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.

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

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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 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 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 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 as low as reasonably achievable (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. <u>Identified LEAKAGE</u>

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)

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 ≥ 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. <u>Identified LEAKAGE</u> 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

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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%

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 ($\leq$ 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 CheckPlus$_{TM}$ 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 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 set point. The set point 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 set point 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 reactor coolant system (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 reactor coolant pumps (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 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 4.3.3).

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 4.3.1. 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.

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

3.1.1 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.

3.2.1 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).

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

1. 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/HPI 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 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 high energy line break (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 SectionIV.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 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 SectionIV.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 LTOP limits remain valid for the MUR power uprate. (Reference 4.3.7)

4.2.3.4 Low-Temperature Overpressure Protection (LTOP) (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

10CFR 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 Correspondence 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 the 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 EC 65626 and additional Main Steam pressure and temperature instrumentation by Engineering Change (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 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 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 LPI/DH 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 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.
2. 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

3. 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.

4. 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.

5. 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)

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

Emergency and abnormal operating procedures (EOP and AOP) 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.

7.2.2.2 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 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 License Amendment Request (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

AREVA

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 32- 9042687 -001

Title Heat Balance Uncertainty for CR-3 MUR

PREPARED BY:

NAME J A Weimer

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TITLE Advisory Engineer DATE 4/17/07

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METHOD: X DETAILED CHECK ☐ INDEPENDENT CALCULATION

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TM STATEMENT: REVIEWER INDEPENDENCE

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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:

CODE/VERSION/REV	CODE/VERSION/REV

THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK

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:

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

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;

---

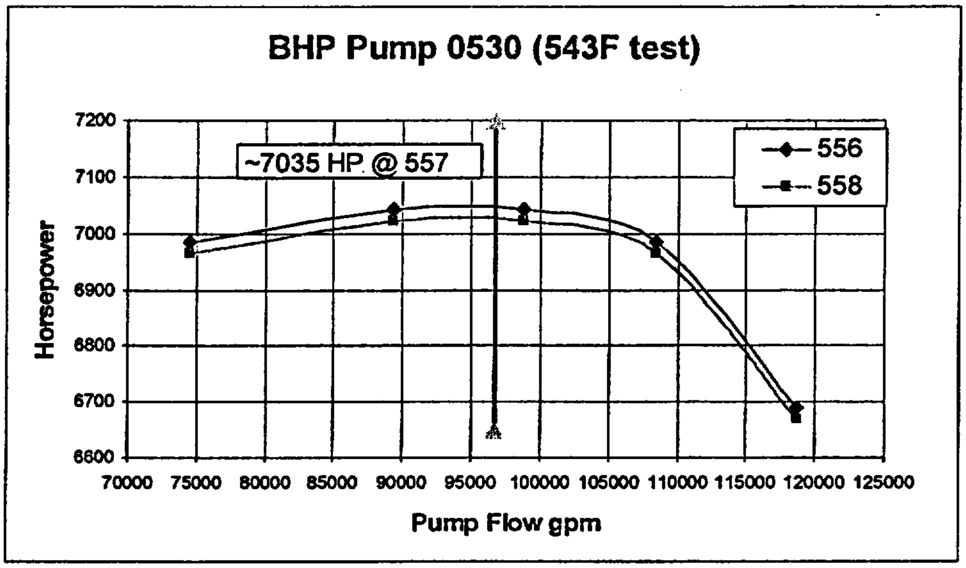
[2] This uncertainty was based on the judgment that any instrumentation used to actually measure/calculate the primary system ambient losses would likely have less than the 0.75MWt random uncertainty (since it is approximately +/- 50% of an anticipated value).

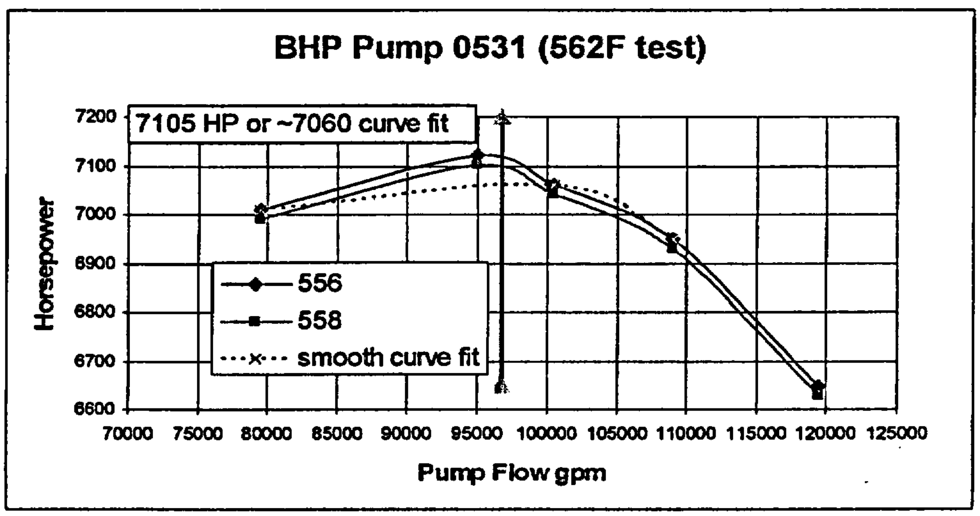
Impeller ID 0532				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
80350	6990.25	6970.22	6950	560
97800	7141.99	7121.53	7080	562
100700	7061.29	7041.07	7000	562
111800	6864.59	6844.92	6805	562
120150	6658.04	6638.97	6610	561

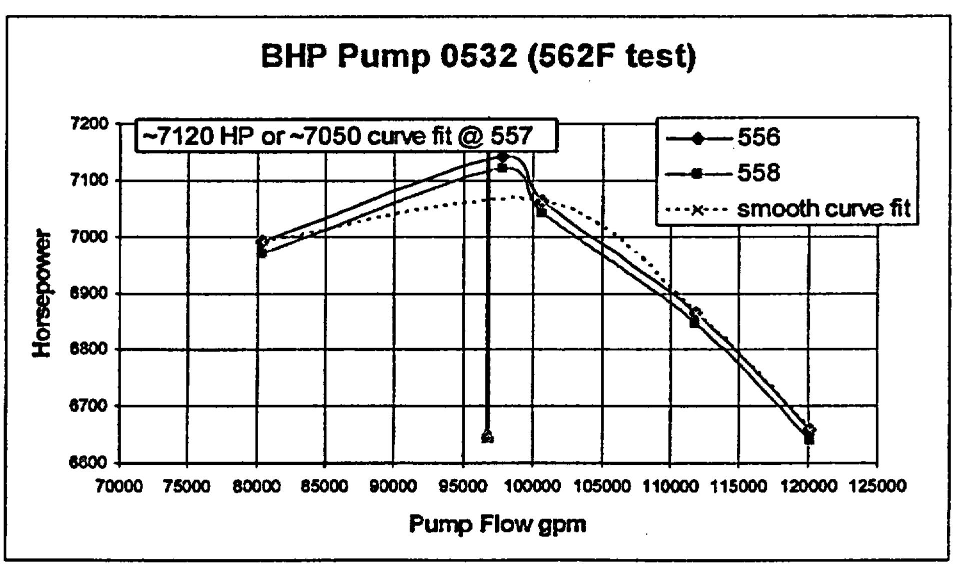
Impeller ID 0531				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
79550	7010.85	6990.77	6950	562
95000	7121.82	7101.42	7060	562
100400	7061.29	7041.07	7000	562
109050	6950.33	6930.42	6890	562
119500	6647.97	6628.92	6600	561

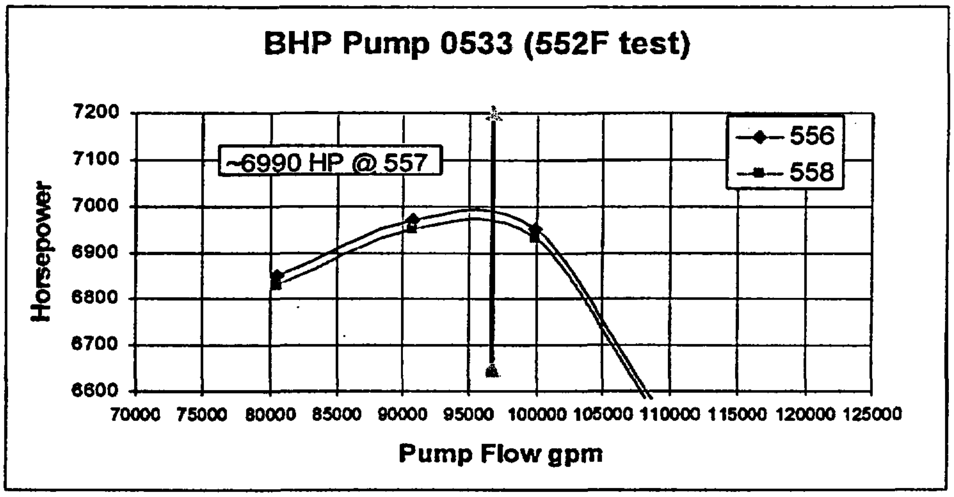
Impeller ID 0530				
	Corrected to temp		Test Data	
Flow(gpm)	BHP(556)	BHP(558F)	BHP	Temp (F)
74500	6984.49	6964.48	7110	543
89350	7043.43	7023.26	7170	543
98750	7043.43	7023.26	7170	543
108500	6984.49	6964.48	7110	543
118800	6688.80	6669.64	6800	544

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

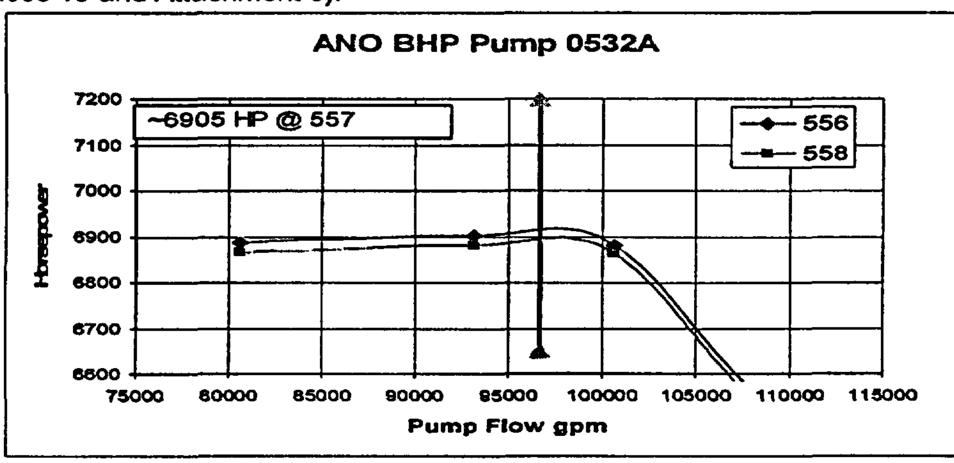

BHP Pump 0530 (543F test)
~7035 HP @ 557
556
558
Horsepower
7200
7100
7000
6900
6800
6700
6600
70000 75000 80000 85000 90000 95000 100000 105000 110000 115000 120000 125000
Pump Flow gpm

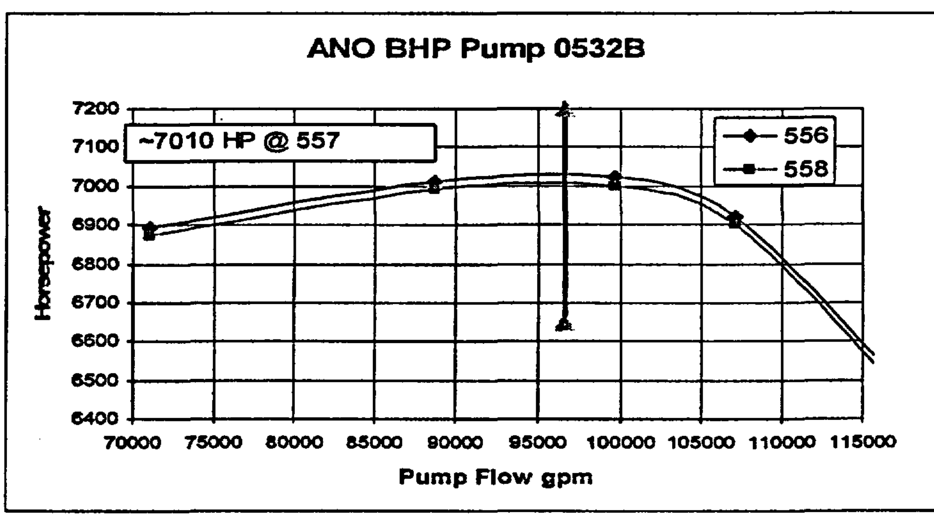

BHP Pump 0531 (562F test)
7105 HP or ~7060 curve fit
556
558
smooth curve fit
Horsepower
7200
7100
7000
6900
6800
6700
6600
70000 75000 80000 85000 90000 95000 100000 105000 110000 115000 120000 125000
Pump Flow gpm

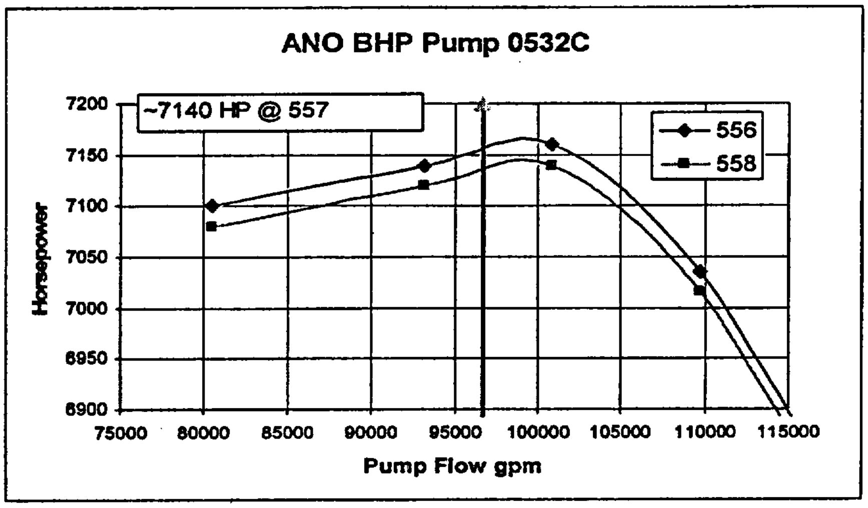

BHP Pump 0532 (562F test)
~7120 HP or ~7050 curve fit @ 557
556
558
smooth curve fit
Horsepower
7200
7100
7000
6900
6800
6700
6600
70000 75000 80000 85000 90000 95000 100000 105000 110000 115000 120000 125000
Pump Flow gpm

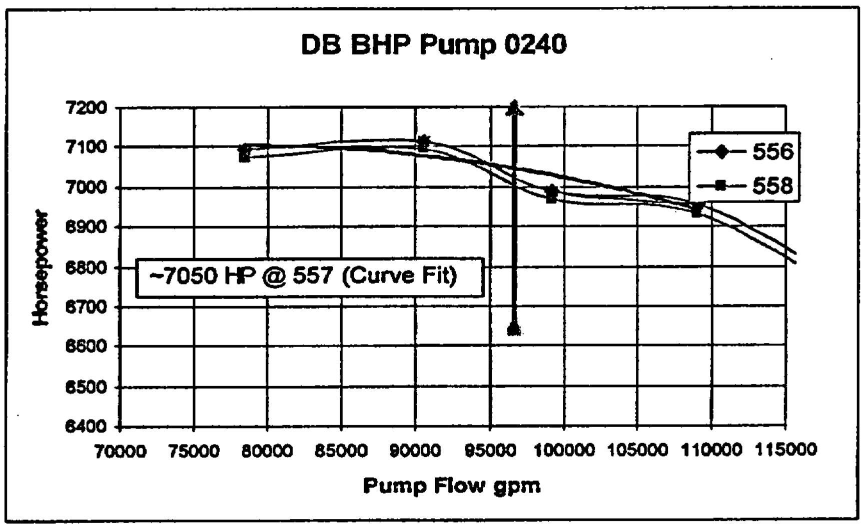


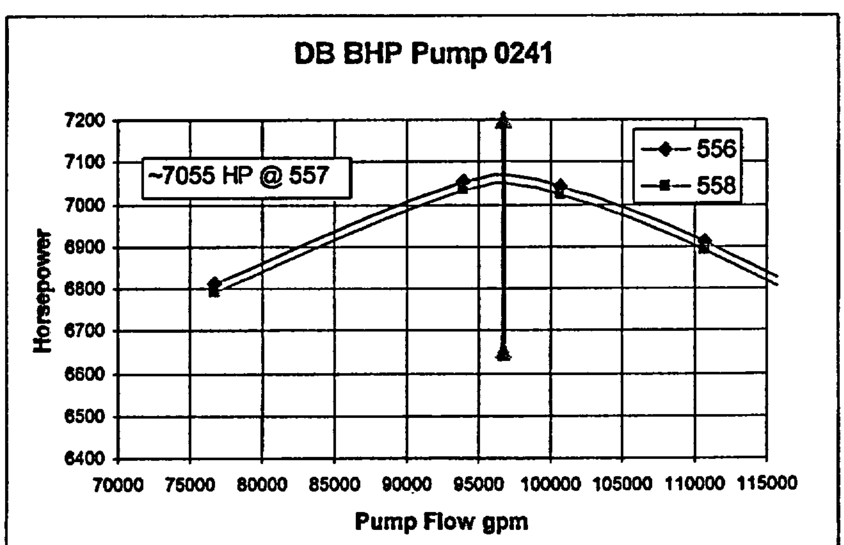
Finally, note that the Byron Jackson (Flow Server) design was relatively consistent in the other impellers tested for ANO-1 and DB-1. Some of these impeller BHP curves are shown below (based on References 13 and Attachment 3).

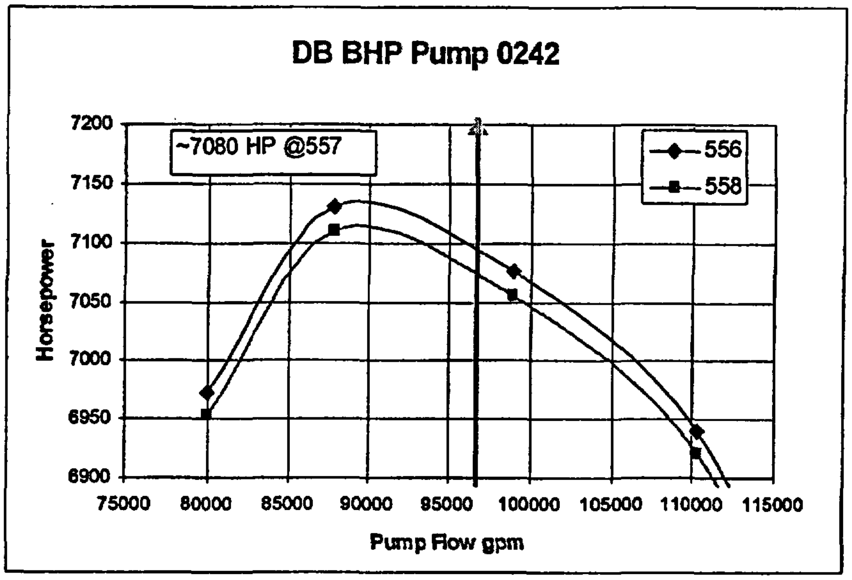


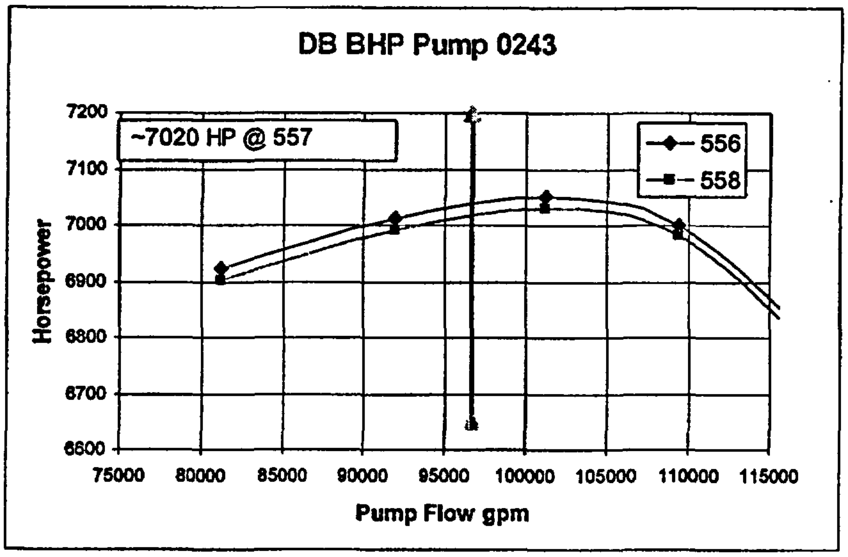



ANO BHP Pump 0532C
~7140 HP @ 557
556
558
Horsepower
7200
7150
7100
7050
7000
6950
6900
75000
80000
85000
90000
95000
100000
105000
110000
115000
Pump Flow gpm


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 ε_{WfwT} = (0.34/100) * 8.905E9 = 3.028E7 BTU/hr, the systematic uncertainty contribution is:

$[\varepsilon_{WfwT}]^2$ = 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 [signature] 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
										100.0%	10.28
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		
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							

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

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	TESTED BY
30 x 33 x 38 DFSS		RM FJ

IMPELLER DRAWING	DIAMETER	VANES	UNDER		EYE		R.P.M.	G.P.M.	HEAD	PUMP EFF.	OVERALL EFF.	NPSH	SP. GR.	WITNESSED BY
R-3803	34 5/8 x 37 5/8	6	5/8 x 8		1	RATED SPEED	1185	90600	358				0.74	FOR BABCOCK & WILCOX - TOLEDO
						TEST SPEED								CUSTOMER'S ORDER NO. 022247-240

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

ITEM NO. ... JOB NO. 701 ... 0240

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

NOTES: ... S/A 15-81 P34

MOTOR EFF. ... BAROMETRIC PRESSURE ... IN. HG.

HOT WATER PERFORMANCE

READING NO.	SUCTION HEAD GAUGE 3000# PSI READ	PSI CORR.	DISCHARGE HEAD GAUGE 3000# PSI READ	PSI CORRECTED	DIFF. PRESS.	ELEVATION HEAD	DIFFERENTIAL VELOCITY HEAD	TOTAL DYNAMIC HEAD IN FEET OF WATER	CAPACITY MEASURED WITH 36 x 24 VENTURI HEAD	G.P.M.	WATER HORSE POWER	BRAKE HORSE POWER	WATT METER READING	EQUIV. KW INPUT	VOLTS	AMPS	PUMP EFFICIENCY	WATER TEMP °F
1	2042	2045	2185	2185.8	133.3	0	0	412	15.8	78500	6105	7260	5650	7570	13090	280	84.1	539
2	2050	2055	2171	2171.7	121			379	20.0	90500	6400	7290	5700	7640	13050	280	87.7	538
3	2051	2051.5	2161	2161.7	110			340	25.2	99200	6370	7160	5600	7500	13050	280	89.0	538
4	2033	2033.5	2129	2129.6	100			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	5405	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	2030.5	2135	2135.7	125.1			369	22.5	91700	6690	7480	5850	7840	13050	280	84.4	503

BHP @ sp. gr. - 0.74

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

BORG-WARNER CORPORATION

PUMP TEST DATA

Pump size and type: 33 x 33 x 38 DFSS

Impeller drawing	Diameter	Vanes			Stage
R-3823	34 1/2 x 37 5/8	5	7/8 x 3		1

Guaranteed pumping conditions

	R.P.M.	G.P.M.	T.D. Head				Sp. Gr.
Field speed	1186	96679	558				.074
Test speed							

Diff. vel. hd. between gages at 36 I.D. Disch. and 36 I.D. Suct. = 0

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

Motor eff.

Barometric pressure in. Hg.

Tested by: MA

Witnessed by:

For: BABCOCK & WILCOX

Customer's order no. 022297-LW

Item no. / Job no. 701-N-0241

Notes: MTR S/N S-81P34

HOT WATER PERFORMANCE

Suct. HD gauge 3000; Discharge head gauge 3000; Capacity measured with 36 x 24 VENTURI

Test No.	Suct. PSI read	PSI corr.		Disch. PSI read	PSI corrected	Diff. press. Δ PSI	Elevation head	Differential velocity head	Total dynamic head	Venturi read	G.P.M.	Water horse power	Brake horse power	Watt meter reading	Equiv. KW input	Volts	Amps	Pump efficiency		Water temp. °F	Pump speed R.P.M.
1	2117	4.6		2228	2222.8	111.2	0	0	356	21.8	94000	6090	6970	5450	7300	13100	240	87.3		564	1189
2	2118	4.6		2222	2217.8	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	110700	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		2187	2182.8	66.2			211.5	40.5	127800	4930	6330	4950	6630	13070	230	77.9		562	1191
6			1	STOP	START	AT	RUNOUT	FLOW	SPIN	DOWN	TIME	2 MIN	- 48	SECS							
7	1982	2.4		2138	2138.7	156.3	0	0	466	4.3	50400	3655	6590	5150	6900	13070	250	55.5		570	1190
8	2006	3.4		2154	2154.7	148.5			441.0	11.0	64300	5550	7000	5475	7340	13050	260	79.4		[illegible]	1189
9	2043	4.5		2180	2180.3	137.2			410	15.6	76700	6140	7200	5600	7540	13030	260	85.3		512	1188
10	2035	4.8		2148	2148.7	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 @ SP.GR. = 0.74

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

	Corrected to temp		Test Data	
GPM	556	558	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. Ilievich Date 4/12/07

Date: | Pump serial no. 701-N-0241 | T-33340

P-366 12/71

Jackson Pump Division
Borg-Warner Corporation

PUMP TEST DATA

PUMP SIZE: 33 x 33 x 33 DFSS

GUARANTEED PUMPING CONDITIONS

R-3873 | 5 | 33 | 48 | 90620 | 558 | 074

TESTED BY: MAT DX

WITNESSED BY: TOH BABCOCK & WILCOX – TOLEDO

CUSTOMER'S ORDER NO.: D22297

JOB NO.: 70-N-0242

36 | 36

NOTES: MTR S/N 1 > 8 P35

9000 HP 6 POLE 13200

HOT WATER PERFORMANCE

SUCT. HD. GAUGE Psi READ	Psi READ (corr.)		DISCHARGE HEAD GAUGE Psi READ	Psi CORRECTED	DIFF PRESS. Barton Δ PSI				CAPACITY VENTURI 36 x 24 HEAD	GPM			POWER		VOLTS	AMPS	EFFICIENCY	BHP 556	WATER TEMP.	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	5530	7440	13160	250	87.6	7240	559	1188
2176	2161.4		2271	2269.4	107.2			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			242	41.0	128300	4990	6270	4890	6550	13160	220	75.9	6370	559	1190
2 STOPS — 2 STARTS AT RUNOUT FLOW										SPIN DOWN TIMES			2 MIN 37 SECS				2 MIN 50 SECS			
[illegible]	2070.4		2225	2223.4	153.0	10	0	456	7.0	51400	4590	6810	5330	7140	13100	240	67.9	6500	510	1188
[illegible]	2070.4		2221	2219.4	149.5			447	8.5	56700	4950	6890	5390	7210	13100	250	71.7	6610	515	1188
[illegible]	2070.4	--	2212	2210.4	140.5			422	12.8	70000	5720	7125	5570	7460	13100	250	83.3	6885	520	1188
2085	2070.4		2204	2202.4	132			400	16.7	80000	6150	7260	5670	7600	13100	260	86.6	7055	525	1188
[illegible]	2070.4		2198	2196.4	126.5			386	19.0	84500	6250	7200	5650	7560	13100	260	86.2	7070	520	1188

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

GPM	Corrected to temp 556	558	Test Data 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

D.S. Ulevich Date 4/12/07

DATE OF TEST

SERIAL No. 70-N-0242

P-366 12/71

T-33522

PUMP TEST DATA

PUMP SIZE AND TYPE: 33x33x38 DFSS

Impeller Drawing	Diameter	Vanes			Stage
E 5323	36 27/32	5	5/6 x 3		1

GUARANTEED PUMPING CONDITIONS

	R.P.M.	G.P.M.	T.D. Head				
Field Speed	1186	90670	358				674
Test Speed							

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

TEST MOTOR NO. 9000 HP 6 POLES 13200 VOLTS

MOTOR EFF.

BAROMETRIC PRESSURE

TESTED BY: MA - H7

WITNESSED BY:

FOR: BABCOCK & WILCOX - TOLEDO

CUSTOMER'S ORDER NO. 022297 - LW

ITEM NO.

JOB NO. 701-N-0243

NOTES: MT-SA-81P0035-2

HOT WATER PERFORMANCE

Reading No.	Suct. Hd. Gauge HG2 3000# PSI Read	PSI Corr		Disch. Head Gauge HG2 3000# PSI Read	PSI Corrected	Diff Press 045041 Barton Δ PSI	Elevation Head	Differential Velocity Head	Total Dynamic Head in Feet of Water	Capacity Measured with 36x24 Venturi Head	Capacity G.P.M.	Water Horse Power	Brake Horse Power	Watt Meter Reading	Equiv. KW Input	A.C. Volts	Amps	Pump Efficiency	BHP @ SP.GR. 0.74	Water Temp °F	Pump Speed RPM
1	2121	2123		2241	2240	117.0	0	0	366	2.35	91900	6255	7080	5525	7410	13000	270	88.5	7100	549	1189
2	2120	2122		2229	2228	106.0			332	2.60	101300	6260	7120	5560	7460	13000	270	88.0	7150	549	1189
3	2118	2120		2217	2216	96.0			301	3.04	109450	6140	7070	5525	7410	13000	270	86.9	7100	549	1189
4	2118	2120		2204	2203	82.5			258.5	3.61	119300	5750	6820	5325	7140	13000	260	84.4	6850	549	1190
5	2116	2118		2191	2190	71.5			224	4.67	127500	5310	6640	5210	6980	13000	250	79.7	6690	549	1190
6																					
7	2095	2097		2247	2246	150.0	0	0	455	6.6	50900	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	538	1189
12																					

DP READINGS 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 32-9042687-001 Pg 35

0243	Corrected to temp		Test Data	
GPM	556	558	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

Date 4/12/07

D. S. Ulevitch

DATE OF TEST

PUMP SERIAL No. 701-N-0243

P-366 12/71

T-33547-1

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 (ºF)	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.

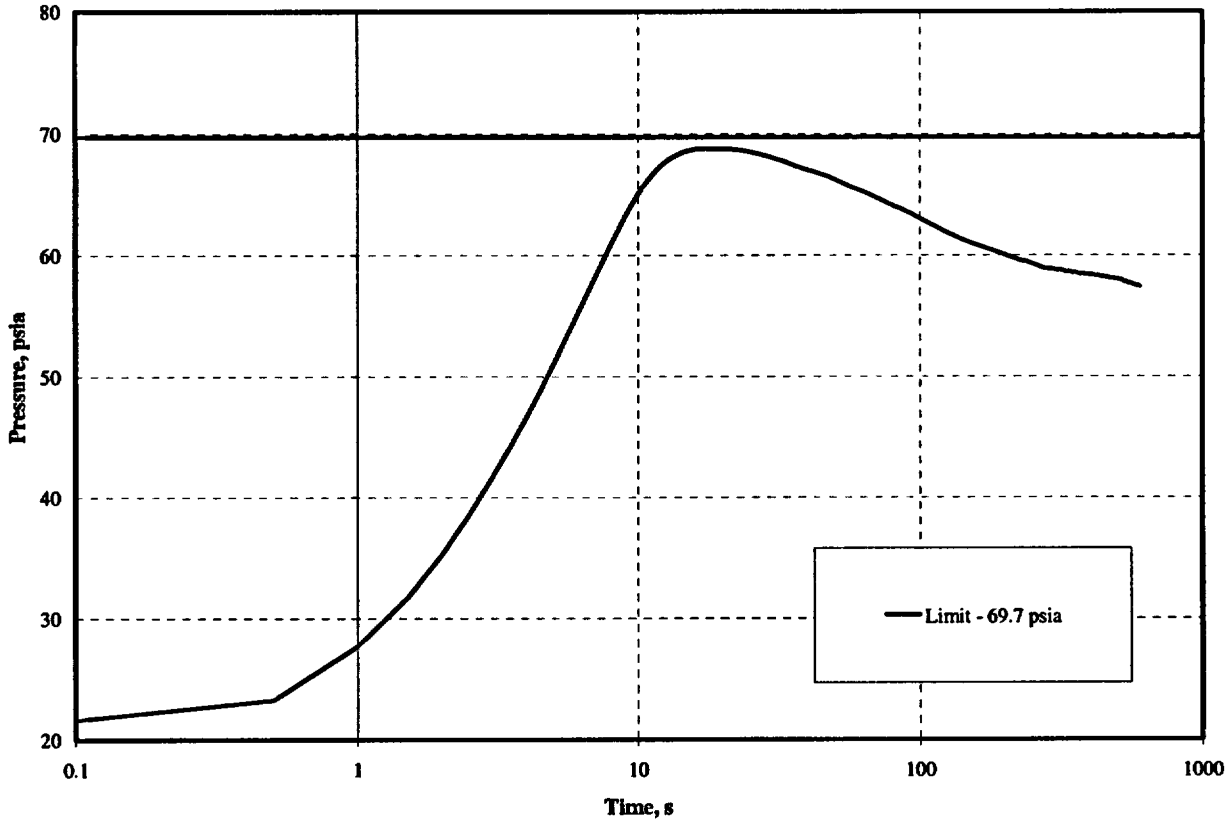


Figure F-1 – GOTHIC Predicted Peak Pressure for a Double-Ended Break at the SG Inlet

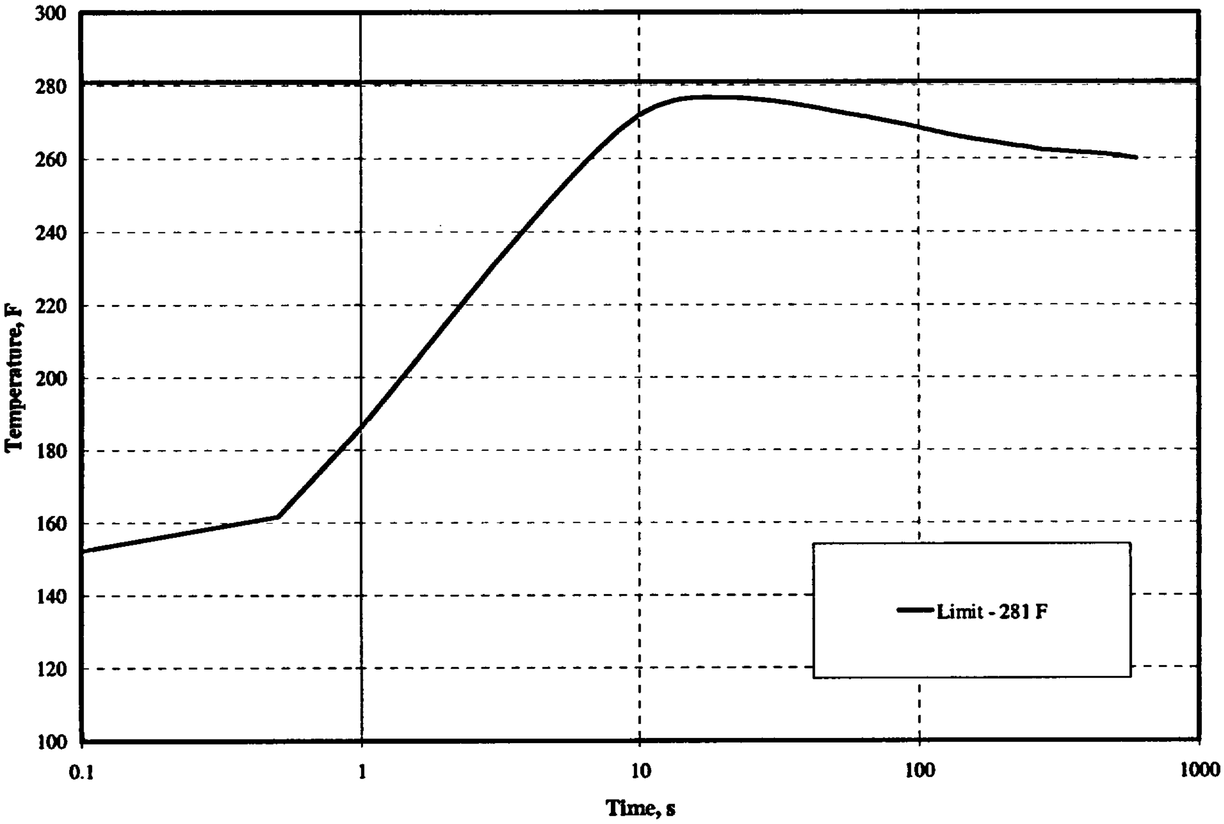


Figure F-2 – GOTHIC Predicted Peak Temperature for a Double-Ended Break at the SG Inlet

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)	$+0.315 \times 10^{-4}$
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

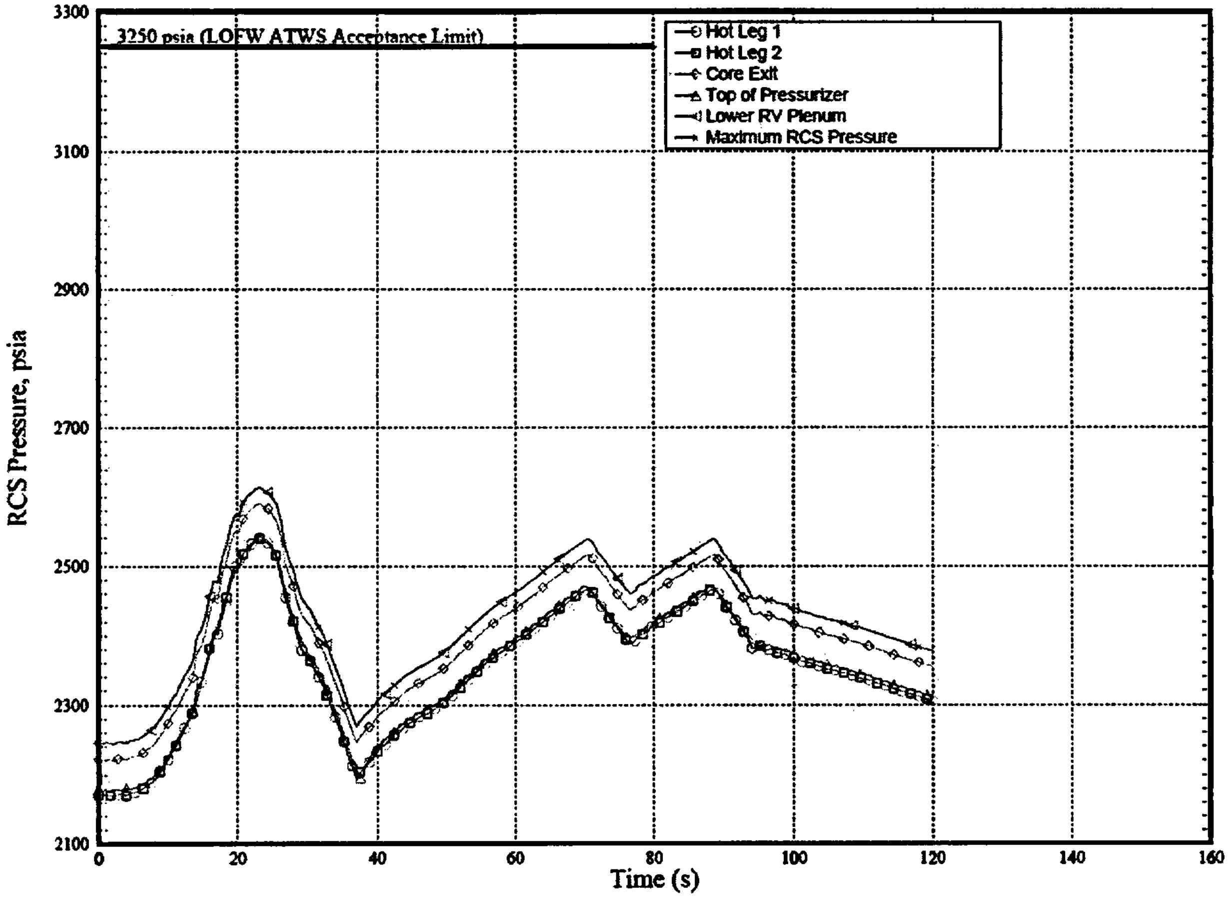


Figure G-2: RCS Temperature for LOFW ATWS for AMSAC with DSS at 52% Power

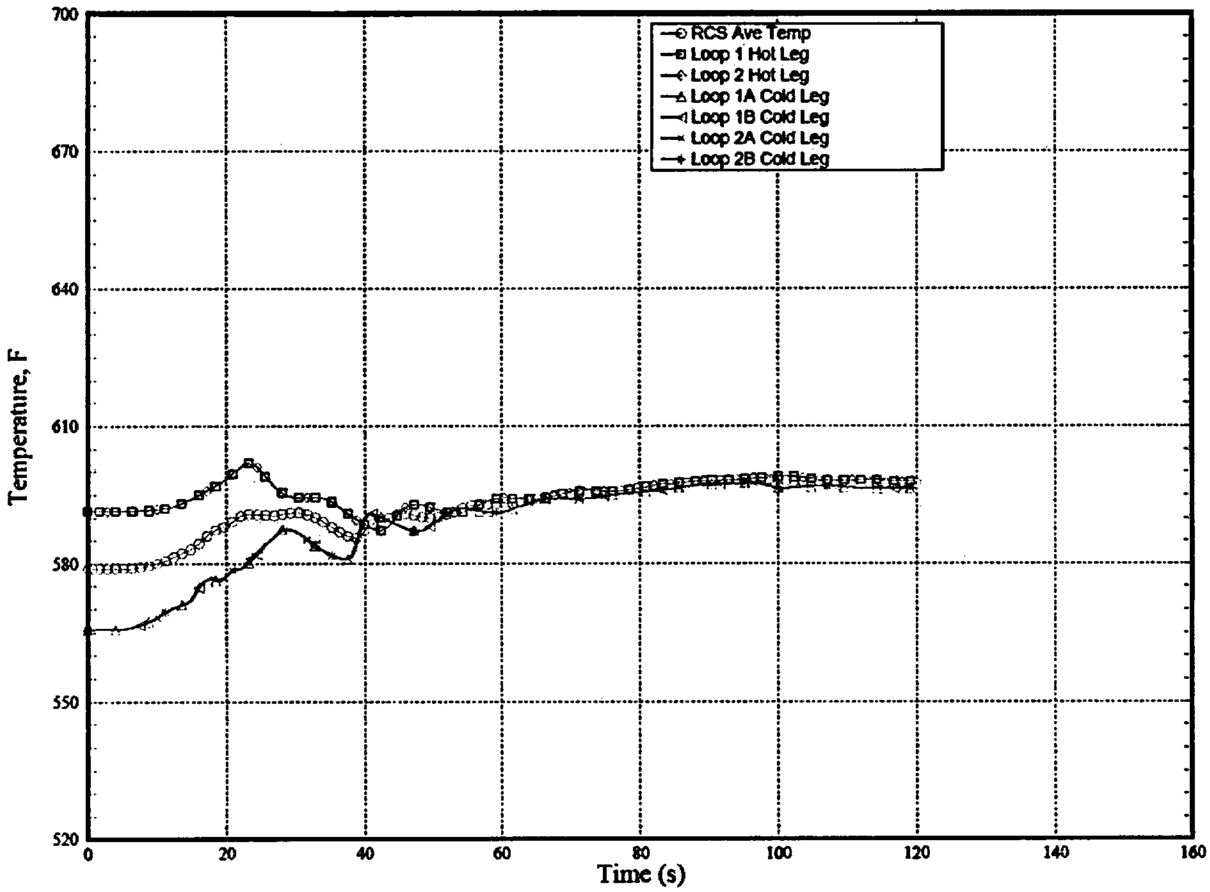
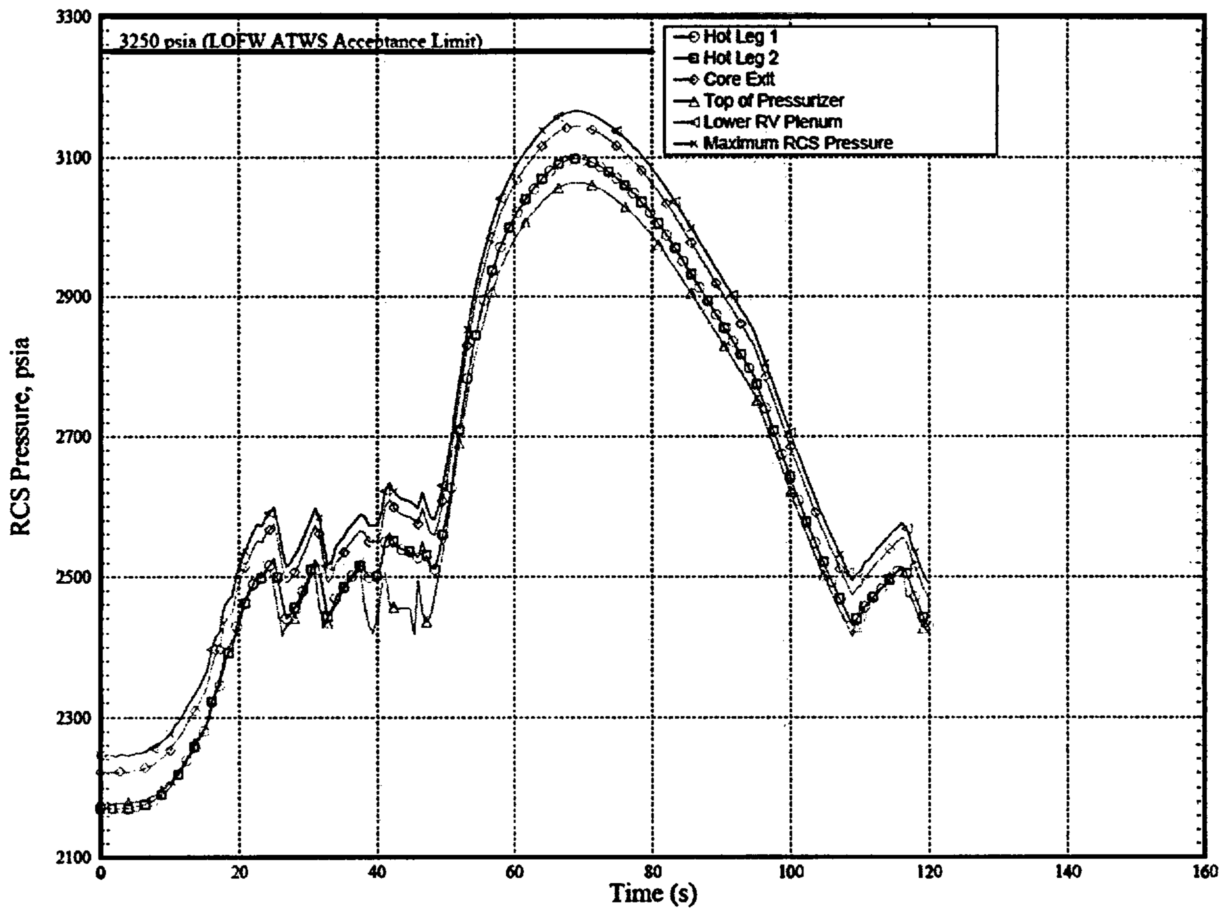


Figure G-3: RCS Pressure for LOFW ATWS for AMSAC without the DSS at 52% Power



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 ≤ 2568 MWt within 12 hours and reduce the nuclear overpower - high setpoint to ≤ 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