Proposed Change, Background, Technical Analysis, Determination of No Significant Hazards Considerations, and the Environmental Assessment
- B. Proposed Operating License and Technical Specification Pages – Strikeout and Shadowed Text Format
- C. Proposed Operating License and Technical Specification Pages - Revision Bar Format.
- D. Response to RIS 2002-03 Questions
- E. Uncertainty Calculation
- F. LOCA Mass and Energy Releases
- G. Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) Arming Setpoint Evaluation
- H. List of Regulatory Commitments

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector
State Contact

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 25 day of April, 2007, by Dale E. Young.

Lisa Gail Crain

Signature of Notary Public
State of Florida

Notary Public State of Florida
Lisa Gail Crain
My Commission DD390922
Expires 01/30/2009

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT A

DESCRIPTION OF THE PROPOSED CHANGE,

BACKGROUND, TECHNICAL ANALYSIS,

DETERMINATION OF NO SIGNIFICANT HAZARDS

CONSIDERATIONS, AND THE

ENVIRONMENTAL ASSESSMENT

**DESCRIPTION OF THE PROPOSED CHANGE
BACKGROUND, TECHNICAL ANALYSIS, NO SIGNIFICANT HAZARDS
CONSIDERATION AND THE ENVIRONMENTAL ASSESSMENT**

1.0 DESCRIPTION OF PROPOSED CHANGE:

The proposed change would revise the Crystal River, Unit 3 (CR-3) Operating License to read as follows:

> 2.C.(1) Maximum Power Level
>
> Florida Power Corporation is authorized to operate the facility at a steady state reactor core power level not in excess of **2609** Megawatts (100 percent of rated core power level).

The proposed change would revise ITS definitions to read as follows:

> EFFECTIVE FULL POWER DAY (EFPD) – EFPD shall be the ratio of the number of hours of production of a given THERMAL POWER to 24 hours, multiplied by the ratio of the given THERMAL POWER to the RTP. One EFPD is equivalent to the thermal energy produced by operating the reactor core at RTP for one full day. (One EFPD is **2609** MWt times 24 hours or **62616** MWhr.)
>
> RATED THERMAL POWER (RTP) – RTP shall be a total reactor core heat transfer rate to the reactor coolant of **2609** MWt.

The proposed change would also revise ITS Section 5.6.2.20, Containment Leak Rate Testing Program to read as follows:

> The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is **54.04** psig. The containment design pressure is 55 psig.

2.0 BACKGROUND

CR-3 was initially licensed to operate at a maximum of 2452 Megawatts-Thermal (MWt). In Amendment 41, dated July 21, 1981, the NRC approved operation of CR-3 up to 2544 MWt. By letter dated June 5, 2002 (3F0602-05), CR-3 requested an increase in maximum Rated Thermal Power (RTP) to 2568 MWt. At that time, several transient and accident analyses (moderator dilution accident, letdown line failure, loss of feedwater event and small break loss-of-coolant accident) were reevaluated at 102% of 2568 MWt. The request was approved on December 4, 2002 as Amendment 205 to the CR-3 facility Operating License. Since the evaluations in 2002 were performed, few changes to plant systems, structures or components (SSC) have occurred. Two analyses, the Main Steam Line Break (MSLB) and High Energy Line Break (HELB) had previously been evaluated at nominal full power (2568 MWt) without considering heat balance uncertainty. These analyses were re-performed at 102% of 2568 MWt in order to support the Measurement Uncertainty Recapture (MUR) uprate. Other analyses had already been performed at 102% of 2568 MWt or higher.

CR-3 has determined that implementation of the Caldon Leading Edge Flowmeter CheckPlus™ System and other changes are an effective way to obtain additional power from the plant without significantly changing current reactor core operations. This will permit the recovery of approximately 41 MWt by using more accurate instrumentation to calculate core power production. A spool piece will be installed in each of the two feedwater pipes, containing 16 ultrasonic, multi-path, transit time transducers. The CheckPlus™ system provides more accurate feedwater flow and temperature. New feedwater pressure and main steam temperature and pressure instrumentation will also be installed. The currently installed flow instruments will continue to provide inputs to other indication and control systems.

3.0 TECHNICAL ANALYSIS

The licensed rated thermal power level for CR-3 is proposed to be increased from 2568 MWt to 2609 MWt. The uprate evaluation addressed the following categories: Nuclear Steam Supply System (NSSS) performance parameters, design transients, systems, components, accidents, and nuclear fuel, as well as interfaces between the NSSS and balance-of-plant (BOP) systems. Most of these evaluations are contained in Attachment D, Response to RIS 2002-03 Questions, which is based largely on Areva Technical Report 51-9042409-000. No new analytical techniques, with the exception of the use of the GOTHIC code for containment response, were used to support the power uprate project. The methodology includes the use of well-defined analysis input assumptions/parameter values and currently approved analytical techniques, and takes into consideration applicable licensing criteria and standards, including Regulatory Issue Summary 2002-03.

All current analyses were performed considering a maximum power output of 2568 MWt or higher. 10 CFR 50, Appendix K, analyses were done at 2619 MWt (102 percent of 2568 MWt) to account for the two percent uncertainty previously assumed in power measurement. This continues to be valid based on 100.4% of 2609MWt. Some analyses were performed at higher power levels (generally 2772 MWt) because they were performed generically to bound all Babcock and Wilcox plants. All of these analyses were reviewed to provide assurances that they remain the bounding or limiting analyses. Some analyses were revised using more current methods. These analyses have been either approved by the NRC or were performed using methods or processes that were approved by the NRC. Most of the systems, structures, and components (SSC) evaluated for the previous uprate, and discussed in the Safety Evaluation for Amendment 205, are virtually unchanged from the SSC that currently exist in the plant.

This section discusses the revised NSSS design thermal and hydraulic parameters that changed as a result of the power uprate and that serve as the basis for all of the NSSS analyses. A detailed assessment of the accident analyses performed for the steam generator tube rupture, Loss of Coolant Accident (LOCA), and non-LOCA areas was performed. The Main Steam Line Break and Loss of Coolant Mass and Energy releases and resultant containment response analyses were rerun. The radiological consequence evaluation is bounded by the current analysis since the radiological source term had not increased, due to the analytical limit of 2619.4 MWt not changing. The fuel was also evaluated for its ability to perform at the uprated power level. CR-3 concludes that the changes to the plant design basis and transient analyses are acceptable. Each of the NSSS systems and components were evaluated for the uprated conditions. The effects of the uprate on the BOP (secondary) systems, electrical power systems, control systems and instrumentation systems were also evaluated. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met and that the plant requires minimal additional design changes other than setpoint adjustments to safely operate at the uprated conditions. A summary of these evaluations and assessments follows.

3.1 Nuclear Steam Supply System (NSSS) Design Parameter

The NSSS parameters are the fundamental parameters which are used as input in all the NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, flows) that are used as the basis for the design transient, system, component and accident evaluations. The parameters for design are established using conservative assumptions in order to provide bounding conditions to be used in the NSSS analyses.

3.1.1 Input Parameters and Assumptions

The total thermal power for the uprate analysis was set at 2609 MWt (core power). This is approximately 1.6 percent higher than the current core thermal power rating of 2568 MWt. Feedwater and main steam pressure, feedwater and main steam temperature, feedwater flow instrument improvements and RCS cold leg temperature (T_{cold}), letdown flow, temperature, and make-up water temperature–uncertainty calculations were required to support this power uprate.

3.1.2 Discussion of Parametric Cases

Table 1 provides the NSSS parameter cases which were generated and used as the basis for the uprate project. Uprated conditions were calculated at current and 20 percent once-through steam generator (OTSG) tube plugging (the CR-3 licensed tube plugging limit) to bound the range of RCS temperatures and steam conditions (flow rate and temperature) which could occur as part of the uprate. This Table provides the values used in the RCS functional specification as well as the calculated uprated conditions at current and 20 percent OTSG tube plugging. The parameters listed in Table 1 have been reviewed against the RCS functional specification. The RCS functional specification bounds the uprated conditions. For reactor coolant flow, the original functional specification design flow was not used for flow-induced vibration analysis. As discussed in BAW-10051, Revision 1, "Design of Reactor Internals and Incore Nozzles for Flow-Induced Vibrations," conservative flow velocities were used. The very slight change in mass flow remains within the velocity used and thus has negligible impact on the components.

3.1.3 Conclusions

Changes to plant operating conditions were determined for the 1.6 percent power uprate (values for current and 20 percent plugging are listed in Table 1). The new operating conditions were compared with original design conditions for the RCS. The power uprate will not result in operation outside the original design conditions. The change in operating conditions and increased power was used to evaluate systems, components, materials, fuel and safety analysis. It has been concluded that the cumulative effect of the evaluations for all systems, components and analyses support the power uprate.

Table 1
Change in Operating Conditions For MUR

Parameter	Case A	Case B	Case C
Core Thermal Power (MWt)	2568	2609	2609
Other RCS Power MWt (RCP-LD) – (QRCS)	16	16	16
Total RCS Power (MWt)	2584	2625	2625
Pressurizer Control Pressure (psig)	2155	2155	2155
SG A/B Tube Plugging (%)	2.4/5.7	2.4/5.7	20/20
Thot (°F)	601.9	602.2	602.9
Tcold (°F)	556.2	555.8	555.1
Tavg (°F)	579	579	579
RCS Mass Flow (E6 lb_m/hr)	144.05	144.08	139.81
RCS Volumetric Flow (gpm)	386,873	386,729	374,896
Steam Temperature (°F)	591.0	590.5	580.7
Steam Superheat (°F)	54.9	54.4	44.6
Feedwater/Steam Flow Rate (E6 lb_m/hr)	10.86	11.07	11.19
OTSG Steam Pressure (psia) (input)	931.7	931.7	931.7
Feedwater Temperature (°F) (input)	456.7	458.4	458.5

Case A – Existing tube plugging at 2568 MWt

Case B – Existing tube plugging at 2609 MWt

Case C – 20 percent tube plugging at 2609 MWt

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power Corporation (FPC) has evaluated the proposed License Amendment Request (LAR) against the criteria of 10 CFR 50.92 to determine if any significant hazards consideration is involved. FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92 (c) criteria is satisfied.

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will increase the maximum core power level from 2568 MWt to 2609 MWt. This increase will only require adjustments and calibrations of existing plant instrumentation and control systems. The only equipment upgrades necessary for this uprate are spool pieces containing multiple ultrasonic flow instruments, which will be installed in each feedwater line, as well as more accurate instrumentation for feedwater pressure and steam pressure and temperature. Indication and control functions will continue to be performed by the currently installed feedwater instrumentation.

Nuclear steam supply systems (NSSS) and balance-of-plant (BOP) systems and components that could be affected by the proposed change have been evaluated using revised NSSS design parameters based on a core power level of 2609 MWt. The results of these evaluations, which used well-defined analysis input assumptions/parameter values and currently approved analytical techniques, indicate that CR-3 systems and components will continue to function within their design parameters and remain capable of performing their required safety functions at 2609 MWt. Since the revised NSSS parameters remain within the design conditions of the Reactor Coolant System (RCS) functional specification, the proposed change will not result in any new design transients or adversely affect the current CR-3 design transient analyses.

The accidents analyzed in Chapter 14 of the CR-3 Final Safety Analysis Report (FSAR) have been reviewed for the impact of the uprate. Based on the power levels assumed in the current safety analyses, it has been determined that all FSAR and supporting analyses bound the uprate. This includes the dose calculations for the design basis radiological accidents, which assume a power level of 2619 MWt (2568 MWt plus an assumed 2 percent measurement uncertainty). Since the proposed change relies on less than 0.4 % uncertainty, the assumed power level of 100.4% of 2609 MWt remains 2619 MWt. Therefore, analyses performed at this power remain bounding.

(2) *Does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

As discussed above, the only equipment upgrades necessary for this uprate are spool pieces containing multiple ultrasonic flow instruments, which will be installed in each feedwater line, as well as more accurate instrumentation for feedwater pressure and steam pressure and temperature. All CR-3 systems and components will continue to function within their design parameters and remain capable of performing their required safety functions. The proposed change does not impact current CR-3 design transients or introduce any new transients. Equipment failure modes are expected to be the same as for existing instruments. Protective and control functions will continue to be performed by the currently installed feedwater instrumentation. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does not involve a significant reduction in a margin of safety

Challenges to the fuel, RCS pressure boundary and containment were evaluated for uprate conditions. Core analyses show that the implementation of the power uprate will continue to meet the current nuclear design basis. Impacts to components associated with RCS pressure boundary structural integrity, and factors such as pressure/temperature limits, vessel fluence, and pressurized thermal shock (PTS) were determined to be bounded by current analyses.

As discussed above, all systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following implementation of the proposed change. Finally, the current CR-3 safety analyses, including the design basis radiological accident dose calculations, bound the uprate. Therefore, this change does not involve a significant reduction in the margin of safety.

5.0 ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if the amendment changes a requirement with respect to use of a facility component within the restricted area provided that (i) the amendment involves no significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this License Amendment Request (LAR) and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is that this amendment does not significantly change feedwater flow instrumentation located inside the restricted area and:

(i) The proposed license amendment does not involve a significant hazards consideration, as described in the significant hazards evaluation.

(ii) As discussed in the Justification for the Request and the No Significant Hazards Evaluation, this change does not result in a significant change or significant increase in the release associated with any Design Basis Accident. The bounding accident involved, the Loss of Coolant Accident, has release rates not significantly affected by the increase in core power. Likewise, there will be no significant change in the types or a significant increase in the amounts of any effluents released offsite during normal operation. The specific activity of the primary and secondary coolant is expected to increase by no more than the percentage increase in power level. Therefore, the amount and specific activity of solid waste is not expected to increase significantly.

Gaseous and liquid effluent releases are expected to increase from current values by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be maintained within the limits of 10 CFR 20 and 10 CFR 50, Appendix I, in accordance with the requirements of the CR-3 Offsite Dose Calculation Manual (ODCM). The ODCM contains methodologies and parameters used in calculation of offsite doses resulting from radioactive gaseous and liquid effluents, the methodologies and parameters used in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and controls for maintaining doses to the public from radioactive effluents As Low As Reasonably Achievable (ALARA) in accordance with 10 CFR 50.36(a). The proposed changes will not result in changes in the operation or design of the gaseous, liquid, or solid waste systems and will not create any new or different radiological release pathways.

Therefore, the proposed LAR will not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite.

(iii) The proposed License amendment does not significantly increase core power and resultant dose rates in the Reactor Building and accessible areas of the plant. Individual worker exposures will be maintained within acceptable limits by the CR-3 ALARA Program. Therefore, the proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT B

Proposed Operating License and Technical Specification Pages

Strikeout and Shadowed Text Format

1.1 Definitions

EFFECTIVE FULL POWER DAY (EFPD) (continued)	reactor core at RTP for one full day. (One EFPD is ~~2568~~ 2609 MWt times 24 hours or ~~61632~~ 62616 MWhr.)
EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME	The EFIC RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its EFIC actuation setpoint at the channel sensor until the emergency feedwater equipment is capable of performing its safety function (i.e., valves travel to their required positions, pump discharge pressures reach their required values, etc.) Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing, that is captured and conducted to collection systems or a sump or collecting tank; or

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and quantified and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; or

(continued)

1.1 Definitions

PHYSICS TESTS (continued)	These tests are: a. Described in Chapter 13, "Initial Tests and Operation" of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.2.19. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature Limits."
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100 \left(\frac{\text{Power In Any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1 \right)$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2568~~ 2609 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or

(continued)

PEF-CR3-0096

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

c. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

d. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, as modified by the following exception:

1. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2006.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is ~~54.2~~ 54.04 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C Tests and $\leq$ 0.75 L_a for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq$ 0.05 L_a when tested at $\geq P_a$.

   b. For each door, leakage rate is $\leq$ 0.01 L_a when tested at $\geq$ 8.0 psig.

   The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

   The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

(7) Florida Power Corporation, pursuant to the Act and 10 CFR Parts 30 and 70, to receive and possess, but not separate, that by-product and special nuclear materials associated with four (4) fuel assemblies (B&W Identification Numbers 1A-01, 04, 05, and 36 which were previously irradiated in the Oconee Nuclear Station, Unit No. 1) acquired by Florida Power Corporation from Duke Power Company for use as reactor fuel in the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54, and 50.59 or Part 50, Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

2.C.(1) Maximum Power Level

Florida Power Corporation is authorized to operate the facility at a steady state reactor core power level not in excess of ~~2568~~ 2609 Megawatts (100 percent of rated core power level).

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ~~222~~, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

The Surveillance Requirements contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment 149. The Surveillance Requirements shall be successfully demonstrated prior to the time and condition specified below for each.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT C

Proposed Operating License and Technical Specification Pages

Revision Bar Format

1.1 Definitions

EFFECTIVE FULL POWER DAY (EFPD) (continued)	reactor core at RTP for one full day. (One EFPD is 2609 MWt times 24 hours or 62616 MWhr.)
EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME	The EFIC RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its EFIC actuation setpoint at the channel sensor until the emergency feedwater equipment is capable of performing its safety function (i.e., valves travel to their required positions, pump discharge pressures reach their required values, etc.) Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing, that is captured and conducted to collection systems or a sump or collecting tank; or
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and quantified and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; or

(continued)

1.1 Definitions

PHYSICS TESTS (continued)	These tests are: a. Described in Chapter 13, "Initial Tests and Operation" of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.2.19. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature Limits."
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100 \left(\frac{\text{Power In Any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1 \right)$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2609 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.19 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

c. The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met.

d. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period.

5.6.2.20 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, as modified by the following exception:

1. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the November 7, 1991 Type A test shall be performed no later than November 6, 2006.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 54.04 psig. The containment design pressure is 55 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C Tests and $\leq$ 0.75 L_a for Type A Tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage range is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

   b. For each door, leakage rate is $\leq$ 0.01 L_a when tested at $\geq$ 8.0 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

(7) Florida Power Corporation, pursuant to the Act and 10 CFR Parts 30 and 70, to receive and possess, but not separate, that by-product and special nuclear materials associated with four (4) fuel assemblies (B&W Identification Numbers 1A-01, 04, 05, and 36 which were previously irradiated in the Oconee Nuclear Station, Unit No. 1) acquired by Florida Power Corporation from Duke Power Company for use as reactor fuel in the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54, and 50.59 or Part 50, Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

2.C.(1) Maximum Power Level

Florida Power Corporation is authorized to operate the facility at a steady state reactor core power level not in excess of 2609 Megawatts (100 percent of rated core power level).

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

The Surveillance Requirements contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment 149. The Surveillance Requirements shall be successfully demonstrated prior to the time and condition specified below for each.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT D

Response to RIS 2002-03 Questions

Response to RIS 2002-03 Questions

1.0 Feedwater Flow Measurement Technique and Power Measurement Uncertainty (RIS 2002-03 Section I Questions)

1.1 A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. This description should include:

A. Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

B. A reference to the NRC's approval of the proposed feedwater flow measurement technique

C. A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique

D. The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique

E. A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty

F. Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:
   i. maintaining calibration
   ii. controlling software and hardware configuration
   iii. performing corrective actions
   iv. reporting deficiencies to the manufacturer
   v. receiving and addressing manufacturer deficiency reports

G. A proposed allowed outage time for the instrument, along with the technical basis for the time selected

H. Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level.

1.2 Response to RIS 2002-03 Section I Questions

Detailed description of the Crystal River Unit 3 (CR-3) Implementation of the Caldon LEFM CheckPlus™ Instrumentation and the 1.6% power increase.

The feedwater flow measurement system installed at CR-3 is an LEFM CheckPlus™ ultrasonic, multi-path, transit time flowmeter. This equipment also provides a highly accurate feedwater temperature that will be input to the heat balance. The design of this advanced flow measurement system is addressed in detail by the manufacturer, Caldon, Inc., in Topical Reports ER-80P, Revision 0 (Reference 1.3.1), and ER-157P, Revision 5 (Reference 1.3.2). The current flow instruments will continue to measure main feedwater flow as well.

The LEFM ultrasonic flowmeter system consists of an electronic cabinet located in the Intermediate Building, Elevation 119 feet and two measurement section/spool pieces, also located in the Intermediate Building, in each of the two 18 inch main feedwater flow headers that feed each steam generator. The measurement sections are located between the existing feedwater flow nozzles and their respective upstream straightening vanes. The LEFM flowmeters will be calibrated at the Alden Research Laboratory, Inc. facility using the plant's current piping configuration and variations of the plant's configuration.

Each measurement section consists of sixteen (16) ultrasonic transducer housings, forming the pressure boundary. Each transducer may be removed at full-power conditions without disturbing the pressure boundary. The installation location of these flow elements conforms to the requirements in Topical Reports ER-80P and ER-157P (Reference 1.3.3 and 1.3.4).

The LEFM system measures the transit times of pulses of ultrasonic energy traveling along chordal acoustic paths through the flowing fluid. This technology provides significantly higher accuracy and reliability than the existing flow instruments, which use differential pressure measurements; and temperature instruments, which use conventional thermocouple or resistance thermometers. The sound will travel faster when the pulse traverses the pipe with the flow and will travel slower when the pulse traverses the pipe against the flow, due to the Doppler effect. The LEFM uses these transit times and time differences between pulses to determine the fluid velocity. The LEFM also measures the speed of sound in water and uses this to determine the feedwater temperature.

The system's software employs the ultrasonic transit time method to measure the velocities at precise locations with respect to the pipe centerline. The system numerically integrates the measured velocities. The system's software has been developed and maintained under a verification and validation (V&V) program. The V&V Program has been applied to all system software and hardware, and includes a detailed code review. The mass flow rate and feedwater temperature are displayed on the electronic cabinet and transmitted via Ethernet to the automated unit load demand (AULD) and plant process computer (CP) for use in the calorimetric measurement of reactor thermal output based on an energy balance of the secondary system.

The improved accuracy of measurements of feedwater mass flow, pressure, and temperature as well as main steam temperature and pressure and updated instrument uncertainty calculations for other parameters results in a total uncertainty of less than 0.4 percent of reactor thermal power. This is substantially more accurate than the nominal 2 percent rated thermal power (RTP) assumed in the accident analyses.

The LEFM indications of feedwater mass flow and temperature will be directly substituted for the existing feedwater flow instrumentation and the resistance temperature detector (RTD) temperature inputs currently used in the plant calorimetric measurement calculations. The existing feedwater flow and RTD temperature will continue to be used for feedwater control and other functions that they currently fulfill.

The Caldon Panel has outputs for internally generated system trouble alarms, which will be wired into the plant process computer.

The AULD and the plant process computer (in the fixed incore detector monitoring system (FIDMS), each perform independent plant secondary heat balance calculations. The AULD heat balance is used in conjunction with the integrated control system to automatically control plant power at the operator selected Core Thermal Power (CTP) in megawatts thermal (MWt). The FIDMS heat balance is normally used by the plant operators to calibrate the nuclear instrumentation (NIs) and can be used by the plant operators to manually control reactor power upon loss of AULD. These two software routines are independent but receive identical inputs.

1.2.1 Caldon Topical Reports Applicable to the LEFM CheckPlus™ System (RIS 2002-03 Section I.1.A)

The referenced Topical Reports are:

ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, dated March 1997 (Reference 1.3.1)

ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System," Revision 5, dated October 2001 (Reference 1.3.2)

1.2.2 NRC Approval of Caldon LEFM CheckPlus™ System Topical Reports (RIS 2002-03 Section I.1.B)

The NRC approved the subject Topical Reports referenced above on the following dates:

ER-80P, NRC SER dated March 8, 1999 (Reference 1.3.3)

ER-157P, NRC SER dated December 20, 2001 (Reference 1.3.4)

In addition, the NRC performed additional evaluations on the acceptability of the Caldon LEFMs and these are documented in “Safety Evaluation by the Office of Nuclear Reactor Regulation The Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters Caldon, Inc.,” Project No. 1311, July 5, 2006 (TAC NO. MC6424) (Reference 1.3.11).

1.2.3 CR-3 Implementation of Guidelines and NRC SER for the Caldon LEFM CheckPlus™ System (RIS 2002-03 Section I.1.C)

The LEFM CheckPlus™ system will be installed at CR-3 in accordance with the requirements of Topical Reports ER-80P and ER-157P. This system will be used for continuous calorimetric power determination by serial link with the CP and incorporates self-verification features to ensure that hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

The CR-3 LEFM CheckPlus™ system will be calibrated in a site-specific model test at Alden Research Laboratories, with traceability to National Standards. The LEFM CheckPlus™ system will be installed and commissioned according to Caldon procedures, which include verification of ultrasonic signal quality and hydraulic velocity profiles as compared to those tested during site-specific model testing.

1.2.4 Disposition of NRC Criteria in the SER during Installation (RIS 2002-03 Section I.1.D)

In approving Caldon Topical Reports ER-80P and ER-157P, the NRC established four criteria to be addressed by each licensee. The four criteria and a discussion of how each will be satisfied for CR-3 follow:

Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for unavailable LEFM instrumentation and the effect on thermal power measurements and plant operation.

Response to Criterion 1

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM system. A preventative maintenance program will be developed for the LEFM using the vendor's maintenance and troubleshooting manual. The preventative maintenance activities perform the following checks:

- General inspection of the terminal and cleanliness
- Power Supply inspection of magnitude and noise
- Central Processing Unit inspection
- Acoustic Processor Unit Checks of the 5 MHz clock and LED status
- Analog Input checks of the A/D converter
- Alarm Relay checks
- Watchdog Timer checks that ensures the software is running
- Transducer Cable checks
- Calibration checks of each of the Feedwater pressure transmitters.

The preventative maintenance program and continuous monitoring of the LEFM ensures that the LEFM remains bounded by the analysis and assumptions set forth in the Topical Report ER-80P. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated. Note that the LEFM provides both feedwater flow and temperature inputs to the core thermal power calculation.

Administrative controls will be implemented to provide guidance for plant control room operations staff in the event the LEFM system is unavailable. A requirement in plant compliance procedure CP-500, will state that if either LEFM or any low-uncertainty heat balance input parameters are inoperable, then reduce power to ≤ 2568 MWt within 12 hours and reduce the nuclear overpower - high setpoint to ≤ 103.3% RTP within 48 hours.

Logic will be programmed with the AULD to compare the improved calorimetric with the existing calorimetric and if the two deviate from one another by a pre-determined value, the AULD will be programmed to automatically transfer from Automatic to Manual in order to prevent any potential power excursions resulting from failed improved calorimetric sensors. Administrative guidance will be developed to assist the Operator in determining whether to remain at 2609 MWt in AULD Automatic using the improved calorimetric or to reduce power to the previous rated thermal power of 2568 MWt (or lower) and transfer to AULD Automatic using the existing calorimetric. In addition to the above comparison logic, the AULD will also be programmed to detect an out-of-range condition for any of the calorimetric inputs. This condition will alert the Operator to investigate the validity of any suspect input.

These requirements ensure that an operable low uncertainty input shall be used whenever power is greater than the pre-uprate RTP level of 2568 MWt. With these requirements in place, the effect on plant operations is that power will be reduced and maintained to the pre-uprate level of 2568 MWt or lower, and that the existing flow nozzles and RTDs will be used for the calorimetric until the LEFM is returned to operable status. These requirements return the measurement techniques and maximum steady state power level to the currently licensed conditions.

Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response to Criterion 2

Criterion 2 does not apply to CR-3.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both Venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

The LEFM uncertainty calculation is based on the ASME PTC 19.1 methodology (Reference 1.3.6) and Alden Research Laboratory Inc. calibration tests. This ASME PTC 19.1 methodology was reviewed by the NRC as part of the Seabrook MUR application and Safety Evaluation Report (Reference 1.3.7). The feedwater flow and temperature uncertainties are then combined with other plant measurement uncertainties (steam temperature, steam pressure, feedwater pressure) to calculate the overall heat balance uncertainty.

This LEFM uncertainty calculation method is consistent with the current heat balance uncertainty calculation that uses the feedwater flow nozzles and feedwater RTDs. The current calculation is based on a square-root-sum-squares calculation, as described in ASME PTC 19.1.

FPC will provide the results of the Alden Research Laboratory calibration and testing to the Staff by September 1, 2007.

Criterion 4

For plants where the ultrasonic meter (including LEFM) was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

A bounding uncertainty for the LEFM has been provided for use in the uncertainty calculation described below (Reference 1.3.8). The acceptability of the bounding calibration factor for the CR-3 spool pieces will be established by tests of these spools at Alden Research Laboratory. These include tests of a full-scale model of the CR-3 hydraulic geometry and tests in a straight pipe. An Alden data report for these tests and a Caldon engineering report evaluating the test data will be prepared. The calibration factor used for the LEFM CheckPlus™ at CR-3 will be verified as acceptable against these reports. The site-specific uncertainty analysis (Attachment E) documents these analyses and will be maintained as a CR-3 design basis calculation.

1.2.5 Total Power Measurement Uncertainty at CR-3 (RIS 2002-03 Section I.1.E)

The total power uncertainty using the LEFM CheckPlus™ at CR-3 is 0.4%. This calculation is provided in AREVA NP Calculation 32-9042687-000 (Reference 1.3.5) and is included as Attachment E. The parameters, their uncertainty, and relative contributions are shown in Table 1-1. The ASME Performance Test Code Methodology was used to calculate the expected core thermal power uncertainty to be achieved using the Caldon CheckPlus™ System ultrasonic flow meter. The analysis concluded that using the following instrument uncertainty values, the core thermal power uncertainty would be 0.394% of 2609 Mwt, thus allowing a power uprate of 1.6% to be pursued. The feedwater flow and temperature measurement is the bulk of this uncertainty (0.34% absolute and ~84% of the total uncertainty). The new steam temperature/pressure instrumentation results in ~4% of the total uncertainty, while the steam pressure measurement uncertainty is ~1% of the total. The Reactor Coolant (RC) pumps energy uncertainty, ambient loss uncertainty and an atmospheric pressure correction uncertainty were chosen to be treated as a bias (algebraically added and not square root sum of the squares (SRSS)) and they are ~11% of the total uncertainty. After the final feedwater flow/temperature uncertainty is determined for the CR-3 specific equipment (post fabrication testing), the total uncertainty may be reduced.

Table D 1-1 below summarizes the core thermal power measurement uncertainty.

TABLE D 1.1 – HEAT BALANCE PARAMETER UNCERTAINTY CONTRIBUTIONS

Symbol	Description	Uncertainty Contribution			
		Absolute $(Btu/hr)^2$	Relative (SRSS)	Relative (Total)	Percent of Total Power
WFW/TFW	Feedwater Flow/Temp	9.166E+14	94.2733%	83.7722%	0.3301%
TS	Steam Temperature	4.432E+13	4.5582%	4.0504%	0.0160%
PS	Steam Pressure	8.573E+12	0.8818%	0.7836%	0.0031%
PFW	Feedwater Pressure	6.140E+08	0.0001%	0.0001%	0.0000%
TLD	Letdown Temperature	2.147E+06	0.0000%	0.0000%	0.0000%
PLD	Letdown Pressure	3.117E+10	0.0032%	0.0028%	0.0000%
WLD	Letdown Flow Rate	2.726E+12	0.2804%	0.2492%	0.0010%
TMU	Makeup Temperature	2.995E+10	0.0031%	0.0027%	0.0000%
	TOTAL	9.723E+14	100%		
	Bias Corrections	**(Btu/hr)**			
QRCP	RCP Power	9.215E+04	NA	2.6261%	0.0103%
QLOSS	Ambient Heat Loss	2.560E+06	NA	7.2948%	0.0287%
PATMOS	P_{gage} to $P_{absolute}$	4.274E+05	NA	1.2181%	0.0048%
Totals				100%	0.394%

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1.2.6 Calibration and Maintenance Procedures of All Instruments Affecting the Power Calorimetric (RIS 2002-03 Section I.1.F)

Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

1.2.6.1 Maintaining Calibration

Calibration of the LEFM will be ensured by preventative maintenance activities previously described in Section 1.2.4, Response to Criterion 1.

New instruments that contribute to the power calorimetric will be maintained according to required calibration and maintenance procedures. The other instruments that contribute to the power calorimetric were unaffected by the addition of the LEFM and will be maintained according to existing calibration and maintenance procedures.

1.2.6.2 Controlling Hardware and Software Configuration

Hardware configuration will be controlled in accordance with Progress Energy procedures, including EGR-NGGC-00D, "Engineering Change," and EGR-NGGC-0012, "Equipment Data Base".

LEFM software will be properly classified in accordance with Progress Energy procedure CSP-NGGC-2507, "Software Documentation and Testing". AULD software will be classified, developed, tested, and controlled in accordance with EGR-NGGC-0157, "Engineering of Plant Digital Systems and Components". Implementation of the AULD software will be performed under the design control process governed by EGR-NGGC-0005, "Engineering Change". Software control will be in accordance with EGR-NGGC-0157, "Engineering of Plant Digital Systems and Components". Software control will be in accordance with EGR-NGGC-0157, "Engineering of Plant Digital Systems and Components".

Instruments that affect the power calorimetric, including the Caldon LEFM CheckPlus™ System inputs, are monitored by CR-3 personnel. Equipment problems for plant systems, including the Caldon LEFM CheckPlus™ System equipment, fall under site work control processes. Conditions that are adverse to quality are documented under the corrective action program. Corrective action procedures, which ensure compliance with the requirements of 10 CFR 50, Appendix B, include instructions for notification of deficiencies and error reporting.

1.2.6.3 Performing Corrective Actions

Corrective actions will be monitored and performed in accordance with Progress Energy procedures CAP-NGGC-0200, "Corrective Action Program," and ADM-NGGC-0104, "Work Management Process".

1.2.6.4 Reporting Deficiencies to the Manufacturer

Reporting deficiencies to the manufacturer will be performed in accordance with Progress Energy procedure CAP-NGGC-0200, "Corrective Action Program".

1.2.6.5 Receiving and Addressing Manufacturer Deficiency Reports

Manufacturer deficiency reports will be received and addressed in accordance with Progress Energy procedure REG-NGGC-0013, "Evaluating and Reporting of Defects and Noncompliance in Accordance with 10 CFR 21".

1.2.7 Allowed Outage Time and Technical Basis (RIS 2002-03 Section I.1.G)

The Nuclear Instrumentation (NI) indicated power is compared against heat balance power on a daily basis. In the event that the LEFM or any low uncertainty heat balance input parameters becomes unavailable, it must be restored to operable status or the plant power will be reduced to 98.4% RTP (≤ 2568 MWt) within 12 hours (see Item 1.2.8 below). The justification for the allowed outage time of the LEFM is that the NIs were compared to the last known good heat balance calculation using the LEFM measurement, do not routinely require adjustments and thus can continue to be relied upon for power measurement until the next daily comparison. The time period is reasonable based on the functions and the capability of performing a calorimetric calculation without using the LEFM.

1.2.8 Actions for Exceeding Allowed Outage Time and Technical Basis (RIS Section I.1.H)

Administrative controls will be placed in CR-3 procedure CP-500 to address LEFM or any low-uncertainty heat balance input parameter unavailability. Should the LEFM system become unavailable, the current flow nozzle-based feedwater flow and RTD feedwater temperature instrumentation will be used as input to the core power calorimetric, and the core power will be limited to the current licensed power level of 2568 MWt. The reactor operators will be provided with procedural guidance for those occasions when the LEFM CheckPlus™ or any low-uncertainty heat balance input parameter is not available.

SECTION 1.0 REFERENCES

1.3.1 ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Caldon, Inc., dated March 1997.

1.3.2 ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System," Caldon, Inc., dated October 2001.

1.3.3 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry, TU Electric, Comanche Peak Steam Electric Station, Units 1 and 2 – Review of Caldon Engineering Topical Report ER 80P, Improving Thermal Power Accuracy and Plant Safety while Increasing Power Level Using the LEFM System (TAC Nos. MA2298 and 2299), dated March 8, 1999.

1.3.4 Letter from S. A. Richards, NRC, to M. A. Krupa, Entergy, Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station – Review of Caldon, Inc. Engineering Report ER-157P (TAC Nos. MB2397, MB 2399 and MB2468), dated December 20, 2001.

1.3.5 32-9042687-000, "CR-3 Heat Balance Uncertainty for CR-3 MUR."

1.3.6 ASME PTC 19.1-1998, Test Uncertainty, Instruments and Apparatus, American Society of Mechanical Engineers, NY, NY, 1998.

1.3.7 NRC Letter, G. Edward Miller to Gene F. St. Pierre, Subject: Seabrook Station, Unit No. 1 – Issuance of Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MC8434), May 2006.

1.3.8 ER-579 Rev. 0, "Bounding Uncertainty Analysis for Thermal Power Determination at Crystal River Unit 3 Using the LEFM✓+ System," dated February 2007

1.3.9 NRC Letter, B. E. Thomas to E. M. Hauser, "Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check AND $\text{CheckPlus}_{\text{TM}}$ Ultrasonic Flow Meters (UFMs)" (TAC NO. MC6424) July 5, 2006

2.0 Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level (RIS 2002-03 Section II Questions)

2.1 A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

A. Identify the transient or accident that is the subject of the analysis

B. Confirm and explicitly state that

i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant

ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC

iii. the analyses of record are not changed by the requested power uprate

C. Confirm that bounding event determinations continue to be valid

D. Provide a reference to the NRC's previous approvals discussed in Item B. above.

2.2 Response to RIS 2002-03 Section II Questions

In order to support the CR-3 MUR Power Level Uprate, with respect to the accident analyses, a review of the Final Safety Analysis Report (FSAR), Reference 4.3.1, Chapters 6 and 14 and other related sub-sections was performed. Evaluations were performed on other analyses as well and it was determined there was no impact from the MUR. The purpose of the review was to confirm that the analysis results, as currently presented in the FSAR, were performed conservatively and bound the proposed power uprate. All of the analyses that are included in the FSAR have been performed using NRC-approved tools and methods. If any event was determined to be not bounded by the current FSAR analyses, then a new analysis was performed as discussed in Section 3.0.

For the Babcock and Wilcox (B&W)-designed plants, the heat balance uncertainty is accounted for in the initial core power level that is modeled in the accident analyses and included in the determination of the nuclear overpower reactor trip setpoint. For the implementation of the MUR, the intent is to use a higher accuracy feedwater flow measurement devise to reduce the secondary side heat balance uncertainty from 2% to 0.4%. The reduced uncertainty would then be used to increase the rated thermal power level of the plant from 2568 MWt to 2609 MWt. Ideally, no new accident analyses would be required as long as the new error-adjusted power level does not exceed what was modeled in the analyses (i.e., 102% of 2568 MWt).

As stated above, the heat balance uncertainty is also used to determine the overpower reactor trip setpoint specified in the technical specifications for Crystal River Unit 3. The current overpower reactor trip setpoint modeled in the accident analyses is 112% of 2568 MWt (or ~2876 MWt). The methodology used to derive the technical specification trip setpoint is described in Section 7 of BAW-10179P-A. The uncertainty associated with power measurement, including the secondary side heat balance uncertainty, is applied to the analysis value to derive the current plant technical specification setpoint allowable value of 104.9% of 2568 MWt (or ~2694 MWt). Therefore, in order to ensure that the over power limit modeled in the accident analyses, in terms of absolute megawatts, is preserved, the overpower analysis setpoint must be reduced proportionally to 110.2% (or 2876 MWt / 2609 MWt * 100%). When the power measurement uncertainties with the reduced heat balance uncertainty are applied to 110.2% analysis value, the technical specification allowable value will be 104.9% of 2609 MWt. Although the allowable value will remain at the same percentage, the actual reactor trip will occur at a higher absolute power (104.9 % of 2609 MWt rather than 104.9 % of 2568 MWt). The analysis determined that the 104.9 % allowable value still supports the assumption of a reactor trip prior to 2876 MWt. This allowable value remains valid as long as the higher accuracy feedwater flow measurement system is operable.

If the higher accuracy feedwater flow measurement system becomes inoperable, the secondary heat balance uncertainty will return to the 2% value associated with the main feedwater flow nozzles. Accordingly, the plant power level and reactor overpower trip setpoints in terms of absolute megawatts must be returned to the pre-MUR values. That is, core power must be reduced to 2568 MWt and the Nuclear Overpower High Flux trip setpoint will be reduced to 103.3% of RTP.

A summary of each accident is provided below and is summarized in Table 2-1 (All information is taken from Reference 2.3.2).

2.2.1 Uncompensated Operating Reactivity Changes (FSAR Section 14.1.2.1)

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion, burnable poison depletion, and changes in fission product poison concentration. These reactivity changes, if left uncompensated, can cause the operating limits to be exceeded. In all cases, the Reactor Protection System (RPS) setpoints are placed to prevent the safety limits from being exceeded. No damage occurs from these conditions.

There are two acceptance criteria for this accident. First, the rate of reactivity will be much less than the rate at which the operator can compensate for the addition. Second, the rate of temperature change will be much less than the rate at which the automatic control system can compensate for the change.

The plant and control system response to reactivity changes resulting from fuel depletion, burnable poison depletion, and changes in fission product poison concentration are not significantly affected by the initial core power level. As a result, the change in the magnitude of reactivity changes caused by fuel depletion, burnable poison depletions, and/or changes in fission product poison concentration will be negligible. The analysis was initiated at 2575 MWt and is insensitive to initial core power. An increase in the analyzed power to 102% of 2568 MWt will not result in any appreciable change in the accident as previously analyzed.

The analysis of record for this accident was accepted by the NRC as part of the approval of the original CR-3 FSAR, Reference 2.3.1. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205. The individual accidents are discussed below.

2.2.2 Startup Accident (FSAR Section 14.1.2.2)

The startup accident is a moderate frequency event that results from a spurious control rod withdrawal from hot zero power conditions. The acceptance criteria for the event are that the peak reactor coolant system (RCS) pressure does not exceed 2750 psig, and the maximum allowed core power does not exceed 112% of rated thermal power. Therefore, the primary reactor protection system (RPS) trip functions that are credited for this event are the high RCS pressure and core over-power. This reactivity addition event is considered a heat-up transient that results in the pressurization of the RCS. The startup accident is the limiting event in ensuring overpressure protection of the RCS as outlined in the FSAR. The transient is initiated from hot zero power conditions and as a result, the MUR power uprate has no effect on the initial conditions within the RCS.

The startup accident credits the reactor trip on high neutron flux. In analytical space, the high neutron flux set point is presently defined as 112% of 2568 MWt (2876.16 MWt). For MUR conditions, the absolute power of 2876.16 MWt will remain the analytical limit for the high neutron flux setpoint. The setpoint expressed as a percent of the rated power condition will be reduced to 110.2% of rated power at MUR conditions. Using the same absolute power for the setpoint ensures the same protection at MUR conditions that currently exist for the rated power condition at 2568 MWt.

The analysis of record for this accident is reflected in the CR-3 FSAR, and remains acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.3 Rod Withdrawal at Rated Power Operation Accident (FSAR Section 14.1.2.3)

The rod withdrawal accident is a moderate frequency event that results from a spurious control rod withdrawal from rated power conditions. The acceptance criteria for the event are that the peak RCS pressure does not exceed 2750 psig and the maximum allowed core power does not exceed 112% of rated power. Therefore, the primary RPS trip functions that are credited for this event are the high RCS pressure and core over-power. This reactivity addition event is considered a heat-up transient that results in pressurization of the RCS.

The initial core power level for the current rod withdrawal at power accident analysis is 100% of 2568 MWt. With the MUR power uprate, the RCS average temperature and initial pressurizer level will not change. The steam space in the pressurizer will also not be affected.

A spectrum of reactivity insertion rates (RIRs) is simulated to demonstrate compliance with the event acceptance criteria. For slow RIRs, the neutron and thermal power increase at nearly the same rate. The RCS temperature, and hence reactor pressure, increases rapidly to the high pressure trip setpoint. A different RIR will become limiting, but the MUR power uprate will still be bounded by the current plant FSAR analysis from the peak pressure perspective.

For fast RIRs, reactor protection is provided by the over-power trip setpoint. The transient response will be governed by the power difference between the initial core power and the over-power trip setpoint. The larger the difference between these values will result in a more severe transient.

In the current FSAR analysis, the power difference (or the net energy added) is 114% of 2568 MWt (Reference 2.3.1, Section 14.1.2.3.2). As discussed in the FSAR Section 14.1.2.3.2, the over-power trip setpoint was reduced to the current analytical setpoint of 112% of 2568 MWt due to fuel densification issues. This change took place after the original FSAR analysis and no new analyses were performed.

In order to provide the same over-power protection under MUR conditions, the over-power setpoint used in the analysis will be reduced to 110.2% of the MUR power level of 2609 MWt, or approximately 2876 MWt, which is the same net value as 112% of 2568 MWt. Since the initial core power level would be higher with the MUR, the net energy added to the RCS before the reactor trip setpoint is reached would be less than as in the current FSAR analysis. Therefore, the rod withdrawal at power accident analyses described in the FSAR will remain applicable for the MUR power uprate from the peak power perspective.

The analyses of record for this accident are reflected in the CR-3 FSAR and remain acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.4 Moderator Dilution Accident (FSAR Section 14.1.2.4)

The moderator dilution accident (MDA) is a moderate frequency event and results from an uncontrolled dilution of the primary coolant. The dilution of the moderator will result in a positive reactivity addition to the core and a corresponding heatup and pressurization of the RCS. The acceptance criteria for this accident relate to peak RCS pressure, maximum allowed power, and minimum subcritical margin.

The transient progression is determined by the combinations of the dilution flow rate and the cycle-specific reactivity parameters. Conservative reactivity parameters and dilution flow rates are modeled to ensure a bounding analysis. These cycle-specific parameters are validated during each reload analysis to ensure the bounding analyses remain conservative. The analysis was performed at 102% of 2568 MWt, and complies with the acceptance criteria that peak power not exceed 112% of rated thermal power and peak RCS pressure not exceed 110% of design pressure.

Also for the dilution event, a minimum shutdown margin must be maintained during refueling conditions. Compliance to the shutdown margin requirement is demonstrated as part of the cycle-specific reload calculations because no system level transient is simulated and the results are largely unaffected by the MUR power uprate.

The CR-3 FSAR discusses an unterminated dilution event through the decay heat removal system. A plant modification was performed to prevent the possibility of dilution by sodium hydroxide (NaOH) addition. Therefore, this event is no longer possible.

The moderator dilution accident credits the reactor trip on high neutron flux. In analytical space, the high neutron flux se point is presently defined as 112% of 2568 MWt (2876.16 MWt). For MUR conditions, the absolute power of 2876.16 MWt will remain the analytical limit for the high neutron flux setpoint. The setpoint expressed as a percent of the rated power condition will be updated to 110.2% of rated thermal power at MUR conditions. Using the same absolute power for the setpoint ensures the same protection at MUR conditions that currently exist for the rated power condition at 2568 MWt.

The analysis of record for this accident is reflected in the CR-3 FSAR and remains acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.5 Cold Water Accident (FSAR Section 14.1.2.5)

This transient results from the startup of an idle loop while the plant is operating at reduced power. The cold water accident (CWA) is a moderate frequency event. The acceptance criteria for the event are that the peak RCS pressure does not exceed 2750 psig and the maximum departure from nucleate boiling ratio (MDNBR) does not decrease below 1.30.

The analysis assumed that the plant was operating with one reactor coolant pump in each loop at 50% of rated power when the remaining two pumps were started. The increase in primary coolant flow and negative reactivity coefficients results in a positive reactivity insertion and subsequent increase in core power. The increase in core power limits the primary coolant temperature decrease and the plant reaches equilibrium at a new power level of approximately 65% which is still less than the rated power. No RPS trip setpoints are challenged. The increase in coolant flow combined with an increase in power to 65% (thermal) does not result in an unacceptable minimum DNBR. The RCS pressure increases approximately 137 psi and remains well below the high pressure reactor trip setpoint. The MUR will not impact the results of this analysis.

The analysis of record for this accident is reflected in the CR-3 FSAR and remains acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.6 Loss-of-Coolant-Flow Accident (FSAR Section 14.1.2.6)

The loss of coolant flow (LOCF) accidents result from either loss of power or mechanical failure of one or more of the RCPs. The LOCF accidents are comprised of three different transients. The simultaneous coastdown of all four RCPs is considered an infrequent event. The single locked pump rotor is considered a limiting fault transient. Although the four pump coastdown is considered an infrequent event, it is typically analyzed to the more restrictive criteria of the moderate frequency event category. For the locker rotor transient, no fuel cladding failure is allowed. These events are evaluated for each new fuel reload. The acceptance criteria for these events relate to the minimum allowed DNBR based on the applicable critical heat flux correlation for the fuel design being analyzed. These events were analyzed at 102% of 2568 MWt and include a 2% power measurement uncertainty in the calculations. In addition, the DNBR calculations are verified for each new core design.

The analysis of record for this accident is reflected in the CR-3 FSAR and remains acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.7 Stuck-Out, Stuck-In, or Dropped Control Rod Accident (FSAR Section 14.1.2.7)

The dropped rod accident is the limiting event in the group of transients identified with misaligned control rods. A misaligned Control Rod Assembly (CRA) is defined as the deviation of a CRA from its group reference position by more than nine inches (indicated). This definition encompasses both the action of having a single CRA stick while moving its associated group or dropping a single CRA. With respect to stuck CRAs, core design requirements ensure that a Shutdown Margin (SDM) of a least 1.0% $\Delta K/K$ exists with the greatest worth CRA full withdrawn from the core. On the other hand, should a CRA stick while pulling its associated group, control systems will function to sound an alarm, inhibiting all CRAs out-movement. The consequences of stuck CRA accidents are therefore limited in severity because of the restrictions associated with rod movement and core design. Thus, the dropped CRA accident, which has restrictions for rod insertion, is the limiting CRA misalignment event.

The dropped control rod accident is a moderate frequency event and the acceptance criteria for this event relate to peak RCS pressure and MDNBR.

The FSAR analysis of record is based on a core power level of 2772 MWt and a core design with steady-state peaking factors allowed by implementation of the statistical core design. The power level bounds the MUR. A cycle specific DNBR evaluation is addressed in the maneuvering analysis during the standard reload process.

2.2.8 Load Rejection Accident (Turbine Trip) (FSAR Section 14.1.2.8)

The plant was originally design to withstand the effects of a load rejection transient without reactor or turbine trip. The reactor power would automatically be runback to the power level corresponding to the steam generator low level limit. The power operated relief valve (PORV) was available to relieve pressure to prevent a reactor trip. The acceptance criteria for this event are that fuel damage would not occur and that the RCS pressure would not exceed the core pressure limit of 110% of the design pressure. Fuel is not expected to fail during the load rejection analysis, and therefore the dose consequences are bounded by the Main Steam Line failure accident. The Analysis of Record (AOR) for the original load rejection accident is discussed in the FSAR.

The current plant response to a load rejection is different than the description presented above because the PORV lift setpoint has been raised above the high reactor coolant pressure reactor trip setpoint. A load rejection from 100% power with the higher PORV lift setpoint would result in a reactor trip on high reactor coolant pressure. The plant response to a load rejection under this configuration is similar to a turbine trip, but is less severe because the closure of the turbine stop valves during a turbine trip causes a more rapid pressurization. It is noted that the Anticipatory Reactor Trip System (ARTS) is not credited in the turbine trip analysis of record.

The turbine trip accident from full power bounds the load rejection accident. This analysis was evaluated over a range of power levels up to 112% of 2568 MWt. The analysis concluded that a 3% maximum tolerance for the main steam safety valves (MSSV) for one inoperable MSSV was sufficient for power levels up to 112% of 2568 MWt, which is equivalent to the maximum allowed power for the MUR (110% of 2609 MWt).

The analyses of record for this accident are reflected in the CR-3 FSAR and remain acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.9 Station Blackout Accident (FSAR Section 14.1.2.9)

The original Station Blackout Accident (SBO) analysis was evaluated as a Loss of AC Power (LOAC) event. This analysis was performed to show that the plant would transition to a stable condition in which decay heat would be removed by the steam generators via natural circulation. During this event, the loss of AC power will initiate a reactor trip, RC pump trip, a turbine trip and the turbine stop valves (TSV) will close. As a result, the secondary side pressure will increase to the main steam safety valve setting which limits the secondary heat removal capacity. This causes an initial reactor coolant heatup and pressure increase. This analysis is historical and has been superseded by a calculation that was prepared in response to the recommendations of the Nuclear Management and Resources Council (NUMARC) to determine the capability of a nuclear plant to cope and to recover from a SBO event for four hours.

The acceptance criteria for a SBO event dictate that fuel damage shall not occur, the reactor coolant system shall not exceed core pressure limits, and the accident doses shall be within the 10 CFR 50.67 limits. The original LOAC power event, evaluated at 100% of 2568 MWt, is discussed in the FSAR. The NUMARC analyses are documented in the FSAR and were evaluated at 100% of 2772 MWt and therefore bound the MUR power uprate conditions.

2.2.10 Steam Line Failure Accident (FSAR Section 14.2.2.1)

The steam line break is a rupture in the steam lines between the steam generators and the turbine. The rapid depressurization causes an increase in the main feedwater flow rate. The increase in steam flow to the break and the turbine results in a large overcooling of RCS. The steam line break accident is the most severe overcooling transient. The acceptance criteria relates to effective core cooling, offsite dose release, reactor coolant system integrity, and containment vessel integrity.

For the core response, as documented in the FSAR, the core power was evaluated at 100% of 2568 MWt, to minimize the heat input to the reactor coolant system. The heat balance uncertainty of 2% was accounted for in the steam generator mass inventories. Therefore, the FSAR analysis bounds the MUR power uprate for the core and reactor coolant system. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

The dose release calculations are evaluated on a reload basis at a power level of 102% of 2568 MWt.

For the containment response, the steam line break event was re-evaluated to support MUR power uprate conditions at 2619.4 MWt. The analysis demonstrated that the MUR had a negligible effect on core decay heat. Consequently, the peak containment pressure increased by only 0.1 psi. Therefore, the conclusions reported in the FSAR regarding compliance to the reactor building pressure limits remain valid.

2.2.11 Steam Generator Tube Rupture Accident (FSAR Section 14.2.2.2)

The Steam Generator Tube Rupture (SGTR) is a postulated double-ended rupture of a steam generator tube with unrestricted discharge from both ends of the tube. The acceptance criteria are related to offsite dose and further degradation of the primary-to secondary boundary beyond the affected tube.

The SGTR is a breach of the reactor coolant pressure boundary and results in a transfer of primary coolant to the secondary system. The core protection aspects of a SGTR are bounded by small break LOCA. Therefore, the SGTR event is analyzed to determine the offsite doses resulting from the release of contaminated primary coolant into the steam generator and to the atmosphere via the main steam safety valves.

The system response for the SGTR analysis of record is based on a constant leak rate. The leak flow rate is based on critical flow from each end of the ruptured tube. The leak rate was assumed to be constant until the plant was cooled down to the decay heat removal cut-in temperature. This is conservative because it does not credit the decrease in the leakage rate with RCS depressurization or the secondary side pressurization following the reactor trip and turbine trip. The SGTR calculation is independent of power level based on the analytical method used. Therefore, there is no impact on the system response due to uprate.

The acceptance criteria for the evaluation of this accident are public radiological doses must not exceed the allowable limits prescribed by 10 CFR 50.67 and Regulatory Guide 1.183 (2.5 rem Total Effective Dose Equivalent (TEDE) for a coincident iodine spike). Additionally, the event must not result in additional tube failures and further degradation of the integrity of the reactor coolant pressure boundary caused by the effects of temperature gradients.

The analyses of record for this accident was evaluated at a power level of 102% of 2568 MWt as reflected in the CR-3 FSAR, and remains acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.12 Fuel Handling Accident (FSAR Section 14.2.2.3)

Mechanical damage to a fuel assembly is postulated during refueling operations. The analyses for this accident consider an accident inside containment and outside containment. The core power level is used to determine the activity levels in the fuel-to-clad region prior to the accident.

The acceptance criteria for the Fuel Handling Accident are based on the requirements of 10 CFR 50.67 and Regulatory Guide 1.183. The analyses of record for this accident was evaluated at a power level of 102% of 2568 MWt as reflected in the CR-3 FSAR, and remains acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment.205.

2.2.13 Rod Ejection Accident (FSAR Section 14.2.2.4)

The rod ejection event is a postulated event involving a physical failure of a pressure barrier component in the Control Rod Drive assembly and subsequent ejection of the control rod. The event is classified as an infrequent event. The acceptance criteria for the Rod Ejection from full power event relate to peak RCS pressure and peak fuel enthalpy.

The ejection of a control rod with the reactor at full power causes a rapid positive reactivity insertion. Core power and fuel temperatures increase rapidly. The rapid fuel temperature rise produces negative Doppler reactivity feedback that terminates the power excursion. A reactor trip occurs on over-power and the reactor is returned subcritical by control rod insertion. The primary safety valves provide steam relief to limit the peak RCS pressure to less than the acceptance criterion. Limiting the reactivity worth of a given rod in the fuel design and the initial fuel enthalpy at full power will ensure that the peak fuel enthalpy does not exceed the maximum allowable limit.

At hot zero power conditions, a rod ejection accident initiated from zero power is not directly impacted by the MUR power uprate.

The rod ejection at hot full power conditions were originally evaluated at 100% of 2568 MWt as documented in the FSAR. The neutron power response during a control rod ejection accident is not sensitive to the initial power conditions. Due to the rapid ejection time of 0.15 seconds, the transient is defined by the ejected rod worth and the kinetics parameters. The MUR will cause a slight increase in the fuel heat up of approximately 2 calories per gram (cal/g). There is approximately 80 cal/g margin to the fuel enthalpy limit for an ejected rod worth of 0.7%ΔK/K. The reload core design ensures that maximum ejected rod worth will not exceed 0.65%ΔK/K including a 15% uncertainty. Therefore, the current fuel enthalpy margin more than compensates for the small power increase.

The radiological analyses for the rod ejection accident assumed that the fuel gap activity for 14% of the fuel rods is completely released. The dose release calculations are evaluated on a reload basis at a power level of 102% of 2568 MWt. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.14 Loss of Coolant Accident (FSAR Section 14.2.2.5)

A spectrum of break sizes and break locations is postulated in the primary coolant piping. The LOCAs are considered limiting fault transients, events that are not expected to occur, but are postulated because of the potential for large releases of radiation. The acceptance criteria relate to ensuring adequate core cooling for the short and long term post-LOCA, containment vessel pressure and temperature, and offsite dose consequences.

For compliance to adequate core cooling, the large and small break loss of coolant accident analyses were evaluated at 102% of 2568 MWt as documented in the FSAR.

For compliance to offsite dose consequences, the loss of coolant accidents were evaluated at radioactive nuclide inventories consistent with 102% of 2568 MWt These analyses address the maximum hypothetical accident discussed in Section 14.2.2.7 of the FSAR.

In addition, post-LOCA boron control management analyses were performed as discussed in the FSAR. These analyses were evaluated at 102% of 2568 MWt. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

A revised containment analysis based upon new mass and energy releases was performed. The description of this analysis is provided in Section 5.0.

2.2.15 Makeup System Letdown Line Failure Accident (FSAR Section 14.2.2.6)

Regulatory Guide 1.70, Table 15-1, indicates breaks in lines connected to the reactor coolant system that carry reactor coolant outside containment should be evaluated for dose consequences. The most severe piping rupture for which radioactivity release is postulated during normal plant operation is in the letdown line of the Makeup and Purification System. The acceptance criteria for this accident are described in 10 CFR 50.67.

The reactor is operating at 102% of 2568 MWt. The rupture is modeled as a complete severance of the 2½ inch nominal diameter letdown line at a location downstream of the outboard isolation valve. A single emergency diesel generator is assumed to fail, and no credit is taken for the operators to increase the steam generator levels. Operators are assumed to isolate the letdown line at 10 minutes after the hot leg reaches saturated conditions.

The analyses of record for this accident was evaluated at a power level of 102% of 2568 MWt as reflected in the CR-3 FSAR and remains acceptable for the MUR power uprate. Radiological consequences are assessed during the standard reload process. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.16 Waste Gas Decay Tank Rupture Accident (FSAR Section 14.2.2.8)

The waste gas decay tank is used in the radioactive waste disposal system to store radioactive gaseous waste from the station until such time that the radioactive decay renders the gas safe for release to the site environment. Rupture of a waste gas tank would result in the premature release of its radioactive contents to the station ventilation system and to the atmosphere through the station vent.

The acceptance criteria for the Waste Gas Decay Tank Rupture Accident (WGDTRA) are based on the requirements of 10 CFR 50.67 and Regulatory Guide 1.183. The analysis of record for this accident is reflected in the CR-3 FSAR. The analysis conservatively assumes that all three available waste gas decay tanks rupture. Each tank is assumed to contain the maximum curie inventory allowed by the Offsite Dose Calculation Manual. The WGDTRA would release more radioactivity to the atmosphere than any other credible radwaste system accident. The dose assessment for the WGDTRA is based on WGDT inventories of radioactive nuclides and are independent of power level. Therefore, the WGDTRA analysis is not affected by operation of CR-3 at 2609 MWt.

2.2.17 Loss of Feedwater and Main Feedwater Line Break Accident (FSAR Section 14.2.2.9)

A loss of feedwater accident results from either a reduction in or the complete loss of secondary feedwater to the steam generators. The loss of feedwater may be caused by pump failure, valve closure, or a feedwater line break. The acceptance criteria are that fuel failure shall not occur, the peak RCS pressure will not exceed code pressure limits of 110% of the design pressure, and offsite dose consequences remain less than the limits specified in 10 CFR 50.67. The loss of feedwater and feedwater line break accidents were evaluated at 102% of 2568 MWt. The loss of feedwater accident is also used to establish the minimum required emergency feedwater (EFW) flow rate of 550 gpm. The feedwater line break accident is considered a limiting fault event. However, the analysis is analyzed with an imposed minimum DNBR limit to prevent fuel failures.

The analyses of record for these accidents are reflected in the CR-3 FSAR and remain acceptable for the MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

2.2.18 ATWS Transients (DSS, AMSAC) (FSAR Section 7.5)

The Anticipated Transients Without Scram (ATWS) events are evaluated in compliance to 10 CFR 50.62. An ATWS event is an anticipated operational occurrence followed by the failure of the reactor trip portion of the reactor protection system.

For compliance to 10 CFR 50.62 criteria, CR-3 has installed a Diverse Scram System (DSS) and an ATWS Mitigating System Actuating Circuitry (AMSAC) system. DSS provides an interruption of power to the control systems at high reactor pressure, and AMSAC provides an actuation of emergency feedwater and trips the turbine at power levels above approximately 50% of rated and feedwater flow below 17% of rated. Both systems are independent of the reactor protection system (RPS), and both are operable during a loss of offsite power.

The design basis transient for the DSS is the loss of main feedwater (LOFW) with a failure of the RPS reactor trip. DSS actuates on a high RCS pressure of 2450 psig (FSAR Section 7.5). The LOFW transient was evaluated generically at 2772 MWt and ensured the peak RCS pressure remained below 3250 psia. Therefore, the current analysis of record for full power operation bounds the conditions for MUR power uprate. This analysis was also reviewed and accepted by the Staff during the review for Amendment 205.

The approval for the analyses of record for these accidents is contained in Reference 2.3.4.

A separate analysis was performed to confirm that the current AMSAC arming setpoint remains valid. The description of this analysis is provided in Section 3.2.2.

2.2.19 ARTS Transients (FSAR Section 7.2.3.2.4)

The Anticipatory Reactor Trip System (ARTS) will trip the reactor if a turbine trip occurs with reactor power above 45%. ARTS was implemented after the Three Mile Island (TMI-2) accident to minimize the challenges to the pressurizer power operated relief valve (PORV) after a turbine trip. The ARTS trip function is not credited in the design bases accidents.

The ARTS power level setpoint is based upon the maximum core thermal power, wherein a reactor runback is capable of minimizing system pressures below the PORV setpoint. This power level is sensitive to the flow capacities of the turbine bypass (TBV) and the main steam safety valves (MSSV), and the reactor kinetics.

The current design basis analysis is a generic evaluation performed at a rated power condition of 2772 MWt and therefore bounds the MUR (Reference 2.3.3).

2.3 Section 2.0 References

2.3.1 CR-3 Final Safety Analysis Report, Rev. 30.1.
2.3.2 51-9036887-000, "CR-3 MUR Summary Report."
2.3.3 BAW 1893A, "Basis for Raising Arming Threshold for Anticipatory Trip on Turbine Trip," August 1986.
2.3.4 Safety Evaluation Crystal River, Unit 3 Compliance With ATWS Rule 10 CFR 50.62, Docket No. 50-302, April 1989.

TABLE 2-1 Crystal River Unit 3 FSAR Accident Analyses

FSAR Section(s)	Event	Initial Core Power Used in FSAR Analysis (% of 2568)	Bounded by Current FSAR Analysis	Supported / Bounded by Other Analyses	Discussion
14.1.2.1	Uncompensated Operating Reactivity Changes	100.2% (2575 MWt)	X		This accident was originally analyzed to demonstrate the ability of control systems and operators to compensate for slow variations in reactivity. The analysis was evaluated at 2575 MWt in Reference 2.3.1. A slight increase in power to the MUR conditions will not result in any appreciable change in the accident as previously analyzed. The reactivity changes for this event are also bounded by the reactivity changes in the startup accident. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.1.2.2 (4.3.7)	Startup Accident	10^{-7} %	X		The analytical high flux reactor trip setpoint will be reduced from 112% (of 2568 MWt) to 110.2% (of 2609 MWt) to ensure that the reactor is tripped at the same net power level. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.1.2.3	Rod Withdrawal at Rated Power Operation Accident	100%	X		The analytical high flux reactor trip setpoint will be reduced from 112% (of 2568 MWt) to 110.2% (of 2609 MWt) to ensure that the reactor is tripped at the same net power level. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.1.2.4	Moderator Dilution Accident	102%	X		The analytical high flux reactor trip setpoint will be reduced from 112% (of 2568 MWt) to 110.2% (of 2609 MWt) to ensure that the reactor is tripped at the same net power level. The shutdown margin calculation is evaluated as part of the standard reload process. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.

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TABLE 2-1 Crystal River Unit 3 FSAR Accident Analyses

FSAR Section(s)	Event	Initial Core Power Used in FSAR Analysis (% of 2568)	Bounded by Current FSAR Analysis	Supported / Bounded by Other Analyses	Discussion
14.1.2.5	Cold Water Accident	50%	X		Current FSAR analysis remains bounding as analyzed. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.1.2.6	Loss-of-Coolant Flow Accident	102%	X		The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate. In addition, the DNBR response is verified for each new fuel cycle. Current cycle analyses support the MUR power uprate.
14.1.2.7	Stuck-Out, Stuck-In, or Dropped Control Rod Accident	108%	X		The analysis of record is provided in Reference 2.3.1 and is based on a core power level of 2772 MWt. The analysis of record remains acceptable for the MUR power uprate.
14.1.2.8	Load Rejection Accident (Turbine Trip)	112%	X		Under the current configuration, the turbine trip accident bounds the consequences of a load rejection accident. The turbine trip analyses were evaluated at 112% of 2568 MWt. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.1.2.9	Station Blackout Accident	108%	X		SBO was evaluated at 2722 MWt. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.

TABLE 2-1 Crystal River Unit 3 FSAR Accident Analyses

FSAR Section(s)	Event	Initial Core Power Used in FSAR Analysis (% of 2568)	Bounded by Current FSAR Analysis	Supported / Bounded by Other Analyses	Discussion
14.2.2.1	Steam Line Failure Accident	100%/102%	X		For the core response, as documented in the FSAR, the core power was evaluated at 100% of 2568 MWt, to minimize the heat input to the reactor coolant system. The heat balance uncertainty of 2% was accounted for in the steam generator mass inventories. Therefore, the FSAR analysis bounds the MUR power uprate for the core and reactor coolant system response.
	Steam Line Failure Mass & Energy Releases	100%	See Discussion		For the containment response, the steam line break event was re-evaluated to support MUR power uprate conditions at 2619.4 MWt. The analysis demonstrated that the MUR had a negligible effect on core decay heat, consequently the peak containment pressure increased by only 0.1 psi. Therefore, the conclusions reported in the FSAR regarding compliance to the reactor building pressure limits remain valid.
14.2.2.2	Steam Generator Tube Rupture Accident	102%	X		The SGTR was evaluated at 102% of 2568 MWt. The dose consequences are evaluated each cycle as part of the standard reload process. These analyses bound the MUR power uprate. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.2.2.3	Fuel Handling Accident	102%	X		The Fuel Handling accident was evaluated at 102% of 2568. The dose consequences are evaluated each cycle as part of the standard reload process. These analyses bound the MUR power uprate The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.

TABLE 2-1 Crystal River Unit 3 FSAR Accident Analyses

FSAR Section(s)	Event	Initial Core Power Used in FSAR Analysis (% of 2568)	Bounded by Current FSAR Analysis	Supported / Bounded by Other Analyses	Discussion
14.2.2.4	Rod Ejection Accident	0.1%	X		The Hot Zero Power analyses are not impacted by a change in rated power level. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
		100%	X		The rod ejection at hot full power conditions were originally evaluated at 100% of 2568 MWt as documented in the FSAR. The neutron power response during a control rod ejection accident is not sensitive to the initial power conditions. Due to the rapid ejection time of 0.15 seconds, the transient is defined by the ejected rod worth and the kinetics parameters. The MUR will cause a slight increase in the fuel heat up of approximately 2 cal/g. There is approximately 80 cal/g margin to the fuel enthalpy limit for an ejected rod worth of 0.7%ΔK/K. The reload core design ensures that maximum ejected rod worth will not exceed 0.65%ΔK/K including a 15% uncertainty. Therefore, the current fuel enthalpy margin more than compensates for the small power increase. The radiological analyses for the rod ejection accident assumed that the fuel gap activity for 14% of the fuel rods is completely released. The dose release calculations are evaluated on a reload basis at a power level of 102% of 2568 MWt.

TABLE 2-1 Crystal River Unit 3 FSAR Accident Analyses

FSAR Section(s)	Event	Initial Core Power Used in FSAR Analysis (% of 2568)	Bounded by Current FSAR Analysis	Supported / Bounded by Other Analyses	Discussion
14.2.2.5	Loss-of-Coolant Accident	102%	X		The spectrum of LOCAs was analyzed for CR-3 at 102% of 2568 MWt. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
	LOCA Mass & Energy Releases	100%	--		The mass and energy release analyses for compliance to containment pressure and temperature criteria was re-evaluated to support operations at 102% of 2568 MWt and is discussed in Section 3.0 of this report.
14.2.2.6	Makeup System Letdown Line Failure Accident	102%	X		The dose consequences were evaluated at 102% of 2568 MWt. These analyses bound the MUR power uprate. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.
14.2.2.8	Waste Gas Decay Tank Rupture Accident	102%		X	The analysis conservatively assumes that all three available waste gas decay tanks rupture. Each tank is assumed to contain the maximum curie inventory allowed by the Offsite Dose Calculation Manual. The dose assessment for the WGDTRA is based on WGDT inventories of radioactive nuclides and are independent of power level.
14.2.2.9	Loss of Feedwater and Main Feedwater Line Break Accident	102%	X		The total loss of feedwater accident and the feedwater line break was evaluated at 102% of 2568 MWt. The analysis of record is provided in Reference 2.3.1 and remains acceptable for the MUR power uprate.

TABLE 2-1 Crystal River Unit 3 FSAR Accident Analyses

FSAR Section(s)	Event	Initial Core Power Used in FSAR Analysis (% of 2568)	Bounded by Current FSAR Analysis	Supported / Bounded by Other Analyses	Discussion
7.5	ATWS/DSS Setpoint	108% (2772 MWt)	X		The ATWS transients are considered beyond the original design basis of the B&W-designed plants. The analyses were performed using nominal values and was evaluated at 2772 MWt. The approval of the analysis of record for CR-3 is provided in Reference 2.3.4 and remains acceptable for the MUR power uprate.
	ATWS/AMSAC Enabling Setpoint	49.53% (50% of 2544 MWt)	--		The ATWS LOFW was re-analyzed at a power level of 52% of 2609 MWt to validate the arming setpoint of the AMSAC system and is discussed in Section 3.0.
7.2.3.2.4	ARTS	108% (2772 MWt)	--		The current design basis analysis is a generic analysis performed at a rated power condition of 2772 MWt. The analysis of record is provided in Reference 2.3.3 and remains acceptable for the MUR power uprate.
N/A	Flooding	102%	X		As discussed above, the various analyses applicable to flooding have been performed at 102% of 2568 MWt. The analyses of record are provided in Reference 2.3.1 and remain acceptable for the MUR power uprate.
N/A	Natural Circulation Cooldown	N/A	N/A		The natural circulation cooldown time will increase slightly based upon the power uprate from 2568 MWt to 2609 MWt. The time to cool the plant to 200°F will increase from 68.54 hrs to 70.38 hrs. This is still less than the 72 hour Appendix R requirement.

3.0 Accidents and Transients for Which the Existing Analyses of Record do not Bound Plant Operation at the Proposed Uprated Power Level. (RIS 2002-03 Section III Questions)

1. This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding).

2. For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:

   A. Identify the transient/accident that is the subject of the analysis

   B. Provide an explicit commitment to re-analyze the transient / accident, consistent with the reload methodology, prior to implementation of the power uprate

   C. Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59

   D. Provide a reference to the NRC's approval of the plant's reload methodology

3. For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should:

   A. Identify the transient or accident that is the subject of the analysis

   B. Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate

   C. Confirm that the limiting event determination is still valid for the transient or accident being analyzed

   D. Identify the methodologies used to perform the analyses, and describe any changes in those methodologies

   E. Provide references to staff approvals of the methodologies in Item D. above

   F. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology

   G. Describe the sequence of events and explicitly identify those that would change as a result of the power uprate

   H. Describe and justify the chosen single-failure assumption

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head (NPSH), valve isolation capabilities) required to support the analysis

K. Discuss the results and acceptance criteria for the analysis, including any changes from the previous analysis

3.1 Response to RIS 2002-03 Section III Questions

All analyses of record bound the MUR power uprate except LOCA mass and energy release – containment response and Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC), which are discussed below and in Attachments F and G.

3.1.1 LOCA Mass and Energy Release – Containment Response

The LOCA Mass and Energy Release and containment response were reanalyzed for the CR-3 MUR. The new analyses followed the NRC-approved methodology detailed in BAW-10252P-A (Reference F.1). The blowdown mass and energy release data were generated with the RELAP5/MOD2-B&W computer code (Reference F.2) and the containment pressure and temperature responses were generated with GOTHIC (Reference F.3). A more detailed discussion is provided as Attachment F. The initial re-analysis was performed with input parameters the same as in the current analysis of record and the results exceeded the current design limit of 69.7 psia. (55.0 psig). The analysis was then re-performed using acceptable but slightly less conservative input parameters. This analysis predicted a peak containment pressure of 68.74 psia (54.04 psig). This compares favorably to the current calculated peak of 68.9 psia (54.2 psig) reported in the FSAR against the current design limit of 69.7 psia (55 psig).

3.1.2 AMSAC Analysis

To ensure that the AMSAC system arming setpoint remains valid for the MUR power level, a new analysis was performed based on a core power level of 52% of 2609 MWt. The limiting ATWS transient for the B&W-designed plant is a loss of feedwater initiated event. The nominal plant setpoint is 50% power. At a lower value, the AMSAC system would not be armed. The purpose of the transient is to demonstrate that the without AMSAC, the peak pressure will not exceed 3250 psia. A more detailed discussion is provided in Attachment G of this report.

4.0 Mechanical/Structural/Material Component Integrity and Design (RIS 2002-03 Section IV Questions)

A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.

A. This discussion should address the following components:
   i. reactor vessel, nozzles, and supports
   ii. reactor core support structures and vessel internals
   iii. control rod drive mechanisms
   iv. Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles
   v. balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, and containment systems)
   vi. steam generator tubes, secondary side internal support structures, shell, and nozzles
   vii. reactor coolant pumps
   viii. pressurizer shell, nozzles, and surge line
   ix. safety-related valves

B. The discussion should identify and evaluate any changes related to the power uprate in the following areas:
   i. stresses
   ii. cumulative usage factors
   iii. flow induced vibration
   iv. changes in temperature (pre- and post-uprate)
   v. changes in pressure (pre- and post-uprate)
   vi. changes in flow rates (pre- and post-uprate)
   vii. high-energy line break locations
   viii. jet impingement and thrust forces

C. The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel with respect to:
   i. pressurized thermal shock calculations
   ii. fluence evaluation
   iii. heatup and cooldown pressure-temperature limit curves
   iv. low-temperature overpressure protection
   v. upper shelf energy
   vi. surveillance capsule withdrawal schedule

D. The discussion should identify the code of record being used in the associated analyses, and any changes to the code of record.

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.

F. The discussion should address whether the effect of the power uprate on steam generator tube high cycle fatigue is consistent with NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988.

4.1 Response to RIS 2002-03 Section IV Questions

Table 4-1 (Reference 4.3.1) contains a summary of changes in operating conditions as a result of the MUR. As can be seen from Table 4-1, there are only minor changes in operating conditions resulting from the uprate at the current OTSG plugging limit.

Table 4-1 Change in Operating Conditions for MUR

Parameter	Case A	Case B	Case C
Core Thermal Power (MWt)	2568	2609	2609
Other RCS Power (MWt) (RCP-LD)	16	16	16
Total NSSS Power (MWt)	2584	2625	2625
Pressurizer Control Pressure (psig)	2155	2155	2155
SG A/B Tube Plugging %	2.4%/5.7%	2.4%/5.7%	20%/20%
Thot (°F)	601.9	602.2	602.9
Tcold (°F)	556.2	555.8	555.1
Tavg (°F)	579	579	579
RCS Mass Flow (E6 lb_m/hr)	144.05	144.08	139.81
RCS Vol. Flow (gpm)	386,873	386,729	374,896
Steam Temperature (°F)	591.0	590.5	580.7
Steam Superheat (°F)	54.9	54.4	44.6
Feedwater/Steam Flow Rate (E6 lb/hr)	10.86	11.07	11.19
OTSG Steam Pressure (psia) **(Input)**	931.7	931.7	931.7
Feedwater Temperature (°F) **(Input)**	456.7	458.4	458.5

Case A Existing Tube Plugging at 2568 MWt
Case B Existing Tube Plugging at 2609 MWt
Case C 20 Percent Tube Plugging at 2609 MWt

4.2.1 Effect of Power Uprate on Major Components (RIS 2002-03 Section IV.1.A)

4.2.1.1 Reactor Vessel Structural Evaluation (RIS 2002-03 Section IV.1.A.i)

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor vessel. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor vessel remain applicable for the uprated power conditions. (Reference 4.3.6).

4.2.1.2 Reactor Vessel Internals Structural Evaluation (RIS 2002-03 Section IV.1.A.ii)

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor vessel internals. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. (Reference 4.3.6)

4.2.1.3 Fuel Assembly

The Crystal River 15x15 Mark-B fuel design was evaluated to determine the impact of the power uprate on the fuel assembly structural integrity. Since the core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, there is no impact on the fuel assembly seismic/LOCA structural evaluation. The power uprate does not increase operating and transient loads such that they will adversely affect the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected, and the seismic and LOCA evaluations of the 15x15 Mark-B fuel design are still applicable for the power uprate.

4.2.1.4 Control Rod Drive Mechanism Structural Evaluation RIS Section 2002-03 Section IV.1.A.iii)

The revised design conditions were reviewed for impact on the existing design basis analyses for the control rod drive mechanisms. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the control rod drive mechanism remain applicable for the uprated power conditions. (Reference 4.3.6)

4.2.1.5 Reactor Coolant Piping and Supports Structural Evaluation (RIS 2002-03 section IV.1.A.iv)

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor coolant piping and supports. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor coolant piping and supports remain applicable for the uprated power conditions. (Reference 4.3.6)

4.2.1.6 Balance-of-Plant (BOP) Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems) (RIS 2002-03 Section IV.1.A.v)

The structural analyses of the piping attached to the RCS (decay heat line, makeup/high pressure injection lines) use anchor motions from the RCS structural analyses. As discussed in Section 4.2.1.4, these anchor motions do not change due to the uprated power conditions. The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor coolant system attached piping and supports. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid.

The revised design conditions were reviewed for impact on the existing design basis analyses for the main steam and main feedwater piping and supports. No significant changes in OTSG design or operating pressure were made as part of the power uprate. The changes in the operating temperatures and flow rates due to the MUR power uprate have been evaluated. These changes were determined to have a negligible effect on the existing design basis analyses. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. (Reference 4.3.6)

4.2.1.7 Steam Generator Tubes, Secondary Side Internal Support Structures, Shell and Nozzles (RIS 2002-03 Section IV.1.A.vi)

The revised design conditions were reviewed for impact on the existing design basis analyses for the steam generator. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the steam generator remain applicable for the uprated power conditions.

Topical report BAW-10146 (Reference 4.3.2) established the minimum required steam generator tube wall thickness for the B&W 177-FA plants. Tube loads were calculated for normal operating and faulted conditions. Normal operating tube loads were determined using design operating transients and were combined with tube geometry to calculate minimum allowable tube wall thickness that satisfy the acceptance criteria of NRC Draft RG 1.121. Faulted condition tube loads are those arising from a safe shutdown earthquake, a loss of coolant accident, a main steam line break (MSLB) and a feedwater line break (FWLB). These loads were used to calculate minimum wall thickness based on the limits of NRC Draft RG 1.121 and ASME Code, Section III, Appendix F. The MUR Power Uprate Program operating conditions were compared with the existing design conditions. The comparison showed that the power uprate by itself will not result in operation outside the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the minimum required tube wall thickness for normal operating conditions will not be affected by the power uprate.

Tube loads for the faulted conditions were calculated for LOCA, MSLB, and FWLB accident conditions considering thermal and pressure loads on the steam generator. The MUR Power Uprate Program operating temperatures were compared with the existing design temperatures. The comparison showed that the existing design temperatures bound the power uprate temperatures. This means that the existing tube loads due to LOCA, MSLB and FWLB will not change as a result of the power uprate.

In addition, a review of calculations performed which assessed the integrity of tubes containing flaws of various types when subjected to operating and accident loads was conducted. This review ensured that existing structural margins are maintained for the MUR Power Uprate Program design conditions. (Reference 4.3.6)

4.2.1.8 Reactor Coolant Pump Structural Evaluation (RIS 2002-03 Section IV.1.A.vii)

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor coolant pumps. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor coolant pumps remain applicable for the uprated power conditions. (Reference 4.3.6)

4.2.1.9 Pressurizer Structural Evaluation (RIS 2002-03 Section IV.1.A.viii)

The revised design conditions were reviewed for impact on the existing design basis analyses for the pressurizer. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the pressurizer remain applicable for the uprated power conditions. (Reference 4.3.6)

4.2.1.10 Safety-Related Valves (RIS 2002-03 Section IV.1.A.ix)

The revised design conditions were reviewed for impact on the existing design basis analyses for the safety-related valves. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Safety analysis confirmed the installed capacities and lift setpoints of the RCS and Main Steam relief valves to be valid for the MUR Conditions. Therefore, the existing loads remain valid and the stresses and fatigue values also remain valid. Safety-related valves were reviewed within the system and program evaluations. None of the safety-related valves required a change to their design or operation as a result of the MUR. (References 4.3.9 and 4.3.10)

4.2.2 Effect of Power Uprate on Stresses, Operating Conditions, and HELB (RIS 2002-03 Section IV.1.B)

4.2.2.1 Stresses (RIS 2002-03 Section IV.1.B.i)

The revised design conditions were reviewed for impact on the existing design basis. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The stress reports including the tabulation of maximum stress intensities/stress ranges with a comparison to stress allowables, cumulative usage factors, and other special stress limits were reviewed. The MUR power uprate conditions are bounded by the design conditions. (Reference 4.3.6)

4.2.2.2 Cumulative Usage Factors (RIS 2002-03 Section IV.1.B.ii)

The revised design conditions for the NSSS components, piping and interface systems were reviewed for impact on the existing design basis analyses. For NSSS components, the evaluation showed that the operating conditions due to the MUR uprate are bounded by those used in the existing analyses. Further, since the evaluated transients listed in FSAR Table 4-8 will not change as a result of the power uprate, the existing loads remain valid and the stresses and fatigue values (cumulative usage factors) also remain valid. (Reference 4.3.6)

4.2.2.3 Flow Induced Vibration (FIV) (RIS 2002-03 Section IV.1.B.iii)

As shown in Table 4-1, the RCS flow rate changes insignificantly compared to the RCS flow rate prior to the uprate for the same steam generator tube plugging conditions. These flow rates were evaluated against flow rates used in Topical Report BAW-10051 (Reference 4.3.3), which presents the design analysis of the RV internals and incore instrument nozzles subjected to operational flow-induced vibration loading for the B&W 177-FA plants. A comparative analysis was performed to evaluate the effects of the operating conditions. This evaluation concluded that those components remain structurally adequate for the observed flow conditions. (Reference 4.3.6)

An evaluation was performed (Reference 4.3.13) that concluded that there currently exists a minimum of 13.6% margin against detrimental effects inside the OTSG due to flow induced vibrations for the 2609 MWt uprate considering 20% tube plugging. The limiting FIV mechanism is turbulence and the resulting mid-span tube impacts.

4.2.2.4 Changes in Temperature (Pre- and Post-Uprate) (RIS 2002-03 Section IV.1.B.iv)

4.2.2.4.1 Temperature Changes

The changes in operating temperatures are provided in Table 4-1. The average temperature is unchanged and the cold leg decreases 0.4°F while the hot leg temperature increases 0.3°F. These changes as discussed elsewhere have minimal impact on the MUR.

4.2.2.4.2 Evaluation of Potential for Thermal Stratification

Thermal stratification in the lines attached to the primary side of the RCS occurs mainly during heatup and cooldown. The 100% power hot and cold leg temperatures that the plant has been designed to are essentially the same as those for the MUR Power Uprate Program. This means that the effects of thermal stratification will not change as a result of the power uprate.

NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems", addresses the issue of thermal stresses in piping attached to the primary loop that cannot be isolated. The temperature changes as a result of the MUR Power Uprate Program compared to current operation are negligible and will not have an effect on existing or potential thermal stratification conditions. In addition, the design RCS flow rates are essentially the same as those for the MUR Power Uprate Program and thus the effects of turbulent penetration will not change as a result of power uprate.

NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," addresses the issue of surge line thermal stratification. Thermal stratification in the surge line occurs mainly during plant heatup and cooldown and is driven by the temperature difference between the hot leg and the pressurizer. The current operating temperature of the hot leg will increase very slightly due to the MUR Power Uprate Program. A higher hot leg temperature gives a lower temperature differential between the hot leg and the pressurizer which in turn lessens the stratification effects. This means that stress and fatigue in the surge line which is attributed to thermal stratification is bounded by the existing analyses. (Reference 4.3.6)

4.2.2.5 Changes in Pressure (Pre- and Post-Uprate) (RIS 2002-03 Section IV.1.B.v)

The changes in operating pressures are provided in Table 4-1. As discussed in Section 2.2 the accident analyses is unchanged. The RCS pressure and pressurizer pressure control setpoint remains the same.

4.2.2.6 Changes in Flow Rates (Pre- and Post-Uprate) (RIS 2002-03 Section IV.1.B.vi)

The changes in RCS flow are provided in Table 4-1. The MUR power uprate does not have an appreciable effect on RCS mass flow (<0.1%). Therefore, the changes in mass flow rates (pre- and post-uprate) will have a negligible impact on core design and safety analyses.

4.2.2.7 High Energy Line Break Locations (HELB) (RIS 2002-03 Section IV.1.B.vii)

An engineering evaluation was performed (Reference 4.3.12) which evaluated the impacts of HELB systems inside and outside containment at CR-3. High energy piping is defined as piping carrying fluid above 275 psig and 200°F inside containment and above 275 psig and/or 200°F outside containment. The HELB evaluations were performed at 2619 MWt to bound the expected range of operation resulting from the MUR uprate.

There are no HELB impacts on the systems reviewed inside containment, nor for flooding inside containment. For high energy systems reviewed outside containment, there are no outliers as a result of the proposed MUR uprate. Flooding events outside containment in the Intermediate Building and the Auxiliary Building are not affected by the uprate.

There were no new line breaks postulated for current HELB systems inside or outside containment as pressures and temperatures did not increase. There are no new systems inside or outside containment that qualify as HELB systems as a result of the uprate.

4.2.2.8 Leak-Before-Break Evaluation

The Leak-Before-Break (LBB) concept applies known mechanisms for flaw growth to piping designs with assumed through-wall flaws and is based on the plant's ability to detect an RCS leak. Topical Report BAW-1847, Rev. 1, (Reference 4.3.4) presents the LBB evaluation of the RCS primary piping. It showed that a double-ended guillotine break will not occur and that postulated flaws producing detectable leakage exhibit stable growth, and thus, allow a controlled plant shutdown before any potential exists for catastrophic piping failure. The major areas that contributed to this evaluation were: RCS piping structural loads; leakage flaw size determination; flaw stability analysis; and, RCS piping material properties. An evaluation was performed which determined the impact of the MUR uprate design conditions on the inputs to the LBB analyses is negligible and the LBB conclusions remain unchanged. (Reference 4.3.6)

4.2.2.9 Reactor Coolant System Loss of Coolant Accident Forces Evaluation

Topical report BAW-1621 (Reference 4.3.5) addresses the RCS components for primary break Loss-of-Coolant Accident (LOCA) loadings. The breaks considered were limited break ruptures of the primary piping. Due to LBB qualification of the hot and cold legs, the RCS was requalified for snubber removal. The MUR Power Uprate Program design conditions were reviewed for impact on the existing hydraulic forcing functions and the HELB locations in the primary RCS piping and the piping attached to the primary RCS to the first anchor. The evaluation showed that the asymmetric cavity pressure forces, thrust loads, and jet impingement loads remain bounded by the values in the existing analyses. The evaluation also showed that there are no additions or changes to the HELB locations or loads. (Reference 4.3.6)

4.2.3 Effect of Power Uprate on Reactor Vessel Integrity (RIS 2002-03 Section IV.1.C)

4.2.3.1 Pressurized Thermal Shock (PTS) (RIS 2002-03 Section IV.1.C.i)

The reference temperature for pressurized thermal shock (RT_{PTS}) values in support of a power uprate applicable to the projected end-of-life period (32 EFPY) for the reactor vessel beltline materials were re-evaluated. These values were calculated in accordance with the requirements in 10 CFR 50.61. A 7% increase in 32 EFPY neutron fluence was used to bound the effects of the MUR power uprate on RT_{PTS}. The limiting reactor vessel beltline material has a RT_{PTS} value of 206°F at 32 EFPY. The screening criterion for this weld metal is 270°F. Therefore, the reactor vessel will remain within its limits for PTS after the MUR power uprate. (Reference 4.3.7)

4.2.3.2 Fluence Evaluation (RIS 2002-03 Section IV.1.C.ii)

The impact of a MUR power uprate on the high energy neutron leakage (neutrons with energies greater than 1.0 million electron volts (MeV) or E > 1.0 MeV) from the core to the internals and reactor pressure vessel will be minimal. The neutron leakage directly impacts the pressurized thermal shock criteria, the pressure - temperature (P-T) limits (including those for low temperature over pressurization), and the baffle bolts or other internals. A 7% increase in 32 EFPY end of life fluences was used to bound the effects of the MUR power uprate. Clearly, the assumed 7% increase in neutron fluence conservatively bounds the actual anticipated increase of the reactor thermal power of 2% or less based on the MUR power uprate. (Reference 4.3.7)

4.2.3.3 Heatup and Cooldown Pressure / Temperature Limit Curves (RIS 2002-03 Section IV.1.C.iii)

The current P-T limit curves are licensed through 32 effective full power years (EFPY) and are based on adjusted reference temperatures at the ¼-thickness (¼T) and ¾-thickness (¾T) wall locations for the limiting reactor vessel beltline material. Adjusted reference temperature (ART) values were calculated in accordance with Regulatory Guide 1.99, Revision 2. Inputs affecting the adjusted reference temperatures and P-T curves remain unchanged under the MUR power uprate, with the exception of neutron fluence. Changes to the core power level will affect neutron flux, which will affect neutron fluence, and could have ultimately affected the validity period of the current P-T curves.

The impact of the MUR power uprate on the P-T curves was assessed by performing a revised 32 EFPY ART calculation in accordance with Regulatory Guide 1.99, Revision 2, which considered recent reactor vessel surveillance data and an assumed 7% increase in 32 EFPY fluence due to a power uprate. The assumed 7% increase in neutron fluence conservatively bounds the actual anticipated increase of neutron fluence at 2% or less based on the MUR power uprate. Based on the additional credible reactor vessel surveillance data, the chemistry factors utilized in the ART calculations were reduced leading to an overall reduction in ART at 32 EFPY. The limiting ART values at ¼T and ¾T were reduced from 213°F and 144.5°F to 195.7°F and 144.1°F, respectively. Therefore, the existing 32 EFPY P-T curves and Low Temperature Overpressure Protection (LTOP) limits remain valid for the MUR power uprate. (Reference 4.3.7)

4.2.3.4 Low-Temperature Overpressure Protection (RIS 2002-03 Section IV.1.C.iv)

As described above, the current LTOP limits in the 32 EFPY P-T curves do not need to be modified for the MUR. (Reference 4.3.7)

4.2.3.5 Effect on Low Upper Shelf Energy (RIS 2002-03 Section IV.1.C.v)

Due to the increase in fluence from a power uprate, low upper-shelf toughness was evaluated to ensure compliance with Appendix G to 10 CFR Part 50. If the limiting reactor vessel beltline material's Charpy upper-shelf energy (USE) is projected to fall below 50 ft-lb, an equivalent margins assessment must be performed. The limiting reactor vessel beltline materials for CR-3 are welds WF-70 (upper shell to lower shell circumferential weld) and WF-8/WF-18 (upper shell longitudinal weld).

An equivalent margin assessment was performed for these welds in a 1994 B&W Owners Group generic analysis. These welds were evaluated for ASME Boiler & Pressure Vessel Code Levels A, B, C, and D Service Loadings based on the evaluation acceptance criteria of ASME Section XI, Code Case N-512, which later became ASME Section XI, Appendix K.

The analysis demonstrated that the limiting reactor vessel beltline welds at CR-3 satisfy the ASME Code requirements of ASME Code Case N-512 (ASME Section XI, Appendix K) for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 32 EFPY considering a fluence which bounds the MUR power uprate. (Reference 4.3.7)

4.2.3.6 Surveillance Capsule Withdrawal Schedule (RIS 2002-03 Section IV.1.C.vi)

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. FPC has completed withdrawal of capsules for CR-3. As discussed above, projections based upon these withdrawals has been factored into fluence calculations and have demonstrated acceptable operation through 32 EFPY.

4.2.4 Code of Record Used in Associated Analyses (RIS 2002-03 Section IV.1.D)

No new structural or fluence analyses were performed. Analyses and codes of record remain unchanged except as discussed in Section 3.0.

4.2.5 Impact of Uprate on Inspection and Testing Programs Including Erosion/Corrosion Programs (RIS 2002-03 Section IV.1.E)

4.2.5.1 Alloy 600 Primary Water Stress Corrosion Cracking (PWSCC)

The effects of an RCS temperature increase resulting from the power uprate on Alloy 600 PWSCC have been evaluated. For the limiting case of 20% OTSG tube plugging, it is estimated that the increase of T_{hot} from 601.7°F to 603.3°F decreases the time to PWSCC initiation by 6% and increases the crack growth rate by 4%. Because the power uprate does not increase the T_{cold} and T_{avg}, or the RCS pressure and T_{sat}, the impact is limited to Alloy 600 components and welds operating near T_{hot}. Examination of the AREVA NP Alloy 600 ranking model shows that the current relative PWSCC ranking of Alloy 600 components will not change after the power uprate. The current top three most PWSCC susceptible components are all in the pressurizer, and these components continue to be the most susceptible after the power uprate. Hence, the impact of the power uprate on Alloy 600 PWSCC is considered very limited and addressed by current CR-3 aging management programs for Alloy 600.

4.2.5.2 Inservice Testing (IST) Program

10 CFR 50.55a(f), "Inservice Testing Requirements," requires the development and implementation of an Inservice Testing (IST) Program. CR-3 has developed, and is implementing an Inservice Testing (IST) Program for Pumps and Valves per the applicable requirements. This evaluation reviewed the impact to the Inservice Testing Program as part of the MUR uprate conditions up to the original licensed reactor thermal power of 2609 MWt and concluded that the MUR uprate is bounded by current analysis and any changes are insignificant.

4.2.5.3 Inservice Inspection (ISI) Program

10 CFR 50.55a(g), "Inservice Inspection Requirements," requires the development and implementation of an Inservice Inspection (ISI) Program. The applicable program requirements

are specified in ASME Section XI. CR-3 has developed and is implementing an Inservice Inspection (ISI) Program per these requirements. This evaluation evaluated the impact to the Inservice Inspection Program as part of the MUR uprate conditions up to of the original licensed reactor thermal power 2609 MWt and concluded that the MUR uprate is bounded by current analysis and no changes are required.

4.2.5.4 Erosion / Corrosion (FAC) Program

The CR-3 FAC model has been revised to reflect the 1.6% MUR conditions. Therefore, the predicted increases in maximum component wear rates and reductions in service lives will be managed by the CR-3 FAC program. The most limiting piping segment is in the feedwater system. It was explicitly re-evaluated based on the revised model. The results support continued operation until its scheduled replacement concurrent with steam generator replacement.

4.2.6 Impact of NRC Bulletin 88-02 "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes" and NRC Information Notice 2002-02 (including Supplement 1) "Recent Experience with Plugged Steam Generator Tubes" Upon the CR-3 MUR Power Uprate (RIS 2002-03 Section IV.1.F)

NRC Bulletin 88-02 implements actions to be taken by the holders of operating licenses of Westinghouse Replacement Steam Generator designs (Specifically models 13, 27, 44, 51, D1, D2, D3, D4 and E) to minimize the potential for a steam generator tube rupture event caused by rapidly propagating fatigue cracks such as occurred at North Anna Unit 1 on July 15, 1987. The tube rupture occurred in the ubend region of a row 9 tube at the top Tube Support Plate (TSP). The cause of the rupture was high cycle fatigue. The source of the loads was a combination of a high mean stress level in the tube and a superimposed alternating stress. The mean stress was produced by denting of the tube at the upper most TSP and the alternating stress was the result of out-of-plane defection of the ubend portion of the tube above the uppermost support plate caused by flow-induced vibration.

The most significant contributors to this occurrence was a high fluid-elastic stability ratio (not margin as addressed in this document) resulting from a reduction in damping at the tube-to-tube support plate intersection caused by denting and a locally high flow velocity caused by non-uniform anti-vibration bars penetrations into the u-bend tube bundle region.

Since the NRC Bulletin 88-02 is not applicable to the OTSG designs, there is no impact upon the Appendix K power uprate and no action is required. A more relevant NRC Generic Communication for OTSG designs to consider would be Information Notice 2002-02, "Recent Experience with Plugged Steam Generator Tubes," dated January 2002 and July 2002 for Supplement 1. EPRI Topical Report 1008438, "Three Mile Island Plugged Tube Severance (A Study of Damage Mechanism)," addresses the concerns identified with Information Notice 2002-02.

The results and findings of the EPRI Report 1008438 concluded that certain types of tube degradation can continue to occur in any steam generator after the tube has been taken out of service. For the B&W OTSGs, it was concluded that the only real vulnerability for tube severance is the growth of circumferential cracks due to high cycle fatigue. However, for a swollen, plugged tube, any degradation mechanism has the potential to provide an initiating site for failure.

In response to the findings, AREVA NP has implemented steam generator plugging and de-plugging maintenance procedures that will prevent such incidences from occurring in the future and CR-3 has complied with these and all other recommendations to mitigate the consequences of over-pressurized tubes in the OTSGs. To address tubes that were plugged prior to NRC Information Notice 2002-02 that may be susceptible to tube swelling, CR-3 has plugged and stabilized all of the adjacent/neighboring tubes.

To address the possibility of circumferential tube cracks eventually severing due to high cycle fatigue, the OTSG stabilization criteria have historically required stabilization of all circumferential crack-like indications regardless of the radial location or elevation. In addition, the OTSG stabilization criteria have historically required stabilization of circumferentially-oriented volumetric indications in regions of high cross flows. Therefore, the finding of the EPRI Report 1008438 have always been employed for these degradation types.

Therefore, there are no Flow Induced Vibration concerns related to the tube bundle associated with the Appendix K power uprate relevant to findings provided by NRC Information Notice 2002-02 or the EPRI Report 1008438 that have not already been evaluated in this and earlier revisions of this document. (Reference 4.3.8)

4.3 Section 4.0 References

4.3.1 32-5012972-001, "CR-3 Power Uprate Operating Conditions."
4.3.2 Topical Report BAW-10146, "Determination of Minimum Required Tube Wall Thickness for 177-FA Once-Through Steam Generators," November 1980.
4.3.3 Topical Report BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration," September 1972.
4.3.4 Topical Report BAW-1847, Rev. 1, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985.
4.3.5 Topical Report BAW-1621, "Effects of Asymmetric LOCA Loads – Phase II," July 1980.
4.3.6 51-9036246-000, "CR-3 MUR Power Uprate RCS Structural Assessment."
4.3.7 51-9040378-000, "CR-3 MUR RV Integrity Summary."
4.3.8 51-5000475-002, "CR-3 OTSG FIV Margins."
4.3.9 51-9041016-000, "CR-3 MUR Power Uprate (2609 MWt) Mechanical Systems Review."
4.3.10 51-9036887-000, "CR-3 MUR Summary Report."
4.3.11 Amendment No. 205 to the Crystal River Unit 3 Facility Operating License (TAC NO. MB 5289), dated December 4, 2002
4.3.12 51-9046833-000, "CR-3 MUR 101.6% HELB Evaluation"
4.3.13 51-5000475-002, "CR-3 OTSG FIV Margins"

5.0 Electrical Equipment Design (RIS 2002-03 Section V Questions)

1. A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:

   A. emergency diesel generators

   B. station blackout equipment

   C. environmental qualification of electrical equipment

   D. grid stability

   E. transformers

5.1 Response to RIS 2002-03 Section V Questions

5.2.1 Emergency Diesel Generators (RIS 2002-03 Section V.1.A)

The emergency diesel generator system (System Code – EG) provides emergency electrical power for the plant Engineered Safeguards (ES) plus selected balance of plant emergency loads. Margin currently exists on each emergency diesel generator (EGDG-1A and EGDG-1B) and the alternate AC diesel. The uprate will not change the loading of the emergency diesel generators or the alternate AC diesel. Therefore, EG System equipment capacity and capability for plant operation under MUR power uprate conditions are bound by the generator loading tables which are supported by the existing analysis of record. As a result, the EG System will continue to have adequate capacity and capability to operate the plant equipment. Relative to the EG System, there are no changes to plant technical specifications, protection system settings, and/or emergency system settings needed to support the MUR power uprate. (Reference 5.3.1)

The alternate AC diesel, with its separate fuel supply, has the capability of being aligned to either safety-related AC distribution bus. This provides additional assurance that AC power remains available. The alternate AC diesel provides defense in depth and this diesel was not credited in the Station Blackout analysis.

5.2.2 Station Blackout Equipment (RIS 2002-03 Section V.1.B)

The DC power system (System Code – DP) supplies required and expected loads (during the 4 hour load profile) in the event of a Station Blackout. The MUR uprate will have no impact on the design of or the loads supplied from the DP System. Therefore, DP System equipment capacity and capability for plant operation under MUR power uprate conditions are bound by the load profiles which are supported by the existing analysis of record. As a result, the DP System will continue to have adequate capacity and capability to operate the plant equipment. Relative to the DP System, there are no changes to plant technical specifications, protection system settings, and/or emergency system settings needed to support the MUR power uprate. (Reference 5.3.1)

5.2.3 Environmental Qualification of Electrical Equipment (RIS 2002-03 Section V.1.C)

The limiting post-accident reactor building conditions were demonstrated to not be increased as part of the LOCA Mass and Energy and Reactor Building analyses described earlier. Thus, the accident profile to which equipment is qualified remains bounding.

5.2.4 Grid Stability (RIS 2002-03 Section V.1.D)

It should be noted that grid stability is somewhat less of a nuclear safety concern for CR-3 than most other plants since off-site power is supplied from the 230 kV system and the CR-3 output is to the 500 kV system, which are not locally interconnected. Nevertheless, the grid's stability is being thoroughly evaluated to address the impacts of this and planned subsequent uprates to CR-3 as well as the potential impact of new generation sited relatively nearby. Preliminary results of that evaluation indicate that the impact of the MUR are negligible. When the formal analysis is completed the results and report will be forwarded to the NRC.

5.2.5 Station Auxiliary Electric Power Distribution System

The AC power system (System Code – AC) will experience minor load changes (additions) as a result of the MUR uprate. The installation of the Caldon equipment by Engineering Change (EC) 65626 and additional Main Steam pressure and temperature instrumentation by EC 65629 will add negligible loads which will be addressed in the modification documents. Condensate pump motor load will increase slightly but remain within the design rating of the motor, associated electrical components and protective relay settings. Feedwater booster pump motor load will increase slightly; however, the motor power required at the uprate condition will remain well within design. Therefore, the AC System will continue to have adequate capacity and capability to operate the plant equipment. Relative to the AC System, there are no changes to plant technical specifications, protection system settings, and/or emergency system settings needed to support the MUR power uprate. (Reference 5.3.1)

5.2.6 Step-Up and Auxiliary Transformers

The Main Power Transformer (Step-Up) is being replaced during the upcoming refueling outage (RFO-15). At that time the Step-Up Transformer will have the nominal rating of 1200 MVA. The transformer will have more than enough capability to accept the approximately 14 MVA from the MUR uprate. Current Step-Up transformer rating is approximately 950 MVA, while current load is approximately 900MWe. The Unit Auxiliary Transformer is capable of handling full in-house loads before and after uprate.

5.3 Section 5.0 References

5.3.1 51-9037444-000, "CR-3 MUR Power Uprate (2609 MWt) BOP Electrical Systems Review."

6.0 System Design (RIS 2002-03 Section VI Questions)

1. A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems:

   A. NSSS interface systems for pressurized-water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling-water reactors (BWRs) (e.g., suppression pool cooling), as applicable

   B. containment systems

   C. safety-related cooling water systems

   D. spent fuel pool storage and cooling systems

   E. radioactive waste systems

   F. Engineered safety features (ESF) heating, ventilation, and air conditioning systems

6.1 Response to RIS 2002-03 Section VI Questions

A comparison between operating requirements for the 2609 MWt MUR conditions generated by the PEPSE heat balance and the 2568 MWt heat balance conditions demonstrates that the major plant systems that meet the requirements identified in 4.1 above and discussed below have sufficient design and operational margin to accommodate the MUR uprate.

A review of the FSAR (Reference 6.3.1) Chapter 14 accidents was performed to determine if the analyses of record for CR-3 remained applicable and bounding for the power uprate. The results of this review are described herein. It was concluded that the existing analyses as described in the FSAR were performed with a bounding core power level or the consequences for a given event were bounded by other analyses presented in the FSAR. Based on the results of this review and in order to ensure protection for the high flux analytical limit of 2876 MWt, the reactor over power limit value will be adjusted to reference the new Reactor Thermal Power value of 2609 MWt. This will require a change from the current value of 112% RTP (2568 MWt) to 110.2% RTP (2609 MWt). This change maintains the reactor trip at the same net power level modeled in the safety analysis. After accounting for Measurement Uncertainty, instrumentation, and process errors, with the reduced heat balance uncertainty using the LEFM CheckPlus™ system, the Technical Specification RPS High Flux trip allowable value can be maintained at 104.9% of the new RTP with four reactor coolant pumps operating. These limits are applicable when power range nuclear instrumentation is verified consistent with the heat balance results calculated using the LEFM system for feedwater flow measurement. Specific requirements for operation with inoperable LEFMs will be placed in CR-3 procedure CP-500.

The individual systems are discussed in more detail below.

6.2.1 NSSS Interface Systems (RIS 2002-03 Section VI.1.A)

6.2.1.1 Main Steam (MS) System

The MS System performs the following safety functions; provides automatic isolation of the OTSGs after a steam line failure, provides overpressure relief capacity in the event of accidents, provides pressure control for decay heat removal in case of accidents, provides steam to the Emergency Feedwater (EF) System as required for accidents and provides capability for RCS cooldown following a steam generator tube rupture event. As discussed in Sections 4 and 5 above, the MS system will support the MUR and the safety functions of this system are not impacted by the uprate. The MS System also functions during normal operation. While steam flow increases with increasing power no changes in design are required and all parameters remain within design. (Reference 6.3.2)

6.2.1.2 Steam Dump

The CR-3 equivalent of a steam dump system includes the Atmospheric Dump Valves (ADVs) and Turbine Bypass Valves (TBVs).

6.2.1.2.1 Atmospheric Dump Valves

An ADV is located in each of the two Main Steam Lines, upstream of the MSIVs (MSV-25 in Steam Line A-1 and MSV-26 in Steam Line B-2). The valve function is to provide a controlled path for venting of main steam to the atmosphere. These valves were evaluated for power uprate impact on three functions: (1) close to isolate containment; (2) open and modulate to relieve steam to the atmosphere; and, (3) maintain pressure boundary to transport steam to safety and non-safety related loads. There are no changes in function. Power uprate conditions are bounded by existing design. The evaluation concludes the functional performance requirements of the Main Steam ADVs will be unaffected by the power uprate. (Reference 6.3.2)

6.2.1.2.2 Turbine Bypass Valves

Four TBVs are located in the Main Steam lines downstream of the MSIVs. The Turbine Bypass Valves are piston-operated globe valves which actuate in response to a hand generated signal or an Integrated Control System (ICS) generated signal. The valves primary function is to maintain stable turbine header pressure during load swing events. The flow rate is not being changed and the function of the TBVs is not being changed. For the power uprate, the ICS control will use the existing TBVs. Power uprate parameters are bounded by existing design conditions at 102% (2619 MWt). There is no impact on the TBVs for the MUR power uprate. (Reference 6.3.2)

The MUR power uprate conditions remain bounded by the design basis of the CR-3 FSAR.

6.2.1.3 Condensate (CD) System

The primary function of the CD system is to supply preheated condensate to the FW System. The Condensate system was evaluated for a power uprate from 2568 MWt to 2609 MWt. The uprate will have no impact on the design functions of the CD system. The condenser load limiting back pressure is 9 inches of mercury (" Hg) absolute and the current maximum operating pressure has been 3-4" Hg absolute. No design changes will be required. (Reference 6.3.2)

The feedwater heaters were evaluated and determined to be adequate for the 2609 MWt operating conditions. No operational changes are required. The Condensate system analysis of record is not impacted by the MUR power uprate. (Reference 6.3.2)

6.2.1.4 Main Feedwater (FW) System

The FW System provides isolation capability of the feedwater during accidents. It also provides feedwater to the OTSGs during normal operation. The CR-3 accident analyses are discussed in Sections 2 and 3 above. The safety functions of this system are not impacted by the uprate. In addition, the main feedwater pumps and the booster pumps have been determined to have adequate margin for a 1.6% power increase. The feedwater heaters were evaluated and determined to be adequate for the 2609 MWt operating conditions. No changes in design are required and all parameters remain within design. The FW system will therefore support a power uprate to 2609 MWt. (Reference 6.3.2)

6.2.2 Containment Systems (RIS 2002-03 Section VI.1.B)

The containment systems include the building spray system, penetrations and hatches. The building spray system's functions are to remove fission products from the post-accident containment atmosphere, and to assist in post-accident pressure and temperature control. The safety function of the penetrations and hatches is to maintain containment integrity under accident conditions. As indicated in Sections 2 and 3 above, the transients continue to be maintained within design limits. As such, these systems are not impacted by the MUR. (Reference 6.3.3)

6.2.3 Safety-Related Cooling Water Systems (RIS 2002-03 Section VI.1.C)

6.2.3.1 Decay Heat Closed Cycle Cooling (DC)

The DC System removes heat from the reactor core via the Low Pressure Injection/Decay Heat system as well as various pumps and motors following a LOCA and transfers it to the Raw Water (RW) system. The applicable CR-3 accident analyses were evaluated at 102% reactor thermal power and bound the 1.6% power increase (Sections 2 & 3). Therefore, the safety functions of this system are not impacted by the uprate. There are no design changes required. As such, this system is not impacted by the MUR. (Reference 6.3.2)

6.2.3.2 Nuclear Services Closed Cycle Cooling (SW)

The SW System removes heat from various safety-related equipment following ES actuation and transfers this heat to the RW system. The applicable CR-3 accident analyses were evaluated at 102% reactor thermal power and bound the 1.6% power increase (Sections 2 and 3). Therefore, the safety functions of this system are not impacted by the uprate. There are no design changes required. As such, this system is not impacted by the MUR. (Reference 6.3.2)

6.2.3.3 Nuclear Services & Decay Heat Seawater (RW)

The RW System provides cooling water to the SW and DC Systems for heat removal during accidents and normal operation. The CR-3 accident analyses were evaluated at 102% reactor thermal power and bound the 1.6% power increase (Sections 2 and 3). Therefore, the safety functions of this system are not impacted by the uprate. There are no design changes required. As such, this system is not impacted by the MUR. (Reference 6.3.2)

6.2.3.4 Emergency Feedwater System (EF)

The EF System provides emergency feedwater in the event of loss of main feedwater. The CR-3 accident analyses were evaluated at 102% reactor thermal power and bound the 1.6% MUR power uprate (Sections 2 & 3). There are no design changes required for the EF system to operate at 2609 MWt. There are no design changes required. As such, this system is not impacted by the MUR. (Reference 6.3.3)

6.2.4 Spent Fuel Pool Storage and Cooling Systems (RIS 2002-03 Section VI.1.D)

The principal function of the Spent Fuel (SF) system is to provide for the cooling and storage of irradiated fuel. The system is described Section 9.3 of Reference 6.3.1. The functions of the system were reviewed and were found to be unaffected by the MUR uprate. There are no design changes required. As such, this system is not impacted by the MUR. (Reference 6.3.3)

6.2.5 Radioactive Waste Systems (RIS Section 2002-03 Section VI.1.E)

The Waste Decay (WD) system provides the means to sample, collect, process, store/hold, re-use or release gaseous and liquid low-level effluents generated during normal operation. The WD system consists of the gaseous waste disposal (WD-GW) and the liquid waste disposal (WD-LW) sub-systems. These systems are discussed below.

6.2.5.1 Gaseous Waste Disposal (WD-GW)

The WD-GW system is used to control low-level gas releases to the environment, and to permit the venting of excess gas to the Reactor Building in a post-accident situation. Portions of the system are required to be operational and intact to provide containment isolation upon an Engineered Safeguard (ES) actuation signal. This system is unaffected by the MUR uprate. There are no design changes required. As such, this system will support the MUR. (Reference 6.3.3)

6.2.5.2 Liquid Waste Disposal (WD-LW)

The WD-LW system is required to collect, store and process, for disposal or reuse, radioactive liquid waste. The WD-LW system provides a means to process radioactive liquid waste prior to release. The WD-LW system tank volumes and processing components (i.e., demineralizers and filters) capacity are adequate to process radioactive liquid waste prior to release. The volume of liquid waste is primarily dependent on RC bleed and SG draindown as well as leakage from various components; the volume generated during normal operation is not expected to change due to the uprate. However, the activity of fission products in the liquid waste is dependent on the power level and will increase slightly due to the uprate. However, the impact of the higher activity on the operation of the WD-LW system will not be significant. The resins in the demineralizers may require replacement or regeneration at slightly higher frequencies, which would affect the volume of generated solid waste slightly, but this would not be a constraint to implementing the uprate. There are no design changes required. As such, this system will support the MUR. (Reference 6.3.3)

6.2.6 Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning Systems (RIS 2002-03 Section VI.1.F)

The ESF Heating, Ventilation, and Air Conditioning Systems remain bounded by the design basis (102% of 2568 MWt) of the CR-3 FSAR (Reference 6.3.1) for MUR power uprate conditions. The CR-3 accident analyses were evaluated at 102% reactor thermal power and bound the 1.6% power increase (Sections 2 and 3). Therefore, the safety functions of these systems are not impacted by the uprate. There are no expected changes in containment cooling operation at the MUR uprate power level. The containment accident analysis has been performed at a bounding power level with the containment air coolers and fan flow rates and found to be acceptable. The containment cooling system has adequate margin to cool the containment at MUR conditions. There are no design changes required for any of the ESF Heating, Ventilation, and Air Conditioning Systems. As such, these systems will support the MUR. (Reference 6.3.4)

6.3 Section 6.0 References

6.3.1 CR-3 Final Safety Analysis Report, Rev. 30.1.
6.3.2 51-9036486-000, "CR-3 MUR Power Uprate (2609 MWt) BOP Mechanical Systems Review."
6.3.3 51-9041016-000, "CR-3 MUR Power Uprate (2609 MWt) NSSS Mechanical Systems Review."
6.3.4 51-9036250-000, "Crystal River 3 MUR HVAC System Evaluation Report."

7.0 Other (RIS 2002-03 Section VII Questions)

1. A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.
1. A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that

are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:

A. emergency and abnormal operating procedures

B. control room controls, displays (including the safety parameter display system) and alarms

C. the control room plant reference simulator

D. the operator training program

2. A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate.

3. A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2 percent to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

4. A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:

   A. A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

   B. A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.

7.1 Response to RIS 2002-03 Section VII Questions

7.1.1 Operator Actions (RIS 2002-03 Section VII.1)

Operator actions that are sensitive to the power uprate, including any effects of the time available for operator actions are being reviewed. It is anticipated that the Operator Actions required to support the MUR uprate will be bounded and supported by the current analysis. It is anticipated that the power uprate will require no additional operator actions, that the additional time required to perform certain operator actions will have no adverse effects and that the time available for critical operator action has not been reduced. FPC will inform the NRC if there are any operator actions that change these conclusions.

7.1.2 Modifications Associated With the Power Uprate (RIS 2002-03 Section VII.2)

7.1.2.1 Emergency and Abnormal Operating Procedures (RIS 2002-03 Section VII.2.A)

Emergency and Abnormal Operating Procedures (EOPs and AOPs) will be reviewed for potential impact from the proposed power uprate. No adverse impact on these procedures with this power uprate is expected. Any EOP changes will be documented and implemented as part of the Engineering Change process.

7.1.2.2 Control Room Controls, Displays (including safety parameter display system) and Alarms (RIS 2002-03 Section VII.2.B)

The following changes/modifications are associated with implementation of the power uprate that affect control room controls:

- Power Production Heat Transfer will be increased slightly but within the ability of the operator to maintain prescribed parameters less than the required limits, thus having no adverse effect on defense in depth or safety margins.
- Cooldown time on Decay Heat is increased slightly, which increases the amount of time the operators must control the cooldown of the RCS. The increase in the amount of time Decay Heat is controlled is within the ability of the operators thus, having no adverse effect on defense in depth or safety margins.
- Changes to be made to the calibration of the nuclear instrumentation due to power uprate will be accommodated by corresponding changes being made to the Integrated Control System (ICS) due the range of the instruments remaining the same whether maintaining 2568 MWt or 2609 MWt. The change in nuclear instrumentation calibration will have no effects on any control room controls or the operator's ability to monitor core power production, thus having no adverse effect on defense in depth or safety margins.
- Required changes to the settings of the pre-identified ICS modules are associated with maintaining the plant within normal operating parameters, thus having no adverse affect on defense in depth or safety margins.

The following modifications are associated with implementation of the power uprate that affect operator displays (including Safety Parameter Display System):

- The Fixed Incore Monitoring System functions may require changes to the plant computer software. The software changes will be transparent to the operators; their response to abnormal indications by the software will remain unchanged, thus having no adverse affect on defense in depth or safety margins.
- The AULD Panel PC displays will be modified to provide both the improved calorimetric and the existing calorimetric values. The Operator will use the modified AULD display to select which calorimetric to be used in the AULD as the controlled parameter used to establish thermal demand. The AULD display, in conjunction with the plant annunciator, will also alert the Operator when the AULD has automatically transferred out of Automatic upon the detection of a sufficient differential in the available secondary heat balance calculations. The Operator will not be allowed by the AULD logic to return the AULD to Automatic using the improved calorimetric as long as this differential exists. The AULD will be allowed to be placed in Automatic using the existing calorimetric with a maximum core thermal power setpoint of 2568 MWt.

The following modifications are associated with implementation of the power uprate that affect alarms:

- A control room alarm has been added to alert the operators when the LEFM system has self diagnosed a condition that has resulted in an internal alert or failure.

The following modifications are required to support the MUR but are not expected to otherwise have significant operational impact:

- Installation of the Caldon system.
- Addition of new Feedwater and Main Steam pressure and Main Steam Temperature instrumentation.
- Modification of AULD software.

7.2.2.3 Control Room Plant Reference Simulator (RIS 2002-03 Section VII.2.C)

The Control Room plant reference simulator will be modified due to this uprate. While there is minimal impact on plant response due to this uprate, changes are needed to be properly modeled on the simulator. These include modifying Integrated Control System function curves to match the increased power output of the plant as well as the equipment modifications discussed above. The simulator modifications will be completed in time for operator training support.

7.2.2.4 Operator Training Program (RIS 2002-03 Section VII.2.D)

The Operations Department has been integrated into the uprate process. An Operations Department representative has been assigned to the uprate team. The design change process requires the Operations Department review and sign-off on design change packages.

The Operator Training program will need to be modified due to this uprate. While there will be minimal impact due to this uprate, changes are being made that the Operator will need to be properly trained on. Training on operation and maintenance of the Caldon LEFM CheckPlus™ System, will be developed and completed prior to implementation of the MUR.

7.2.2.5 Confirmation of Intent to Complete Modifications and Training (RIS 2002-03 Section VII.3)

FPC will complete all modifications identified above (including the training of operators), prior to implementation of the power uprate.

7.2.3 Confirmation of Intent to Revise Operating Procedures Related to Power Levels (RIS 2002-03 Section VII.4)

FPC will revise existing plant operating, maintenance, alarm response, and other procedures to reflect the modifications noted above, as well as appropriate administrative controls necessary to assure timely response to loss of equipment availability.

7.2.4 10 CFR 51.22 Discussion (RIS 2002-03 Section VII.5)

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions

eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (A.1) involve a significant hazards consideration; (A.2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or, (B) result in a significant increase in individual or cumulative occupational radiation exposure.

7.2.5.1 Environmental Assessment (RIS 2002-03 Section VII.5.A)

It has been determined that this License Amendment Request (LAR) meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the no significant hazards evaluation (Attachment 1) for this LAR

2. The proposed changes will allow CR-3 to operate at an uprated power level of 2609 Megawatts Thermal (MWt). This represents an increase of approximately 1.6 percent over the current 100 percent power level of 2568 MWt.

The proposed changes do not significantly impact installed equipment performance or require significant changes in system operation. Changes in maintenance and operational practices will not impact the release of solid, liquid or gaseous effluents. The specific activity of the primary and secondary coolant is expected to increase by no more than the percentage increase in power level. Therefore, the amount and specific activity of solid waste is not expected to change significantly.

Gaseous and liquid effluent releases are expected to increase from current values by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be maintained within the limits of 10 CFR 20 and 10 CFR 50, Appendix I, in accordance with the requirements of the CR-3 Offsite Dose Calculation Manual (ODCM). The ODCM contains offsite dose calculation methodologies, the radioactive effluent controls program, and radiological environmental monitoring activities. The ODCM contains the methodologies and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, the methodologies and parameters used in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and the controls for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable in accordance with 10 CFR 50.36a. The proposed changes will not result in changes in the operation or design of the gaseous, liquid or solid waste systems, and will not create any new or different radiological release pathways.

Therefore, the proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site.

7.2.5.2 Environmental Assessment (RIS Section 2002-03 Section VII.1.5.B)

The proposed changes will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR 20. Radiation levels in the plant are expected to increase by no more than the percentage increase in power level. Individual worker exposures will be maintained within acceptable limits by the CR-3 as-low-as-reasonably-achievable (ALARA) program. Therefore, the proposed license amendment will not result in a significant increase to the individual or cumulative occupational radiation exposure.

8.0 Changes to Technical Specifications, Protection System Settings, and Emergency System Settings (RIS 2002-03 Section VIII Questions)

1. A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:

    A. a description of the change
    B. identification of analyses affected by and/or supporting the change
    C. justification for the change, including the type of information discussed in Section 3, above, for any analyses that support and/or are affected by change

8.1 Response to RIS 2002-03 Section VIII Questions

8.1.1 There will be no technical specification changes resulting from this LAR, other than the changes to Section 1, DEFINITIONS, and Section 5.2.6.20, Containment Leakage Rate Testing Program.

8.1.2 There will not be any protection system or emergency system setpoint changes resulting from this LAR, although several instruments will require rescaling in order to support implementation of this LAR

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT E

UNCERTAINTY CALCULATION

20697-10 (3/30/06)

AREVA

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 32- 9042687 -001

Title Heat Balance Uncertainty for CR-3 MUR

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PURPOSE AND SUMMARY OF RESULTS.

PURPOSE

The objective of this calculation is to determine the full-power reactor core power uncertainty value, also referred to as the "heat balance uncertainty," for Crystal River Unit 3 (CR-3) based on the installation of CALDON ultrasonic feedwater flow metering equipment and other new secondary side instruments and equipment. The "Square Root Sum of the Squares" (SRSS) approach will be used in this calculation along with request from CR-3 to apply some of the uncertainties as "biases" (algebraically added to the SRSS total) This calculation will also (1) predict the nominal RC pump heat into the RCS and the uncertainty associated with pump heat, and (2) the energy added to the MU tank flow from the makeup pump.

RESULTS

The ASME Performance Test Code Methodology was used to calculate the expected core thermal power uncertainty to be achieved using the Caldon CheckPlus™ System ultrasonic flow meter. The analysis concluded that using the following instrument uncertainty values, the core thermal power uncertainty would be 0.394% of 2609 Mwt, thus allowing a power uprate of 1 6% to be pursued. The feedwater flow and temperature measurement is the bulk of this uncertainty (0 34% absolute and ~84% of the total uncertainty) The new steam temperature/pressure instrumentation results in ~4% of the total uncertainty and the steam pressure measurement uncertainty is ~1% of the total. The RC pumps energy uncertainty, ambient loss uncertainty and an atmospheric pressure correction uncertainty were chosen to be treated as a bias (algebraically added and not SRSS) and they are ~11% of the total uncertainty (see Attachment 1). After the final feedwater flow/temperature uncertainty is determined for the CR-3 specific equipment (post fabrication testing), the total uncertainty may be reduced

Other pertinent output of this calculation include (1) the RC pump power (to be used in the HB equation) is 20.96 MWt, (2) the temperature increase due to the MU pump to be added to the makeup measured temperature is 5.6F, (3) the pressure to be used for MU enthalpy is a constant (2400 psia), and the letdown pressure will be the measured RCS pressure minus 20 psi (noting that this input has essentially no impact on the final HB uncertainty).

Revision 1 replaces revision 0 (editorial changes).

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:		THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK
CODE/VERSION/REV	CODE/VERSION/REV	X YES
		NO

RECORD OF REVISION

Revision	Description	Date
00	Original release	04/2007
01	Minor editorial changes	04/2007

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1.0 PURPOSE

The objective of this calculation is to determine the full-power reactor core power uncertainty value, also referred to as the "heat balance uncertainty," for Crystal River Unit 3 (CR-3) based on the installation of Caldon ultrasonic feedwater flow metering equipment and other new secondary side instruments and equipment. The "Square Root Sum of the Squares" (SRSS) approach will be used in this calculation along with request from CR-3 to apply some of the uncertainties as "biases" (algebraically added to the SRSS total). This calculation will also (1) predict the nominal RC pump heat into the RCS and the uncertainty associated with pump heat, and (2) the energy added to the MU tank flow from the makeup pump.

2.0 ASSUMPTIONS

2.1 Key Assumptions

There are no key assumptions that need to be verified to use the results of this calculation. The assumptions used in these calculations are presented in this section.

2.2 Other Assumptions

(1) Since CR-3 will modify the Automatic unit load demand (AULD) to simulate the online computer program "Fixed Incore Detector Monitoring System" (FIDMS - to be installed), this heat balance uncertainty will be applicable to the heat balance performed in both calculations.

(2) All the uncertainties discussed herein are in absolute values (i.e., psi absolute for steam pressure at the maximum possible value, not percent of full power pressure). Therefore, the full power values are not pertinent to the uncertainty values. However, since the partial derivatives used in the uncertainty calculation require approximate full power values, they will be presented or calculated herein. Since these derivatives (i.e., $\partial H/\partial T_{FW}$) will not change significantly within 10% of the anticipated values, these approximate plant values at full power are acceptable.

(3) Letdown mass flow rate will be assumed equal to makeup (MU) plus seal injection (SI). This is a reasonable steady state assumption since pressurizer level is essentially constant. Also, since the contribution of the MU/LD energy to the total heat balance uncertainty is ~0.5% after SRSS (~0.5% of 0.39% or 0.002% of the total HB uncertainty), a small variation in the MU vs. LD flow will be negligible on the final uncertainty.

(4) Since the ambient RCS loss is not a "measured" value and therefore has no "instrument" uncertainty, the uncertainties associate with this value will be estimated in this document. CR-3 has requested that the ambient heat loss uncertainty be applied as a bias and not be SRSS with the other random uncertainties.

(5) The RC pump heat into the RCS is based on the brake horsepower test values of the original pump impellers. Based on BHP test data from different Byron Jackson (BJ) pumps at CR-3 and other B&W plants, it is assumed that the BHP of replacement impellers at CR-3 (also made by BJ) are very similar to be the original BJ impellers (tested). The similarity of BJ pump impellers is shown below.

(6) The pump power input is based on the BHP test data which in turn is based on cold leg temperature, pump flow rate, and any test instrument uncertainty. The uncertainty in the BHP (impeller characteristics) will be based on the uncertainty in the testing instrumentation (of which DB-1 and ANO-1 testing accuracies were found and used for

CR-3). The RC flow uncertainty on BHP at CR-3 is shown below to be negligible. Since lower cold leg temperatures are conservative for the BHP power in the HB equation, the actual BHP at full power will be based on a maximum expected cold leg temperature (in all four cold legs) at full power. This is 559F (2F greater than the nominal expected 557F).

(7) Water/steam properties in AULD will be made to be essentially identical to the values generated in FIDMS and therefore no uncertainties or adjustments will be needed to account for water/steam properties. Water properties used in the file are based on Reference 12 which is also the basis for the 1997 ASME published values (noting that the IFC-67 standard used in the ASME 1967 version has been declared obsolete by the ASME).

(8) The 14.7 psi (22.93 in Hg) constant added to the gage pressure to convert to absolute pressure (psia) will vary slightly as atmospheric pressure varies. CR-3 has chosen to address this error and apply it as a bias (not a random, independent uncertainty). Attachment 4 shows a summary of 7 years worth of average barometric pressure indications in central Florida (as provided by CR-3). This data shows the maximum pressure in this time period was 0.67 in Hg greater than 29.93 and CR-3 has chosen to bias this impact in the HB uncertainty. It is applied only for steam properties only since this pressure difference will impact FW enthalpy or RCS cold leg enthalpies insignificantly. Should all the steam pressure transmitters be calibrated during a high atmospheric pressure, the steam pressure could be in error by .67 in Hg (.33 psi). Using the 1.29E6 BTU/hr/psi (1.284E6 calculated in section 5.0 below), this is an error of 425700 BTU/hr. This results in a HB uncertainty of 4.257E5/8.905E9 BTU/hr or .000048 (.0048%).

(9) The FW pressure has previously never been measured near the SG inlet at CR-3 and therefore no plant data is available for the nominal full power value (at the new FW pressure instrument is located at the FE-34A and FE-34B flow measurement locations). The indicated full power FW pressure is estimated to be ~980 psia at the new measurement location (engineering judgment). This does not impact the uncertainty on this pressure since it is based on 1200 psig (maximum range pressure). Also, since the partial derivative used for the impact on heat balance is based on a range from 925 to 1025 psia, this assumption will have no impact on the partial derivative and consequently no impact on the final HB uncertainty.

(10) Reference 8 shows the expected steam pressure at the bundle outlet is 930 psia. Reference 9 shows that the pressure drop from the tube bundle outlet through the SG nozzle is ~5 psid. The velocity head in the steam pipe is ~5 psi. This results in a steam pressure of ~920 psia in the piping just outside the SG. Note that the uncertainty for this value is conservatively based on 1000 psig (and not percent of full power) and therefore this parameter is presented only to show expected range at full power. This is used only to determine the partial derivatives. These partial derivatives will not change substantially in the 800-1000 psi range and is therefore an acceptable assumption for this calculation.

(11) No "pressure" location correction is needed for the steam temperature indication since they are assumed to be within ~2 to 3 feet of the steam pressure indication. This assumption has been verified verbally and the appropriate drawing will be referenced when it is available.

(12) These uncertainty calculation are for the Caldon LEFM equipment fully operational (not in the maintenance mode).

3.0 METHODOLOGY

A discussion of heat balance uncertainty methodology used for the CR-3 heat balance uncertainty is presented herein. This information is presented to show the acceptability of the "Square Root Sum of the Squares" (SRSS) approach for this type of calculation. Some of the uncertainties were chosen to be biased uncertainties (a non-random constant error in the HB equation).

3.1 Industry Standard

The ASME provides a standard methodology for estimating instrument-related uncertainties, Reference 1. Both individual instruments as well as resultants from multiple instruments are treated in this reference.

3.2 AREVA NP Experience

AREVA NP has performed secondary heat balance calculations including uncertainty calculations for secondary thermal power, core thermal power, and RCS flow for a number of B&W plants. The methodology used in these calculations is consistent with those of the ASME, Reference 1. The governing equation is presented and then differentiated with respect to the contributing measurements. The products of the partial derivatives and individual measurement uncertainties are squared, summed, and then square-rooted to solve for the core thermal power uncertainty. For example, from Reference 6, the uncertainty in steam generator secondary power is:

$$E(Q) = [(\partial Q/\partial W_{fw} \times \varepsilon_{Wfw})^2 + (\partial Q/\partial T_s \times \varepsilon_{Ts})^2 + (\partial Q/\partial T_{fw} \times \varepsilon_{Tfw})^2 + (\partial Q/\partial P_s \times \varepsilon_{Ps})^2 + (\partial Q/\partial P_{fw} \times \varepsilon_{Pfw})^2]^{0.5}$$

Where
E(Q) = steam generator thermal power uncertainty
Q = steam generator thermal power
Wfw = feedwater flow
Ts = steam temperature
Tfw = feedwater temperature
Ps = steam pressure
Pfw = feedwater pressure
ε_i = measurement uncertainty for feedwater flow, feedwater pressure, feedwater temperature, steam pressure, and steam temperature

Since the FW flow and temperature uncertainties were combined by CALDON (Attachment 2) into one power uncertainty ($\varepsilon_{Wfw/T}$) in BTU/hr, the equation becomes

$$E(Q) = [\varepsilon_{Wfw/T})^2 + (\partial Q/\partial T_s \times \varepsilon_{Ts})^2 + (\partial Q/\partial P_s \times \varepsilon_{Ps})^2 + (\partial Q/\partial P_{fw} \times \varepsilon_{Pfw})^2]^{0.5}$$

The nominal heat balance uncertainty would be:

E^2 (core) $=[E_Q^2 + E_{RCpump\ energy}^2 + E_{ambient\ heat\ loss}^2 + E_{Makeup/letdown}^2]$ if all components were assumed independent and random (with the secondary heat balance "E_Q " comprising the large majority of the uncertainty).

Since the measured pressures (in particular the steam pressure) require a conversion to psia by adding the atmospheric pressure, CR-3 has requested an atmospheric pressure uncertainty be added to this calculation. They also requested that the ambient heat loss uncertainty and the RC pump energy uncertainty be treated as bias error (added algebraically to the SRSS error).
This final error in the heat balance uncertainty equation will be
E (core) $=[E_Q^2 + E_{Makeup/letdown}^2]^{0.5} + E_{ambient\ heat\ loss} + E_{Atmos\ press} + E_{RC\ Pump}$

3.3 Crystal River 3 Heat Balance Equations

Crystal River 3 uses software in the AULD module to calculate core thermal power. This calculation is based on the following equation:

$$Q\ core = W_{FWA}\left(H_{SA} - H_{FWA}\right) + W_{FWB}\left(H_{SB} - H_{FWB}\right) + Q_{LD\text{-}MU} - Q_{RCP} + Q_{LOSS}$$

Where;

W_{FWA}, W_{FWB}	Feedwater flows in Loop A & B
H_{SA}, H_{FWA}, H_{SB}, H_{FWB}	Steam & feedwater enthalpies for Loops A & B
$Q_{LD\text{-}MU}$	Heat loss due to primary side letdown flow
Q_{RCP}	Heat added due to RC pumps
Q_{LOSS}	Ambient heat losses from the RCS

3.4 Crystal River 3 Heat Balance Instruments

A listing of Crystal River 3 AULD computer points that are input to the current core thermal power calculation is provided for information In Table 1.

Table 1 - Crystal River 3 Heat Balance Input Listing

Instrument Description	Units	Range
FW 1 COMP FLOW, Caldon	KPPH	*
FW 2 COMP FLOW, Caldon	KPPH	*
RCS LETDOWN FLOW	GPM	0-160 (Ref. 25)
RCS PRESSURE	PSIG	1700-2500 (Ref 15,19)
TEMP @ LETDOWN FLOW MEASUREMENT	Deg F	0-200(Ref. 23)
MAKEUP TEMPERATURE (AT MU TANK)	Deg F	40-200(Ref. 23)
LETDOWN TEMPERATURE (AT CL LD NOZZLE)	Deg F	520-620(Ref. 24)
SG 1 FW PRESS FW-297-PT	PSIG	600-1200(Ref. 7)
SG 1 OUT STM PRESS, MS-114-PT	PSIG	800-1000(Ref. 4)
SG 1 OUT STM PRESS, MS-115-PT	PSIG	800-1000(Ref. 4)
SG 2 FW PRESS FW-298-PT	PSIG	600-1200(Ref. 7)
SG 2 OUTLET STM PRESS, MS-116-PT	PSIG	800-1000(Ref. 4)
SG 2 OUTLET STM PRESS, MS-117-PT	PSIG	800-1000(Ref. 4)
SG 1 OUTLET STM TEMP MS-001-TE1,MS-001-TE2	Deg F	To 610F(Ref. 5)
SG 1 OUTLET STM TEMP MS-002-TE1,MS-002-TE2	Deg F	To 610F(Ref. 5)
SG 2 OUTLET STM TEMP MS-003-TE1,MS-003-TE2	Deg F	To 610F(Ref. 5)
SG 2 OUTLET STM TEMP MS-004-TE1,MS-004-TE2	Deg F	To 610F(Ref. 5)
SG 1 FW TEMP Caldon	Deg F	>250F
SG 2 FW TEMP Caldon	Deg F	>250F

*The range of this instrument (at the specified accuracy) is limited only to the feedwater being above 250°F (Reference 3)

4.0 INPUTS TO THE UNCERTAINTY EVALUATION

The inputs to the final heat balance uncertainty are (1) instrument uncertainties, (2) non instrument uncertainties (pump heat, ambient losses, atmospheric pressure), and (3) the full power partial derivative. These three groups of inputs are taken from references or calculated below.

4.1 Referenced Instrumentation Inputs

4.1.1 CALDON Equipment

The Caldon LEFM CheckPlus™ System ultrasonic feedwater flow meter provides a measurement of the feedwater flow and feedwater temperature (Reference 17) with appropriate reference page in Attachment 2. The preliminary values used are for the combine temperature and flow uncertainty converted to a power uncertainty:

Feedwater Flow Rate Combined with Feedwater Temperature Uncertainty = ±0.34% of nominal Power

This uncertainty is based on the total power (both SGs combined).

Note that the instrument specific value is expected to decrease when final testing of these instruments is performed. The value used herein is a nominal (maximum) value expected.

4.1.2 New CR-3 Installed Instrumentation

The following uncertainties for steam temperature, steam pressure, and feedwater pressure were provided by Crystal River 3, References 4 (pg 6), 5 (pg 6), and 12. These are new instrumentation installed at CR-3.

The feedwater pressure instrument is in the FW piping approximately 150 feet (with pressure losses due to friction and ~10-20 elbows) before the feedwater ring header and at the same approximate elevation (see Assumption 9). The FW pressure at the point where primary energy is being added (at the spray nozzle) is the most correct pressure to use for the FW enthalpy, however, the FW enthalpy dependency on FW pressure is very small and the location of the pressure input will not change the heat balance significantly. This DP will have no impact on the partial derivatives calculated herein for the final heat balance uncertainty.

The steam pressures and temperatures are measured approximately just before the piping to the MSSVs (4 places). The pressure at point where primary energy stops being added to the FW is just inside the SG steam nozzles will be ~10-12 psi higher than the measured point. This change will not impact the heat balance as long as the pressure and temperature are measured at approximately the same location (which it is based on assumption 11) and therefore will have no impact on the final heat balance uncertainty.

Steam Temperature Uncertainty (per instrument) = ±2.02°F (up to 610°F) (see Reference 5, eight Indications per Table 1).

Steam Pressure Uncertainty = ±4.57 psi (uncertainty of maximum range [200 psid] at 1000 psig with four indications per Table 1, and Reference 4).

Feedwater Pressure Uncertainty = 7.11 psi (per FW line at maximum range [600 psid] at 1200 psig and one indication per Table 1, and Reference 7). As discussed above in the assumptions section, the atmospheric pressure change will be insignificant on this portion of the calculation.

The steam temperature is monitored by eight RTDs. There are two RTDs at each measurement location and the combined uncertainty using the square root sum of the squares is $(2*2.02^2)^{0.5}/2$ or 1.43°F in each location. Since both SG outlet pipe temperatures are the same, combining the two pipes results in $(2*1.43^2)^{0.5}/2$ or 1.01°F uncertainty for each SG. Combining the two pressure sensors for each SG, the uncertainty is $(2*4.57^2)^{0.5}/2$ or 3.23 psi uncertainty for the total heat balance per SG.

Summarizing the Total Uncertainty of pressure and temperature per SG (statistically combining the two steam line values)

Steam Temperature (per SG) = ±1.01°F (up to 610°F)
Steam Pressure (per SG) = ±3.23 psi (up to 1200 psig)

Feedwater Pressure has only one indication and will not change from above (7.11 psi)

Since the FW flow and temperature uncertainties are for both SGs (total RCS uncertainties), as are the letdown flow, RC pump uncertainties, the other secondary heat balance parameters will also be calculated for both SGs. Therefore, when the heat balance is performed on the entire RCS, all the secondary side instruments can be used (and statistically combined again). The resulting uncertainties are;

Total Uncertainty (statistically combining the two steam line values)

Steam Temperature (total both SGs) = 0.71°F (up to 610°F)
Steam Pressure (total both SGs) = 2.28 psi (up to 1000 psig)
Feedwater Pressure (total both SGs) = 5.03 psi (up to 1200 psig)

4.1.3 Letdown/Makeup Energy

The letdown energy is based on the letdown flow times the LD-MU energy (enthalpy). The equation for this uncertainty is;

$$E(Q_{LD}) = [(\partial Q/\partial W_{LD} \times \varepsilon_{WLD})^2 + (\partial Q/\partial T_{LD} \times \varepsilon_{TLD})^2 + (\partial Q/\partial T_{MU} \times \varepsilon_{TMU})^2 + (\partial Q/\partial P_{LD} \times \varepsilon_{PLD})^2]^{0.5}$$

This assumes that the MU plus SI flow (lb/hr) is equal to the letdown flow (see assumption 3).

Where
$E(Q_{LD})$ = Letdown energy loss uncertainty
Q_{LD} = Letdown thermal power Loss
W_{LD} = Letdown Flow (mass flow)
T_{LD} = Letdown temperature (Cold leg temperature)
T_{MU} = Makeup temperature
P_{LD} = Letdown Nozzle Pressure in RCS
ε_i = measurement uncertainty for letdown flow, letdown temperature, and makeup temperature.

Temperature Inputs

The makeup temperature is a measured parameter, however HPI/MU pump heat added at the typical flow rates is not insignificant. Reference 20 shows the MU/HPI pump curves at CR-3. The pertinent data from these curves is the efficiency and total head. Reference 21 notes that the minimum allowed pump flow (recirculation back to the MU tank) is 100 gpm. With the nominal 35000 lb/hr MU/SI flow rate (see Table 2 below), this is also ~70 gpm SI/MU flow resulting in 170 gpm total pump flow. At 170 gpm Reference 20 shows the pump efficiency is between 49% and 52% and a pump head of 6400 ft.

Per Reference 22, the pump heat input is calculated as

T(degree rise) = [total head (ft)]/778 (1/efficiency -1)

T1 = 6400/778(1/.5 – 1) = 8.22 x 1 = 8.2F

At 240 gpm pump flow (~140 gpm LD flow),
T2 ~ 6000/778(1/.58 – 1) = 7.7 x 0.72 = 5.6F

Therefore, since high LD/MU energy is conservative for the core power calculation, and low MU temperature results in high MU/LD energy, 5.6F will be a constant added to the measured MU temperature to account for pump heat.

Reference 23 calculates the maximum uncertainty on the measured MU temperature at +/-4.82F (increasing to 5.0F per Table 4 below).

The letdown temperature is the cold leg temperature and the uncertainty is based on the full span of 100F (520F-620F) and is 3.42F per Reference 24. As discussed below, this will be rounded up to 4F in the uncertainty calculation.

Pressure Inputs

The MU pressure is essentially at the MU pump discharge where the ~5.6F temperature energy is added to the MU water. This is typically atmospheric pressure plus the pump head or between 2350 and 2400 psia. At 110F MU temperature[1], 2350 and 2400 psia, the enthalpies are 84.106 and 84.235 BTU/lb. At 557F LD temperature and, 2230 psia LD pressure, the enthalpy is 555.993 BTU/lb. The LD-MU enthalpy difference is 471.887 BTU/lb at a 2350 psia MU pressure and 471.758 BTU/lb at a 2400 psia MU pressure. This is 471.758/471.887 = .99973 difference in the total MU/LD energy (typically ~3 Mwt) or ~.0008 Mwt which is insignificant relative to the 2609 MWt core power. Therefore, a constant pressure of 2400 psia along with the measured MU temperature (plus a constant 5.6F) will be used to determine the MU enthalpy.

In order to be consistent with FIDMS, the letdown pressure is the measured pressure (measured at the "A" hot leg tap) minus ~20 psi (the reduction in pressure after passing through the SG at full power plus an elevation pressure increase) to the let down nozzle. The uncertainty in the LD pressure per Reference 19 is 22.48 psi, (use 23 psi). This uncertainty and location correction will essentially have no impact the final HB uncertainty and since even +/- 100 psi will not impact the HB uncertainty (see discussion on MU pressure), no further discussion will is required on this parameter.

Flow Inputs

The HB equation requires mass flow (lb/hr) and the measured flow at CR-3 is volumetric (gpm). This will introduce two uncertainties associated with LD flow, one in the measurement of GPM and one in the measurement of the temperature (needed to convert to lb/hr). The maximum flow uncertainty from Reference 25 is 6.77 gpm (based on the maximum flow of 160 gpm). The maximum temperature uncertainty on the location where the flow is measured is 5.64F

[1] This is an assumed nominal temperature (see assumptions 2) at the MU pump discharge and is used only for relative difference in heat inputs.

(Reference 23) based on the maximum range temperature of 200F. At a typical maximum LD temperature of ~135F, 5.64F will impact the density by 61.5 lb/ft^3 [135F, 150 psia]/61.6 lb/ft^3 [129F, 150 psia] =0.999. At a maximum of 140 gpm flow (see Reference 25), this is 0.001 x 140 ~ 0.2 gpm. Therefore, the flow uncertainty will be 6.77 + 0.2 = 7 gpm. This temperature uncertainty is essentially a bias added to the flow measurement uncertainty (and not SRSS with the flow measurement).

4.1.4 Ambient Heat Losses

Reference 6 estimated an ambient loss uncertainty at 2.5E6 BTU/hr (out of 5.12E6 BTU/hr [1.5 MWt] estimated total heat loss). This uncertainty is essentially 50% of the expected maximum heat losses predicted in Reference 16 (which already had ~±20% uncertainty included in the 1.5 MWt) and was based on engineering judgment and has no analytical basis. The 0.75 MWt uncertainty[2] was a conservatively high estimate added again to this ambient loss since it did not impact the final HB uncertainty when it was SRSS with the other plant parameters. CR-3 has chosen to treat this uncertainty as a bias of 0.75/2609 = .00029 or 0.029%.

Note that this ambient loss does not include pressurizer ambient losses because the pressurizer heaters cycle on to compensate for these losses.

4.2 Calculated Uncertainty Inputs

4.2.1 RC Pump Heat Input and Uncertainty

The brake horsepower is a function of flow (gpm) and temperature and is the total energy the impeller puts into the water and is a published (test) value of each original impeller at CR-3. The temperature function is the density difference between the test temperature and the temperature expected during operation (at 100% power). The flow function is shown in the following curves based on the test data. Note that the impellers were designed to have the approximate peak BHP at the expected flow and at this flow, the BHP is relatively steady (not changing significantly in a +/- 2% flow range). The attached BHP Curves (test data) originally installed at CR-3 show the following (Attachment 3 shows original data transcribed for clarity). Some conclusions drawn from this data are;

1. The BHP was apparently designed to peak at approximately the initial design flow (~90000 gpm)

2. Each Impeller provides approximately the same total energy at 96720 gpm (CR-3 best estimate flow at power) or ~5.24 MWt (or 1.8E7 Btu/hr, ~7030 brake horsepower) per pump. This is the average of 12 BJ pump impellers shown below. This results in a best estimate total energy is 20.96 MWt (or 7.16E7 Btu/hr) for the 4 pumps.

3. A 2°F difference in Tcold is equivalent to ~0.008 MWt per pump or ~0.03 MWt for 4 pumps and a +/- 2% difference in RCS flow will not change BHP significantly (4-pump only).

Density correcting the test data to 556°F and 558°F (the range of cold leg temperatures expected at CR-3 at full power and 2250 psia) are;

---

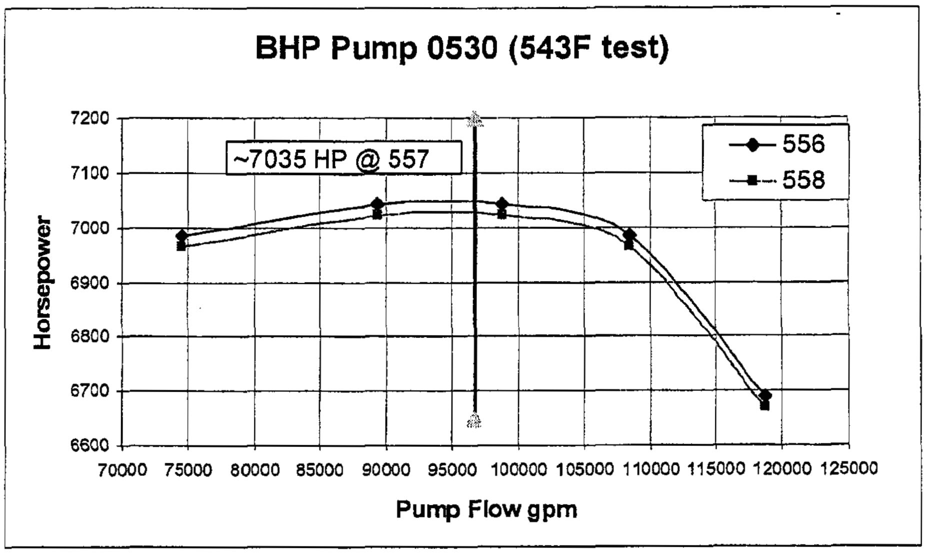
[2] This uncertainty was based on the judgment that any instrumentation used to actually measure/calculate the primary system ambient losses would likely have less than the 0.75MWt random uncertainty (since it is approximately +/- 50% of an anticipated value).

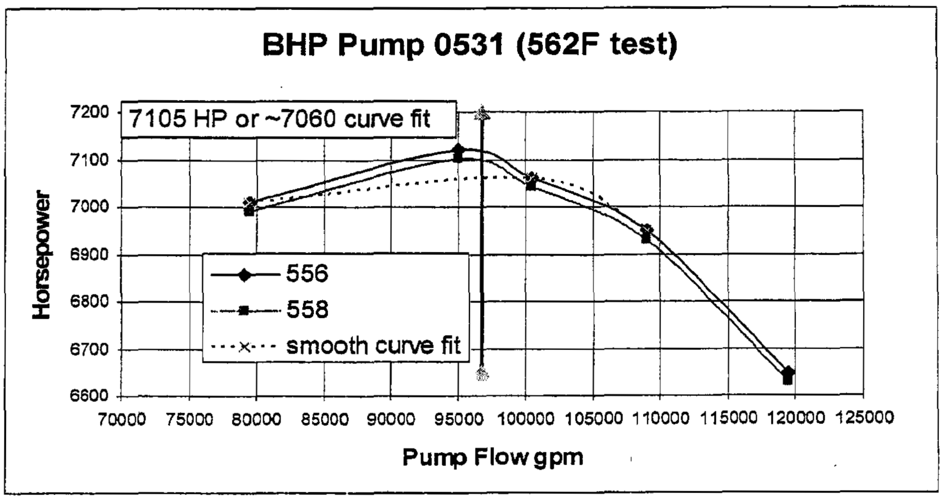
Impeller ID 0532				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
80350	6990.25	6970.22	6950	560
97800	7141.99	7121.53	7080	562
100700	7061.29	7041.07	7000	562
111800	6864.59	6844.92	6805	562
120150	6658.04	6638.97	6610	561

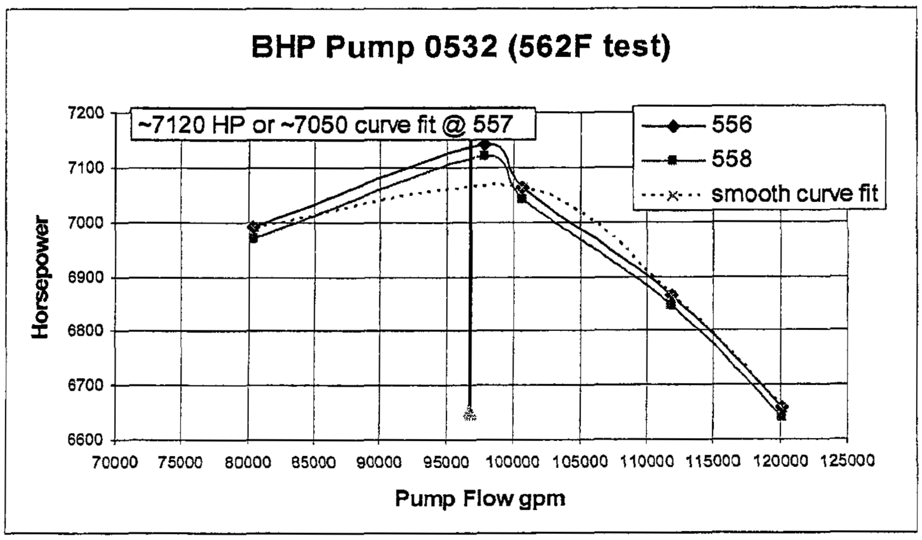
Impeller ID 0531				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
79550	7010.85	6990.77	6950	562
95000	7121.82	7101.42	7060	562
100400	7061.29	7041.07	7000	562
109050	6950.33	6930.42	6890	562
119500	6647.97	6628.92	6600	561

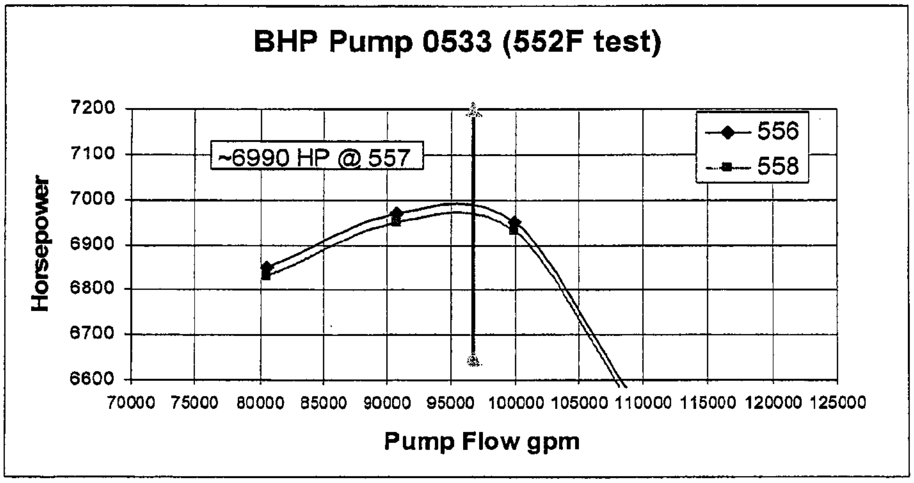
Impeller ID 0530				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
74500	6984.49	6964.48	7110	543
89350	7043.43	7023.26	7170	543
98750	7043.43	7023.26	7170	543
108500	6984.49	6964.48	7110	543
118800	6688.80	6669.64	6800	544

Impeller ID 0533				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
70800	6592.78	6573.89	6630	552
80600	6851.32	6831.69	6890	552
90750	6970.64	6950.68	7010	552
99950	6951.41	6931.49	7010	550
108850	6574.58	6555.75	6630	550

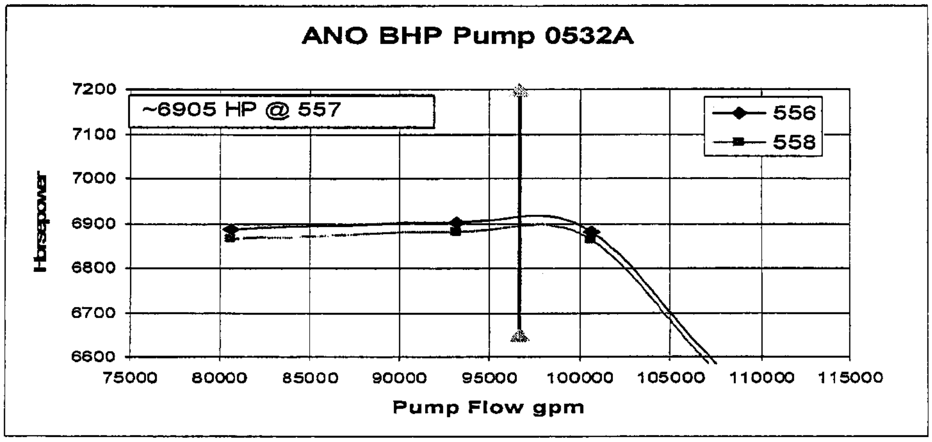

BHP Pump 0530 (543F test)
~7035 HP @ 557
556
558
Horsepower
7200
7100
7000
6900
6800
6700
6600
70000 75000 80000 85000 90000 95000 100000 105000 110000 115000 120000 125000
Pump Flow gpm

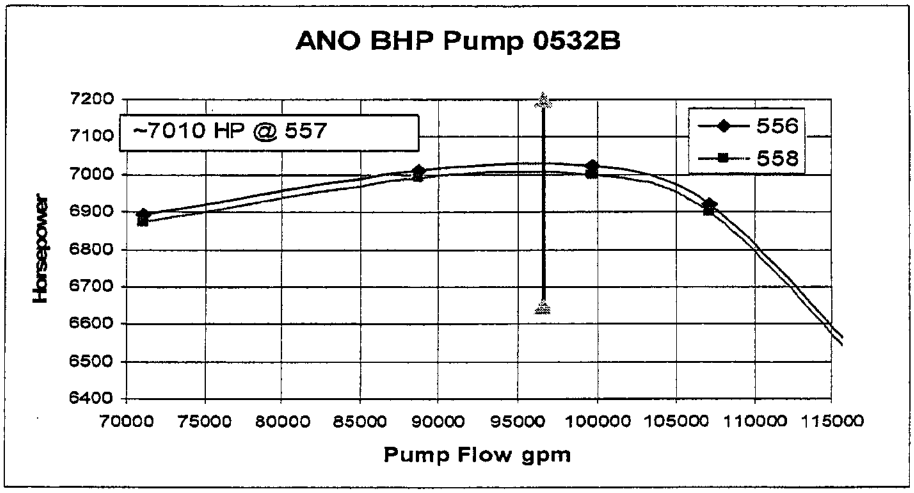

BHP Pump 0531 (562F test)
7105 HP or ~7060 curve fit
556
558
smooth curve fit
Horsepower
7200
7100
7000
6900
6800
6700
6600
70000 75000 80000 85000 90000 95000 100000 105000 110000 115000 120000 125000
Pump Flow gpm

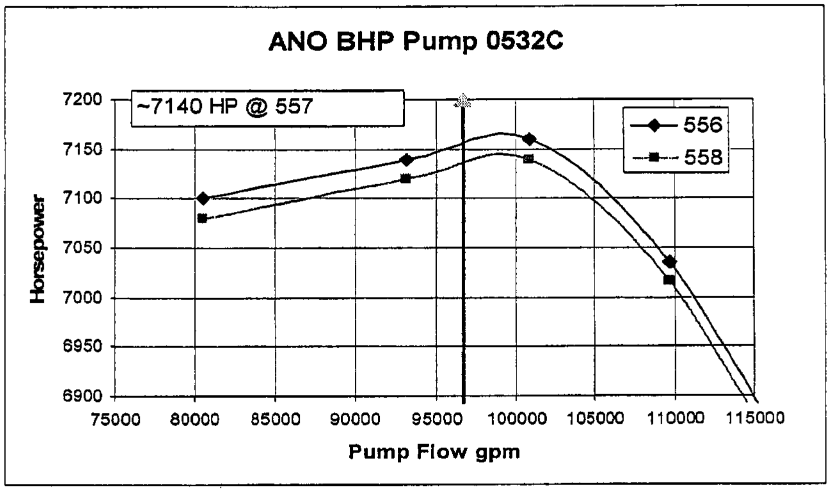

BHP Pump 0532 (562F test)
~7120 HP or ~7050 curve fit @ 557
556
558
smooth curve fit
Horsepower
7200
7100
7000
6900
6800
6700
6600
70000 75000 80000 85000 90000 95000 100000 105000 110000 115000 120000 125000
Pump Flow gpm

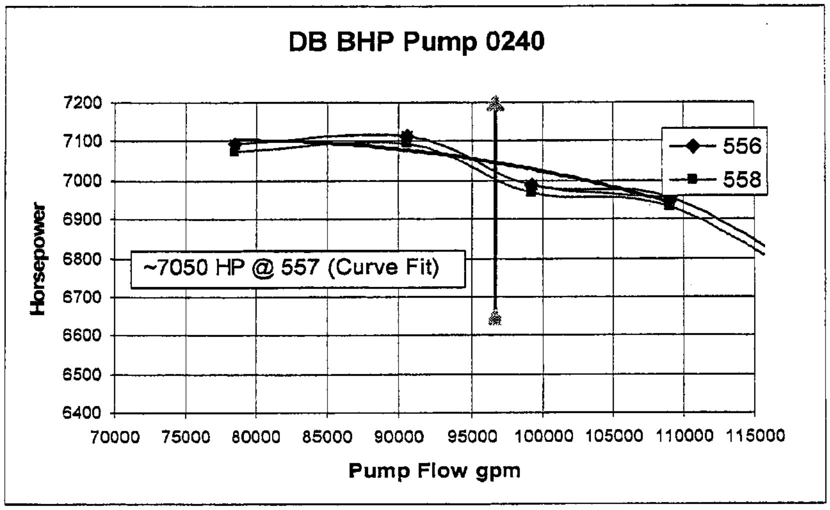


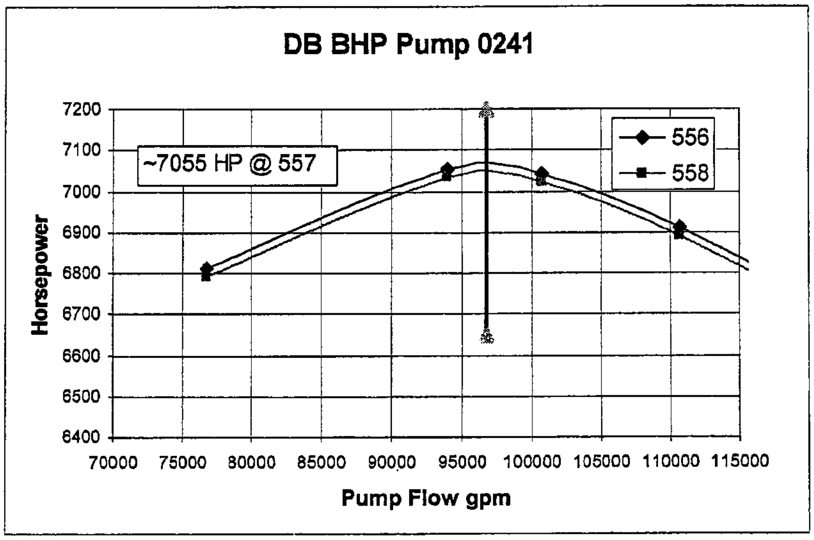
Finally, note that the Byron Jackson (Flow Server) design was relatively consistent in the other impellers tested for ANO-1 and DB-1. Some of these impeller BHP curves are shown below (based on References 13 and Attachment 3).

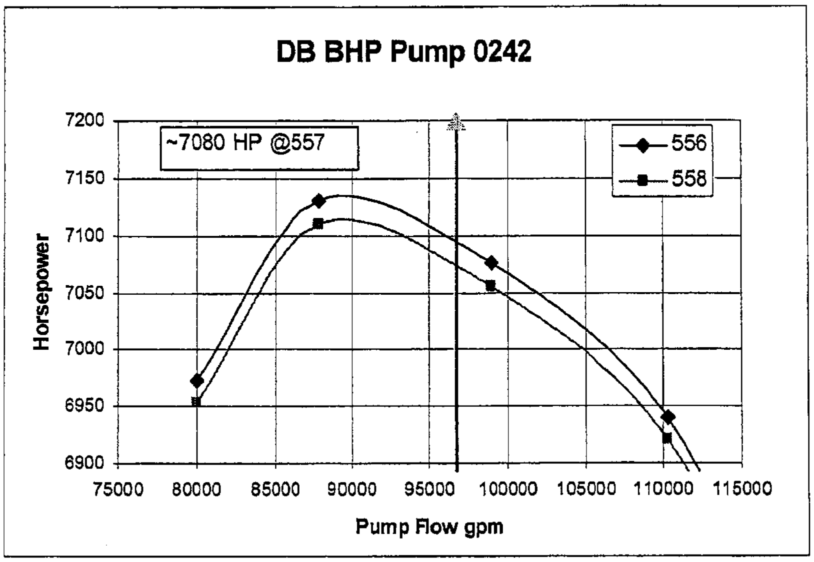


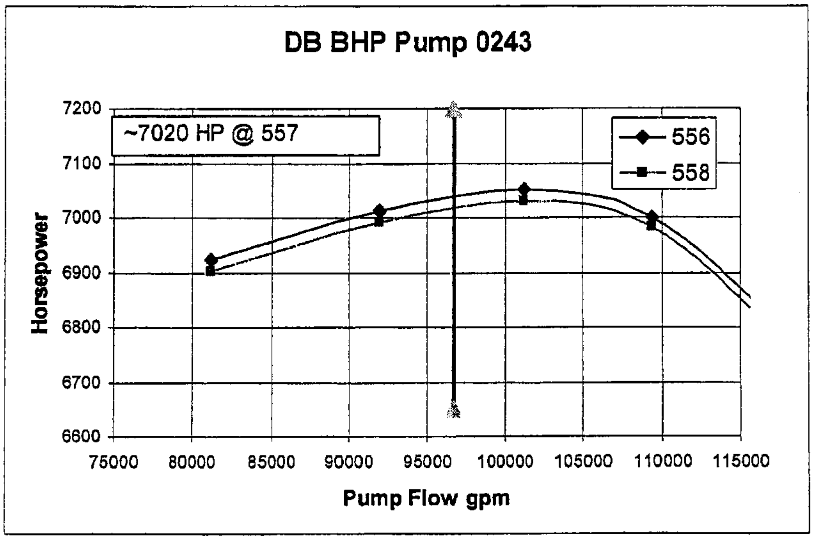



ANO BHP Pump 0532C
~7140 HP @ 557
556
558
Horsepower
7200
7150
7100
7050
7000
6950
6900
75000
80000
85000
90000
95000
100000
105000
110000
115000
Pump Flow gpm


DB BHP Pump 0240
~7050 HP @ 557 (Curve Fit)
556
558
Horsepower
7200
7100
7000
6900
6800
6700
6600
6500
6400
70000
75000
80000
85000
90000
95000
100000
105000
110000
115000
Pump Flow gpm


DB BHP Pump 0241
~7055 HP @ 557
556
558
Horsepower
7200
7100
7000
6900
6800
6700
6600
6500
6400
70000
75000
80000
85000
90000
95000
100000
105000
110000
115000
Pump Flow gpm





The following table shows the average of these 12 impellers.

	HP	MWt
DB	7050	5.26
	7055	5.26
	7020	5.23
	7080	5.28
ANO	7140	5.32
	6905	5.15
	6940	5.18
	7010	5.23
CR-3	7060	5.26
	7035	5.25
	6990	5.21
	7050	5.26
Avg	7027.917	5.24
Avg (CR-3 only)	7033.75	5.25

Given that all the BJ test data on impellers averaged 5.24 MWt range at the CR-3 flow rate, it will be assumed that the replacement impellers (manufactured by BJ) will all be about the same and the original impeller data will be valid (see assumption 5). Therefore, the total heat input of the CR-3 pumps will be 4 x 5.24 = 20.96 Mwt

Uncertainty On Pump Power
The uncertainty in the BHP used for CR-3 is based on two components (1) the validity of the test data and the applicability of the tested impellers relative to the actual replacement impellers at CR-3, and (2) the actual cold leg temperature during full power operation.

Test Data:
At the CR-3 expected RCS flow rate (see curves above), the range of BHP spans ~5.15 to 5.32 MWt per pump. Data from the ANO-1 and Davis Besse pump test (see Attachment 3) revealed the flowing instrument uncertainties in the test instrumentation used by Byron Jackson.

Kilowatt Meter	+/- 0.5%
Potential Transformers	+/- 0.5%
Current Transformers	+/- 0.5%
Flow rate	+/- 1%
Temperature	+/- 0.5%
Pressure	+/- 0.25%

The uncertainty on the power measuring Instrumentation requires two potential transformers[3], the current transformer, and the watt meter resulting in $(0.5^2 + 0.5^2 + 0.5^2 + 0.5^2)^{0.5}$ = 1.0% of the pump energy or 0.0524 Mwt per pump

The flow uncertainty of 1% of the nominal flow (1000 gpm) has essentially no impact on the final pump energy since as shown above, the BHP at CR-3 is in the "flat" portion of the curve of BHP vs flow. The temperature impact is ~0.005 x 560F = 2.8F (use 3F). As shown on the table for pump Impeller ID 0532 at 97800 gpm, the impact of 3F is (7142-7121)3/2 HP = 31.5 HP = .023 Mwt per pump.

The impact of the RCS pressure uncertainty (.0025 x 2250 psia= 6 psia) will change the pump power by less than 1 horse power and will be ignored. This is based on the cold leg density impact of 2250 vs. 2256 psia. This is ~46.437/46.433 x ~7000 HP < 7001 HP.

The potential variability in the pump impellers actually installed at CR-3 will use a conservative application of the nominal pump power minus the minimum measured power (the conservative direction for the HB). This is 5.24-5.15= 0.09 Mwt

Since each of these uncertainties are independent and random, the total pump energy uncertainty is $(0.0524^2 + .023^2 + .09^2)^{0.5}$ = 0.107 MWt. Since the four pumps are independent, the total uncertainty on four pumps is $(4 \times .107^2)^{0.5}$ = 0.21 Mwt.

Plant Conditions
Similar to the discussion above, the uncertainty in the actual RCS flow (within ~ ± 2%) and RCS pressure will both have an insignificant impact on the final total pump heat. The pump heat calculated above (20.96 Mwt) is based on 557F Tcold leg and actual Tcold is the dominant uncertainty parameter in the pump heat calculation during operation. At full power, Tcold can

[3] Reference 18 pg 16-54 describes a normal measurement setup for 3-phase current with a balanced load (which includes a motor). This normal setup requires two potential transformers and one current transformer.

vary due to the ICS control over reactor vessel Tavg and loop A to loop B delta Tc. If delta Tc (cold leg delta temperature difference) varies at a constant Tavg, the two pumps with an increased temperature and the two pumps with a decreased temperature will essentially offset each other, maintaining a constant total pump energy. During an end of cycle coastdown (reduced Tcold), the actual pump power will increase. If this increase was used in the HB equation, the allowable Qsec would actually increase, resulting in the constant Qpump at 557F conservative for this situation. An increase Tcold will reduce pump energy which in turn will decrease the allowable SG energy to maintain a 2609 Mwt core power making the 20.96 Mwt constant pump power (at 557F) non-conservative if Tavg (Tcold) were to increase.

During normal full power operation, the ICS uses all the cold leg RTDs and the hot leg RTDs to establish Tavg. The normal operation permits Tavg +/- 2F. At +2F Tavg the cold leg temperature would be 559F. Per pump impeller data above, 2F equates to a little less than 21 HP or 0.015 MWt per pump or 0.06 Mwt for 4 pumps. The total pump energy uncertainty is 0.21+0.06 = 0.27 Mwt. Per CR-3 request, this will be conservatively applied as a bias or it will add 0.27/2609 =0.01% to the uncertainty.

4.2.2 Partial Derivatives at Nominal Full Power Heat Balance Parameters

The partial derivatives in the uncertainty equation are based on water/steam properties at or near the values they are calculated for. Table 2 presents the nominal expected full power values at 2609 MWt.

TABLE 2 - Nominal Heat Balance Parameter Values

Symbol	Description	Units	Nom. Value	Basis
	(at ~location of measurement)			
WFW	Feedwater Flow Rate	Lbm/hr	1.107E+07	Ref. 8
TS	Steam Temperature	F	590.5	Ref. 8
PS	Steam Pressure	psia	~920	Assumption 10
TFW	Feedwater Temperature	F	458.4	Ref. 8
PFW	Feedwater Pressure	psia	~980	Assumption 8
WLD	Letdown Flow Rate	Lbm/hr	35000	70 gpm @~100F, 150 psia*
		gpm	70	Ref 6
TLD	Letdown Temperature	F	557	Ref. 8 (CL temp)
TMU	Makeup (tank+pump heat) temperature	F	110	See Sec 4.1.3
PMU	Makeup Press	psia	2400	See Sec 4.1.3
PLD	Letdown Pressure	psia	2230	See Sec 4.1.3
QRCP	RCP Power	Btu/hr	7.158E+07	See Sec 4.2.1
QLOSS	Ambient Heat Loss	Btu/hr	5.12E+06	See Sec 4.1.4

*This neglects density increases due to boron which is acceptable since these are approximate full power nominal operating conditions.

Note that if the actual values of these parameter vary a few percent when the AULD monitors them, the partial derivatives calculated below will remain applicable.

The water/steam properties at these nominal values are;

Steam Enthalpy = 1250.386 Btu/lbm at 590.5°F and 920 psia
Feedwater Enthalpy = 439.904 Btu/lbm at 458.4°F and 980 psia
Letdown Enthalpy = 555.993 Btu/lbm at 557°F and 2230 psia

Makeup Enthalpy = 84.235 Btu/lbm at 110°F and 2400 psia

The partial derivatives based on these values are:

For steam at 920 psia:
At T = 585°F, H = 1245.687 Btu/lbm
At T = 595°F, H = 1254.157 Btu/lbm

$\partial H/\partial T_S \cong (1254.157-1245.687)/(595 - 585) = 0.847$ Btu/lbm/°F

For steam at 590.5°F:
At P = 915 psia, H = 1250.965 Btu/lbm
At P = 925 psia, H = 1249.805 Btu/lbm

$\partial H/\partial P_S \cong (1250.965 - 1249.805)/(915 - 925) = -0.11601$ Btu/lbm/psia

For feedwater at 458.4°F:
At P = 1025 psia, H = 439.9242 Btu/lbm
At P = 925 psia, H = 439.8797 Btu/lbm

$\partial H/\partial P_{fw} \cong (439.9242 - 439.8797)/(1025 - 925) = 4.45E\text{-}4$ Btu/lbm/psia

For MU at 2400 psia:
At T = 105°F, H = 79.289 Btu/lbm
At T = 110°F, H = 84.235 Btu/lbm

$\partial H/\partial T_{MU} \cong (79.289 - 84.235)/(105 - 110) = 0.989$ Btu/lbm/°F

For LD at 2250 psia:
At T = 555°F, H = 553.442 Btu/lbm
At T = 560°F, H = 559.749 Btu/lbm

$\partial H/\partial T_{LD} \cong (559.749 - 553.442)/(560 - 555) = 1.261$ Btu/lbm/°F

For LD at 557 F:
At P = 2200 psia, H = 556.048 Btu/lbm
At P = 2250 psia, H = 555.957 Btu/lbm

$\partial H/\partial P_{LD} \cong (556.048 - 555.957)/(2250 - 2200) = -1.82E\text{-}3$ Btu/lbm/psi

The following terms are include for potential future use but are not used in this calculation since the feedwater flow and temperature were combined by CALDON into one power uncertainty

For feedwater at 975 psia:
At T = 455°F, H = 436.085Btu/lbm
At T = 465°F, H = 447.348 Btu/lbm
$\partial H/\partial T_{fw} \cong (447.348 - 436.085)/(465 - 455) = 1.126$ Btu/lbm/°F

The water property derivatives are summarized in Table 3 below.

TABLE 3 - WATER PROPERTY DERIVATIVES

	∂H/∂T, Btu/(lbm°F)	∂H/∂P, Btu/(lbm psi)
Steam (590.5°F, 920 psia)	0.847	-0.116
Feedwater (458.4°F, 975 psia)	1.126	4.45E-4
Makeup(110°F, 2400 psia)	0.989	NA (Const Press per Sec 4.1.3)
Letdown(557°F, 2230 psia)	1.261	-1.82E-3

5.0 CALCULATION OF HEAT BALANCE UNCERTAINTY

Inputs were calculated for operating conditions for the Appendix K power uprate. Reference 1 provides step-by-step instructions for calculating the uncertainty of a result. The independent measurement parameters and their nominal values are comprised of the values in Table 2 above.

The expression for core power in terms of a secondary side heat balance is re-stated below.

$$Q_C = W_{FWA}(H_{SA} - H_{FWA}) + W_{FWB}(H_{SB} - H_{FWB}) + Q_{LD-MU} - Q_{RCP} + Q_{LOSS}$$

Where:

W_{FWA}, W_{FWB}	Feedwater flows in Loop A & B
H_{SA}, H_{FWA}, H_{SB}, H_{FWB}	Steam & feedwater enthalpies for Loops A & B
Q_{LD-MU}	Heat loss due to primary side letdown flow
Q_{RCP}	Heat added due to RC pumps
Q_{LOSS}	Ambient heat losses from the RCS
W_{LD}, W_{MU}	Letdown and Makeup Flow Rates
H_{LD}, H_{MU}	Letdown and Makeup Enthalpies

Since the SG uncertainties were combined to one uncertainty

$$Q_C = Wfw(Hs - Hfw) + Q_{LD-MU} - Q_{RCP} + Q_{LOSS}$$

The uncertainties for each parameter are also summarized below:

TABLE 4 – HEAT BALANCE PARAMETER UNCERTAINTY VALUES

Symbol	**Description**	**Units**	**Uncertainty Value**	**Basis**
WFW/TFW	Combined Uncertainty	BTU/lb	0.34% of nominal flow (both SGs)	Section 4.1.1
TS	Steam Temp	F	1.43(single) [2.02F per RTD, 4 RDTs per SG] 1.01 (one SG) 0.71 (both SGs)	Section 4.1.2
PS	Steam Pressure	psi	4.57 (single) 3.23 (one SG) 2.28 (both SGs)	Section 4.1.2
PFW	Feedwater Press	psi	7.11 (one SG) 5.03 (both SGs)	Section 4.1.2
WLD	Letdown Flow	lbm/hr	7 gpm @ 135F, 150 psia or 3.5E3 lb/hr	Section 4.1.3
TLD	Letdown Temp(CL)	F	3.42 (use 4.0 to assure worst case)	Ref. 24
TMU	Makeup Temp	F	4.82 (use 5.0 to assure worst case)	Section 4.1.3

PLD	Letdown Press	psia	22.48(use 23)	Section 4.1.3
QRCP	RCP Power	Btu/hr	9.22E5 (0.27 Mwt)	Section 4.2.1
QLOSS	Ambient Heat Loss	Btu/hr	2.5E6	Section 4.1.4
QATMOS	Psig to Psia error	psi	0.33 (converts to 4.257E5 BTU/hr -Steam Press)	Section 2.2

The uncertainty calculation from section 3.2 for the secondary side heat balance is

$$E(Q) = [\ \varepsilon_{Wfw/T})^2 + (\partial Q/\partial T_s \times \varepsilon_{Ts})^2 + (\partial Q/\partial P_s \times \varepsilon_{Ps})^2 + (\partial Q/\partial P_{fw} \times \varepsilon_{Pfw})^2]^{0.5}$$

The MU/LD uncertainty is per section 4.1.3

$E_{letdown} = [(\partial Q/\partial W_{LD} \times \varepsilon_{WLD})^2 + (\partial Q/\partial T_{LD} \times \varepsilon_{TLD})^2 + (\partial Q/\partial T_{MU} \times \varepsilon_{TMU})^2\ ]^{0.5}$
$E_{ambient}$ = Constant
E_{RCpump}= Constant
E_{ATMOS} = Constant

The final heat balance uncertainty will be:

$E\ (core) = [E_Q^2 + E_{letdown}^2]^{0.5} + E_{ambient\ heat\ loss} + E_{Atmos\ press} + E_{RC\ Pump}$
with E_{QA} comprising the large majority of the uncertainty

The core thermal power equation was differentiated with respect to the individual measured parameters to yield the following sensitivity coefficients (values from Section 4.2.2):

$\theta_{Pfw} = \partial Qc/\partial P_{FW} = W_{FW}\ \partial H/\partial P_{FW}$ = (1.107E+07 lb/hr x 4.45E-4 Btu/lb/psi) = 4.926E3 Btu/hr/psi

$\theta_{Ps} = \partial Qc/\partial P_S = W_{FW}\ \partial H/\partial P_S$ = (1.107E+07 lb/hr x -0.11601 Btu/lb/psi) = -1.284E6 Btu/hr/psi

$\theta_{Ts} = \partial Qc/\partial T_S = W_{FW}\ \partial H/\partial T_S$ = (1.107E+07lb/hr x 0.847 Btu/lb/°F) = 9.376E6 BTU/hr F

$\theta_{Wld} = \partial Qc/\partial W_{LD} = H_{LD} - H_{mu}$ = 555.993-84.235 = 471.76 BTU/lbm

$\theta_{TMU} = \partial Qc/\partial T_{MU} = W_{MU}\ \partial H/\partial T_{MU}$ = (35000 lb/hr x 0.989 Btu/lb/°F) = 3.462E4 BTU/hr F

$\theta_{PLD} = \partial Qc/\partial P_{LD} = W_{MU}\ \partial H/\partial P_{LD}$ = (35000 lb/hr x -0.00182 Btu/lb/psi) = -6.37E1 BTU/hr/psi

$\theta_{TLD} = \partial Qc/\partial T_{LD} = W_{LD}\ \partial H/\partial T_{LD}$ = (35000 lb/hr x 1.261 Btu/lb/°F) = 4.414E4 BTU/hr F

$\theta_{Qrcp} = \partial Qc/\partial Q_{RCPs} = 1$

$\theta_{Qloss} = \partial Qc/\partial Q_{LOSS} = 1$

The following terms are include for potential future use but are not used in this calculation since the feedwater flow and temperature were combined by CALDON into one power uncertainty.
$\theta_{Wfw} = \partial Qc/\partial W_{FW} = (H_S - H_{FW})$ = (1250.384 Btu/lb - 439.904 Btu/lb) = 810.080 Btu/lbm
$\theta_{Tfw} = \partial Qc/\partial T_{FW} = W_{FW}\ \partial H/\partial T_{FW}$ = (1.107E+07 lb/hr x1.126 Btu/lb/°F) =1.246E7 BTU/hr F

Sensitivity Coefficients and Uncertainty Contributions

The sensitivity coefficients and the uncertainty contributions were calculated using the values in Tables 2 and 3 as follows:

Feedwater Flow Rate and Feedwater Temperature

Using the uncertainty of $\varepsilon_{Wfw/T}$ = (0.34/100) * 8.905E9 = 3.028E7 BTU/hr, the systematic uncertainty contribution is:

$[\varepsilon_{Wfw/T}]^2$ = 9.166E14 $(Btu/hr)^2$

Feedwater Pressure

Using the uncertainty of ε_{Pfw} = 5.03 psi, the systematic uncertainty contribution is:

$[\theta_{Pfw} * \varepsilon_{Pfw}]^2 = [4.926E3 * 5.03]^2 = 6.140E8\ (Btu/hr)^2$

Steam Pressure

Using the uncertainty of ε_{Ps} = 2.28 psi, the systematic uncertainty contribution is:

$[\theta_{Ps} * \varepsilon_{Ps}]^2 = [-1.284E6 * 2.28]^2 = 8.573E12\ (Btu/hr)^2$

Steam Temperature

Using the uncertainty of ε_{Ts} = 0.71°F, the systematic uncertainty contribution is:

$[\theta_{Tsa} * \varepsilon_{Ts}]^2 = [9.376E6 * 0.71]^2 = 4.432E13\ (Btu/hr)^2$

Letdown Flow
Using the uncertainty of ε_{WLD} = 3.5E3 lbm/hr

$[\theta_{WLD} * \varepsilon_{WLD}]^2 = [471.76 * 3.5E3]^2 = 2.726E12\ (Btu/hr)^2$

Letdown Pressure
Using the uncertainty of ε_{PLD} = 23 psi

$[\theta_{PLD} * \varepsilon_{PLD}]^2 = [-63.7 * 23]^2 = 2.147E{+}06\ (Btu/hr)^2$

Letdown (CL) Temperature
Using the uncertainty of ε_{TLD} = 4.0 °F, the systematic uncertainty contribution is:
$[\theta_{TLD} * \varepsilon_{TLD}]^2 = [4.414E4 * 4.0]^2 = 3.117E10\ (Btu/hr)^2$

Makeup Temperature
Using the uncertainty of ε_{TLD} = 5.0 °F, the systematic uncertainty contribution is:
$[\theta_{TLD} * \varepsilon_{TLD}]^2 = [3.462E4 * 5.0]^2 = 2.995E10\ (Btu/hr)^2$

RCP Power
This is a biased uncertainty calculated in section 4.2.1 of 0.27 Mwt.

Ambient Heat Loss
This is a biased uncertainty calculated in section 4.1.4 of 0.75 Mwt.

Pgage to Pabsolute Conversion
This is a biased uncertainty shown in section 2.2 of 425700 BTU/hr (0.125 Mwt)

The uncertainty contributions are summarized below in Table 5.

TABLE 5 – HEAT BALANCE PARAMETER UNCERTAINTY CONTRIBUTIONS

Symbol	Description	Uncertainty Contribution			
		Absolute $(Btu/hr)^2$	Relative (SRSS)	Relative (Total)	Percent of Total Power
WFW/TFW	Feedwater Flow/Temp	9.166E+14	94.2733%	83.7722%	0.3301%
TS	Steam Temperature	4.432E+13	4.5582%	4.0504%	0.0160%
PS	Steam Pressure	8.573E+12	0.8818%	0.7836%	0.0031%
PFW	Feedwater Pressure	6.140E+08	0.0001%	0.0001%	0.0000%
TLD	Letdown Temperature	2.147E+06	0.0000%	0.0000%	0.0000%
PLD	Letdown Pressure	3.117E+10	0.0032%	0.0028%	0.0000%
WLD	Letdown Flow Rate	2.726E+12	0.2804%	0.2492%	0.0010%
TMU	Makeup Temperature	2.995E+10	0.0031%	0.0027%	0.0000%
	TOTAL	9.723E+14	100%		
	Bias Corrections	**(Btu/hr)**			
QRCP	RCP Power	9.215E+04	NA	2.6261%	0.0103%
QLOSS	Ambient Heat Loss	2.560E+06	NA	7.2948%	0.0287%
PATMOS	P_{gage} to $P_{absolute}$	4.274E+05	NA	1.2181%	0.0048%
	Totals			100%	0.394%

6.0 SUMMARY OF RESULTS

The ASME Performance Test Code Methodology was used to calculate the expected core thermal power uncertainty to be achieved using the Caldon CheckPlus™ System ultrasonic flow meter. The analysis concluded that using the following instrument uncertainty values, the core thermal power uncertainty would be 0.394% of 2609 Mwt, thus allowing a power uprate of 1.6% to be pursued. The feedwater flow and temperature measurement is the bulk of this uncertainty (0.34% absolute and ~84% of the total uncertainty). The new steam temperature/pressure instrumentation results in ~4% of the total uncertainty and the steam pressure measurement uncertainty is ~1% of the total. The RC pumps energy uncertainty, ambient loss uncertainty and an atmospheric pressure correction uncertainty were chosen to be treated as a bias (algebraically added and not SRSS) and they are ~11% of the total uncertainty (see Attachment 1). After the final feedwater flow/temperature uncertainty is determined for the CR-3 specific equipment (post fabrication testing), the total uncertainty may be reduced.

Other pertinent output of this calculation include (1) the RC pump power (to be used in the HB equation) is 20.96 MWt, (2) the temperature increase due to the MU pump to be added to the makeup measured temperature is 5.6F, (3) the pressure to be used for MU enthalpy is a constant (2400 psia), and the letdown pressure will be the measured RCS pressure minus 20 psi (noting that this input has essentially no impact on the final HB uncertainty).

7.0 REFERENCES

(1) ASME PTC 19.1-1998, Test Uncertainty, Instruments and Apparatus, American Society of Mechanical Engineers, NY, NY, 1998.

(2) Caldon, Inc. Engineering Report-80P Revision 0 (Proprietary Version), Topical Report - "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," March 1997. (For Information Only)

(3) Caldon Topical Report Caldon, Inc. Engineering Report-157P Revision 5 , Topical Report - "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or LEFM CheckPlus™ System," February 2001.

(4) AREVA Doc. 32-9040357-001 "Loop Accuracy Calculation for CR3 Steam Press Computer Ind for MUR Heat Balance Computation"

(5) AREVA Doc 32-9040364-000 "Loop Accuracy Calculation for CR3 Steam Temperature Computer Ind for MUR Heat Balance Computation,"

(6) AREVA Document 32-5001078-01, "CR-3 Heat Balance Uncertainty Calc," March 1998.

(7) AREVA Document 32-9041510-001 "CR3 MUR Feedwater Pressure Uncertainty Calculation"

(8) AREVA Document 32-5012972-01, "CR-3 Power Uprate Operating Conditions"

(9) AREVA Document 32-5018220-02 "ANO-1 ETOSG Secondary Pressure Distribution"

(10) AREVA Document 32-9034910-00, "CR-3 Main Steam Line DP Estimate"

(11) Deleted

(12) WinSteam program for water properties based on IAPWS-IF95 by ChemicaLogic Corporation.

(13) AREVA Document 32-1174376-00, "ANO-1 RCS Delta Pressure," 1989.

(14) Deleted

(15) CR-3 Document I91-0023 Revision 2 "Narrow Range RC Pressure (PIR) Instrument Loop Accuracy"

(16) AREVA Document 32-1240860-00, "Ambient Heat Loss estimate" 12/1995

(17) Caldon, Inc. Engineering Report: ER-579 Revision 1, "Bounding Uncertainty Analysis for Thermal Power Determination at crystal River Unit 3 Using the LEFM✓+ System," February 2007. (Pertinent pages attached as Attachment 4).

(18) Standard Handbook for Electrical Engineering Eleventh Edition 1978, McGraw Hill Publishing

(19) CR-3 Document I01-0002 Revision 0 "TRICON Instrument Accuracy", 10/2001

(20) AREVA Document 32-1257388-00 "CR-3 HPI Pump HQ Response" 1996

(21) AREVA Document 32-5002731-03 "CR-3 HPI Hydraulics Analysis" 2000

(22) Karrassik, Igor J. Centrifugal Pump Clinic, Marcel Dekker Inc. (1981)

(23) AREVA Document 32-9043792-000 "Loop Accuracy Calculation for CR-3 Makeup Tank (MUT) Temperature and Letdown Flow temperature Computer indications for MUR Heat balance Computation," 3/2007

(24) AREVA Document 32-9043797-000 "Loop Accuracy Calculation for CR-3 T-Cold Temperature indications for MUR Heat balance Computation," 3/2007

(25) AREVA Document 32-9049930-000 "Loop Uncertainty Calculation for Let down Flow to AULD Plant Computer," 3.2007

References 15 and 19 are retrievable in the CR-3 documentation system Dallas Scott Date 4/20/07

ATTACHMENT 1 – HEAT BALANCE SPREADSHEET

The methodology developed in Section 4 was programmed in Excel for ease of evaluating various inputs. The Excel spreadsheet was verified by comparing the results of these results with data listed in Section 5.

			BASE CASE								
Description	Units	Nominal Value	Absolute Uncertainty		Absolute Sensitivity		Absolute Uncertainty Contribution(squared)		Relative Uncertainty Contribution	Relative Uncertainty Contribution	Contribution in Mwt
									(SRSS Only)	Tot Uncrt	
Feedwater Flow Rate& Temp	BTU/hr	8.905E+09	3.028E+07	BTU/hr	3.028E+07	BTU/hr	9.166E+14	BTU/hr^2	94.2733%	83.7722%	8.613
Steam temperature	F	590.5	0.71	F	9.376E+06	BTU/hr /F	4.432E+13	BTU/hr^2	4.5582%	4.0504%	0.416
Steam Pressure	psia	920	2.28	psi	-1.284E+06	BTU/hr/psi	8.573E+12	BTU/hr^2	0.8818%	0.7836%	0.081
Feedwater Pressure	psia	980	5.03	psi	4.926E+03	BTU/hr/psi	6.140E+08	BTU/hr^2	0.0001%	0.0001%	0.000
Letdown Pressure	psia	2230	23	psi	-6.370E+01	BTU/hr/psia	2.147E+06	BTU/hr^2	0.0000%	0.0000%	0.000
Letdown Temperature	F	557	4	F	4.414E+04	BTU/hr/ F	3.117E+10	BTU/hr^2	0.0032%	0.0028%	0.000
Letdown Flow Rate	lbm/hr	35000	3.50E+03	lbm/hr	471.76	BTU/lb	2.726E+12	BTU/hr^2	0.2804%	0.2492%	0.026
Makeup Temperature	F	110	5	F	3.462E+04	BTU/hr/ F	2.995E+10	BTU/hr^2	0.0031%	0.0027%	0.000
						SUM=	9.723E+14		100.0%		
Bias for pump power					9.215E+05	BTU/hr				2.6261%	0.270
Bias for Ambient Loss					2.560E+06	BTU/hr				7.2948%	0.750
Bias for Atmospheric Press					4.274E+05	BTU/hr				1.2181%	0.125
Description	Nominal Value (MWt)	Nominal Value (Btu/hr)	Absolute Uncertainty (Btu/hr) (SRSS Components)		Relative Uncertainty (%) SRSS components	Bias for Pump Power	Bias for 0.75 Mwt ambient loss	Atmospheric pressure bias	Total Uncertainty	100.0%	10.28
Core Thermal Power	2609	8.905E+09	3.118E+07		**0.35017%**	**0.0103%**	**0.0287%**	**0.0048%**	**0.3941%**		
									3.509E+07	**BTU/hr**	
Steam Enthalpy	Btu/lbm	1250.386	Steam	$\partial H_S/\partial T$	0.847	$\partial H_S/\partial P$	-0.11601		**10.28**	**MWt**	
Feedwater Enthalpy	Btu/lbm	439.904	Feedwater	$\partial H_{FW}/\partial T$	1.126*	$\partial H_{FW}/\partial P$	4.450E-04				
Makeup Enthalpy(110,2400)	Btu/lbm	84.235	Makeup	$\partial H_{MU}/\partial T$	0.989	Nom FW Flow	11070000	lb/hr (both SGs)			
Letdown Enthalpy(557,2230)	Btu/lbm	555.993	Letdown	$\partial H_{LD}/\partial T$	1.261	$\partial H_{LD}/\partial P$	-0.00182				
				* Not used in this spreadsheet							

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ATTACHMENT 2 – CALDON UNCERTAINTY INPUTS –PRELIMINARY VALUES

Page 2

2.0 SUMMARY

For Crystal River Unit 3, Revision 0 results are as follows:

1. The mass flow uncertainty approach is documented in Reference 3. The uncertainty in the LEFM✓+'s mass flow of feedwater is as follows:
   - Fully Functional LEFM✓+ system mass flow uncertainty is ± 0.32%
   - Maintenance Mode LEFM✓+ system mass flow uncertainty is ± 0.38%.

   Note: The LEFM✓+ system is in maintenance mode when only one of the two LEFM✓+ subsystems is fully functional, i.e., LEFM✓+ System is operating as an LEFM✓ System. The uncertainty of the LEFM✓+ when in maintenance mode may be re-evaluated and will likely be reduced after site specific hydraulic experience has been taken into account.
2. The uncertainty in the LEFM✓+ feedwater temperature is as follows:
   - Fully Functional LEFM✓+ system temperature uncertainty is ±0.57°F (±0.32°C)
   - Maintenance Mode LEFM✓+ system the uncertainty is ±0.57°F (±0.32°C)
3. The thermal power uncertainty approach is documented in Reference 3. The uncertainty in the calculation of thermal power due to the LEFM✓+ is as follows:
   - Fully Functional LEFM✓+ system thermal power uncertainty is ± 0.34%
   - Maintenance Mode LEFM✓+ system thermal power uncertainty is ± 0.40%.

   Note: Because some elements of the temperature uncertainty are systematic, the total power uncertainty due to the LEFM✓+ is *not* the root sum squares of the uncertainties due to items 1 and 2 above.
4. For an overall thermal power uncertainty analysis in which mass flow and temperature errors are to be treated separately (i.e., the thermal power uncertainty above is not used), the bounding mass flow and temperature errors must be divided into components that are systematically and randomly related to the mass flow error, as follows:
   - The fully operational LEFM✓+ systematic temperature error related to the mass flow error is ±0.07°F (±0.04°C) and the random temperature error related to the mass flow error is ±0.56°F (±0.31°C). The mass flow uncertainty remains at ±0.32%. The thermal power error due to this term is combined as the root sum square with other elements of the thermal power uncertainty.
   - The maintenance mode LEFM✓+ systematic temperature error related to the mass flow error is ±0.05°F (±0.03°C) and the random temperature error related to the mass flow error is ±0.57°F (±0.32°C). The mass flow uncertainty remains at ±0.38%. The thermal power error due to this term is combined as the root sum square with other elements of the thermal power uncertainty.

ATTACHMENT 3 TRANSCRIBED RC PUMP DATA

Impeller 0530

Pump Discharge pressure Indicated Press psig	Corr Press	Pump DP psid	flow gpm	WHP	BHP	Watt Reading	Eq HP Input	Volts	Amps	Pump Eff.	Temp
2242	2242.9	122.3	89350	6370	7170	5650	7560	6400	532	88.9	543
2236	2236.9	110.2	98750	6350	7170	5650	7560	6400	532	88.5	543
2223	2223.8	57.1	108500	6150	7110	5600	7500	6400	528	86.5	543
2193	2143.8	81.2	118800	5630	6800	5350	7160	6400	510	82.8	543
2153	2153.7	60.1	129700	4520	6350	5000	6700	6400	480	71.2	544
2254	2254.9	136.3	79500	6325	7110	5600	7500	6400	525	88.3	543
2256	2256.9	143.3	69650	5830	6900	5430	7280	6400	512	89.5	542
2246	2246.9	149.3	57950	5050	6540	5150	6900	6400	489	77.2	541
2245	2245.9	153.3	49030	4390	6410	5050	6760	6400	485	68.5	540
2264	2264.9	119.2	91300	6350	7150	5030	7540	6400	530	88.8	540

Impeller 0533

Pump Discharge pressure Indicated Press psig	Corr Press	Pump DP psid	flow gpm	WHP	BHP	Watt Reading	Eq HP Input	Volts	Amps	Pump Eff.	Temp
2312	2303.0	118.2	90750	6250	7010	5500	7370	6500	519	89.0	552
2320	2311.1	128.3	80600	6040	6890	5400	7240	6500	515	87.6	552
2321	2312.1	135.3	70800	5590	6630	5200	6970	6500	500	84.2	552
2307	2298.0	140.3	59790	4885	6250	4900	6560	6500	470	78.1	552
2306	2299.0	146.3	50700	4330	6120	4800	6430	6500	455	70.7	552
2317	2308.0	105.2	99950	6140	7010	5500	7370	6500	510	87.5	550
2275	2266.0	99.2	108850	5860	6630	5200	6970	6500	505	88.4	550
2248	1238.9	75.2	119550	5240	6630	5200	6970	6500	490	79.0	550
2223	2213.8	60.1	128150	4500	6250	4900	6560	6500	470	72.0	550
2306	2297.0	113.2	94700	6255	7010	5500	7370	6500	520	89.3	549

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Impeller 0531

Pump Discharge pressure Indicated Press psig	Corr Press	Pump DP psid	flow gpm	WHP	BHP	Watt Reading	Eq HP Input	Volts	Amps	Pump Eff.	Temp
2273	2270.5	109.8	95000	6090	7060	5540	7420	6780	517	86.2	562
2265	2262.5	104.8	100900	6190	7000	5500	7360	6780	518	87.6	562
2248	2295.5	94.8	109050	6030	6890	5400	7240	6780	510	87.5	562
2212	2209.6	75.9	119500	5290	6600	5175	6940	6770	485	80.0	562
2173	2170.6	59.8	128050	4965	6120	4800	6440	6800	448	73.0	561
2293	2290.4	129.7	79550	6015	6950	5450	7300	6800	510	86.5	560
2295	2292.4	136.7	69650	5560	6850	5370	7200	6800	500	81.3	560
2298	2295.4	142.7	59400	4950	6530	5130	6880	6800	480	75.8	560
2297	2294.4	146.7	49600	4250	6280	4940	6610	6800	460	67.7	560
2282	2279.4	115.7	42500	6220	7120	5575	7480	6790	519	87.4	559

Impeller 0532

Pump Discharge pressure Indicated Press psig	Corr Press	Pump DP psid	flow gpm	WHP	BHP	Watt Reading	Eq HP Input	Volts	Amps	Pump Eff.	Temp
2242	2239.5	108.8	97800	6210	7080	5550	7440	7000	525	87.8	562
2236	2233.5	105.8	100700	6220	7000	5490	7350	6950	510	88.7	562
2215	2212.6	89.8	111800	5855	6805	5340	7150	6950	500	86	562
2196	2193.6	75.8	120150	5310	6610	5180	6950	6950	485	80.3	561
2181	2178.6	64.8	127950	4820	6400	5040	6710	6975	470	75.4	560
2265	2262.5	127.8	80350	5960	6950	5450	7300	6975	510	86	560
2278	2275.4	136.7	70100	5600	6720	5270	7060	7000	490	83.2	560
2291	2288.4	143.7	60150	5040	6360	4990	6690	7000	465	79.1	560
2300	2297.4	147.7	49350	7240	6000	4700	6300	7000	440	70.7	561
2255	2252.5	114.8	92850	6210	7010	5500	7360	7000	515	88.6	562

BJ Test Data Uncertainties for ANO-1

FIGURE II
PUMP LOOP TEST INSTRUMENTATION LIST

PARAMETER	INSTRUMENT	RATED ACCURACY
Motor		
Kilowatt Input	Weston Polyphase Wattmeter	± 0.5%
Voltage	Weston Voltmeter Model 341	± 0.25%
Amperage	Weston Ammeter Model 370	± 0.25%
Current Transformers	Westinghouse Type CLA-10	
Potential Transformers	Westinghouse Type PTM-15	
Winding Temp.	Leeds & Northrop Ohmmeter & 10 RTD	± 0.5%
Bearing Temp.	Honeywell Recorder and T.C.	± 1.5°F.
Cooling Water Flow	System Calibration ΔP vs Q	± 2%
Cooling Water Temp.	Honeywell Recorder and T.C.	± 1.5°F
Pump Flow		
Venturi (four)	Herschel Type 24" x 17-1/2"	± 1%
	Validyne X-Ducer Mod. DP15	± 0.5%
	Barton ΔP Unit Model 200	± 0.5%
Static Pressures		
Pump Suction and Discharge	Crosby Bourdon Tube Type	± 0.5%
	Roylyn Coil Tube Type	± 0.25%
System Loop Water Temp.	Bailey Recorder and 100 ΩRTD	± 0.5%
Pump Speed	Hewlett Packard Elec. Counter	± 0.5%
Pump Seals		
Controlled Bleed-Off Flow	Fischer Porter Float Type	± 2%
Controlled Bleed-Off Temp.	Honeywell Recorder and T.C.	± 1.5°F
Seal Cavity Pressure (3)	Bailey Recorder Model E101	± 0.5%
Seal Cavity Temp.	Honeywell Recorder and T.C.	± 1.5°F
Cooling Water Flow	Fischer Porter Float Type	± 2%
Cooling Water Temp. Inlet and Outlet	Honeywell Recorder and T.C.	± 1.5°F
Differential Pressure Across Pump	Valydyne X-Ducer Mod. DP. 15	± 0.5%
	Bailey ΔP Unit Model 200 -0-150PSI	± 0.5%
Additional Static Pressure measurements	Crosby Bourdon Tube Type	± 1.0%
Additional Temperature Measurements	Honeywell Recorder and T.C.	
	Up to + 200°F	± 1.5°F
	200°F to 700°F	± 0.75%
Pump and Motor Vibration	B & K Accelerameter	± 2%
	I.R.D. Model 306 Meter	± 2%

BJ Test Data Uncertainties for Davis Besse

Exerpt From Byron Jackson Report
On Davis Besse Pump Tests
June 1, 1973

PARAMTER	MFG.	ACCURACY
Killowatt Input	Weston	0.5%
Pump Capacity	Barton (DP)	0.75%
	Byron Jackson (Venturi)	0.5%
Loop Water Temperature	Baily Meter	0.5%
Static Pressure		
Pump Suction	Roylyn Linsay	0.25%
Pump Discharge	Roylyn Linsay	0.25%

BORG-WARNER CORPORATION

PUMP TEST DATA

PUMP SIZE AND TYPE	GUARANTEED PUMPING CONDITIONS	
33 x 33 x 38 DFSS	Field speed: R.P.M. 1180; G.P.M. 90670; T.D. HEAD 358; SP. GR. 0.74	TESTED BY: MA FJ
Impeller drawing: R-3823; Diameter: 34 1/16 x 37 3/16; Vanes: 5; Area under: 5/8 x 8; Stage: 1	Test speed	WITNESSED BY:
	DIFF. VEL. HD. BETWEEN GAUGES AT 36 I.D. DISCH. AND 36 I.D. SUCT. = 0 FT. AT — G.P.M.	FOR: BABCOCK & WILCOX - TOLEDO
	TEST MOTOR NO. 9000 H.P. 6 POLES 13200 VOLTS	CUSTOMER'S ORDER NO. 022247 LW
	MOTOR EFF.	ITEM NO. — JOB NO. 701-N-0240
	BAROMETRIC PRESSURE IN. HG.	NOTES: Mtr S/N 15-81P34
		HOT WATER PERFORMANCE

Reading No.	Suct. Hd. Gauge #31 3000# Psi Read	Suct. Psi Corr.	Discharge Gauge #35 3000# Psi Read	Discharge Psi Corrected	Diff. Press. Δpsi	Elevation Head	Differential Velocity Head	Total Dynamic Head in Feet of Water	Capacity 36x24 Venturi Head	Capacity G.P.M.	Water Horse Power	Brake Horse Power	Watt Meter Reading	Equiv. KW Input	Volts	Amps	Pump Efficiency	Water Temp. °F	Pump Speed RPM
1	2042	2042.5	2195	2195.8	133.3	0	0	412	15.8	78500	6105	7260	5650	7570	13090	250	84.1	539	
2	2050	2050.5	2171	2171.7	121.2			379	21.0	90500	6400	7290	5700	7640	13050	280	87.7	538	
3	2051	2051.5	2161	2161.7	110.2			340	25.2	99200	6370	7160	5600	7500	13050	280	89.0	538	
4	2023	2023.5	2123	2123.6	100.1			309	30.5	109100	6360	7125	5570	7470	13080	270	89.3	538	
5	2017	2017.4	2103	2103.6	86.2			266	36.5	119500	6000	6910	5400	7240	13090	260	86.9	538	
6	2011	2011.4	2085	2085.6	74.2			229	41.0	126500	5475	6630	5180	6950	13100	250	82.5	538	
7	2020	2020.5	2174	2174.8	154.3			454	8.7	56900	5110	6860	5370	7200	13050	260	74.5	505	
8	2020	2020.5	2153	2153.7	133.2			392	17.7	81000	6290	7420	5800	7770	13050	290	84.7	500	
9																			
10	2030	2030.6	2155	2155.7	125.1			369	22.5	91700	6690	7480	5850	7840	13050	280	84.4	503	

BHP @ 540°F

7190
7205
7085
7050
6840
6560
6500
7000
7070

THE ONLY DATA USED FROM THE PAGE WAS INTERPRETED CORRECTLY FROM THE ORIGINAL COPY OF THIS PAGE AND IS SHOWN BELOW

Impeller ID 32-9042687-001 Pg 32

0240	Corrected to temp		Test Data	
GPM	556	558	BRAKE HP	Temp (F)
78500	7094.77	7074.45	7260	539
90500	7114.94	7094.56	7290	538
99200	6988.06	6968.05	7160	538
109100	6953.91	6933.99	7125	538
119500	6744.07	6724.75	6910	538

[signature] Date 4/12/07
D.S. Ulerich

DATE OF TEST 10-27-72

PUMP SERIAL No. 701-N-0240

T-33255-1

P-366 12/71

PEF-CR3-0190

PUMP TEST DATA

Pump size and type	33 x 33 x 38 DFSS
Impeller drawing	R-3323
Diameter	34 1/4 x 37 5/8
Vanes	5
Fillets	5/8 x 3
Other	1

Guaranteed pumping conditions

	R.P.M.	G.P.M.	T.D. Head	Pump Eff.	Overall Eff.	BHP	Sp. Gr.
Field speed	1186	96000	358				0.74
Test speed							

Diff. vel. hd between gauges at 36 I.D. disch. and 36 I.D. suct. = 0 ft at — G.P.M.

Test motor no. 9000 H.P. 6 poles 13200 volts

Motor eff.

Barometric pressure — in. Hg.

Tested by: MA

For: BABCOCK & WILCOX

Customer's order no. 022297-LW

Job no. 701-N-0241

Notes: MTR S/N 15-81834

HOT WATER PERFORMANCE

Reading No.	Suct. Hd. Gauge #5 (5000) PSI Read	Suct. Hd. PSI Corr.	Discharge Head Gauge #22 (3000 #) PSI Read	Discharge PSI Corrected	Diff. Press. Δ PSI	Elevation Head	Differential Velocity Head	Total Dynamic Head in Feet of Water	Capacity (Measured with 36 x 24 Venturi) Read	G.P.M.	Water Horse Power	Brake Horse Power	Watt Meter Reading	Equiv. H.P. Input	Volts	Amps	Pump Efficiency	Water Temp. °F	Pump Speed RPM
1	2117	4.6	2228	2228.8	111.2	0	0	356	21.8	94000	6090	6970	5450	7300	13100	240	87.3	564	1189
2	2118	4.6	2222	2222.8	104.2			333	25.0	100700	6100	6950	5425	7270	13050	240	87.6	564	1189
3	2114	4.6	2205	2205.8	91.2			292	30.2	110000	5880	6820	5335	7150	13050	240	86.1	565	1190
4	2117	4.6	2196	2196.8	79.2			255	35.4	120000	5550	6655	5200	6965	13070	240	83.4	565	1190
5	2116	4.6	2182	2182.8	66.2			211.5	40.5	127800	4930	6330	4950	6630	13090	230	77.9	562	1191
6	1 STOP START AT RUNOUT FLOW SPIN DOWN TIME 2 MIN - 48 SECS																		
7	1982	4.4	2138	2138.7	156.3	0	0	466	4.3	40000	3655	6590	5150	6900	13070	250	55.5	510	1190
8	2006	4.4	2154	2154.7	148.3			441.0	11.0	64300	5550	7000	5475	7340	13050	260	79.4	[illegible]	1189
9	2043	4.5	2180	2180.7	137.2			410	15.6	76700	6140	7200	5640	7540	13030	260	85.3	512	1188
10	2033	4.5	2148	2148.0	115.2			345	24.8	97000	6520	7330	5730	7680	13050	270	89.0	515	1188
11	1 STOP - START AT DESIGN FLOW SPIN DOWN TIME - 3 MIN - 05 SECS																		

BHP @ 558 °F

7160
7140
7030
6850
6480
6290
6690
6900
7030

THE ONLY DATA USED FROM THE PAGE WAS INTERPRETED CORRECTLY FROM THE ORIGINAL COPY OF THIS PAGE AND IS SHOWN BELOW

Impeller ID 0241 — **32-9042687-001** — **Pg** 33

GPM	Corrected to temp 556	Corrected to temp 558	Test Data BRAKE HP	Temp (F)
76700	6811.80	6792.29	7200	512
94000	7052.03	7031.83	6970	564
100700	7042.40	7022.23	6950	565
110700	6910.67	6890.88	6820	565
120000	6743.48	6724.16	6655	565

D.S. Ulevich Date 4/12/07

PUMP SERIAL No. 701-N-0241

T-33340-1

PEF-CR3-0191

BORG WARNER — BW Jackson Pump Division

PUMP TEST DATA

Pump size and type: 33 x 33 x 35 DFSS

Impeller drawing: R 3823; Diameter: 34 [illegible] 5; Impeller pattern: 5 [illegible] D

Guaranteed pumping conditions: 178 | 90000 | 358 | [illegible] | 074

Diff. ht. between gauges at 36 D. disch. and 36 D. suction: 0

Test motor no.: 9000 HP 6 poles 13200 volts

Tested by: [illegible] DX

Witnessed by: [illegible]

For: BABCOCK & WILCOX — TOLEDO

Customer's order no.: 022297-CWD

Job no.: 701-N-0242

Notes: MTR S/N 15-81P35

HOT WATER PERFORMANCE

Suct. HD. Gauge 3000# PSI Read	Suct. PSI Read (corr.)	Disch. Gauge 3000# PSI Read	Disch. PSI Corrected	Diff. Press. 0–150 ΔP Barton ΔPSI	Elevation Head	Velocity Head Differential	Total Head ft	Capacity (36x24 Venturi) Read	Capacity GPM	Water Horse Power	Brake Horse Power	Watt Meter Reading	Equiv. H.P.	Volts	Amps	Pump Efficiency	BHP 568.0°F	Water Temp °F	Pump Speed RPM
2174	2159.4	2280	2278.4	119.5	0	0	380	19.2	87850	6115	7100	5550	7440	13160	250	86.2	7240	559	1188
2177	2162.4	2277	2275.4	113.5			361	22.0	94000	6220	7100	5550	7440	13160	250	87.6	7240	559	1188
2176	2161.4	2271	2269.4	102.2			341	24.4	99000	6200	7045	5500	7365	13130	250	88.1	7180	559	1189
2172	2157.4	2251	2249.4	91.5			291	25.4	110300	5890	6910	5400	7240	13160	250	85.1	7050	559	1188
2172	2157.4	2240	2238.4	81.0			257.5	35.8	118600	5600	6660	5210	6980	13160	240	84.0	6800	559	1189
2172	2157.4	2225	2223.4	66.5			212	41.0	128300	4990	6250	4890	6550	13160	220	79.9	6370	559	1190
2 STOPS — 2 STARTS AT RUNOUT FLOW										SPIN DOWN TIMES 2 MIN 37 SECS								2 MIN 50 SECS	
2085	2070.4	2225	2223.4	153.0	0	0	456	7.0	51400	4590	6810	5330	7140	13100	240	67.4	6500	510	1188
2085	2070.4	2221	2219.4	149.5			447	8.5	56700	4950	6890	5390	7210	13100	250	71.7	6610	515	1188
2085	2070.4	2212	2210.4	140.5			422	12.8	70000	5720	7125	5570	7460	13100	250	80.3	6885	520	1188
2085	2070.4	2204	2202.4	132			400	16.7	80000	6150	7260	5670	7600	13100	260	84.6	7055	525	1188
		2198	2196.4	126.5			386	19.0	84500	6250	7260	5650	7560	13100	260	86.2	7070	520	1188

[illegible] WITH BARTON METER #200-903

WITH SUCTION AND DISCHARGE GAGES

THE ONLY DATA USED FROM THE PAGE WAS INTERPRETED CORRECTLY FROM THE ORIGINAL COPY OF THIS PAGE AND IS SHOWN BELOW

Impeller ID 0242 — 32-9042687-001 Pg 34

	Corrected to temp		Test Data	
GPM	556	558	BRAKE HP	Temp (F)
80000	6972.638	6952.665	7260	525
87850	7130.715	7110.289	7100	559
99000	7075.477	7055.209	7045	559
110300	6939.893	6920.014	6910	559
118600	6688.812	6669.651	6660	559

[signature] Date 4/12/07

D.S. Ulevich

SERIAL No. 701-N-0242

T-33522-1

DATE OF TEST

P-366 12/71

PEF-CR3-0192

PUMP TEST DATA

PUMP SIZE AND TYPE: 33x33x38 DFSS

IMPELLER DRAWING	DIAMETER	VANES	UNDER	FILLED	TRIM
P-3863	37 3/4 x 37 1/2	5	3/16 x 3		1

GUARANTEED PUMPING CONDITIONS

	R.P.M.	G.P.M.	T.D. HEAD				SP. GR.
FIELD SPEED	1186	90670	358				.071
TEST SPEED							

DIFF. VEL. HD BETWEEN GAUGES AT 36 I.D. DISCH. AND 36 I.D. SUCT. = 0 FT AT G.P.M.

TEST MOTOR NO. 9000 H.P. 6 POLES 13200 VOLTS

MOTOR EFF.

BAROMETRIC PRESSURE IN. HG.

TESTED BY: MA-HT

WITNESSED BY:

FOR: BABCOCK & WILCOX TOLEDO

CUSTOMER'S ORDER NO. 022297 LW

ITEM NO.

JOB NO. 701-N-0243

NOTES: MTR S/A 81P0035-L

HOT WATER PERFORMANCE

READING NO.	SUCT. HD. GAUGE #31 3000# PSI READ	PSI CORR	DISCHARGE HEAD GAUGE #32 3000# PSI READ	PSI CORRECTED	DIFF PRESS 0150 (BARTON) Δ PSI	ELEVATION HEAD	DIFFERENTIAL VELOCITY HEAD	TOTAL DYNAMIC HEAD IN FEET OF WATER	CAPACITY MEASURED WITH 36x24 VENTURI HEAD	G.P.M.	WATER HORSE POWER	BRAKE HORSE POWER	WATT METER READING	EQUIV. H.P. INPUT	VOLTS	AMPS	PUMP EFFICIENCY	PHT SP GR = 0.71	WATER TEMP. °F	PUMP SPEED RPM
1	2121	2123	2241	2240	117.0	0	0	366	61.35	91960	6255	7080	5525	7410	13000	270	88.5	7160	549	1189
2	2120	2122	2229	2228	106.0			332	46.10	101300	6260	7120	5560	7460	13000	270	88.0	7150	549	1189
3	2118	2120	2217	2216	96.0			301	30.4	109450	6140	7070	5525	7410	13000	270	86.9	7100	549	1189
4	2118	2120	2204	2203	82.5			258.5	36.1	119300	5750	6820	5325	7140	13000	260	84.4	6850	549	1190
5	2116	2118	2191	2190	71.5			224	46.1	127500	5310	6660	5210	6980	13000	250	79.7	6690	549	1190
6																				
7	2095	2097	2247	2246	150.0	0	0	455	6.6	50400	4405	6550	5110	6850	13100	250	67.4	6360	525	1191
8	2097	2099	2247	2246	146.0			444	9.6	60900	5170	6800	5310	7120	13100	260	76.0	6640	528	1190
9	2098	2100	2240	2239	139.5			426	12.4	69100	5605	6910	5400	7240	13100	260	81.1	6790	532	1189
10	2100	2102	2233	2232	129.3			397	17.0	81250	6110	7120	5560	7460	13100	270	85.8	7030	535	1189
11	2110	2112	2234	2233	121.0			374	20.6	89750	6350	7210	5630	7550	13100	280	87.9	7140	558	1189
12																				

DETERMINED WITH BARTON METER #200-903

THE ONLY DATA USED FROM THE PAGE WAS INTERPRETED CORRECTLY FROM THE ORIGINAL COPY OF THIS PAGE AND IS SHOWN BELOW

Impeller ID 0243 — 32-9042687-001 Pg 35

GPM	Corrected to temp 556	558	Test Data BRAKE HP	Temp (F)
81250	6922.59	6902.76	7120	535
91900	7011.214	6991.131	7080	549
101300	7050.826	7030.629	7120	549
109450	7001.311	6981.256	7070	549
119300	6250.086	6232.182	6820	549

D.S. Ulevitch Date 4/12/07

PEF-CR3-0193

PUMP SERIAL No. 701-N-0243

T-33547-1

P-366 12/71

ATTACHMENT 4 ATMOSPHERIC PRESSURE DATA FROM CR-3

The attached data gives a high over a 7 year period from Ocala and Gainesville Florida and 4 years from Holder Florida. The high for that period was 0.67 in Hg. Use this number. It is the only number that can be defended based on engineering data.

14.7 2.036022 29.930

OCF	low "Hg	high "Hg	avg "hg	range "Hg	neg "Hg	pos "hg	neg psi	pos psi
2000	29.66	30.49	30.04	0.83	-0.27	0.56	-0.13	0.28
2001	29.54	30.50	30.16	0.96	-0.39	0.57	-0.19	0.28
2002	29.67	30.55	30.13	0.88	-0.26	0.62	-0.13	0.30
2003	29.57	30.54	30.10	0.97	-0.36	0.61	-0.18	0.30
2004	28.94	30.55	30.15	1.61	-0.99	0.62	-0.49	0.30
2005	29.50	30.59	30.16	1.09	-0.43	0.66	-0.21	0.32
2006	29.66	30.53	30.13	0.87	-0.27	0.60	-0.13	0.29
GNV								
2000	29.60	30.49	30.02	0.89	-0.33	0.56	-0.16	0.28
2001	29.60	30.50	30.15	0.90	-0.33	0.57	-0.16	0.28
2002	29.63	30.56	30.15	0.93	-0.30	0.63	-0.15	0.31
2003	29.54	30.53	30.10	0.99	-0.39	0.60	-0.19	0.29
2004	29.11	30.56	30.12	1.45	-0.82	0.63	-0.40	0.31
2005	29.45	30.60	30.15	1.15	-0.48	0.67	-0.24	0.33
2006	29.63	30.55	30.12	0.92	-0.30	0.62	-0.15	0.30
Holder								
2003	29.68	30.55		0.87	-0.25	0.62	-0.12	0.30
2004	28.79	30.55		1.76	-1.14	0.62	-0.56	0.30
2005	29.51	30.59		1.08	-0.42	0.66	-0.21	0.32
2006	29.66	30.52		0.86	-0.27	0.59	-0.13	0.29
				Max	-1.14	0.67	-0.56	0.33

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT F

LOCA MASS and ENERGY RELEASES

LOCA Mass and Energy Releases – Containment Response

1.0 DISCUSSION

The CR-3 LOCA mass and energy (M&E) and containment pressure temperature response was recalculated based on a core power level of 102% of 2568 MWt (2619.4 MWt) which bounds the MUR. The calculations follow the NRC-approved methodology described in BAW-10252P-A (Reference F.1). The RELAP5/MOD2-B&W computer code (Reference F.2) was used to generate the M&E release data and the Gothic Code (Reference F.3) was used for the containment pressure and temperature response.

The RELAP5/MOD2-B&W digital computer code was used to generate the blowdown and refill portions of the transient. The base model that was used in these analyses was identical to the model that is used to calculate the fuel clad response to the postulated spectrum of break sizes to demonstrate compliance with 10CFR 50.46. Modifications to the base model were made to maximize the blowdown mass and energy release data in compliance with NRC guidance as described in Appendix A of BAW-10252P-A. The calculation complies with the limits and restrictions that have been placed on the approved topical reports.

NRC approved methods and tools were used in this analysis. The results of these calculations will be incorporated in the CR-3 FSAR. The key input assumptions include:

- no SG tube plugging
- hot expanded volumes in the RCS
- minimum and maximum Emergency Core Cooling System flow rates cases
- offsite power available and loss of offsite power cases
- reactor coolant pumps powered and delayed pump trip cases
- nitrogen entering the RCS and the containment via emptying of the core flood tanks

The data was generated for the same postulated spectrum of breaks evaluated for peak containment pressure as listed the FSAR. A number of additional sensitivity studies, as described above, and hot leg breaks at the SG inlet were analyzed.

The key boundary conditions imposed on the calculations to generate the mass and energy release data is provided in Table F-1.

Table F-1. KEY PARAMETERS for LOCA MASS AND ENERGY RELEASE

Parameter	Value
Initial Core Power	2620 MWt
Initial RCS Average Temperature	579 °F
Initial RCS Pressure (hot leg)	2170 psia
Initial Pressurizer Level	220 inches
BWST Temperature	120 °F
CFT Liquid Temperature	130 °F

Mass and energy release data for the spectrum of break sizes and locations were calculated for the first 600 seconds of the transient. The limiting break location that results in the maximum pressure and temperature response were the double-ended breaks of the hot leg piping. The limiting case was a double-ended break at the SG inlet and it resulted in a peak pressure and temperature of 68.74 psia and 276.6 F, respectively. However, the variation in peak pressure for all the hot leg break sizes was only 1.4 psi, 67.35 to 68.74 psia. The cold leg break locations resulted in a 3 to 4 psi lower peak pressure as compared to the hot leg breaks. The peak pressure and temperatures for all of the cases that correspond to the FSAR results are shown in Table F-2. A plot of the pressure and temperature response for the limiting cases is shown in Figures F-1 and F-2. The M&E release data for the limiting hot leg break case is provided in Table F-3. The data includes both average and integrated mass and energy release with averaged instantaneous enthalpies for each of the selected breaks.

This mass and energy release rate data, generated with the RELAP5/MOD2-B&W code, was used to develop the containment building pressure and temperature response using the GOTHIC code. The same containment parameter assumptions were used as described in Table 14-45 of the FSAR. The exception is that painted internal steel surface area was increased from the FSAR value of 106,110 ft^2 to 212,220 ft^2. This value is still conservative relative to the estimated value of 409,817 ft^2 identified in Reference F.4. A comparison to the existing peak containment pressure reported in the FSAR indicates the calculated peak pressure to be 54.04 psig (68.74 psia), which is less that the current design limit of 55 psig (69.7 psia) and less the current calculated peak pressure of 54.2 psig (68.9 psia) reported in the FSAR.

REFERENCES

F.1 BAW-10252P-A, Revision 0, "Analysis of Containment Response to Postulated Pipe Failures Using GOTHIC."

F.2 BAW-10164P-A, Revision 4, "RELP5/MOD 2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

F.3 "GOTHIC – Containment Analysis Package Technical Manual", Version 7.2, NAI 8907-06, Rev 15, EPRI, Palo Alto, CA, September 2004.

F.4 Gilbert and Associates, Inc, Report No. 1889, "ECCS Passive Heat Sink Data and Information."

Table F-2: Summary of Results of all cases

(Peak Pressure Allowed = 69.7 psia, Peak Temperature Allowed = 281 F)

Break Location	Break Size (ft^2)	Peak Pressure (psia)	Peak Temperature (°F)	Time of Peak Pressure / Temperature (sec)
CLPD	8.6822[1]	64.51	271.0	17.6 / 17.4
CLPS	8.6822[1]	65.11	271.9	19.2 / 19.2
	7.0	65.01	271.7	20.0 / 20.0
	5.13	64.89	271.5	23.0 / 23.0
	3.0	64.28	270.7	31.0 / 31.0
	2.0	64.50	270.6	47.0 / 46.0
	0.5	62.30	267.4	139.0 / 139.0
HL at RV	14.352[1]	68.22	276.0	16.0 / 15.8
HL at SG	14.352[1]	68.74	276.6	17.2 / 17.2
	11.0	68.42	276.2	16.8 / 16.6
	8.55	68.27	276.0	18.8 / 18.8
	5.0	67.35	274.8	24.0 / 24.0

[1] The original FSAR mass and energy release data was based on a model that contained nominal (cold) dimensions, 8.55 ft^2 for a double-ended break in the cold leg piping and 14.14 ft^2 for the hot leg piping. The revised data is based on a model with hot expanded dimensions.

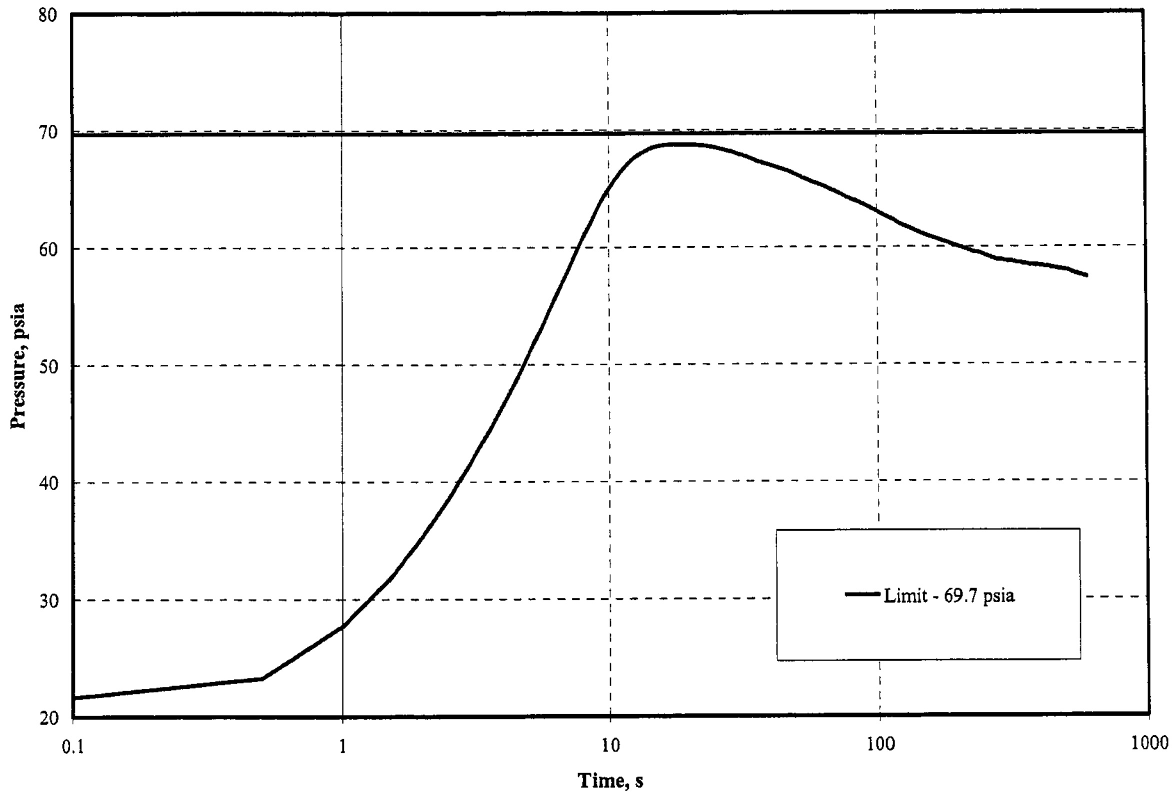


Figure F-1 – GOTHIC Predicted Peak Pressure for a Double-Ended Break at the SG Inlet

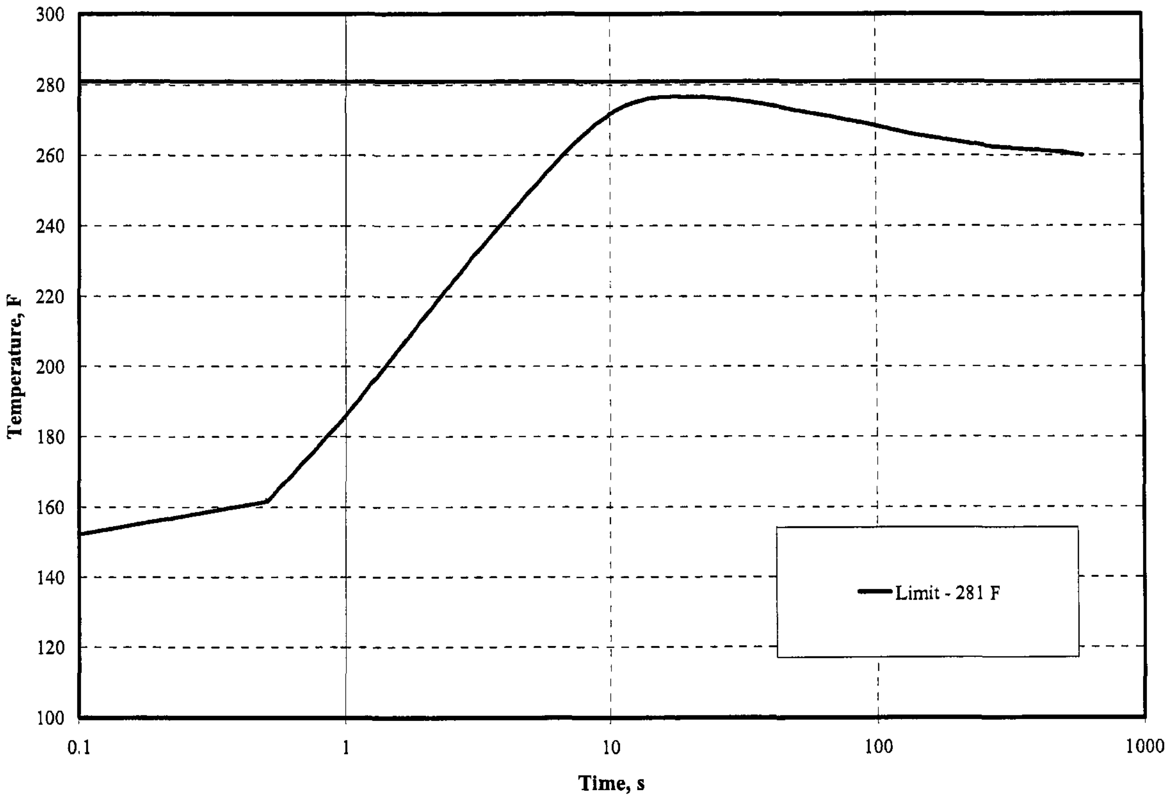


Figure F-2 – GOTHIC Predicted Peak Temperature for a Double-Ended Break at the SG Inlet

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT G

ATWS MITIGATION SYSTEM ACTUATION CIRCUITRY (AMSAC) ARMING SETPOINT EVALUATION

ATWS Mitigation System Actuation Circuitry (AMSAC) Arming Setpoint Evaluation

1.0 DISCUSSION

In response to 10 CFR 50.62, the ATWS rule, the B&W-designed plants were required to install a diverse scram system (DSS) that was independent of the existing reactor trip system. In addition, a means to initiate emergency feedwater and trip the turbine was required. At CR-3, the ATWS Mitigation System Actuation Circuitry (AMSAC) was installed. The current arming setpoint is based on 50% of 2544 MWt. To ensure that the AMSAC system arming setpoint remains valid for the MUR power level, a new analysis was performed based on a core power level of 52% of 2609 MWt. The limiting ATWS transient for the B&W-designed plant is a loss of feedwater initiated event. At a lower power level, the AMSAC system would not be armed. Therefore, the purpose of the transient is to demonstrate that the without AMSAC actuation, the peak pressure will not exceed 3250 psia.

The RELAP5/MOD2-B&W computer code (Reference G 2.1) was used to reanalyze the loss of feedwater AMSAC transient. RELAP5/MOD2-B&W has been reviewed and approved for the safety analyses for the once-through steam generator plants, Reference G.2.2. The base model was the same model used for the current loss of main feedwater transient described in the CR-3 FSAR, Section 14.2.2.9. The RELAP5/MOD2-B&W model contains a detailed representation of the CR-3 nuclear steam supply system (NSSS) and includes:

- Reactor vessel and core
- Hot legs, cold legs, and reactor coolant pumps
- Pressurizer
- Main and emergency feedwater
- Steam generators
- Main steam piping and valves
- Reactor protection system/DSS

The model was reinitialized to an initial core power level of 52% of 2609 MWt for this analysis. A list of the other key input parameters and initial conditions are provided in Table G-1.

Table G-1: Key Inputs and Boundary Conditions

Parameter	LOFW ATWS for AMSAC Arming Value
Core Power (MWt) (52% of 2609 MWt)	1356.68
Core Decay Heat	1.0*ANS71 + B&W actinides
RCP Heat Addition, net (MWt per pump)	4.0
Average RCS Temperature (°F)	579
RCS Pressure (psig) – hot leg	2155
Initial Pressurizer Level (in)	220
Spray Setpoint - on/off (psig)	2205/2155
Spray Capacity - design (gpm)	190
PORV Setpoint - open/close (psig)	2450 / 2380
PORV Capacity - nominal (lbm/hr) @ 2450 psig + 3% accumulation	148,306 @2538 psia
Number of Pressurizer Code Safety Valves (PSV)	2
PSV Setpoint - nominal (psig)	2500
PSV Setpoint Tolerance (%)	+2.0 / -4.0
PSV Capacity - nominal (lbm/hr/valve)	317,973 @2750 psig
MSSV Setpoint Lift Tolerance (%)	+1.0
MSSV Accumulation (%)	+3.0
MSSV Blowdown (%)	-0.0
Tube Plugging - average (%)	20
MFW coastdown time (sec)	7
Min. EFW Flow Rate (gpm)	550
EFW Delay Time (sec)	60
Low SG level setpoint for EFW actuation (in. above UFLTS)	6
Moderator Temperature Coefficient (ΔK/K/°F)	$+0.315\times10^{-4}$
Doppler Coefficient (ΔK/K/°F)	-1.17×10^{-5}

Parameter	LOFW ATWS for AMSAC Arming Value
Group 5-7 Insertable Worth for DSS trip (%ΔK/K)	1.7
RPS High RCS Pressure Setpoint (psia)	N/A
DSS High RCS Pressure Setpoint (psia)	2464.7
DSS High RCS Pressure Response Time (s)	2.0
Offsite Power	Available
Single Failure	RPS
Operator Actions	None

A 120-second LOFW ATWS event transient with the AMSAC disabled was simulated with RELAP/MOD2-B&W to confirm that the peak RCS pressure does not exceed the conservative estimate of the ASME Service Level C limit, 3250 psia. A loss of main feedwater was simulated at time zero with a 7-second coastdown. The RCS temperature and pressure begin to increase due to the reduction in heat transfer. The RCS reaches the DSS trip setpoint and the PORV open setpoint within the first 20 seconds. The peak RCS pressure reached a maximum of 2616.0 psia at approximately 22.9 seconds, which is below the pressure limit of 3250 psia. The EFIC low SG level EFW actuation setpoint is reached at about 33 seconds with EFW available 60 seconds later. These results confirm that the AMSAC is not required at or below 52% of 2609 MWt to prevent the RCS pressure from exceeding ASME acceptance criteria. Therefore, the 50% power arming setpoint remains valid for CR-3. The sequence of events for this transient is provided in Table G-2. A plot of the RCS pressure and temperature responses are provided in Figures G-1 and G-2. A second case was also run assuming that the DSS was not available. In this case, the peak RCS pressure was 3164.9 psia, which is also less than the 3250 psia limit. The pressure response for the second case is included as Figure G-3.

Table G-2: Sequence of Events – LOFW ATWS w/ AMSAC Disabled

Parameter	Transient Time, seconds
Loss of Main Feedwater	0.0
MFW terminated	7.0
Pressurizer Spray flow begins	~13.0
DSS High RCS Pressure signal	18.6
PORV initial lift	~19.0
Control Rods begin to fall	20.6
Peak RCS Pressure (2615.5 psia)	22.9
SG Level Reaches 6 inches, EFIC Setpoint	~33.2
EFW initiated SG-A/SG-B	~93.2
Peak RCS Average Temp.	~97.0
Analysis Terminated	120.0

G.2.0 REFERENCES

G.2.1 BAW-10164P-A, Revision 4, "RELP5/MOD 2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

G.2.2 BAW-10193P-A, Revision 0, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors."

Figure G-1: RCS Pressure for LOFW ATWS for AMSAC with DSS at 52% Power

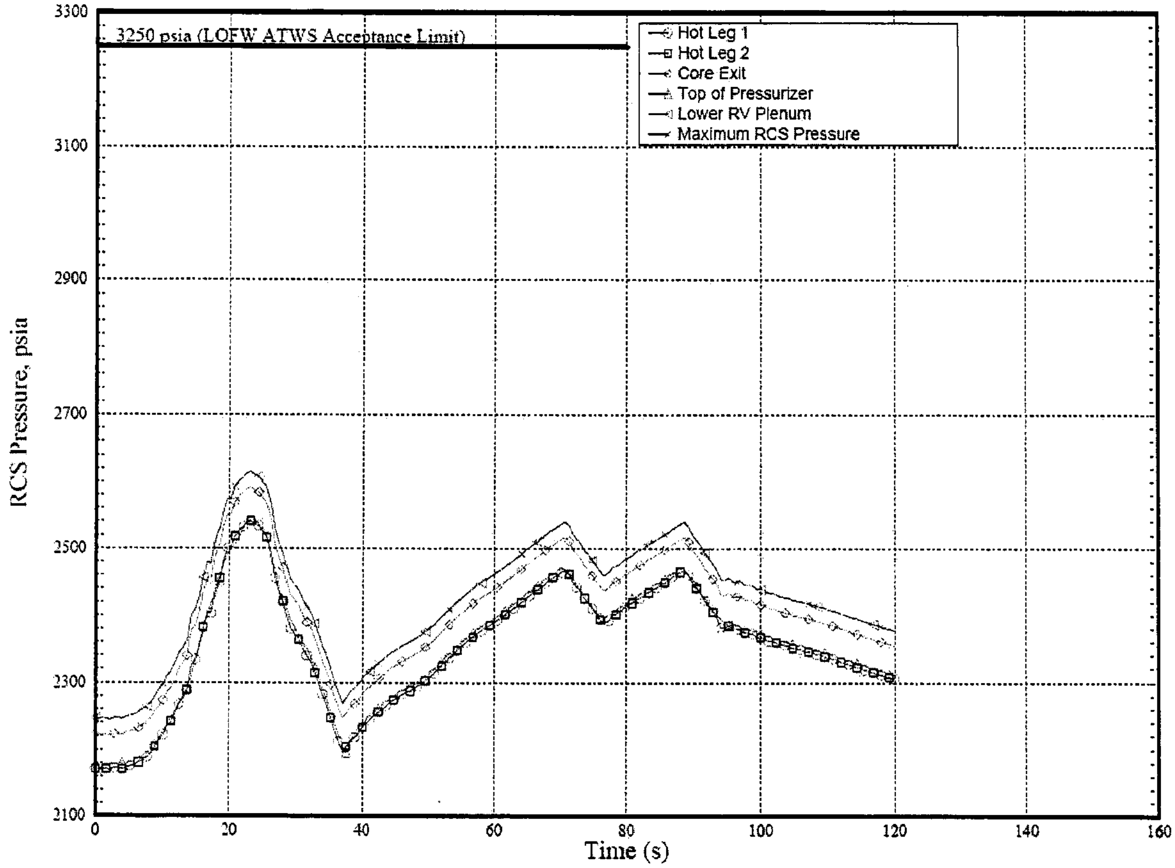


Figure G-2: RCS Temperature for LOFW ATWS for AMSAC with DSS at 52% Power

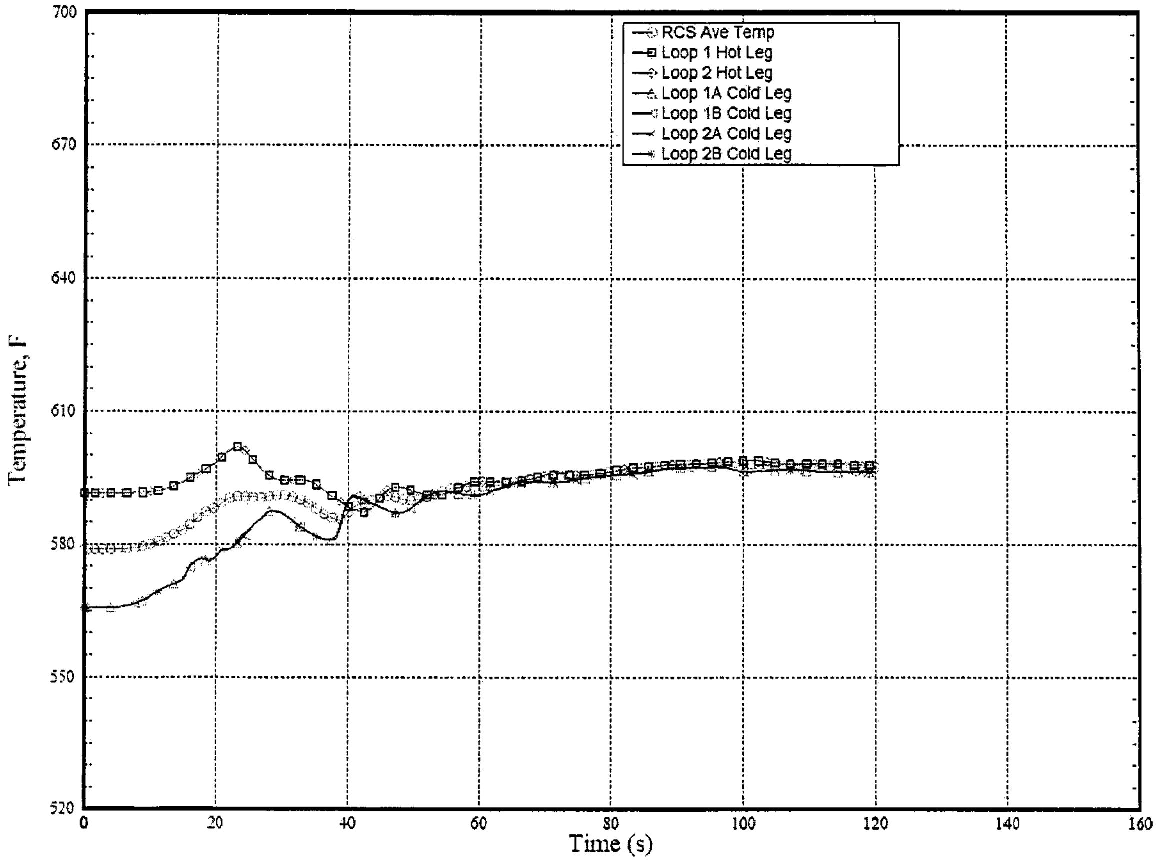
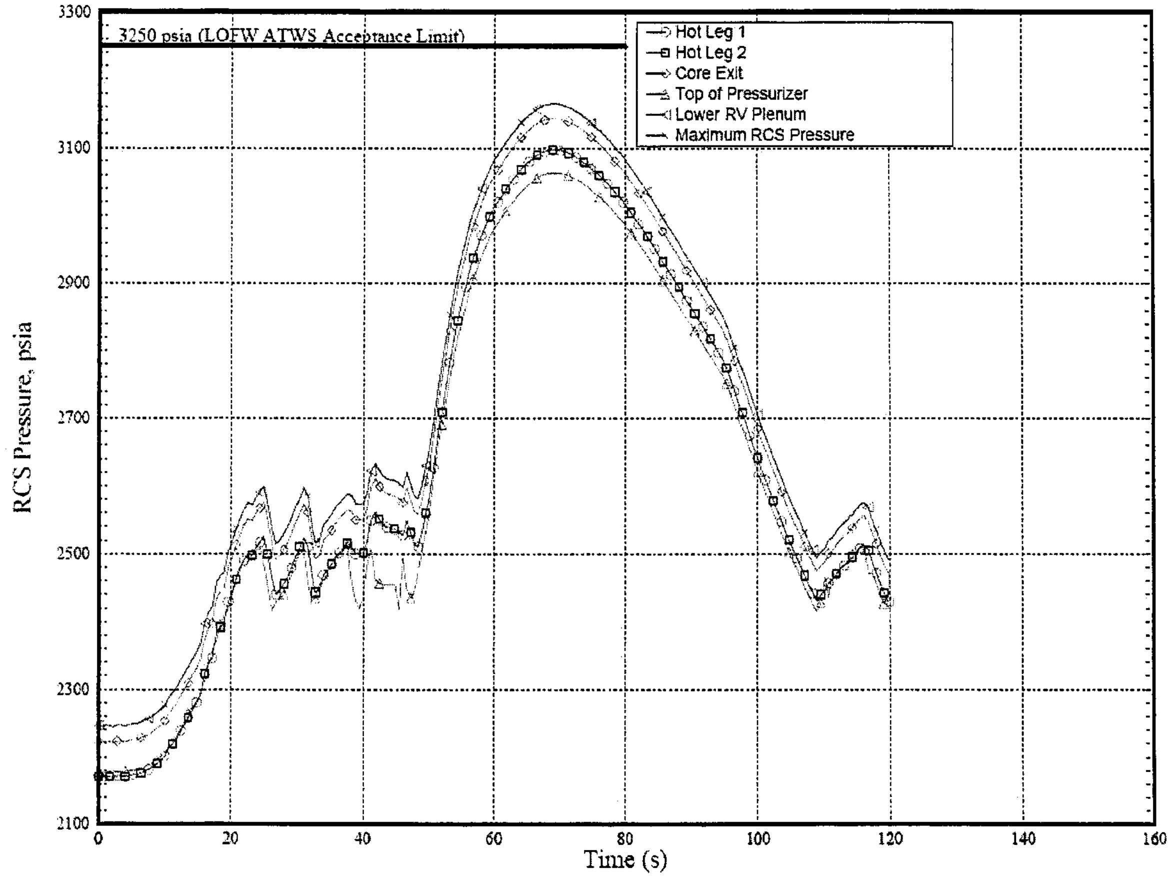


Figure G-3: RCS Pressure for LOFW ATWS for AMSAC without the DSS at 52% Power



FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 0

ATTACHMENT H

LIST OF REGULATORY COMMITMENTS

List of Regulatory Commitments

The following table identifies those actions committed to by Florida Power Corporation (FPC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Paul Infanger, Supervisor, Licensing & Regulatory Programs at (352) 563-4796.

Regulatory Commitments	Due Date
Administrative controls will be added to CP-500 for situations where the LEFM CheckPlus™ system or other specific heat balance uncertainty inputs is unavailable. These controls will address maximum power and inputs into the heat balance calculation as well as the allowed outage time for the LEFM CheckPlus™ system to be inoperable. This requirement will state that if either LEFM or any low-uncertainty heat balance input parameters are inoperable, then reduce power to $\leq$ 2568 MWt within 12 hours and reduce the nuclear overpower - high setpoint to $\leq$ 103.3% RTP within 48 hours. Should the LEFM system become unavailable, the current flow nozzle-based feedwater flow and RTD feedwater temperature instrumentation will be used as input to the core power calorimetric, and the core power will be limited to the current licensed power level of 2568 MWt.	prior to implementation
A grid reliability study will be completed and submitted to the NRC at the MUR power level of 2609 MWt.	09/01/2007
A preventative maintenance program will be developed for the LEFM using the vendor's maintenance and troubleshooting manual. This includes verifying the calibration of the 5 MHz clock in the Acoustic Processor unit and power supplies.	prior to implementation
CR-3 will complete all LEFM and associated modifications. • Installation of the Caldon system. • Addition of new Feedwater and Main Steam pressure and Main Steam Temperature instrumentation. • Modification of AULD software. • The Control Room plant reference simulator	prior to implementation
CR-3 will complete the training of operators on the LEFM modification and actions to be taken if the system is inoperable.	prior to implementation
Alden Labs calibration and test data will be provided to the NRC once completed.	09/01/2007
FPC will inform the NRC if there are any changes to critical operator actions.	09/01/2007

Title	Ultrasonic Feedwater Flow Measurement System Specification
RE	George Hildebrandt

A.1 EC Folder Contents

PCHG-DESG template 12/07/05

EC FOLDER UTILITIES (Rev 4)	
Open selected doc files from the folder	Administrative Review Report
Print selected doc files from the folder	Close & save changes to all doc files opened from the folder
Close & discard changes to all doc files opened from the folder	Update "EC Folder Contents" table and update headers, footers, & chapters of doc files
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Instructions: Click in the appropriate cell; hit F2 to execute

A.2 List of Hard Copy Only Pages

Section/Pages which have blank page place holders for hard copy only originals: None

A.3 Revision Summary

Original Revision.

A.4 Problem Statement

Objective:

The objective of this EC is to issue design specification CR3-M-0008 for the procurement of an Ultrasonic Feedwater Flow Measurement System. This system will consist of two(2) 18 inch pipe spool pieces with ultrasonic transducer and RTD assemblies for installation in the Feedwater System piping upstream of the OTSG and a single electronics cabinet. The electronics cabinet will contain the equipment necessary to operate the ultrasonic transducers and RTD assemblies and provide to plant computer systems accurate mass feedwater flowrates to each OTSG.

History:

The Appendix K power uprate is based on the accuracy of the new feedwater flow and temperature measurement instrumentation used in the secondary plant calorimetric power calculation. Per calculation M98-0007 "CR3 Heat Balance Uncertainty", the existing plant instrumentation's accuracy is within the assumed +/- 2.0 percent stated in Appendix K. By purchasing and installing the Ultrasonic Feedwater Flow Measurement System detailed in specification CR3-M-0008, this uncertainty will be reduced, thus providing the basis for a license power uprate at Crystal River 3.

Root Cause Evaluation:

Not applicable.

EC Team Members:

The EC team members include the responsible engineer (G. V. Hildebrandt), the specification verifier (M. D. Lord) and the authorizing supervisor (K. L. Allen). Reviews of the specification have been performed by various departments as detailed in Section H, and as noted in the EC milestones approvals.

A.5 Solution Statement

Options, Costs, and Benefits Evaluated:

These attributes are not within the scope of this EC, but have been addressed by management in the decision to proceed with the Measurement Uncertainty Recapture (MUR) Power Uprate Project.

Solution Statement:

The solution proposed by this EC is to issue specification CR3-M-0008, Ultrasonic Feedwater Flow Measurement System, so that procurement and installation of this equipment can be accomplished to support the MUR Power Uprate Project.

Note: Engineering Change Sections D-"Installation, E-"Testing", F-"Turnover" and G-"Sketches" have been omitted from this EC Folder since they are not needed for issue of a specification.

Title	Ultrasonic Feedwater Flow Measurement System Installation
RE	George Hildebrandt

A.1 EC Folder Contents

PCHG-DESG template 12/07/05

PEF-CR3-0215

Z16	0		pdf		308-102	1	1
Z17	0		doc		CR3-E-379	1	1
Z18	0		pdf		IDS	1	40
Z19	0		pdf		S-FN-1860-1	1	1
Z20	0		pdf		Sketch SK-1plus data	1	1
Z21	0		pdf		CDS	1	6
Z22	0		pdf		Cbl Pull Shts	1	49
Z23	0		pdf		Cbl Term Shts	1	49
Z24	0		pdf		51-9040323-000	1	22
Z25	0		xls		SCM-001 Enc. 1.xls	1	2
Z26	0		xls		SCM-001 Enc. 2.xls	1	1
Z27	0		xls		SCM-001 Enc. 3.xls	1	3
Z28	0		xls		SCM-001 Enc. 7.xls	1	1
Z29	0		doc		I06-0001 (AREVA 32-9041501-001)	1	88
Z30	0		pdf		APC UPS	1	4
Z31	0		pdf		APC Application Note #51	1	20
Z32	0		pdf		FWH-075	1	1
Z33	0		pdf		FWH-076	1	1
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Z35	0		pdf		FWH-099A	1	1
Z36	0		pdf		201B058F	1	3
Z37	0		pdf		201B070D	1	3
Z38	0		pdf		201B079F	1	3
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Z41	0		pdf		202B142-	1	2
Z42	0		pdf		202B166-	1	2
Z43	0		pdf		202B177A	1	2
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Z45	0		pdf		202B190C	1	5
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Z52	0		pdf		202B711	1	4
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Z54	0		pdf		202B717B	1	5
Z55	0		pdf		202B718C	1	8
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Z57	0		pdf		202B720B	1	4
Z58	0		pdf		PL202B626-01	1	1
Z59	0		pdf		PL202B637-02	1	3
Z60	0		pdf		PL202B719-02	1	1
Z61	0		pdf		PL202B720-01	1	2
Z62	0		pdf		Caldon Leak Report	1	?
Z63	0		pdf		Caldon B31.1 Code Reconcilation	1	7
Z64	0		pdf		Burns RTD Data	1	5

Z65	0		pdf		High Temp Transducer Cable Data	1	1
Z66	0		pdf		RTD Cable Data	1	1
Z67	0		pdf		EFP08R1	1	2
Z68	0		pdf		EFP18 Rev 11	1	4
Z69	0		pdf		EFP59 Rev 5	1	3
Z70	0		pdf		EFP61 Rev 17	1	34
Z71	0		pdf		LEFM Output Data	1	48
Z72	0		doc		CR3 Cabinet Temp Output Data	1	1
Z73	0		pdf		LEFM-VVP-1 Rev 3	1	7
Z74	0		pdf		ER-119 Rev 1 FMEA	1	39
Z75	0		pdf		TP203 FAT	1	65
Z76	0		pdf		Caldon User Manual	1	43
Z90	0		pdf		Kickoff Meeting Attendee List	1	1
Z91	0		pdf		30%ChallengeAttendeesList	1	1

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A.2 List of Hard Copy Only Pages

Section/Pages which have blank page place holders for hard copy only originals:

A.3 Revision Summary

Original Revision.

A.4 Problem Statement

Objective:

The objective of this EC is to install an ultrasonic feedwater flow measurement system in each of the 18 inch feedwater lines to the two(2) OTSG at Crystal River Unit #3. Using the ultrasonic time of transit method, the flow measuring system will continuously calculate a volumetric flow-rate of the feedwater to each OTSG. The ultrasonic feedwater flow measurement system will also use the relationship of acoustical velocity to fluid temperature to calculate a accurate feedwater temperature. By installing a pressure transmitter to measure the fluid pressure at the point of volumetric flow-rate and temperature measurement, a mass flow-rate of feedwater to each OTSG will also be continuously calculated by the system. This ultrasonic feedwater flow measurement system will be more accurate than the existing feedwater flow nozzles and temperature detectors. Additionally, more accurate Main Steam temperature and pressure instrumentation will be installed at the same time by EC 65629. All of this additional instrumentation will be used to calculate a more accurate secondary plant heat balance, thus providing the basis for an increase in the licensed core thermal power output. The uncertainty value of the new secondary plant heat balance and the resultant power level increased will be addressed by EC 65627.

Background:

Prior to this engineering change, the NRC per 10CFR50 Appendix K accepted that the secondary plant heat balance instrumentation at CR3 was accurate to within a +/- 2.0% uncertainty. The plant LOCA analysis was performed assuming this uncertainty value, and as such the plant thermal output at the onset of the accident was set to 102% of 2568 MWt or 2619.4 MWt. By installing more accurate secondary plant heat balance instrumentation (accurate to within a +/- 0.4% uncertainty), the plant could be licensed to operate at a higher thermal output level, still being bounded by the previously performed LOCA analysis.

Root Cause Evaluation:

Not applicable, since the EC is considered a plant enhancement, and as such is not be installed to correct a know plant defect or problem.

EC Team Members:

Responsible Engineer	George Hildebrandt (AREVA-NP)
Operations	Dave Jones
Maintenance	Dalton Brass/Rich Wiemann
Plant Support (Procedures)	Russ Harvey

Scheduling	Ivan Wilson/John Foley
Licensing	Shannon Burke
Training	Steve Smith
Simulator	Lee Linton
Nuclear Information Tech.	Susan McPherson/Bob Taylor
Engineering Programs	John Mueller
Radiation Protection	Ken Young
Chemistry	Rocky Thompson
Procurement	Steve Taylor
Welding	Jeff Finell
Design Eng. Electrical	Abid Khan
Design Eng. I&C	Steve Fox
Design Eng. Mechanical	Mark Livingston
Design Eng. Structural	Don Eng
System Eng. (ICS)	Bob Muzzi
System Eng. (FW/MS)	Tom Salute
Reactor Eng.	Mike Culver
Nuclear Q.C.	Bradley Kelly
Appendix R	Charlie Kish

A.5 Solution Statement

Options, Costs and Benefits Evaluated:

The procurement and installation of the ultrasonic feedwater flow measurement system was approved as part of the MUR (Measurement Uncertainty Recapture) Project which is a sub-project of the EPU (Extend Power Uprate) Project. The ultrasonic feedwater flow measurement system select is a Caldon LEFM CheckPlus System. This vendor has been also selected for similar equipment to be installed at all of the Progress Energy nuclear power plants.

Industry OE (Operating Experience)

Progress Energy installed a Caldon LEFM CheckPlus system at Robinson Unit 2 through the implementation of EC 47152. That installation experienced a weld failure located at the transducer housing tube to housing tip connection. Additionally in 2002 a weld failure was experienced at Peach Bottom Unit 2. A root cause analysis was performed by Caldon addressing both of these failures. This analysis can be found in Caldon's Customer Information Bulletin CIB 107 Rev. 0 dated April 2003 (Attachment Z62). The corrective action was the implementation of new weld procedures as discussed in this bulletin. These new weld procedures will be implemented during the manufacturing of the Crystal River Unit 3 LEFM CheckPlus chordal spool pieces. In addition, augmented quality control inspections will performed by Progress Energy at the manufacturing vendor's facility.

Solution Statement:

This EC will purchase and install a Caldon LEFM CheckPlus ultrasonic feedwater flow measurement system as specified in specification CR3-M-0008 "Caldon LEFM CheckPlus Ultrasonic Feedwater Flow Measurement System" issued by EC 65566.

Title	Power Uprate to 2609 Megawatt-Thermal
RE	Virgilio Esquillo

A.1 EC Folder Contents

PCHG-DESG template 1/1/03

Z13	0		doc	Z.14	Attachment N Electrical Impact Assessm	1	3
Z14	0		tif	Z.15	Attachment O Program Impact Reviews	1	64
Z15	0		tif	Z.16	Attachment P System Impact Reviews	1	245
Z16	0		doc	Z.17	Attachment Q Design Review Board	1	7
Z17	0		doc	Z.18	Attachment R Deaerator Uprate Evaluation	1	4

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A.2 List of Hard Copy Only Pages

Section/Pages which have blank page place holders for hard copy only originals:

EC Package Print Report

A.3 Revision Summary

Initial Issue

A.4 Problem Statement

Objective:

The objective of this EC is to evaluate a thermal power uprate of the Crystal River 3 nuclear plant to achieve an increase in the reactor core thermal power output from 2568 MWt to 2609 MWt (1.6% increase) and resultant increases in electrical generation output. Current 10 CFR Part 50 regulations allow the plant to recover the difference between 2% and the demonstrated uncertainty of thermal power measurement made possible with the installation of more accurate ultrasonic feedwater flow instrumentation. As part of the 2002 uprate to 2568 MWt, CR-3 anticipated obtaining a measurement uncertainty recapture (MUR) uprate. As such, the 2002 uprate evaluated systems up to 2619.4 MWt, reflecting the application of the full 10 CFR Part 50, Appendix K 2% uncertainty on the current base power level.

!0 CFR Part 50, Appendix K, requires licenses to assume that the reactor has been operating at a power level at least 102% of the licensed power level when performing loss-of-coolant (LOCA) and emergency core cooling system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR Part 50 allows licenses to assume a power level lower than 102% of the licensed power level provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties.

Uncertainty in feedwater flow measurement is the most significant contributor to core power measurement uncertainty. The installation of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus ultrasonic feedwater flow measurement (UFM) system will reduce the calorimetric core power measurement uncertainty to ≤ 0.4%. Based on this Crystal River 3 will reduce the power measurement uncertainty required by 10 CFR Part 50, Appendix K, to permit an increase of 1.6% in the licensed power level.

This EC consolidates and evaluates the activities of ECs 49623, 65626, 65628, 65629, and 65630 for the purpose of attaining a power uprate based on a Measurement Uncertainty Recapture (MUR) effort. EC 65627 is a documentation-only EC that will implement the uprate once regulatory approval is obtained.

Historical Information:

Crystal River Unit 3 was initially licensed to operate at a maximum of 2452 MWt. In Technical Specification Amendment 41, dated July 21, 1981, the NRC approved operation of CR-3 up to 2544 MWt.

Florida Power Corporation (FPC) began the power upgrade process several years ago with a feasibility study which evaluated the overall benefit of performing various power level upgrades in terms of the gain in electrical power output and cost. The study examined various expected plant operation conditions to confirm that appropriate design, operating and safety criteria could be met. The evaluations were done to support preliminary decisions concerning operating plant conditions at the increased power level and to identify potential major hardware modifications. At the completion of this feasibility study phase, FPC decided that increasing the allowed rated thermal power to 2568 MWt was feasible and cost beneficial in 1994.

In pursuant to the feasibility study, a detailed engineering evaluation was undertaken. This earlier engineering effort provided a detailed analysis and assessment of potentially affected areas of plant operation at the 2568 MWt power level. This evaluation covered the Nuclear Steam Supply Systems, Balance of Plant Systems and the Reactor Building Structure to assure their adequacy at the increased power level. The results of this detailed evaluation were shared at an NRC/FPC meeting, which occurred in August 31, 1994. Shortly thereafter, letter 3F0994-08 dated September 30, 1994 was submitted to the NRC requesting CR-3's license be amended to increase the reactor core thermal power from 2544 MWt to 2568 MWt. In May 1996 the submitted request was withdrawn because at the time several transient and accident analyses were being design reconstituted at the 2568 MWt power level.

Since 1996, several analyses have been revised. The loss-of-coolant accident (LOCA) analyses were revised to account for plant modifications, increased steam generator tube plugging and a change in the analysis of record from the CRAFT2 evaluation model to RELAP5/mod2-B&W. Other analyses were also revised to reflect the design improvements made during the1996 to 1998 design outage and subsequent refueling outage (R11) in the Fall of 1999. All of the revised analyses were performed considering maximum power output of 2568 MWt or higher. 10CFR50 Appendix K analyses were done at 2619.4 MWt (102 percent of 2568 MWt) to account for the two percent uncertainty assumed in power measurement. Some analyses were performed at higher power levels (generally 2772 MWt) because they were performed generically to bound all Babcock and Wilcox 177 Fuel Assembly plants. All of these analyses were approved by the NRC or were performed using methods or processes that were approved by the NRC.

Crystal River Unit 3 (CR3) has been continuously evaluating various options for increasing the power output of the plant. The Babcock and Wilcox 177 Fuel Assembly (FA) Nuclear Steam Supply System has been licensed to operate as high as 2772 MWt with most facilities operating at 2568MWt. Even though Crystal River Unit 3 did not plan to operate at 2772 MWt power level in the past; many of the original Babcock and Wilcox licensing topical reports, design documents and equipment performance specifications were developed based on operation at this power level, and were used as the basis for the design and licensing of CR3. Consequently, during the CR3's original licensing application, the NRC review and licensing actions were based on a majority of the Final Safety Analysis Report, Chapter 14 accident analyses that were performed at a thermal power level of 2568 MWt or greater.

Root Cause Evaluation:

Root Cause evaluation is not applicable, since this engineering change does not address an adverse condition. This engineering change deals with a plant improvement in generation capacity.

EC Team Members:

The EC team members include the responsible engineer (V. Esquillo), the design verifier (Later) and the authorizing supervisor (Ted Williams).

A.5 Solution Statement

Options, Costs and Benefits Evaluated:

Crystal River Unit 3 will be able to generate additional power with minimal plant impact. The Progress Energy Economic Model Version XX.X resulted in a benefit/cost ratio of XX.X and a payback period of XX years based on fuel savings and capacity deferral. (Reference: Phase Project authorization Form for Project #XXXXXXX)

Solution Statement:

Nuclear Regulatory Commission (NRC) has issued License Amendment 20X to Facility Operating License No. DPR-72 for Crystal River Unit 3 to increase the maximum allowed rated thermal power (RPT) for Crystal River Unit 3 from 2568 Megawatt – Thermal (MWt) to 2609 Megawatt – Thermal (MWt), an increase of 41 MWt over the current licensed rated thermal power of 2568 MWt. This represents a power uprate increase of 1.6%. The 2609 MWt Power Uprate will be managed under five engineering design changes, listed as follows:

EC49623 This engineering change replaces the Nuclear Application Software (NAS) incore surveillance program with the Fixed Incore Detector Monitoring System (FIDMS).

EC65626 This engineering change provides for the installation of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus Ultrasonic Feedwater Flow Measurement (UFM) system including the spool pieces with integrated instrumentation, electronics cabinet with cooling system, and associated software. This change also provides for all associated electrical system modifications to power the Caldon LEFM CheckPlus system cabinet and cooling unit.

EC65628 This change modifies the affected parameters in the Automated Unit Load Demand (AULD) software and the AULD Display software. The resulting database changes to the ranges and/or descriptors for the parameters supplied by the TRICON to the Plant Process Computer System (PPCS) will also be accomplished as part of this EC.

EC65629 This engineering change provides for the installation of one new high accuracy pressure transmitter and one new thermowell and dual element RTD assembly in each of the four main steam lines. The instrumentation is selected to be significantly more accurate than existing instrumentation that is used to determine heat balance per existing calculation I98-0002. Additionally, the new MUR installed pressure transmitters will be calibrated for a narrow range of 800-1000 psig. These equipment changes and calibration changes will significantly reduce the uncertainty of the OTSG pressure and temperature measurements.

EC65630 This engineering change modifies Integrated Control System (ICS) modules in order to support the increase in licensed thermal generation resulting from the MUR power uprate. ICS modules that are affected are those whose settings are referenced to the licensed thermal power limit, the maximum continuous rating (MCR) or corresponding nominal electrical power value, and the full load main feedwater flow value.

Based on the above and the evaluation performed in Section B.6, the solution response for this Engineering Change is for Crystal River Unit 3 to implement the power uprate. The NRC has issued License Amendment 20X to Facility Operating License No. DPR-72 for Crystal River Unit 3 to increase core rated thermal power from 2568 MWt to 2609 MWt. The license amendment permits CR3 to operate at the 2609 MWt power level for the remaining operating license life (XX years) and until the end of the extended plant license life.

Title	Automated Unit Load Demand (AULD) Modification Supporting Measurement Uncertainty Recapture (MUR) Power Uprate
RE	M. D. Lord

A.1 EC Folder Contents

PCHG-DESG template 1/15/07

EC FOLDER UTILITIES (Rev 4)	
Open selected doc files from the folder	Administrative Review Report
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A.2 List of Hard Copy Only Pages

EC Package Print Report
Section/Pages which have blank page place holders for hard copy only originals:
N/A

A.3 Revision Summary

Original Revision.

Section/Pages affected by this revision:
N/A

A.4 Problem Statement

Objective:
The objective of this EC is to modify the Automated Unit Load Demand (AULD) portion of the Integrated Control System (ICS) to support an increase in the thermal, and consequently the electrical generation of the station. This EC is not addressing any plant problem, but rather involves a plant enhancement to increase the electrical output of the unit.

Historical Information:
Reference Design Inputs under the Design Specification section.

Root Cause Evaluation:
N/A

EC Team Members:
EC team members include representatives from I&C Design Engineering, I&C Systems Engineering, Operations, and Maintenance.

A.5 Solution Statement

Options, Costs and Benefits Evaluated:
The only option evaluated in response to the above problem statement is to modify the AULD software running in the TRICON and the AULD Display software running in the associated Panel PC in order to support the increase in licensed thermal generation of CR3 authorized by Engineering Change 65627. The result of these changes is an expected increase in the thermal and electrical generation of CR3 of 41 MWt. The cost of this modification has been addressed in the Request for Project Approval for this phase of the power level upgrade project. The primary benefit to be derived from this EC is an increase in electrical generation of CR3.

Solution Statement:
Based on the above evaluation, the solution promoted by this EC is to modify the AULD software and the AULD Display software to increase thermal and electrical generation.

Title	MUR Upgrade- MS Pressure and Temperature Instrumentation
RE	Robert E. Wagner

A.1 EC Folder Contents

PCHG-DESG template 12/07/05

Z15	0		pdf		Enclosure Layout in Intermediate Building	1	1
Z16	0		pdf		Instrument string calib	1	12
Z17	0		pdf		Rosemount transmitter mounting & dimensions	1	3
Z18	0		xls		SCM-001 MSCustomerDataTable	1	1
Z19	0		xls		SCM-001 MSPointTable	1	1
Z20	0		xls		SCM-001 MSAlarms-AnalogTable	1	1
Z21	0		xls		SCM-001 MSRealTimeTable	1	1
Z22	0		pdf		Conduit Support Load sheets	1	13
Z23	0		pdf		Corning fiber equipment	1	26
Z24	0		pdf		VTMA 02503-001-Rosemount Series 0078 RTD	1	77
Z25	0		pdf		VTMA 02504-001- Rosemount 3051N transmitter	1	91
Z26	0		pdf		VTMA 02505-001 Moore Ind & Acopian	1	65
Z27	0		pdf		VTMA 02506-001 Cisco Switch and Power Supply	1	96
Z28	0		pdf		Hoffman Enclosure dimensions	1	5
Z29	0		pdf		PS cal data sheets	1	6
Z30	0		pdf		PS Instrument Data Sheets	1	6
Z31	0		pdf		Swagelok Installation Instructions	1	1
Z32	0		doc		CR 3 Steam Temperature Instrument Uncertainty	1	50
Z33	0		doc		CR3 Steam Pressure Instrument Uncertainty	1	49

EC FOLDER UTILITIES (Rev 4)	
Open selected doc files from the folder	Administrative Review Report
Print selected doc files from the folder	Close & save changes to all doc files opened from the folder
Close & discard changes to all doc files opened from the folder	Update "EC Folder Contents" table and update headers, footers, & chapters of doc files
Save all EC Folder doc files as read-only	Allow change to all EC Folder doc files
Select an open doc and insert an 11x8.5 page following its current page	Select an open doc and insert a 17x11 page following its current page

Instructions: Click in the appropriate cell; hit F2 to execute

A.2 List of Hard Copy Only Pages

Section/Pages which have blank page place holders for hard copy only originals:

None

A.3 Revision Summary

Original Revision.

Section/Pages affected by this revision: None. This is Rev. 0

A.4 Problem Statement

.The existing CR3 ITS allowed reactor power level is reduced by 2% due to the potential error in instrumentation uncertainties for those flow, temperature, pressures, and RCP power input used in determining the reactor heat production. Significant inputs to determination of reactor power are the OTSG secondary side parameters of flow and enthalpy of steam exiting the steam generators. The steam enthalpy is determined by measurement of the steam pressure and steam temperatures exiting the steam generators.

The existing CR3 heat balance utilizes an average of four main steam pressure transmitters (SP-6A/6B-PT1/PT2) located at the outlet of the steam generator in the reactor building with four main steam pressure transmitters located at the high pressure turbine steam chest (SP-10A/10B-PT1/PT2). Additionally, the thermocouple temperature elements (MS-14,15,16,17-TE) used in conjunction with the pressure transmitters are located at neither the reactor building nor the high pressure turbine but in the intermediate building. The existing methodology thus measures steam pressures at two locations that have different pressure readings with a temperature that is at a third location.

Excel Spreadsheet evaluation using ASME steam table data reveal that this existing methodology of determining OTSG outlet steam enthalpy creates a heat balance error that results in a higher apparent/calculated steam enthalpy than actually exits from the OTSG. This effectively creates a condition in which the secondary side heat balance indicates a higher BTU/hr heat transfer from the primary reactor coolant system to the feedwater/steam secondary systems than has actually occurred. This creates an unaccounted for condition in which power is limited to a lower value than is necessary for the 2.0% instrument uncertainty limit in determining reactor power.

Furthermore I98-0002, "CR 3 Heat Balance Calculation Input Uncertainties" assign a +/- 18 psig (+/- 1.5%) steam generator pressure loop uncertainty to a single pressure transmitter loop with a 9.0 psig average of four steam line uncertainty. I98-0002 also evaluates the thermocouple temperature elements as having a 6.0°F single loop uncertainty and a +/- 4.5°F average of two temperature inputs.

The Measurement Uncertainty Recapture Project has a goal to recover/eliminate1.6% of the existing 2.0% uncertainty of the CR3 heat balance. The existing steam instrumentation does have a significant impact on determination of secondary side steam enthalpy and thus on determination of heat produced by the reactor and limited by plant technical specifications. Neither the location, methodology, nor the accuracy specifications of existing steam pressure and temperature instrumentation are desirable from the viewpoint of reducing heat balance uncertainty. There are more accurate instrumentation devices available than those presently used in heat balance calculations

A.5 Solution Statement

This EC will install additional steam pressure transmitter and steam temperature RTD devices that are significantly more accurate than those presently used in heat balance calculations. One new pressure transmitter and one new RTD will be installed in each of the four main steam lines. The new steam pressure transmitter and the steam temperature measurement will be at essentially the same location (in Intermediate Building elevation 119') which will eliminate some potential steam enthalpy error of measuring steam pressure at locations different from the temperature measurement..

Four of the existing pressure transmitters used in the existing 2568 MWth heat balance have documented reference accuracies of 0.25% span and temperature effect of 0.5%URL +0.5% span per 100°F.

The new pressure transmitters installed by this EC have documented reference accuracies of 0.075% span and temperature effect of 0.025%URL + 0.125% span per 100°F. The pressure transmitter 4-20ma input will be to a Moore Industries Net Concentrator System AIM (Analog Input Module) that has specifications of accuracy of 0.01% span and stability/drift uncertainty of 0.081% span for 3 years. This AIM will convert the analog signal to an Ethernet signal. The remainder of the pressure transmitter signal loop to the AULD/Triconex and plant computer will be Ethernet/digital with no documented error.

The existing thermocouples used in the existing 2568 MWth heat balance have a documented element error of up to +/-4.6°F with an instrument string error of 6.0F°.

The new RTDs installed by this EC will be obtained with IEC 751 Class A option with accuracy of +/- 1.426 F° at 610F° from the standard IEC 751, 100 ohm platinum, alpha= 0.0038505 curve. The new RTDs will also have small errors of about 0.18°F for 10 cycle temperature effect, about 1.05°F for 30 month stability, and a self heating effect of less than 0.002°F (when using the Moore Industries TIM module with 250 uamp excitation current).

The RTD ohm input will be to a Moore Industries Net Concentrator System TIM (Temperature Input Module) that has specifications of +/- 0.18°F with an ambient temperature accuracy change of up to 0.245°F for a 70°F intermediate building temperature change. The stability/drift uncertainty for this TIM is documented as 0.023% for 3 year interval. The remainder of the RTD signal loop to the AULD/Triconex and plant computer will be Ethernet/digital with no documented error.

CR3 Calculation I07-0002 (Attachment Z32R0) documents a loop uncertainty of +/- 2.02F° for a single element of the RTD. To reduce instrument loop uncertainty without creating administrative burdens of matching RTDs with specially calibrated/programmed input modules, dual element RTDs are being used in each steam line thermowell. Thus when evaluated for a single steam line, the +/-2.02F° of a single element becomes +/- 1.43F° for the two elements of the dual element RTD.

To minimize steam piping weld requirements and to minimize new pressure transmitter tubing, the new pressure transmitters will utilize existing pressure transmitter tubing for the EFIC transmitters. Appropriate isolation valves will be installed between the new transmitters and the EFIC tubing to minimize any potential for pressure disturbance during calibration of the new pressure transmitters. Additionally this is 900 psig tubing and the instrumentation will be valved in and out per SP-123 . The potential for inadvertent actuation of EFIC transmitters on an OTSG low pressure signal is very low. The OTSG differential pressure FOGG logic signal must also have a concurrent "vector enable" signal for any EF valves to travel to close position. Thus this potential is also very low.

The existing steam pressure transmitters and steam thermocouples used in the 100% power/2568 MWth heat balance calculation will not be disturbed and will remain in service for use in backup contingency 2568MWth heat balance calculations in the event the new 2609MWth target heat balance equipment fails on-line.

Title	Integrated Control System (ICS) Scaling to Support Increase to 2609MWt
RE	George Hildebrandt (Michael Speziali)

A.1 EC Folder Contents

PCHG-DESG template 1/15/07

EC FOLDER UTILITIES (Rev 4)	
Open selected doc files from the folder	Administrative Review Report
Print selected doc files from the folder	Close & save changes to all doc files opened from the folder
Close & discard changes to all doc files opened from the folder	Update "EC Folder Contents" table and update headers, footers, & chapters of doc files
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Instructions: Click in the appropriate cell; hit F2 to execute

A.2 List of Hard Copy Only Pages

Section/Pages which have blank page place holders for hard copy only originals:

A.3 Revision Summary

Original Revision.

Section/Pages affected by this revision:

N/A

A.4 Problem Statement

Objective:
The objective of this EC is to modify the Integrated Control System (ICS) to support an increase in the thermal, and consequently the electrical generation of the station. This EC is not addressing any plant problem, but rather involves a plant enhancement to increase the electrical output of the unit.

Background:

As a result of EC 65627 which is increasing the Plant Output to 2609 MWt and 914 MWe, this EC will support the required rescaling of the system modules to reflect the new plant operating parameters as a result of the increase. Previously EC 49344R0, EC 59702R0 and EC 62207R0 provide the methodology for this EC in support of a previous uprate to 2568MWt and 903 MWe.

Root Cause Evaluation:

Not applicable, since the EC is considered a plant enhancement, and as such is not be installed to correct a know plant defect or problem.

EC Team Members:

Responsible Engineer	George Hildebrandt (AREVA-NP) (Michael Speziali)
Operations	Dave Jones
Maintenance	Dalton Brass/Rich Wiemann
Plant Support (Procedures)	Russ Harvey
Scheduling	Ivan Wilson/John Foley
Licensing	Shannon Burke
Training	Steve Smith
Simulator	Lee Linton
Engineering Programs	John Mueller
Radiation Protection	Ken Young
Chemistry	Rocky Thompson
Procurement	Steve Taylor
Design Eng. Electrical	Abid Khan
Design Eng. I&C	Steve Fox
Design Eng. Mechanical	Mark Livingston
Design Eng. Structural	Don Eng
System Eng. (ICS)	Bob Muzzi
System Eng. (FW/MS)	Tom Salute
Reactor Eng.	Mike Culver
Nuclear Q.C.	Bradley Kelly
Appendix R	Charlie Kish

A.5 Solution Statement

Options, Costs and Benefits Evaluated:

To support the Power Level Upgrade as provided by EC 65627 the rescaling of the ICS modules to support the increase in plant parameters previously performed by EC 49344R0 is considered the only viable option to reflect the new plant parameters.

Industry OE (Operating Experience)

Progress Energy experienced ICS module failures which have caused plant trips. To preclude the potential for a plant trip, refurbished modules will be utilized in the replacements/rescaling to provide reasonable assurance that a plant trip is avoided.

EC 49344R0 along with EC 59702R0 and EC 62207R0 were successfully implemented which provide the methodology for this EC.

Solution Statement:

This EC will provide refurbished modules as required and rescale the modules in the ICS to reflect the new plant operating parameters.

Progress Energy

AREVA NP INC.
P.O. Box 10935
3315-A Old Forest Rd.
Lynchburg, VA 24506-0935

CONFIDENTIAL

Attention: Mr. Don Lightfoot

CONTRACT NO. 101659
AMENDMENT NO. 06
EFFECTIVE JANUARY 1, 2007

[redacted]

Sincerely,

T.R. Lineback

T. R. Lineback
Corporate Services

TRL/jeb
Attachment

Accepted:

FRAMATOME ANP, INC.

By: David F. Gura

Name (printed): DAVID F. GUZA

Title: Secretary

Date: 12/20/06

CONTRACT
JAN 04 2007 CS
ACCEPTANCE

Should the person's title who is executing this document not indicate that he/she is a corporate officer, an affidavit signed by a corporate officer shall be provided stating that the person whose name appears above is duly authorized to execute Contracts on behalf of the firm.

MASTER CONTRACT No. 101659

BETWEEN

PROGRESS ENERGY SERVICE COMPANY, LLC

Not in its individual capacity but solely as agent for

Progress Energy, Carolinas, Inc.,
also known as Carolina Power & Light Company
and Progress Energy, Florida, Inc.,
also known as Florida Power Corporation

AND

FRAMATOME ANP, INC.

GENERAL TERMS AND CONDITITONS
FOR FURNISHING EQUIPMENT AND SERVICES

Progress Energy Services Company, LLC
411 Fayetteville Street Mall
Raleigh, North Carolina 27602

PEF DOCUMENTS RESPONSIVE TO
CITIZEN'S FIRST REQUEST TO PRODUCE
DOCUMENTS TO
PROGRESS ENERGY FLORIDA, INC. (NO. 2)

DOCUMENTS IDENTIFIED WITH
BATES NOS.
PEF-CR3-0240 THROUGH PEF-CR3-0303
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

CONFIDENTIAL

[redacted]

FRAMATOME, ANP, INC.

BY: [signature]

NAME (printed): Andrew G. Cook

TITLE: Vice President

DATE: 8/28/03

PROGRESS ENERGY SERVICE COMPANY, LLC
Not in its individual capacity, but solely as Agent for, Progress Energy, Carolinas, Inc. and Progress Energy, Florida, Inc.

BY: Joseph J Corey, Jr

Joseph J. Corey, Jr., Program Leader
Supply Chain Management - Contracts

DATE: AUGUST 11, 2003

Indicate your Social Security Number (SS#) OR your Employer Identification Number (EIN). This number shall correspond with the FRAMATOME name indicated above and shall be the same TIN under which you report income. COMPLETE ONLY ONE.

TIN 54-1536465

The Internal Revenue Service (IRS) requires us to obtain certain information from you to meet IRS Form 1099 reporting and filing requirements.

If you do not provide your correct Taxpayer Identification Number (TIN), your payments may be subject to 20% backup withholding.

Under penalties of perjury, I certify that the TIN shown above is correct for the FRAMATOME named.

Donald Lightfoot
(FRAMATOME to fill in name and title)
is appointed as the person to whom all official correspondence to FRAMATOME concerning this Contract should be directed.

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DOCUMENTS IDENTIFIED WITH
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PEF-CR3-0305 THROUGH PEF-CR3-0336
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
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CR No. 615972

Progress Energy

AREVA NP, INC.
3315 Old Forest Road
Lynchburg, VA 24506

Attention: Gary Mignogna

CONFIDENTIAL

CONTRACT NO. 101659
WORK AUTHORIZATION NO. 61
EFFECTIVE SEPTEMBER 21, 2006

Progress Energy Service Company, LLC
P.O. Box 1551
Raleigh, NC 27602

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DOCUMENTS IDENTIFIED WITH
BATES NOS.
PEF-CR3-0338 THROUGH PEF-CR3-0345
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

CONFIDENTIAL

[redacted]

Sincerely,

CSHinnant

C.S. Hinnant
Senior Vice President and Chief Nuclear Officer

Accepted:

AREVA NP INC.

By: Gary Mignogna
with clarification as noted in AREVA-06-3351

Gary Mignogna
Vice President, Engineering

Date: December 18, 2006

CONTRACT
JAN 02 2006 CS
ACCEPTANCE

Should the person's title who is executing this document not indicate that he/she is a corporate officer, an affidavit signed by a corporate officer shall be provided stating that the person whose name appears above is duly authorized to execute Contracts on behalf of the firm.

Dallas T. Scott, Project Manager
is appointed as the person to whom all official correspondence to Contractor concerning this Contract should be directed.

In accordance with the Federal Acquisition Regulation section 52.219, please check all that apply to your company. Please provide supporting documentation or certification to confirm the status for any categories checked under Small/Diverse Vendors.

[ ] Certified small business*
[ ] Veteran-owned business*
[ ] Service-disabled veteran-owned business*
[ ] Not a Small Business
[ ] HUBZone, 8(a) or disadvantaged business*
[ ] Minority-owned business * *
[ ] Women-owned small business * *

* As defined by the Small Business Administration (SBA): www.sba.gov
* * Certified by Progress Energy and as defined by SBA.

Register online at www.progress-energy.com/supplierdiversity

248525

PEF DOCUMENTS RESPONSIVE TO
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DOCUMENTS TO
PROGRESS ENERGY FLORIDA, INC. (NO. 2)

DOCUMENTS IDENTIFIED WITH
BATES NOS.
PEF-CR3-0347 THROUGH PEF-CR3-0356
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

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PROGRESS ENERGY FLORIDA, INC. (NO. 2)

DOCUMENTS IDENTIFIED WITH
BATES NOS.
PEF-CR3-0358 THROUGH PEF-CR3-0363
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

NuFlo Technologies Sales Co.
1070 Banksville Avenue
Pittsburgh, PA 15216

CONFIDENTIAL

Attention: Ernie Hauser

CONTRACT NO. 44867
AMENDMENT NO. 7
EFFECTIVE NOVEMBER 30, 2006

CONFIDENTIAL

[redacted]

Sincerely,

CSHinnant

C. S. Hinnant
Senior Vice President
Chief Nuclear Officer

Accepted:

NuFlo Technologies ~~Sales Company~~

By: ______________________

Name (printed): Tom Simms

Title: Vice President Finance

Date: 11/28/06

Should the person's title who is executing this document not indicate that he/she is a corporate officer, an affidavit signed by a corporate officer shall be provided stating that the person whose name appears above is duly authorized to execute Contracts on behalf of the firm.

In accordance with the Federal Acquisition Regulation section 52.219, please check all that apply to your company. Please provide supporting documentation or certification to confirm the status for any categories checked under Small/Diverse Vendors.

[ ] Certified small business*
[ ] Veteran-owned business*
[ ] Service-disabled veteran-owned business*
[X] Not a Small Business
[ ] HUBZone, 8(a) or disadvantaged business*
[ ] Minority-owned business * *
[ ] Women-owned small business * *

* As defined by the Small Business Administration (SBA): www.sba.gov
* * Certified by Progress Energy and as defined by SBA.

Register online at www.progress-energy.com/supplierdiversity

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DOCUMENTS IDENTIFIED WITH
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PEF-CR3-0365 THROUGH PEF-CR3-0401
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PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

CR No. 58857

MEASUREMENT UNCERTAINTY RECAPTURE

CONTRACT

PASSPORT NO. 44867

BETWEEN

CAROLINA POWER AND LIGHT

AND

FLORIDA POWER CORP.

AND

CALDON, INC.

PEF DOCUMENTS RESPONSIVE TO
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DOCUMENTS IDENTIFIED WITH
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PEF-CR3-0403 THROUGH PEF-CR3-0442
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

CONFIDENTIAL

[redacted]

CALDON, INC.	Carolina Power & Light
BY:	By: C. S. Hinnant
NAME (printed): ERNEST M. HAUSER	Senior VP and Chief Nuclear Officer
TITLE: PRESIDENT - CALDON NUCLEAR	Date 11-12-01
DATE: 11-14-01	Florida Power Corp.
	By: C. S. Hinnant
	Senior VP and Chief Nuclear Officer
	Date 11-12-01

Indicate your Social Security Number (SS#) OR your Employer Identification Number (EIN). This number shall correspond with the Contractor name indicated above and shall be the same TIN under which you report income. COMPLETE ONLY ONE.

EIN 25-1591461 SS# ___-___-____

The Internal Revenue Service (IRS) requires us to obtain certain information from you to meet IRS Form 1099 reporting and filing requirements.

If you do not provide your correct Taxpayer Identification Number (TIN), your payments may be subject to 20% backup withholding.

Under penalties of perjury, I certify that the TIN shown above is correct for the Contractor named.

ERNEST M. HAUSER, CALDON NUCLEAR PRESIDENT
(Contractor to fill in name and title)
Is appointed as the person to whom all official correspondence to Contractor concerning this Contract should be directed.

PEF DOCUMENTS RESPONSIVE TO
CITIZEN'S FIRST REQUEST TO PRODUCE
DOCUMENTS TO
PROGRESS ENERGY FLORIDA, INC. (NO. 2)

DOCUMENTS IDENTIFIED WITH
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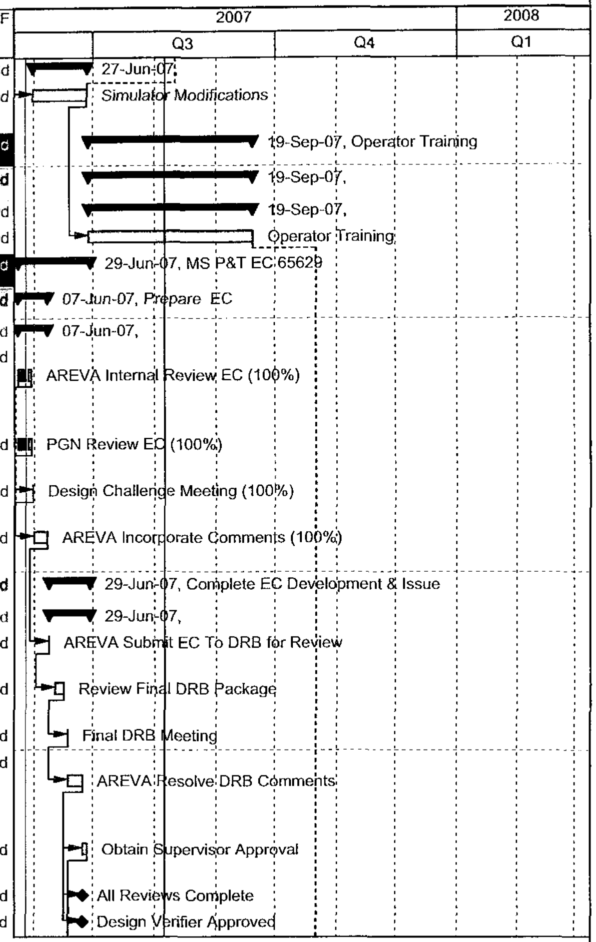
PEF-CR3-0444 THROUGH PEF-CR3-0462
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

Activity ID	Activity Name	RD	Physical % Complete	RESP	Start	Finish	Predecessors	Successors	TF
		20d			31-May-07	27-Jun-07			28d
CRSM000	Simulator Modifications	20d	0%	CR3	31-May-07	27-Jun-07*	CRTH060, CRPM000	CREDBD000, CROT000	29d
Operator Training		58d			28-Jun-07	19-Sep-07			29d
		58d			**28-Jun-07**	**19-Sep-07**			**29d**
		58d			28-Jun-07	19-Sep-07			29d
CROT000	Operator Training	60d	0%	CR3	28-Jun-07	19-Sep-07	CRSM000	CRST000	29d
MS P&T EC 65629		24d			24-May-07 A	29-Jun-07			144d
Prepare EC		**8d**			**24-May-07 A**	**07-Jun-07**			**144d**
		8d			24-May-07 A	07-Jun-07			144d
CR-226-E01	AREVA Internal Review EC (100%)	2d	0%	DUNGAN	24-May-07 A	30-May-07	CR-224-E01, CR-225-E01	CR-228-E01, CR-230-E01, CR-232-E01	149d
CR-230-E01	PGN Review EC (100%)	2d	0%	CR3	24-May-07 A	30-May-07	CR-226-E01	CR-232-E01	149d
CR-232-E01	Design Challenge Meeting (100%)	1d	0%	CR3/AREVA	31-May-07	31-May-07	CR-230-E01, CR-226-E01	CR-228-E01	149d
CR-228-E01	AREVA Incorporate Comments (100%)	5d	0%	WAGNER	01-Jun-07	07-Jun-07	CR-226-E01, CR-232-E01	CR-242-E01	149d
Complete EC Development & Issue		**16d**			**08-Jun-07**	**29-Jun-07**			**144d**
		16d			08-Jun-07	29-Jun-07			144d
CR-242-E01	AREVA Submit EC To DRB for Review	0d	0%	WAGNER	08-Jun-07	08-Jun-07	CR-228-E01	CR-243-E01	149d
CR-243-E01	Review Final DRB Package	5d	0%	CR3/AREVA	11-Jun-07	15-Jun-07	CR-242-E01	CR-244-E01	149d
CR-244-E01	Final DRB Meeting	0d	0%	CR3/AREVA	18-Jun-07	18-Jun-07	CR-243-E01	CR-245-E01	149d
CR-245-E01	AREVA Resolve DRB Comments	6d	0%	WAGNER	18-Jun-07	25-Jun-07	CR-244-E01	CR-248-E01, CR-247-E01, CR-246-E01	149d
CR-248-E01	Obtain Supervisor Approval	3d	0%	WAGNER	25-Jun-07	27-Jun-07	CR-245-E01	CR-249-E01	149d
CR-246-E01	All Reviews Complete	0d	0%	CR3/AREVA		25-Jun-07	CR-245-E01		153d
CR-247-E01	Design Verifier Approved	0d	0%	CR3/AREVA		25-Jun-07	CR-245-E01		153d



Actual Work | Critical Remaining Work | Su... | Remaining Work | Milestone

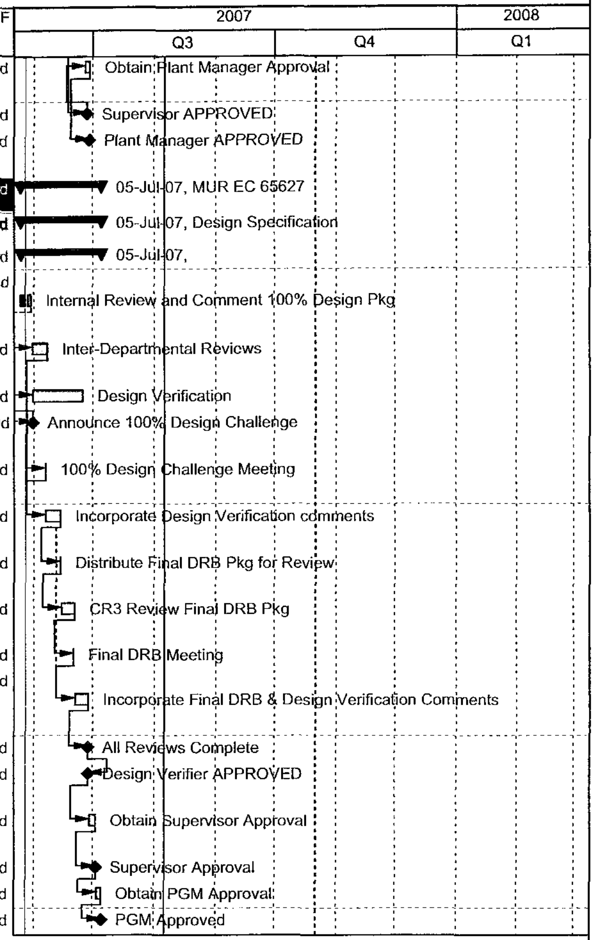
DATA DATE: 28-May-07

TASK filter: Remaining Work.

(c) Primavera Systems, Inc.

PEF-CR3-0464

Activity ID	Activity Name	RD	Physical % complete	RESP	Start	Finish	Predecessors	Successors	TF
CR-250-E01	Obtain Plant Manager Approval	3d	0%	WAGNER	27-Jun-07	29-Jun-07	CR-249-E01	CR-252-E01	149d
CR-249-E01	Supervisor APPROVED	0d	0%	CR3		27-Jun-07	CR-248-E01	CR-250-E01	149d
CR-252-E01	Plant Manager APPROVED	0d	0%	CR3		29-Jun-07	CR-250-E01		149d
MUR EC 65627		27d			25-May-07 A	05-Jul-07			141d
Design Specification		**27d**			**25-May-07 A**	**05-Jul-07**			**141d**
		27d			25-May-07 A	05-Jul-07			141d
CRMEC150	Internal Review and Comment 100% Design Pkg	2d	0%	ESQUILLO	25-May-07 A	30-May-07		CRMEC145	1d
CRMEC142	Inter-Departmental Reviews	6d	0%	CR3	31-May-07	07-Jun-07	CRMEC145	CRMEC155, CRMEC160	151d
CRMEC151	Design Verification	18d	0%	CR3	31-May-07	25-Jun-07	CRMEC145		148d
CRMEC145	Announce 100% Design Challenge	0d	0%	ESQUILLO		31-May-07*	CRMEC140, CRMEC150	CRMEC151, CRMEC142	0d
CRMEC155	100% Design Challenge Meeting	0d	0%	ESQUILLO	07-Jun-07	07-Jun-07	CRMEC142	CRMEC160	141d
CRMEC160	Incorporate Design Verification comments	6d	0%	ESQUILLO	07-Jun-07	14-Jun-07	CRMEC155, CRMEC142	CRMEC168, CRMEC165	141d
CRMEC165	Distribute Final DRB Pkg for Review	1d	0%	ESQUILLO	14-Jun-07	14-Jun-07	CRMEC160	CRMEC166	141d
CRMEC166	CR3 Review Final DRB Pkg	5d	0%	CR3	15-Jun-07	21-Jun-07	CRMEC165	CRMEC167	141d
CRMEC167	Final DRB Meeting	0d	0%	ESQUILLO	21-Jun-07	21-Jun-07	CRMEC166	CRMEC168	141d
CRMEC168	Incorporate Final DRB & Design Verification Comments	5d	0%	ESQUILLO	22-Jun-07	28-Jun-07	CRMEC167, CRMEC160	CRMEC169	141d
CRMEC169	All Reviews Complete	0d	0%	ESQUILLO		28-Jun-07	CRMEC168	CRMEC170	141d
CRMEC170	Design Verifier APPROVED	0d	0%	ESQUILLO		28-Jun-07	CRMEC169	CRMEC171	141d
CRMEC171	Obtain Supervisor Approval	2d	0%	ESQUILLO	29-Jun-07	02-Jul-07	CRMEC170	CRMEC172	141d
CRMEC172	Supervisor Approval	0d	0%	ESQUILLO		02-Jul-07	CRMEC171	CRMEC173	141d
CRMEC173	Obtain PGM Approval	2d	0%	ESQUILLO	03-Jul-07	05-Jul-07	CRMEC172	CRMEC174	141d
CRMEC174	PGM Approved	0d	0%	ESQUILLO		05-Jul-07	CRMEC173		141d



Actual Work | Critical Remaining Work | Su... | Remaining Work | Milestone

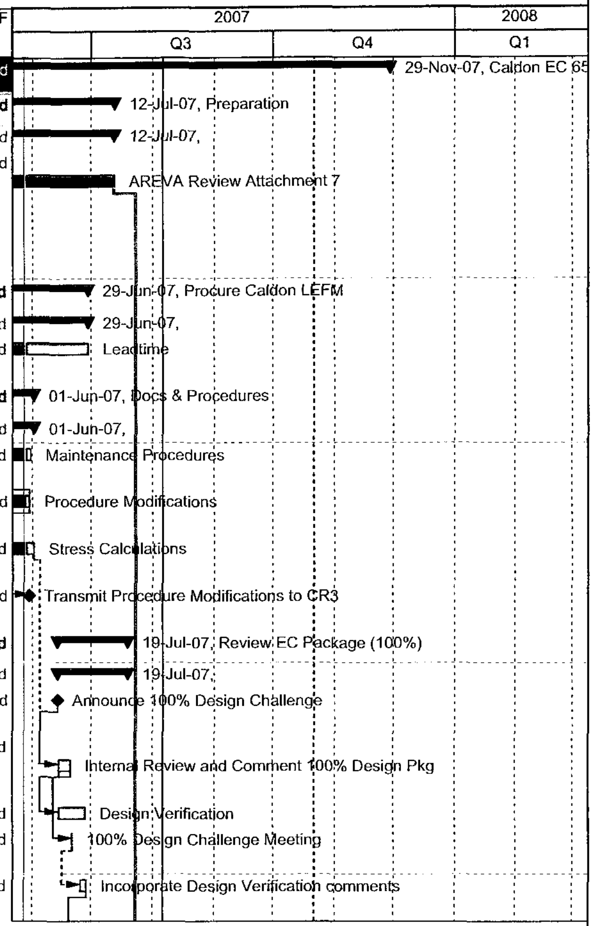
DATA DATE: 28-May-07

TASK filter: Remaining Work.

(c) Primavera Systems, Inc.

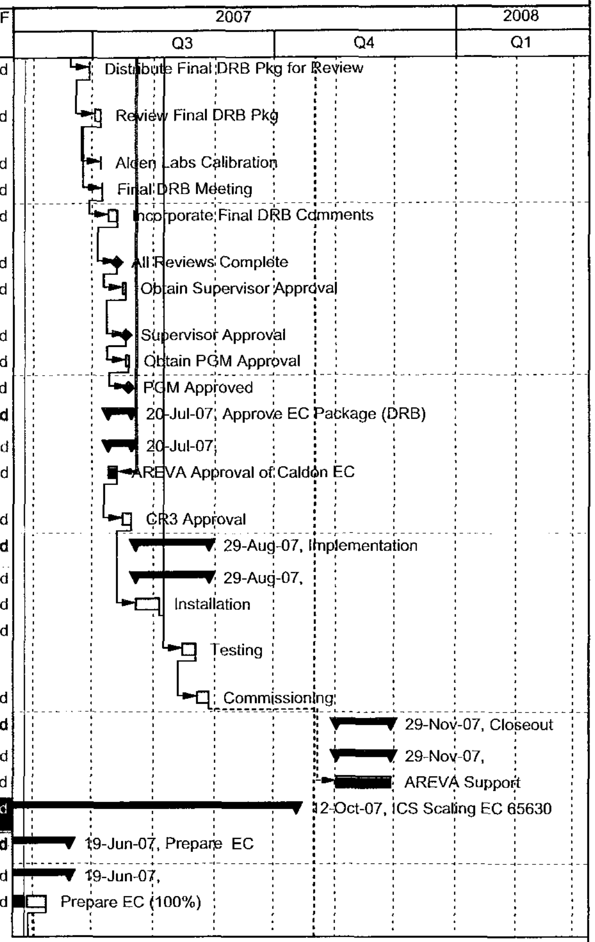
PEF-CR3-0465

Activity ID	Activity Name	RD	Physical % complete	RESP	Start	Finish	Predecessors	Successors	TF
Caldon EC 65626		129d			06-Nov-06 A	29-Nov-07			39d
Preparation		**32d**			**14-Dec-06 A**	**12-Jul-07**			**0d**
		32d			14-Dec-06 A	12-Jul-07			0d
CRECPR040	AREVA Review Attachment 7	32d	0%	MOORE	14-Dec-06 A	12-Jul-07	CRECPR000, CRECPR010, CRECPR020, CRECPR030	CRECA000	0d
Procure Caldon LEFM		**24d**			**06-Nov-06 A**	**29-Jun-07**			**144d**
		24d			06-Nov-06 A	29-Jun-07			144d
CRECPL000	Leadtime	24d	0%	CR3	06-Nov-06 A	29-Jun-07	CRECDS070		144d
Docs & Procedures		**4d**			**24-Jan-07 A**	**01-Jun-07**			**164d**
		4d			24-Jan-07 A	01-Jun-07			164d
CRECAD070	Maintenance Procedures	3d	0%	MOORE	24-Jan-07 A	31-May-07	CRECAD060		165d
CRPM000	Procedure Modifications	3d	0%	CR3/AREVA	12-Feb-07 A	30-May-07	CRECAD060	CRSM000, CRPM001	22d
CRECAD090	Stress Calculations	4d	0%	MOORE	09-Apr-07 A	01-Jun-07	CRECAD080	CREC020	140d
CRPM001	Transmit Procedure Modifications to CR3	0d	0%	AREVA		30-May-07*	CRPM000		22d
Review EC Package (100%)		**25d**			**13-Jun-07**	**19-Jul-07**			**131d**
		25d			13-Jun-07	19-Jul-07			131d
CREC010	Announce 100% Design Challenge	0d	0%			13-Jun-07*		CREC020	0d
CREC020	Internal Review and Comment 100% Design Pkg	5d	0%		14-Jun-07	20-Jun-07	CREC010, CRECAD090	CREC030, CREC040	132d
CREC030	Design Verification	10d	0%		14-Jun-07	27-Jun-07	CREC020		146d
CREC040	100% Design Challenge Meeting	0d	0%		21-Jun-07	21-Jun-07	CREC020	CREC050	132d
CREC050	Incorporate Design Verification comments	4d	0%		25-Jun-07*	28-Jun-07	CREC040	CREC060	131d



(c) Primavera Systems, Inc.

Activity ID	Activity Name	RD	Physical % complete	RESP	Start	Finish	Predecessors	Successors	TF
CREC060	Distribute Final DRB Pkg for Review	1d	0%		29-Jun-07	29-Jun-07	CREC050	CREC070	131d
CREC070	Review Final DRB Pkg	3d	0%		02-Jul-07	05-Jul-07	CREC060	CREC080, CREC065	131d
CREC065	Alden Labs Calibration	0d	0%		05-Jul-07	05-Jul-07	CREC070		151d
CREC080	Final DRB Meeting	0d	0%		06-Jul-07	06-Jul-07	CREC070	CREC090	131d
CREC090	Incorporate Final DRB Comments	5d	0%		09-Jul-07	13-Jul-07	CREC080	CREC100	131d
CREC100	All Reviews Complete	0d	0%			13-Jul-07	CREC090	CREC120	131d
CREC120	Obtain Supervisor Approval	2d	0%		16-Jul-07	17-Jul-07	CREC100	CREC130	131d
CREC130	Supervisor Approval	0d	0%			17-Jul-07	CREC120	CREC140	131d
CREC140	Obtain PGM Approval	2d	0%		18-Jul-07	19-Jul-07	CREC130	CREC150	131d
CREC150	PGM Approved	0d	0%			19-Jul-07	CREC140		131d
Approve EC Package (DRB)		**10d**			**09-Jul-07**	**20-Jul-07**			**53d**
		10d			09-Jul-07	20-Jul-07			53d
CRECA000	AREVA Approval of Caldon EC	5d	0%	MOORE	09-Jul-07	13-Jul-07*	CRECPR040	CRECA010	0d
CRECA010	CR3 Approval	5d	0%	CR3	16-Jul-07	20-Jul-07	CRECA000	CRECI000	53d
Implementation		**28d**			**23-Jul-07**	**29-Aug-07**			**45d**
		28d			23-Jul-07	29-Aug-07			45d
CRECI000	Installation	10d	0%	CR3	23-Jul-07	03-Aug-07	CRECA010	CRECI010	53d
CRECI010	Testing	5d	0%	MOORE	16-Aug-07	22-Aug-07	CRECI000, CRFARL010, CRFARL005	CRECI020	45d
CRECI020	Commissioning	5d	0%	CR3/AREVA	23-Aug-07	29-Aug-07	CRECI010	CRECCO000	45d
Closeout		**18d**			**02-Nov-07**	**29-Nov-07**			**0d**
		18d			02-Nov-07	29-Nov-07			0d
CRECCO000	AREVA Support	18d	0%	MOORE	02-Nov-07	29-Nov-07*	CRECI020		0d
ICS Scaling EC 65630		97d			08-Mar-07 A	12-Oct-07			71d
Prepare EC		**16d**			**08-Mar-07 A**	**19-Jun-07**			**77d**
		16d			08-Mar-07 A	19-Jun-07			77d
CR-224-E03	Prepare EC (100%)	8d	0%	SPEZIALI	08-Mar-07 A	07-Jun-07	CR-202-E03	CR-226-E03, CR-225-E03	76d



Actual Work | Critical Remaining Work | Su... | Remaining Work | Milestone

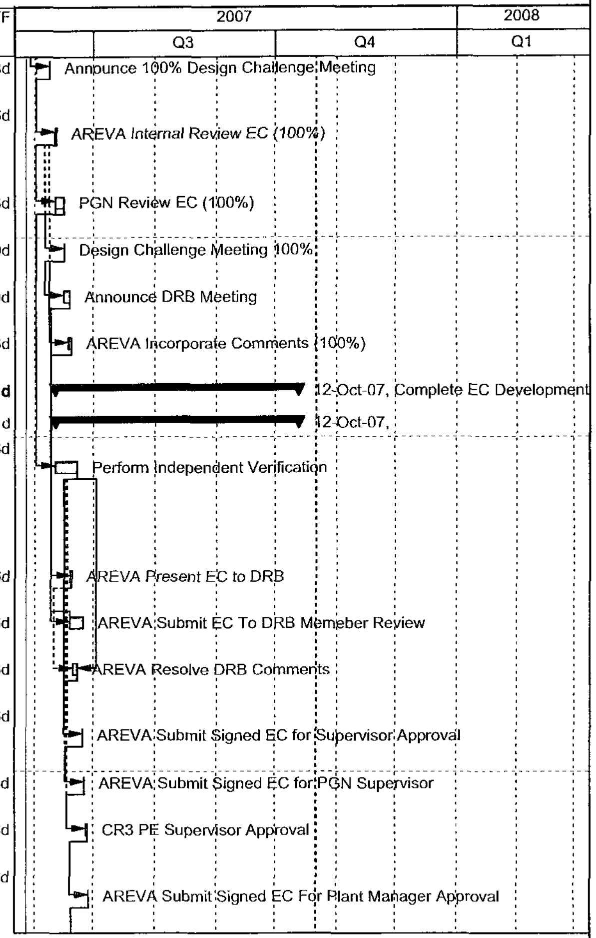
DATA DATE: 28-May-07

TASK filter: Remaining Work.

(c) Primavera Systems, Inc.

PEF-CR3-0467

Activity ID	Activity Name	RD	Physical % complete	RESP	Start	Finish	Predecessors	Successors	TF
CR-225-E03	Announce 100% Design Challenge Meeting	0d	0%	SPEZIALI	08-Jun-07	08-Jun-07	CR-224-E03	CR-226-E03	78d
CR-226-E03	AREVA Internal Review EC (100%)	2d	0%	MOORE	11-Jun-07	12-Jun-07	CR-224-E03, CR-225-E03	CR-231-E03, CR-228-E03, CR-230-E03	76d
CR-230-E03	PGN Review EC (100%)	5d	0%	CR3	11-Jun-07	15-Jun-07	CR-226-E03	CR-231-E03, CR-234-E03	76d
CR-231-E03	Design Challenge Meeting 100%	0d	0%	CR3/AREVA	15-Jun-07	15-Jun-07	CR-230-E03, CR-226-E03	CR-233-E03, CR-228-E03	79d
CR-233-E03	Announce DRB Meeting	1d	0%	CR3	15-Jun-07	18-Jun-07	CR-231-E03	CR-242-E03, CR-243-E03	79d
CR-228-E03	AREVA Incorporate Comments (100%)	2d	0%	SPEZIALI	18-Jun-07	19-Jun-07	CR-226-E03, CR-231-E03	CR-242-E03	76d
Complete EC Development & Issue		**88d**			**11-Jun-07**	**12-Oct-07**			**71d**
		88d			11-Jun-07	12-Oct-07			71d
CR-234-E03	Perform Independent Verification	10d	0%	SPEZIALI	11-Jun-07	22-Jun-07	CR-230-E03	CR-236-E03, CR-246-E03, CR-248-E03, CR-244-E03	76d
CR-243-E03	AREVA Present EC to DRB	1d	0%	SPEZIALI	19-Jun-07	19-Jun-07	CR-242-E03, CR-233-E03	CR-246-E03	76d
CR-242-E03	AREVA Submit EC To DRB Memeber Review	5d	0%	SPEZIALI	19-Jun-07	25-Jun-07	CR-233-E03, CR-228-E03	CR-243-E03	76d
CR-246-E03	AREVA Resolve DRB Comments	3d	0%	SPEZIALI	20-Jun-07	22-Jun-07	CR-243-E03, CR-234-E03	CR-244-E03	76d
CR-244-E03	AREVA Submit Signed EC for Supervisor Approval	1d	0%	SPEZIALI	25-Jun-07	25-Jun-07	CR-246-E03, CR-234-E03	CR-248-E03	76d
CR-248-E03	AREVA Submit Signed EC for PGN Supervisor	1d	0%	SPEZIALI	26-Jun-07	26-Jun-07	CR-234-E03, CR-244-E03	CR-236-E03	76d
CR-236-E03	CR3 PE Supervisor Approval	1d	0%	CR3	27-Jun-07	27-Jun-07	CR-234-E03, CR-248-E03	CR-250-E03	76d
CR-250-E03	AREVA Submit Signed EC For Plant Manager Approval	1d	0%	SPEZIALI	28-Jun-07	28-Jun-07	CR-236-E03	CR-252-E03	76d



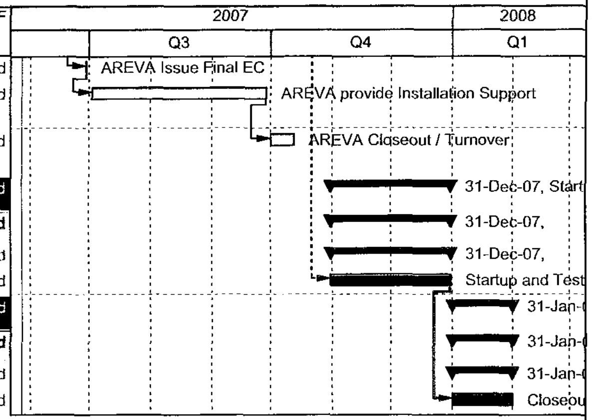
Actual Work | Critical Remaining Work | Su... | Remaining Work | Milestone

TASK filter: Remaining Work.

(c) Primavera Systems, Inc.

PEF-CR3-0468

Activity ID	Activity Name	RD	Physical % complete	RESP	Start	Finish	Predecessors	Successors	TF
CR-252-E03	AREVA Issue Final EC	1d	0%	SPEZIALI	29-Jun-07	29-Jun-07	CR-250-E03	CR-238-E03	76d
CR-238-E03	AREVA provide Installation Support	63d	0%	SPEZIALI	02-Jul-07	28-Sep-07	CR-252-E03	CR-240-E03	76d
CR-240-E03	AREVA Closeout / Turnover	10d	0%	SPEZIALI	01-Oct-07	12-Oct-07	CR-238-E03		76d
Startup and Testing		37d			31-Oct-07	31-Dec-07			0d
		37d			**31-Oct-07**	**31-Dec-07**			**0d**
		37d			31-Oct-07	31-Dec-07			0d
CRST000	Startup and Testing	44d	0%	CR3/AREVA	31-Oct-07	31-Dec-07*	CROT000	CRCO000	0d
Closeout		22d			01-Jan-08	31-Jan-08			0d
		22d			**01-Jan-08**	**31-Jan-08**			**0d**
		22d			01-Jan-08	31-Jan-08			0d
CRCO000	Closeout	23d	0%	CR3/AREVA	01-Jan-08	31-Jan-08*	CRST000		0d



Actual Work | Critical Remaining Work | Su...
Remaining Work ◆ | ◆ Milestone

TASK filter: Remaining Work.

(c) Primavera Systems, Inc.

PEF-CR3-0469

Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
MUR Uprate		3d	06-Nov-06	08-Nov-06	317d			
Kick-Off / Data Gathering		**3d**	**06-Nov-06**	**08-Nov-06**	**317d**			
		3d	06-Nov-06	08-Nov-06	317d			
CR000	Kick Off Meeting	3d	06-Nov-06*	08-Nov-06	321d			
Caldon Engineering Change		277d	02-Oct-06 A	29-Nov-07	43d			
Design Specification		**19d**	**02-Oct-06 A**	**30-Nov-06**	**0d**			
		19d	02-Oct-06 A	30-Nov-06	0d			
CRECDS000	Scope	0d	02-Oct-06 A	06-Oct-06 A		DEVENDORF		CRECDS010
CRECDS010	References	0d	02-Oct-06 A	06-Oct-06 A		DEVENDORF	CRECDS000	CRECDS020
CRECDS020	Design Input	0d	09-Oct-06 A	13-Oct-06 A		DEVENDORF	CRECDS010	CRECDS030
CRECDS030	Assumptions	0d	09-Oct-06 A	13-Oct-06 A		DEVENDORF	CRECDS020	CRECDS040
CRECDS040	Issue Draft Spec	0d	13-Oct-06 A	13-Oct-06 A		DEVENDORF	CRECDS030	CRECDS050
CRECDS050	Finalize Spec/EC	5d	16-Oct-06 A	10-Nov-06	0d	DEVENDORF	CRECDS040	CRECDS060
CRECDS060	Issue Spec/EC to Procurement	1d	15-Nov-06	15-Nov-06	0d	DEVENDORF	CRECDS050	CRECDS070
CRECDS070	PO to Caldon (by CR3)	9d	16-Nov-06	30-Nov-06	0d	DEVENDORF	CRECDS060	CRECIR000, CRECPL000
Preparation		**14d**	**21-Nov-06**	**08-Dec-06**	**266d**			
		14d	21-Nov-06	08-Dec-06	266d			
CRECPR000	CR3 RE Assemble EC Team	3d	21-Nov-06*	27-Nov-06	257d	DEVENDORF		CRECPR010, CRECPR020...
CRECPR010	CR3 RE Create EC in Passport	3d	21-Nov-06	27-Nov-06	266d	DEVENDORF	CRECPR000	CRECPR040
CRECPR020	CR3 RE Create EC Folder	3d	21-Nov-06	27-Nov-06	266d	DEVENDORF	CRECPR000	CRECPR040
CRECPR030	CR3 RE Identify NIT	6d	21-Nov-06	30-Nov-06	263d	DEVENDORF	CRECPR000	CRECPR040
CRECPR040	AREVA Review Attachment 7	12d	21-Nov-06	08-Dec-06	257d	DEVENDORF	CRECPR000, CRECPR0...	CRECK000
Procure Caldon LEFM		**198d**	**01-Dec-06**	**06-Sep-07**	**0d**			
		198d	01-Dec-06	06-Sep-07	0d			
CRECPL000	Leadtime	172d	01-Dec-06*	09-Aug-07	0d		CRECDS070	CRECPL010
CRECPL010	Alden Labs Calibration	19d	10-Aug-07	06-Sep-07*	0d		CRECPL000	CRECI000
Kick-Off / Data Gathering		**9d**	**12-Dec-06**	**22-Dec-06**	**270d**			
		9d	12-Dec-06	22-Dec-06	270d			
CRECK000	AREVA Attend KO Meeting	1d	12-Dec-06	12-Dec-06	257d	DEVENDORF	CRECPR040	CRECK020, CRECK030, C...
CRECK020	AREVA Identify ADs & AEs	3d	13-Dec-06	15-Dec-06	262d	DEVENDORF	CRECK000	CRECIR000, CRECTR000
CRECK030	AREVA Identify Inteface Reviews	3d	13-Dec-06	15-Dec-06	262d	DEVENDORF	CRECK000	CRECIR000, CRECTR000
CRECK040	CR3 RE Confirm ADs & AEs	3d	13-Dec-06	15-Dec-06	262d	DEVENDORF	CRECK000	CRECIR000, CRECTR000
CRECK010	AREVA Gather all ADs	8d	13-Dec-06	22-Dec-06	257d	DEVENDORF	CRECK000	CRECIR000, CRECTR000
Install Requirements		**5d**	**03-Jan-07**	**09-Jan-07**	**265d**			
		5d	03-Jan-07	09-Jan-07	265d			
CRECIR000	AREVA Prepare Install Requirements	5d	03-Jan-07*	09-Jan-07	256d	DEVENDORF	CRECDS070, CRECK02...	CRECTR000
Test Requirements		**5d**	**03-Jan-07**	**09-Jan-07**	**265d**			
		5d	03-Jan-07	09-Jan-07	265d			
CRECTR000	AREVA Prepare Test Requirements	5d	03-Jan-07	09-Jan-07	256d	DEVENDORF	CRECIR000, CRECK020,..	CRECTO000
OPS Turnover		**5d**	**10-Jan-07**	**16-Jan-07**	**265d**			
		5d	10-Jan-07	16-Jan-07	265d			

TASK filter: All Activities

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PEF-CR3-0470

Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRECTO000	AREVA Prepare Caldon EC	5d	10-Jan-07	16-Jan-07	256d	DEVENDORF	CRECTR000	CRECRR000
Determine Required Reviews (30% Package)		**5d**	**17-Jan-07**	**23-Jan-07**	**265d**			
		5d	17-Jan-07	23-Jan-07	265d			
CRECRR000	AREVA Prepare Required Reviews	3d	17-Jan-07	19-Jan-07	256d	DEVENDORF	CRECTO000, CRECP110	CRECRR010
CRECRR010	CR3 Confirm for 30% DRB	2d	22-Jan-07	23-Jan-07	256d	DEVENDORF	CRECRR000	
AREVA Prepare Engineering Change Package		**140d**	**17-Jan-07**	**31-Jul-07**	**130d**			
		140d	17-Jan-07	31-Jul-07	130d			
CRECP110	Issue 30% DRB	3d	17-Jan-07*	19-Jan-07	267d	DEVENDORF		CRECRR000
CRECP000	Revision Summary	5d	24-Jan-07*	30-Jan-07	0d	DEVENDORF		CRECP010, CRECP020, C...
CRECP010	Table of Contents	5d	24-Jan-07	30-Jan-07	221d	DEVENDORF	CRECP000	CRECP040
CRECP020	Problem Statement	5d	24-Jan-07	30-Jan-07	221d	DEVENDORF	CRECP000	CRECP030
CRECP120	Prepare / Issue 70% DRB	75d	24-Jan-07	08-May-07	0d	DEVENDORF	CRECAD060	CRECAD090
CRECP040	ADL - AEL	5d	31-Jan-07	06-Feb-07	221d	DEVENDORF	CRECP030, CRECP010	CRECP050
CRECP030	Design Specification	20d	31-Jan-07	27-Feb-07	221d	DEVENDORF	CRECP020	CRECP040
CRECP050	Installation Description	5d	07-Feb-07	13-Feb-07	221d	DEVENDORF	CRECP040	CRECP060
CRECP060	Testing Requirements	5d	14-Feb-07	20-Feb-07	221d	DEVENDORF	CRECP050	CRECP070
CRECP070	Turnover / Closeout Summary	5d	21-Feb-07	27-Feb-07	221d	DEVENDORF	CRECP060	CRECP080
CRECP080	Self Assessment Records	5d	28-Feb-07	06-Mar-07	221d	DEVENDORF	CRECP070	CRECP090
CRECP090	10CFR50.59	5d	07-Mar-07	13-Mar-07	221d	DEVENDORF	CRECP080	
CRECP135	Prepare / Issue 100% Draft DRB for CR3 Review	60d	02-May-07	24-Jul-07	0d	DEVENDORF	CRECR000	CRECR050
CRECP100	Issue Caldon EC to Crystal River	0d		31-Jul-07*	0d	DEVENDORF	CRECA000	
Affected Documents (70% Package)		**75d**	**24-Jan-07**	**08-May-07**	**10d**			
		75d	24-Jan-07	08-May-07	10d			
CRECAD060	Operations Procedures	5d	24-Jan-07	30-Jan-07	65d	DEVENDORF	CRECAD000	CRECAD070, CRECP120
CRECAD070	Maintenance Procedures	5d	24-Jan-07	30-Jan-07	65d	DEVENDORF	CRECAD060	CRECR000
CRECAD010	Vendor Documents	15d	24-Jan-07	13-Feb-07	35d	DEVENDORF	CRECAD000	CRECAD020
CRECAD000	Elect / I&C Dwgs	30d	24-Jan-07	06-Mar-07	35d	DEVENDORF	CRECP000	CRECAD010, CRECAD060
CRECAD080	Mech / Struct Dwgs	50d	24-Jan-07	03-Apr-07	0d	DEVENDORF	CRECP000	CRECAD090
CRECAD020	System Description	5d	14-Feb-07	20-Feb-07	35d	DEVENDORF	CRECAD010	CRECAD030
CRECAD030	DBD	5d	21-Feb-07	27-Feb-07	35d	DEVENDORF	CRECAD020	CRECAD040
CRECAD040	Elect / I&C Calculations	25d	28-Feb-07	03-Apr-07	35d	DEVENDORF	CRECAD030	CRECAD050
CRECAD050	FSAR	5d	07-Mar-07	13-Mar-07	35d	DEVENDORF	CRECAD040	CRECR000
CRECAD100	Mechanical Calculations	20d	04-Apr-07	01-May-07	0d	DEVENDORF	CRECAD090	CRECR000
CRECAD090	Stress Calculations	25d	04-Apr-07	08-May-07	0d	DEVENDORF	CRECAD080, CRECP120	CRECAD100
Review EC Package (100% Package)		**60d**	**02-May-07**	**24-Jul-07**	**0d**			
		60d	02-May-07	24-Jul-07	0d			
CRECR000	CR3 RE Upload EC in Passport	10d	02-May-07	15-May-07	0d	DEVENDORF	CRECAD100, CRECAD0...	CRECR010, CRECP135
CRECR010	CR3 Interface Reviews	9d	16-May-07	29-May-07	0d	DEVENDORF	CRECR000	CRECR020
CRECR020	CR3 RE Transmit Comments to AREVA	10d	30-May-07	12-Jun-07	0d	DEVENDORF	CRECR010	CRECR030
CRECR030	AREVA Resolve Interface Comments	10d	13-Jun-07	26-Jun-07	0d	DEVENDORF	CRECR020	CRECR040
CRECR040	AREVA Review EC	9d	27-Jun-07	10-Jul-07	0d	DEVENDORF	CRECR030	CRECR050
CRECR050	AREVA Design Verfication	10d	11-Jul-07	24-Jul-07	0d	DEVENDORF	CRECR040, CRECP135	CRECA000

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
Approve EC Package (DRB)		**10d**	**25-Jul-07**	**07-Aug-07**	**0d**			
		10d	25-Jul-07	07-Aug-07	0d			
CRECA000	AREVA Approval	5d	25-Jul-07	31-Jul-07*	0d	DEVENDORF	CRECR050	CRECA010, CRECP100
CRECA010	CR3 Approval	5d	01-Aug-07	07-Aug-07*	0d	DEVENDORF	CRECA000	CRECI000
Implementation		**20d**	**05-Oct-07**	**01-Nov-07**	**0d**			
		20d	05-Oct-07	01-Nov-07	0d			
CRECI000	Installation	10d	05-Oct-07	18-Oct-07	0d	CR3	CRECPL010, CRECA010	CRECI010
CRECI010	Testing	5d	19-Oct-07	25-Oct-07	0d	DEVENDORF	CRECI000, CRFARL010,...	CRECI020
CRECI020	Commissioning	5d	26-Oct-07	01-Nov-07	0d	CR3/AREVA	CRECI010	CRECCO000
Closeout		**20d**	**02-Nov-07**	**29-Nov-07**	**0d**			
		20d	02-Nov-07	29-Nov-07	0d			
CRECCO000	AREVA Support	18d	02-Nov-07	29-Nov-07*	0d	DEVENDORF	CRECI020, CRMEC000, ...	
MUR EC		277d	02-Oct-06 A	29-Nov-07	0d			
Design Specification		**277d**	**02-Oct-06 A**	**29-Nov-07**	**0d**			
		277d	02-Oct-06 A	29-Nov-07	0d			
CRMEC000	MUR EC	266d	02-Oct-06 A	29-Nov-07	0d	DEVENDORF		CRECCO000
Main Steam RTD Installation EC		277d	02-Oct-06 A	29-Nov-07	0d			
Design Specification		**277d**	**02-Oct-06 A**	**29-Nov-07**	**0d**			
		277d	02-Oct-06 A	29-Nov-07	0d			
CRMSEC000	Main Steam RTD Installation EC	266d	02-Oct-06 A	29-Nov-07	0d	DEVENDORF		CRECCO000
I&C Upgrade to Support MUR EC		277d	02-Oct-06 A	29-Nov-07	0d			
Design Specification		**277d**	**02-Oct-06 A**	**29-Nov-07**	**0d**			
		277d	02-Oct-06 A	29-Nov-07	0d			
CRICEC000	I&C Upgrade to Support MUR EC	266d	02-Oct-06 A	29-Nov-07	0d	DEVENDORF		CRECCO000
License Amendment Request		306d	11-Sep-06 A	11-Jan-08	14d			
		276d	**15-Sep-06 A**	**28-Nov-07**	**44d**			
		276d	15-Sep-06 A	28-Nov-07	44d			
CRLAR000	Request NRC Docket Position	0d	15-Sep-06 A	29-Sep-06 A		SCOTT		
CRLAR002	System Interviews	0d	10-Oct-06 A	13-Oct-06 A		SCOTT		
CRLAR005	Initial Walkdowns	0d	10-Oct-06 A	13-Oct-06 A		SCOTT		CRLAR034
CRLAR030	LOCA M&E	74d	26-Oct-06 A	15-Feb-07	1d	SCOTT	CRSA040	CRSA041
CRLAR100	Prepare Technical Specification Changes	10d	19-Jan-07	01-Feb-07	0d	SCOTT		CRLAR035
CRLAR110	Prepare FSAR Changes (What is required date?)	10d	19-Jan-07	01-Feb-07	0d	SCOTT		CRLAR035
CRLAR120	Licensing Information to Licensing for Review (LAR, TS, FS...	10d	19-Jan-07	01-Feb-07	0d	SCOTT		CRLAR035
CRLAR035	AREVA LAR Inhouse Review	10d	02-Feb-07	15-Feb-07*	0d	SCOTT	CRLAR100, CRLAR110, ...	CRLAR034
CRLAR032	Issue LOCA M&E to CR3 for Review	10d	16-Feb-07	01-Mar-07	230d	SCOTT	CRSA041	CRLAR033
CRLAR034	Submit LAR to CR3	0d		28-Feb-07*	-8d	SCOTT	CRLAR005, CRLAR035, ...	CRLAR130
CRLAR130	CR3 Interdiscipline Review	20d	01-Mar-07	28-Mar-07	20d	SCOTT	CRLAR034, CRSENE110	CRLAR140
CRLAR033	Incorporate Comments and Issue LOCA M&E	10d	02-Mar-07	15-Mar-07	230d	SCOTT	CRLAR032	
CRLAR140	CR3 PORC	20d	29-Mar-07	25-Apr-07	20d	SCOTT	CRLAR130	CRLAR040

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRLAR040	Submit LAR for MUR to NRC	1d	24-May-07	24-May-07*	0d	SCOTT	CRLAR140	CRLAR045
CRLAR045	Respond to NRC RAIs	133d	25-May-07	27-Nov-07*	0d	SCOTT	CRLAR040	CRLAR050
CRLAR050	NRC SER	1d	28-Nov-07	28-Nov-07*	0d	SCOTT	CRLAR045	
Thermal-Hydraulics (T-H) Models/Operating Conditions		**103d**	**11-Sep-06 A**	**31-Mar-07**	**110d**			
		103d	11-Sep-06 A	31-Mar-07	110d			
CRTH010	Develop MUR Heat Balance Uncertainty Calulation (32)	13d	11-Sep-06 A	22-Nov-06	62d	CLAUNCH		CRTH060, CRTH100
CRTH020	Benchmark and Revise PEPSE Model (12)	13d	11-Sep-06 A	22-Nov-06	62d	CLAUNCH		CRTH060, CRTH100
CRTH100	Prepare input for MUR Summary Report	33d	11-Sep-06 A	28-Feb-07	-8d	CLAUNCH	CRTH010, CRTH020, CR...	CRTH060, CRLAR034
CRTH040	Develop MUR Operating Conditions (32)	6d	29-Sep-06 A	13-Nov-06	69d	CLAUNCH		CRTH080, CRTH100
CRTH000	Establish New Operating Condition Targets	0d	19-Oct-06 A	20-Oct-06 A		CLAUNCH		CRTH100
CRTH080	CR-3 choose ICS Header Pzr and Turbine Cycle Conditions f...	6d	23-Oct-06 A	13-Nov-06	69d	CR3	CRTH040	CRTH060, CRTH100
CRTH050	Calculate FIV scaling factors.	29d	06-Nov-06 A	04-Dec-06	74d	CLAUNCH		CRTH100
CRTH110	Prepare USAR Updates	33d	06-Nov-06	20-Dec-06	42d	CLAUNCH		CRTH060, CRLAR034
CRTH120	Prepare TS Updates	33d	06-Nov-06	20-Dec-06	42d	CLAUNCH		CRTH060, CRLAR034
CRTH030	Compare MUR and 2% BOP Heat Balance	22d	15-Dec-06*	15-Jan-07	-8d	DEVENDORF		CRTH060, CRTH090, CRT...
CRTH070	CR-3 Acceptance of MUR Operating Conditions & Heat Bala...	22d	15-Dec-06	15-Jan-07	24d	CR3	CRTH030	CRTH060, CRTH100
CRTH090	Prepare/update DHR Cooldown Calculation	54d	15-Dec-06	28-Feb-07	-8d	CLAUNCH	CRTH030	CRTH060, CRTH100
CRTH060	Define Simulator Inputs	27d	01-Mar-07	31-Mar-07*	131d	CLAUNCH	CRTH080, CRTH010, CR...	CRSM000
Safety Analysis - Analysis Services Unit (ASU)		**171d**	**03-Oct-06 A**	**04-Jul-07**	**149d**			
		171d	03-Oct-06 A	04-Jul-07	149d			
CRSA010	Prepare FA Thermal-Hydraulic Input for MUR	0d	03-Oct-06 A	25-Oct-06 A		SEALS		CRSA033
CRSA030	Review Technical Specification and USAR	0d	19-Oct-06 A	01-Nov-06 A		SEALS		CRSA033
CRSA031	Review Previous Power Uprate Repts 51-5013615-00 and 5...	0d	19-Oct-06 A	01-Nov-06 A		SEALS		CRSA033
CRSA032	Review AORs and Cycle 15 Task 14 Reload Evaluation	0d	19-Oct-06 A	01-Nov-06 A		SEALS		CRSA033
CRSA020	Perform Evaluation of Safety System Setpoints	20d	20-Oct-06 A	01-Dec-06	29d	SEALS		CRSA033
CRSA040	Convert LOCA PCT Model to M&E Release Model	19d	26-Oct-06 A	30-Nov-06	1d	SEALS		CRSA041, CRLAR030
CRSA033	Prepare disposition of Events (DOE)	10d	26-Oct-06 A	15-Dec-06	29d	SEALS	CRSA020, CRSA032, CR...	CRSA100
CRSA061	Initialize Model & Run LOFW and Turbine Trip 50% Power C...	13d	30-Oct-06 A	22-Nov-06	4d	CR3		CRSA062
CRSA060	Develop AIS for AMSAC, ARTS, & MSSV Operablility Study f...	19d	30-Oct-06 A	30-Nov-06	-2d	CR3		CRSA062
CRSA050	Develop GOTHIC Model for CR3 Containment	48d	15-Nov-06*	19-Jan-07	-2d	SEALS		CRSA051
CRSA062	Perform & Document LOFW and Turbine Trip 50% Power A...	42d	01-Dec-06	29-Jan-07	-2d	CR3	CRSA061, CRSA060	CRSA100
CRSA041	Perform and document M&E Analyses	55d	01-Dec-06	15-Feb-07	1d	SEALS	CRSA040, CRLAR030	CRLAR032, CRSA100
CRSA080	Prepare Tech Spec and Bases Changes	27d	15-Jan-07*	20-Feb-07	-2d	SEALS		CRSA100
CRSA090	Prepare USAR Updates	27d	15-Jan-07*	20-Feb-07	247d	SEALS		
CRSA051	Perform & Document Containment Pressure & Temp Respo...	22d	22-Jan-07	20-Feb-07	-2d	SEALS	CRSA050	CRSA100
CRSA100	Prepare LAR Input	16d	30-Jan-07	20-Feb-07	-2d	SEALS	CRSA033, CRSA062, CR...	CRLAR034
CRSA070	Perform VLPT Setpoint Analysis	22d	05-Jun-07*	04-Jul-07	151d	SEALS	CRFAMA050	
Fuel America		**306d**	**02-Oct-06 A**	**11-Jan-08**	**14d**			
MUR Analysis		149d	02-Oct-06 A	04-Jun-07	171d			
CRFAMA075	Fuels Input to LAR	74d	02-Oct-06 A	15-Feb-07	1d	SUHOCKI		CRLAR034
CRFAMA030	Task 12: Fuel Performance (CFM, Pin Pressure, etc.)	151d	02-Oct-06 A	04-Jun-07	75d	SUHOCKI	CRFAMA000	CRFAMA080
CRFAMA050	Task 13/15: T-H DNB and VLPT Analysis	151d	02-Oct-06 A	04-Jun-07	151d	SUHOCKI	CRFAMA000	CRSA070

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRFAMA000	Task 76: Fuel Cycle Design Model	30d	06-Nov-06 A	15-Dec-06	46d	SUHOCKI		CRFAMA010, CRFAMA030...
CRFAMA120	Task 87: Approved Model & Methodology Evaluation	90d	06-Nov-06 A	09-Mar-07	234d	SUHOCKI		
CRFAMA060	Task 68: Fuel Mechanical Analysis	10d	06-Nov-06 A	22-Dec-06	289d	SUHOCKI	CRFAMA010	
CRFAMA110	Task 75: Fuel Project Management	10d	06-Nov-06 A	04-Jun-07	173d	CREASY	CRFAMA080	
CRFAMA010	Task 11: Nuclear Analysis Evaluation	65d	11-Dec-06	09-Mar-07	204d	SUHOCKI	CRFAMA000	CRFAMA020, CRFAMA060
CRFAMA020	Task 73: Maneuvering Analysis Evaluation	95d	11-Dec-06	20-Apr-07	204d	SUHOCKI	CRFAMA010	CRFAMA070
CRFAMA070	All Fuels Tasks: TS Changes, LAR, Engr, Sumry Rep Inputs	15d	02-Apr-07	20-Apr-07	204d	SUHOCKI	CRFAMA020	
CRFAMA080	FSAR Markups (All Tasks)	20d	08-May-07	04-Jun-07	75d	SUHOCKI	CRFAMA030	CRFARL010, CRFAMA110
Reload Licensing Work (at Uprated Power)		171d	18-Dec-06	15-Aug-07	46d			
CRFARL000	Revised FFCD	34d	18-Dec-06	01-Feb-07	46d	SUHOCKI	CRFAMA000	CRFARL015
CRFARL015	Cycle 16 Licensing / Draft Reload Rept / Dual COLR	116d	02-Feb-07	13-Jul-07	46d	SUHOCKI	CRFARL000	CRECI010, CRFARL010
CRFARL010	Final Reload Report	23d	16-Jul-07	15-Aug-07	46d	SUHOCKI	CRFAMA080, CRFARL015	CRECI010
FIDMS		306d	06-Nov-06	11-Jan-08	14d			
CRFANAS000	AREVA Start FIDMS Software Modifications	84d	06-Nov-06	01-Mar-07	240d	CREASY		CRFANAS030
CRFANAS030	FIDMS Software Changes Installed / Run site ISTs per SCN ...	6d	22-Feb-07	01-Mar-07	240d	CREASY	CRFANAS000	
CRFANAS010	AREVA Start FIDMS DBU	5d	03-Sep-07*	07-Sep-07	104d	CREASY		
CRFANAS040	FIDMS DBU at 2568 Installed / Run Site ISTS	8d	01-Nov-07*	12-Nov-07	58d	CREASY		
CRFANAS050	FIDMS DBU at 2609 Installed / Run Site ISTS per SCN & Sit...	8d	02-Jan-08*	11-Jan-08	14d	CREASY		
System Evaluations		81d	11-Sep-06 A	28-Feb-07	239d			
		81d	**11-Sep-06 A**	**28-Feb-07**	**239d**			
		81d	11-Sep-06 A	28-Feb-07	239d			
CRSE000	Review Issue restraints Letter SE01-0154	0d	11-Sep-06 A	22-Sep-06 A		SCOTT		
CRSE010	Compare System List w/ Previous Evaluation	0d	11-Sep-06 A	22-Sep-06 A		SCOTT		
CRSE020	Perform System Interviews	0d	10-Oct-06 A	13-Oct-06 A		SCOTT		
CRSE030	Prepare System Evaluation Report	55d	16-Oct-06 A	19-Jan-07	-26d	SCOTT	CRHV080, CRNFS140, C...	CRBFS230, CRHV090, CR...
CRSE040	CR Owner Acceptance	21d	31-Jan-07*	28-Feb-07	0d	SCOTT		
NSSS I&C Systems		**72d**	**10-Oct-06 A**	**15-Feb-07**	**248d**			
		72d	10-Oct-06 A	15-Feb-07	248d			
CRSEIC010	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		CHEATHAM	CRSEIC000	CRSEIC020
CRSENE020	Nuclear Instrumentation and Incore Monitoring	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE030	Non-Nuclear Instrumentation (NN+SP)	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE040	Emergency Feedwater Initiation & Control System	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE050	ATWS	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE060	Reactor Protection System (RP)	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE070	Control Rod Drive Control System (DR)	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE080	Integrated Control System (IC)	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE090	Radiation Monitoring (RM)	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSENE100	ECCS/Engineered Safe Guards	8d	16-Oct-06 A	15-Nov-06	3d	KEY		CRSEIC040
CRSEIC020	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		CHEATHAM	CRSEIC010	CRSEIC030, CRSEIC040, ...
CRSEIC030	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		CHEATHAM	CRSEIC020	
CRSEIC040	Draft System Evaluations (resolve followup issues as necess...	0d	31-Oct-06 A	20-Nov-06	1d	CHEATHAM	CRSEIC020, CRSENE01...	CRSEIC050
CRSENE010	Anticipatory Reactor Trip System	6d	06-Nov-06	13-Nov-06	5d	KEY		CRSEIC040

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRSEIC000	Plant Computer (PICS,SPDS,CP)	10d	06-Nov-06	17-Nov-06	1d	CHEATHAM		CRSEIC010
CRSEIC050	Internal AREVA Review & Comment Incorporation	13d	20-Nov-06	06-Dec-06	1d	CHEATHAM	CRSEIC040	CRSEIC060
CRSEIC060	*Transmit to Progress Energy for Comments*	1d	07-Dec-06	07-Dec-06	1d	CHEATHAM	CRSEIC050	CRSEIC070
CRSEIC070	Progress Energy Review	18d	08-Dec-06	02-Jan-07	1d	CR3	CRSEIC060	CRSEIC080
CRSEIC080	Resolve and Incorporate PE Comments	15d	03-Jan-07	23-Jan-07	1d	CHEATHAM	CRSEIC070	CRSEIC090, CRSENE110
CRSENE120	Prepare USAR Updates	2d	22-Jan-07*	23-Jan-07	32d	KEY		CRSENE110
CRSENE130	Prepare TS Updates	2d	22-Jan-07*	23-Jan-07	32d	KEY		CRSENE110
CRSENE110	Prepare input for MUR Summary Report	14d	24-Jan-07	12-Feb-07	32d	KEY	CRSEIC080, CRSENE13...	CRLAR130
CRSEIC090	Sign off and input into PE Records	17d	24-Jan-07	15-Feb-07	1d	CHEATHAM	CRSEIC080	CRLAR034
BOP Electrical Systems		**62d**	**10-Oct-06 A**	**01-Feb-07**	**11d**			
		62d	**10-Oct-06 A**	**01-Feb-07**	**11d**			
CRSEBE100	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		BRUCE		CRSEBE120
CRSEBE120	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		BRUCE	CRSEBE100	CRSEBE130, CRSE030
CRSEBE130	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		BRUCE	CRSEBE120	CRSEBE140
CRSEBE070	Auxiliary Transformer (UAT, SAT)	19d	06-Nov-06	30-Nov-06	4d	BRUCE		CRSEBE140
CRSEBE110	Station Auxiliary Electrical Power Distribution System	23d	06-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE140	Draft System Evaluations (resolve followup issues as necess...	23d	06-Nov-06	06-Dec-06*	0d	BRUCE	CRSEBE080, CRSEBE0...	CRSEBE150
CRSEBE020	Main Generator	12d	21-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE030	*Emergency Diesel Generator*	12d	21-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE050	Main Transformer and Isophase Bus	12d	21-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE060	DC Power	12d	21-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE080	Prepare input for BOP summary report	12d	21-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE040	Station Blackout Diesel Generator	6d	29-Nov-06	06-Dec-06	0d	BRUCE		CRSEBE140
CRSEBE090	Grid Stability	1d	06-Dec-06	06-Dec-06	0d	CR3		CRSEBE140
CRSEBE150	Internal AREVA Review & Comment Incorporation	10d	07-Dec-06	20-Dec-06	11d	BRUCE	CRSEBE140	CRSEBE160
CRSEBE160	Transmit to Progress Energy for Comments	1d	21-Dec-06	21-Dec-06	11d	BRUCE	CRSEBE150	CRSEBE170
CRSEBE170	Progress Energy Review	10d	22-Dec-06	04-Jan-07	11d	CR3	CRSEBE160	CRSEBE180
CRSEBE180	Resolve and Incorporate PE Comments	10d	05-Jan-07	18-Jan-07	11d	BRUCE	CRSEBE170	CRSEBE190
CRSEBE190	Sign off and input into PE Records	10d	19-Jan-07	01-Feb-07	11d	BRUCE	CRSEBE180	CRLAR034
NSSS Fluid Systems		**70d**	**10-Oct-06 A**	**13-Feb-07**	**3d**			
		70d	**10-Oct-06 A**	**13-Feb-07**	**3d**			
CRNFS130	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		*ESQUILLO*		*CRNFS140*
CRNFS140	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		ESQUILLO	CRNFS130	CRNFS160, CRSE030
CRNFS160	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		ESQUILLO	CRNFS140	CRNFS150
CRNFS045	*Chemical Addition (includes CA-LS, PASS)*	0d	30-Oct-06 A	03-Nov-06 A		ESQUILLO		CRNFS150
CRNFS050	Core Flood	0d	30-Oct-06 A	03-Nov-06 A		ESQUILLO		CRNFS150
CRNFS055	RB Spray (BS)	5d	06-Nov-06 A	10-Nov-06	18d	ESQUILLO		CRNFS150
CRNFS060	Make up and Purification	5d	06-Nov-06 A	10-Nov-06	18d	ESQUILLO		CRNFS150
CRNFS070	Control Rod Drive Mechanism	5d	06-Nov-06 A	10-Nov-06	18d	ESQUILLO		CRNFS150
CRNFS150	Draft System Evaluations (resolve followup issues as necess...	23d	06-Nov-06 A	08-Dec-06*	-2d	ESQUILLO	CRNFS010, CRNFS110, ...	CRNFS170
CRNFS030	Decay Heat Removal	5d	13-Nov-06*	17-Nov-06	13d	ESQUILLO		CRNFS150
CRNFS080	Spent Fuel Cooling	5d	13-Nov-06*	17-Nov-06	13d	ESQUILLO		CRNFS150
CRNFS040	OTSG	5d	20-Nov-06*	24-Nov-06	8d	ESQUILLO		CRNFS150

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRNFS010	Reactor Coolant System	6d	20-Nov-06*	27-Nov-06	7d	ESQUILLO		CRNFS150
CRNFS090	Containment Isolation	5d	04-Dec-06*	08-Dec-06	-2d	ESQUILLO		CRNFS150
CRNFS110	Emergency Feedwater	5d	04-Dec-06*	08-Dec-06	-2d	ESQUILLO		CRNFS150
CRNFS170	Internal AREVA Review & Comment Incorporation	5d	11-Dec-06	15-Dec-06	5d	ESQUILLO	CRNFS150	CRNFS180
CRNFS180	Transmit to Progress Energy for Comments	5d	18-Dec-06	22-Dec-06	5d	ESQUILLO	CRNFS170	CRNFS190
CRNFS190	Progress Energy Review	20d	27-Dec-06*	23-Jan-07	3d	CR3	CRNFS180	CRNFS200
CRNFS200	Resolve and Incorporate PE Comments	10d	24-Jan-07	06-Feb-07	3d	ESQUILLO	CRNFS190	CRNFS210
CRNFS210	Sign off and input into PE Records	5d	07-Feb-07	13-Feb-07	3d	ESQUILLO	CRNFS200	CRLAR034
NSSS Structual Analysis		**71d**	**10-Oct-06 A**	**14-Feb-07**	**2d**			
		71d	10-Oct-06 A	14-Feb-07	2d			
CRNSA220	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		SCOTT		CRNSA230
CRNSA230	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		SCOTT	CRNSA220	CRNSA250, CRSE030
CRNSA250	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		SCOTT	CRNSA230	CRNSA240
CRNSA240	Draft System Evaluations (resolve followup issues as necess...	55d	30-Oct-06 A	19-Jan-07*	-32d	SCOTT	CRNSA010, CRNSA040, ...	CRNSA270
CRNSA040	NSSS Piping & Support Branch Nozzles	20d	25-Dec-06	19-Jan-07	-32d	GREGORY	CRNSA080, CRNSA100	CRNSA240
CRNSA080	NSSS Valves (MOV's,OV's,Check Valves)	20d	25-Dec-06	19-Jan-07	-32d	GRAMBAU		CRNSA040
CRNSA130	High Energy Line Break	20d	25-Dec-06	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA140	Jet Impingement & Thrust Forces	20d	25-Dec-06	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA100	Stresses	18d	27-Dec-06	19-Jan-07	-32d	GRAMBAU		CRNSA040
CRNSA090	Core Flow Induced Vibration	15d	01-Jan-07	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA060	RC Pumps	13d	03-Jan-07	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA010	Reactor Vessel, Nozzle & Supports	10d	08-Jan-07	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA110	Usage Factors	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA120	Flow, Temperature, Pressure	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA150	Pressurized Thermal Shock	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA160	Fluence	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA170	Heat up & Cool down	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA180	Low Temperature Over Pressure	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA190	Upper Shelf Energy	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA200	Surveilance Capsule Withdrawal	10d	08-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA020	Core Support Structure & Vessel Internals	5d	15-Jan-07	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA030	Control Rod Drive Mechanism	5d	15-Jan-07	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA050	OTSG	5d	15-Jan-07	19-Jan-07	-32d	GRAMBAU		CRNSA240
CRNSA070	Pressurizer Shell, Nozzle& Surge line	1d	19-Jan-07	19-Jan-07	-32d	GREGORY		CRNSA240
CRNSA270	Transmit to Progress Energy for Comments	1d	19-Jan-07	19-Jan-07	2d	SCOTT	CRNSA240	CRNSA280, CRLAR034
CRNSA280	Progress Energy Review	5d	22-Jan-07	26-Jan-07	2d	CR3	CRNSA270	CRNSA290
CRNSA290	Resolve and Incorporate PE Comments	8d	29-Jan-07	07-Feb-07	2d	SCOTT	CRNSA280	CRNSA300
CRNSA300	Sign off and input into PE Records	5d	08-Feb-07	14-Feb-07	2d	SCOTT	CRNSA290	CRLAR034
BOP Fluid Systems		**53d**	**10-Oct-06 A**	**19-Jan-07**	**20d**			
		53d	10-Oct-06 A	19-Jan-07	20d			
CRBFS210	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		STEWART		CRBFS220
CRBFS220	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		STEWART	CRBFS210	CRBFS240, CRSE030
CRBFS240	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		STEWART	CRBFS220	CRBFS230

TASK filter: All Activities

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRBFS010	Nuclear Services & Decay Heat Seawater	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS020	Circulating Water	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS030	Main Turbine / Hydrogen Cooling	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS060	Main Steam	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS070	Condensate	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS080	Condensate Polishers	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS090	Feed Water	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS100	Condenser Air Removal	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS110	Main Condenser	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS120	Nuclear Services Closed Cycle Cooling	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS130	Secondary Services Closed Cycle cooling	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS140	Extraction Steam / HD / MSR / RH	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS160	Decay Heat Closed Cycle Cooling	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS170	Diesel Generator (Fuel,Jacket Cooling, Lube Oil, Air, Exhaust)	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS175	Industrial Cooling System	25d	23-Oct-06 A	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS230	Draft System Evaluations (resolve followup issues as necess...	25d	23-Oct-06 A	08-Dec-06*	-2d	STEWART	CRBFS010, CRBFS080, ...	CRBFS250
CRBFS180	Pipe Spec Comparison to PEPSE	21d	10-Nov-06*	08-Dec-06	-2d	STEWART		CRBFS230
CRBFS250	Internal AREVA Review & Comment Incorporation	10d	11-Dec-06	22-Dec-06	20d	STEWART	CRBFS230	CRBFS260
CRBFS260	Transmit to Progress Energy for Comments	5d	25-Dec-06	29-Dec-06	20d	STEWART	CRBFS250	CRBFS270
CRBFS270	Progress Energy Review	10d	01-Jan-07	12-Jan-07	20d	CR3	CRBFS260	CRBFS280
CRBFS280	Resolve and Incorporate PE Comments	5d	15-Jan-07	19-Jan-07	20d	STEWART	CRBFS270	CRBFS290
CRBFS290	Sign off and input into PE Records	1d	19-Jan-07	19-Jan-07	20d	STEWART	CRBFS280	CRLAR034
HVAC Systems		**73d**	**10-Oct-06 A**	**16-Feb-07**	**0d**			
		73d	10-Oct-06 A	16-Feb-07	0d			
CRHV070	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		STEWART		CRHV080
CRHV080	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		STEWART	CRHV070	CRHV100, CRSE030
CRHV100	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		STEWART	CRHV080	CRHV090
CRHV010	Main Reactor Building Cooling	5d	06-Nov-06 A	10-Nov-06	9d	STEWART		CRHV090, CRHV020, CRH...
CRHV090	Draft System Evaluations (resolve followup issues as necess...	21d	06-Nov-06 A	05-Jan-07*	-23d	STEWART	CRHV030, CRHV050, C...	CRHV110
CRHV020	ESF HVAC – (Emergency FW Pump Bldg AH)	8d	13-Nov-06	22-Nov-06	9d	STEWART	CRHV010	CRHV090
CRHV030	Emergency Diesel Generator Building HVAC	5d	27-Nov-06*	01-Dec-06	2d	STEWART		CRHV090
CRHV040	EFIX Room Cooling	5d	04-Dec-06*	08-Dec-06	-3d	STEWART		CRHV090
CRHV050	Control Complex Chillers	5d	11-Dec-06*	15-Dec-06	-8d	STEWART		CRHV090
CRHV060	Control Complex Normal HVAC, CREVS, & Habit Environment	8d	13-Dec-06*	22-Dec-06	-13d	STEWART	CRHV010	CRHV090
CRHV065	Decay Heat Pump Cooling	13d	20-Dec-06*	05-Jan-07	-23d	STEWART		CRHV090
CRHV110	Internal AREVA Review & Comment Incorporation	10d	08-Jan-07	19-Jan-07	0d	STEWART	CRHV090	CRHV120
CRHV120	Transmit to Progress Energy for Comments	5d	22-Jan-07*	26-Jan-07	0d	STEWART	CRHV110	CRHV130
CRHV130	Progress Energy Review	10d	29-Jan-07	09-Feb-07	0d	CR3	CRHV120	CRHV140
CRHV140	Resolve and Incorporate PE Comments	5d	12-Feb-07	16-Feb-07	0d	STEWART	CRHV130	CRHV150
CRHV150	Sign off and input into PE Records	1d	16-Feb-07	16-Feb-07	0d	STEWART	CRHV140	CRLAR034
Programs		**65d**	**10-Oct-06 A**	**06-Feb-07**	**8d**			
		65d	10-Oct-06 A	06-Feb-07	8d			
CRPRG160	System Engineer Interviews	0d	10-Oct-06 A	13-Oct-06 A		SCOTT		CRPRG170

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
CRPRG170	Prep Engineer Interview Forms & Transmit to CR3 Engineers	0d	16-Oct-06 A	20-Oct-06 A		SCOTT	CRPRG160	CRPRG190, CRSE030
CRPRG190	CR3 Engineers Review & Comment on Interview Forms	0d	23-Oct-06 A	27-Oct-06 A		SCOTT	CRPRG170	CRPRG180
CRPRG040	Air Operated valves	21d	06-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG180	Draft System Evaluations (resolve followup issues as necess...	21d	06-Nov-06	04-Dec-06	8d	SCOTT	CRPRG010, CRPRG040,...	CRPRG200
CRPRG080	Appendix R	20d	07-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG100	ISI- IWE/IWL	18d	09-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG090	Flow Accelerated Corrosion	15d	14-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG060	IST Pump and Valves	13d	16-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG010	Station Black Out	10d	21-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG020	In-Service Inspection - Repair & Replacement	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG030	Motor Operated Valves	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG050	Check Valves	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG110	ISI Snubber / Pipe Support	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG120	EQ	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG130	HELB / MELB	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG140	Flooding	5d	28-Nov-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG070	ISI Pressure Testing	1d	04-Dec-06	04-Dec-06	8d	CR3		CRPRG180
CRPRG200	Internal AREVA Review & Comment Incorporation	10d	05-Dec-06	18-Dec-06	8d	SCOTT	CRPRG180	CRPRG210
CRPRG210	Transmit to Progress Energy for Comments	1d	19-Dec-06	19-Dec-06	8d	SCOTT	CRPRG200	CRPRG220
CRPRG220	Progress Energy Review	15d	20-Dec-06	09-Jan-07	8d	CR3	CRPRG210	CRPRG230
CRPRG230	Resolve and Incorporate PE Comments	10d	10-Jan-07	23-Jan-07	8d	SCOTT	CRPRG220	CRPRG240
CRPRG240	Sign off and input into PE Records	10d	24-Jan-07	06-Feb-07	8d	SCOTT	CRPRG230	CRLAR034
Calculation Package Modifications		462d	06-Nov-06	18-Aug-08	-142d			
		462d	**06-Nov-06**	**18-Aug-08**	**-142d**			
		462d	06-Nov-06	18-Aug-08	-142d			
CRCPM000	Review List of Calc. Pkg. Changes from Previous Uprate	10d	06-Nov-06	17-Nov-06	-142d	KANE		CRCPM010
CRCPM010	Prepare Calculation Package Modifications	243d	20-Nov-06	24-Oct-07	-142d	KANE	CRCPM000	CRCPM020
CRCPM020	Owner's Acceptance	213d	25-Oct-07	18-Aug-08	-142d	KANE	CRCPM010	
Procedure Modifications		78d	13-Jun-07	28-Sep-07	87d			
		78d	**13-Jun-07**	**28-Sep-07**	**87d**			
		78d	13-Jun-07	28-Sep-07	87d			
CRPM000	Procedure Modifications	78d	13-Jun-07	28-Sep-07*	89d	CR3	CRSM000	
EDBD Modifications		78d	13-Jun-07	28-Sep-07	87d			
		78d	**13-Jun-07**	**28-Sep-07**	**87d**			
		78d	13-Jun-07	28-Sep-07	87d			
CREDBD000	EDBD Modifications	78d	13-Jun-07*	28-Sep-07	89d	AREVA	CRSM000	
Simulator Modifications		20d	03-Sep-07	28-Sep-07	22d			
		20d	**03-Sep-07**	**28-Sep-07**	**22d**			
		20d	03-Sep-07	28-Sep-07	22d			
CRSM000	Simulator Modifications	20d	03-Sep-07*	28-Sep-07	22d	CR3	CRTH060	CRPM000, CREDBD000, C...

TASK filter: All Activities

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Activity ID	Activity Name	RD	ES	EF	Total Float	RESP	Predecessors	Successors
Operator Training		60d	09-Jul-07	28-Sep-07	22d			
		60d	**09-Jul-07**	**28-Sep-07**	**22d**			
		60d	09-Jul-07	28-Sep-07	22d			
CROT000	*Operator Training*	60d	09-Jul-07	28-Sep-07	22d	CR3	CRSM000	CRST000
Startup and Testing		42d	31-Oct-07	31-Dec-07	0d			
		42d	**31-Oct-07**	**31-Dec-07**	**0d**			
		42d	31-Oct-07	31-Dec-07	0d			
CRST000	Startup and Testing	44d	31-Oct-07	31-Dec-07*	0d	CR3/AREVA	CROT000	CRCO000
Closeout		23d	01-Jan-08	31-Jan-08	0d			
		23d	**01-Jan-08**	**31-Jan-08**	**0d**			
		23d	01-Jan-08	31-Jan-08	0d			
CRCO000	Closeout	23d	01-Jan-08	31-Jan-08*	0d	CR3/AREVA	CRST000	

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Crystal River Unit 3

Power Uprate Project

Phase I:

Measurement Uncertainty Recapture

Project Plan

MASTER PROJECT NUMBER: 20058849

Revision History

Revision	Date	Project Manager	Project Sponsor
0	4/27/07	Ted Williams	Danny Roderick

PEF-CR3-0480

Change Basis for Revision

Original Issue

Executive Summary

Progress Energy plans to increase the electrical power output of Crystal River 3 in order to minimize cost to our customers and enhance shareholder value. Currently operating at a licensed core power level of 2568 MW_t and 903 MW_e, Crystal River 3 intends to achieve a power uprate of approximately 17%, enabling the plant to safely operate at 3014 MW_t and 1080 MW_e. The Crystal River 3 Power Uprate Project will be implemented in three phases over the course of three refueling outages. Each phase enumerated below contains definitive commercial and technical objectives:

- **Phase I – Measurement Uncertainty Recapture**

  Improvements in process instrument accuracy enable a power increase of approximately 1.6% or 14 MW_e to be attained. Existing plant safety analyses remain in effect as analyzed core power does not change.

  Phase I implementation is scheduled for 2007, during Refueling Outage (RFO) 15.

- **Phase II – Balance of Plant Efficiency**

  Replacement of low pressure turbines coincident with the planned steam generator replacement will increase the thermal efficiency of the secondary side of the plant to attain a 3% increase in electrical power or 28 MW_e, with no change in thermal power.

  Phase II implementation is scheduled for 2009, during RFO 16.

- **Phase III – Extended Power Uprate**

  Numerous plant modifications and a significant Licensing effort will enable reactor thermal output to increase by 17% to a total power of 3014 MW_t with a corresponding electrical output of approximately 1080 MW_e.

  Phase III Implementation is scheduled for 2011, following RFO 17.

This Project Plan focuses on Phase I – Measure Uncertainty Recapture. Identified herein are the project management tools, techniques and instructions necessary to achieve project goals and objectives while minimizing cost and risk. Management plans to implement work activities include:

- Project Vision, Mission Statement and Charter
- Scope, Schedule, Cost and Risk Management Plans
- Contract, Procurement and Quality Management Plans
- Human Resource and Communication Management Plans
- Safety, Human Performance and Environmental Management Plans

Based on Progress Energy Nuclear Generation Group procedures and management guidelines, this Project Plan incorporates operating experience and the best project management practices of the industry.

ACRONYMS

ADV	Atmospheric Dump Valves	MSR	Moisture Separator Reheater
AIMS	Action Item Management System	MUR	Measurement Uncertainty Recapture
ALARA	As Low As Reasonably Achievable	MUP	Make-up Pump
AMSAC	ATWS Mitigation Safety Actuation Circuitry	MW_e	Megawatts Electric
ASME	American Society of Mechanical Engineers	MW_t	Megawatts Thermal
ATWS	Anticipated Transient Without Scram	NCR	Nonconformance Report
AULD	Automatic Unit Load Demand	NGG	Nuclear Generation *Group*?
BFP	Boiler Feed Pump	NRC	Nuclear Regulatory Commission
BOP	Balance of Plant	NTM	Nuclear Task Management
CDP	Condensate Pump	NAS	Nuclear Application Software
CR3	Crystal River 3	NSSS	Nuclear Steam Supply System
CWP	Cooling Water Pump	OE	Operating Experience
DRB	Design Review Board	OMT	Outage Management Team
DTP	Detailed Task Plan	PEPM	Progress Energy Project Manager
EC	Engineering Change	PI	Performance Indicator
EDBD	Engineering Design Bases Document	PMT	Preventative Maintenance Test
EDG	Emergency Diesel Generator	PPA	Phased Project Authorization
EPU	Extended Power Uprate	PRG	Project Review Group
FIDMS	Fixed In-Core Detector Monitoring System	QA	Quality Assurance
FWHE	Feedwater Heater Exchanger	RAI	Request for Additional Information
HB	Heat Balance	RB	Reactor Building
HP	High Pressure	RCP	Reactor Coolant Pump
HPT	High Pressure Turbine	RPS	Reactor Protection System
ICS	Integrated Control System	SCP	Service Condensate Pump?
KPI	Key Performance Indicator	SPI	Schedule Performance Index
LAR	License Amendment Request	SSC	System, Structure or Component
LEFM	Leading Edge Flow Meter	TBV	Turbine Bypass Valve
LOCA	Loss of Coolant Accident	WBS	Work Breakdown Structure
LPT	Low Pressure Turbine	XFMR	Transformer
M&E	Mass & Energy	ZTEF	Zero Tolerance Equipment Failures
MFP	Main Feed Pump		

Table of Contents

Appendices

1.0 Integration Management

1.1 Introduction and Business Case

Progress Energy plans to increase the electrical power output of Crystal River 3 (CR3) in order to reduce overall costs to customers and enhance shareholder value. Currently operating at a licensed core power level of 2568 MW_t and electrical output of 903 MW_e, CR3 intends to achieve a power uprate of 17.4 %, enabling the plant to safely operate at 3014 MW_t and 1085 MW_e. The business case for a series of power up-rates was developed to seek funding from either corporate sources or through the Fuel Adjustment Clause, and is included in the Phased Project Authorization. The Florida Public Service Commission is currently reviewing a request for approval to utilize the Fuel Adjustment Clause as a source of funding for this project. The strategy to minimize risk and cost exposure is to increase power level in three distinct phases:

- **Phase I – Measurement Uncertainty Recapture (MUR)**, the subject of this Project Plan, supports an increase of approximately 1.6% or 14 MW_e. Key safety analyses supporting the licensed core power assume a two percent (2%) heat balance uncertainty. Improved instrument measurement accuracy reduces the heat balance uncertainty, which enables a licensed power uprate based on the measurement uncertainty recapture (MUR). No change in plant safety analyses is required.

  Implementation of the MUR is scheduled for 2007, following Refueling Outage (RFO) 15.

- **Phase II – Balance of Plant (BOP) Efficiency** will increase the thermal efficiency of the secondary side of the plant by replacing the Low Pressure (LP) Turbines and Moisture Separator Reheaters (MSR), and upgrading the turbine generator. It is expected the output of the plant can be increased by 26 MW_e or more, from the same licensed power level.

  Implementation of the BOP Efficiency is scheduled for 2009, following RFO 16.

- **Phase III – Extended Power Uprate (EPU)** consists of numerous plant modifications and a significant Licensing effort to maximize the power output of the facility. It is expected that the electrical output of the plant can be increased by 140 MW_e or more, with a License Amendment Request that increases power to 3014 MW_t.

  Implementation of the EPU is scheduled for 2011, following RFO 17.

This Project Plan addresses all activities associated with Phase I – MUR.

1.2 Project Implementation Approach

The CR3 Major Projects Organization will manage the MUR project, utilizing in-house and contractor personnel as needed to perform individual activities. This Project Plan provides the overall project direction and identifies management implementing strategy and tactics to achieve project goals and objectives. The overall implementation approach is based on industry recognized management techniques along with the requirements specified in

PEF-CR3-0486

NGGM-PM-0018 and other fleet procedures. In support of the project, the following assignments have been made:

- Project Sponsor: Danny Roderick
- Project Manager: Ted Williams
- Project Classification: Medium for MUR (< $5 million)

The Project is in a concurrent study and design phase. The study phase was significantly truncated since MURs are common in the industry. In addition, the evaluations performed to support the CR3 uprate from 2544 MW_t to 2568 MW_t in 2002, considered the potential to pursue the MUR and therefore are used as a basis for validating systems and programs. Further, the Robinson MUR, "Power Uprate Feasibility Study" and work performed by AREVA for Davis-Besse support the viability of implementing a MUR at CR3.

The MUR Project organization, which is presented in Appendix B and discussed in more detail in Section 4, is staffed by:

- Progress Energy Project Manager (PEPM).
- Progress Energy Regulatory Affairs
- Design engineering personnel from both Progress Energy and AREVA.
- Contractor expertise in the area of equipment design, work package development and craft management for equipment installation

The Project requires system and program evaluations to support the license amendment request (LAR). In addition, plant modifications to install more accurate feedwater flow instrumentation and other inputs to the Secondary Heat Balance will be implemented. Appropriate changes to plant control systems and computing platforms will be developed and implemented. Industry and internal operating experience is factored into evaluations and Engineering Change (EC) packages.

The Project will integrate all aspects of planning and construction management into plant on-line and outage schedules with an expectation to complete as much work on-line as possible. The Project will also include scheduling of resources required for plant support, such as reviews, owner acceptance, procedure review and revision, etc. Periodic meetings, as defined in the communications plan, will be held to keep personnel informed of Project progress. Weekly telecoms with AREVA will confirm progress toward completion of schedule activities in a manner that meets Progress Energy quality expectations. Monthly meetings between Progress Energy and AREVA management are also being held

The MUR Project will result in a power up-rate of approximately 1.6% and is expected to be fully implemented for Cycle 16 operation, following Refuel 15 in 2007.

1.3 Project Vision

Crystal River 3 Major Projects will contract and manage the design, procurement, and installation of components and equipment necessary to implement a power uprate within the approved project budget and schedule; and secure timely NRC approval of MUR and EPU License Amendment Requests that are supported by fundamentally sound technical justification and uncompromised focus on nuclear, public and employee safety.

1.4 Mission Statement

Crystal River 3 Nuclear Station
Measurement Uncertainty Recapture Uprate Project
Mission Statement

Our mission is to plan, design and successfully increase the electrical power output of Crystal River 3 thereby reducing overall costs to our customers. The project will be accomplished in a manner that is consistent with industry best practices in the areas of:

Cost Control
Industrial Safety
Human Performance
Foreign Material Exclusion
Engineering Product Quality
Control of Non-Station Personnel
Schedule Development and Adherence

We will achieve this by working together with common goals. The project team will be comprised of utility and contract staff working in a professional environment where information is shared, team members are mutually supported, contributors are accountable, and project success is achieved through the efforts of the entire organization.

Vice President Nuclear Projects &Construction	**Project Sponsor**	**Project Manager**

PEF-CR3-0488

1.5 Charter

Crystal River 3 Nuclear Station
Measurement Uncertainty Recapture Uprate Project
Team Charter

Project Success Goals include:

Industrial Safety –	A safe work environment maintained by all employees
Quality –	Processes and products that meet or exceed specifications
Radiological Safety –	A culture that strives to reduce occupational exposure to ALARA levels
Cost –	Organizations working for best possible project cost performance
Schedule –	A project planned and completed according to schedule

Governing Values include:

- Safety First
- Working to common goals
- Implementation of innovative ideas
- Effective risk management
- Mutual respect and trust
- Effective communication
- Empowered decision making at appropriate levels
- Conservative, timely and balanced decisions
- Behaviors are appropriately recognized and addressed
- Intrusive planning and thorough walkdowns

Vice President
Nuclear Projects &Construction

Project Sponsor

Project Manager

PEF-CR3-0489

1.6 Scope Overview

1.6.1 Scope Summary

The scope of the MUR Project is to perform work necessary to increase the electrical power output of CR3 by approximately 14 MW_e. The following summarizes the current list of activities that support this Project.

1.6.1.1 Plant Modifications

a) New Caldon Check-Plus Feedwater flow meters: "Leading Edge Flow Meters" (LEFM), and associated instrumentation cabinet will be installed. This requires removal of a pipe section from each Feedwater line and replacement with a pipe section that contains the Caldon LEFM. Prior to installation, the equipment will be tested at Alden Labs in a full-flow test loop that simulates the CR3 piping configuration.

b) New Main Steam Pressure and Temperature high accuracy transmitters will be installed at appropriate locations to reduce heat balance uncertainty.

c) Instrument outputs will be routed to the control complex via an Ethernet hub/switch.

d) Fixed In-Core Detector Monitoring System (FIDMS) Software 1.8 will be installed. The new FIDMS software is capable of utilizing data inputs from the new Caldon LEFM and the high accuracy Main Steam pressure and temperature transmitters. This software will replace the existing Nuclear Application Software (NAS). The change to Caldon LEFM and high accuracy Main Steam pressure and temperature transmitters also enables modification to the Automatic Unit Load Demand (AULD) heat balance calculations.

e) In conjunction with the above changes, rescaling of various Integrated Control System (ICS) modules will be specified and implemented.

f) System reviews and other information will be captured in an overall Engineering Change (EC) to support the increase in licensed power output.

1.6.1.2 System Evaluations

The systems that may be impacted by the power uprate or are required by the NRC to be discussed in the Licensing Amendment Request (LAR) will be evaluated. The intent is to use previous evaluations as a start point.

1.6.1.3 Program Evaluations

The programs that may be impacted by the power uprate or are required by the NRC to be discussed in the LAR will be evaluated. The intent is to use previous evaluations as a start point.

1.6.1.4 Fuel Analysis

The fuel load for Cycle 16 must be designed with adequate energy to support the increased power.

1.6.1.5 Calculation Package Upgrade

All calculations that may be impacted by the MUR power uprate or are required by the NRC to be included in the LAR will be evaluated and modified as necessary. The planned change to the Secondary Heat Balance Uncertainty Calculation forms the fundamental basis for the MUR Project and associated LAR. Other calculations and supporting documents will be evaluated including:

- Feedwater Pressure Calculation
- Main Steam Pressure Calculation
- Main Steam Temperature Calculation
- Letdown Flow Uncertainly Calculation
- Makeup Tank Temperature Uncertainty Calculation
- T_{cold} Narrow Range Uncertainly Calculation
- Reactor Coolant Pump (RCP) Input Calculation
- Radiative Losses Calculation
- Secondary Heat Balance Uncertainty Calculation

Revised calculations will have posted changes for the MUR and be revised as part of the EPU. A LAR will be developed in accordance with RIS 2002-03 "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications", and submitted to the NRC as shown in the schedule of Appendix C.

1.6.1.6 Enhanced Design Basis Document and Updated Final Safety Analysis Report

These documents will be revised as necessary.

1.6.1.7 Operating Procedure Revisions

This project will require significant changes to normal plant operating procedures. Dependent on NRC approval of the LAR for Cycle 16 startup, two sets of operating procedures need to be developed to address plant operation with or without the availability of the Caldon LEFM and other equipment.

1.6.1.8 Simulator Modifications

The replacement of the NAS with FIDMS as the core monitoring software system, will result in minor changes to Group Displays in both the Control Room and at the Simulator. The software changes to the AULD and the replacement of the NAS with the FIDMS are applicable to the simulator and will be implemented as either stimulated or simulated inputs.

1.6.1.9 Operator Training

Operator training will be impacted. CR3 currently operates to the heat balance power. The only change to the heat balance will be the selected power level. There are likely to be more prescriptive requirements with regard to operator and procedure response when LEFM inputs are unavailable to support the lower uncertainty heat balance. These changes will be addressed in appropriate administrative, alarm and operating procedures, and Operations will be appropriately trained to the procedure changes.

1.6.1.10 Installation and Testing

Each EC will specify appropriate installation and testing requirements. The normal Caldon scope of supply includes support for testing and commissioning of the system as well as support for the first year in-service.

1.6.1.11 Work Breakdown Structure

The official project scope is recorded in the Work Breakdown Structure (WBS), which is given in Appendix A. A summary of the individual WBS scope is listed below:

a) Project Management for scheduling, budgeting, coordinating, staff resources, communication and reporting of all aspects of the Task. Preparation, review and approval of detailed activities described in the WBS, (e.g., Communications, Risk Management, Chemical Control, FME Plans, etc.)

b) Preparation, review and approval of Industrial Safety plan(s).

c) Development, review and approval of appropriate Detailed Task Plans (DTP) as needed.

d) Planning, scheduling, preparation, review and approval of work packages required to implement all pre-outage, and outage activities.

e) Training of personnel required to support the Project activities including access and technical training. Preparation, review, and approval of all training procedures, lesson plans and mock-up training required to support project implementation.

f) Procurement of materials required to support preparations for activities, including replacement piping, consumables, bolts/nuts, parts and spares.

g) Fabrication and delivery of components.

h) Preparation of ECs, Temporary Load Restraints and Facility Change packages required for implementation and construction phases of Task Projects, including components, piping material changes, and piping/pipe support modifications.

i) Tests including: Code required testing and post modification testing such as pre-service operational tests, In-Service Inspection, start-up tests and warranty or incentive tests following start-up.

k) Identification, evaluation and preparation of procedure changes resulting from component replacement or modification.

l) Evaluating and improving reliability of structures, systems and components related to or interfacing with the modification.

m) Preparation, review and approval of Project Closeout Plan including demobilization of crews, equipment and facilities after Project completion.

n) Preparation, review and approval of Project Quality Assurance Plan that includes both Vendor and Progress Energy Quality Assurance.

o) Oversight of non-station personnel to Progress Energy standards for safety, production, and cost.

1.6.2 Assumptions

1.6.2.1 The MUR Power Uprate Project can achieve the expected increase in core power of approximately 1.6%, from a current licensed power of 2568 MW_t to 2609 MW_t.

1.6.2.2 The NRC will approve the LAR in time for implementation at startup of Cycle 16 operation

1.6.2.3 MUR related design modifications will be implemented no later than RFO-15 in fall of 2007.

1.6.2.4 MUR Project installation activities will not impact the schedule for RFO 15 with the exception of startup testing.

1.6.2.5 There will be no schedule interruptions due to work stoppages related to safety concerns, employee lost days of work due to injuries, or OSHA reportable events.

1.6.3 Critical Success Factors

Critical Success Factors are identified based on sound project management principals and industry operating experience. These factors, which are incorporated throughout this project plan, in the mission statement and the project charter are:

1.6.3.1 Involvement of station and corporate management.

1.6.3.2 Effective use of Project Management fundamentals.

1.6.3.3 Focus on Industry Experience and Lessons Learned from previous MUR power uprates.

1.6.3.4 Implementation of a comprehensive return to service and test plan.

1.6.3.5 Thorough assessments of station configuration and process controls.

1.6.3.6 Early and successful scheduling of licensing submittals with the NRC.

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1.6.3.7 Anticipation of problems and effective contingency planning (Risk Management).

1.6.3.8 Excellent Engineering Product Quality.

1.6.3.9 Provide qualified and trained company and contract labor.

1.6.3.10 Uncompromised focus on nuclear and industrial safety

1.6.4 Technical Objectives

1.6.4.1 Achieve an increase in core power of approximately 1.6% from the current licensed power of 2568 MW_t to 2609 MW_t

1.6.4.2 Design and procure required components in accordance with plant design and licensing requirements..

1.6.4.3 Develop engineering analyses, engineering change packages, implementation instructions and test plans necessary to design and implement the project activities.

1.6.4.4 Provide facilities, tools, equipment and materials needed to implement the work and support operation of modified System, Structure or Component (SSC).

1.6.4.5 Safely remove and dispose of old components.

1.6.4.6 Safely transport and install the components.

1.6.4.7 Modify existing or establish new programs and procedures that are required to operate the SSC through the balance of the licensed plant life which, after License Renewal approval, is 2036..

1.6.4.8 Provide licensing submittal that reflects changes made to support the MUR Power Uprate.

1.6.5 Performance Objectives

Specific goals will be developed based on industry best practices in the following areas. Qualitative goals are indicated until more specific objectives can be established.

1.6.5.1 Nuclear Safety

Nuclear Safety Risk, an element of Risk Management, will be managed by the station Operations group using principles and procedures that are based on conservative decision making.

The schedule will follow the normal review and approval process from an outage risk perspective using WCP-102; Outage Risk Assessment, and AI-504; Guidelines for Cold Shutdown and Refueling. Identified conditions meeting the threshold for Nonconformance Reports (NCR) will be documented in the Corrective Action Program.

1.6.5.2 Personnel Safety

Safe work practices will be utilized based on Progress Energy procedures, and government and industry standards. Personnel training will reinforce concepts of a safety conscious work ethic.

Goals will be established by all permanent, craft, and subcontractors consistent with station goals and expectations. There will be no work suspensions due to safety concerns and no employee injuries that result in lost days of work or OSHA reportable events. Rigorous management oversight of project work activities will reinforce the use of safe work practices.

1.6.5.3 Quality

The Project will comply with all codes, standards, and regulations as well as applicable corporate and plant procedures. Quality will be designed and built into each stage of the project by the Project Team. Quality Control Inspections will provide verification that quality has been designed and built into the project. The established NGG Corrective Action Program will be used to disposition all conditions adverse to quality. Appendix B suppliers may use their corrective action process, if contracted to do so. In these cases, the project team will ensure the end product is acceptable. Performance indicators will be developed to measure and monitor performance in this area.

1.6.5.4 Teamwork

The Project Team is made up of station personnel and contract personnel. This composite organization will function and succeed as one team. Incentives will reinforce the team concept and common goals.

1.6.5.5 Communications

Communications within the team, with the line staff, with primary contractors (AREVA, Bechtel and Caldon), and the NRC will be given regular attention. Critical challenges will be communicated with those who can expeditiously resolve them. Weekly project team meetings internal to the project and with AREVA will assure consistent progress. Pre-and post-LAR submittal meetings with the NRC will assure timely review and approval.

1.6.5.6 Radiation Control Practices

The Feedwater system is treated as a potentially contaminated system as a result of primary-to-secondary leakage. An appropriate ALARA plan will be developed and implemented to reduce exposure and spread of contamination.

1.6.5.7 Financial Controls

The Project Team will comply with all Progress Energy financial standards. A rigorous scope control process will be followed to control the project costs. Monthly financial reporting will be the standard for the design and planning process. Earned Value Analysis techniques and specific Cost Performance Indicators will be used whenever possible and appropriate. The objective is to implement and complete the project at or under budget without compromising quality or safety.

1.6.5.8 Planning

Effective use of risk management techniques is a critical success factor. Risk mitigation strategies and contingency plans will be developed for moderate to high risk activities and evolutions. Implementation of these strategies will effectively manage overall project risk. In general, the greater the risk of an activity or evolution, the more detailed the risk mitigation strategy item should be.

The project is set up using an organization structure that relies heavily on contractor support. Detailed plans and schedules will be developed for each activity of the project. The Project Manager will play a key role in risk management and planning as well as activity planning for the project.

1.6.5.9 Schedule

Schedules will be developed for each activity of the project. The schedule is aggressive but achievable. More detailed schedules will be developed as specific activities are firmed up. The implementation schedule will have input from the line organizations, particularly the Outage and Scheduling work group and station Operations.

1.6.5.10 Human Performance

Human Performance standards will be used to ensure station and non-station personnel are working to identify and correct error likely conditions and implement barriers to preclude human performance events.

1.6.5.11 Commercial Risk

An element of commercial risk is associated with failure to obtain NRC approval by end of RFO 15 in 2007. This and other commercial risks will be evaluated using risk evaluation techniques and risk mitigation actions. Contingency plans will be developed where appropriate. These are discussed in greater detail in Section 6.

1.6.6 Goals

The following high level goals have been established for the Project.

- Obtain NRC approval of the LAR before the end of RFO-15 in 2007.
- Obtain greater than 12 MW_e increase in plant output from 900 to 912 MW_e.
- Complete the Project with no impact to the duration of the overall outage schedule, and minimal impact to plant operations and restart.
- Complete an extended operation cycle without Foreign Material Exclusion (FME) issues attributable to project related modifications.
- Outage specific goals will be generated with respect to industrial safety, human performance, outage duration, and cost.
- Complete the project within the cost estimate.

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1.6.7 Project Acceptance Criteria

Acceptance Criteria will evolve as the project progresses. Currently, the following criteria have been established:

1.6.7.1 Satisfactory completion of all testing and inspection requirements specified in the EC's and Work Authorization packages

1.6.7.2 Caldon performance is in accordance with agreed upon performance criteria

1.6.7.3 Adherence to Progress Energy financial standards and cost control measures.

1.6.7.4 Timely closeout of project in accordance with all applicable station programs and procedures for work described in the EC package.

2.0 Scope Management Plan

2.1 Scope Statement

The scope of the CR3 Power Uprate Project Phase 1 – MUR is summarized as follows:

2.1.1 AREVA shall provide Turn Key Analysis and Design of the items below:

- **Ultrasonic Leading Edge Flow Meters** - Reduces Heat Balance Uncertainty from 2% to 0.5% (1.5% Recovery)
- **Main Steam Pressure Temperature Instrumentation** - Reduces Instrument Uncertainty (0.1% Recovery)
- **Fixed In-Core Detector Monitoring System** – Modified core monitoring software (FIDMS Revision 1.8) that incorporates new Heat Balance Algorithm
- **Automatic Unit Load Demand** - Modification to include New and Old Heat balance (Note: Work in conjunction with CR3 Design Engineering)
- **FSAR Chapter 14** –Analysis from 2568 MW_t to 2609 MW_t. Safety Analysis remains the same up to 2619 MW_t

2.1.2 Contracted labor shall perform field installation services.

2.1.3 CR3 Major Projects Group shall provide overall project management including responsibility for PMT testing and modification turn over.

2.2 Work Breakdown Structure (WBS)

2.2.1 The Phased Project Authorization provides an executive summary that establishes a common understanding between executive management, the project sponsor and other stakeholders regarding the project goals and objectives for the defined scope of work.

2.2.2 A detailed description of the project scope is presented in Section 1.6 – Scope Overview. Based on this understanding of project scope, a Work Breakdown Structure (WBS) is developed, which explicitly identifies project deliverables that comprise the project scope. The WBS is prepared and approved as part of the Design Phase Project

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Authorization, and is presented in Appendix A. Proposed changes to the approved WBS are processed using the Power Up-rate Project Integrated Change Control Plan Form, which is part of the Action Item Management System (AIMS), as discussed in Section 2.4 - Scope Control Process. The WBS can only be changed through this scope control process

2.3 WBS Dictionary

The WBS Dictionary provides the detailed descriptions associated with each WBS item. It forms the foundation for what work is to be done and thereby controls or eliminates scope creep. It also preserves the boundaries on what work is included in the task or Work Package. The WBS dictionary evolves in detail as the project, tasks and WBS items develop. Items typically detailed in the WBS Dictionary include:

- Task ID
- Assigned Individual
- Task Objectives
- Product deliverable
- Assumptions

The WBS dictionary is controlled by the Project Controls Lead individual, who is responsible to ensure that the scope and intent of each WBS item is preserved, while WBS Dictionary continues to evolve.

2.4 Scope Control Process

2.4.1 The formal process for controlling project scope and cost resides in the Phased Project Authorization Procedure. This procedure, however, is too high a level to address and control minor changes in project scope. Therefore AIMS will be used to document and track minor scope changes.

A scope change or impact is anything not currently enveloped by a WBS activity, as described in the WBS dictionary, and meets the following criteria:

- The change or impact will result in a consequential impact to the project approved budget, (i.e., can not be handled within the activity allotted funding).
- The change or impact will result in an outage duration or schedule extension.

2.4.2 Scope is controlled by using the WBS and WBS Dictionary as the reference. The following assignments are established for scope change approval:

- CR3 Project Manager: Ted Williams
- AREVA Project Manager: Dallas Scott
- CR3 Project Sponsor: Danny Roderick

Additions, deletions, or changes to the WBS and WBS Dictionary are dispositioned in a three step process as follows:

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a) The Project Manager must present preliminary cost and budget estimates for the proposed scope change to Project Sponsor. The Project Sponsor must evaluate the recommendation and make a determination of tentative approval for the request to proceed. If tentative approval is not achieved, the request is considered rejected.

b) If tentative approval is obtained, the Project Manager will authorize a formal cost estimate and schedule to be developed via the AIMS process. Once the cost estimate and schedule impact are identified, the Project Manager and Project Sponsor will determine if:

- The change in scope is encompassed by the general scope understanding established by the Scope Statement
- The cost of the scope change can be absorbed by the existing project budget as approved in the Phased Project Authorization
- The outage duration or other critical project schedule milestones remain unaffected.

c) If the answer to the above three conditions is affirmative in each case, Project Management may approve the change in scope. The WBS and WBS Dictionary will then be modified and the scope appropriately planned, scheduled, and implemented. The revised WBS and WBS Dictionary is submitted by the Project Control Lead and approved by the Project Controls Supervisor to become the official record.

If the answer to any of the above conditions is negative, Project Management will reject the request, and the WBS and WBS Dictionary will remain unchanged.

Rejected requests may receive further consideration by senior station management if the scope change is driven by:

- An unanticipated Regulatory requirement or request, or
- A technical issue that is not encompassed by contracted vendor supplied services and, if not addressed, could have a strategic impact to achievement of fundamental project goals and objectives, or
- An unanticipated industry event, operating experience or other issue that, if not addressed, could have a strategic impact to achievement of fundamental project goals and objectives.

In such event, the outcome of any decision is made at the discretion of Outage Management in the case of an outage schedule impact, or senior plant management in other cases. Revision of the Project Authorization, if necessary, will be in accordance with ACT-SUBS-00261.

2.4.3 The contract with AREVA has contingency dollars for emergent work:

- HELB analysis - approved scope
- Feedwater large bore piping analysis – approved scope
- Two additional calculations to support Secondary Heat Balance Uncertainty Calculation - approved scope.

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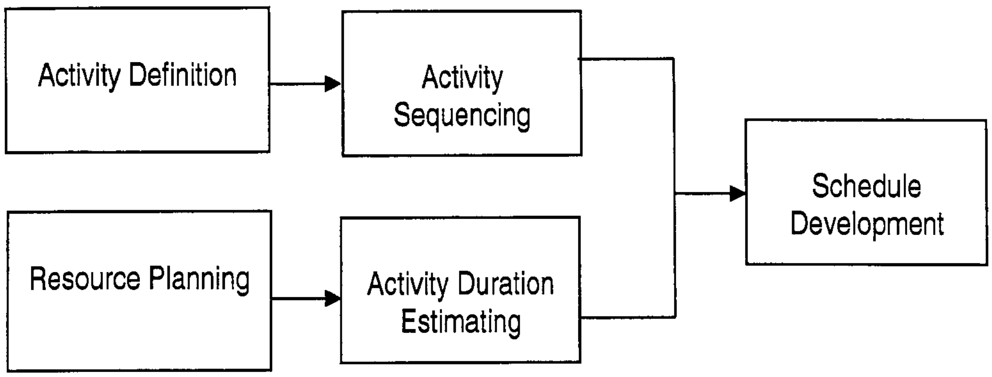
3.0 Schedule Development and Management Plan

3.1 Schedule Development Planning

Principles of time management put forth by the Project Management Institute identify four planning processes beyond the definition of project scope, that are fundamental to schedule development. These processes are shown schematically below:



The WBS identifies the project deliverables and work elements that need to be produced to achieve project objectives. Based on the WBS, activities are defined and sequenced to establish schedule logic. At the same time, the project Organization Plan (See Section 4.0) is integrated with the schedule development processes to establish activity durations. Experience obtained from similar work activities may be used to form the basis for resource allocation and activity durations. The outcome is a resource loaded schedule that identifies the following items:

- Activity
- Duration
- Due Date
- Interdependencies
- Resources needed/assigned

Throughout the life of the project, the schedule will continue to evolve utilizing a rolling-wave planning and development process. Regular updates of the Project Plan will ensure inputs to this and other management processes identified herein remain current reflecting the requirements of any revisions to Progress Energy fleet procedures and incorporating the latest industry operating experience.

3.2 Schedule Management

The initial draft Level 1 schedule is provided in Appendix C. This schedule is prepared to provide an initial outline of the overall Project and to facilitate work activities until additional details are developed.

Subsequent schedules are compared to the established baseline schedule for planning and are reviewed on a weekly basis and reported regularly in the

Monthly Report. Schedule control during the implementation outage will be in the form of Schedule Performance Index (SPI), since cost information is not as readily available on a real time basis as that of schedule performance. The baseline outage schedule will be developed in conjunction with the Outage and Scheduling organization in accordance with the milestones for schedule development for that outage. Schedules will be updated at the following frequencies:

- Design phase – once per week
- Outage phase – once per shift

The Project baseline schedule will be developed by the end of October 2006. It will continue to be refined and updated as the project evolves. This schedule will include resource loading and leveling in order to support management decisions throughout the project. Key inputs and assumptions for the schedule are as follows:

a) The WBS and WBS Dictionary are used to define and control scope and provide the activity list for the Project Schedule.

b) The Project Schedule precedence relationships and levels will be provided by the Project Scheduler with input from the core project team.

c) The Project Scheduler will incorporate activities into the Project Schedule using the WBS and the detailed activity list as approved for scope additions.

d) Risks, Assumptions, Constraints, and station milestones will be incorporated into the schedule as appropriate from the Risk Matrix, Project Plan and input from the Project Team. The following project critical milestones are identified:

    - The placement of procurement contracts for engineering analysis (Areva) and installation work.
    - Approval of ECs
    - Approval of the LAR by the NRC prior to the end of RFO 15

3.2.1 Scheduling Methodology and Assumptions

a) ProjectView® scheduling software will be the primary scheduling tool.

b) Common Production, Working & Baseline versions will be used in order to facilitate the Multi-Project features in Project View.

c) Craft resources and source will be identified as contracts are issued and estimates are provided. The Schedule will be updated using Progress Reporter.

d) Updates will be provided weekly during the Design phase of work, daily during the Pre-Outage phase, and Shift updates will be provided during Outage periods.

3.2.2 Schedule Monitoring

Schedule will be monitored by comparing schedule performance against schedule milestones.

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4.0 Resources Management

4.1 Organizational Structure

The CR3 MUR Power Level Uprate Project organization that is shown in Appendix B, reports to the Project Manager. The Project Manager, while administratively reporting to the Manager of Major Projects, functionally reports to the Project Sponsor. Titles used in this chart are functional descriptions of the organizational responsibilities rather than specific titles used in People Soft®, and as such may differ from job description titles that were used for previous projects at other NGG stations. The CR3 MUR Power Uprate Project organization will evolve to support project tasks as they arise. The following descriptions are representative of the Project.

4.2 Project Sponsor

The roles and responsibilities for the Project Sponsor are described in NGGM PM-0018, Project Management Manual, Section 3.3; and thus, need not be repeated in this project plan. Any additional expectations for the sponsor's interaction with the project team are identified herein.

4.3 Progress Energy Project Manager (PEPM)

The Project Management roles and responsibilities for the PEPM are described in PM-0018, Project Management Manual, Section 3.4. Expectations for the PEPM are described throughout the program manual.

The PEPM is becoming qualified as a Project Manager – Large Projects.

- The PEPM is responsible for initial planning, organizing, staffing, directing and controlling the project activities. This includes: Provide leadership and vision for the team.
- Establish (with stakeholders) the project objectives and success criteria.
- Organize the people and other resources to successfully accomplish the project objectives, goals, and strategies.
- Facilitate staffing the project team from a variety of sources both internal and external during the initial phase of the Project.
- Build the team, establish team culture, roles, and an environment of maximum productivity and minimal conflict.
- Direct resources as required to keep the project moving toward its goals and objectives.
- Delegate appropriate tasks and ensure project control and adherence to project plan.
- Participates in Contractor selection.
- Provides Progress Energy direct management of the Project to the subcontractors.
- Communicate across interfaces; ensure that the team members communicate effectively.
- Keep the, sponsor and senior management informed.

- Conduct periodic evaluation of project results.

Per NGGM-PM-0018; Project Management, the PEPM has the ultimate responsibility, authority, and accountability for all aspects of the project.

4.4 Functional Organization – Roles and Responsibilities

The following placeholders have been left in as representatives of duties and responsibilities.

4.4.1 Supervisor, Power Up-Rate Engineering

This individual is the functional lead for the engineering organization. Supervisor Power Up-rate Engineering is responsible for planning, organizing, staffing, directing, and controlling design activities. It will be necessary to closely interface with line organizations to obtain appropriate levels of part-time support required in support of the engineering deliverables. The Power Uprate design organization is intended to function as a subset of Major Projects engineering. This is a stand-alone and parallel engineering organization to that of the CR3 Engineering – Design group. Responsibilities for this position include but are not limited to:

- Provide engineering input to Project Plan
- Monitor and Control Engineering Scope of Work
- Perform Owners Reviews
- Review & Approve EC's
- Review Engineering Specifications
- Comment Screening/Resolution
- Pre-Implementation walkdowns
- Review Field Change Requests
- Overview Calculations, Design, & Implementation for Shielding Activities
- Preparation, Review, Processing of License Documents
- Incorporation of ALARA into Design & Implementation Activities
- Post Modification Testing
- Metallurgical and welding support
- Interface with site and corporate engineering organizations.
- Coordinate document updates with Configuration Control
- Ensure adherence to engineering schedules.

The position has the authority to:

- Direct the activities of all assigned personnel.
- Approve engineering related documentation as identified in the WBS.
- Develop and approve expenditures for the Project Engineering budget.

4.4.2 Engineering Team Member Common Duties:

- Identify stakeholders and stakeholder requirements.
- Develop an integrated Work Breakdown Structure and schedule.
- Identify risk areas and develop risk management strategy.
- Pre-outage and outage planning
- Support component replacement pre-outage planning
- Support component replacement outage planning
- 10CFR50.59 Reviews
- Employee and Management presentations as required
- Design and Implementation Meetings and Reviews.
- Pre-implementation Readiness Reviews
- Develop and implement contingency plans
- Safety Program Support
- Corrective Action Program Support
- Execute the project plan, solve problems, and remove roadblocks.
- Monitor, review, and provide input to project plan updates.
- Resolve conflicts quickly and amicably.
- Support establishing of priorities and negotiate trade-offs.
- Forecast end results.
- Ensure accurate and timely completion of project records and documentation.
- Identify training, and testing programs to achieve smooth, full operation

4.4.3 Project Lead – Procurement /Contract Management

Shared with the Replacement Once Through Steam Generator (ROTSG) Project. See Section 9.0

4.4.4 Project Lead – Project Controls (Shared with ROTSG Project)

Responsible for all financial controls and scheduling activities associated with the project. This individual compiles the elements of Cost Performance and Schedule Performance to generate Earned Value of the project, tasks, and activities. Scope control is accomplished in the Project Controls area. Risk Management is coordinated through the Project Controls area. This individual also coordinates training needs with respect to PM fundamentals for the core project team. The Project Lead – Project Controls has the authority to:

- Direct activities of the assigned scheduling and financial personnel.
- Develop and control the budget.
- Control the development of the WBS, WBS dictionary, and Work

Structure.

- Develop and approve schedule development and work processes to ensure integration into the PE Passport processes.

4.4.5 Project Lead – Quality Planning and Implementation (Shared with ROTSG)

The Project Lead is responsible for oversight of quality planning, quality assurance, and quality control. In addition, all self-evaluation initiatives are coordinated through this position. Tracking and trending of observations and NCR's is coordinated by this position. Interface with NAS, Internal Audits, and external organizations are handled through the Project Quality – functional lead. This position has the authority to:

- Direct the activities of all personnel functionally assigned to the Quality organization of the Project.
- Develop and control the budget for the group.
- Add/remove hold and witness points in work packages.
- Stop work as necessary to preserve the quality of the work.

4.5 <u>Training Qualification</u>

The Training Plan for the Project will be a part of the Training Plan for all of the up-rate projects. It will include ongoing training for PE staff as well as that necessary for contractor staff. The Training Plan is expected to include:

4.5.1 Project Management Training

4.5.2 Project team position specific training needs

4.5.3 Vendor training needs

4.5.4 Craft and Technical Training

4.5.5 End-User training needs

a) The Project Team will work with the Training Advisory Board to develop a training plan for updating each plant unit on important details of the Project.

b) Training requirements will be evaluated when new employees report to the project team. Consideration should be given to presenting work group specific training as part of the following training programs:

- Licensed Operator Re-qualification and Non-Licensed Operator Training
- Engineering Support Program
- Maintenance Continuing Training
- Other groups as appropriate

5.0 Cost Management

5.1 Financial Controls

MUR Power Up-rate Project funding is being sought from the Florida Public Service Commission through Fuel Adjustment funds. The specific work for the Project and individual activities will be authorized under a Three Phase Project Authorization Form by the Crystal River Unit 3 and Progress Energy-Florida management teams. The document approval will define the source of funds to be made available on an annual basis through the budget process.

5.2 Project Cost Estimate

The Power Uprate Project Costs are Company Confidential and will not be published in the body of this project plan. Cost data is maintained by Project Controls and is available on request to authorized personnel. The project cost is now an estimate of the total cost required to achieve the Project objectives.

The project budget is the amount authorized for the Project. The estimated Project cost and the Project budget may be different; that difference represents Cost Risk. Cost Risk will be documented in the Project Risk Matrix.

The cost estimates were developed based on a bottom-up estimate. The estimate was used as input into a Phased Project Authorization (PPA). The PPA revision is the basis for cash flows for the remainder of the project.

Costs will be itemized by Company Labor, Contractor Labor, Materials (such as component milestone payments), and Other Miscellaneous Costs (such as travel). Costs will be monitored monthly, and reports will be issued monthly through the life of the project. Costs will be accrued for each activity that has been invoiced or can be identified by a task manager or vendor as complete.

The following information is available from the MUR Power Uprate Project Controls Financial Analyst on request:

- Project Authorizations
- Project Authorization Revisions
- Detailed Project Cost Estimates
- Project Budget
- Cash flows

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6.0 Risk Management

Integral to the project is the process for evaluating and managing risk. Each Task Manager is responsible for identifying potential risk items, documenting the risk items in the AIMS database, and evaluating each risk item. Risk items identified will be documented in DTP's and dispositioned using one of three methods:

- Mitigation planning (Risk Matrix – RED area)
- Develop Contingency plans (Risk Matrix – YELLOW area)
- Accept the risk (Risk Matrix - GREEN area)

Each evaluation is complete when actions are identified that move the risk item into a green area as shown in the Risk Matrix discussed in this section of the Project Plan. The process is described in the following sub-section.

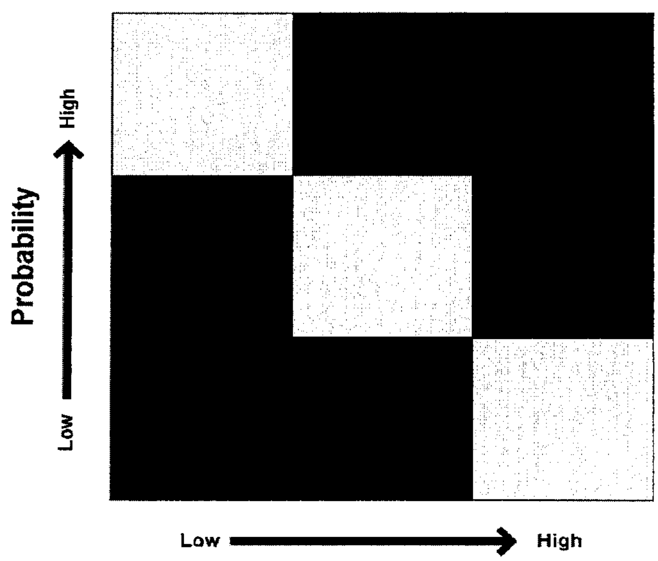
6.1 Risk Management Process

The evaluation may likely involve an iterative process. The Rick Management tool is a subset of AIMS and use of the tool is described in SGR-005.

The process is described as follows:

Step 1 is to perform the initial risk evaluation using the risk matrix developed and controlled in the AIMS database. Each risk item will be identified as an If–Then conditional statement. The item will then be evaluated independently with respect to the probability of the occurrence taking place and the consequences if the event does occur. The evaluation will assess each item as low, medium, or high for probability and consequences. The evaluation matrix is shown for discussion purposes.

Risk Matrix



Consequences

Step 2: Risk items that fall into the red or yellow area require mitigation actions to reduce the risk. A follow up evaluation will occur assuming the mitigation actions are implemented. The follow-up evaluation should result in

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reduced risk such that the result will fall into the yellow or green risk area. If the item results in a green risk, the evaluation is complete and the mitigation actions will be tracked in AIMS to closure. If the initial or secondary evaluation (assuming the mitigation plan is fully implemented) yields a result in the yellow area, a contingency plan must be established to deal with the event, if it occurs.

6.2 Decision Analysis

Decision Analysis is a tool used to formally evaluate decision options related to the Project. The Decision Analysis process is a systematic technique that will lead to decision outcomes that form the basis of a recommendation to the PEPM and appropriate station management.

The Decision Analysis process is based on the following principles:

- A consistent and documented process is used to support recommendations.
- The decision statement will be clearly defined to frame the item.
- A clear understanding of the business case relative to the project will be established.
- Realistic alternatives will be established.
- Subject Matter Experts, Operating Experience, Lessons Learned, and other reliable information will be used in the process.
- Actions will be documented using the appropriate tool for tracking.

6.3 Known Risks

6.3.1 MUR License Amendment Request (LAR) not being approved by end of RFO 15 in 2007. This designated as a Yellow risk category with medium probability and medium consequence.

6.3.1.1 Mitigation – a different set of Operation's procedures will be developed and ready if LAR is not approved.

6.3.2 Integration of hardware and software communication related to MUR plant changes. This is designated as a Yellow risk category with low probability and high consequence.

6.3.2.1 Mitigation – a dedicated IT engineer is hired to coordinate all the hardware and software modifications related to the MUR project.

6.3.3 Quality Assurance

6.3.3.1 Mitigation – see Section 8

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7.0 Procurement Management

7.1 **Fixed Price Contract for**:

7.1.1 Nuclear Steam Supply System (NSSS) and Fuel Activities performed by AREVA:

Systems evaluations, System Engineer-Design Engineer Interviews, Calculation and EDBD Update, Safety Analyses (revise LOCA M&E, update FSAR and Tech Specs.), Thermal Hydraulics (new operating conditions, FSPLIT, VAGEN and PEPSE Models,), Structural evaluation (NSSS Attached piping, review stress report summaries), Fuel America Evaluations (prepare CR3 analyses for higher power level), MUR EC.

7.1.2 BOP Mechanical and Structural Activities performed by AREVA:

System evaluations, system engineer/design engineer interviews, structural evaluations, calculation and Enhanced Design Bases Document (EDBD) update.

7.1.3 Caldon Leading Edge Flow Meters (LEFM) EC Activities performed by AREVA:

Engineering and Design for the Caldon feedwater flow meter, feedwater temperature and feedwater pressure instruments to improve heat balance uncertainty, including Caldon Spec.

7.1.4 Main Steam Pressure and Temperature Replacement EC Activities performed by AREVA:

Engineering and Design for the Main Steam pressure and temperature sensor replacement to improve heat balance uncertainty.

7.1.5 Reactor Protection Systems (RPS), ATWS Mitigation Safety Actuation Circuitry (AMSAC), ICS Function Curves, EC Activities performed by AREVA:

Revise the RPS High Flux setpoint and AMSAC setpoint, Implement revised ICS function curves.

7.2 **Fixed Price with Time and Materials Pricing to be performed by AREVA**

7.2.1 The fixed price with time and materials pricing shall consist of the following components.

- A firm fixed price for activities not designated as time and materials per the contract
- Time and materials activities are to include response to NRC requests for additional information (RAIs), contingency
- Creation of uncertainty calculations for new or installed components that will feed the LEFM heat balance, contingency
- Emergent work, contingency.

7.3 **Incentive / Penalty Targets**

Incentive/penalty targets for megawatts electric gain, Caldon EC, MUR EC, LAR Draft,

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LOCA M&E and RB Response, FIDMS Version 1.8, and Main Steam Pressure and Temperature EC to be assessed against AREVA. Reconciliation of the deliverable portion will be invoiced with the deliverables. Reconciliation of the MW_e increase will be made after actual power increase to the new licensed power level.

7.4 All procurement of material and services will be performed by Progress Energy Procurement Department.

8.0 Quality Plan

8.1 Quality Plan

One of the goals of the Project is to perform the Project in accordance with industry-best quality requirements set forth in this Project Plan. This project is neither safety-related nor Zero Tolerance Equipment Failure (ZTEF) Critical, and is not sufficiently complex to warrant a separate Quality Plan. Individual plans will be prepared, reviewed and approved as required, for individual activities that include focus areas such as receipt inspection, vendor facility surveillance, documentation and records, etc., in accordance with NGGM-PM-20 – Non Safety Related Critical Component.

Software Quality Assurance will be in accordance with NGGC EGR-157. Project assessments, audits and reviews will be scheduled as part of the Project QA Plan and will be integrated into the Project schedule. The reviews will be performance driven. Schedules and review frequency will be revised as required based on Project performance.

Throughout the various phases of the Project, testing and validation of the design and implementation will be conducted. Testing will include construction, initial operation, Code required testing and examination, functional and warranty testing. Each engineering change or component fabricated will have a plan developed that will specify required testing. Personnel conducting testing will be qualified and trained to the requirements specified by the various Codes and qualification/certification programs and standards.

The detailed Project schedule will provide for each of the required plans, controls and indicators, reviews, etc. The schedule will be developed and will be available for review through access to a shared location on the Company intranet.

8.2 Self-Evaluation Plan

The Project Plan will contain the cornerstone elements of self-evaluation.

8.2.1 Corrective Action Program

NCRs shall be used when the criteria for NCR initiation of CAP-NGGC-0200, Corrective Action Program, is met.

8.2.2 Self-Assessment Program

All self assessments will be in accordance with CAP-NGGC-0201, Self Assessment and Benchmark Programs.

- Formal and informal self-assessments will be performed periodically throughout the duration of the project.

- The first self assessment will be performed in June 2007.

8.2.3 Benchmarking

All benchmarking will be in accordance with CAP-NGGC-0201, Self Assessment and Benchmark Programs.

- Two places will be chosen to benchmark with one being Davis-Besse Nuclear Power Station.
- The first benchmarking trip will be schedule for July 2007.

8.2.4 Industry and Internal Operating Experience (OE)

The MUR project will use the Progress Energy procedure CAP-NGGC-2002, Operating Experience Program, to identify and transfer lessons learned from industry and internal events, such that those lessons are shared between NGG sites and the nuclear industry. The following OE are presented in Appendix D and will be assessed by the project team::

- INPO SER 05-01 – Large Project Implementation Strategies
- INPO SER 05-02 – Lessons Learned from Power Uprates
- INPO SER 03-04 – Reactor Overpower Events Associated with Ultrasonic Feedwater Flow Measurement Systems

8.2.5 Performance Indicators (PIs)

Develop PIs for:

- Manufacturer's performance
- Cash flow
- Schedule adherence
- Engineering change packages
- Installation work packages

8.3 Engineering Product Quality

Engineering product quality reviews will be conducted in accordance with the Engineering Product Quality Review Procedure, EGR-NGGC-0011. The project is developing the Engineering Change packages in accordance with ERG-NGGC-0005. The procedures that accompany established engineering controls will be used to ensure the engineering process is consistent with the station processes. Contracted engineering services will meet the following requirements:

8.3.1 Vendors must be on Procurement's "Approved Vendors List".

8.3.2 Vendors will prepare documents under their approved Appendix B QA program.

8.3.3 Vendors' documents will be prepared using Progress Energy's processes and templates.

8.3.4 CR3 will perform 'Owner's Acceptance" of all engineering documents.

9.0 Communication Management

9.1 Information Management

9.1.1 Action Item Management System (AIMS)

AIMS is a tool that has been specifically developed to provide enhanced search and sort capabilities for tracking and reporting information below the threshold of Nonconformance Report (NCR) and Nuclear Task Management (NTM). These are typically internal action items, lessons learned items, tracking of benchmark actions, risk management actions, etc.

AIMS will not be used for tracking actions assigned to external/line organizations. These will be tracked using the NTM process. Conditions adverse to quality, significant conditions adverse to quality, and process enhancements will be tracked using the NCR process. Detailed discussion of AIMS and its use and capabilities are discussed in document SGR-005.

9.1.2 Implementing Plans and Procedures

Control documents will be developed for consistent application of process. These will continue to be developed as the project evolves. The original copies can be found at the Department Administrators office.

9.1.3 Key Performance Indicators

Key Performance Indicators (KPIs) will be established and are controlled by the Project Controls - Functional Lead.

9.1.4 Project presentation and schedule:

a) Project Review Group (PRG) – quarterly

b) Outage Management Team (OMT) – monthly

c) Monthly Executive Review Meeting (AREVA/CR3 senior management) - monthly

d) Sponsor – weekly

e) Project Team (phone call/schedule/action item) – weekly

10.0 Safety Management

10.1 One member of the Project Team (typically the on-site Task Leader) will have the responsibility to ensure that the project construction and installation phases are performed safely. This individual will:

a) provide safety training to project team members as required or necessary;

b) be knowledgeable of the safety procedures and practices;

c) have the responsibility to ensure that safety procedures and standards are adhered to during the project;

d) ensure that all persons working on the project have and properly use the required personal protective equipment;

e) make routine inspections of all project work areas to ensure safe work practices are being followed;

f) respond to all identified safety concerns;

g) work with project management to correct any safety concern or problem in an efficient manner; and

h) direct (as necessary) work stoppage to correct safety violations and prevent injury to personnel.

10.2 Applicable site and fleet processes will be used throughout the project.

11.0 Human Performance Plan

11.1 AI-1851, Site Observation Program, describes the program to be used by the MUR Project to perform human performance observations. This program is applicable to all MUR personnel and is to be performed by management and peers alike.

11.2 The MUR Project will use the existing AI-1851, Site Observation Program to:

- Provide a process to observe, review, and document human performance behaviors
- Provide information which management can use to improve human performance
- Provide a process through which human performance data is compiled for identification of project trends
- Ensure management expectations are achieved and effectively communicated, and awareness is promoted between interfacing work groups by observation of performance and expectations

12.0 Environmental Management

All site and fleet processes will be used throughout the project.

13.0 Financial Management

Items addressed by this section are included in the Cost Management Section of this project plan, or are available in the Project Manager Book of Knowledge in project files.

14.0 Contract Management

14.1 Contract Development

All site and fleet processes will be used throughout the project.

14.2 Incentive Basis

Where appropriate, incentives may be established. Due to the nature of the work, the complexity of the work, and the timing of the work, contract incentives may vary. In every case, incentives are intended to have subcontractor support organizations aligned with the overall goals of the station and project team. Success is realized when the project is successful.

14.3 Claim Management

Addressed in each contract.

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Appendix A

MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M	**Measurement Uncertainty Recapture**					**64%**
M1	**Project Management**	**Ted Williams**				**42%**
M1.1	Project Plan *Notes* Issue Rev 0 of Staffing Plan	Ted Williams	Ken Wilson			50%
M1.2	Staffing Plan *Notes* Issue Rev. 0 of Staffing Plan	Ted Williams	George Englert			50%
M1.3	Construction Vendor Selection	Ted Williams	Robin Kibler			50%
M1.4	Divisions of Responsibilities *Notes* Issue Contract WA 51 To AREVA Contract 101659	Ted Williams	Debbi Hanna			100%
M1.5	Management Presentations *Notes* Establish Management Update Frequency.	Ted Williams	Ted Williams			50%
M1.6	**Project Controls**	**Gene Flavors**				**67%**
M1.6.1	Develop WBS *Notes* Issue Rev. 0 of MUR WBS	Gene Flavors	Cal Goering			50%
M1.6.2	Develop WBS Dictionary *Notes* Issue Rev. 0 of MUR WBS Dictionary	Gene Flavors	Cal Goering			50%
M1.6.3	Integrated Change Control Plan *Notes* Incorporate SGR Change Control Plan Into MUR Project	Gene Flavors	Cal Goering			100%
M1.7	**Schedule Development**	**Gene Flavors**				**33%**
M1.7.1	Level I Project Schedule *Notes* Issue Initial Level I Schedule	Gene Flavors	Cal Goering			100%
M1.7.2	**Level III Project Schedule**	**Gene Flavors**				**20%**
M1.7.2.1	Licensing Prep Schedule *Notes* Establish Initial Lvl III Engineering Schedule	Gene Flavors	Cal Goering			50%

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M1.7.2.2	EC Engineering Schedule *Notes* Establish Initial Lvl III EC Schedule	Gene Flavors	Cal Goering			50%
M1.7.2.3	Work Planning Schedule *Notes* Establish Initial Lvl III WO Schedule	Gene Flavors	Cal Goering			0%
M1.7.2.4	Field Construction Schedule *Notes* Establish Initial Lvl III Construction Schedule	Gene Flavors	Cal Goering			0%
M1.7.2.5	Outage Schedule Input *Notes* Establish Initial Lvl III Outage Schedule	Gene Flavors	Cal Goering			0%
M1.8	**Cost Management Plan**	**Gene Flavors**				**13%**
M1.8.1	Design Phase PA *Notes* Signed PA	Gene Flavors	Tina Suits			50%
M1.8.2	Implementation Phase PA *Notes* Signed PA	Gene Flavors	Tina Suits			0%
M1.8.3	**Budget Preparation**	**Gene Flavors**				**0%**
M1.8.3.1	Bottoms Up Estimate Field Construction *Notes* Establish Bottoms Up Estamates for Field Activities	Gene Flavors	Robin Kibler			0%
M1.8.4	Monthly cost Reports *Notes* Issue Initial Monthly Cost Reports	Gene Flavors	Tina Suits			0%
M1.9	**Contract Administration**	**Gene Flavors**				**33%**
M1.9.1	**Establish Contracts**	**Gene Flavors**				**33%**
M1.9.1.1	Caldon Contract *Notes* Issue Contract #44867 to Caldon	Gene Flavors	Debbie Hanna			100%
M1.9.1.2	AREVA Contract *Notes* Issue Contract WA 61 To AREVA Contract 101659	Gene Flavors	Debbie Hanna			100%

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M1.9.1.3	UT Contract (QC) *Notes* Issue Contract	Gene Flavors	Debbie Hanna			0%
M1.9.1.4	Bechtel Contract *Notes* Issue Contract	Gene Flavors	Debbie Hanna			0%
M1.9.1.5	Intercom Contract *Notes* Issue Contract	Gene Flavors	Debbie Hanna			0%
M1.9.1.6	Radiography Contract (QC) *Notes* Issue Contract	Gene Flavors	Debbie Hanna			0%
M1.10	**Risk Management**	**Gene Flavors**				**100%**
M1.10.1	Develop Risk Analysis Process *Notes* Incorporate SGR AIMS into MUR Project	Gene Flavors	Gene Flavors			100%
M2	**Admin Support**	**Gene Flavors**				**0%**
M2.1	**Records management Plan**	**Gene Flavors**				**0%**
M2.1.1	Determine Scope of Documents Requiring Record Keeping *Notes* Establish List of Required Records	Gene Flavors	George Engert			0%
M2.1.2	Identify Document Services Requirements *Notes* Establish Required Document Services	Gene Flavors	Janet Schroeder			0%
M2.2	**Process Operating Experience**	**Ken Wilson**				**0%**
M2.2.1	OE Review *Notes* Establish OE Plan	Ken Wilson	Ken Wilson			0%
M2.2.2	Benchmarking Plan *Notes* Establish Benchmarking Plan	Ken Wilson	Ken Wilson			0%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M2.2.3	Corrective Action Review *Notes* Establish Corrective Action Review Plan	Ken Wilson	Ken Wilson			0%
M2.2.4	Lessons Learned Review *Notes* Establish Lessons Learned Review Plan	Ken Wilson	Ken Wilson			0%
M2.3	**Quality Control**	**Bil Nielsen**				**0%**
M2.3.1	Quality Control Plan *Notes* Establish Quality Control Plan	Bil Nielsen	Bil Nielsen			0%
M2.3.2	Evaluate Contractor NDE Quals *Notes* Establish Contractor NDE Quals Plan	Bil Nielsen	Bil Nielsen			0%
M3	**Procurement**	**Gene Flavors**				**67%**
M3.1	Caldon Material Specification EC # 65566 *Notes* Issue EC # 65566	Gene Flavors	George Hildebrandt			100%
M3.2	Caldon Equipment Purchase Order *Notes* Issue Contract to Caldon	Gene Flavors	Debba Hanna			100%
M3.3	Caldon Equipment Receipt Inspection Plan	Gene Flavors	Andy Harmon			0%
M4	**Engineering** *Notes* Cr3 to Provide References (Design Manual, Piping Spec., Calculations, etc.	**George Englert**				**88%**
M4.1	**System Evaluations** *Notes* CR3 to Provide list of Design Mod's & Plant Operational Changes Since 2002 CR3 to Provide System Evaluations/Program Changes CR3 to Provide a List of New or Revised Calculations	**George Englert**				**99%**
M4.1.1	**NSSS Fluid Systems**	**George Englert**				**100%**
M4.1.1.1	Control Rod Drive Mech *Notes* AREVA WBS # 5.05.01	George Englert	George Englert			100%

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MUR WBS Dictionary as of Fri 2/16/07
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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
"Control Rod Drive Mech" continued	Notes Owner Acceptance of Doc # 51-XXXXXXX-000					
M4.1.1.2	Core Flood Notes AREVA WBS # 5.05.02 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.3	Make-Up and Purification Notes AREVA WBS # 5.05.04 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.4	Chemical Addition Notes AREVA WBS # 5.05.06 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.5	Liquid Sampling Notes Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.6	Hydrogen Notes Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.7	Decay Heat Removal Notes AREVA WBS # 5.05.07 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.8	Reactor Coolant Notes AREVA WBS # 5.05.08 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.9	Building Spray Notes AREVA WBS # 5.05.09 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.10	Spent Fuel Pool Cooling Notes AREVA WBS # 5.05.10	George Englert	George Englert			100%

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
"Spent Fuel Pool Cooling" continued						
	Notes Owner Acceptance of Doc # 51-XXXXXXX-000					
M4.1.1.11	Containment Isolation Notes AREVA WBS # 5.05.11 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.1.12	Emergency Feedwater Notes AREVA WBS # 5.05.12 Owner Acceptance of Doc # 51-XXXXXXX-000	George Englert	George Englert			100%
M4.1.2	**NSSS Components (mechanical/structural/material)**	**George Englert**				**100%**
M4.1.2.1	Reactor Vessel, Nozzles & Supports Notes AREVA WBS # 5.08.01 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.2	Core Support and Internals Notes AREVA WBS # 5.08.02 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.3	CRDM Notes AREVA WBS # 5.05.01 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.4	NSSS Piping, Pipe Supports, Branch Nozzles Notes AREVA WBS # 5.08.03 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.5	Once Through Steam Generator - Tubes etc. Notes AREVA WBS # 5.05.05 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.6	Reactor Coolant Pump Supports Notes AREVA WBS # 5.08.04 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.1.2.7	Pressurizer Shell, Nozzles, Surge Line *Notes* AREVA WBS # 5.08.06 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.8	Safety Related Valves *Notes* AREVA WBS # 5.08.14 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.9	Fluance Evaluation *Notes* AREVA WBS # 5.08.10 Owner Acceptance of Doc # 51-9038958-000	George Englert	George Englert			100%
M4.1.2.10	Heatup and Cooldown P-T Curve Update *Notes* AREVA WBS # 5.08.11 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.11	PWSCC/Alloy 600/PTS/Upper Shelf Energy/Surveillance Capsule Withc *Notes* AREVA WBS # 5.08.12 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.2.12	OTSG Tubes, Secondary Side Internal Support Structures, Shell & Sup *Notes* AREVA WBS # 5.08.13 Owner Acceptance of Doc # 51-9036246-000	George Englert	George Englert			100%
M4.1.3	**NSSS I&C and Electrical Systems** *Notes* CR3 to Provide Distribution System Calc CR3 to Provide Electric Plant Loading Model	**George Englert**				**100%**
M4.1.3.1	Anticipated Transient Without Scram *Notes* AREVA WBS 5.07.01 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.2	ECCS/Engineered Safeguards *Notes* AREVA WBS 5.07.10 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.1.3.3	Emergency Feedwater Initiation & Control *Notes* AREVA WBS 5.07.04 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.4	Integrated Control *Notes* AREVA WBS 5.07.06 & 7.01 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.5	Non-Nuclear Instrumentation *Notes* AREVA WBS 5.07.03 & 7.01 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.6	Nuclear Instrumentation *Notes* AREVA WBS 5.07.02 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.7	Reactor Protection System *Notes* AREVA WBS 5.07.05 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.8	Control Rod Drive System *Notes* AREVA WBS 5.07.07 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.9	Incore Monitoring System *Notes* AREVA WBS 5.07.02 Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.3.10	Radiation Monitoring *Notes* Owner Acceptance of Doc # 51-9035360-000	George Englert	George Englert			100%
M4.1.4	**BOP Fluid Systems**	**George Englert**				**100%**
M4.1.4.1	Main Steam *Notes* AREVA WBS 6.15 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%

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MUR WBS Dictionary as of Fri 2/16/07
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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.1.4.2	Condensate Notes AREVA WBS 6.18 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.3	Condenser & Condenser Vacuum Notes AREVA WBS 6.19 & 6.21 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.4	Feedwater (Main, Aux & FW Heaters) Notes AREVA WBS 6.16 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.5	Main Turbine & Auxiliaries Notes AREVA WBS 6.09 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.6	Moisture Separator Reheaters Notes AREVA WBS 6.13 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.7	Extraction Steam Notes AREVA WBS 6.25 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.8	Heater Drains Notes AREVA WBS 6.26 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.9	Circulating Water Notes AREVA WBS 6.08 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.10	Decay Heat Closed Cycle Notes AREVA WBS 6.28 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.1.4.11	Nuclear Services & Decay Heat Seawater *Notes* AREVA WBS 6.07 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.12	Secondary Services Closed Cycle Cooling *Notes* AREVA WBS 6.24 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.13	Piping & Supports *Notes* Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.14	Liquid Waste Disposal *Notes* Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.15	Diesel Fuel & Cooling & Lube Oil *Notes* AREVA WBS 6.30 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.16	Hydrogen Cooling *Notes* AREVA WBS 6.10 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.17	Stator Cooling *Notes* AREVA WBS 6.11 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.4.18	Condensate Polishing Demineralizer *Notes* AREVA WBS 6.20 Owner Acceptance of Doc # 51-9036484-000	George Englert	George Englert			100%
M4.1.5	**BOP Electrical and I&C** *Notes* CR3 to Provide Distribution System Calc CR3 to Provide Electric Plant Loading Model	**George Englert**				**95%**

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WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.1.5.1	DC Electrical Notes AREVA WBS 7.01 Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.2	Diesel Generator Notes AREVA WBS 7.04 CR3 to Provide EDG Loading Calc. Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.3	**Turbine Generator**	**George Englert**				**100%**
M4.1.5.3.1	Main Generator Notes AREVA WBS 7.03 Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.3.2	500 KV Tie Lines Notes Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.3.3	Step UP Transformers Notes AREVA WBS 7.06 Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.3.4	Unit Auxiliary Transformer Notes AREVA WBS 7.06 Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.3.5	Isolated Phase Bus Duct & Cooling Notes AREVA WBS 7.06 & 6.12 Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.4	AC Vital Bus Notes Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%
M4.1.5.5	Motor Control Feeder Busses Notes Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			100%

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Appendix A

MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.1.5.6	Grid Stability Notes AREVA WBS 7.10 Owner Acceptance of Doc # 51-9037444-000	George Englert	George Englert			50%
M4.1.6	**HVAC**	**George Englert**				**100%**
M4.1.6.1	Reactor Building Main Cooling Notes AREVA WBS 6.50.01 Owner Acceptance of Doc # 51-9036250-000	George Englert	George Englert			100%
M4.1.6.2	ESF HVAC - (Emergency FW Pump Bldg AH) Notes AREVA WBS 6.50.02 Owner Acceptance of Doc # 51-9036250-000	George Englert	George Englert			100%
M4.1.6.3	Control Complex Normal HVAC, CREVS, & Habit Environmental Notes AREVA WBS 6.50.03 Owner Acceptance of Doc # 51-9036250-000	George Englert	George Englert			100%
M4.1.6.4	Post LOCA Combustible Gas Notes AREVA WBS 6.50.04 Owner Acceptance of Doc # 51-9036250-000	George Englert	George Englert			100%
M4.2	**Thermal-Hydraulics (T-H) Models/Operating Condition** Notes CR3 to Provide a List of New of Revised Calculations	**George Englert**				**71%**
M4.2.1	Establish New Operating Condition Targets Notes AREVA WBS 3.01 Owner Acceptance of New Operating Condition Targets	George Englert	George Englert			100%
M4.2.2	**Develop MUR Heat Balance Uncertainty Calculation** Notes AREVA WBS 3.02 Owner Acceptance of MUR EC # 65627	**Ken Wilson**	**George Hildebrandt**			**50%**
M4.2.2.1	I&C F/W Pressure Calculation	Ken Wilson	George Hildebrandt			50%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.2.2.2	I&C M/S Pressure Calculation Notes Owner Exceptance of Calculation	Ken Wilson	George Hildebrandt			50%
M4.2.2.3	I&C M/S Temperature Calculation Notes Owner Exceptance of Calculation	Ken Wilson	George Hildebrandt			50%
M4.2.2.4	Letdown Flow Uncertainty Calculation Notes Owner Exceptance of Calculation	Ken Wilson	George Hildebrandt			50%
M4.2.3	Benchmark & Revise PEPSE Model (12) Notes AREVA WBS 3.03 CR3 to Provide Most Current PEPSE Model Address PEPSE Model Issues Compare MUR and 2% BOP Heat Balance Develop MUR Operating Conditions (32) Define Simulator Inputs	George Englert	George Englert			100%
M4.2.4	**CR3 Acceptance of MUR Operating Conditions & Heat** Notes AREVA WBS 3.07	**George Englert**				**100%**
M4.2.4.1	CR3 Choose ICS Header Pressure & Turbine Cycle Notes Establish ICS Header Pressure & Turbine Cycle	George Englert	George Englert			100%
M4.3	**Fuel Analysis (Fuel America)** Notes CR3 to Provide a List of New or Revised Calculations	**Ken Wilson**				**78%**
M4.3.1	Task 11: Nuclear Analysis Notes AREVA WBS 4.01 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			50%
M4.3.2	Task 12: Fuel Rod Thermal Design & Performance Analysis Notes AREVA WBS 4.02 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			50%

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Appendix A

MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.3.3	Task 13: Core Hydraulics *Notes* AREVA WBS 4.03 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			100%
M4.3.4	Task 15: Core Thermal-Hydraulics (Total for Task 12/13/15) *Notes* AREVA WBS 4.04 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			50%
M4.3.5	Task 65: Core Follow & DSU Updates *Notes* AREVA WBS 4.05 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			100%
M4.3.6	Task 66: Fuel Rod Mechanical Design & Performance *Notes* AREVA WBS 4.06 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			50%
M4.3.7	Task 71: Fuel Assembly Mechanical Performance *Notes* AREVA WBS 4.07 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			100%
M4.3.8	Task 73: Maneuvering Analysis *Notes* AREVA WBS 4.08 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			100%
M4.3.9	Task 75: Fuel Project Management *Notes* AREVA WBS 4.09 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			100%
M4.3.10	Task 76: Fuel Cycle Design & Analysis *Notes* AREVA WBS 4.10 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			100%
M4.3.11	Task 81: Summary Reports *Notes* AREVA WBS 4.11	Ken Wilson	Mike Devoe			50%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
"Task 81: Summary Reports" continued	Notes Owner Acceptance of Cycle Reload Report					
M4.3.12	Task 87: Approved Model & Methodology Evaluation Notes AREVA WBS 4.13 Owner Acceptance of Cycle Reload Report	Ken Wilson	Mike Devoe			90%
M4.4	**Program Evaluations** Notes CR3 to Provide System Evaluations/Program Changes	**Ken Wilson**				**100%**
M4.4.1	**NSSS Design**	**Ken Wilson**				**100%**
M4.4.1.1	Environmental Qualification Notes Completion of NTM 219642.	Ken Wilson	Danny Adler			100%
M4.4.1.2	Station Blackout Notes Completion of NTM 219642.	Ken Wilson	Roy Lane			100%
M4.4.2	**Fire Protection**	**Ken Wilson**				**100%**
M4.4.2.1	Fire Protection (Base Program) Notes Completion of NTM 219642.	Ken Wilson	John Curham			100%
M4.4.2.2	Fire Protection (Appendix R) Notes Completion of NTM 219642.	Ken Wilson	John Curham			100%
M4.4.3	**Components & Systems**	**Ken Wilson**				**100%**
M4.4.3.1	Appendix J Notes Completion of NTM 219642.	Ken Wilson	Brent Tolan			100%
M4.4.3.2	Cooling Water Reliability Notes Completion of NTM 219642.	Ken Wilson	Jim Lane			100%
M4.4.3.3	IST Notes Completion of NTM 219642.	Ken Wilson	Brent Tolan			100%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.4.3.4	Maintenance Rule *Notes* Completion of NTM 219842.	Ken Wilson	Louis Barbieri			100%
M4.4.3.5	Pridictive Maintence (PdM) *Notes* Completion of NTM 219842.	Ken Wilson				100%
M4.4.3.6	**Valves**	**Ken Wilson**				**100%**
M4.4.3.6.1	AOV's *Notes* Completion of NTM 219842.	Ken Wilson	Larry Ganstine			100%
M4.4.3.6.2	Check Valves *Notes* Completion of NTM 219842.	Ken Wilson	Larry Ganstine			100%
M4.4.3.6.3	MOV's *Notes* Completion of NTM 219842.	Ken Wilson	Don Vaccaro			100%
M4.4.4	**Material Integrity**	**Ken Wilson**				**100%**
M4.4.4.1	Boric Acid Corrosion Control (BACC) *Notes* Completion of NTM 219842.	Ken Wilson	John Mueller			100%
M4.4.4.2	Flow Accelerated Corrosion (FAC) *Notes* Completion of NTM 219842.	Ken Wilson	John Mueller			100%
M4.4.4.3	**ISI Programs**	**Ken Wilson**				**100%**
M4.4.4.3.1	IWE/IWL *Notes* Completion of NTM 219842.	Ken Wilson	Rick Portman			100%
M4.4.4.3.2	Pressure Testing *Notes* Completion of NTM 219842..	Ken Wilson	Mike Brannin			100%
M4.4.4.3.3	Repair Replacement *Notes* Completion of NTM 219842.	Ken Wilson	Jeff Finnell			100%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M4.4.4.3.4	Snubbers / Supports *Notes* Completion of NTM 219642.	Ken Wilson	Rick Portman			100%
M4.4.4.3.5	Weld Exams *Notes* Completion of NTM 219642.	Ken Wilson	Mike Brannin			100%
M4.4.4.5	Safety Related Coatings *Notes* Completion of NTM 219642.	Ken Wilson	Chris Beechwood			100%
M4.4.4.6	Steam Generator Life Cycle Management *Notes* Completion of NTM 219642.	Ken Wilson	Scot Stewart			100%
M4.5	Safety Analysis - (AREVA Analysis Unit (ASU)) *Notes* CR3 to Provide a List of New or Revised Calculations CR3 to Provide Current FSAR CR3 to Provide Current Tech Spec. Prepare Fuel America Thermal-Hydraulic Input for MUR Perform Evaluation of Safety System Set Points Review Technical Specification and FSAR Review Previous Power Uprate Reports 51-5013615 Review AORs & Cycle 15 Task 14 Reload Evaluation Prepare Disposition of Reports (DOE) Perform & Document M & E Analysis Perform & Document Containment Pressure & Temperature Response Perform & Document ARTS Arming Set Point, AMSAC Enabled Set Point Prepare Tech Spec & Bases Changes Prepare FSAR Changes Owner Acceptance of Core Reload Report Prepare Final Licensing Report Input	Ken Wilson	Ken Wilson			50%
M4.6	Obtain License Amendment From NRC *Notes* CR3 to Provide Current FSAR CR# to Provide Current Tech. Spec. Safety Analysis Input Prepare MUR Summary Report Prepare Tech Spec Markups & Related FSAR Markups AREVA Review of MUR Summary Report CR3 Review of MUR Summary Report	Ken Wilson	Ken Wilson			50%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
"Obtain License Amendment From NRC" continued	*Notes* AREVA Resolve Comments Prepare LAR AREVA Review LAR CR3 Review LAR Submit LAR to NRC AREVA/CR3 Resolve NRC RAI's Obtain Docketed SER From the NRC					
M4.7	**EC Development**	**George Englert**				**23%**
M4.7.1	Develop Measurement Uncertainty Recovery EC # 65627 *Notes* Separate EC ENG: AREVA / Esquilio Should Correspond to LAR Inputs and Include 50.59, Formal CR3 Quality Heat Balance Uncertainty Calculation, etc. No WR MUR Summary Report FSAR Markups TS Markups Marked-Up EDBD's EC 65566 Approved	George Englert	Virg Esquilio			50%
M4.7.2	Develop Caldon Spool Piece, Cabinet & Circuitry Installation EC # 6562 *Notes* Master EC ENG: AREVA / Hildebrandt This will authorize all installation activities in IB including FW Temp RTD & MS Pressure Transmitters for Caldon. Will also need TLR's or preinstalled lifting and rigging points. Multiple WR's (at least FW spool, RTD, Transmitter, Cabinet Installation, Cabinet Power, lifting/rigging points EC # 65626 Approved	George Englert	George Hildebrandt			50%
M4.7.3	Caldon Spec EC 65566 *Notes* Master EC ENG: AREVA / Hildebrant No WR SPEC only Approved EC # 65566	George Englert	George Hildebrandt			100%
M4.7.4	Inputs to FIDMS EC # 49623 *Notes* Separate EC ENG: PE - Fuels / NIT	George Englert				0%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
"Inputs to FIDMS EC # 49623" continued	*Notes* Will require on-line WR to complete FIDMS installation and Outage WR for changes due to MUR Approved EC # 49623					
M4.7.5	Changes to AULD EC # 65628 *Notes* Separate EC ENG: PE-Lord Will require upload to TRICONEX Software Project. Place holder WR during the Outage EC # 65628 Approved	George Englert	Mike Lord			0%
M4.7.6	MS Pressure & Temperature Thermocouple RTD EC # 65629 *Notes* Master EC ENG: AREVA / TBD Both are critical inputs to heat Balance Uncertainty that need to be replaced to improve uncertainty and ability to routinely calibrate, etc. 5 WR's 1 for Pre-Outage and 4 for installation of each device. EC # 65629 Approved	George Englert	Bob Wagner			50%
M4.7.7	ICS Scaling Function Curves EC # 65630 *Notes* EC # 65630 Approved	George Englert				0%
M4.7.8	RPS *Notes* Child of EC (Inputs to FIDMS) ENG: AREVA /	George Englert				0%
M4.7.9	SPDS *Notes* Child of EC (Inputs to FIDMS) ENG: AREVA /	George Englert				0%
M4.7.10	Caldon Software EC (F) ######## *Notes* Separate EC ENG: AREVA SQA of Caldon Software/Firmware to meet ENG-NGGC-0157 requirements. No WR	George Englert				0%
M4.7.11	Ethernet Hub EC (G) ##### *Notes* Separate EC ENG: NIT Will Support data transfer to FIDMS and AULD	George Englert				0%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M5	**Planning**	**Gene Flavors**				**8%**
M5.1	Location Walkdown	Ted Williams	Robin Kibler			100%
M5.2	**Work Package Planning**	**Gene Flavors**				**0%**
M5.2.3	**Pre-Outage Work Order Planning**	**Gene Flavors**				**0%**
M5.2.3.1	Plan WO xxxxxx for EC 65626 Pre-Outage Work LEFM *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.3.2	Plan WO xxxxxx for EC 65629 Pre-Outage Work MS Press & Temp RT *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.3.3	Plan WO xxxxxx for EC 49623 Pre-Outage Work FIDMS *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4	**Outage Work Order Planning**	**Gene Flavors**				**0%**
M5.2.4.1	Plan WO xxxxxx for EC 65626 Outage Work LEFM Installation *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4.2	Plan WO xxxxxx for EC 65629 Outage Work MS Temp & Press RTD Ini *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4.3	Plan WO xxxxxx for EC 49623 Outage Work FIDMS *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4.4	Plan WO xxxxxx for EC 65628 Outage Work AULD *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4.5	Plan WO xxxxxx for EC 65630 Outage Work ICS Scaling *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4.6	Plan WO xxxxxx for EC XXXXX Outage Work RPS *Notes* Planned WO	Gene Flavors	Robin Kibler			0%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M5.2.4.7	Plan WO xxxxxx for EC XXXXX Outage Work SPDS *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M5.2.4.6	Plan WO xxxxxx for EC XXXXX Outage Work Caldon Soft Ware *Notes* Planned WO	Gene Flavors	Robin Kibler			0%
M6	**Inplementation & Testing**	**Robin Kibler**				**0%**
M6.1	**Pre-Outage WO Implementation**	**Robin Kibler**				**0%**
M6.1.1	Implement WO xxxxxx for EC 65626 Pre-Outage Work LEFM *Notes* Scaffolding Installation Cable Pulls Cable Initialization and Testing Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS Scaffolding Removal Job Site Clean-Up Work Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.1.2	Implement WO xxxxxx for EC 65629 Pre-Outage Work MS Temp & Pre *Notes* Scaffolding Installation Cable Pulls Terminations Cable Initialization and Testing Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS Scaffolding Removal Job Site Clean-Up Work Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.1.3	Implement WO xxxxxx for EC 49623 Pre-Outage Work FIDMS *Notes* Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS Job Site Clean-Up Work Completion & Closure	Robin Kibler	Robin Kibler			0%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M6.2	**Outage WO Implementation**	**Robin Kibler**				**0%**
M6.2.12	Implement WO xxxxxx for EC 65626 Outage Work LEFM Notes Scaffolding Installation Pipe Cut Flow Element Welding Cable Pulls Terminations Cable Initialization and Testing Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS Scaffolding Removal Job Site Clean-Up WO Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.2.13	Implement WO xxxxxx for EC 65629 Outage Work MS Temp & Press R Notes Scaffolding Installation Pipe Cut Flow Element Welding Cable Pulls Terminations Cable Initialization and Testing Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS Scaffolding Removal Job Site Clean-Up WO Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.2.14	Implement WO xxxxxx for EC 49623 Outage Work FIDMS Notes Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS WO Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.2.15	Implement WO xxxxxx for EC 65628 Outage Work AULD Notes Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS WO Completion & Closure	Robin Kibler	Robin Kibler			0%

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MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M6.2.16	Implement WO xxxxxx for EC 65630 Outage Work ICS Scaling	Robin Kibler	Robin Kibler			0%
M6.2.17	Implement WO xxxxxx for EC XXXXX Outage Work RPS *Notes* Field Calibrations Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS WO Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.2.18	Implement WO xxxxxx for EC XXXXX Outage Work SPDS *Notes* Field Calibrations Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS WO Completion & Closure	Robin Kibler	Robin Kibler			0%
M6.2.19	Implement WO xxxxxx for EC XXXXX Outage Work Caldon Soft Ware *Notes* Software Update Modification Testing Pri Zero Affected Document List Distribution EC RTS WO Completion & Closure	Robin Kibler	Robin Kibler			0%
M7	**Project Closure**					**0%**
M7.1	Work Order Closure *Notes* All MUR Project WO's Closed & Vaulted	Ted Williams	Robin Kibler			0%
M7.2	Pri One Affected Document List Distribution *Notes* All Pri One Affected Documents Distributed	Ted Williams	Ted Williams			0%
M7.3	EC Closure *Notes* All MUR Project EC's Closed	Ted Williams	George Englert			0%
M7.4	Final Invoicing and Payment *Notes* All MUR Project Invoices Recieved & Paid	Ted Williams	Ted Williams			0%

Appendix A

MUR WBS Dictionary as of Fri 2/16/07
MEASUREMENT UNCERTAINTY RECAPTURE WBS

WBS	Name	Functional Area	Task Manager	Project Cost Code	WO Numbe	PC
M7.5	Project Critique *Notes* MUR Project Critiques Cataloged	Ted Williams	Ted Williams			0%
M7.6	Lessons Learned Issued	Ted Williams	Ted Williams			0%

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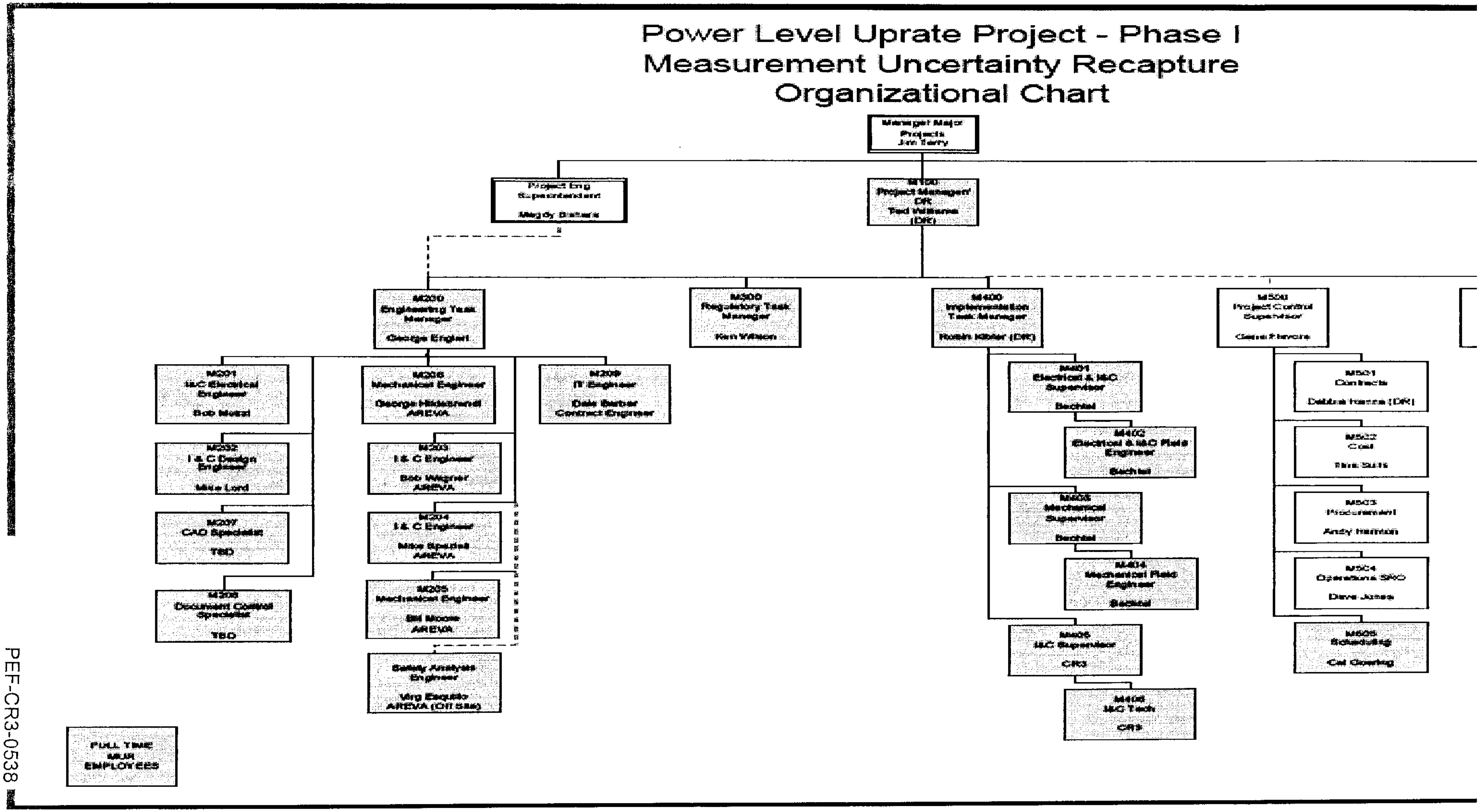
Appendix B
Organization Structure



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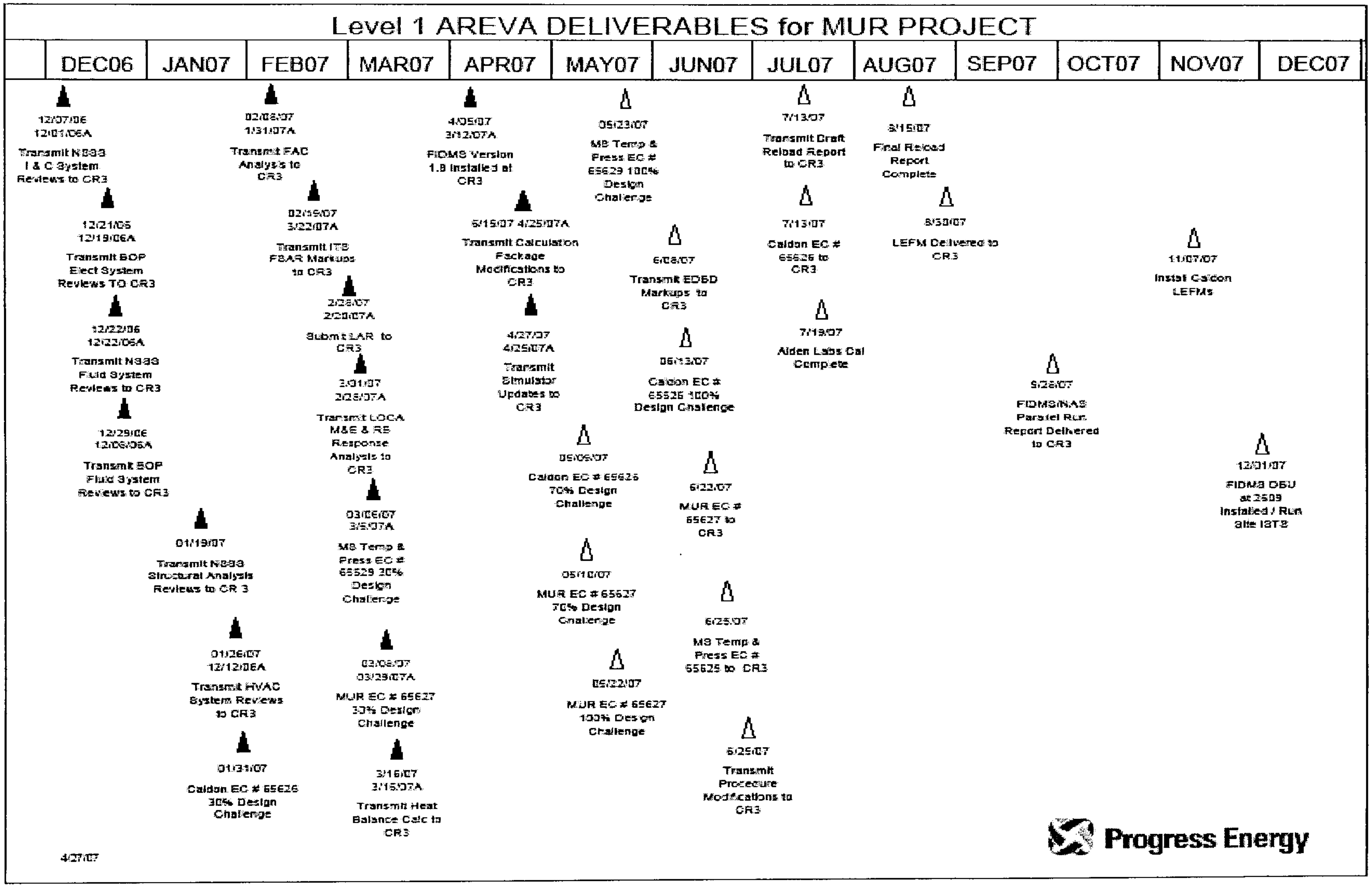
Appendix C
MUR Uprate Schedule



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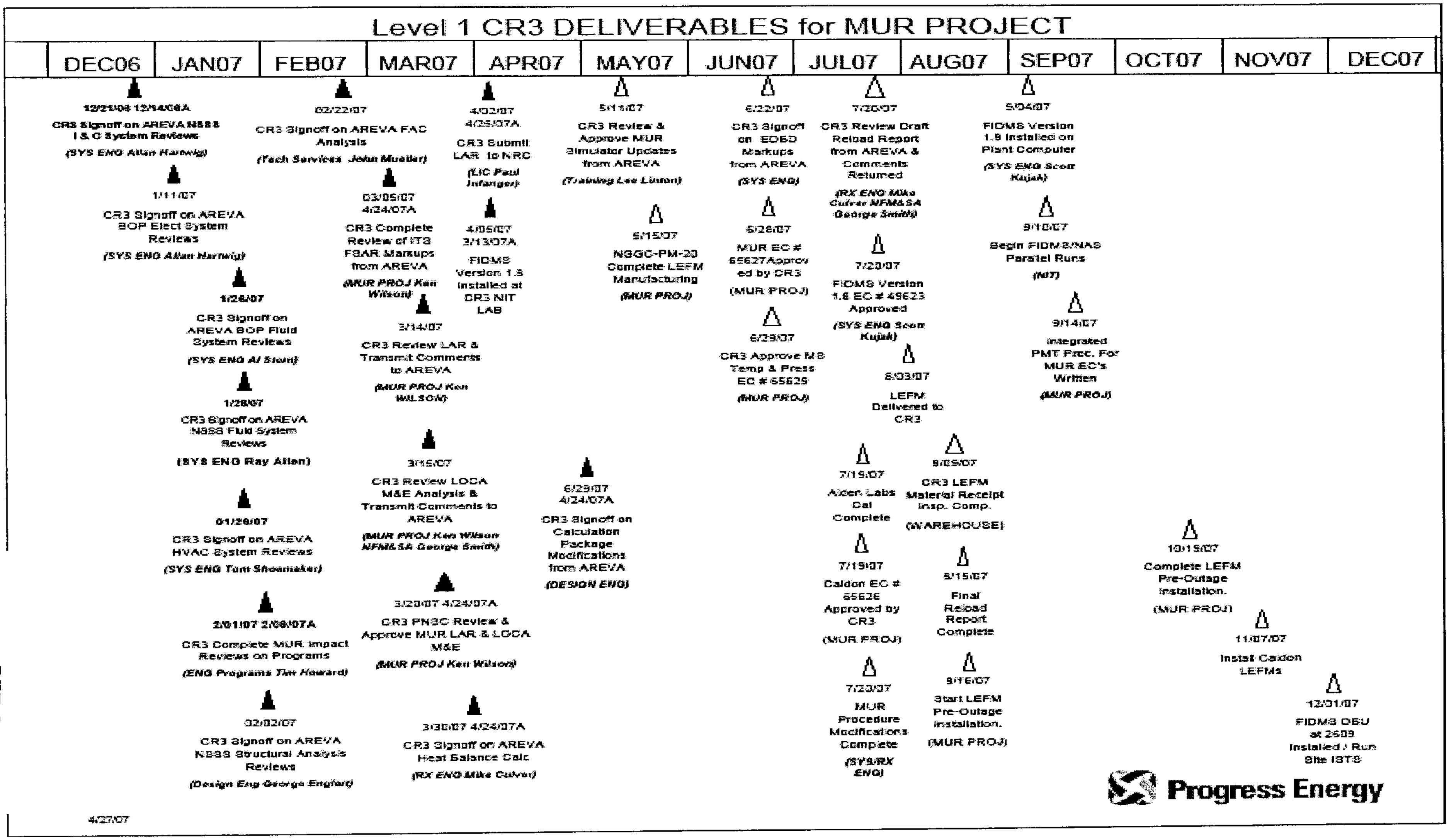
Appendix C
MUR Uprate Schedule



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Appendix D

Operating Experience

REACTOR OVERPOWER EVENTS ASSOCIATED WITH ULTRASONIC FEEDWATER FLOW MEASUREMENT SYSTEMS

Limited Distribution June 18, 2004

Executive Summary

Errors in core power calculations based on data provided by ultrasonic flow measurement instruments resulted in extended operation above licensed power limits at Byron Station and River Bend Station in 2003, resulting in significant events. Between 2000 and 2003, the industry experienced 14 events involving reactor overpower or potential overpower conditions attributed to the use of ultrasonic flow measurement systems. The overpower conditions resulted in a reduction to safety margins. The major contributors to these events were over reliance on vendor expertise, lack of a questioning attitude by station personnel, and inadequate verification testing.

Significant aspects of these events include the following:

- Byron Unit 1 operated at power levels in excess of 100 percent rated thermal power for over three years.

- Several opportunities were missed to identify and resolve the issue of operation at excess power levels over the three-year period.

- River Bend operated above 100 percent power for nearly seven years. Reactor power exceeded the analyzed limit of 102 percent power for approximately 15 months over a two-year period.

- Plant personnel were unaware that the reactor was operating at these power levels, in part because independent methods of correlating reactor power with other power-dependent plant parameters were not rigorously performed.

NOTE: Training materials developed by Exelon Nuclear containing a case study lesson plan and slide presentation are included with this document. These materials discuss the events at Byron and may be useful to other stations to communicate operating experience.

Background

Ultrasonic flow measurement (UFM) systems are used for reactor power level calculations or for periodic calibration of feedwater flow venturis used in power level calculations. The systems can be operated on line to provide continuous input to calorimetric power calculations or intermittently to calibrate or adjust existing feedwater flow instruments. The systems are used at both boiling water reactors and pressurized water reactors and have resulted in overpower events at both types of reactors.

Although UFM technology has been in use for many years, few events were reported between 1991 and 1999. An increasing trend in the number of overpower events related to UFM began in 2000, with 14 events occurring between 2000 and 2003. This resulted primarily because UFM systems allowed operation closer to design limits for core thermal power.

Overpower Events

Byron and Braidwood Stations – August 2003

In August 2003, Byron and Braidwood stations reported exceeding their licensed maximum power levels because of inaccuracies in the ultrasonic flow measurement system. The inaccuracies were attributed to signal noise caused by pressure pulses in the individual feedwater piping to each steam generator. The inaccuracies had existed since the systems were placed in service.

Byron Experience

Byron Station units 1 and 2 installed the UFM system (UFMS) in May 1999. During testing, measurements of the venturi correction factors indicated an unexpected difference between Braidwood Station Unit 1 and Byron Station Unit 1 feedwater venturi flow rates and the venturi correction factors. This resulted in differences in the electrical output between the two identical units, with Byron Unit 1 producing approximately 15 MWe more than Braidwood Unit 1 at 100 percent power.

An evaluation conducted to determine the difference between Byron Unit 1 and Braidwood Unit 1 was inconclusive. The evaluation did verify the UFMS was installed correctly and was operating within the established design criteria. Consequently, a decision was made to implement feedwater venturi correction factors using the UFMS in May 2000.

Over the next several months, additional internal and external evaluations were conducted in an attempt to understand the discrepancy between the Byron and Braidwood units. The results of these investigations once again concluded that the UFMS was operating in accordance with criteria established by the vendor.

A broader test plan was developed in early 2003 to continue the investigation. This included operating the UFMS in a continuous measuring mode instead of the intermittent (periodic) mode to improve trending of system performance. In May 2003, a flow comparison test was conducted at Braidwood Unit 1, which compared the feedwater header flow in the common feedwater header with the sum of the four individual feedwater branch lines. The results of this test were within the expected criteria.

An identical test was performed at Byron Unit 1 in August 2003. The difference between the sum of the UFMS measurements in the four individual feedwater branch lines and the common feedwater header was outside the acceptance criteria. Based on this, Byron Station reduced power and returned the venturi correction factors to 1.0 on both units, pending resolution of the issue.

Investigation by site and vendor personnel determined that signal noise in the UFMS was affecting the flow velocity calculations, which in turn affected the venturi correction factors. With noise-contaminated correction factors uploaded into the calorimetric calculation, a non-conservative power measurement resulted. The overpower condition potentially existed since the initial implementation of the UFMS in May 2000.

In 2004, a flow measurement test using a radioactive tracer was performed on the Byron Unit 1 and Unit 2 main feedwater systems to help diagnose a configuration and alignment sensitivity issue with a new UFMS installation on the common feedwater header. The test validated the accuracy of the installed venturi flow measurement system. The tracer test confirmed there was a non-conservative bias in the UFMS installed on the common main feedwater header.

The estimated worst-case overpower conditions for Byron units 1 and 2 reported in September 2003 were based on the premise that the UFMS installed on the common main feedwater header was providing accurate flow measurements. However, based on the subsequent tracer tests validating the accuracy of the venturi flow measurement, the estimated worst-case overpower based on the maximum UFMS correction factor applied to the venturi flow measurement could have been as high as 102.62 percent for Unit 1 and 101.88 percent for Unit 2.

Braidwood Experience

Braidwood Station units 1 and 2 installed the UFMS in May 1999. Braidwood implemented the venturi correction factors using UFMS in June 1999. The originally reported worst-case overpower condition on Braidwood Unit 2 in September 2003 was based on the premise that the UFMS on the common main feedwater header was providing accurate flow measurements. However, based on the results of the tracer test on Byron units 1 and 2 validating the accuracy of the venturi flow measurement at Byron, the estimated worst-case overpower was based on the maximum UFMS correction factor applied to the venturi flow measurement. It was assumed that similar results would be obtained on both of the Braidwood units, resulting in estimated worst-case overpower conditions that could be as high as 101.07 percent for Braidwood Unit 1 and 101.21 percent for Braidwood Unit 2.

Evaluations starting in April 2003, including continuous monitoring of the UFMS and frequency spectrum analysis, indicated that the Byron and Braidwood overpower events were caused by noise contamination of the UFMS ultrasonic signal for the meters installed on the individual feedwater lines. Feedwater flow pressure pulses occurred at frequencies that affected the UFMS signal and resulted in a bias in the determination of the venturi flow correction factors. This noise caused the UFMS to indicate lower than actual feedwater flow, which resulted in a non-conservative calorimetric calculation of reactor thermal power.

River Bend Station – May 2003

On May 10, 2003, reactor feedwater flow instrumentation used to calculate reactor power was found to be providing nonconservative data, resulting in the reactor being operated above its licensed thermal power limit. At the time this condition existed, River Bend was licensed at 3,039 megawatts thermal

(MWth) and was analyzed at 102 percent of licensed thermal power. The primary factor affecting the reactor power calculation is feedwater flow. Prior to 1996, reactor feedwater flow was measured using a calibrated venturi in each of the two feedwater headers.

In February 1996, a leading edge flow meter (LEFM) was installed to correct the feedwater flow venturi data in the plant process computer. The LEFM data was used to eliminate reactor heat balance conservatism caused by built-in allowances to account for venturi fouling. The LEFM data would result in maximizing plant efficiency and electrical output.

The first venturi correction factors were implemented in late February 1996, resulting in an increased core power thermal power of approximately 1.5 percent of rated thermal power. The correction factors were checked monthly to compare feedwater venturi flow indication and LEFM flow indication. New correction factors were calculated and installed if the difference between venturi flow and LEFM flow reached a predetermined limit or each time the plant was shut down and restarted.

During the March 2003 refueling, a more accurate LEFM unit was installed to replace the externally mounted LEFM. Because of the increased accuracy of the new LEFM, the station expected to obtain an additional 1.7 percent increase in core thermal power. During the subsequent startup and power ascent, the indicated feed flow data from the new LEFM was reviewed against feedwater venturi data. Comparison of these flows with the externally mounted LEFM data indicated that reactor power had exceeded the licensed power limit by a maximum of 2.7 percent.

A detailed analysis was performed to determine the magnitude of the overpower condition. From the installation of the externally mounted LEFM in February 1996 until the high-pressure (HP) main turbine rotor replacement in 1999, there was no indication that the 102 percent accident analysis limit was exceeded. Estimates for the period between the HP turbine rotor replacement in 1999 and 2003 indicated the following:

- Between the HP turbine rotor replacement in 1999 and the 5 percent power uprate in October 2000, the licensed power limit was exceeded. However, there was no indication that the 102 percent accident analysis limit was exceeded.

- The period between October 2000, when new correction factors were installed following the 5 percent uprate, and April 2001, when new correction factors were installed following a forced outage, represented the bounding case for the overpower condition. During this time, reactor power level was apparently as high as 102.7 percent.

- From April 2001, when correction factors were changed following a forced outage, to the October 2001 refueling outage, reactor power was as high as 102.5 percent.

- Following the 2001 refueling outage, the 102 percent analyzed limit was not exceeded again until January 2002, when a scheduled power reduction for control rod sequence exchange and turbine valve testing occurred. This condition lasted until May 2002, when correction factors were revised following a planned outage. The plant operated above 100 percent power until January 2003 and the power coastdown leading to the spring 2003 refueling outage.

Based on this analysis, the reactor was operated in excess of the 102 percent accident analysis limit for approximately 15 months.

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St. Lucie Nuclear Power Plant Unit 2 – October 2000

On September 25, 2000, St. Lucie Unit 2 operated at an average power level greater than 100 percent for more than 8 hours. This resulted from an instrumentation failure on the leading edge flow meter (LEFM) used by the digital data processing system (DDPS) to determine calorimetric power level. A transducer failure in the A channel of the LEFM instrument resulted in a decreasing feedwater flow value.

Detailed analysis of plant performance for September 24-25, 2000 indicated an unexpected decrease in the channel A feedwater flow input to the DDPS, without a corresponding decrease in a venturi-based feedwater flow. A direct and corresponding decrease in DDPS calorimetric power was noted from the decreased output of channel A, without a corresponding decrease in the output of channel B or the feedwater venturi.

The control room crew questioned this anomaly and initiated corrective action based on the observation that DDPS calorimetric power level was decreasing. During the LEFM instrument failure, the calorimetric power level indication used by the operating crew did not exceed 100.0 percent for more than eight hours. The LEFM transducer was repaired and returned to service on October 1.

Based on subsequent analysis of channel B LEFM flow and the A and B channel venturi flow, the change in reactor power was approximately 0.2 percent power. The maximum power level was approximately 99.9 percent on September 24 and approximately 100.1 percent, with a maximum of 100.2 percent, on September 25. An increase to 100.2 percent power was well within the uncertainty for DDPS calorimetric power of 1.3 percent and the initial power level assumed in the plant safety analysis of 2 percent.

Disassembly of the LEFM transducer assemblies revealed significant cracking in the non-metallic wedges on the surfaces that contact the feedwater pipe. Cracking in the wedges alters the sound wave that changes the transit time from the upstream to the downstream transducer, and vice versa. Wedges were replaced in several locations by the LEFM original equipment manufacturer and station personnel.

The root causes of the wedge cracking may be attributed to three effects:

- time in service (more than six years for all four A pipe wedges and more than four years for one of the B pipe wedges)
- high stress placed on the wedge by the presently installed mounting fixture and the zinc pad method of coupling the wedge to the pipe
- harsh environment of 430°F pipe surface temperature

Salem/Hope Creek Generating Station – June 2002

On June 28, 2002 a detailed analysis determined that ultrasonic flow detection equipment used to detect feedwater flow was malfunctioning because of calcium silicate insulation lodged between the clamp and pipe. The malfunction of the cross-correlation instrumentation correction factor resulted in the reactor exceeding its licensed power level by an average of 0.25 percent for eight hours.

PEF-CR3-0546

On July 30, 2001 Hope Creek received approval from the NRC to perform a 1.4 percent power uprate. The approval was partially based on the installation of a cross-correlation ultrasonic flow measurement system and its ability to achieve increased accuracy in measuring feedwater flow.

On May 26, 2002, operators identified that the feedwater cross-correlation correction factor was lower after returning to normal operations from a downpower evolution. Balance-of-plant parameter anomalies led the operators to suspect that indicated reactor power, with the correction factor applied, was nonconservative. Detailed analyses confirmed that a malfunction of the cross-correlation instrumentation correction factor caused the plant to operate above its licensed power level by 0.25 percent for eight hours.

The event was attributed to cracked insulation becoming lodged between the cross-correlation instrument clamp and the feedwater line. Immediate corrective actions included removing the cross-correlation instrument from service and reducing power below the 1.4 percent uprate value. Additional corrective actions included the installation of new cross-correlation transducers and validation of existing cross-correlation performance. The mounting configuration was also changed.

Analysis

The trend in actual reactor overpower events and potential overpower events has been increasing since 2000, and this has resulted in reduced margins to core thermal power limits. Fourteen events involving issues with feedwater flow measurements were recently analyzed by INPO and published in Topical Report TR4-34, "Review of Feedwater System Ultrasonic Flowmeter Problems," in March 2004. The following key conclusions were reached:

- Reactor power indication was directly affected in 10 of the 14 events, with seven units exceeding their 100 percent licensed thermal power limits and one BWR conservatively estimated to have operated in excess of the 102 percent thermal power limit used in the accident analyses.
- Problems were experienced with both cross-correlation flow instrumentation and transit time (leading edge) instrumentation. Four units using cross-correlation instruments exceeded their 100 percent licensed power limits. Three units using transit time instrumentation also exceeded their licensed power limits.
- Nine of the 14 events involved personnel error. Seven of 9 events involved performance deficiencies by vendors.
- Overreliance on vendor expertise, lack of a questioning attitude by station personnel, and inadequate acceptance testing were major contributors to many of the events.

Lessons Learned

The following lessons learned are excerpted from root cause investigations provided by stations experiencing long-term undetected overpower conditions caused by problems with ultrasonic feedwater flow measurement systems.

Signal Noise

- UFMS sensor signals should be checked for noise before the system is placed in service.

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- On-line UFM systems should be checked for noise following major changes in the feedwater system, such as shifting main feedwater pumps, or changes in feedwater heater operation.

- Vendor diagnostic software programs should allow for station personnel to evaluate sensor noise.

Transducer Problems

- Externally-mounted leading edge flow meters can be affected by changes in velocity profiles that are greater than the calibration basis of the flow meter.

- Incorrect feedwater piping measurements and corrosion can cause changes in the effective acoustic path of the transducer, resulting in signals being outside of the desired range.

- Thermal cycling of the feedwater piping caused by more frequent plant startups and shutdowns or by large power changes can affect transducer spacing. Transducer spacing also can be affected by loose mounting brackets.

- Leading edge flow meters installed in welded spool pieces can be affected by flow bypass.

Operational Considerations

- Small changes in power, such as the installation of new turbine rotors, new HP turbine inlet nozzles, and flow or pressure uprates in BWRs, can mask changes in data being provided by the UFMS.

- Without an independent, highly accurate means of verifying feedwater flow after UFMS installation, the plant relies solely on the vendor or an outside source to verify that the UFMS data is correct. Often, this has resulted in only determining that the UFMS continues to meet the uncertainty values specified for the system, without resolving the cause of questionable readings.

- Other nuclear steam supply system and balance-of-plant power-dependent parameters can be used to make correlations to core thermal power. For example, turbine first-stage pressure can be used with UFMS to corroborate performance.

- Heat balance calculation (calorimetric) procedures should provide criteria for ensuring feedwater flow venturi correction factors were reasonable. Maximum power capability tests also should contain criteria for comparing calculated core thermal power with other power-dependent parameters.

- Critical parameters that provide redundant methods for verifying reactor power may be limited because they are single-channel instruments and their accuracy can only be verified by an outside source.

- A computer point using multiple power-dependent parameters can provide a statistical best estimate of core thermal power for comparison with heat balance calculations based on UFMS signals.

- Operations personnel need to question and seek resolutions to anomalous or unexpected indications in plant operating parameters that may indicate unrecognized inaccuracies in ultrasonic flow measurement systems.

Management Considerations

- Roles and responsibilities for UFMS installation and testing should be well defined and coordinated among various organizations. Station personnel should be well trained and closely involved through ownership and accountability.

- Evaluations of complex technical issues with conflicting indications should include the application of diverse testing and analysis methodologies to confirm the extent and nature of the condition.

- Efforts to address technical issues should include a focus on maintaining the plant in a known safe state until the uncertainties are eliminated.

- Oversight groups and problem reporting and resolution processes should provide additional barriers for identifying and addressing anomalous conditions.

- Utility oversight and questioning of vendor information and designs should be rigorous to avoid being solely reliant on vendors for verification of critical data.

References

1. INPO Events Database Report 454-030828-1, "Licensed Maximum Power Level Exceeded Due to Inaccuracies in Feedwater Ultrasonic Flow Measurements Caused by Signal Noise Contamination," August 2003.

2. INPO Events Database Report 412-010710-1, "Secondary Calorimetric Inaccuracy Resulting from Main Feedwater Mass Flowrate Inconsistencies," July 2001.

3. NRC Morning Report 37403, "Discovery That Reactor Power Exceeded 100% for a Period of Greater Than 8 Hours Due to an Instrument Failure on the Leading Edge Flow Meter," October 2000.

4. INPO Events Database Report 458-030510-1 "Licensed Thermal Power Limit Was Exceeded Due To An Error In Feedwater Flow Measurement," May 2003.

5. INPO Events Database Report 354-020628-1, "Potential to Exceed Licensed Power Level Due to Malfunction of The Crossflow Correction Factor Instrumentation," June 2002.

6. INPO Trend Report TR4-34, "Review of Feedwater System Ultrasonic Flowmeter Problems," March 2004.

Utilities are requested to provide feedback on similar occurrences and solutions at their plants or on their equipment to the information contact listed below.

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PREVENT EVENTS

Learning from Industry Experience

PREVENT EVENTS is intended for use by personnel during morning meetings, prejob briefings, and work unit meetings to communicate key industry experience.

Between 2000 and 2003, the industry experienced 14 events involving reactor overpower or potential overpower conditions attributed to the use of ultrasonic flow measurement systems. For example, overpower events occurred in August 2003 at Exelon Nuclear Corporation's Byron and Braidwood stations and resulted in a significant event at Byron Unit 1. In addition, River Bend Station operated for an extended period above 102 percent power, resulting in a significant event. These events and others are discussed in INPO Significant Event Report 3-04, "Reactor Overpower Events Associated with Ultrasonic Feedwater Flow Measurement Systems," June 18, 2004.

Engineering Supervisors

1. What questions have been asked regarding UFMS vendor assumptions associated with your plant-specific ultrasonic feedwater flow measurement system?
2. How have you evaluated the UFMS software codes proposed or installed by the vendor using a software quality assurance program?
3. What validation and verification have been completed on UFMS software that will interface with the plant process computer?
4. What members of your organization have reviewed the contents and recommendations of EPRI TR-112118, "Nuclear Feedwater Flow Measurement Application Guide"?
5. How will you verify UFMS operability following mechanical repairs to housings, spool pieces, and other hardware?
6. How will you evaluate the effects on UFMS transducer coupling following leak repairs, waterhammer events, or thermal cycling?
7. How have you evaluated the effects of changes in ambient temperature near UFMS cabinets?
8. How have you evaluated the performance of the UFMS under different feedwater system configurations?
9. What testing have you performed using independent or more accurate measurement techniques to validate UFMS accuracy?

Training Managers

1. What training has been provided to station engineering, operations, and maintenance personnel on the UFMS?

Management

1. How have you verified that roles and responsibilities associated with UFMS installation are defined and understood by all organizations involved? How have you defined ownership and accountability for the project?

2. What diverse testing methods will be used to verify the accuracy and reliability of the UFMS? How will unresolved technical questions be handled?

3. What oversight groups will be used to provide additional barriers in identifying and resolving issues that may arise during and after UFMS installation?

4. What oversight will be provided to evaluate vendor data analysis?

Attachments

Training materials developed by Exelon Nuclear containing a case study lesson plan and slide presentation are included with this document. These materials discuss the events at Byron and may be useful to other stations to communicate operating experience.

Case Study Lesson Plan

Slide Presentation

Large Project Implementation Strategies – A Leadership Perspective of Power Uprates

Institute of Nuclear Power Operations

January 2005

INPO 05-001
(Preliminary)

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PEF-CR3-0553

Introduction

This document provides implementation insights and practical examples of how stations have achieved excellence during development and installation of large-scale projects such as power uprates. Examples used in the document were identified during benchmarking activities, INPO plant evaluations, and industry meetings from 2002 to the present. An industry working group of power uprate project managers and engineering leaders participated in the document's development and ensured the conclusions balanced high standards with practical achievability. While the document focuses on power uprates, the success factors could be applied to any large project.

Striving for excellence in implementing power uprates is an ongoing industry effort. In addition to this document, INPO endeavors in this area include the following:

- In August 2004, a power uprate working meeting was held that focused on management of ultrasonic feedwater flow instrumentation concerns and extended power uprate lessons learned. The presenters and participants reviewed and discussed the organizational factors that contributed to events associated with power uprates. Techniques and compensatory measures to mitigate power uprate events were addressed and captured into this document.
- In June 2004, INPO Significant Event Report 3-04, "Reactor Overpower Events Associated with Ultrasonic Feedwater Flow Measurement Systems," was issued to highlight errors in core power calculations based on data from the ultrasonic feedwater flow instruments. Overreliance on vendor expertise, lack of a questioning attitude by station personnel, insufficient training, and inadequate acceptance testing were contributors to many of the events.
- In September 2003, a power uprate working meeting was held for project managers and engineering leaders to discuss their collective experiences, concerns, and best practices. Another purpose of the meeting was to collect information to incorporate into this document that highlights the key attributes that enable power uprate projects to be implemented successfully.
- In August 2002, INPO Significant Event Report 5-02, "Lessons Learned from Power Uprates," was issued. The report discussed more than 40 events over the previous five years. Those events resulted from inadequate analysis, design, or implementation of plant power uprates. Many of the events involved damage to equipment, unanticipated responses to plant conditions, or challenges for operating staff. The number and types of events indicate that more significant consequences could result from future events if power uprates are not carefully controlled.

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- In June 2002, a power uprate working meeting convened that focused on issues of stretch and extended power uprates on both PWRs and BWRs. Some of the most often cited lessons learned from power uprate project management and implementation included an emphasis on organizational communication, involvement of the operations and training staffs, and the need to perform benchmarking of other sites.

The body of this report is written for project managers and senior station managers and focuses on power uprate implementation. As such, these characteristics may be useful for self-assessment activities. In addition, appendixes for each of the seven success factors provide examples of methods stations have used to achieve success in those areas.

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Executive Summary

This document summarizes common success factors observed at U.S. nuclear plants that implemented power uprates. Key characteristics of each success factor are described. In addition, appendixes provide examples of methods stations have used to achieve success in these areas.

The seven common success factors and their characteristics are as follows:

Success Factor 1 ***Involve Site and Corporate Management***

- Senior management clearly communicates throughout the organization the goals, priorities, and expectations for the power uprate activity, emphasizing its importance to nuclear safety, plant economic viability, and personnel quality of life.

Success Factor 2 ***Manage Projects Effectively***

- The station uses the fundamentals of project management, with tools for scope control, phased work approach, schedule, risk management, and budget, to plan and implement the power uprate.

Success Factor 3 ***Assess Actual Plant Conditions***

- The as-built design margins and performance capabilities of the plant are well documented and understood and are used in power uprate project planning.

Success Factor 4 ***Focus on Operating Experience***

- Station efforts are focused on using operating experience and industry expertise to review power uprate planning and progress at key project milestones.

Success Factor 5 ***Implement a Post-Uprate Testing and Monitoring Plan***

- Efforts are made to predict how plant performance can change as a result of power uprate and defining which key parameters are tested and trended. When actual performance deviates from predictions, the station aggressively seeks to understand the differences.

Success Factor 6 ***Anticipate Design Shortfalls***

- The project predicts and plans for the contingency of reconstituting some aspects of the original plant design.

Success Factor 7 ***Promote Effective Engineering Human Performance***

- Station leaders ensure a robust engineering human performance culture is present to promote and expect superior engineering product quality.

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Success Factor 1 — *Involve Site and Corporate Management*

***Characteristic*: Senior management clearly communicates throughout the organization the goals, priorities, and expectations for the power uprate activity, emphasizing its importance to nuclear safety, equipment reliability, and plant economic viability.**

- The site leadership team supports the project by early assignment of high-performing and experienced personnel who are dedicated full time to the project. Team turnover and attrition are closely monitored and compensatory adjustments made. A senior management sponsor is kept actively engaged in the power uprate project.
- Well-defined roles and responsibilities for the project team members are established.
- Senior management makes key inputs and decisions for the power uprate project and communicates those decisions to the site leadership team.
- The site leadership team reviews the aggregate uprate work scope, to better allocate the necessary resources.
- The project team and station management work together to address risks and make risk-based decisions.
- The site leadership team focuses on design and operating margins and keeps open communications on margin concerns.
- An implementation plan is developed and is approved by the site leadership team.
- The site leadership team receives routine reports on the project status, updates, and areas requiring management attention to ensure success of the project.
- Periodic management reviews and challenge boards are conducted with the project team.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 1.

Success Factor 2 — *Manage Projects Effectively*

***Characteristic:* The station uses the fundamentals of project management with tools for scope control, phased work approach, schedule, risk management, and budget, to plan and implement the power uprate.**

- The project team secures site leadership team approval and priority early in the planning phases. Feasibility studies are conducted prior to project plan development.
- Project goals and priorities are clearly communicated and may include a combination of gaining megawatts, maintaining or increasing margins, addressing equipment obsolescence, or planning for plant life extension. The project team and the site leadership team partner in communicating the project goals and objectives to the station staff. Training and change management plans are developed to ensure site understanding of the project scope and objectives.
- Clear roles and responsibilities for project team members and site staff supporting the power uprate project are developed that reflect ownership and accountability. Clear agreements with the vendors on expected inputs, outputs, and deliverables are established.
- A detailed scope, including expectations and a clear understanding of site procedures and processes, is well defined when contractual agreements are entered into with vendors. Added rigor to the purchasing process in the form of detailed purchase specifications reduces quality issues.
- A comprehensive risk matrix is developed to enable management to see all of the project technical and commercial risks. Management reviews the matrix on a recurring basis.
- Scope change occurs after a documented, risk-informed decision process. The project team develops a culture of resisting scope reductions with the intent of meeting deadlines. An issue resolution process is developed to resolve emergent technical and commercial issues quickly.
- A resource-loaded schedule is developed to establish confidence that appropriate resources are being assigned to the project. Critical milestones and decision points are established to avoid undue time pressures and to facilitate operational decision-making. The site leadership team approves the milestones and commits needed resources.
- A transition plan is developed for ending the project and turning the completed power uprate project over to the site.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 2.

Success Factor 3 — *Assess Actual Plant Conditions*

Characteristic: **The as-built design margins and performance capabilities of the plant are well documented and understood and are used in the power uprate project planning.**

- Design and operating margins are well managed and understood. Actual operating and design margins may differ from original margins as the plant ages.
- The site leadership team is engaged with decisions to accept and manage lower margins and directs efforts to restore margins.
- Actual plant performance data is used in power uprate planning. For example, current thermal performance trends are used in the power uprate planning, rather than relying on design information for the balance-of- plant systems.
- System engineering and program engineering interviews are conducted to validate current plant performance data. After the conceptual design phase is complete, the assumptions and inputs are revalidated with the system and program engineers.
- Known equipment reliability shortfalls are rigorously identified and fixed prior to implementation of a power uprate. Experience from several extended power uprates (EPUs) indicates that implementing a power uprate may unmask lower-level equipment problems and make them more significant.
- Plant personnel search for latent issues through a structured method to identify vulnerabilities.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 3.

Success Factor 4

Focus on Operating Experience

Characteristic: **Station efforts are focused on using operating experience and industry expertise to review power uprate planning and progress at key project milestones.**

- Plants that previously implemented a power uprate are benchmarked. The lessons learned are factored into the station's plan for each phase of the power uprate project. The following link to the INPO Power Uprate page provides several sources of benchmarking information. One of them, the NRC Power Uprate Web page, lists the plants that have done the various types of uprate: http://www.inpo.org/operatingexperience/pwruprate/main.asp
- Peer assessments are conducted at various stages of the uprate project.
- Power uprate experiences from fleet, peer, and sister plants are collected and analyzed.
- Maintenance work order history of the equipment most influenced by the power uprate is reviewed to better understand its current condition and health.
- Previous Requests for Additional Information (RAIs) from other licensing submittals are reviewed to determine if the same issues require specific consideration in the submittal report.
- The effects of generic industry regulatory issues on the uprate submittal and the activities of EPRI and the Nuclear Energy Institute are considered.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 4.

Success Factor 5

Implement a Post-Uprate Testing and Monitoring Plan

Characteristic: **Efforts are made to predict how plant performance can change as a result of power uprate and defining which key parameters are trended. When actual performance deviates from predictions, the station aggressively seeks to understand the differences.**

- New risk factors are predicted and monitoring plans are developed.
- Pre-uprate data collection and baseline equipment conditions are factored into the project schedule. In some cases, this may involve planning inspections two years prior to the actual uprate implementation.
- Monitoring plans place special emphasis on collection and analysis of vibration data, both before and after power uprate.
- Augmented thermography monitoring may be necessary in the switchyard and in some generator support systems.
- Some modification work, well prior to the power uprate implementation date, may be necessary to install monitoring instrumentation such as test taps or revenue grade metering to monitor MWe output.
- Consideration is given to monitoring for higher area temperatures and to how the increased temperatures affect equipment.
- Ambient noise levels that may affect industrial safety postings are monitored.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 5.

Success Factor 6 — *Anticipate Design Shortfalls*

Characteristic: **The project predicts and plans for the contingency of reconstituting some aspects of the original plant design.**

- The effects of being unable to locate an existing basis for the plant design or the impact of creating a new design basis is anticipated, and the team is prepared to address any such issues.
- The project team develops strategies of margin management and margin recovery in dealing with design shortfalls.
- Changes in operating strategies and the effects on both normal and off- normal operating procedures are understood.
- Operations personnel are provided frequent updates of the project plan and the margin management strategies. Feedback is solicited from operations to confirm that planned actions are acceptable. Simulator updates and operator training are completed prior to project implementation.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 6.

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Success Factor 7 — *Promote Effective Engineering Human Performance*

Characteristic: **Station leaders ensure a robust engineering human performance culture is present to promote and expect superior engineering product quality.**

- Engineering leaders ensure that those assigned to perform reviews and provide technical oversight possess the right technical expertise to challenge the final product. Training is used as a means to enhance the performance of those who oversee supplemental personnel.
- Design inputs are validated to ensure they are appropriate and current for the application.
- Station personnel review all vendor products to ensure assumptions, design inputs, and actual plant operating history have been used appropriately. In some cases, generic vendor products may not apply to all plants. In cases where the station staff lacks sufficient technical expertise, third-party reviewers are used. The team ensures effective vendor oversight and source surveillances are conducted.
- The station guards against an overreliance on and overconfidence in the vendor for resolving technical issues. When complex technologies are involved, the use of third-party experts is pursued. Gaps in expertise are identified, and outside resources are used as needed. Training is used where necessary to ensure the station has the skills for sustainable performance after the power uprate.
- Human performance error-reduction tools and differing professional opinion processes are used by supplemental personnel.

Examples of methods stations have used to achieve success in these areas are provided in Appendix 7.

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Appendix 1

Involve Site and Corporate Management

Characteristic: **Senior management clearly communicates throughout the organization the goals, priorities, and expectations for the power uprate activity, emphasizing its importance to nuclear safety, plant economic viability, and personnel quality of life.** Examples are as follows:

1) The Calvert Cliffs Nuclear Power Plant measurement uncertainty power uprate was made part of the site business plan. Responsibilities were defined, individuals named, and due dates specified in the plan. This high-level focus made the project's priority more visible to the other managers and stakeholders. The risks and benefits of the project were discussed at all management levels, and the concerns were rolled up as part of the communications plan.
2) The Palisades Nuclear Plant site management team strongly supported the instrument uncertainty power uprate by naming a dedicated power uprate champion and by treating the uprate as a major facility change. The champion coordinated and drove all efforts to develop, review, approve, and implement the power uprate safely and successfully. The project was staffed with very experienced personnel and had strong corporate management support.
3) Entergy Nuclear power uprate projects developed and communicated a comprehensive list of management goals that included the purpose of the uprate modifications. The goals included gains in megawatts, replacement of obsolete equipment, improvement of margins, and better positioning of the plant for life extension.
4) The Tennessee Valley Authority (TVA) Browns Ferry Nuclear Plant Unit 1 restart presented a unique EPU challenge. While Unit 1 was currently being modified for restart, it was also being uprated to 120 percent. A strategic goal by the corporate and site management was to accomplish both 100 percent and 120 percent design functions and licensing requirements for the plant. This required ongoing involvement of both the recovery design team and the EPU evaluation team to achieve the end goal.
5) Progress Energy senior leaders kept a high priority on restoring and maintaining margins during the Brunswick Steam Electric Plant power uprate. Brunswick used a phased approach in the power uprate project that included maintaining an N+1 philosophy for maintaining design and operating margins. Existing design and operating margins were maintained and improved.

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Appendix 2

Manage Projects Effectively

Characteristic: **The station uses the fundamentals of project management with tools for scope control, phased work approach, schedule, risk management, and budget to plan and implement the power uprate.** Examples are as follows:

1) The project teams at several sites identified all key stakeholders affected by a power uprate. They developed communication and contingency plans to share with these stakeholders before, during, and after implementation of the power uprate. Effective communication helped to reduce error-likely situations and contributed to event-free operations. Error-likely situations that could result from a power uprate include the following:

    a. changes that affect industrial, radiological, environmental, or nuclear safety

    b. management and organizational changes necessary for implementation

    c. changes to programs or processes that control how the work is done during implementation

2) Brunswick Steam Electric Plant performed a feasibility study to lay out the options involved with its power uprate. Based on the plant limitations identified, equipment modifications required to support raising power beyond various power plateaus were evaluated. The feasibility study also included an assessment of interrelationships with other plant improvement projects, outage schedule impact, regulatory requirements, and evaluation of the economic benefit associated with achievable power uprates.

3) Many plants generated and published a thorough list of project milestones and schedules that were monitored with periodic reports to the site leadership team. Performance indicators were used for cost, schedule, quality, and earned value.

4) The Donald C. Cook Nuclear Plant feasibility study addressed the primary and secondary sides of the plant separately. For the primary side, the study used an expert panel to identify fuel and accident analyses limitations and budgetary costs for resolving the limitations. For the secondary side, a documentation search, walkdowns, and plant personnel interviews were used to identify potential "pinch points" related to the secondary side design limitations. When combined, the primary and secondary side evaluations were documented in a graduated approach and compared on a cost-per-MWe installed basis.

5) D. C. Cook also used a project scope change form that requires at least the project manager's approval for all scope changes that could impact project objectives. The project sponsor's approval including a discussion of the risk associated with performing more work or deleting scope, is also required.

6) Several plants used a budget checkbook to track the use of contingency funds. Risk and contingency funds are used after a thorough review of the alternatives.

Appendix 3

Assess Actual Plant Conditions

Characteristic: **The as-built design margins and performance capabilities of the plant are well documented and understood and are used in the power uprate project planning.** Examples are as follows:

1) Actual plant conditions were used to review the options for the Brunswick Steam Electric Plant power uprate. During project planning and scope identification, reactor water cleanup, feedwater heater, and reactor building closed cooling water heat exchangers with degraded tubes were identified that would not support the power uprate. Main transformer and high-pressure turbine upgrades were also required to achieve an extended power uprate. This prompted the station to include modifications and extraordinary maintenance to maintain or regain operating margins.
2) Palisades Nuclear Plant performed interviews with the systems and design basis owners to understand current system performance as a component for the margin uncertainty power uprate. As the design change package was prepared, more engineering program reviews identified several calculations on secondary-side large motor loads that needed to be updated to account for the increased flows expected in the post-power-uprate condition.
3) The Browns Ferry Nuclear Plant approach to the integration of operating conditions in addition to the design basis evaluations for EPU resulted in several significant modifications to restore and increase operating margins for EPU. Examples are the upgrading of the condensate, condensate booster, and feedwater pump capacities to improve transient performance, reliability, and operating flexibility. Browns Ferry identified the need to perform certain upgrades through detailed system-based component reviews. This involved a system impact review by considering the effect the power uprate would have on individual components. Browns Ferry applied the lessons learned from the Quad Cities Station and Dresden Station EPUs.
4) Browns Ferry made notable investment in the improvement of equipment reliability and availability while achieving unit recovery and EPU objectives. For example, Unit 1 will incorporate new low-pressure turbine rotors, new high-pressure turbine rotors, and new upgraded reactor feedwater pump turbine rotors for life extension and outage reduction benefits.

5) Exelon performed an extent-of-condition review of design and operating margins to address issues from the Dresden and Quad Cities power uprates. Over 50 systems were reviewed and 101 recommendations, which included more monitoring, preventive maintenance activities, inspections, modifications, and analyses, were produced by the review. System engineers, operators, and maintenance workers were interviewed, and walkdowns were conducted. Many of these were to address vibration concerns, but some included inspections to account for additional heat and stress on various components (such as the main generator and large motors).

   For plants that experience a significant increase in feedwater iron content, compensatory measures to vacuum additional control rod drive mechanisms were recommended. Exelon wrote a project instruction to describe the process to perform the system reviews that identified and mitigated EPU vulnerabilities. The review evaluated the following:

   a. process parameters, such as flow, temperature, pressure, moisture, and fluid state, that have been affected by EPU
   b. affected components and subcomponents
   c. potential of a component/subcomponent failure to result in unacceptable consequences such as a scram or entry into a technical specification limiting condition of operation
   d. affected characteristics of components and subcomponents, such as vibration levels, stress, and impact velocity
   e. vulnerabilities and failure modes, such as high cycle fatigue, wear, erosion, and aging, resulting from the affected characteristics
   f. prioritization and recommended actions of identified vulnerabilities

6) Palo Verde Nuclear Generating Station collected plant data using an engineering test procedure prior to implementing a 3 percent power uprate. This test also gathered data on the low-pressure (LP) rotor that was used to determine efficiency gains with the LP rotor modification. A large number of secondary plant systems were evaluated to determine margin on critical systems (feedwater heaters, heater drain system, and cooling systems) and determine vibration levels on piping lines in both the primary and secondary systems.

7) At Calvert Cliffs Nuclear Power Plant, the thermal performance model was used to document the as-built configuration. Then the model was extrapolated to document post-uprate temperatures, pressures, and flows. The data was then sent to all affected systems managers and the design engineers for evaluation. The results helped focus the station on comparing actual plant test data to the expected post-uprate condition and address any differences.

   The BWR Owners Group (BWROG) and GE Nuclear reported in their lessons learned document that they have changed their management approach for performing extended power uprates. GE is now working with plant personnel to ensure that the project is evaluated from an overall equipment reliability perspective. More detailed work is performed to assess the actual condition of the plant. The specific attributes were as follows:

   a. a plant component assessment during the evaluation phase of an EPU project
   b. a review and evaluation of plant operational data, trends, and other operating experience (including interviews with the plant staff)
   c. An evaluation of the current operating margin (as opposed to initial design margin), and an estimate of margin available at EPU conditions
   d. an evaluation of the potential system and component vulnerabilities caused by EPU implementation
   e. the development of recommendations to improve operating margin

8) The BWROG developed a recommendation to ensure that the preventive maintenance (PM) on system components that may be impacted by EPU is evaluated prior to EPU. For example, the BWROG identified the need to increase PMs on components that currently require frequent maintenance or replacement and on components for which the required level of maintenance is increasing.

Appendix 4

Focus on Operating Experience

Characteristic: **Station efforts are focused on using operating experience and industry expertise to review power uprate planning and progress at key project milestones.** The following are examples:

1) Quad Cities Station used the power uprate startup lessons learned from its sister station, Dresden. Quad Cities changed procedure steps for turbine warming such that operators ensured the first-stage pressure limits were not encroached by the bypass scram setpoint limit. Another lesson incorporated by Quad Cities was to make procedure changes to ensure the feedwater suction pressure did not decrease to the condensate pump auto-start setpoint that starts the fourth condensate pump to restore suction pressure. Using these lessons learned avoided startup events.

2) During the Dresden Station power uprate, relevant operating experience was collected and analyzed as part of the modification package. Where appropriate, adjustments were made to the design to accommodate lessons learned from modifications at other utilities.

3) Brunswick Steam Electric Plant incorporated industry operating experience into a plant newsletter to augment the chemistry technical training program. The lessons learned focused on new plant equipment.

4) The Browns Ferry Nuclear Plant extended power uprate project team performed extensive benchmarking of BWRs that had evaluated and implemented EPU for their facilities. This included multiple site visits to targeted plants and the development of a comprehensive EPU interview survey. Good practices and lessons learned were identified throughout the industry's collective experience.

5) To minimize the number of regulatory information requests, Palo Verde Nuclear Generating Station and Calvert Cliffs Nuclear Power Plant reviewed the regulatory requests for information that other plants answered. This ensured a more streamlined submittal and minimized the number of regulatory information requests.

6) The Palo Verde shift technical advisors reviewed internal plant operating experience to identify systems needing closer review.

7) The BWROGs performed an extensive review of events that may have been related to EPU and also conducted a survey of plants that had performed EPU to look for common vulnerabilities. The summary of key recommendations developed by this team include the following:

    a. Prior to EPU implementation, identify existing material condition deficiencies that are affected by EPU-related changes. For example, a resolution to an existing overpressure condition on low-pressure heaters and drain coolers would be needed if the plant had a history of normally lifting the relief valves during a reactor trip. Identify mitigating actions or justify the risk of proceeding with the known deficiency.

    b. Review and disposition issues resulting from the GE/Exelon

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extent-of-condition review to ensure that increased component wear following implementation of EPU does not adversely impact plant reliability. Obtain the recommended pre-EPU baseline data (including vibration data), and compare this data with the post-EPU implementation data and EPU predictions to ensure that unanticipated effects are not occurring.

c. Consider steam dryer acoustic loads evaluation and complete appropriate upgrades/modifications prior to implementation of EPU.

d. Update operational procedures and preventive maintenance strategies related to systems, equipment, and components that are susceptible to changes as a result of EPU implementation.

Appendix 5

Implement a Post-Uprate Testing and Monitoring Plan

Characteristic: **Efforts are taken to predict how plant performance can change as a result of power uprate and defining which key parameters are trended. When actual performance deviates from predictions, the station aggressively seeks to understand the differences.** Examples are as follows:

1) River Bend Station developed a tool to trend various power-dependent parameters and compare those to core thermal power. This methodology offers a second check to watch for ultrasonic flow instrument deviations from other plant parameter values. River Bend shared this with the industry through Nuclear Exchange® document NX-1057, "Configuration Management: River Bend Station Best Estimate Core Thermal Power." Other vendors have similar tools available to evaluate heat balance and approximate core thermal power.

2) The Calvert Cliffs Nuclear Power Plant environmental qualification program was reviewed, and several areas required confirmation of the base assumptions. A program was developed to perform temperature monitoring throughout the plant. This is an ongoing program that will capture seasonal swings in temperature.

3) As part of the extent-of-condition review of events following power uprate projects within Exelon, a thorough review of the project implementation was performed, with the following conclusions:

   a. Increased feedwater flow has increased the fatigue loading on some vessel internals, which may require more frequent inspections of susceptible components.

   b. Increased core differential pressure has changed the jet pump flow and consequently the loading on the jet pump support components. These components require accelerated inspections.

   c. Changed operating conditions increased component wear, which required implementation of enhanced PMs.

   d. Increased feedwater flow, steam flow, and recirculation pump speed result in increased vibration on the system piping and components. Components with known preventive or corrective maintenance as a result of vibration were evaluated for potential increased vulnerability.

   e. Elimination of the standby feedwater and condensate pumps and operation of these pumps at non-optimum flow conditions have introduced gradual component degradation

   f. Balance-of-plant valves and internal components were assessed for the effect of increased feedwater and condensate flow.

   g. Increased feedwater flow has increased the effects of flow-accelerated corrosion.

   h. Known system deficiencies were not corrected prior to EPU implementation, resulting in more pronounced operational challenges.

   i. Post-EPU operating and analytical margins have been reduced.

4) Exelon also studied different flow-induced vibration failures and developed a

strategy to address the vulnerabilities. Specifically, Exelon instituted upgrades and accelerated preventive maintenance (PM) tasks on solenoid-operated relief valves. Generic vibration monitoring recommendations included the following:

a. For planned vibration monitoring data points, obtain pre-EPU baseline data. This should be performed prior to any uprate modifications that would change system performance (turbine modifications).

b. Evaluate main steam and feedwater components for potential wear-related degradation. Look for components with increased PM frequency as a result of known vibration vulnerabilities.

c. Perform baseline inspections and walkdowns either during or prior to EPU outage. Based on maintenance history, include any components identified as vulnerable to vibration degradation. Inspect mechanical joints to ensure that proper locking mechanisms are in place.

5) Palo Verde Nuclear Generating Station used special startup test controlling procedures after the combined power uprate and steam generator replacement outage. The procedure provided the operators with expected operating bands of many key parameters, such as reactor coolant system (RCS) temperature, feedwater temperature, feedwater flow for the venturi and ultrasonic instrumentation, first-stage turbine pressure, RCS mass flow rate, and steam generator pressure.

6) Palo Verde used thermography to check for proper installation of the steam generator insulation following the power uprate and steam generator replacement outage.

7) The Boiling Water Reactor Owners Group (BWROG) EPU lessons learned and General Electric (GE) experience revealed that the testing results were very dependent on the unique equipment characteristics of the individual plant. Consideration should be given to assigning an EPU test director who is an experienced senior reactor operator with experience in control system testing. This is key to successful pressure and feedwater level control testing. Also, it is recommended that the EPU procedure writer be a member of the EPU test team to simplify procedure changes when required.

8) D. C. Cook Nuclear Plant and others' power uprate projects used separate test procedures, modeled after operations procedures, that included power level plateau hold points to document and evaluate power ascension plant operating parameter data.

Appendix 6

Anticipate Design Shortfalls

A. ***Characteristic:*** **The project predicts and plans for the contingency of reconstituting some aspects of the original design.** Examples are as follows:

1) Palisades Nuclear Plant identified low operating margin with the moisture separator reheaters pressure relief valve setpoint. Analysis and compensatory measures were taken to raise relief valve setpoints an outage prior to the power uprate. Identifying this design margin issue avoided an inadvertent relief valve lift or a leaking relief valve with the unit at higher power.

2) The Brunswick Steam Electric Plant management team focused on margins during the power uprate project. The margin management strategy provided increased flexibility for the operators and improved some of the probabilistic safety analysis results. Brunswick Unit 1 was the first GE plant to reach the full 120 percent uprate. Brunswick focused on regaining margin during the project and installed higher-capacity feedwater pumps that enabled the plant to tolerate the loss of a feedwater pump without a scram. Brunswick also increased the enrichment of boron in the standby liquid control system and was able to decrease core damage frequency and large early release frequency to be lower than the pre-EPU values. The increase in boron concentration for the standby liquid control system improved margin and reduced the required number of pumps from two to one during an anticipated transient without scram event.

3) Tennessee Valley Authority (TVA) conducted critical reviews of plant design basis as part of the EPU. The EPU evaluations uncovered limiting impacts on design bases that needed to be updated or created and affected downstream design products. Comprehensive reviews incorporating multidisciplined and organizational contributions led to higher-quality products.

4) TVA comprehensively reviewed fuel-related concerns for the EPU analyses affecting transients.

5) Calvert Cliffs Nuclear Power Plant created a power uprate document that presented the remaining margin of all balance-of-plant reviewed equipment. In many cases, this was the first time a basis had been documented and retained for some balance-of-plant components. It provided an excellent starting point for future power uprate work.

6) Several plants did not take the entire benefit for their ultrasonic feedwater flow measurement uncertainty power uprates. By withholding a small fraction of the uncertainty, they allowed themselves some additional margin.

7) The BWROG recommended that in the plant evaluation process for EPU, the effects on existing plant system and equipment margins be evaluated and incorporated into the uprated plant cost-beneficial design enhancements. This would ensure that sufficient margin is maintained for reliable plant performance.

Appendix 7

Promote Effective Engineering Human Performance

A. *Characteristic:* **Station leaders ensure a robust engineering human performance culture is present to promote and expect superior engineering product quality.** Examples are as follows:

1) TVA ensures that all supplemental station personnel participate in the weekly quiz on site procedures, plant initiatives, and human performance fundamentals. This keeps the supplemental staff current on station events and reinforces expectations.

2) On large design projects, Comanche Peak Steam Electric Station allows time for personnel to walk down and validate design drawings prior to starting on the design work. This ensures the staff is using as-built field information and minimizes the potential of substandard design work.

3) The Progress Energy procedure on engineering product quality uses a checklist for vendor product quality management. It provides a number of good items to consider when working with a vendor. For example, the checklist prompts the person involved with the oversight to understand the vendor's modification procedure, including the corrective action process and the vendor's use of error-reduction tools.

4) Palo Verde Nuclear Generating Station used a third party to review low-pressure rotor modification test procedure and test results. The use of the third party helped to identify testing shortfalls in the turbine and provided greater insight into the test results.

5) At several utilities, leadership listens closely to contrary viewpoints and ensures positions are resolved. Exelon established a technical human performance process to increase the awareness of how engineering contributes to the technical conscience of the plant.

6) Exelon and Progress Energy developed processes and tools to facilitate vendor and supplemental personnel oversight. Vendor or equipment suppliers are questioned and challenged by the plant staff to gain a complete understanding of any complex technologies being implemented. Also, the leadership team should ensure that those monitoring the projects have the right skill sets to oversee vendor work. In one case, the quality review involves the manager reviewing the questions being asked and the answers given between design teams and supplemental personnel to verify that the questions and responses are of sufficient depth and technical rigor. Some plants make broad use of third-party reviews and vendor surveillances to raise product quality.

7) Waterford 3 Steam Electric Station established a core team that

reviewed vendor calculations and evaluations following vendor quality assurance review but prior to final approval. Comments were documented and resolved prior to vendor approval. This enabled factoring the core team knowledge and experience into the product without the need for rework after the product had been approved. Although time-consuming, this resulted in a better documented, higher-quality design basis.

8) Browns Ferry Nuclear Plant maintains site standards on completing work to support the refueling outage milestones. There is industry experience in which EPU project teams made a conscious decision to disregard key outage milestones and did not take sufficient compensatory measures to identify shortfalls. Browns Ferry established decision points in a project schedule to decide if the project was able to continue or if it needed to be deferred. If milestones are missed, TVA will not compress engineering review time to stay on schedule. Rather, the completion date milestone is extended.

9) D. C. Cook Nuclear Plant used dedicated, experienced primary-side and secondary-side project engineers to validate design inputs, coordinate owners' acceptance reviews of vendor products, and develop and implement the power ascension test procedure. Supplemental personnel were contracted for uncertainty calculation preparation, software specification development, and software implementation testing.

10) Brunswick Steam Electric Plant relied on a wide range of training initiatives to prepare for the power uprate implementation. Simulator training for the operating crews was conducted to support the post-extended-power-uprate startup, and the simulator model shifted between the two units to allow the crews more focus on the unit with the power uprate. Vendor training for the technical training representatives was also conducted so that the on-site training programs could be updated.

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Lessons Learned from Power Uprates

Limited Distribution August 21, 2002

Executive Summary

More than 40 events have occurred over the past five years as a result of inadequate analysis, design, or implementation of plant power uprates. Many of the events involved equipment damage, unanticipated responses to plant conditions, or challenges for operating staff. The number and types of events indicate that more significant consequences could result from future events if power uprates are not conducted in a thorough and carefully controlled manner.

Significant aspects of these events include the following:

- An extended, unplanned shutdown was required to retrieve several loose parts as a result of a flow-induced, high-cycle fatigue failure of a steam dryer cover plate.
- Operational transients and equipment damage have occurred as a result of weaknesses in identifying, communicating, and training the plant staff on expected changes to secondary plant operating characteristics.
- Unanticipated operating challenges and degraded equipment performance have resulted from reductions in operating and design margins.
- Some units have operated beyond their licensed power levels for extended periods because of errors in reactor thermal power calculations following uprates that changed secondary plant operating characteristics.

Planning, resource, implementation, testing, and data analysis errors have contributed to many of these events. Since the majority of industry experience is from uprates of less than 10 percent power, these events may be precursors to more severe transients following larger power uprates that are planned for the future.

Background

Cheap and Easy Megawatts?

Improved measurement and analysis techniques have allowed utilities to increase the licensed power limits of existing plants as a cost-effective method for adding nuclear plant generating capacity. A number of utilities are planning to implement power uprates within the next few years. While there are considerable economic benefits to power uprates, the complexity and significance of issues associated with power uprate projects make additional megawatts gained from uprate projects anything but "cheap and easy." As noted in the following quote from Zack Pate, WANO chairman, at the 2002 WANO Biennial General Meeting, there is a need to be wary of the potential nuclear safety impact of power uprate projects:

"The U.S. National Energy Strategy, recently issued by President Bush, specifically encourages nuclear power plant uprates. This is a good thing, and it is certainly not my intent to discourage uprates. We must rely on our suppliers, regulators, and our own engineers to retain the necessary margins. At any one plant, an uprate is a good thing. But in the aggregate worldwide, these uprates increase the possibility of breaching the fuel cladding. ***By decreasing our margins, we are relying more and more heavily on our operators, engineers, and managers to make the right decisions, and to make them in a timely manner.****"*

Zack T. Pate

Types of Uprates

Power uprates are typically classified by their size, as follows:

- **Measurement Uncertainty Recapture Power Uprates:** uprates of 1 to 2 percent power, typically achieved using more precise techniques for measuring feedwater flow
- **Stretch Power Uprates:** uprates of 5 to 7 percent power, typically achieved by changing instrumentation setpoints, with few major plant modifications
- **Extended Power Uprates:** uprates of up to 20 percent power, achieved through advanced core design and by significant modifications to major plant equipment

Event Descriptions

Steam Dryer Damage at Quad Cities

Following an extended power uprate, increased steam flow rates led to a high-cycle fatigue failure of a steam dryer cover plate. The plate broke into several pieces, resulting in a 10-day forced outage to retrieve the loose parts. This condition was not anticipated because the effects of the increased steam flow conditions in combination with existing steam dome forces on the steam dryer were not well understood. **(See reference 1.)**

Quad Cities Unit 2 completed an 18 percent extended power uprate (EPU) in the first quarter of 2002. On June 7, operators noted a reactor vessel pressure decrease from 1,001.1 psig to 998.8 psig and a tenfold increase in moisture carryover to the turbine from 0.028 percent to 0.27 percent. On June 20, the A channel of reactor vessel water level indication showed level to be 4 inches lower than the other channels, and moisture carryover peaked at 0.735 percent. On June 30, operators observed a decrease in the A main steam line flow and a 6 psi increase in reactor pressure. During the next week, steam pressure increased to 15 to 20 psi above the initial pressure. Plant management directed that the plant be shut down on July 11 because of the concern that loose parts may have exited the reactor vessel and traveled into the main steam lines.

Upon inspection, a section of the steam dryer outer bank hood cover plate was found to be missing. The cover plate, 10 feet long by 16 inches wide and ¼ inch thick, had separated into three large sections and several small pieces. One section was found on top of the steam separator; another section was partially separated, but still attached to the steam dryer; and the third section was lodged in the A main steam line venturi nozzle. This piece was about 12 inches wide and 18 inches long. One small piece was found downstream of the venturi, and several small pieces were found in the turbine stop valve strainer. Visual inspections identified impingement damage to the A main steam line nozzle, minor surface damage on steam line piping, and minor damage to the A main steam line flow venturi nozzle. Station personnel determined the damaged steam dryer cover plate had allowed steam to bypass the dryer and exit the reactor directly through the A main steam line.

The station investigation established the preliminary failure cause as a high-cycle fatigue failure of the steam dryer cover plate. The high-cycle fatigue failure conditions resulted from the increased steam flow rates associated with the EPU. Specifically, the higher flow rates increased dynamic effects in the area adjacent to the cover plate. The resulting acoustic standing wave matched the natural frequency of the cover plate, causing a cyclic fatigue failure. The station used conventional modeling to evaluate steam flow conditions through the dryers as part of the uprate design and, therefore, did not predict this condition. The condition was later understood when the vendor developed a physical model to mimic these flows using air.

The damaged cover plate and the similar cover plate on the opposite side of the steam dryer were replaced with new, ½-inch thick cover plates with enhanced welding.

Extended Operation in an Overpower Condition at Monticello

Monticello was operated for an extended period at greater than 100 percent power because changes to the process computer calibration constants for feedwater flow were not identified when the feedwater transmitters were changed. **(See reference 2.)**

On April 4, 2000, with Monticello at approximately 100 percent power, a technician calibrating the feedwater flow transmitters noted a small mismatch between the transmitter calibration values and the corresponding values generated by the plant process computer. The investigation determined that the span of the feedwater transmitters was changed on October 22, 1998, following a 6 percent power uprate. However, the process computer calibration constants for feedwater flow were not changed when the spans of the corresponding transmitters were changed. These computer points are used in the reactor thermal power calorimetric calculations. This error resulted in calculated reactor power being approximately 3.7 MWt, or 0.2 percent below actual power. The net effect of this error was that actual reactor power exceeded the maximum licensed power by 0.2 percent for 316 days. Other overpower events related to power uprate projects are discussed in LER 278-97001, "Nonconservative Feedwater Temperature Instrument Calibrations Result in Operation at Power Levels Greater Than Licensed Maximum," and NRC Plant Event Report Number 38579, "24-Hour Report Due to Potential Operation in Excess of Operating License Condition 2.C(1)."

Main Steam Isolation Signal Actuations During Turbine Stop Valve Testing at Dresden

Following an extended power uprate, several partial actuations of the main steam isolation logic occurred as a result of reduced margin between main steam pressure and the associated isolation setpoint. The reduced operating margin increased the potential for inadvertent main steam isolation while operating at rated power. **(See reference 3.)**

Dresden Unit 2 implemented a 17 percent EPU in late 2001. As part of the uprate, the high-pressure turbine was modified to accept higher steam flows. These modifications reduced main steam throttle pressure and provided a decreased margin between normal operating pressure and the main steam line low-pressure isolation setpoints. Following the uprate, partial actuations of the main steam isolation logic occurred on several occasions during main turbine stop valve testing and during steady-state operation at the new maximum attainable power level.

Although GE Service Information Letter 130 provides a recommended operational margin of 125 psig between rated main steam throttle pressure and the low-pressure main steam line isolation setpoints, the Dresden EPU modifications reduced this margin. The margin reduction, combined with normal instrumentation drift and other calibration issues, resulted in an actual operational margin of only 16 psig. The margin reduction was such that minor fluctuations in main steam pressure were sufficient to initiate intermittent isolation signals on one channel of the isolation logic.

The reduced operating margin was not evident during power ascension testing because reactor pressure was adjusted for a relatively high pressure setting before main turbine stop valve testing at each power level. This approach provided the largest possible margin between actual steam pressure and the isolation setpoint. During power ascension, the main steam line pressure switches were calibrated to the low end of their setting tolerance band, providing an additional 6 psig margin between rated main steam throttle pressure and the low-pressure main steam line isolation setpoints. These settings were maintained throughout subsequent testing. The main steam line low-pressure isolation pressure switches were then recalibrated, with two switches adjusted to settings closer to the setpoints. After the isolation signals were received, two pressure switches were found to have drifted high in the setting tolerance band, and the reduction in margin enabled main steam line pressure fluctuations to cause intermittent isolation signals.

Unexpected Feedwater Heater Problems at Browns Ferry

Existing feedwater heater materiel condition issues were recognized and considered in the power uprate project feasibility study, but they were included as a risk item and were not included in the original project budget. The need to repair feedwater heater shells and nozzles to correct these issues increased the initially approved scope and cost of the power uprate project.

The original power uprate feasibility study at Browns Ferry concluded that the feedwater heaters may not be adequate for a planned uprate to 120 percent power. However, the decision was made not to include replacement of feedwater heaters in the power uprate project cost/benefit study, but to track it as a risk item. Subsequent detailed analysis associated with the power uprate project revealed that feedwater heater materiel condition on both units was marginal for uprated conditions.

Feedwater heater shell thinning was present in all high-pressure and some low-pressure feedwater heaters, especially near the nozzle areas. The problems were caused by the original equipment manufacturer's fabrication methods. The feedwater and steam inlet and outlet piping were directly welded to the feedwater heater shells without collars or sleeves. As a result, 50 percent of the nozzles on the feedwater heaters (36 nozzles total for both units) require repair. Although the problem was identified before an actual plant event could occur, the power uprate project was impacted. This problem increased the scope and added expense to the power uprate project.

Events involving feedwater heater damage resulting from increased steam flows are discussed in OE 14135, "Multiple Heater Tube Leaks on Four Series Feedwater Heaters"; OE 8880, "Tube Leaks in Low Pressure Feedwater Heaters"; and OE 13006, "Broken Extraction Steam Impingement Plate Leads to Excessive Tube Failures in Feedwater Heater."

Turbine Control System Changes Result in Unanticipated Operational Challenges at Byron

Operators experienced difficulty controlling turbine speed and generator load following a power uprate that included installation of a high-efficiency turbine rotor. The need for new operating strategies was not recognized during preparations for the uprate. **(See reference 4.)**

On October 13, 2000, following installation of a higher-efficiency turbine rotor to support a 5 percent power uprate modification, Byron Unit 1 operating personnel found it difficult to control the speed of the turbine when synchronizing the generator to the grid. Subsequent investigation determined that the higher turbine efficiency required less steam to achieve the same rotor speed, causing the governor valves to operate closer to the full-closed position. (With the valve only slightly open, turbine speed control was difficult for the fine speed adjustments needed for paralleling the generator to the grid.)

Another unanticipated impact was identified when the turbine control valves on both units were required to be nearly full open at the new uprated full-power level, rather than operating in a partially throttled position. This condition reduced the ability of the turbine control system to automatically respond to system changes and required a new operating strategy by operations. The new strategy required operators to increase power until all four turbine control valves were fully open, then reduce generator load until the number four control valve was partially closed. This strategy maintained the ability of the turbine control system to respond to load changes. Subsequent review determined that the uprate project team and the power uprate vendor were uncertain of the postmodification control valve position at 100 percent power. Other similar events include OE12280, "Transfer of Turbine Control from Manual to Automatic Results in Unplanned Power Excursion," and OE9684, "Turbine Control Valve Oscillations Following Power Uprate."

Reduced Stator Cooling Water System Operating Margin Contributes to a Power Reduction at Diablo Canyon

A power reduction was required as a result of stator cooling water system flow and temperature problems following a power uprate. Operation with reduced stator cooling water system differential temperature operating margin following a power uprate contributed to this event. **(See reference 5.)**

Between November 28 and December 14, 2000, while Diablo Canyon Unit 1 was at 100 percent power, an unexpected increase occurred in the stator cooling water system differential temperature. Chemical cleaning had been done immediately prior to the last refueling outage. The maximum differential temperature in the Unit 1 stator rose from 13.8 degrees Fahrenheit on November 28, 2000 to 18 degrees Fahrenheit on December 14, 2000. Power was reduced from 1,155 MWe to 1,007 MWe over the following four days to stabilize the differential temperature at 17-18 degrees Fahrenheit. The maximum allowable differential temperature is 21 degrees Fahrenheit.

The cause of the increased differential temperature was the formation of copper oxides that lowered the flow in the stator cooling water system. However, some of the differential temperature increase was attributed to an increase in electrical output from 1,130 MWe to 1,155 MWe following the power uprate project. Station personnel noted a nearly linear relationship between differential temperature and generator electrical output.

Analysis

The events described, and the much larger body of available industry operating experience associated with power uprates, were reviewed to identify areas where problems are commonly encountered following power uprates. The following list highlights system and component problems related to power uprates:

- Main Steam Systems: Unanticipated resonant vibrations in a steam dryer cover plate occurred as a result of the increased steam flow. Also, decreased margins in steam line pressure setpoints resulted in intermittent, partial main steam line isolation signals.
- Feedwater Heater and Extraction Steam Systems: Feedwater heater equipment problems have occurred following power uprates because of higher steam flow through the feedwater heaters. Typical problems include tube damage, extraction steam impingement plate damage, and shell thinning.
- Main Turbine Control Systems: Changes to turbine control systems and/or plant operating conditions have resulted in transients and have challenged the ability of the operating staff to control load or turbine speed following plant uprates. In many cases, the operational effects of the changes were not fully understood before the uprates were implemented.
- Feedwater Flow and Temperature Measurement: Modifications to improve feedwater flow and temperature measurement were sometimes implemented without the modification's impact on reactor thermal power measurements being fully assessed and validated.

- Main Generator Cooling Systems: Problems have been encountered with high gas and stator cooling water temperatures as a result of power uprates. Contributing to this condition is that, in many cases, the main generator is the power-limiting component following power uprate projects and is operated near the bounds of the generator capability curve.
- Isolated Phase Bus Duct and Transformer Cooling Systems: In some cases, a reduction in main transformer oil temperature or isolated phase bus air temperature operating margin (operating closer to high temperature limits) has challenged the operating staff and made the plant more vulnerable to equipment problems during high seasonal ambient temperature conditions.
- Grid Stability: Information shared during a power uprate working meeting held at INPO on June 11 and 12, 2002 noted that many plants performing extended power uprates must address the impact on grid stability of the additional power generated. In some cases, additional modifications were required to address grid stability issues. In addition, a discussion of grid loss issues can be found in SOER 99-1, "Loss of Grid."
- Reactor Core: Power uprates usually involve a transition to a higher energy core. Recommendations for addressing issues associated with transitioning to higher energy cores can be found in SOER 96-2, "Design and Operating Considerations for Reactor Cores."

Lessons Learned

INPO personnel visited several site and corporate organizations with extensive uprate experience to discuss the problem areas described above. Information obtained during these visits, along with event and evaluation information, indicates that many of the undesirable outcomes listed above can be avoided if the uprate project organization is well resourced, staffed with personnel with extensive plant experience (particularly in the area of plant operations), and focused on identifying and resolving potential operational and other impacts.

Power Uprate Project Team Organization

- Most uprate project teams included a full-time project manager, and the project manager and team members had no additional duties.
- Most uprate project teams involved the operations and training departments early in the project. The most successful project teams included a senior reactor operator on a full-time basis from the beginning of the project.
- Successful uprate project schedules include sufficient lead time to allow adequate data analysis, identification of procedure changes, and reviews by plant personnel during all stages of the project. Sufficient time is also allotted for training station personnel and for incorporating changes in the control room simulator.

Feasibility Study Phase

- Late scope additions and outage planning challenges were minimized when feasibility studies considered existing equipment problems or limitations. Feasibility studies that used original design parameters often underestimated the scope of work required. As a result, project schedules were challenged when detailed analyses identified major additional equipment modifications.
- Following the feasibility study, projects that took a deliberate approach to implementation greatly increased the ability to support standard outage scheduling milestones. This also

contributed to more effective reviews of power uprate-related plant changes, modifications, training, and procedure revisions.

Detailed Analysis/Design Phase

- Power uprate projects benefited from plant engineering and operations personnel performing detailed reviews of power uprate project analyses. These reviews focused on ensuring that present plant conditions and equipment performance problems were addressed during system and components design reviews.
- Some power uprate license amendment requests used two separate amendments, one for the new fuel design and one for the power uprate. Some utilities found this project strategy to be advantageous because the power uprate amendment was not approved until after the implementation refueling outage was complete.
- Power uprate license amendments benefited from a review and incorporation of issues discussed in NRC "Request for Additional Information" (RAI) documents from other recent power uprate submittals. Also, because there is no standard NRC review plan, many stations have found that the RAIs provide insight into the critical topical areas for the license amendments. Experience has shown that NRC RAIs can significantly add to a power uprate team's workload when the project team resources are least available.

Implementation Phase

- Some extended power uprate projects used a two-stage implementation strategy, making use of two refueling outages to implement the changes. This was done to ensure that plant resources were not overburdened to review and implement a large number of plant changes in a short period of time. However, some utilities noted that additional administrative controls were necessary to address the interim state of the changes during the operating cycle between the two outages.
- Successful power ascension tests included an approach in which power is increased in increments, with holdpoints at predetermined intervals. These holdpoints can last several days while operators gain experience and resolve technical issues at the new power level. During this testing, actual plant parameters are compared to expected values, and contingency plans are available to identify actions needed to address unexpected plant parameters.

Ongoing Post-Power Uprate Operation

The project teams placed a great deal of emphasis on making operating staff and other station personnel aware of reduced operating margins and changes to various plant systems and components. Contingency plans have also been helpful in providing guidance to the plant staff when expected or unexpected conditions are encountered. Examples of conditions that often occur following uprates are provided below.

- Following power uprates, some plants now operate with secondary pumps and heat exchangers in service that were previously used as spares. Thus, removal of these components from service for maintenance following power uprates will require a power reduction from rated power. This has prompted changes to on-line maintenance strategies to accommodate the new operating limitations.
- Power uprate projects often require operating strategy changes. The traditional operating philosophy of operating at 100 percent reactor thermal power may not be applicable following

many power uprates because a plant system or component may be the power-limiting factor. Often, main generator limitations require that some plants operate at a constant electrical output and allow reactor thermal power to vary with efficiency.

- There is greater potential for temporary generation reductions at power-uprated units during hot summer weather. Various system or component limitations can be the primary causes for these power reductions. However, the most typical plant limitation contributing to this situation is condenser circulating water temperature and the associated increase in condensate temperature and decrease in condenser vacuum.
- Ongoing equipment problems may occur due to flow-accelerated corrosion (FAC) and vibration. Because of greater feedwater and steam flows associated with power uprates, there is an increased potential for FAC that could lead to failure. Additionally, increased vibration of components in systems experiencing increased flow rates has caused fatigue-induced failures. These conditions may not be readily identified during the analysis phase of the uprate project. Thus, the scope and frequency of FAC and vibration monitoring programs may need to be reconsidered following power uprates.

Economic evaluations for adding electrical generation under present industry conditions are likely to show justification for continued power uprating of existing nuclear facilities. Industry experience has shown that power uprates can be implemented safely and successfully. However, industry experience illustrates the need for additional focus on the importance of using a thorough, deliberate approach when planning and executing power uprates to avoid undesirable consequences.

References

1. Nuclear Network® Operating Experience Report OE14300, "Reactor Steam Dryer Found Damaged at Quad Cities Unit 2" July 19, 2002.
2. LER 263-00009, "Operation Above Licensed Power Due to Missed Process Computer Setpoint Changes," May 3, 2000.
3. Nuclear Network® Operating Experience Report OE14079, "Main Steam Isolation Valve Isolation Signal During Turbine Stop Valve Testing and Steady-State Operation after Power Uprate," June 25, 2002.
4. Byron 1 EPIX Report 376, "Unit 1 Generator Trip During Synchronization Attempt," October 13, 2000.
5. Nuclear Network® OE11839, "DCPP Unit 1 Stator Cooling Water Max DT Approached After Unit Uprating," January 18, 2001.

Additional Power Uprate-Related Operating Experience:

6. LER 278-97001, "Nonconservative Feedwater Temperature Instrument Calibrations Result in Operation at Power Levels Greater Than Licensed Maximum," February 19, 1997.
7. NRC Plant Event Report Number 38579, "24-Hour Report Due To Potential Operation In Excess Of Operating License Condition 2.C(1)," December 18, 2001.
8. Nuclear Network® OE14135, "Multiple Heater Tube Leaks on Four Series Feedwater Heaters," June 28, 2002.
9. Nuclear Network® OE8880, "Tube Leaks in Low Pressure Feedwater Heaters," March 26, 1998.

10. Nuclear Network® OE13006, "Broken Extraction Steam Impingement Plate Leads to Excessive Tube Failures in Feedwater Heater," November 30, 2001.

11. Nuclear Network® OE12280, "Transfer of Turbine Control from Manual to Automatic Results in Unplanned Power Excursion," May 22, 2001.

12. Nuclear Network® OE9684, "Turbine Control Valve Oscillations Following Power Uprate," February 22, 1999.

13. Nuclear Network® OE14373, "Main Steam Piping Low Point Drain Line Fails Due to Vibration Related to Extended Power Uprate (EPU) Modifications," August 2, 2002.

Event Criteria

Other plant safety or reliability events

Cause Categories

- Work organization/planning (insufficient time to prepare or to perform, or maintenance not scheduled)
- Change management (inappropriate plant modifications, lack of change-related retraining, procedures, or documents)
- Design configuration (inappropriate layout of systems or subsystems, inappropriate component orientation, or component omission)
- Analysis (errors in assumptions, methods, or calculations during design or establishment of operational limits)
- Equipment manufacturing (improper heat treatment, machining, or casting)

Utilities and participants are requested to provide feedback on similar occurrences and solutions at their plants or on their equipment to the information contact listed below.

Information Contact: Allen Smith (770) 644-8474, smithga@inponn.org

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Keywords

ANALYSIS, DESIGN CHANGE, DIFFERENTIAL PRESSURE, FEEDWATER HEATER, FLOW INDUCED VIBRATION, LOOSE PART, MAIN STEAM, MAIN STEAM LINE, POWER ASCENSION TEST, POWER UPRATE, PRESSURE SWITCH, SETPOINT, STEAM DRYER, TRAINING, TURBINE, UPRATE, VIBRATION

PREVENT EVENTS

Learning from Industry Experience

PREVENT EVENTS is intended for use by personnel during morning meetings, prejob briefings, and work unit meetings to communicate key industry experience. Recent industry events have occurred either in whole or in part as a result of power uprate projects. Many of these events involved loss of operating margin or unanticipated system, component, and instrumentation response to operation under increased flow or temperature conditions. These events are discussed in SER 5-02, "Lessons Learned From Power Uprates," August 21, 2002.

Management

1. What methods will be used to provide management oversight and direction for key power uprate decisions that affect plant operating strategy and reductions in operating margins? Who reviews the power uprate project milestones for resource conflicts, overloading of work groups, or activity/priority conflicts?
2. What will be the roles of the plant on-site and off-site safety review committees in the power uprate process?
3. What is the plan for identifying and coordinating changes in the on-line maintenance and equipment performance monitoring programs? What organization will lead this effort?

Power Uprate Project Managers

1. How will the power uprate project team be staffed? How will the team composition change as the project moves toward implementation?
2. What actions are we taking to ensure the feasibility study is complete and reviewed by operations and engineering?
3. How do we ensure that station management is familiar with the power uprate project plan and potential barriers? How do we communicate the operational impact of the uprate to operations, engineering, and training personnel?
4. What important systems or components were we not able to model during detailed design? How will we monitor these systems or components following the uprate to identify unexpected conditions?
5. How are industry lessons learned reviewed and incorporated into the power uprate project?
6. What is our strategy for obtaining NRC approval and scheduling the modifications for upcoming outages?

Operations

1. What support will operations provide to the power uprate project team? How will we ensure a senior licensed person will be an early participant with the power uprate team?
2. What power uprate training is planned for the operators? How will lessons learned from industry power uprate-related events be incorporated into training?
3. How will we identify the large number of procedure changes associated with a power uprate project? How will operations prepare inputs to the procedures for power ascension testing?
4. What operating strategies and special precautions are planned for power ascension testing following the power uprate? How will these strategies and precautions be addressed in operator training?

Engineering

1. How will the site engineering organization support and interface with the power uprate project team? What additional resources will be required to accomplish this?
2. How will the engineering organization support review of the modifications related to the power uprate while continuing to manage existing engineering programs?
3. What changes to the station's performance monitoring programs (such as vibration and flow-accelerated corrosion) are planned as a result of the power uprate project?
4. What special training will be provided to thermal performance and reactor engineers?

Training

1. How will major changes to plant equipment and operating strategies as a result of power uprates be identified for incorporation into the control room simulator? How will simulator changes be completed in time for operators to receive simulator training before assuming duties on the uprated unit?
2. How will power uprate post-startup data (changes to various plant parameters) be incorporated into simulator modeling and training programs?
3. How will the training department address initial and continuing training on an uprated unit and a non-uprated unit during interim implementation periods when all units have not been fully uprated? (for two- or three-unit sites)

Please provide feedback on the usefulness of PREVENT EVENTS to Bob Heublein, INPO Events Analysis Department manager, at (770) 644-8671 or mailto:heubleinrm@inponn.org.

Information Contact: Allen Smith, (770) 644-8474, smithga@inponn.org

PEF-CR3-0593

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AGENDA

CR3 MUR
EXECUTIVE MEETING
12-08-06

Project Status

- Schedule Overview
- Draft Deliverable Status
- Action Items
- Issues and Restraints
- Key Issues
  - RPS High Flux Trip Setpoint
  - Heat Balance Uncertainty Calculation
  - FIDMS vs Tricon

Proposal Status

- Overall Status
- Metrics
- Payment Plan

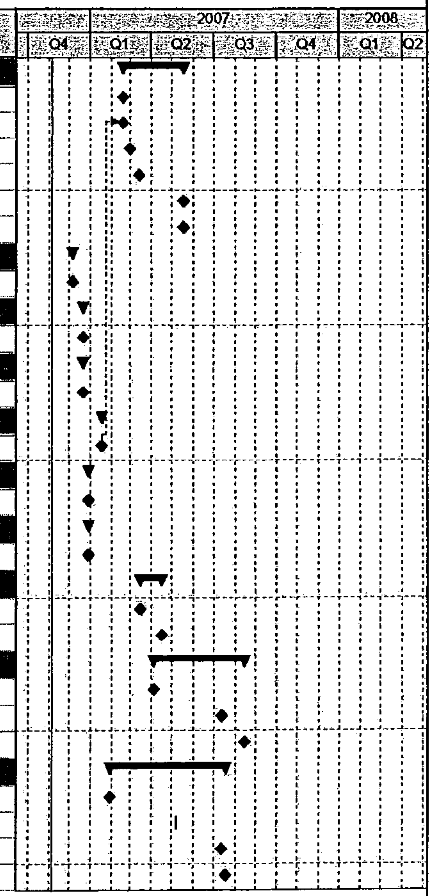
Summary

Project Status

Schedule Overview

MUR MILESTONE SCHEDULE — 06-Dec-06 17:17

Activity ID	Activity Name	RD	ES	EF	Total Float	CR3 Date	RESP
		64d	19-Feb-07	18-May-07	29d		
CRLAR126	Issue TS and FSAR Changes to CR3	0d		19-Feb-07*	0d	19-Feb-07	SCOTT
CRLAR129	Submit LAR to CR3	0d		19-Feb-07*	7d	28-Feb-07	SCOTT
CRLAR034	Transmit LOCA M&E to CR3 for Review	0d		01-Mar-07*	0d	01-Mar-07	SCOTT
CRLAR036	Issue Final LOCA M&E to CR3	0d		15-Mar-07*	0d	15-Mar-07	SCOTT
CRCPM011	Transmit Calculation Package Modifications to...	0d		18-May-07*	19d	15-Jun-07	ESQUILLO
CRPM001	Transmit Procedure Modifications to CR3	0d		18-May-07*	30d	29-Jun-07	CR3
NSSS I&C Systems		0d	07-Dec-06	07-Dec-06	0d		
CRSEIC060	Transmit to Progress Energy for Comments	0d		07-Dec-06*	0d	07-Dec-06	CHEATHAM
BOP Electrical Systems		0d	21-Dec-06	21-Dec-06	0d		
CRSEBE160	Transmit to Progress Energy for Comments	0d		21-Dec-06*	0d	21-Dec-06	BRUCE
NSSS Fluid Systems		0d	22-Dec-06	22-Dec-06	0d		
CRNFS180	Transmit to Progress Energy for Comments	0d		22-Dec-06*	0d	22-Dec-06	ESQUILLO
NSSS Structual Analysis		0d	18-Jan-07	18-Jan-07	0d		
CRNSA270	Transmit to Progress Energy for Comments	0d		18-Jan-07*	0d	19-Jan-07	SCOTT
BOP Fluid Systems		0d	29-Dec-06	29-Dec-06	0d		
CRBFS260	Transmit to Progress Energy for Comments	0d		29-Dec-06*	0d	29-Dec-06	STEWART
HVAC Systems		0d	29-Dec-06	29-Dec-06	19d		
CRHV120	Transmit to Progress Energy for Comments	0d		29-Dec-06*	20d	26-Jan-07	STEWART
Thermal-Hydraulics (T-H) Models/Operating Conditions		22d	16-Mar-07	17-Apr-07	7d		
CRTH031	Transmit Heat Balance Uncertainty Calc to CR3	0d		16-Mar-07*	0d	16-Mar-07	DEVENDORF
CRTH061	Transmit Simulator Inputs to CR3	0d		17-Apr-07*	8d	27-Apr-07	CLAUNCH
Fuel America		92d	05-Apr-07	15-Aug-07	0d		
CRFANAS031	FIDMS Version 1.8 Installed at CR3	0d		05-Apr-07*	0d	05-Apr-07	CREASY
CRFARL006	Transmit Draft Reload Report to CR3 for Review	0d		13-Jul-07*	0d	13-Jul-07	SUHOCKI
CRFARL011	Issue Final Reload Report to CR3	0d		15-Aug-07*	0d	15-Aug-07	SUHOCKI
AREVA Prepare Engineering Change Package		119d	31-Jan-07	19-Jul-07	8d		
CRECP110	Issue 30% Design Challenge	0d		31-Jan-07*	0d	31-Jan-07	DEVENDORF
CRECP125	CR3 Confirm for 70% Design Challenge	0d	09-May-07	09-May-07*	1d	10-May-07	DEVENDORF
CRECP099	Issue 100% Design Challenge	0d		12-Jul-07*	8d	24-Jul-07	DEVENDORF
CRECP100	Issue Caldon EC to Crystal River	0d		19-Jul-07*	8d	31-Jul-07	DEVENDORF



Actual Work — Critical Remaining Work — Summary
Remaining Work — Milestone

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PEF-CR3-0596

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Draft Deliverable Status

Deliverable	Due Date	Draft Date	CR3 Review Date
New Operating Conditions	**11/13/06**	**11/13/06**	**12/08/06**
Updated PEPSE	**11/22/06**	**11/29/06**	**12/08/06**
NSSS I&C System Evaluation	**12/1/06**	**12/01/06**	**12/15/06**
EC's Closed Since 2002	**12/6/06**	**12/6/06**	**12/20/06**
Electrical System Evaluations	12/21/06		
NSSS Fluid System Evaluation	12/22/06		
BOP Fluid Systems Evaluation	12/29/06		
HVAC Systems Evaluation	12/29/06		
NSSS Structural Evaluation	1/19/07		

Project Status

Action Items - Handout

Key Issues

RPS High Flux Trip Setpoint – Davis-Besse Issue

1. Operate at current trip setpoint of 104.9%. Have an LCO to reduce setpoint to 103.3% and reduce core power to 98.4% on loss of LEFM.
2. Operate all the time at a reduced setpoint of 103.3% and reduce core power to 98.4% on loss of LEFM.
3. Operate at higher setpoint all the time, but have LCO to recalibrate NIs to 2568 MWt = 100% on loss of LEFM.

Recommend we proceed with first option.

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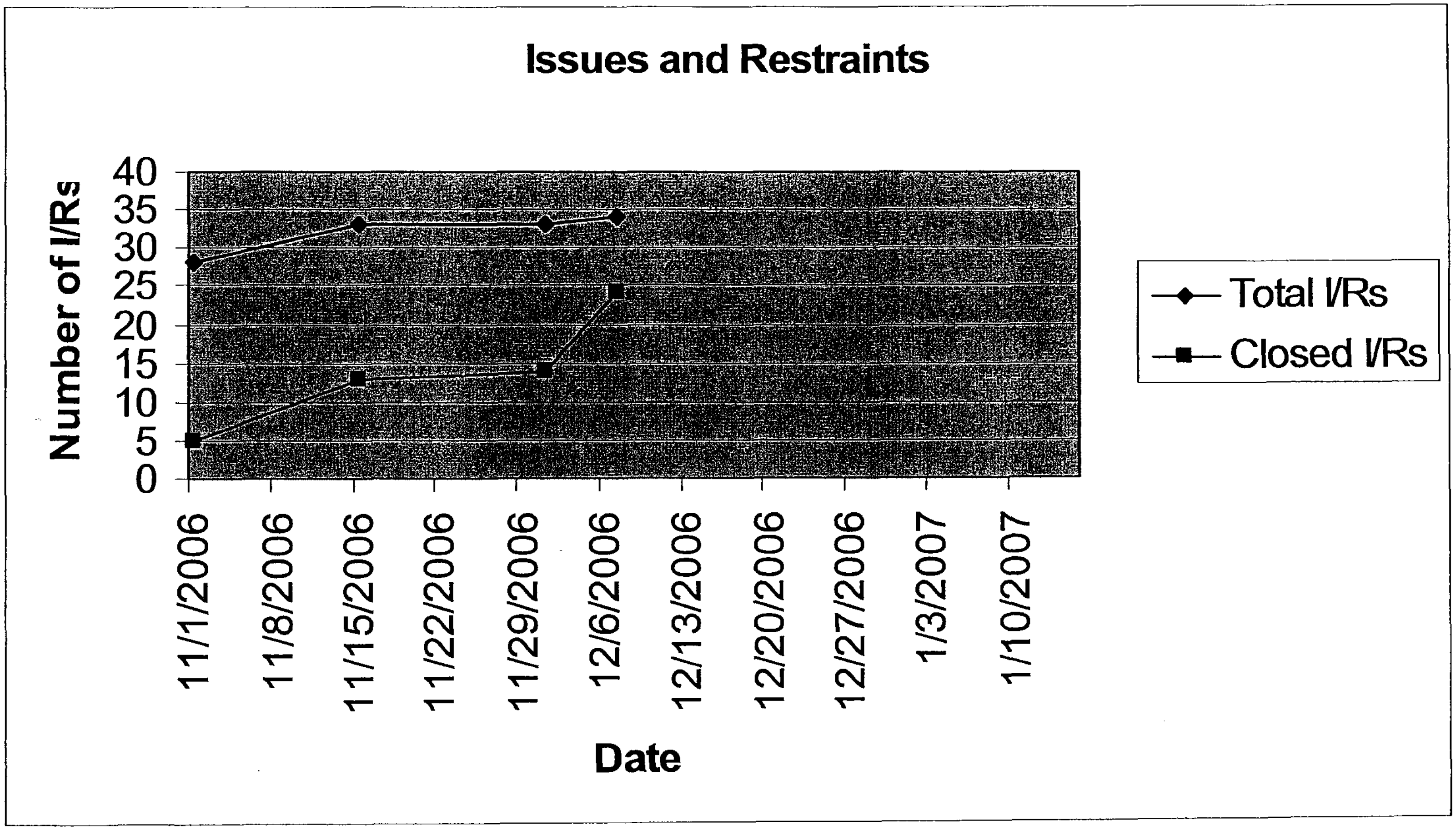
Key Issues

Heat Balance Uncertainty Calculation

- **Initial Heat Balance uncertainty input calculation schedule does not support LAR submittal date.**
  - **Need input uncertainty calculations**
    - **MS Temp, MS Press, FW Temp, FW Press, LD Flow, MU Flow, MU Temp, RCS Tcold**
- **Schedule to be resolved by 01/04/07.**

Issues and Restraints Status



Key Issues

Symbol	Description	Units	Preparer	Draft Calc Date	Reviewer	Complete Internal Review	CR3 Review Complete	Final Doc Sign Off	Comments
WFW	Feedwater Flow Rate	lbm/hr	Caldon						
TS	Steam Temperature	F	Wagner	3/15/07					
PS	Steam Pressure	psia	Wagner	2/04/07					
TFW	Feedwater Temperature	F	Caldon						
PFW	Feedwater Pressure	psia	Wagner	12/22/06					
WMU	Makeup Flow Rate	lbm/hr							
TMU	Makeup Temperature	F							
PMU	Makeup Pressure	psia							
WLD	Letdown Flow Rate	lbm/hr							
TLD	Cold Leg Temperature	F							
PLD	Letdown Pressure	psia							
QRCP	RCP Power	Btu/hr	Weimer						Included in Uncertainty Calc
QLOSS	Ambient Heat Loss	Btu/hr	Weimer?						
	Heat Balance Uncertainty Calc		Weimer						

Issues and Restraints Status

Item #	Issue	Source	Applicable System or Document	Responsible Individual	Scheduled Completion Date	Actual Completion Date	Comments
1	Turbine Lube Oil Temperature	SE01-0154 I/R #1	SC	N/A	N/A	10/20/2006	**Resolved**: Cleaned Lube oil cooler
2	Hydrogen Cooler Temperature	SE01-0154 I/R #2		N/A	N/A		**Resolved**: Simultaneously with I/R 14
3	Iso-Phase Bus Duct Humidity	SE01-0154 I/R #3					**Resolved**: Humidity sensor replaced. Not an MUR issue
4	Condenser	SE01-0154 I/R #4	CD	Ted Williams	N/A	12/6/2006	Resolved: Condenser dT is 25F. Will need to reduce to approximately 85%RTP and monitor dT across water box
5	CDHE-1/2/3 Above Design Parameters	SE01-0154 I/R #5				11/15/2006	**Resolved**: EC 49524
6	FWHE-2/3 Above Design Parameters	SE01-0154 I/R #6				11/15/2006	**Resolved**: EC 49524
7	Condensate Demineralizers Above Design Flow	SE01-0154 I/R #7				11/14/2006	**Resolved**: DBD was changed to reflect 2619.4 MWt flows
8	PEPSE Model Secondary Plant Performance Evaluation	SE01-0154 I/R #8	All	Steve Claunch		11/2/2006	**Resolved**: Scientech upgraded model in 2002

Issues and Restraints Status

Item #	Issue	Source	Applicable System or Document	Responsible Individual	Scheduled Completion Date	Actual Completion Date	Comments
9	Step-up Transformer Loading	SE01-0154 I/R 9		N/A	10/31/2006	11/1/2006	Resolved: EC-64327
10	RCP Load Transfer	SE01-0154 I/R 10		N/A		9/18/2006	Resolved: EC-61858 RCPs restored to Unit Aux Bus
11	BAST Volume	SE01-0154 I/R 11		Virg Esquillo	1/26/2006		
12	HDVs Above Design Pressure/Flow	SE01-0154 I/R 12		Don Ryan	1/15/2006		
13	Discharge Temperature Increase	SE01-0154 I/R 13				10/23/2006	Resolved: Helper cooling towers
14	SC System Performance	SE01-0154 I/R 14				11/15/2006	Resolved: Nitrogen injection is not MUR related issue
15	ICS EC - Who will do this	System Interviews	IC	Ted Williams		11/9/2006	Resolved: PE will do EC for TRICON and AREVA will do ICS.
16	Main Steam Thermowell/ Jet Impingement Shield	Initial Walkdown	NNI	G. Hildebrandt	completed	12/6/2006	Resolved: EC 65629 will locate RTD in accessible location.
17	HELB Analysis Outside Containment	System Interviews	HELB Program	Scott/Williams	In Progress		Define Responsibility
18	FIDMS installation in summer - fall 2007	Discussions Lynchburg and Raleigh Fuels	Plant computer	Larry Creasy		11/1/2006	Resolved: FIDMS 1.8 to be installed in the spring 2007.

Issues and Restraints Status

Item #	Issue	Source	Applicable System or Document	Responsible Individual	Scheduled Completion Date	Actual Completion Date	Comments
19	Fuel Schedule does not support Nov. 2007 SER	Fuel Discussions	COLR	Creasy/Morgan	10/31/2006	11/15/2006	**Resolved:** AREVA Fuels will work to an agreed upon revised schedule
20	Early LAR Submittal	Progress Energy Licensing Recommendation	LAR	Kane/Wilson	11/17/2006	Complete	**Resolved:** Level I schedule developed that shows Mid May submittal date. Earlier date will be dependent upon accelerated reviews of one week instead of two which may reduce submittal date to End of April.
21	Appendix R analysis performed at 2568 MWt	System Interviews	Appendix R	Scott/Seals		Complete	**Resolved:** AREVA to perform scoping evaluation to ensure 72 hour comp time met
22	FAC analysis limiting at feedwater to fdw. Ring header	System Interviews	FAC / FW	Scott		Complete	**Resolved:** AREVA performing FAC analysis. Identified in Level I schedule
23	Heat Balance Pump Heat	General Discussions	CTP	Weiner		Complete	**Resolved:** 21MWth will be used in HB uncertainty analysis based on BHP.
24	Progress Energy Support Team	N/A	N/A	D. Roderick	11/20/2006	11/6/2006	**Resolved:** Ted Williams is PM
25	Verify Progress Energy will do installation	General Discussions	ECs	D. Roderick	10/16/2006	10/16/2006	**Resolved:** PE will install
26	Seismic Qualification of Caldon Equipment	PO Review	Caldon PO	B. Devendorf	10/29/2006	10/19/2006	**Resolved:** Equipment will be qualified
27	Review and Approval Process for Documents	General Discussions	All Documents	Ken Wilson	1/15/2006	12/6/2006	**Resolved:** AREVA provides mark ups and final AREVA docs. CR# performs Owner acceptance reviews and records submittal

Issues and Restraints Status

Item #	Issue	Source	Applicable System or Document	Responsible Individual	Scheduled Completion Date	Actual Completion Date	Comments
28	T/G Governor Valve Programming Curve	System Interviews	TG				Modify if required
29	Condensate Pump Resonate Speed	System Interviews		Don Ryan	12/12/2006		
30	Feedwater Start-up valve stroke margin	System Interviews		Don Ryan	12/12/2006		
31	Condenser backpressure not evaluated in 2001	System Interviews		Don Ryan	12/12/2006		
32	ADV and TBV Margin Acceptability	System Interviews		Don Ryan	12/12/2006		
33	RPS High Flux Trip - How to handle	Davis Besse	RPS	Wilson	12/8/2006		Choose Path
34	H. B. Uncertainty and Daughter Calc Schedule	Schedule Dev.	HB Calc	Bondurant	1/4/2007		Add resources and resolve schedule issues

Project Status

CONFIDENTIAL

Overall Status

> Approximately [redacted]

> System Evaluations are approximately 75% complete.

> Fuels is approximately 70% complete for LAR.

> EC's are approximately 20% complete.

> Schedule is being met.

Work Scope Defined

Schedule Defined

Deliverables Defined

Metrics Defined

Payment Schedule Defined

Proposal Presented to CR3

PEF DOCUMENTS RESPONSIVE TO
CITIZEN'S FIRST REQUEST TO PRODUCE
DOCUMENTS TO
PROGRESS ENERGY FLORIDA, INC. (NO. 2)

DOCUMENTS IDENTIFIED WITH
BATES NOS.
PEF-CR3-0609 THROUGH PEF-CR3-0618
HAVE BEEN DESIGNATED CONFIDENTIAL
IN ACCORDANCE WITH
PEF'S FIRST REQUEST FOR CONFIDENTIAL
CLASSIFICATION

Summary

CONFIDENTIAL

- ❖ Level 1 schedule defined.
- ❖ Initial deliverables are proceeding.
- ❖ Heat Balance Uncertainty and emergent supporting calculations are critical path.
- ❖ [redacted] proposal presented.
- ❖ Metrics resolved.

AREVA

Crystal River 3
Measurement Uncertainty Recapture (MUR) Power Uprate

AREVA Project Team

September 13, 2006

AREVA

Agenda

- Introduction and Background - Roderick
- Caldon Measurement Uncertainty Recapture (MUR) Project Scope– Project Team
  - ✓ Introduction – Franch
  - ✓ EC - Installation – Devendorf
  - ✓ License Support
    - Fuels Evaluations – Creasy
    - Systems and Documentation – Kane
    - Safety Analysis – Scott
    - LAR - Scott
- MUR Preliminary Schedule - Kane
- Preliminary Support Needs from CR - Scott
- Summary - Franch

AREVA

Introduction - Franch

> **Purpose**

- **To demonstrate a success path for delivering an MUR total installation solution for CR3**

> **Process**

- **Review necessary scopes, schedule and potential division of responsibilities for activities to support delivery of the solution.**
  - **Installation in fall '07**
  - **Uprate implementation in cycle 16**

> **Payoff**

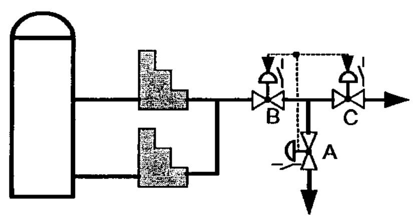
- **A better understanding of what needs to be accomplished to support the fall '07 installation and implementation in cycle 16!**

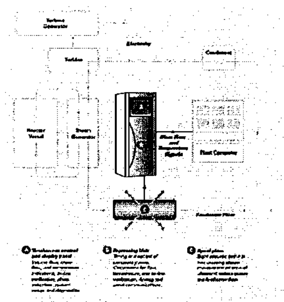
Upon completion CR3 will realize ~ 1.5% Power Uprate

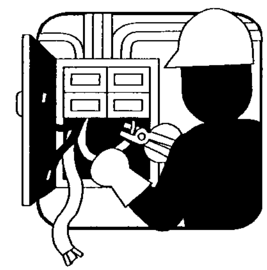
PEF-CR3-0623

Successful Integration is …

Progress Energy/AREVA | Caldon | Progress Energy/AREVA







Mechanical and I&C System | LEFM | Installation / Commissioning

- Licensing
- EC's
- Calculations/Analysis
- Procedures
- Work Packages
- Equipment Installation
- Start-up Testing

"Understanding and managing the Interfaces between Partners"

PEF-CR3-0624

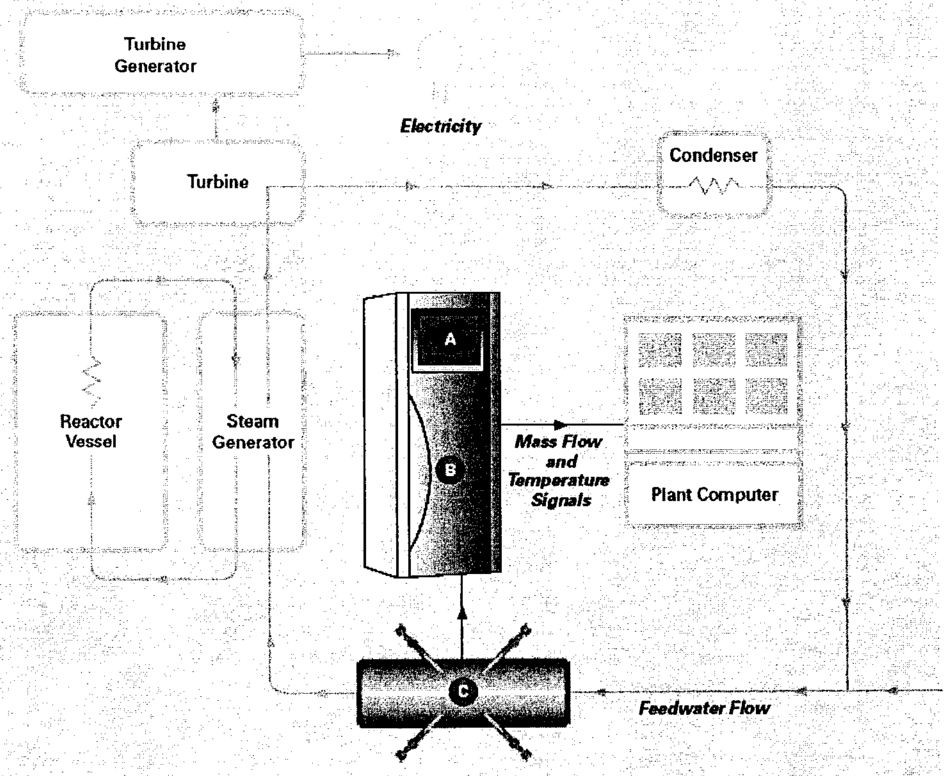
AREVA

EC - Installation – Devendorf

- Caldon Leading Edge Flow Meter (LEFM)
  - ✓ Measurement Uncertainty Recapture (MUR)
  - ✓ System Overview and Installation
  - ✓ Unit Operation
- AREVA Engineering Support of LEFM
  - ✓ Modification Development
  - ✓ Division of Responsibility
  - ✓ Schedule
- LEFM Implementation
  - ✓ Installation
  - ✓ Testing
  - ✓ Commissioning

PEF-CR3-0625

EC - Installation – Devendorf

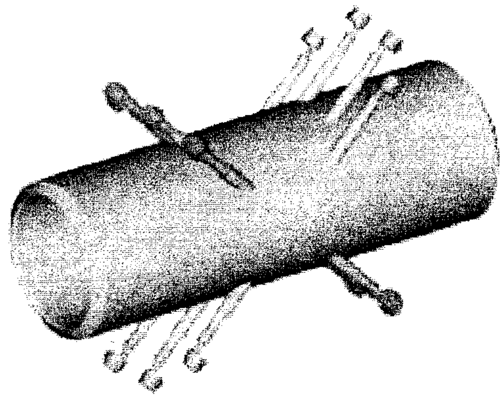


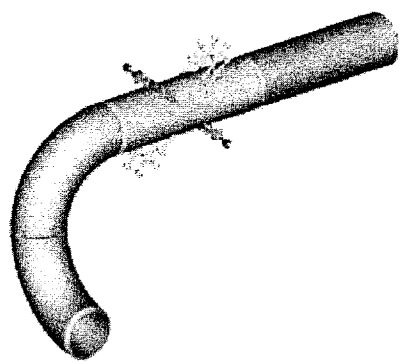
A Touchscreen control and display panel Volume flow, mass flow, and temperature indications, on-line verification, alarm indicators, system setup, and diagnostics.

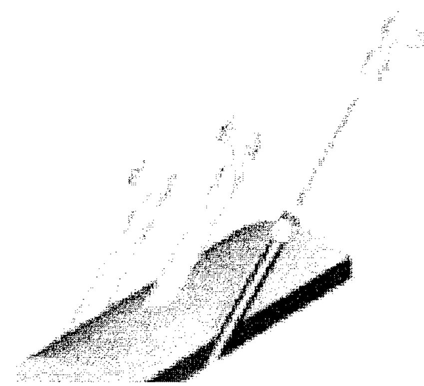
B Processing Unit Timing and control of ultrasonic pulses. Calculations for flow, temperature, and on-line verification. Analog and serial communications.

C Spool piece Eight acoustic paths in two crossing planes measure transit time of ultrasonic pulses across the feedwater flow.

EC - Installation – Devendorf







PEF-CR3-0627

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Fuels Evaluations – Creasy

- **Thermal-Hydraulic Analysis**
  - ✓ New Statistical Core Design (SCD) DNB Analyses to address the higher nominal core power level
    - DNB core safety limits
    - Limiting Condition I/II transients and Ejected Rod
    - Steady-state Maximum Allowable Peaking (MAPs)
    - Operating Limit MAPs
    - Other MAPs (Dropped rod, misaligned rod)
  - ✓ Implement the Statistical Fuel Assembly Hold Down Methodology at the new RTP for the "at power" conditions
  - ✓ Evaluations and Dispositions – to address analyses/documentation for CR-3 that currently assume operation at 2568 MWt
    - verify continued applicability and adequacy of margin to accommodate the higher nominal power level in the cycle-specific reload analysis
    - if disposition is not possible, perform analysis (e.g., burnup limit for end-of-life fuel rod internal pressure)
  - ✓ AOA Evaluation
- **Fuel Mechanical Analyses**
  - ✓ evaluate capability of stress/strain, rod growth, creep collapse, shoulder gap, fatigue
  - ✓ validate adequacy of Mech. Maneuv. Recommendations and assembly structure

Fuels Evaluations – Creasy

- **Fuel Cycle Design Model**
  - ✓ update T/H model and critical reactivity benchmark
  - ✓ cycle 15 design will be basis for comparisons between current & uprate RTP
  - ✓ Cycle 16 FFCD will be done at power uprate for licensing analysis (PFCD already performed at 2568 MWt)
- **Nuclear Analysis**
  - ✓ review bases document, calculate key physics parameters for impact on safety analysis
  - ✓ re-evaluate boron requirements for boric acid reduction (heat trace de-termination project)
- **Maneuvering Analysis**
  - ✓ review error adjustment equation for imbalance and tilt, review methods and peaking factors
  - ✓ evaluate PSC resolutions, generic EOC Tavg study, shutdown flexibility

PEF-CR3-0629

Fuels Evaluations – Creasy

- **All tasks**
  - ✓ review Tech Specs and UFSAR; prepare engineering summary report input
- **Most Reload Scheduled Deliverables to slip approx. 3 months – but avoids redoing reload licensing later at uprated power level**
- **Reload Report would be based on power uprate & contain COLR figures for both power levels**
- **PTM and OLC DBU basis for cycle 16 TBD – may require mid-cycle update to the uprated RTP**

Systems and Documentation - Kane

Approach

- Define Systems, Programs and Documentation to be Evaluated
- Use Evaluations Made for 2568 MWt Uprate as Start
- Perform Interviews with Cognizant Personnel
- Perform Gap Analysis
- Document Results

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Systems and Documentation - Kane

- Systems Evaluation
  - ✓ NSSS Fluid Systems
  - ✓ NSSS I&C and Electrical
  - ✓ NSSS Structural
  - ✓ BOP Fluid Systems
  - ✓ BOP I&C and Electrical
  - ✓ BOP Structural
  - ✓ HVAC

Systems and Documentation – Kane

- **Program Evaluations**
  - ✓ Erosion Corrosion
  - ✓ Inservice Test Program
  - ✓ Inservice Inspection Program
  - ✓ Other Programs

Systems and Documentation - Kane

- Calculations
  - ✓ New Plant Operating Conditions
  - ✓ Heat Balance Uncertainty
  - ✓ Decay Heat Cooldown
  - ✓ Other Calculations

Systems and Documentation - Kane

- Procedure Modifications
- Engineering Design Basis Documents
- Simulator Updates

PEF-CR3-0635

Safety Analysis – Scott

- Review Previous Submittals
- Review AMSAC, ATWS, DSS
- Review Chapter 14 of USAR
- Review Technical Specifications
- Update ATWS
- Update ARTS Arming Setpoint, AMSAC, MSSV Operability
- Prepare USAR/TS Updates
- LOCA M&E Release Data
- Update Containment Response
- Prepare MUR Summary Report Input

License Amendment Activities – Scott

- New Operating Conditions
- Heat Balance Uncertainty
- Safety Analysis Evaluations
- NSSS/BOP Components
- System/Program Evaluations
- Summary Report and LAR Input
- LAR preparation Support to NRC RIS 2002-003
- LAR Review Support (Interdiscipline Review, PRC, Corporate Review)
- NRC RAI Response Support

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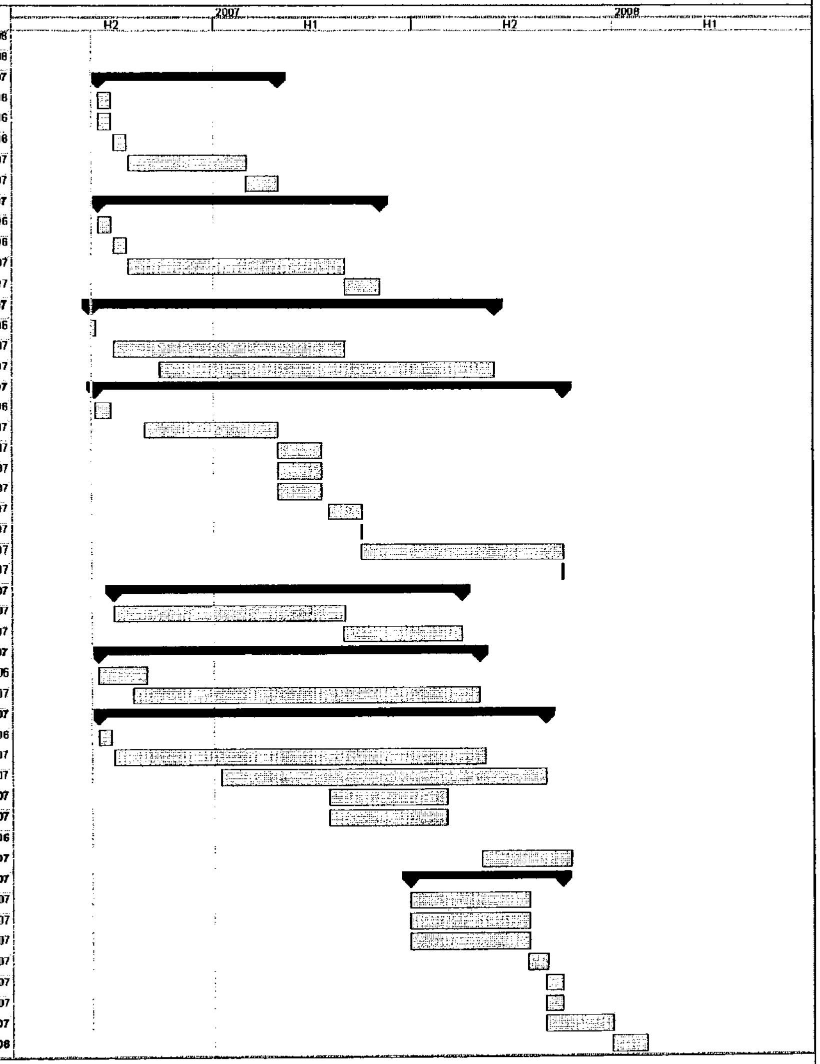
MUR Preliminary Schedule - Kane

MUR Power Uprate Draft Level 1 Schedule

ID		Task Name	Start	Finish
1		Project Plan Preparation	Mon 9/11/06	Mon 9/11/06
2		Kick Off Meeting	Mon 9/11/06	Mon 9/11/06
3		**System Evaluations**	**Mon 9/18/06**	**Wed 2/28/07**
4		Review Issues/Restraints Letter SE01-0154	Mon 9/18/06	Fri 9/29/06
5		Compare System List with Previous Evluations	Mon 9/18/06	Fri 9/29/06
6		Perform System Interviews	Mon 10/2/06	Fri 10/13/06
7		Prepare System Evaluation Report	Mon 10/16/06	Wed 1/31/07
8		CR Owner Acceptance	Wed 1/31/07	Wed 2/28/07
9		**Program Evaluations**	**Mon 9/18/06**	**Fri 6/1/07**
10		Program Selection	Mon 9/18/06	Fri 9/29/06
11		Interviews	Mon 10/2/06	Fri 10/13/06
12		Program Reviews	Mon 10/16/06	Mon 4/30/07
13		Owner Acceptance	Tue 5/1/07	Fri 6/1/07
14		**Fuel Analysis**	**Mon 9/11/06**	**Fri 9/14/07**
15		Pick Power Level	Mon 9/11/06	Fri 9/15/06
16		FOAK	Mon 10/2/06	Mon 4/30/07
17		Reload PT Uprate	Mon 11/13/06	Fri 9/14/07
18		**LAR Preparation**	**Fri 9/15/06**	**Fri 11/16/07**
19		Request NRC Docket Position	Fri 9/15/06	Fri 9/29/06
20		Prepare LAR	Tue 10/31/06	Wed 2/28/07
21		Prepare Technical Specification Changes	Thu 3/1/07	Mon 4/9/07
22		Prepare FSAR Changes (What is required date?)	Thu 3/1/07	Mon 4/9/07
23		Licensing Information to Licensing for Review (LAR, TS, FSAR Changes)	Thu 3/1/07	Mon 4/9/07
24		CR3 LAR Inhouse Review	Mon 4/16/07	Wed 5/16/07
25		Submit LAR for MUR to NRC	Wed 5/16/07	Wed 5/16/07
26		Respond to NRC RAIs	Wed 5/16/07	Fri 11/16/07
27		NRC SER	Fri 11/16/07	Fri 11/16/07
28		**Engineering Change Package Preparation**	**Mon 10/2/06**	**Wed 8/15/07**
29		Prepare EC for Caldon	Mon 10/2/06	Mon 4/30/07
30		Owner Acceptance	Mon 4/30/07	Wed 8/15/07
31		**Procurement**	**Mon 9/18/06**	**Fri 8/31/07**
32		Prepare Caldon PO	Mon 9/18/06	Tue 10/31/06
33		Procure Caldon	Fri 10/20/06	Fri 8/31/07
34		**Calculation Package Modifications Resulting From MUR**	**Mon 9/18/06**	**Wed 10/31/07**
35		Review List of Calc. Pkg. Changes from Previous Uprate	Mon 9/18/06	Fri 9/29/06
36		Prepare Calculation Package Modifications	Mon 10/2/06	Wed 9/5/07
37		Owner's Acceptance	Mon 1/8/07	Wed 10/31/07
38		**Procedure Modifications**	**Mon 4/16/07**	**Wed 8/1/07**
39		**EDBD Modifications**	**Mon 4/16/07**	**Wed 8/1/07**
40		**Simulator Modifications**	**Mon 9/11/06**	**Mon 9/11/06**
41		**Operator Training**	**Mon 9/3/07**	**Fri 11/23/07**
42		**Construction and Installation (By Construction Company)**	**Fri 6/29/07**	**Thu 11/15/07**
43		Prepare Cutting and Welding Work Packages for Caldon Equipment	Fri 8/29/07	Mon 10/15/07
44		Prepare Cable Pull Work Packages	Fri 8/29/07	Mon 10/15/07
45		Prepare Termination Work Packages at Computer and Caldon Eq	Fri 8/29/07	Mon 10/15/07
46		Install Cables (Non Outage Activity)	Mon 10/15/07	Thu 11/1/07
47		Cut Feedwater Lines and Weld in Caldon Equipment	Wed 10/31/07	Thu 11/15/07
48		Terminate Cabling	Wed 10/31/07	Thu 11/15/07
49		**Startup and Testing**	**Wed 10/31/07**	**Mon 12/31/07**
50		**Closeout**	**Mon 12/31/07**	**Thu 1/31/08**



Page 1

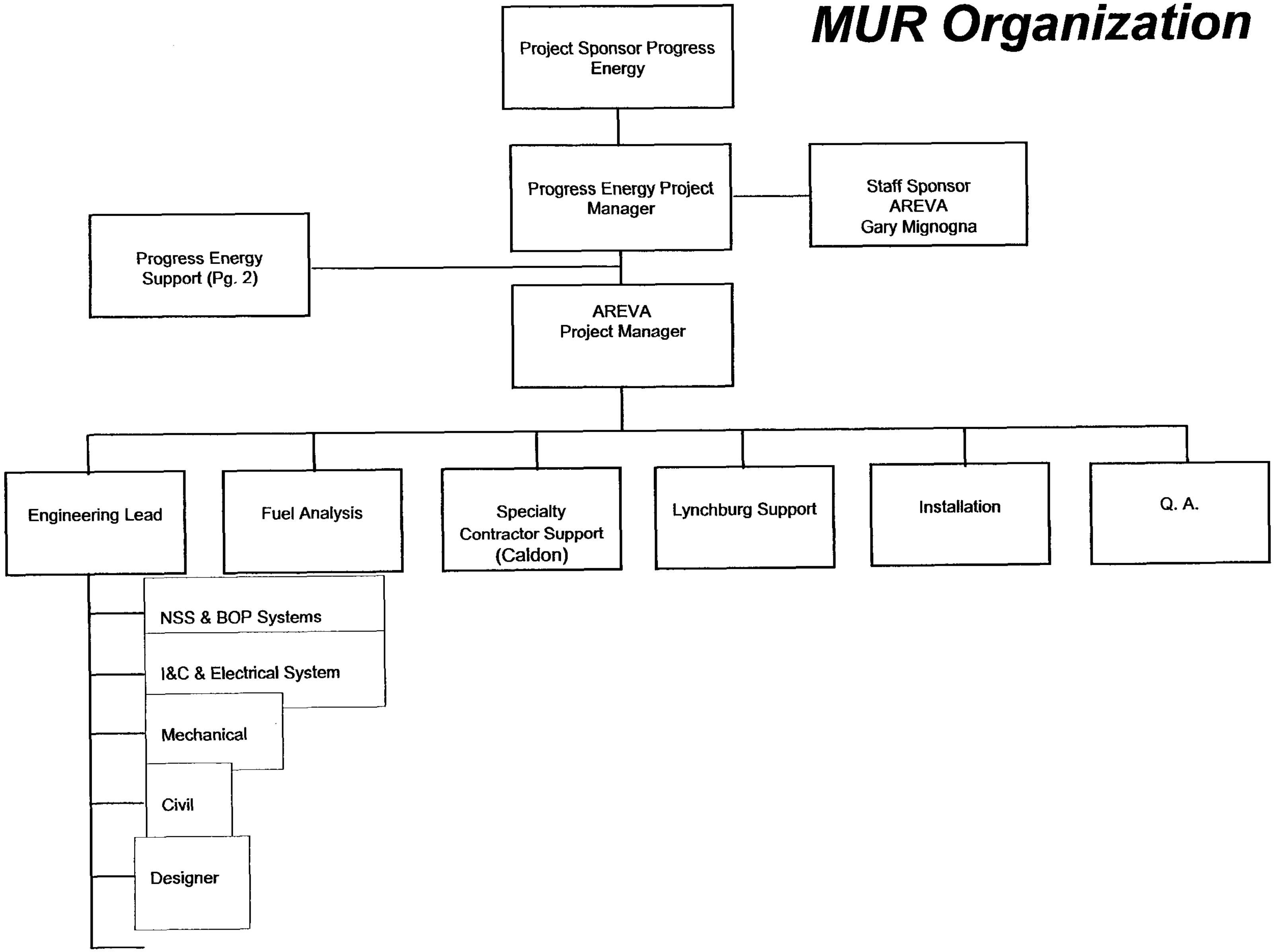
PEF-CR3-0638

Preliminary Support Needs from CR - Scott

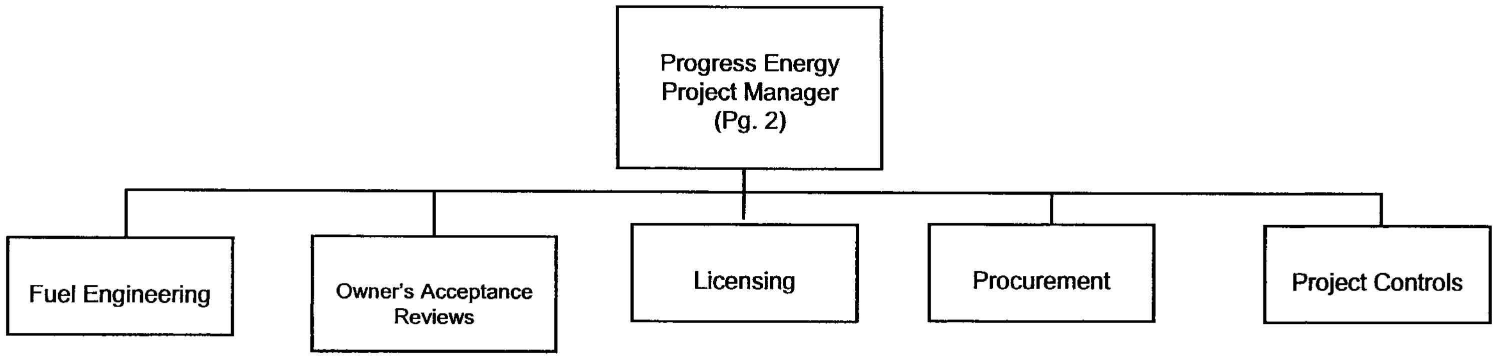
- Project Management
- Procurement Support
- Modification Support
  - ✓ Software V&V
  - ✓ Procedure Changes
  - ✓ NAS Modification Request
- System Engineer/Design Engineer Interview Support
- System and Program Evaluation Review and Acceptance
- Grid Stability Analysis
- Environmental Evaluation
- LAR Preparation, Review and Submittal
  - ✓ Licensing Lead
  - ✓ Progress Energy Review and Approval
- RAI Response Preparation, Review and Submittal

AREVA

MUR Organization



MUR Organization



SUMMARY - Franch

- Schedule Is critical but achievable based on October 1st, 2006 Start Date
- Initial Detailed Planning and Coordination is Crucial to Project Success
- AREVA is Committed to Provide Resources to Support Accelerated Schedule

WE will apply our lessons learned from Davis Besse to CR3 which ensures schedule and budget performance

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- Purpose- The purpose of this meeting is to outline information and approach going forward on the Crystal River 3 Power Uprate, supporting power increase objectives

- Process- As a team, we will review the status of the MUR and EPU

- Payoff – Clear understanding of the MUR and EPU Status supporting CR 3 power increase objectives.

PEF-CR3-0644

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AGENDA

Introduction	Gary Mignogna	1:00-1:15pm
MUR Update	Dallas Scott/ Tom Bilger	1:15-2:00pm

NON-RESPONSIVE

PEF-CR3-0645

AREVA

AGENDA

CR3 MUR STATUS 3-13-07

- **Project Status**
  - MUR Project Schedule Overview
  - Deliverable Status
  - Issues and Restraints
  - Key Issues
    - RPS High Flux Trip Setpoint
    - Surface for M&E Calculation
    - Heat Balance Uncertainty Calculation
    - FIDMS vs AULD
- **Contract Status**

Summary

AREVA

Project Status

- **MUR Schedule Overview**
  - Schedule Milestones within targets
  - Schedule handout showing remaining tasks provided for later review
  - LAR DRAFT submitted and CR3 comments received
  - LOCA M/E Containment Response complete
  - Additions to HB Uncertainties (AULD and FIDMS)

MUR is on track for NRC Submittal in April

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Deliverable Status

Deliverable	Document Number	Draft Due Date	Draft Issue Date	Review Date Requested	CR3 Comments Issued	Rev. 000 Issued to CR3
New Operating Conditions	32-5012972-001	11/13/06	11/13/06	12/8/06	12/6/06	12/19/06
Updated PEPSE	12-9036301-000	11/22/06	11/29/06	12/8/06	12/6/06 12/13/06	12/31/07
NSSS I&C System Evaluation	51-9035360-000	12/7/06	12/1/06	12/15/06	12/18/06	2/6/07
EC's Closed Since 2002	51-9036116-000	12/6/06	12/6/06	12/20/06	1/4/07	1/19/07
Pipe Spec Evaluation	51-9037456-001		12/13/06	1/3/07	12/18/06	1/26/07
Electrical System Evaluations	51-9037444-000	12/21/06	12/19/06	1/9/07	1/8/07	1/26/07
NSSS Fluid System Evaluation	51-9041016-000	12/22/06	12/22/06	1/11/07	1/19/07	2/2/07
BOP Fluid Systems Evaluation	51-9036484-000	12/29/06	12/8/06	1/12/07	1/31/07	2/8/07
HVAC Systems Evaluation	51-9036250-000	12/29/06	12/12/06	1/5/07	1/19/07	2/21/07
Fluence Evaluation	51-9038958-000	1/18/07	12/31/06	1/12/07	1/12/07	1/26/07
ChecWorks Model Update (FAC)	51-9039008-000		1/8/07	1/22/07	1/22/07	2/15/07
NSSS Structural Evaluation	51-9036246-000	1/19/07	1/19/07	2/2/07	2/13/07	2/14/07
NSSS Materials Evaluation	51-9040378-000	1/19/07	1/19/07	2/2/07	2/2/07	3/7/07
App R Cooldown	32-9038072-000		1/15/07	1/29/07	1/29/07	1/29/07
OTSG FIV Evaluation	51-5000475-002	1/19/07	1/22/07	2/5/07	2/1/07	2/20/07
LOCA M&E and Cont. Response	32-9040399-000	2/26/07				2/15/07

Deliverable Status

Deliverable	Document Number	Draft Due Date	Draft Issue Date	Review Date Requested	CR3 Comments Issued	Rev. 000 Issued to CR3
FW Pressure Uncertainty	32-9041510-000		3/1/07	3/8/07		
[redacted]						
Letdown Flow Uncertainty	32-9049930-000		3/6/07	3/14/07		
[redacted]						
BOP Piping and Supports	51-9039310-000		2/28/07	3/7/07		
[redacted]						
Makeup Temperature Uncertainty	32-9043792-000		3/6/07	3/14/07		
RC Tcold Uncertainty	32-9043797-000		3/6/07	3/14/07		

CR3 has next Action	[redacted]	Task Complete

❖ RPS High Flux Trip Setpoint

- Operate at current trip setpoint of 104.9%. Have to reduce setpoint to 103.3% and reduce core power to 98.4% on loss of LEFM in CP-0500.

Crystal River 3 NRC Meeting TBD

Heat Balance Uncertainty Calculation

- Input calculations drafted and under review for
  - MS Temp, MS Press, FW Press, LD Flow, MUT Temp, T-cold
- Heat Balance Uncertainty Calculation DRAFTED pending final input calcs
- Discussion of random and bias uncertainties and impact on overall uncertainty in progress.

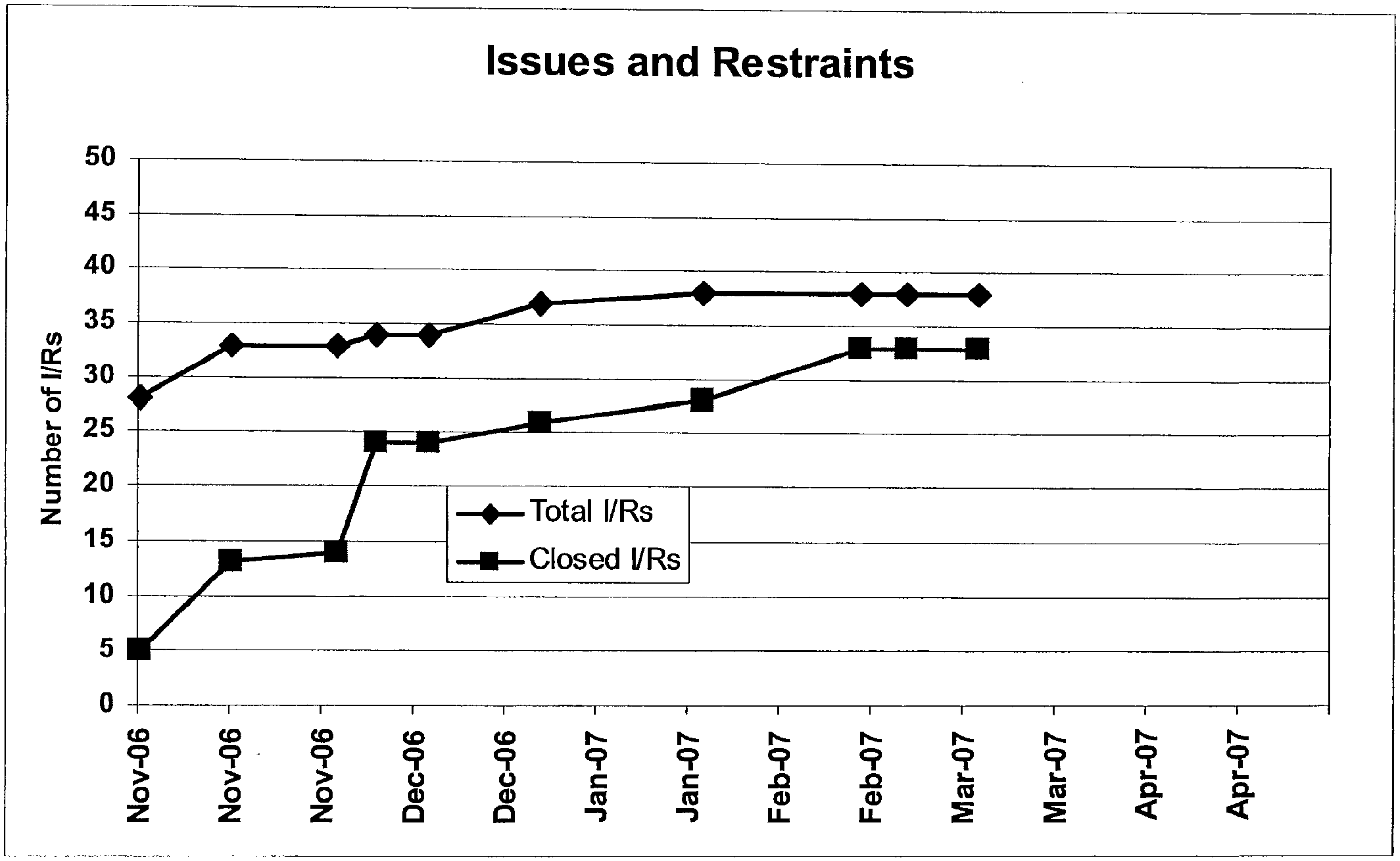
Schedule resolved to support LAR submittal to NRC.

AULD Heat Balance vs FIDMS Heat Balance

- The AULD HB uses constants to calculate core power. The FIDMS HB does not use constants.
- The use of constants produces a less accurate HB
- AULD HB to be reprogrammed by CR3 to mimic FIDMS HB

FIDMS and AULD Heat Balances to be matched

Issues and Restraints Status



Issues and Restraints Status

Item #	Issue	Source	Applicable System or Document	Responsible Individual	Scheduled Completion Date	Actual Completion Date	Comments
9	Step-up Transfmer Loading	SE01-0154 I/R 9		N/A	10/31/2006	11/1/2006	Resolved: EC-64327
10	RCP Load Transfer	SE01-0154 I/R 10		N/A		9/13/2006	Resolved: EC-61858 RCPs restored to Unit Aux Bus
11	BAST Volume	SE01-0154 I/R 11		Lewis Wells	1/26/2007	2/19/2007	Resolved: BAST volume part of normal core reload analysis.
12	HDVs Above Design Pressure/Flow	SE01-0154 I/R 12		Don Ryan	4/15/2007		Proposal sent to CR3 (Ted Williams) on 2-23-07 need to validate conclusion and approve proposal.
13	Discharge Temp Increase	SE01-0154 I/R 13				10/23/2006	Resolved: Helper cooling towers
14	SC System Performance	SE01-0154 I/R 14				11/15/2006	Resolved: Nitrogen injestion is not MUR related issue.
15	ICS EC - Who will do this	System Interviews	IC	Ted Williams		11/9/2006	Resolved: PE will do EC for TRICON and AREVA will do ICS.
16	Main Steam Thermowell/ Jet Impngement Shield	Initial Walkdown	NNI	G. Hildebrandt	completed	12/6/2006	Resolved: EC 65629 will locate RTD in accessable location
17	HELB Analysis Outside RB	System Interviews	HELB Program	Scott Williams	3/15/2007		AREVA work started 2/21
18	FIDMS installation in summer - fall 2007	Discussions Lynchburg and Raleigh Fuels	Plant computer	Larry Creasy		11/1/2006	Resolved: FIDMS 1.8 installed spring 2007.

Issues and Restraints Status

Item #	Issue	Source	Applicable System or Document	Responsible Individual	Scheduled Completion Date	Actual Completion Date	Comments
35	Feedwater line may need re-analysis	General Discussions	Caldon EC	Loseke	5/23/2007		Proposal sent to CR3 2/9/07, updated proposal sent to CR3 2/20/07. Verbal authorization to start.
36	Does existing heat balance uncertainty calc need updating?	General Discussions		Williams	1/8/2007	2/5/2007	**Resolved:** Existing calc is satisfactory.
37	Several outliers exists on the line list	Line list review	BOP	George Englert	4/6/2007		MUR operating conditions were compared to the allowable operating conditions for each piping segment. If the design operating conditions were revised to increase the appropriate design operation conditions as discussed in Section 5.1 of AREVA document 51-903465-000, these piping segments would no longer be outliers for the MUR uprate. An EC will need to be performed by CR3 to revise the Piping Specification.
38	AULD HB must agree with HB in FDMIS	General Discussions		Weimer / Chang / Mike Lord	4/6/2007		Revise the instrument uncertainties and modify the AULD. Proposal due from AREVA for additional calcs.

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Contract Status

- MUR Contract 101659-61
- HELB Contract 101659-61 Amendment 03
- FW Line Analysis Contract 101659-61 Amendment 04 Proposed
- Heater Drain Valves Contract 101659-61 Amendment 05 Proposed

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Project Summary

MUR Overall Status

- ❖ System Evaluations are complete.
- ❖ Fuels is approximately 90% complete for MUR tasks and on schedule.
- ❖ FIDMS programming is installed in CR3 Test Lab.
- ❖ Caldon EC is 40% complete and on schedule.
- ❖ MUR EC is 30% complete and on schedule.
  - ➢ RPS EC part of MUR EC.
- ❖ ICS EC 15% complete and on schedule.
- ❖ MS P&T EC 30% complete and on schedule.

Project Summary

- Approximately [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

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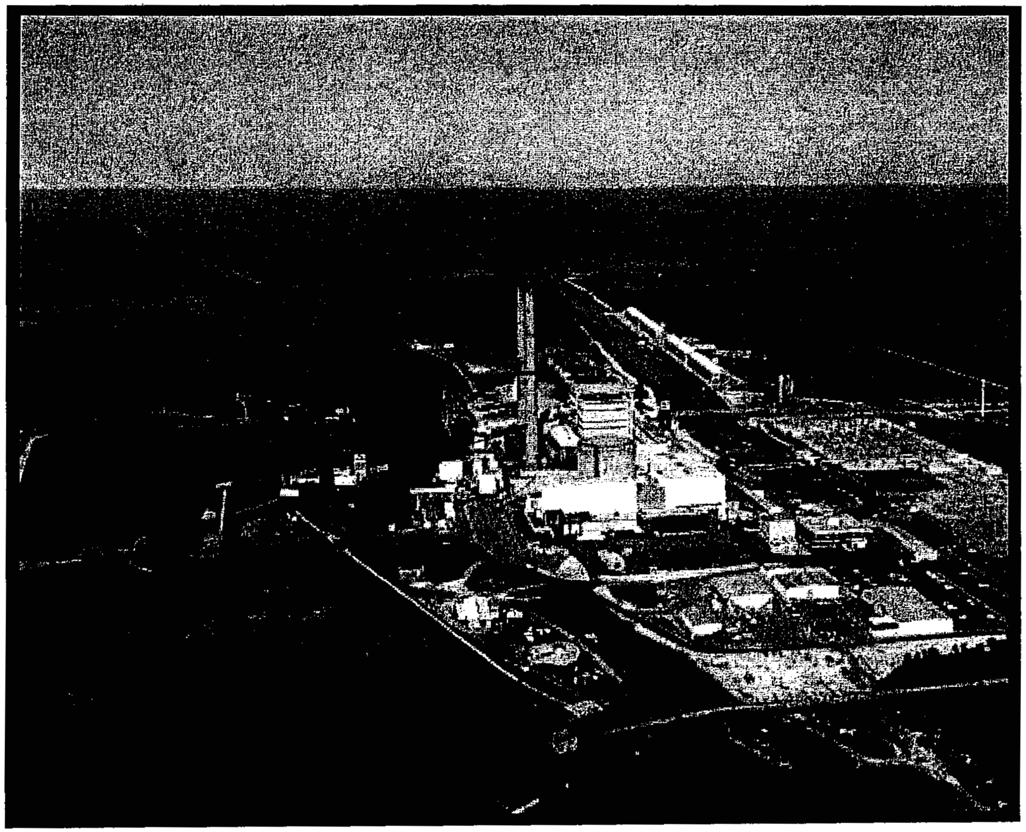
Crystal River 3
Power Uprate
Management Meeting
January 12, 2007



PEF-CR3-0660

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> Purpose- The purpose of this meeting is to outline information and approach going forward on the Crystal River 3 Power Uprate, supporting power increase objectives

> Process- As a team, we will review the status of the MUR, discuss background and experience of AREVA performing BOP and NSSS work in combination and then review the team approach to the EPU

> Payoff – Clear understanding of the team approach to accomplish the MUR and EPU, supporting CR 3 power increase objectives.

PEF-CR3-0661

PEF DOCUMENTS RESPONSIVE TO
CITIZEN'S FIRST REQUEST TO PRODUCE
DOCUMENTS TO
PROGRESS ENERGY FLORIDA, INC. (NO. 2)

DOCUMENTS IDENTIFIED WITH
BATES NOS.
PEF-CR3-0663 THROUGH PEF-CR3-0685
HAVE BEEN DESIGNATED
AS NON-RESPONSIVE TO THIS REQUEST

AREVA

Project Summary

Overall Status

- > System Evaluations are approximately 90% complete and on schedule.
- > Fuels is approximately 80% complete for MUR tasks and on schedule.
- > FIDMS programming on schedule.
- > Caldon EC is 25% complete and on schedule.
- > MUR EC commenced and on schedule.
- > RPS EC commenced and on schedule.
- > ICS EC commenced and on schedule.
- > MS P&T EC commenced and on schedule.
- > Approximately [redacted]

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AGENDA

> Division of Responsibilities
> Price Comparison / Pricing
> Monitoring and Metrics
> Guarantees

CR3 Responsibilities

- NSSS Program Reviews
- BOP Program Reviews
- Tricon EC
- Owner Acceptance of AREVA System Reviews
- LAR Input and Submittal
- Grid Stability
- Final Tech Spec Preparation
- Procedure Revision
- Owner Acceptance of AREVA EDBD Revisions
- Operator Training
- Simulator Revision

AREVA Responsibilities

> NSSS System Review
  - Fluid Systems
  - I&C Systems
  - Structural and Material Systems

> Safety Analysis
  - Review Chapter 14 Analysis
  - Revise LOCA M&E Results
  - Evaluate Safety System Setpoints
  - Revised Containment Pressure and Temperature Response
  - Appendix R Analysis

> Thermal Hydraulics Analysis
  - New Operating Conditions – FSPLIT, VAGIN
  - Heat Balance Uncertainty Calculation
  - Update PEPSE Model
  - Define Simulator Inputs

AREVA Responsibilities

> **Fuel America Activities**
- Nuclear Analysis
- Core Thermal Hydraulics
- Fuel Assembly Mechanical Analysis
- Startup Testing
- Replace NAS with FIDMS

> **BOP Systems**
- Fluid Systems
- Electrical Systems
- Structural Systems
- HVAC Systems
- FAC Analysis

AREVA Responsibilities

> Licensing
- Project Management
- Project Controls
- Scheduler
- Responsible Technical Manager
- NSSS Project Engineer
- BOP Project Engineer
- LAR Enclosure 2 Preparation
- LAR Review
- NRC RAI Response

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ECs to Be Prepared

Engineering Change (EC)	Responsibility
Caldon Instrument Specification EC	AREVA NP
Caldon Instrument EC	AREVA NP
Main Steam RTD EC	AREVA NP
Main Steam Pressure EC	AREVA NP (Not in proposal)
RPS High Flux and AMSAC Trip Setpoint Revision EC	AREVA NP
ICS Function Curve Revision EC	AREVA NP
Plant Computer EC (CP/PICS/SPDS)	AREVA NP
Caldon Software V&V	CR3
Tricon EC	CR3
MUR EC	AREVA NP

Price Comparisons/Pricing

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MUR Price Comparison Summary				
All values in 2007 ($000) except hardware				
4% escalation	2001	2001	2005	2007
	Robinson	Davis Besse	Seabrook	**CR3**
Licensing				
NSSS	[redacted]	[redacted]	[redacted]	[redacted]
Fuel	Unknown	[redacted]	[redacted]	[redacted]
BOP	[redacted]	[redacted]	[redacted]	[redacted]
EC/ Mod Packages	Self performed	included in BOP	[redacted]	[redacted]
Eng Sub Total	[redacted]	[redacted]	[redacted]	[redacted]
Hardware	[redacted]	[redacted]	[redacted]	[redacted]
Installation	[redacted]	[redacted]	[redacted]	[redacted]
Total	[redacted]	[redacted]	[redacted]	[redacted]

* Does not include Fuel or EC cost

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Price Comparisons/Pricing

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NSSS Details

	Robinson		CR-3		
WBS Item	**MHrs.**		**MHrs**	**Cost**	
NSSS					
Operating Conditions	[redacted]		[redacted]	$ [redacted]	
Interviews	[redacted]		[redacted]	$ [redacted]	
NSSS Components	[redacted]		[redacted]	$ [redacted]	RIS 2002-003
NSSS Systems	[redacted]		[redacted]	$ [redacted]	8 systems vs 12 systems
SG FIV	[redacted]		[redacted]	$ [redacted]	No analysis/RIS 2002-003
Safety Analysis	[redacted]		[redacted]	$ [redacted]	RIS 2002-003
I&C Evaluations	[redacted]		[redacted]	$ [redacted]	6 systems vs 12 systems
LAR/USAR input	[redacted]		[redacted]	$ [redacted]	RIS 2002-003
RAI response	[redacted]		[redacted]	$ [redacted]	
Project Management	[redacted]		[redacted]	$ [redacted]	
Sub-total	[redacted]		[redacted]		

Manhour differences due to increased systems resiews and RIS compliance

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Price Comparisons/Pricing

NSSS Delta Scope Robinson and CR3

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	Robinson		CR-3		
WBS Item	**MHrs.**		**MHrs**	**Cost**	
NSSS					
NSSS Programs	[redacted]		[redacted]	$ -	
AFW Analysis	[redacted]		[redacted]	$ -	
Feasibility Study	[redacted]		[redacted]	$ -	Half of Feasibility study effort
DH Calc	[redacted]		[redacted]	$ -	
Sub-total	[redacted]		[redacted]		
Heat Balance Uncertainty	[redacted]		[redacted]	$ [redacted]	
Benchmark and Revise PEPSE	[redacted]		[redacted]	$ [redacted]	
ICS Header Press	[redacted]		[redacted]	$ [redacted]	
Simulator Inputs	[redacted]		[redacted]	$ [redacted]	
NSSS Flu & Rad Analysis	[redacted]		[redacted]	$ [redacted]	
Tech Spec updates	[redacted]		[redacted]	$ [redacted]	
M/E and Containment	[redacted]		[redacted]	$ [redacted]	Performed by Westinghouse
Project Controls	[redacted]		[redacted]	$ [redacted]	
Consultant	[redacted]		[redacted]	$ [redacted]	
Calc/Proc. Update	[redacted]		[redacted]	$ [redacted]	
EDBD's	[redacted]		[redacted]	$ [redacted]	
ARTS	[redacted]		[redacted]	$ [redacted]	
Appendix R Analysis	[redacted]		[redacted]	$ [redacted]	
Sub-total	[redacted]		[redacted]	**$** [redacted]	
					Difference
NSSS Total	[redacted]	[redacted]	[redacted]	$[redacted]	$ [redacted]

* Robinson is a fixed price contract escalated to 2007. CR3 is T&M.

Price Comparisons/Pricing

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NSSS Fuel Details

	Robinson		CR-3		
WBS Item	MHrs.		MHrs	Cost	
Fuel	[redacted]	0 *	[redacted]	$ [redacted]	Compares to 2300 mhrs for HTP cycle 14 for CR3.
FIDMS	[redacted]	$ -	[redacted]	$ [redacted]	
Fuels Total	[redacted]	$ -	[redacted]	$ [redacted]	

* Robinson fuel cost included in separate fuel contract

BOP Details

Area of Interest	Robinson Cost	CR3 MHrs	CR3 Cost
SE/DE Interviews		[redacted]	$ [redacted]
BOP Systems (includes BOP PE)		[redacted]	$ [redacted]
HVAC		[redacted]	$ [redacted]
BOP Electrical and I & C		[redacted]	$ [redacted]
BOP Structural Evaluation		[redacted]	$ [redacted]
Pipe Spec Review		[redacted]	$ [redacted]
Licensing Document Input and Support		[redacted]	$ [redacted]
Calculation update		[redacted]	$ [redacted]
EDBD's		[redacted]	$ [redacted]
FAC Analysis		[redacted]	$ [redacted]
Total BOP Licensing	[redacted]	[redacted]	$ [redacted]

No Robinson manhours available

* Includes T&L and Supervision

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Price Comparisons/Pricing

EC Details

	MHrs.	Cost
Caldon EC*	[redacted]	$ [redacted]
NSSS I&C EC		
RPS/AMSAC	[redacted]	$ [redacted]
ICS Function Curves	[redacted]	$ [redacted]
MS RTD	[redacted]	$ [redacted]
MS Pressure	TBD	AREVA NP
MUR EC	[redacted]	$ [redacted]
Tricon EC		CR3 Scope
RE Caldon	[redacted]	$ [redacted]
RE I&C	[redacted]	$ [redacted]
TOTALS	[redacted]	$[redacted]

* Caldon V&V Justification by CR3

Computer EC in Fuels Scope

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No Comparable Robinson Cost

PEF-CR3-0699

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Price Comparisons/Pricing Summary

> RIS 2002-003 requires more systems to be evaluated and previous NRC approvals documented.

> Significant increased scope for CR3 versus Robinson

> Subcontracted Project results in higher manhour chargeout rate as compared to internal CR3 rates

Monitoring – Budget Performance

Burn Curves

AREVA will develop burn curves for monitoring budget performance. Two curves will be maintained:

(1) Total Project Manhours (Projected and Actual)

(2) Total Project Dollars (Projected and Actual)

> Deliverable Document Progress

AREVA will maintain two curves:

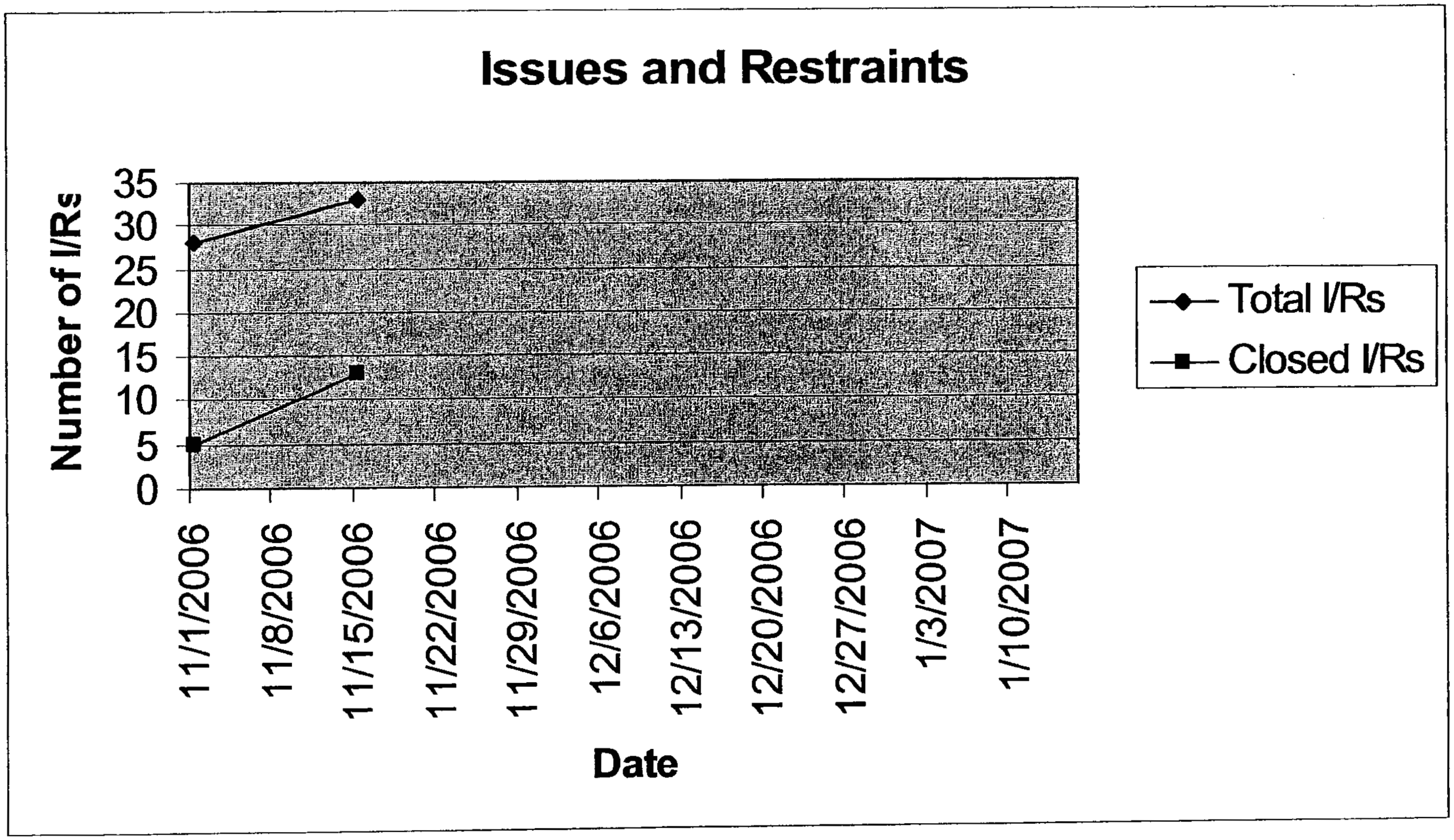
(1) Number of scheduled draft documents vs time

(2) Number of delivered draft documents vs time

> Earned Value

- CPI and SPI Curves will be generated

Issues and Restraints Progress



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Pricing Options

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> **Time and Materials**

[redacted]

> [redacted]

Executive Review Meeting

Measurement Uncertainty Recapture Power Uprate

October 16, 2006

AREVA

Executive Review Meeting
Agenda

> Introductions
> Project Goal
> Key Decisions to be Made
  - Agree on Goal (Fuel schedule to support SER)
  - Recommended Power Level
> Schedule
> Issues and Restraints
> Metrics and Performance Indices
> Action Items
> Summary and Conclusions

MUR Project Goal

> Complete installation of Caldon and support equipment and obtain NRC SER by end of November 2007.

Increase power by 1.6% and 12 MWe to 14 MWe

Key Decision - Project Goal

- > Current evaluation and modification schedule supports an LAR submittal in May 2007.
- > Current schedule supports a possible NRC SER at the end of November 2007.
- > Current fuel analysis and review schedule supports an SER sometime in 2008.
- > Action Required – Improve fuel analysis and review schedule.

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Key Decision - Recommended Power Level

> It is recommended that power level be increased 1.6 percent to 2609 MWt. (requires installation of more accurate RTDs)

> Justification
- Preliminary uncertainty calculations show uncertainty of 0.44 % without thermocouple replacement.
- Replacement of thermocouples reduces uncertainty to about 0.35 %.
- Choosing a power increase of 1.6 % provides margin.

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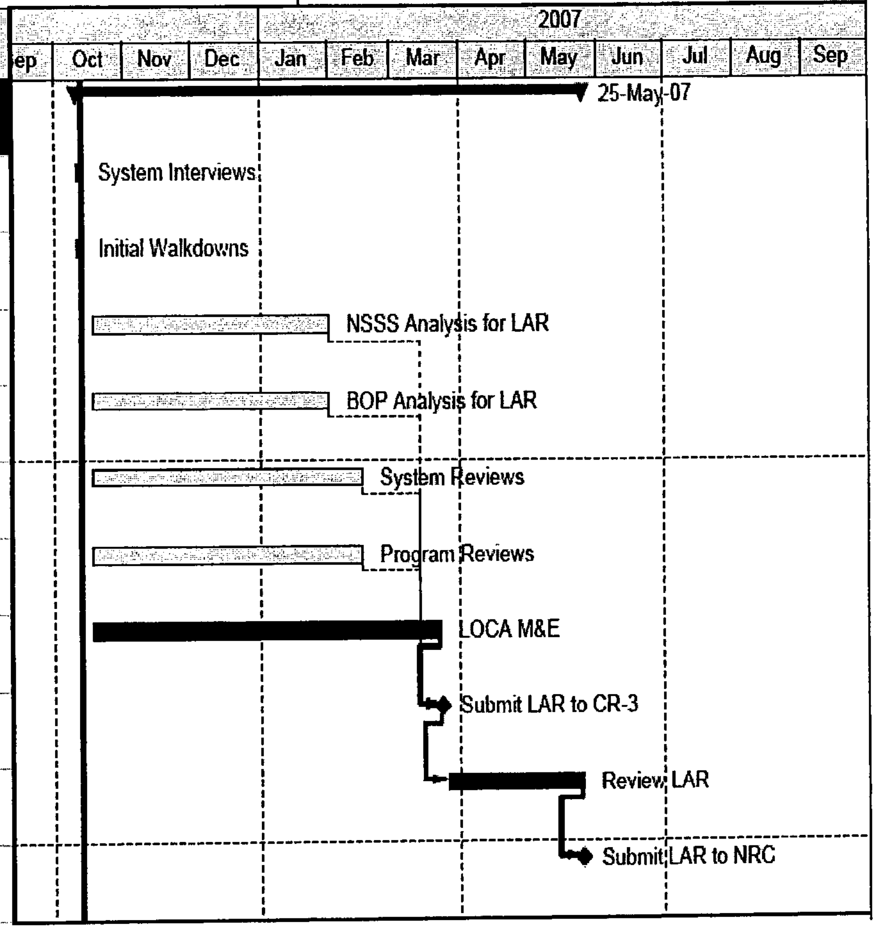
License Amendment Request

13-Oct-06 14:00

Activity ID	Activity Name	% Complete	Start	Finish
License Amendment Request			10-Oct-06 A	25-May-07
A1000	System Interviews	100%	10-Oct-06 A	13-Oct-06 A
A1010	Initial Walkdowns	100%	10-Oct-06 A	13-Oct-06 A
A1020	NSSS Analysis for LAR	0%	17-Oct-06*	31-Jan-07
A1030	BOP Analysis for LAR	0%	17-Oct-06*	31-Jan-07
A1040	System Reviews	0%	17-Oct-06*	15-Feb-07
A1050	Program Reviews	0%	17-Oct-06*	15-Feb-07
A1060	LOCA M&E	0%	17-Oct-06*	22-Mar-07
A1070	Submit LAR to CR-3	0%		23-Mar-07
A1080	Review LAR	0%	26-Mar-07	25-May-07
A1090	Submit LAR to NRC	0%		25-May-07*



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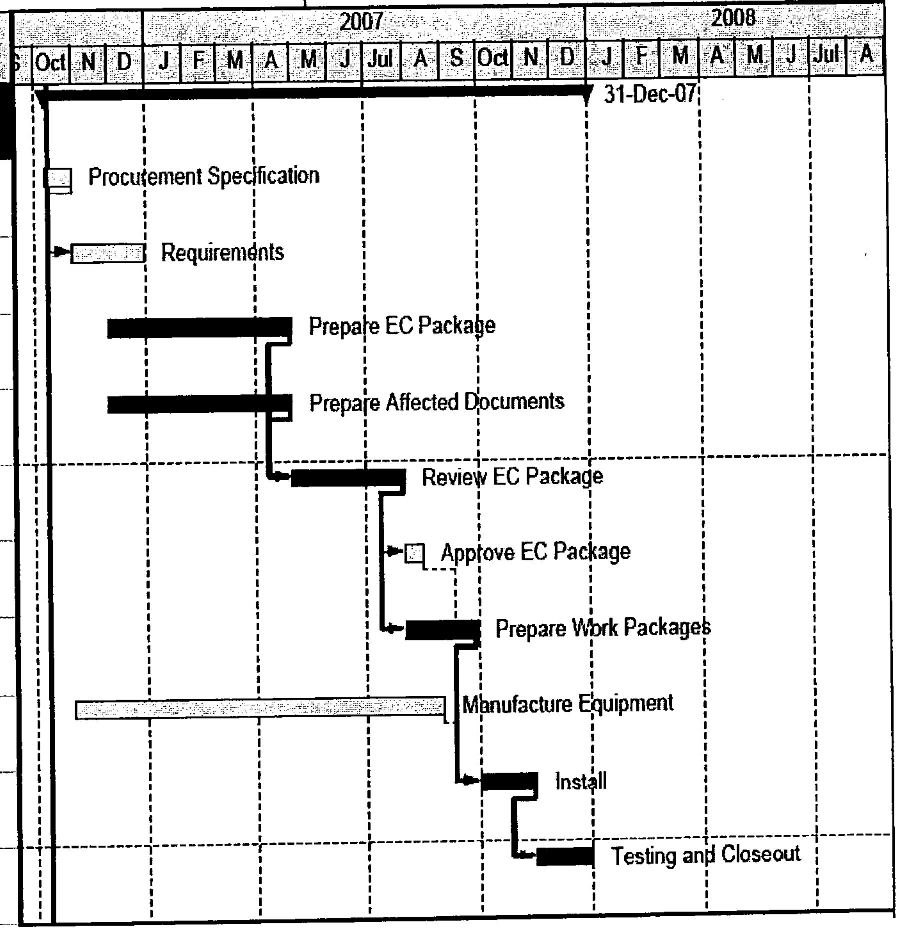
13-Oct-06 14:05

Caldon ECs and Installation

Activity ID	Activity Name	% Complete	Start	Finish
Caldon ECs and Installation			10-Oct-06 A	31-Dec-07
A1100	Procurement Specification	95%	10-Oct-06 A	31-Oct-06
A1110	Requirements	0%	01-Nov-06	29-Dec-06
A1120	Prepare EC Package	0%	01-Dec-06*	30-Apr-07
A1130	Prepare Affected Documents	0%	01-Dec-06*	30-Apr-07
A1140	Review EC Package	0%	01-May-07*	31-Jul-07
A1150	Approve EC Package	0%	01-Aug-07*	15-Aug-07
A1160	Prepare Work Packages	0%	01-Aug-07*	28-Sep-07
A1170	Manufacture Equipment	0%	31-Oct-06*	31-Aug-07
A1180	Install	0%	01-Oct-07	15-Nov-07
A1190	Testing and Closeout	0%	16-Nov-07	31-Dec-07



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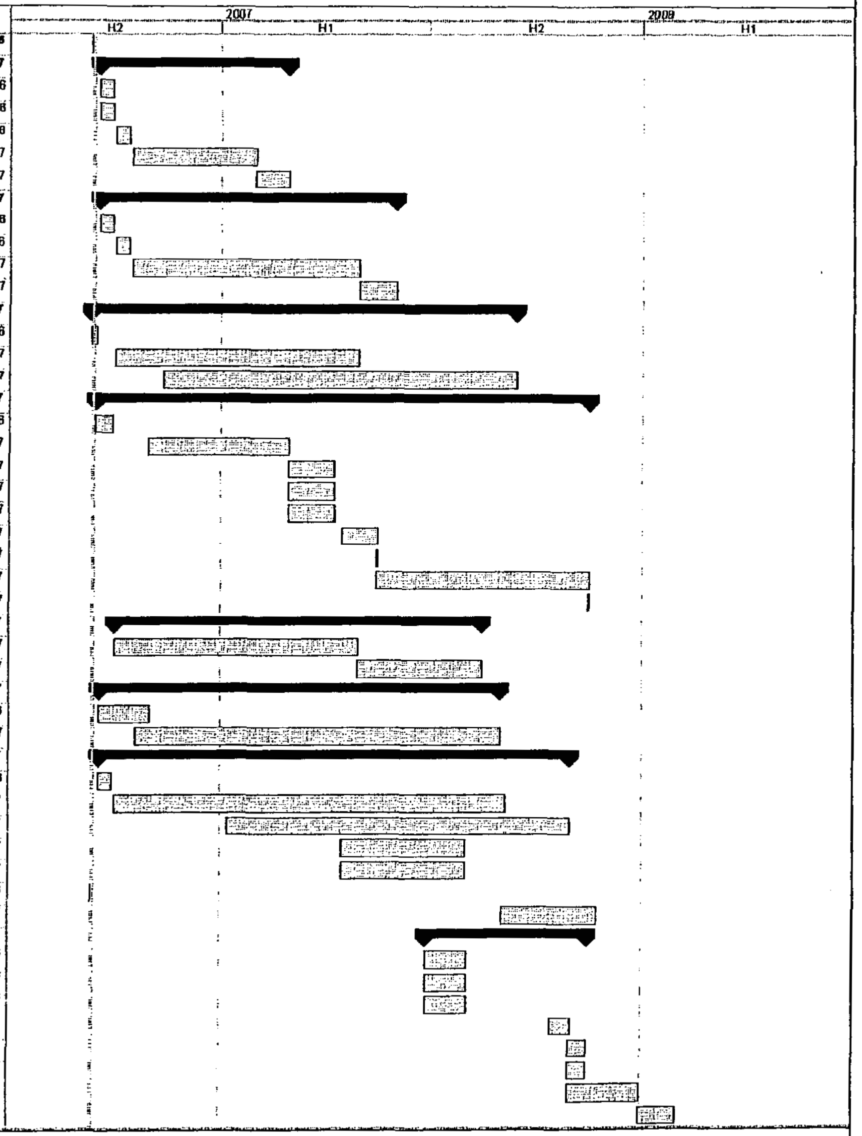
MUR Preliminary Schedule

MUR Power Uprate Draft Level 1 Schedule

ID		Task Name	Start	Finish
1		**Kick Off Meeting**	**Mon 9/11/06**	**Wed 9/13/06**
2		**System Evaluations**	**Mon 9/18/06**	**Wed 2/28/07**
3		Review Issues/Restraints Letter SE01-0154	Mon 9/18/06	Fri 9/29/06
4		Compare System List with Previous Evluations	Mon 9/18/06	Fri 9/29/06
5		Perform System Interviews	Mon 10/2/06	Fri 10/13/06
6		*Prepare System Evaluation Report*	Mon 10/16/06	Wed 1/31/07
7		CR Owner Acceptance	Wed 1/31/07	Wed 2/28/07
8		**Program Evaluations**	**Mon 9/18/06**	**Fri 6/1/07**
9		Program Selection	Mon 9/18/06	Fri 9/29/06
10		Interviews	Mon 10/2/06	Fri 10/13/06
11		Program Reviews	Mon 10/16/06	Mon 4/30/07
12		Owner Acceptance	Tue 5/1/07	Fri 6/1/07
13		**Fuel Analysis**	**Mon 9/11/06**	**Fri 9/14/07**
14		Pick Power Level	Mon 9/11/06	Fri 9/15/06
15		FOAK	Mon 10/2/06	Mon 4/30/07
16		Reload PT Uprate	Mon 11/13/06	Fri 9/14/07
17		**LAR Preparation**	**Fri 9/15/06**	**Fri 11/16/07**
18		Request NRC Docket Position	Fri 9/15/06	Fri 9/29/06
19		Prepare LAR	Tue 10/31/06	Wed 2/28/07
20		Prepare Technical Specification Changes	Thu 3/1/07	Mon 4/9/07
21		Prepare FSAR Changes (What is required date?)	Thu 3/1/07	Mon 4/9/07
22		Licensing Information to Licensing for Review (LAR, TS, FSAR Changes)	Thu 3/1/07	Mon 4/9/07
23		CR3 LAR Inhouse Review	Mon 4/16/07	Wed 5/16/07
24		Submit LAR for MUR to NRC	Wed 5/16/07	Wed 5/16/07
25		Respond to NRC RAIs	Wed 5/16/07	Fri 11/16/07
26		NRC SER	Fri 11/16/07	Fri 11/16/07
27		**Engineering Change Package Preparation**	**Mon 10/2/06**	**Wed 8/15/07**
28		Prepare EC for Caldon	Mon 10/2/06	Mon 4/30/07
29		Owner Acceptance	Mon 4/30/07	Wed 8/15/07
30		**Procurement**	**Mon 9/18/06**	**Fri 8/31/07**
31		Prepare Caldon PO	Mon 9/18/06	Tue 10/31/06
32		Procure Caldon	Fri 10/20/06	Fri 8/31/07
33		**Calculation Package Modifications Resulting From MUR**	**Mon 9/18/06**	**Wed 10/31/07**
34		Review List of Calc. Pkg. Changes from Previous Uprate	Mon 9/18/06	Fri 9/29/06
35		Prepare Calculation Package Modifications	Mon 10/2/06	Wed 9/5/07
36		Owner's Acceptance	Mon 1/8/07	Wed 10/31/07
37		**Procedure Modifications**	**Mon 4/16/07**	**Wed 8/1/07**
38		**EDBD Modifications**	**Mon 4/16/07**	**Wed 8/1/07**
39		***Simulator Modifications***	**Mon 9/11/06**	**Mon 9/11/06**
40		**Operator Training**	**Mon 9/3/07**	**Fri 11/23/07**
41		**Construction and Installation (By Construction Company)**	**Fri 6/29/07**	**Thu 11/15/07**
42		Prepare Cutting and Welding Work Packages for Caldon Equipment	Fri 6/29/07	Fri 8/3/07
43		Prepare Cable Pull Work Packages	Fri 6/29/07	Fri 8/3/07
44		Prepare Termination Work Packages at Computer and Caldon Eq.	Fri 6/29/07	Fri 8/3/07
45		Install Cables (Non Outage Activity)	Mon 10/15/07	Thu 11/1/07
46		*Cut Feedwater Lines and Weld in Caldon Equipment*	*Wed 10/31/07*	*Thu 11/15/07*
47		Terminate Cabling	Wed 10/31/07	Thu 11/15/07
48		**Startup and Testing**	**Wed 10/31/07**	**Mon 12/31/07**
49		**Closeout**	**Mon 12/31/07**	**Thu 1/31/08**



PEF-CR3-0716

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Issues and Restraints List

> SE01-0154 Issues and Restraints from 2568 MWt

14 items were identified as part of 2002 uprate. It is expected that these have all been resolved. We have verified resolution of about 10.

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Issues and Restraints List

- ICS – Will require software modification to Triconex equipment – Recommend that Progress Energy do this EC.
- Main Steam Thermowell – Jet impingement shield is located over thermowell location.
- HELB analysis (outside containment) was performed at 2568 MWt.
- FDMIS version 1.7 or 1.8 will require installation in the summer – fall of 2007 for testing purposes.

Issues and Restraints List

- Fuel Schedule does not support November 2007 SER
- Early LAR submittal
- Appendix R analysis was performed at 2568 MWt.
- FAC analysis is limiting at feedwater pipe to feedwater ring header location.
- Heat balance pump heat.
- Progress Energy support team identification.
- Verify Progress Energy will perform installation.

PEF-CR3-0719

Metrics and Performance Indices

> AREVA will develop a set of performance indices by 11-17-06

> Suggested indices include:

❖ Budget performance

❖ Deliverable document progress

❖ Issues and restraints

❖ Earned value

❖ Quality

❖ Safety

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[redacted]

Action Items

> **Action Items from September 13, 2006 Meeting**

See attached

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Summary and Conclusions

> No Show Stoppers Have Been Identified.
> Support for Interviews Was Excellent.
> SER Schedule Is Tight, but Achievable.