Florida Power
A **Progress Energy** Company

Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

June 5, 2002
3F0602-05

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

Subject: Crystal River Unit 3 - License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt"

References:
1. SECY-01-0124, "Power Uprate Application Reviews," dated July 9, 2001
2. Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

Dear Sir:

Pursuant to 10 CFR 50.90, Florida Power Corporation (FPC) submits a request to increase the maximum allowed rated thermal power (RTP) for Crystal River Unit 3 (CR-3) from 2544 MegaWatts-thermal (MWt) to 2568 MWt. Changes are requested to two definitions in Section 1.1 of the Improved Technical Specifications (ITS) 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.

Reference 1 categorized the power uprates into three types: stretch, measurement uncertainty recapture and extended. This application for CR-3 falls into the "stretch" category. The 24 MWt uprate is less than a 1 percent increase in RTP. No plant systems need to be upgraded or modified to accommodate this power uprate. Several setpoint changes are required to reflect the increased RTP. These setpoint changes can be made while the unit is at power. The existing CR-3 accident analyses were performed at 2568 MWt or higher, therefore, no analytical changes are needed to support this power uprate.

In Reference 1, the NRC Staff emphasized the high priority that is being given to power uprate license amendment applications. Regulatory Issue Summary (RIS) 2002-03, Reference 2, provided guidance for submittals requesting power uprates involving flow measurement uncertainty recapture. The RIS stated that licensee's applications which followed this guidance would require less review time, and could be approved in six months or less. Although the CR-3 power uprate does not involve measurement recapture, the guidance of RIS 2002-03 was used to ensure all areas of concern to the staff were addressed in this submittal. The proposed

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CR-3 power uprate is simpler in nature than the measurement recapture uprates because no evaluation of hardware changes or uncertainty calculations is required. Therefore, FPC requests approval of this request by September 30, 2002 in order to support a fourth quarter implementation.

Attachment A provides the description and assessment of proposed changes including the Environmental Assessment and the Determination of No Significant Hazards Considerations. Attachment B provides a strikeout version for the proposed revised ITS and license pages. Attachment C provides the revised ITS pages with revision lines.

This letter makes no new regulatory commitments.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Daniel L. Roderick
Director Site Operations

DLR/pei

Attachments:

A. Description and Assessment of Proposed Changes

B. Proposed Revised Improved Technical Specification Pages and License Condition – Strikeout Version

C. Proposed Revised Improved Technical Specification Pages – Revision Line Version

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

STATE OF FLORIDA

COUNTY OF CITRUS

Daniel L. Roderick states that he is the Director Site Operations, Crystal River Nuclear Plant for Progress Energy; 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.

Daniel L. Roderick
Director Site Operations
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 5th day of June, 2002, by Daniel L. Roderick

Susan I McDonald

Signature of Notary Public
State of Florida

Susan I. McDonald
Commission # DD 012591
Expires May 11, 2005
Bonded Thru
Atlantic Bonding Co., Inc.

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

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FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #270, REVISION 0
Power Uprate to 2568 MWt

Description and Assessment of Proposed Changes

Description and Assessment of Proposed Changes

1.0 INTRODUCTION

This letter submits a request to increase the Crystal River Unit 3 (CR-3) power level from 2544 MegaWatts-thermal (MWt) to 2568 MWt. The power level is directly referenced by two Improved Technical Specifications (ITS) definitions, Rated Thermal Power (RTP) and Effective Full Power Day (EFPD). The power level is also directly referenced in the plant operating license, paragraph 2.C.(1), Maximum Power Level. The power level is indirectly referenced by numerous specifications that utilize these definitions. For example, the nuclear overpower high setpoint is given as 104.9 percent of RTP in Table 3.3.1-1. Although the setpoint will remain at the same percentage, the actual reactor trip will occur at a higher absolute power (104.9 percent of 2568 MWt rather than 104.9 percent of 2544 MWt). Similar indirect changes will occur for ITS Limiting Conditions for Operation, Applicability, Actions and Surveillance Requirements which utilize the definitions of RTP and EFPD.

2.0 DESCRIPTION

The proposed change would revise the 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 **2568** 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 **2568** MWt times 24 hours or **61,632** MWhr.)
>
> RATED THERMAL POWER (RTP) – RTP shall be a total reactor core heat transfer rate to the reactor coolant of **2568** MWt.

3.0 BACKGROUND

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

September 30, 1994 (3F0994-08), CR-3 requested an increase in maximum 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 2568 MWt. All other analyses had already been performed at 2568 MWt or higher. The request was withdrawn in letter 3F0596-02, dated May 1, 1996, stating that Florida Power Corporation (FPC) would submit a new license amendment request if operation at 2568 MWt was desired.

FPC has been evaluating various options for increasing the power output of CR-3. The Babcock and Wilcox 177 Fuel Assembly (B&W 177 FA) Nuclear Steam Supply System (NSSS) has been licensed to operate as high as 2772 MWt with most facilities operating at 2568 MWt. FPC is evaluating plant modifications to feedwater flow instrumentation and other secondary plant systems that would increase capability above 2568 MWt. While that evaluation is in progress, FPC is making the current request for operation at the previously evaluated limit of 2568 MWt, an increase of 24 MWt.

4.0 TECHNICAL ANALYSIS

The power level for CR-3 is proposed to be increased from 2544 MWt to 2568 MWt. The uprate evaluation addressed the following categories: NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and balance-of-plant (BOP) systems. No new analytical techniques 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.

Since 1996, several analyses have been revised. The loss-of-coolant accident (LOCA) analyses were revised to account for plant modifications, increased steam generator tube plugging and a change in the analysis of record from the CRAFT2 evaluation model to RELAP5/MOD2-B&W (References 19 and 20). Other analyses were also revised to reflect the design improvements made during the 1996 to 1998 design outage and the subsequent refueling outage (R11) in Fall 1999. A detailed listing of topical reports, methodologies and calculations utilized in the CR-3 analyses is given in FSAR Sections 1.5.8 and 14.3. All of the revised 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 assumed in power measurement. Some analyses were performed at higher power levels (generally 2772 MWt) because they were performed generically to bound all B&W 177 FA plants. All of these analyses were approved by the NRC or were performed using methods or processes that were approved by the NRC.

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 and evaluations. A detailed assessment of the accident analyses and evaluations performed for the steam generator tube rupture, LOCA, and non-LOCA areas was performed. The containment accident analyses and evaluations and the radiological consequence evaluations were reviewed. The fuel was also evaluated for its ability to perform at the uprated power level. FPC concludes that no changes to design basis or transient analyses are required to accommodate the revised NSSS design conditions. Each of the NSSS systems and components were evaluated for the uprated conditions. The effects of the uprate on the balance of plant (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 no design changes other than calibrations and setpoint adjustments to safely operate at the uprated conditions. A summary of these evaluations and assessments follows.

4.1 Nuclear Steam Supply System (NSSS) Design Parameters

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.

4.1.1 Input Parameters and Assumptions

The total thermal power for the uprate analysis was set at 2568 MWt (core power). This is approximately 0.9 percent higher than the current core thermal power rating of 2544 MWt. Feedwater/steam flow, RCS hot leg temperature (T_{hot}), and RCS cold leg temperature (T_{cold}) were required to change to allow this power uprate. All other input parameters (e.g., reactor coolant system pressure, average RCS temperature (T_{avg}), steam pressure) remained the same as those used for the current licensing basis.

4.1.2 Discussion of Parametric Cases

Table 1 (Reference 1) provides the NSSS parameter cases, which were generated and used as the basis for the uprate project. Uprated conditions were calculated at 0 percent and 20 percent once-through steam generator (OTSG) tube plugging 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 (Reference 22) as well as the calculated uprated conditions at 0 percent and 20 percent OTSG tube plugging. The parameters listed in Table 1 have been reviewed against those inputs used to develop the RCS functional specification for the design of the plant. 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," (Reference 2), conservative flow velocities were used. The very slight change in mass flow has negligible impact on the components.

4.1.3 Conclusions

Changes to plant operating conditions were determined for the 0.9 percent power uprate (values for 0 percent 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: NSSS Performance Parameters

Parameter		Current Power		0.9% Uprate	
	Original Design Basis	**No OTSG Tube Plugging**	**20% OTSG Tube Plugging**	**No OTSG Tube Plugging**	**20% OTSG Tube Plugging**
Core Thermal Power (MWt)	2568	2544	2544	2568	2568
Other RCS Power (MWt)	16	16	16	16	16
Total Thermal Power (MWt)	2584	2560	2560	2584	2584
T_{hot} (°F)	604.0	601.7	602.5	601.9	602.7
T_{cold} (°F)	554.0	556.3	555.5	556.1	555.3
T_{avg} (°F)	579	579	579	579	579
Minimum RCS Mass Flow Rate (million pounds mass per hour – Mlbm/hr)	131.3	143.8	138.7	143.9	138.8
Maximum Steam Temperature (°F)	594.0	593.1	583.2	593.1	582.9
Maximum Feedwater/Steam Flow Rate (Mlbm/hr)	11.20	10.75	10.86	10.86	10.98
Steam Pressure (psia)	925	925	925	925	925
Maximum Feedwater Temperature (°F)	459.0	457.3	457.2	458.2	458.2

4.2 Design Transients

The uprated conditions in Table 1 are within the design conditions of the RCS functional specification. These values serve as the final conditions for the power escalation transient and initial conditions for full power transients such as reactor trip, load rejection, turbine trip, rapid depressurization, loss of flow, power change, and loss of main feedwater transients. Thus, these transients are not changed by the uprate. In addition, the injection transients, such as actuation of high pressure injection or emergency feedwater, are unchanged since the uprate conditions are bounded by the design transient conditions and the safety analyses. Also, since hot standby (MODE 3) conditions are unaffected by the power uprate, plant heatups and cooldowns, which are the most fatigue significant transients, are unchanged. Therefore, it was concluded that the design transients are not adversely affected by the power uprate.

4.3 Nuclear Steam Supply System (NSSS) Fluid Systems

This section presents the results of the evaluations and analyses performed for the NSSS systems to support the revised operating conditions in Table 1. As indicated above, the parameters all remain within the design basis of the plant. The results and conclusions of each evaluation are presented within each subsection.

4.3.1 Reactor Coolant System (RCS)

The RCS consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains two reactor coolant pumps, which circulate the water through the loops and reactor vessel, and a once through steam generator (OTSG), where heat is transferred to the main steam system (MSS). In addition, the RCS contains a pressurizer which controls the RCS pressure through electrical heaters, water sprays, a power operated relief valve (PORV) and spring loaded safety/relief valves. The steam discharged from the PORV and safety/relief valves flows through interconnecting piping to the reactor coolant drain tank.

As discussed above, the revised operating conditions all remain within the design basis of the plant. Further, the accident analyses have been performed at power levels that bound the uprated conditions. Therefore, the RCS is unaffected by the uprate.

4.3.2 Emergency Core Cooling Systems (ECCS)

The emergency core cooling systems are used to mitigate the effects of postulated design basis events. The basic functions of the systems include providing short and long term core cooling, and maintaining core shutdown reactivity margin. Since the accident analyses are unchanged by the uprate, the uprated conditions have no direct effect on the overall performance capability of the ECCS. These systems will continue to deliver flow at the design basis RCS and containment pressures. The emergency core cooling systems consists of three subsystems.

4.3.3 Core Flood (CF) System

The passive portion of the system is the two Core Flood Tanks (CFT) which are connected to each of the Low Pressure Injection (LPI) lines entering the Reactor Vessel. Each CFT contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when the pressure within the RCS decreases below the operating pressure of each of the CFTs. Since the accident analyses are unchanged by the uprate, this system is not affected by the uprate.

4.3.4 High Pressure Injection (HPI) System

The HPI portion of the active part of the ECCS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Three High Pressure Injection (HPI) pumps are available to take suction from the Borated Water Storage Tank (BWST) and deliver borated water to the reactor vessel via four cold leg connections. Since the accident analyses are unchanged by the uprate, this system is not affected by the uprate.

4.3.5 Makeup Function of High Pressure Injection/Makeup System

The Makeup System function of the HPI/MU provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of letdown water from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the tube side of the Letdown Cooler and then through a letdown orifice. The Letdown Cooler reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low-pressure water leaves the reactor containment and enters the auxiliary building. After passing through one of the mixed bed purification demineralizers, where ionic impurities are removed, coolant flows through the Makeup filter and enters the Makeup tank (MUT). The water is drawn from the MUT by the in-service MU pump and returned to the RCS. The uprated conditions are bounded by the design conditions in the functional specification. Therefore, operation of the HPI/ MU System is unaffected by the uprate.

4.3.6 Low Pressure Injection (LPI) System

The LPI portion of the active part of the ECCS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two Low Pressure Injection pumps are available to take suction from the Borated Water Storage Tank (BWST) or reactor building sump and deliver borated water to the reactor vessel. Since the accident analyses are unchanged by the uprate, this system is not affected by the uprate.

4.3.7 Decay Heat Function of Low Pressure Injection/Decay Heat (DH) System

The Decay Heat function of the LPI/DH System is designed to remove sensible and decay heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the LPI/DH System is used to transfer refueling water between the BWST and the refueling canal at the beginning and end of refueling operations.

The LPI/DH System consists of two decay heat coolers, two LPI/DH System pumps and associated piping, valves and instrumentation. During system operation, coolant flows from one hot leg of the RCS to the LPI/DH System pumps, through the tube side of the decay heat coolers and back to the Reactor Vessel downcomer region via the Core Flood nozzles.

The power uprate will slightly increase decay heat. This increase is small (less than 1 percent) and well within the design capability of the LPI/DH System. Therefore, the increased decay heat will not affect the operation of the LPI/DH System.

<u>4.4 Spent Fuel Cooling (SF) System</u>

The expected decay heat load will be approximately proportional to the power increase. The fuel discharge to the spent fuel pool may have a slightly increased decay heat (less than one percent), but is still well within the design limits of the cooling system. No system design changes are required to accommodate spent fuel resulting from operation at 2568 MWt.

<u>4.5 Building Spray (BS) System</u>

The accident and reactor building analyses have been performed at power levels that bound the uprated conditions. No operating parameters of the BS System will change. Therefore, the BS System is unaffected by the uprate.

<u>4.6 Emergency Feedwater (EF) System</u>

The EF System provides water to the steam generators in the event the Main Feedwater (FW) System becomes unable to perform this function, or as required during accident conditions. Once the EF System is actuated, the Emergency Feedwater Initiation and Control (EFIC) System will automatically control the steam generator level depending on actual plant condition. The EF system was analyzed assuming a maximum power level of 2568 MWt and the resulting decay heat level. The accident analyses are not impacted by the uprate. Therefore, the EF System is unaffected by the uprate.

<u>4.7 Nuclear Steam Supply System (NSSS) Components</u>

The reactor vessel (RV) was evaluated at the uprated conditions for the structural acceptability of the vessel, and for the reactor vessel integrity in terms of the impact due to neutron fluence.

4.7.1 Reactor Vessel Structural Evaluation

The uprated 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 uprated 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.

4.7.2 Reactor Vessel Integrity (Fluence) and Alloy 600 Considerations

The revised design conditions in Table 1 can affect the fluence analyses generally in two ways. One way is that changes in T_{cold} may affect the value used in the various analysis methods. The second way is that the increase in core power can increase the neutron fluences experienced by the vessel. The current 32 Effective Full Power Year (EFPY) fluences for the CR-3 reactor vessel beltline materials are reported in FRA-ANP 86-1266133-01, "CR-3 PT Fluence Analysis Report - Cycles 7-10," (Reference 5). The reactor vessel fluence will increase with the power uprate. To bound the increase in fluence, the current 32 EFPY inside surface fluence values for the CR-3 reactor vessel beltline materials were conservatively increased by 7 percent, i.e., fluence = 1.07 x current fluence, (FRA-ANP 32-5013936-00, "Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate," Reference 6).

4.7.2.1 Heatup and Cooldown Pressure / Temperature (P-T) Limit Curves

The current P-T limit curves are valid through 32 EFPY and are based on adjusted reference temperatures (ARTs) at the ¼-thickness (¼T) and ¾-thickness (¾T) wall locations for the limiting reactor vessel beltline materials (Reference 11). These ART values were calculated in accordance with Regulatory Guide 1.99, Revision 2 (Reference 23). With the implementation of the power uprate, these calculations were re-evaluated based on a 7 percent fluence increase added to the 32 EFPY peak inside surface fluence ($E > 1.0$ MeV) using the guidelines of Regulatory Guide 1.99, Revision 2. For the CR-3 uprate, the ARTs were recalculated considering the increased neutron fluence and the updated chemical composition of the welds (Reference 7). The current P-T limit curves at 32 EFPY were calculated using ART values of 213.0°F and 144.5°F for the ¼T and ¾T wall locations, respectively. The new ART values are 195.7°F and 144.1°F for the respective wall locations. Since the ART's at both locations are less than the ARTs used in generating the current P-T limit curves, the current P-T limit curves for 32 EFPY are still valid for the planned uprate.

4.7.2.2 Surveillance Capsule Withdrawal Schedule

A B&W Owners Group 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. Since the revised fluence projections do not appreciably exceed the fluence projections used in development of the current withdrawal schedules, then the current withdrawal schedules remain valid.

4.7.2.3 Low Temperature Overpressure Protection System (LTOP)

LTOP is designed to protect the RCS from overpressure events when the RCS temperature is below 280°F. Changes to full power operating parameters, such as NSSS power, do not impact LTOP. Thus, the existing LTOP analysis is unaffected.

4.7.2.4 Pressurized Thermal Shock (PTS)

The 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 conservatively re-evaluated using a 7 percent increase in the current fluence values. These values were re-evaluated in accordance with the requirements in the Code of Federal Regulations, Title 10, Part 50.61 (10 CFR 50.61). The controlling beltline materials for the reactor vessel are the upper shell longitudinal welds, WF-8 and WF-18, with RT_{PTS} values of 206.0°F. The screening criterion for these weld metals is 270°F. Therefore, the reactor vessel remains within its limits for PTS at the uprated condition (Reference 8).

4.7.2.5 Alloy 600 Primary Water Stress Corrosion Cracking (PWSCC)

The effect of a temperature increase resulting from the power uprate on Alloy 600 PWSCC has been evaluated (Reference 9). To conservatively estimate the limiting case of 20 percent OSTG tube plugging, it is assumed that T_{hot} increased from 601.7°F to 603.3°F, which decreased the time to PWSCC initiation by 6 percent and increased the crack growth rate by 4 percent. Because the power uprate does not increase the T_{cold}, T_{avg} or the pressure, the impact is limited to Alloy 600 components and welds operating near T_{hot}. Examination of the FRA-ANP 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 therefore not affected by the power uprate. These components in the pressurizer 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 bounded by current B&WOG aging management programs for Alloy 600.

Although the pressurizer components remain limiting, an evaluation of the impact of the power uprate on Alloy 600 PWSCC on the CR-3 reactor pressure vessel head control rod drive mechanism (CRDM) nozzles was performed. The rate of PWSCC growth will increase slightly and the time to PWSCC initiation will decrease slightly due to an increase in T_{hot}. For the proposed power uprate, the T_{hot} increase is expected to be approximately 0.2°F, which will increase the growth rate and decrease time to crack initiation by less than 1 percent. This small increase will have negligible impact on the integrity of the CRDM nozzles and is bounded by the evaluation of PWSCC described above.

4.7.3 Reactor Internals

The reactor internals support and orient the fuel and control rod assemblies, absorb control rod assembly dynamic loads and transmit these and other loads to the reactor vessel. The internals also direct flows through the fuel assemblies, provide adequate cooling to various internals structures and support in-core instrumentation. The changes in the RCS temperatures, reported in Table 1, produce changes in the boundary conditions experienced by the reactor internals components. Also, increases in core power may increase nuclear heating rates in the lower core plate, upper core plate and former plate region. As described in the following section, several evaluations have been performed to demonstrate that the reactor internals can perform their intended design functions at the revised design conditions.

4.7.4 Core Support Structures and Vessel Internals

The revised design conditions were reviewed for impact on the existing design basis analyses for the core support structure and reactor vessel internals. No change in RCS design or operating pressure was made as a part of the power uprate. The conditions analyzed in the existing analyses are based on the RCS functional specification. The uprated conditions are bounded by the conditions in the RCS functional specification. 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.

4.7.5 Reactor Internals Flow Induced Vibration (FIV)

The uprate by itself does not result in an appreciable change in RCS mass flow compared to the current operation value (less than 0.1 percent -- see Table 1). The design bases analyses used conservative flow velocities which bound those resulting from the power uprate. The very slight change in mass flow has negligible impact on the components (Reference 10). Thus, core lift, and flow-induced vibration on reactor vessel internals are not significantly affected by the power uprate.

OTSG tube plugging does reduce the RCS flow. However, steam generator tube plugging does not adversely affect primary component flow induced vibration. Flow induced vibration is a result of the dynamic pressure, or the density-velocity-squared product of the flow. Since the uprated power RCS volumetric flow decreases with additional steam generator tube plugging, the existing reactor internals flow induced vibration analyses remain bounding.

4.7.6 Control Rod Assembly (CRA) Drop Time Analyses

ITS Surveillance Requirement (SR) 3.1.4.3 requires that the CRA drop time be less than or equal to 1.66 seconds to ¾ insertion. The revised design conditions, in particular the reduced T_{cold}, can increase the drop time due to the increased fluid density if the temperature change is significant. For the power uprate, T_{cold} will only decrease by 0.2°F below the current operating condition for 20 percent tube plugging. Periodic testing as required by ITS is performed prior to the reactor going critical, with all four reactor coolant pumps operating and RCS temperature greater than 525°F. Since the density change for a 0.2°F T_{cold} decrease is very small and since periodic testing is performed at conservatively low RCS temperatures to confirm drop time acceptability, no further evaluation is required.

4.7.7 Mechanical Evaluations

The uprated conditions do not affect the current design bases for seismic and LOCA loads. Thus, it was not necessary to re-evaluate the structural affects from seismic loads, and the LOCA hydraulic and dynamic loads. With regards to flow and pump induced vibration, the current analysis uses a mechanical design flow, which did not change for the revised design conditions. The revised design conditions will slightly alter the T_{cold} and T_{hot} fluid densities, which will slightly change the forces induced by flow. However, these changes are insignificant when compared to the current design temperature ranges. Thus, the uprated conditions do not affect the mechanical loads.

4.7.8 Structural Evaluations

The uprated conditions were reviewed (Reference 4) 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 or conservative flow calculations. As noted in Table 1, the uprated 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 analyses for the reactor vessel internals remain applicable for the uprated power conditions.

4.7.9 Control Rod Drive Mechanisms (CRDMs) Structural Evaluation

The uprated conditions were reviewed (Reference 4) 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. As noted in Table 1, the uprated 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 CRDMs remain applicable for the uprated power conditions. This applies to both Type "A" and "C" CRDMs.

4.7.10 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. The RCS mass flowrate increases slightly due the density increase that results from the decrease in T_{cold} (the volumetric flowrate does not increase). This effect on RCS mass flowrate is very small (less than 0.1 percent). Past fretting occurrences for Mark-B fuel has been with peripheral rows of peripheral assemblies. Relatively high cross flow velocities through LOCA slots and holes in the core baffle coupled with grid to rod gaps are considered to be the primary contributors. Since a 0.1 percent change in system mass flow rate will have little to no influence on the cross flow velocities present, there will be no additional susceptibility to grid to rod fretting. The core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, so 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.

4.7.11 Reactor Coolant Loop Piping and Supports Structural Evaluation

The uprated conditions were reviewed (Reference 4) 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. As noted in Table 1, the uprated 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.

4.7.12 Reactor Coolant Pumps (RCPs) Evaluation

The uprated conditions were reviewed (Reference 4) for impact on the existing design basis analyses for the reactor coolant pump. 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. As shown in Table 1, the uprated conditions for RCS mass flow are greater than the minimum assumed in the original design basis and therefore are bounded. 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.

4.7.13 Reactor Coolant Pump Motor Evaluation

The power uprate changes the RCS mass flow only slightly. Thus, the pump head capacity performance and net positive suction head (NPSH) requirements are virtually unchanged. With tube plugging, RCS flow decreases and the developed head increases. NPSH requirements will decrease slightly with the decreased flow.

Since the uprate will not cause a significant RCS mass flow change, the pump power requirements will not change noticeably. With tube plugging, the pump flow will change from approximately 96,585 gpm per pump to 93,040 gpm per pump. The brake horsepower requirements do not perceptibly change over this flow change. Therefore, pump power requirements will not change significantly.

The RCP motors were evaluated based on the uprated conditions for continuous operation. In addition, the uprated conditions will have no effect upon motor operation during pump start and cold loop operation. Reference 1 determined that the power uprate changes the RCS mass flow imperceptibly. Thus, the pump head capacity performance and NPSH requirements are unchanged. Therefore, FPC concludes that the uprated conditions will have a negligible impact on the RC pump and motors.

4.7.14 Once Through Steam Generators (OTSG) Thermal-Hydraulic Performance

The following evaluations and analyses were performed to assess the impact that the revised design conditions had on the thermal-hydraulic performance of the steam generators.

4.7.14.1 Steam Generator Inventory

Within the tube bundle, the water level will not appreciably change due to the power uprate as evidenced by the similar steam temperatures (pre- and post-uprate). The downcomer inventory, however, will increase due to the increased feedwater flow. This is caused by the increase in tube region unrecoverable pressure drop, which must be offset by an increased downcomer water column (inventory). Additional OTSG tube plugging will cause further increases in water level due to the increased boiling lengths required due to the reduction in heat transfer area. Inventory limits on the steam generator are based on safety analyses. The safety analyses were performed using the maximum inventory possible without flooding the aspirator ports. Thus, plant operation will continue to be limited to the current 90 percent level on the OTSG operating range to comply with the technical specification inventory limit of 96 percent level.

The trending of steam generator water levels has shown that tube support plate fouling has a dominant effect on measured water levels. So while some increase in level will occur due to the power uprate and additional OTSG tube plugging, the effects of fouling are greater than the effects of the power uprate and OTSG tube plugging.

4.7.14.2 Steam Generator Temperature

For the uprated conditions, the change in steam temperature is negligible and will not cause a change in the steady state tube-to-shell delta temperature (ΔT). While the tube-to-shell ΔT will increase with plugging, it is still within the 60°F compressive limit. In summary, the uprate will not cause steam generator design temperature values to be exceeded.

4.7.14.3 OTSG Structural Evaluation

The uprated conditions were reviewed (Reference 4) 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. As noted in Table 1, the uprated 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 OTSG remain applicable for the uprated power conditions.

4.7.14.4 OTSG Tube Integrity

The uprated conditions were reviewed (Reference 12) for impact on the existing design basis analysis for the steam generator tubes. An evaluation was performed to demonstrate that the existing structural and fatigue analyses of the steam generator tubes continue to comply with the ASME Code limits for the revised design conditions. This evaluation considered the steam generator tubes with regard to stress and fatigue usage. The evaluation demonstrated that the steam generator tubes continue to comply with the requirements of the ASME Code for the uprated design conditions.

4.7.14.5 OTSG Flow Induced Vibration (FIV)

The best estimate feedwater flow rates, as a function of steam temperature, were determined. These values are based on the uprated thermal power, a feedwater temperature of 458°F, and a steam pressure of 925 psia. The flow rates were determined at the expected steam temperature of 593°F, the current steam temperature of 593°F, and the 20 percent tube plugging temperature of 583°F. The dynamic pressure used for FIV analyses is effectively constant for all the points along the feedwater flow versus steam temperature line.

Rather than perform the subsequent uprate FIV analyses at the best estimate conditions, the analyses (Reference 13) were performed at a greater flow rate. This provides margin for instrument uncertainty, asymmetric OTSG operating levels, changes in steam pressure, and three RCP power operation. The uprated "design" flow rate was specified as 2 percent greater than the previous FIV analyzed condition.

The various degradation indications that were detected in the Crystal River OTSG tubes have been repaired with mechanical sleeves and continue to operate, while others have been taken out of service by plugging. Some of the plugged tubes were stabilized with various stabilizer designs. Both the virgin tubes, sleeved tubes, and plugged and stabilized tubes have been certified, by prior analyses, to be free of flow-induced vibration problems. At a power uprate to 2568 MWt and in consideration of 20 percent tube bundle plugging, the flow velocities in the OTSG will increase correspondingly by approximately 2 percent. The forcing function on the tubes due to fluid flow increases approximately 4 percent during full power operation. The integrity of the tubes, virgin, sleeved or stabilized, were re-assessed with the latest techniques and input parameters in a FIV analysis.

To assess the margins the tubes in the OTSGs have against detrimental flow-induced vibration effects, the tubes with the smallest margins were identified and their present margin of safety was reassessed using the results of the new FIV analyses for the increased flow rate. Because the various hardware used to repair these tubes were developed over a period of 19 years, a thorough review of the past FIV qualification of both the virgin tube, sleeve, and the different stabilizers was performed.

The reassessment shows that the original functional integrity of the installed hardware is maintained for the increased flow rate. The tube bundle in the OTSGs will have a minimum fluid-elastic stability margin of about 8 percent. The minimum margin against excessive turbulence-induced stress in the stabilized tube is about 57 percent. The frequency of tube-tube impacting is determined to be insignificant with a 2 percent increase in the cross flow velocity in all tubes.

4.7.14.6 OTSG Tube Repair Hardware

The OTSG tubes have been repaired using the following products:

- Welded Tube Plugs,
- Mechanical Tube Plugs,
- Mechanical Sleeve Plugs,
- Repair Rolls,
- Mechanical Sleeves, and
- Tube Stabilizers.

The revised operating conditions were reviewed (Reference 13) for impact on the existing qualification reports and design calculations for the repair hardware. The existing steam generator loads remain valid. The evaluations showed that the temperature changes, due to the power uprate, are bounded by those used in the repair rolls, sleeve and plug qualifications and analyses. The affect of the flow increase was also evaluated and showed that all installed tube repair hardware maintained their functional integrity with the increased secondary side flow rates. Therefore, the existing structural and fatigue analyses remain valid for the installed tube repair hardware. Thus, the existing stress reports for the mechanical and welded plugs, mechanical sleeves, and tube stabilizers remain applicable for the uprated power conditions.

4.7.14.7 Tube Plugging and Repair Criteria

Crystal River's current Steam Generator program follows the inspection guidelines contained in the latest revision of the EPRI PWR Steam Generator Examination Guidelines. The modest power uprate will not require a change to the program. Crystal River currently inspects for all active and potential degradation. The pre-outage degradation assessment includes Crystal River specific degradation as well as industry degradation. Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs will be made. Potential degradation growth rate changes will be incorporated into the operational assessment associated with potential effects of the uprate.

The revised operating conditions were reviewed (Reference 13) for impact on the existing analyses that support a 40 percent through-wall plugging criteria for a tube or sleeve. Since the RCS pressure will not change and the Steam Generator outlet pressure will increase slightly, the power uprate has insignificant impact on the existing operating pressure differential across the tube wall for either the 0 percent plugging or the 20 percent tube plugging conditions. The change in tube temperature will have an insignificant effect on the tube strength properties. The current OTSG tube pressure differentials and tube loads during Faulted Conditions remain valid. The effect of a 0.2°F temperature increase may decrease the crack initiation time by approximately 1 percent, which is not considered significant. Thus, the crack growth estimates remain valid and the current 40 percent tube/sleeve plugging criteria remains applicable for the uprated power conditions. However, the higher temperature will be considered in future growth rate analyses.

The current Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes, analyses remain valid under the uprated conditions. The uprated conditions will not impact tube inspection during future outages nor will the methodology (assumptions and parameters used) for condition monitoring and operational assessments be impacted.

4.7.15 Pressurizer Structural Evaluation

The uprated conditions were reviewed (Reference 4) for impact on the existing design basis analyses for the pressurizer. No changes in RCS design or operating pressure are required for the power uprate. The design conditions in the existing analyses are based on the RCS functional specification. The uprated conditions are bounded by the design conditions as described in Table 1. 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.

The uprated conditions were reviewed for impact on thermal stratification in the surge and spray lines. The evaluation showed that the effects of thermal stratification are either bounded by the existing analysis or negligibly affected.

4.7.16 RCS Attached Piping and Supports Structural Evaluation

The uprated conditions were reviewed (Reference 4) 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 are required 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. As noted in Table 1, the uprated 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 system attached piping and supports remain applicable for the uprated power conditions.

4.7.17 Leak Before Break (LBB)

The current LBB evaluation was performed for the RCS primary loops to provide technical justification for eliminating large primary loop pipe rupture as the structural design basis. An evaluation (Reference 4) was performed which determined the impact of the uprated conditions on the LBB margins is negligible, and the LBB conclusions remain unchanged.

4.8 Balance Of Plant (BOP) Systems

Using the uprated NSSS parameters from Table 1, a heat balance on the secondary systems has been performed for the proposed uprate. The results of the heat balance (References 1, 15) are provided in Table 2, BOP Parameters Before and After Power Uprate. The resultant conditions from the 0.9 percent upgrade were then evaluated to assure satisfactory operation of the BOP systems. The BOP systems that were reviewed (Reference 3) are those that are (or could be) directly affected by the uprate. These systems are discussed below.

Table 2: BOP Parameters Before and After Power Uprate

Parameter	Units	Before	After
Total Thermal Power (0% Tube Plugging)	MWt	2560	2584
Main Steam Flow	Mlbm/hr	10.702	10.817
Main Steam Temperature	°F	593.1	593.1
Main Steam Pressure	psia	925.0	925.0
Main Feedwater Flow	Mlbm/hr	10.702	10.817
Main Feedwater Temperature	°F	457.3	458.2
Electric Output	MWe	895.0	903.3
LP Condenser Vacuum	In. Hg	1.92	1.93
Condensate Temperature	°F	101.4	101.6

4.8.1 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 EF System as required for accidents and provides capability for RCS cooldown following a steam generator tube rupture event. As discussed previously, the accident analyses are not impacted by the uprate. No changes in design are required. Therefore, the safety functions of this system are not impacted by the uprate.

The MS System also functions during normal operation. As shown on Table 2, MS temperature and pressure will not be affected by the power uprate. The MS flow will increase with increasing power, but is still within the capabilities of the existing system. It is concluded from a review of this system that no changes in design are needed and that all parameters remain within design requirements.

4.8.2 Main Steam and Main Feedwater Piping

The uprated conditions were reviewed (Reference 14) for impact on the existing design basis analyses for the main steam and main feedwater piping and supports. No changes in OTSG design or operating pressure are required as part of the power uprate. The changes in the operating temperatures and flow rates due to the 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.

4.8.3 Auxiliary Steam (AS) System

The AS System provides a pathway for steam between the MS System and the Emergency Feedwater (EFW) Pump Turbine for emergency operation of the EFW pump. The accident analyses are not impacted by the uprate. No changes in design are required. Therefore, this system is not impacted by the uprate.

4.8.4 Condensate (CD) System

The primary function of the CD System is to supply preheated condensate to the FW System. The Condensate System pressure, temperature and flow rate will change slightly at the uprated power. The condensate pumps have sufficient margin to accommodate the uprate. The condenser was reviewed for the uprated conditions and found acceptable.

4.8.5 Condenser Air Removal (AR) System

The AR System removes non-condensable gases from the condenser to maintain maximum vacuum and thus maintain plant efficiency. The higher power level will not result in an increased air removal burden. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.6 Condensate Polishing Demineralizer (CX)

The CX System removes dissolved solids, corrosion products and suspended solids by ion exchange and filtering through the beds of ion exchange resins. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.7 Feedwater (FW)

The FW System provides isolation capability of the feedwater during accidents. It also provides feedwater to the OTSGs during normal operation. The accident analyses are not impacted by the uprate. In addition, the main feedwater pumps and the booster pumps are currently operating at 80 and 84 percent of capacity, respectively. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.8 Auxiliary Feedwater (AFW)

Auxiliary Feedwater is provided via an Auxiliary Feedwater Pump. AFW is a non-safety related system and is not required to perform any accident mitigation functions. The AFW system is capable of providing the equivalent of 100 percent of the flow of an EFW pump. The sizing of the EFW pumps was based on the decay heat for operation at 2568 MWt. Therefore, the AFW pumps will remain adequate to provide a backup source of secondary cooling after the uprate.

4.8.9 Feedwater Heaters

The Feedwater Heaters provide preheat of the feedwater prior to the feedwater entering the OTSGs. The current system is capable of providing feedwater heating requirements including the increase in demand (an additional ΔT of 0.7°F). Therefore, this system does not need to be modified to accommodate the power uprate.

4.8.10 Main Turbine

The main turbine is designed to accommodate the steam flow resulting from the governor valves wide open with design input pressure and design condenser backpressure. The uprated conditions maintain steam pressure constant. The condenser backpressure will rise slightly as part of the uprate. Currently, the turbine is operating with the fourth governor valve less than 100 percent open providing margin for the uprate. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.11 Generator Gas (GG) System

The GG System is designed to remove heat from the windings of the Main Generator and transfer that heat to the Secondary Services Closed Cycle Cooling System. The increase in power will result in an increase in load by less than 2 percent. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.12 Moisture Separator/Reheater

The Moisture Separator/Reheater (MSR) is designed to remove moisture and reheat the resultant dry steam to a superheated state over two stages. The MSRs were provided with the turbine, which was designed to accommodate full steam flow with the throttle and governor valves full open. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.13 Extraction Steam (EX) System

The EX System piping, instrumentation and controls provide steam from the turbine to the Moisture Separator Reheater Low Pressure Tube Bundles as well as steam from the low and high pressure turbines to the Condensate/Feedwater Heaters. It is concluded from a review of this system that no changes in design are required and that all parameters remain within design.

4.8.14 Heater Drain (HD) System

The HD System provides for drainage of the MSRs and Feedwater Heaters. The current system has been determined to be adequate and is not affected by the power uprate.

4.8.15 Circulating Water (CW) System

The CW System provides seawater as cooling to the main condenser and the SC System. It is concluded from a review of this system that no changes in design are required. There will be a slight increase in discharge temperature. This may result in slightly more frequent operation of the helper cooling towers on the discharge canal during warm summer months. No existing temperature limits will be exceeded.

4.8.16 Decay Heat Closed Cycle Cooling (DC) System

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 RW system. As discussed previously, the accident analyses are not impacted by the uprate. No changes in design are required. Therefore, this system is not impacted by the uprate.

4.8.17 Nuclear Services Closed Cycle Cooling (SW) System

The SW System removes heat from various safety-related equipment following ES actuation and transfers this heat to the RW system. As discussed previously, the accident analyses are not impacted by the uprate. No changes in design are required. Therefore, this system is not impacted by the uprate.

4.8.18 Nuclear Services & Decay Heat Seawater (RW) System

The RW System provides cooling water to the SW and DC System for heat removal during accidents and normal operation. As discussed previously, the accident analyses are not impacted by the uprate. No changes in design are required. Therefore, this system is not impacted by the uprate.

4.8.19 Secondary Services Closed Cycle Cooling (SC) System

The SC System provides cooling water flow to various secondary plant heat loads including the Turbine Generator, Condensate pumps, Lube Oil Coolers, Air Removal System, Instrument Air System and Station Air System. It also provides backup cooling to Control Complex Chillers upon loss of SW. The SC cooling system can limit Turbine Generator output when the heat sink (Gulf of Mexico) temperatures are high and heat load is higher than normal (e.g., high reactive load on generator). CR-3 monitors the temperature of loads serviced by SC. If equipment approaches its design limits, the generator power may need to be reduced. Therefore, under some extreme circumstances, the full benefit of the power uprate may not be realized. System operational enhancements and modifications are being considered to eliminate this potential restriction. Under most operating conditions, the SC System will continue to perform its function at the uprated conditions. The limitations of this system have potential economic consequences due to reduced power generation but do not adversely impact the loads serviced by the SC system or impact plant safety.

4.8.20 Main Reactor Building Fans (AH-XA) System

The AH-XA System maintains reactor building integrity by reducing temperature and pressure inside containment following an accident. It also provides for building cooling during normal operation. As discussed previously, the accident analyses are not impacted by the uprate. Also, since NSSS T_{avg} and pressure are unchanged, normal operating heat loads are not affected. No changes in design are required. Therefore, this system is not impacted by the uprate.

4.9 Electrical Systems

4.9.1 Main Generator

The main generator is a four-pole machine rated at 989.4 MVA and 22 KV with a 0.90 power factor. This rating is based upon 60-psig hydrogen pressure that is supplemented with cooling water for the stator. At the current thermal rating of 2544 MWt, the peak efficiency main generator electrical output is typically 895 MWe. The turbine generator and auxiliaries have been evaluated for operation at the uprated conditions. A review of the applicable generator reactive capability curve confirms that the main generator is capable of operating at a maximum real power output of 989.4 MWe at a 1.0 power factor (zero megavar output). Heat balance studies completed for the uprate identify gross generator output levels less than this maximum (approximately 903 MWe). Machine operation at a lower real output power level and a power factor of 1.0, or less, is permissible provided unit operation remains within the real and reactive power limits defined by the reactive capability curve. No modifications will be required to generate the electrical power associated with the 24 MWt increase.

4.9.2 Generator Isolated Phase Bus Duct (TB)

The Isolated Phase Bus Duct connects the output of the generator with the primaries of the Step-Up and Unit Auxiliary transformers. The power uprate is expected to increase power from about 895 MW at 0.9 power factor to about 903 MW at about 0.91 power factor. The isolated phase bus duct and associated cooling equipment are designed to accept the maximum generator output and therefore will continue to support plant operations at uprated conditions. Therefore, this Isolated Phase Bus Duct supports the uprate to 2568 MWt.

4.9.3 Step-Up and Auxiliary Transformers (MT)

The Main Power Transformers (Step-Up) consist of three single-phase 316.667 MVA units to form a three phase bank with a nominal rating of 950 MVA at 65°C FOA, average rise per single-phase unit. This transformer bank is the main power outlet for the main generator. The transformer receives all output of the generator. With an expected power uprate to about 903 MWe, the increase is within the design limits of the transformer. Therefore, this system supports the uprate to 2568 MWt. The Unit Auxiliary Transformer (UAT) is capable of handling full in-house loads before and after the power uprate.

4.9.4 Emergency Diesel Generator (EDG)

The Emergency Diesel Generators supplies the source of power following a loss of offsite power or degraded grid voltage conditions. The EDG automatically provides AC electrical power to the 4160 volt Engineered Safeguards busses 3A and 3B in order to provide motive and control power to equipment required for safe shutdown of the plant and mitigation and control of accidents. As discussed previously, the accident analyses are not impacted by the uprate. No changes in design are required. Therefore, this system is not impacted by the uprate.

4.9.5 Motor Feeders

The Motor Feeder System was reviewed to determine the impact of the uprate on the system. All components remain within their design limits and no changes in design are required. Therefore, this system supports the uprate to 2568 MWt.

4.9.6 DC Electrical Battery (DP) System

The DC electrical distribution systems were reviewed to identify the major items that may be affected by uprated conditions and to evaluate the potential impact of an uprate on that equipment. System reviews confirmed that no DC powered loads were affected by unit operation at uprated conditions. Additionally, the reviews confirmed that the DC control power for AC loads remained unchanged. Therefore, the DP System is unaffected by unit operation at uprated conditions.

4.9.7 Switchyard and Grid Stability

CR-3's offsite power for the Start-Up, Off-Site Power and Backup Engineered Safeguards Transformers are supplied by a 230 KV switchyard. The CR-3 generator outputs to a step-up transformer which is connected to a separate 500 KV switchyard. Changes to the generator output will not impact the 230 KV switchyard, which supplies all normal and ES busses. The small increase in load to the 500 KV switchyard has been evaluated and determined to be acceptable. The CR-3 power uprate is sufficiently small that it will have no significant impact on grid stability.

4.10 Control Systems And Instrumentation

4.10.1 Anticipated Transients Without Scram (ATWS) System

The Anticipated Transients Without Scram System is comprised of two discrete functions: Diverse Scram System and ATWS Mitigating System Actuation Circuitry. As discussed previously, the existing accident analyses are unchanged. No changes to the design of this system or to the setpoints are required for this uprate. Therefore, this system supports the uprate to 2568 MWt.

4.10.2 Engineered Safeguards Actuation (ES) System

The ES System consists of three independent instrument channels for detection of conditions indicative of an accident and two channels for activation of equipment required to mitigate the consequences of an accident. As discussed previously, the existing accident analyses are unchanged. No changes to the design of this system or to the setpoints are required for this uprate. Therefore, this system supports the uprate to 2568 MWt.

4.10.3 Emergency Feedwater (EF) Initiation and Control (EFIC) System

The EF System provides coolant to the steam generators in the event the Main Feedwater System becomes unable to perform this function, or when emergency feedwater is necessary to provide natural circulation in the RCS. Once the EF System is actuated, the EFIC System will automatically control the steam generator level at one of three possible setpoints, depending on the actual plant conditions. As discussed previously, the existing accident analyses are unchanged. No changes to the design of this system or to the setpoints are required for this uprate. Therefore, this system supports the uprate to 2568 MWt.

4.10.4 Integrated Control System (ICS)

The ICS is a non-safety system that automatically controls the station in response to commands preset by the operator. The ICS provides control rod motion, feedwater control and turbine control when in the fully automatic mode. The operator is provided the capability for manual override control of the station. Software changes will be made to the Automatic Unit Load demand (AULD) programmable logic controller and minor adjustments to several ICS modules will be required to reflect the increased power output of the plant, including an increase in feedwater flow. The required changes can be made while the unit is at power. These changes do not impact the overall operation of the ICS. The ICS also uses percentages of RTP for runback setpoints. These values are nominal and are not affected by the small increase in RTP. Therefore, this system supports the uprate to 2568 MWt.

4.10.5 Non-Nuclear Instrumentation (NN) System

The NN System consists of non-safety process variable sensors, signal processing equipment, and a means of selecting and/or transmitting the derived signals for use by the plant. These signals are input to control and computer systems for monitoring and indication. The power uprate does not impact the performance of this system. Therefore, this system supports the uprate to 2568 MWt.

4.10.6 Nuclear Instrumentation (NI) System

The NI System is comprised of three subsystems: the Incore Monitoring System, the Excore Monitoring System and the Gamma Metrics System. The subsystems are physically separated from each other and are separate in function as well. Each has safety related and non-safety related functions. No design changes are required for these systems. While some minor calibrations will be required, the power uprate does not impact the overall performance of this system. The required changes can be made while the unit is at power. Therefore, this system supports the uprate to 2568 MWt.

4.10.7 Reactor Protection (RP) System

The RP System consists of four redundant instrumentation channels that monitor process parameters related to safe operation. No changes to the RP design or ITS trip setpoints is required. As mentioned earlier, some of the RP setpoints are based on the definition of RTP. These setpoints will not need to be revised because they are based on a percentage of RTP. Several of these will effectively be changed when the new 100 percent power level is set at 2568 MWt. The required changes can be made while the unit is at power. The overpower trip setpoint based on flow and imbalance (ITS 3.3.1, Table 3.3.1-1, Item 8, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE) is located in the Core Operating Limits Report (COLR). The Cycle 13 COLR was developed assuming operation at 2568 MWt, therefore, no changes to the existing COLR will be required. In addition, no changes to the accident analyses are required since they were performed assuming that 100 percent RTP was 2568 MWt or higher.

4.11 Accident Analysis Evaluation

Chapter 14 of the Final safety Analysis Report (FSAR) (Reference 16) has been reviewed for the impact of the uprate on the safety analyses. It has been determined that all FSAR and supporting analyses bound the uprate, or in the case of the pump coastdown events, are reevaluated each reload for cycle specific considerations (Reference 17). The Cycle 13 pump coastdown events have been analyzed at 102 percent of 2568 MWt, as part of the normal reload process. All analyses use previously approved NRC methods and codes. Table 3 provides a summary matrix of the accident analyses. Descriptions of each event and discussions of the impact of the power uprate, if any, follow the table.

Table 3: Accident Analysis Summary

FSAR Section	Accident	Analysis Power Level (MWt)	Uprate Bounded
14.1.2.1	Uncompensated Operating Reactivity Changes	2575	Yes
14.1.2.2	Startup Accident	0	Yes
14.1.2.3	Rod Withdrawal at Power	2568	Yes
14.1.2.4	Moderator Dilution From Full Power	2619	Yes
14.1.2.4	Moderator Dilution Accident During Refueling	0*	Yes
14.1.2.5	Cold Water Accident	1284	Yes
14.1.2.6	Single Pump Coastdown	2619	Yes
14.1.2.6	Locked Rotor	2619	Yes
14.1.2.6	Four-Pump Coastdown	2619	Yes
14.1.2.7	Stuck-Out, Stuck-In, or Dropped Control Rod Accident	2568	Yes
14.1.2.8	Load Rejection/Turbine Trip	2876	Yes
14.1.2.9	Station Blackout	2772	Yes
14.1.2.9	Loss of AC Power	2568	Yes
14.2.2.1	Steam Line Failure Accident	2568	Yes
14.2.2.2	Steam Generator Tube Rupture	2568/2619	Yes
14.2.2.3	Fuel Handling Accident	2619	Yes
14.2.2.4	Hot Zero Power Rod Ejection	0*	Yes
14.2.2.4	Full Power Rod Ejection	2568/2619	Yes
14.2.2.5	Loss-of-Coolant Accidents	2619	Yes
14.2.2.6	Makeup System Letdown Line Failure	2619	Yes
14.2.2.7	Maximum Hypothetical Accident	2619	Yes
14.2.2.8	Waste Gas Tank Rupture Accident	N/A	Yes
14.2.2.9	Loss of Main Feedwater	2619	Yes
14.2.2.9	Total Loss of Feedwater Accident	2827.4	Yes
14.2.2.9	Feedwater Line Break	2619	Yes
N/A	ATWS (LOFW)	2772	Yes
N/A	AMSAC	1272	Yes

*The actual core power modeled is 2.568 watts for the zero power, point kinetics solution.

4.11.1 Uncompensated Operating Reactivity Changes (FSAR Section 14.1.2.1)

Uncompensated reactivity changes occur 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. Automatic control systems are designed to compensate for fuel depletion, burnable poison depletion, and changes in fission product poison concentration due to power level changes. Uncompensated reactivity changes are classified as a normal operational occurrence. The acceptance criteria are that the rate of reactivity addition will be much less than the rate at which the operator can compensate for the addition and that the rate of temperature change will be much less than the rate at which the automatic control system can compensate for the change.

This accident was originally analyzed to demonstrate the slow evolution of this type of event. The ability of the control systems and operators to compensate for these changes was of prime importance in early licensing stages. Following many cycles of operation, the control systems and operators have demonstrated their capabilities to respond to these core reactivity changes.

The reactivity changes for fuel depletion and xenon buildup result in negative reactivity additions to the core. These additions will lead to power reductions if compensating actions are not taken. During normal operation, the control system will take action to increase the core reactivity by an equal amount to maintain a constant power level. The reactivity changes due to xenon burnup result in a positive reactivity addition to the core. This addition will lead to a power increase and a corresponding average coolant temperature increase if left uncompensated. During normal operation, the control system will take action to decrease the core reactivity by an amount equal to the reactivity addition to maintain a constant power level and constant average temperature.

The plant and control system response to reactivity changes resulting from fuel depletion, burnable poison depletion, and changes in fission product poison concentration was analyzed at 2575 MWt. Therefore, the current analyses of uncompensated reactivity changes support CR-3 operation at 2568 MWt.

4.11.2 Startup Accident (FSAR Section 14.1.2.2)

The Startup Accident is a postulated event resulting in withdrawal of control rods with the reactor subcritical at zero power. The event is classified as a moderate frequency event. The acceptance criteria for the Startup Accident relate to peak RCS pressure, peak thermal power, and minimum departure from nucleate boiling Ratio (DNBR).

The withdrawal of control rods with the reactor subcritical at zero power decreases the shutdown margin and eventually results in core criticality. Core power increases rapidly and a primary to secondary heat mismatch develops. The RCS pressure and temperature increase and a reactor trip is initiated on high RCS pressure or high flux. A reactor trip and steam relief through the pressurizer safety valves (PSVs) limits the increase in RCS pressure and temperature. The event is terminated by the reactor trip.

The Startup Accident is initiated from near zero power and is not directly affected by the initial core power level. The high flux setpoint is based on percent of rated power. The Startup Accident analysis of record for CR-3 modeled a high flux setpoint of 112 percent of 2568 MWt. Also, any changes in core parameters such as Doppler coefficient and moderator temperature coefficient that result from the core uprate will be evaluated as part of the normal reload process. The reload evaluation will ensure parameters are within those assumed in the Startup Accident analysis. Therefore, the Startup Accident analysis supports CR-3 operation at 2568 MWt.

4.11.3 Rod Withdrawal at Rated Power Operation (FSAR Section 14.1.2.3)

A Rod Withdrawal at Power accident is the result of the uncontrolled withdrawal of a control rod group while the reactor is operating at rated power. The Rod Withdrawal at Power accident is classified as a moderate frequency event. The acceptance criteria relate to peak RCS pressure and peak core thermal power.

The withdrawal of a control rod group at power, caused by either operator error or equipment failure, results in positive reactivity addition. As the positive reactivity addition increases, core power level increases. The increase in core power causes fuel rod temperatures to rise and increases the heat transferred to the reactor coolant. The increase in core power creates a mismatch between core power generation and secondary heat removal. The heat mismatch causes reactor coolant temperature and pressure to increase. The transient is terminated by a reactor trip on high RCS pressure or high flux. The reactor trip limits the peak core thermal power to an acceptable level. The reactor trip and subsequent steam relief through the PSVs ensures that the peak primary pressure meets the acceptance criterion.

The Rod Withdrawal at Power accident is analyzed at 2568 MWt. The analysis methodology includes a spectrum of reactivity insertion rates chosen to bound the insertion rates produced by withdrawal of the lowest worth rod group and the highest worth group. Additional sensitivity studies were analyzed including Doppler coefficient, moderator coefficient, and trip delay time. These sensitivity studies coupled with bounding reactivity insertion rates defined a bounding analysis. The analyses results indicate that the highest peak RCS pressure and core thermal power is predicted for the case where the high RCS pressure trip and high flux trip arrive at the same time.

Since the initial core power level for the Rod Withdrawal at Power accident analyses is 2568 MWt, the accident analyses support CR-3 operation at 2568 MWt.

4.11.4 Moderator Dilution Accident From Full Power (FSAR Section 14.1.2.4)

The Moderator Dilution Accident From Full Power is an event resulting in reduction of the boron concentration in the primary coolant. The event is classified as a moderate frequency event. The acceptance criteria for the Moderator Dilution Accident relate to peak RCS pressure, peak thermal power, and minimum DNBR.

The decrease in boron concentration causes a positive reactivity addition and an increase in core power. The increase in core power creates a mismatch between heat removed by the steam generators and heat added by the core. As a result, RCS pressure and temperature increase and eventually a reactor trip on high RCS pressure or high flux is reached. A reactor trip and steam relief through the main steam safety valves limits the increase in RCS pressure and temperature.

The Moderator Dilution Accident was analyzed at 102 percent of 2568 MWt, or 2619 MWt. Therefore, the current analysis supports CR-3 operation at 2568 MWt.

4.11.5 Moderator Dilution Accident During Refueling (FSAR Section 14.1.2.4)

The Moderator Dilution Accident During Refueling is an event resulting in reduction of the boron concentration in the primary coolant during refueling. The event is classified as a moderate frequency event. The acceptance criterion for the Moderator Dilution Accident relates to minimum shutdown margin.

The decrease in boron concentration causes a positive reactivity addition and a decrease in shutdown margin. If the dilution source is not isolated, shutdown margin may be lost and the core could reach criticality. During refueling, the available volume of deborated fluid is limited. The volume of deborated fluid is mixed with available RCS fluid and a homogenous boron concentration is determined. The new boron concentration is compared to the value required to maintain the shutdown margin.

The Moderator Dilution Accident During Refueling is evaluated at shutdown conditions and is not affected by core power directly. Any change in boron concentration required to maintain shutdown conditions with an operational core power of 2568 MWt, or higher, will be accounted for in the core reload calculations. The core power level will not affect the results of the Moderator Dilution Accident During Refueling. Therefore, the current analysis supports CR-3 operation at 2568 MWt.

4.11.6 Cold Water Accident (FSAR Section 14.1.2.5)

The Cold Water Accident is analyzed as the startup of two idle RCPs from a reduced power level. The Cold Water Accident is classified as a moderate frequency event. The acceptance criteria are related to peak RCS pressure, peak thermal power, and minimum DNBR.

Even though there is a licensing restriction that prohibits the plant from being critical with less than three reactor coolant pumps operating, the analysis assumed that the plant was operating with one reactor coolant pump in each loop at 50 percent of rated power when the remaining two pumps were started. The increase in primary coolant flow and negative reactivity coefficients result 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 below the rated core power. The increase in coolant flow combined with an increase in power does not result in an unacceptable minimum DNBR. The RCS pressure remains below the high pressure reactor trip setpoint.

The Cold Water Accident is analyzed from 50 percent of 2568 MWt. The core power increases to approximately 80 percent with the startup of an inactive pump. The startup of an inactive pump causes colder fluid in the idle loop to mix with the reactor coolant from the other loops. The colder fluid from the idle loop causes a reduction in the average reactor coolant (RC) temperature. The decrease in the average RC temperature coupled with the negative moderator temperature coefficient causes the core power increase. The reactor is controlled to a constant average RC temperature above 25 percent power. The startup of an inactive pump causes a reduction in the average RC temperature based on the temperature of fluid in the idle cold leg and not based on initial power level. Therefore, the power increase due to the startup of an inactive loop from a higher power level will be similar to the current analysis.

In addition, the licensing restriction on CR-3 operation specifies that three RCPs must be operating when the reactor is critical. The licensing restriction limits the consequences of the Cold Water Accident to the startup of one RCP. Therefore, the Cold Water Accident analysis supports CR-3 operation at 2568 MWt.

4.11.7 Loss of Coolant Flow (FSAR Section 14.1.2.6)

4.11.7.1 Single-Pump Coastdown

The Single-Pump Coastdown event is the loss of one RCP from full power. The Single-Pump Coastdown event is classified as a moderate frequency event. The acceptance criterion relates to minimum DNBR.

The Single-Pump Coastdown event can result from a loss of power to one of the RCPs or a mechanical failure of an RCP. The coastdown of an RCP results in a decrease in reactor coolant flow. Core power remains constant as reactor coolant flow decreases. With a constant core power and decreasing flow, the DNBR decreases to a minimum at the time of control rod insertion due to a reactor trip on power/pump monitors or flux/flow. The reactor trip decreases core power and the DNBR increases again. Decay heat is removed by forced circulation of the three operating RCPs.

The RCS response to the Single-Pump Coastdown event was originally analyzed at 2544 MWt for the FSAR. A later analysis at 102 percent of 2568 MWt examined the flow characteristics for up to 20 percent equivalent tube plugging. The RCS response (core power, coolant temperature, RCS pressure, and coolant flow rate) is utilized to calculate the minimum DNBR reached during the event. The core power response is normalized to the initial full power value and provided as input to the core analysis. The initial coolant average temperature and RCS pressure will not change with the core uprate. The flow coastdown is a function of the RCP inertia and not initial core power. Therefore, starting the event at an initial power of 2568 MWt will not affect system response as far as the evolution of coolant temperature, RCS pressure, and coolant flow. This event is analyzed each reload. For the current reload, an uprated power level of 102 percent of 2568 MWt was used in the reload DNB analysis. Therefore, the analysis for the Single-Pump Coastdown event supports CR-3 operation at 2568 MWt.

4.11.7.2 Locked Rotor

The Locked Rotor event is seizure of an RCP shaft due to mechanical failure or blockage from full power. The Locked Rotor event is classified as a limiting fault event. The acceptance criterion relates to minimum DNBR.

The locked rotor event occurs when the rotor of an RCP seizes. When the rotor seizes, forced flow is no longer provided by the affected RCP. The locked rotor event results in a reduction in reactor coolant flow. The reduction in reactor coolant flow coupled with a constant core power decreases the DNBR to a minimum at the time of control rod insertion due to a reactor trip on flux/delta flux/flow (Nuclear Overpower RCS Flow and Measured Axial Power Imbalance). The reactor trip decreases core power and the DNBR increases again. Decay heat is removed by forced circulation of the three operating RCPs.

The RCS response to the Locked Rotor event was analyzed at 102 percent of 2568 MWt with 20 percent equivalent tube plugging. Therefore, the current RCS response to Locked Rotor event supports CR-3 operation at 2568 MWt.

4.11.7.3 Four-Pump Coastdown

The Four-Pump Coastdown event is the complete loss of forced flow in the reactor coolant system from full power. The Four-Pump Coastdown event is classified as an infrequent event. The acceptance criterion relates to minimum DNBR.

The Four-Pump Coastdown event can result from a loss of power to the four RCPs or a common mechanical failure of four RCPs. A simultaneous common mechanical failure of four RCPs is not considered a credible event. The coastdown of the four RCPs results in a rapid decrease in reactor coolant flow. Core power remains constant as reactor coolant flow decreases. With a constant core power and decreasing flow, the DNBR decreases to a minimum at the time of control rod insertion due to a reactor trip on power/pump monitors or flux/delta flux/flow. The reactor trip decreases core power and the DNBR increases again. Decay heat is removed by natural circulation.

The RCS response to the Four-Pump Coastdown event was originally analyzed at 2544 MWt for the FSAR. A later analysis at 102 percent of 2568 MWt examined the flow characteristics for up to 20 percent equivalent tube plugging. The RCS response (core power, coolant temperature, RCS pressure, and coolant flow rate) is utilized to calculate the minimum DNBR reached during the event. The core power response is normalized to the initial full power value and provided as input to the core analysis. The initial coolant average temperature and RCS pressure will not change with the core uprate. The flow coastdown is a function of the RCP inertia and not initial core power. Therefore, starting the event at an initial power level of 2568 MWt will not affect system response as far as the evolution of coolant temperature, RCS pressure, and coolant flow. For the power uprate, the Four-Pump Coastdown reload DNB analysis was analyzed at 102 percent of 2568 MWt. Therefore, the reload analysis and the current RCS response to the Four-Pump Coastdown event supports CR-3 operation at 2568 MWt.

4.11.8 Stuck or Dropped Control Rod Accident (FSAR Section 14.1.2.7)

The Stuck-Out, Stuck-In, or Dropped Control Rod Accident refers to a control rod misalignment from its proper location. Control rod misalignment can occur on reactor trip if one rod fails to insert and remains stuck in the fully withdrawn position. Control rod misalignment can occur during withdrawal of the control rods if one rod becomes stuck at some position as the other rods continue in motion. Control rod misalignment can occur when a control rod drops into the core due to an electrical or mechanical failure. The Stuck-Out, Stuck-In, or Dropped Control Rod Accident is classified as a moderate frequency event. The acceptance criteria are related to minimum DNBR and peak RCS pressure.

Control rod misalignment caused by failure of a rod to fall on reactor trip is evaluated to ensure that the remaining reactivity contained in the tripped control rods is still sufficient to maintain the reactor subcritical in a hot shutdown condition. This type of control rod misalignment is not dependent on initial core power level.

A dropped control rod is defined as the deviation of a control rod from the average group position by more than an indicated 5 inches (equivalent to a 9 inch absolute error). This definition then covers the action of a stuck-in control rod during withdrawal of the others and a dropped control rod. A stuck-in rod is less limiting due to the time required to raise the control rods. The term "dropped rod" refers to a stuck-in or dropped control rod assembly for the remainder of this section.

Dropping a control rod into the core from full power causes a rapid reduction in power and temperature due to the negative reactivity addition to the core. The magnitude of the power and coolant temperature decrease is a function of the reactivity worth of the dropped rod. The reduction in coolant temperature combined with the negative moderator temperature coefficient provides positive reactivity addition to the core, and the power increases again. The magnitude of the return to power, in consideration of the asymmetric power distribution, could lead to fuel rods experiencing departure from nucleate boiling (DNB). The peak power, coolant temperature, coolant flow rate, and RCS pressure are used as input to core DNB analysis to verify that no fuel pins experience DNB.

The original dropped rod analysis was performed at 2568 MWt. These analyses were representative of beginning and end of life conditions. As a result of Preliminary Safety Concern (PSC) 15-74, a more limiting time in life, near middle of life, was identified. New analyses were performed to determine the core power response for a given dropped rod worth. The normalized power response, along with coolant temperature coolant flow rate, and RCS pressure are provided as input into cycle-specific core DNB analysis to ensure that the event acceptance criteria would not be exceeded. A conservative system response to a dropped rod is used for evaluating the cycle-specific core designs using NRC approved statistical methods. The cycle-specific analyses use a normalized power from the system analysis and apply the appropriate conservatism for core power, i.e., accounts for a 2 percent RTP heat balance error. Therefore, the Stuck-Out, Stuck-In, or Dropped Control Rod accident analyses support CR-3 operation at 2568 MWt.

4.11.9 Load Rejection/Turbine Trip (FSAR Section 14.1.2.8)

The Load Rejection accident occurs under circumstances where the external transmission system deteriorates, as indicated by unit frequency deviation. Under these conditions, the unit will automatically disconnect from the transmission system. The Load Rejection accident is classified as a moderate frequency event. The acceptance criteria are related to peak RCS pressure, minimum DNBR, and offsite doses (original plant design).

The load rejection results in a rapid decrease in steam flow to the turbine. The original plant design included features that enabled the plant to runback reactor power and avoid a reactor trip. The current plant configuration with the pressurizer power operated relief valve (PORV) setpoint above the reactor trip setpoint leads to a reactor trip on high RCS pressure for load rejections. The plant response to a load rejection under the current configuration is similar to a turbine trip event except that it is less severe. The turbine trip is analyzed in place of the load rejection for the current plant configuration.

A turbine trip results in rapid increase in secondary system pressure. The increase in secondary pressure causes a rapid decrease in heat removal from the primary system. The decreased secondary heat removal causes an increase in reactor coolant pressure and temperature. The RCS pressure and temperature increase and a reactor trip on high RCS pressure occurs. The reactor trip coupled with steam release through the MSSVs and PSVs limits the peak RCS pressure to less than the acceptance criterion. The RCS pressure increases during the event and forced primary coolant flow is maintained, therefore, the minimum DNBR remains above the acceptance criterion.

The Turbine Trip accident was analyzed at a power level of 112 percent of 2568 MWt, or 2876 MWt, as part of an B&W Owners Group Program. Therefore, the Turbine Trip analyses support CR-3 operation at 2568 MWt.

4.11.10 Station Blackout Accident / Loss of AC Power (FSAR Section 14.1.2.9)

4.11.10.1 Station Blackout

The Station Blackout event (NUMARC 87-00) is a complete loss of all unit AC power and is considered beyond the original design basis of the plant. It is analyzed to determine if the plant can withstand a station blackout for four hours and recover from the event. The acceptance criteria for this accident are related to peak reactor coolant system (RCS) pressure, minimum departure from nucleate boiling ratio (DNBR), and adequate core cooling.

The complete loss of all unit AC power causes the reactor, turbine, RC pumps, main feedwater pumps, and condensate booster pumps to trip. Decay heat is removed by natural circulation heat transfer to the steam generators with steam relief through the main steam safety valves (MSSVs) and atmospheric dump valves (ADVs). Heat removal by natural circulation with steam relief through the MSSVs and ADVs is adequate to prevent occurrence of fuel damage and prevent excessive RCS pressures and temperatures. The feedwater for decay heat removal through the steam generators is supplied by the Emergency Feedwater (EF) System. The turbine driven emergency feedwater pump (EFP-2) takes suction from the dedicated Emergency Feedwater Tank (EFT-2) and is driven by steam from either or both steam generators. The Class 1E DC system station batteries supply the power for all required indication and control. The diesel driven Emergency Feedwater Pump (EFP-3) also takes suction from EFT-2 and the diesel is started from safety-grade compressed air receivers.

The loss of all AC power results in a loss of RCP seal injection flow. Primary coolant is lost at a specified rate through the RCP seals and RCS inventory decreases throughout the event. The analysis shows that sufficient RCS inventory is maintained to keep the core covered and maintain adequate core cooling throughout the 4-hour coping period. Once AC power is restored, RCP seal injection flow is reestablished. The loss of reactor coolant is terminated and core decay is removed by emergency feedwater with subsequent steam relief through the ADVs.

The Station Blackout event analysis was performed at a power level of 2772 MWt. Therefore, the Station Blackout analyses support CR-3 operation at 2568 MWt.

4.11.10.2 Loss of AC Power

The Loss of AC Power event is initiated by a loss of onsite and offsite AC power. However, limited AC power is supplied to vital equipment and instruments following startup of the plants emergency diesel generators. Historically, it was analyzed to show the plant can transition from power operation to natural circulation cooling. The Loss of AC Power event is classified as a moderate frequency event. The acceptance criteria for this accident are related to peak reactor coolant system (RCS) pressure, minimum DNBR, and natural circulation decay heat removal.

Following the loss of AC power, the turbine stop valves close. The increase in secondary pressure caused by closure of the turbine stop valves results in a decrease in secondary heat removal capacity which results in an initial reactor coolant heatup and pressure increase. Due to the loss of power, the control rods are released and fall into the core, terminating the initial heatup. Steam relief to the condenser via the turbine bypass valves is unavailable due to the inability to maintain a vacuum in the condenser. Therefore, reactor coolant pressures and temperatures are controlled by heat removal via steam relief through the main steam safety and atmospheric dump valves. Excess steam is relieved until the reactor coolant system pressure is below the pressure corresponding to the setpoint of the atmospheric dump valves. Thereafter, the atmospheric dump valves are lifted as needed to remove decay heat.

The reactor coolant system flow decreases after power to the RC pumps is lost. Since the reactor power is reduced due to gravity insertion of the control rods, the decrease in flow does not result in DNB in the core and no fuel damage occurs. Decay heat removal after RC pump coastdown is provided by the natural circulation characteristics of the system.

The concerns related to a Loss of AC Power event are peak RCS pressure, minimum DNBR, and decay heat removal capability under natural circulation. This event was analyzed at 2568 MWt and thus supports the uprate to 2568 MWt.

4.11.11 Steam Line Failure Accident (FSAR Section 14.2.2.1)

The Steam Line Failure Accident is defined as the rupture of a steam line between the steam generator and the turbine. The Steam Line Break Accident is classified as a limiting fault event. The acceptance criteria related to the event are listed below:

a. The core shall remain intact for effective core cooling.
b. The reactor shall not return to a high power level due to a return to criticality following reactor trip.
c. The reactor coolant system pressure shall not exceed code pressure limits.
d. The accident doses shall be within 10 CFR 50.67 and Regulatory Guide 1.183 limits.
e. The steam generator tubes shall not fail due to the loss of secondary side pressure and resultant temperature gradients.
f. The reactor building pressure during a steam line rupture inside containment shall not exceed the reactor building design limit.

The rupture of the steam line causes an increase in steam flow due to critical flow through the break. The loss of secondary coolant through the break causes a decrease in steam pressure, secondary saturation temperature, and an increase in flow across steam generator tubes. The increased flow across the steam generator tubes combined with the decline in secondary side fluid saturation temperature causes a cooldown and depressurization of the reactor coolant system. The cooldown of the reactor coolant system, in conjunction with a negative moderator reactivity coefficient causes positive reactivity addition to the core. The increase in core power coupled with reduced RCS pressure reduces the margin to DNB.

The Steam Line Break Accident was analyzed at 2568 MWt. This power level is consistent with the guidance given in BAW-10193-A (Reference 21). The radionuclide inventory dose analysis was based on 2619 MWt. Therefore, the steam line failure analyses support CR-3 operation at 2568 MWt.

4.11.12 Steam Generator Tube Rupture (FSAR Section 14.2.2.2)

A 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 SGTR event is classified as a limiting fault event.

A 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. The SGTR event is analyzed to determine the offsite and control room doses resulting from the release of contaminated primary coolant into the steam generator and to the atmosphere. The SGTR analysis was performed at 2568 MWt. The radionuclide inventory dose analysis was based on 2619 MWt. Therefore, the SGTR analysis supports CR-3 operation at 2568 MWt.

4.11.13 Fuel Handling Accident (FSAR Section 14.2.2.3)

A Fuel Handling Accident (FHA) involves the dropping of a fuel assembly and breaching of the fuel rod cladding. This event is classified as an infrequent event. There are numerous administrative controls and physical limitations that are imposed to prevent a FHA from occurring during refueling operations. Nevertheless, accident sequences with mechanical damage have been postulated with the objective of assessing the potential risk to the public health and safety. The acceptance criteria for the Fuel Handling Accident are based on the requirements of 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, for the calculated radiological consequences.

Two accident scenarios are considered: (1) refueling accident occurring inside the Reactor Building and (2) refueling accident occurring outside the Reactor Building.

For an FHA inside the reactor building, it is postulated that a fuel assembly is dropped during refueling resulting in breaching of the fuel rod cladding. As a result of the damage, the volatile fission gases contained in the fuel to pellet gap of all 208 fuel rods of one assembly are released to the water. Subsequently, a fraction of the iodine and all of the particulates are absorbed in the water. The escaped gases are assumed to be released to the environment.
The analysis of a Fuel Handling Accident in the Spent Fuel Pool is identical to the FHA inside the Reactor Building. The radioactive nuclide inventories are based on a plant power level of 2619 MWe. Therefore, the FHA analysis supports CR-3 operation at 2568 MWt.

4.11.14 Rod Ejection Accident (FSAR Section 14.2.2.4)

4.11.14.1 Hot Zero Power (HZP)

The Rod Ejection Accident from HZP 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 Accident from HZP event relate to peak RCS pressure and peak fuel enthalpy.

The ejection of a control rod with the reactor at zero power causes a rapid positive reactivity insertion. Core neutron 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 high flux and the reactor is returned to a zero power state by control rod insertion. The PSVs 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 will ensure that the peak fuel enthalpy will be within acceptable limits if the rod is ejected.

The Rod Ejection Accident from HZP is analyzed from zero power and is not directly affected by the initial core power level. The high flux setpoint is based on percent of rated power. Since the neutron power increases rapidly and reaches several orders of magnitude above the high flux reactor trip setpoint, the heatup is a function of the ejected rod worth, the rate it is ejected, and the Doppler reactivity coefficient. The core power increase does not change the ejected rod worth or the ejection rate. The Doppler coefficient is evaluated each cycle to ensure the analysis remains bounding. Therefore, the Rod Ejection Accident from HZP analysis supports CR-3 operation at 2568 MWt.

4.11.14.2 Full Power

The Rod Ejection Accident from Full Power (FP) 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 Accident from FP 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 high flux and the reactor is returned subcritical by control rod insertion. The PSVs 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 will be less than the 280 calorie/gram limit if the rod is ejected.

The Rod Ejection Accident from FP is analyzed at 2568 MWt. The radioactive nuclide inventories for dose consequences are based on a plant power level of 2619 MWe. Therefore, the analysis and dose consequences of the Rod Ejection Accident from FP supports CR-3 operation at 2568 MWt.

4.11.15 Loss-of-Coolant Accidents (LOCAs) (FSAR Section 14.2.2.5)

A Loss-of-Coolant Accident (LOCA) is defined as the postulated failure of the RCS pressure boundary that would allow the loss of reactor coolant into the reactor building (RB) at a rate in excess of the capability of the HPI/MU System. 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, reactor building pressure and temperature, and offsite and control room dose consequences. Reactor trip for all LOCAs is provided by the low RCS pressure trip. The variable low pressure and high reactor building (RB) pressure trips provide backup protection. The high RB pressure trip function is not credited in the analyses.

The LOCA results in a decrease in RCS inventory and RCS pressure. The loss of coolant inventory and depressurization causes a reactor trip on low RCS pressure. The timing of the low RCS pressure trip is a function of the break size, and to a lesser extent, the break location. The loss of reactor coolant results in an increase in RCS and fuel temperatures due to inadequate core cooling. The RCS pressure continues to decrease and engineered safeguards (ES) is initiated on low-low RCS pressure. Actuation of ES is required to establish sufficient RCS inventory to ensure adequate core cooling, to limit the fuel and cladding temperature increases, and to refill the RCS, thereby ensuring fuel cladding integrity.

The LOCA analyses are performed at a power level of 102 percent of 2568 MWt. In addition, the post-LOCA boron control management analyses performed for CR-3 were based on a core power level of 102 percent of 2568 MWt. The LOCA dose analyses are performed assuming a core power level of 2619 MWt and the resulting dose consequences meet the acceptance criteria of 10 CFR 50.67. Therefore, the analyses support CR-3 operation at 2568 MWt.

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

A rupture in the letdown line was evaluated to address the specific requirement for a letdown line rupture analysis in Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 3. A break in fluid-bearing lines that penetrate the reactor containment may result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS that penetrate the containment. However, there are other piping lines such as those associated with the Makeup and Purification (MU) System and the Decay Heat Removal (DH) System that penetrate the containment. For fluid penetrations in piping systems that do not serve to limit the consequences of accidents, leakage is minimized by a double-barrier design to ensure that no single credible failure or malfunction of an active component will result in either unacceptably high leakage or the loss of the capability to isolate a piping break. The installed double barriers consist of closed piping, both inside and outside the containment, and various types of isolation valves.

The most severe piping rupture for which radioactivity release is postulated during normal plant operation is in the letdown line of the MU System. However, as discussed in FSAR section 5.4.4.2, a break in the high energy portion of the letdown line outside containment is not considered a credible event. Nonetheless, the Makeup System Letdown Line Failure Accident is analyzed to demonstrate that the dose consequences from a postulated break in the letdown line outside containment remain below the 10 CFR 50.67 and Regulatory Guide 1.183 limits. Relative to dose consequences, the hypothetical break in the letdown line bounds other postulated breaks in lines connected to the RC System that carry reactor coolant outside containment.

The Letdown Line Rupture analysis was performed at 102 percent of 2568 MWt. Therefore, the event analysis supports CR-3 operation at 2568 MWt.

4.11.17 Maximum Hypothetical Accident (FSAR Sections 14.2.2.7)

A Maximum Hypothetical Accident (MHA) is equivalent to the Design Basis LOCA addressed in 4.11.15. The power level utilized in determining radioactive nuclide inventories was 102 percent of 2568 MWt. Therefore, the MHA analysis supports CR-3 operation at 2568 MWt.

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

The Waste Gas Decay Tank Rupture Accident (WGDTRA) postulates a gross failure, resulting in the release of the entire contents of all three waste gas decay tanks (WGDTs). Each tank is assumed to contain the maximum curie inventory allowed by the Offsite Dose Calculation Manual (ODCM). The WGDTRA would release more radioactivity to the atmosphere than any other credible radwaste system accident. The WGDTRA is that radiation doses are within the 10 CFR 50.67 limits. 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 2568 MWt.

4.11.19 Loss of Main Feedwater (FSAR Section 14.2.2.9)

A Loss of Main Feedwater accident is a complete loss of normal feedwater during power operation. A loss of main feedwater may result from abnormal closure of feedwater isolation valves, abnormal closure of feedwater flow control valves, or a main feedwater pump failure. This event is classified as a moderate frequency event. The acceptance criteria relate to peak RCS pressure and minimum DNBR. Additional criteria may also be imposed, i.e., peak pressurizer liquid level and average steam generator shell to average steam generator tube temperature difference.

A loss of main feedwater results in a reduction in secondary heat removal. Upon the reduction in secondary heat removal, the RCS temperature increases. The increase in RCS temperature causes coolant expansion and an increase in RCS pressure. Increasing RCS temperature and pressure could result in a water solid pressurizer, a failure of the RCS, or a fuel cladding failure. The increase in RCS pressure and temperature is limited by a reactor trip on high RCS pressure, steam relief through the PSVs, and subsequent initiation of emergency feedwater on low steam generator level. The emergency feedwater flow rate is sized to ensure that core decay heat is removed and a steam bubble is maintained in the pressurizer. The steam generator tube-to-shell temperature difference is calculated to ensure that tube stresses in excess of material limits are not caused by cold EF injection into the tube bundle. Excessive tube stresses could lead to failure and the introduction of a primary-to-secondary tube leak.

The Loss of Main Feedwater accident was analyzed at 102 percent of 2568 MWt. Therefore, the current analysis supports CR-3 operation at 2568 MWt.

4.11.20 Total Loss of Feedwater Accident (FSAR Section 14.2.2.9)

The Total Loss of Feedwater event is a complete loss of main feedwater and emergency feedwater. This event is considered beyond the original design basis of the plant. The event was analyzed following the TMI-2 accident to ensure that the core remained covered despite the total loss of feedwater. The acceptance criterion is that the core remains covered and that sufficient time was available for operators to restore feedwater flow or initiate feed and bleed cooling.

A total loss of feedwater results in a reduction in secondary heat removal. Upon the reduction in secondary heat removal, the RCS temperature increases. The increase in RCS temperature causes coolant expansion and an increase in RCS pressure. Increasing RCS temperature and pressure leads to a water solid condition and water discharge through the PSVs. The increase in RCS pressure and temperature is limited by a reactor trip on high RCS pressure and steam relief through the PSVs. Normally, initiation of emergency feedwater on low steam generator level would occur. However, in this event, the emergency feedwater system is assumed unavailable.

As a result of the unavailability of emergency feedwater, the steam generators boil dry and decay heat is removed by liquid/steam relief through the PSVs. The continued loss of primary coolant can lead to inadequate core cooling and core damage. Operator action is taken at 20 minutes to initiate safety injection to replenish lost RCS inventory and ensure long term core decay heat removal via feed and bleed cooling.

The Total Loss of Feedwater accident was analyzed at 102 percent of 2772 MWt, or 2827.4 MWt. Therefore, the current analysis supports CR-3 operation at 2568 MWt.

4.11.21 Feedwater Line Break Accident (FSAR Section 14.2.2.9)

The Feedwater Line Break event is a double-ended break between a steam generator (SG) and the first upstream feedwater check valve in the piping of the main feedwater system. The Feedwater Line Break event is classified as a limiting fault event. The acceptance criteria are related to peak RCS pressure, minimum DNBR, and offsite doses.

A double-ended main feedwater line break between an SG and the first upstream feedwater check valve is analyzed because there is no means of isolating a break in this region. A break in this region results in a blowdown of the affected SG until the steam lines are isolated on low steam line pressure. The loss of secondary coolant inventory continues through the break. In addition, it is postulated that the unaffected steam generator will also experience a loss of feedwater due to the decreased hydraulic line resistance to the affected steam generator diverting the flow from both main feedwater pumps to the break. The loss of secondary coolant results in a decrease in secondary heat removal and an increase in RCS pressure and temperature. The RCS pressure increases to the high RCS pressure trip setpoint and a reactor trip is initiated. The RCS pressure continues to rise and eventually the PSVs lift to limit the peak RCS pressure. Secondary inventory continues to decrease and EF is actuated on low-low SG level. When EF flow begins to the steam generator, the RCS temperature and pressure begin to decrease and decay heat is removed via steam release through the MSSVs on the unaffected steam generator.

The Feedwater Line Break event was reanalyzed for the power uprate at 102 percent of 2568 MWt. The reanalysis credits EFIC to initiate EF. All of the acceptance criteria were met. Therefore, this reanalysis supports CR-3 operation at 2568 MWt.

4.11.22 Anticipated Transient Without SCRAM (ATWS)

The analysis of ATWS events has been performed on a generic basis for the B&W plant design. Ten expected operational transients have been analyzed with a failure of the RPS to shut down the reactor via rod insertion. The transients that were analyzed are listed below.

- Rod Withdrawal
- Boron Dilution
- Loss of Primary Flow
- Inactive Primary Loop Startup
- Loss of Electrical Load
- Loss of Main Feedwater
- Loss of Offsite Power
- Load Increase
- Primary System Depressurization
- Excessive Cooldown

The results of the transient analyses indicated that the loss of main feedwater and loss of offsite power events produce the limiting response with respect to the ATWS event consequences considered. These transients were used to form the analytical basis for the design of the Diverse Scram System (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC). Based on further ATWS analyses, it was determined that the loss of main feedwater ATWS event was the limiting event with respect to peak RCS pressure.

4.11.22.1 Loss of Main Feedwater Anticipated Transient Without SCRAM

A Loss of Main Feedwater (LOFW) Anticipated Transient Without SCRAM is a postulated event initiated by a loss of main feedwater event with a subsequent failure of the RPS and the DSS to trip the reactor. Generic analysis performed to support the design of the DSS assumed a normal power level of 2772 MWt. The generic analysis is applicable to all B&W-designed 177-FA plants including CR-3. Therefore this analysis bounds operation at 2568 MWt.

4.11.22.2 Loss of Main Feedwater (LOFW) ATWS - AMSAC Enable Setpoint

The LOFW ATWS event was analyzed without credit for the DSS to determine an acceptable power level above which the AMSAC must be enabled. The AMSAC is designed to actuate EFW and trip the turbine on low SG liquid level.

The analysis of the LOFW ATWS that determined the power level at which AMSAC must be enabled was performed at a power level of 50 percent of 2544 MWt plus 16 MWt of reactor coolant pump heat. This corresponds to a power level of 49.53 percent of 2568 MWt. The results of the analysis indicate that the peak RCS pressure reached 3005 psia. The peak RCS pressure acceptance criterion adopted for the design of the AMSAC is 4000 psia. An increase in the initial power level to 50 percent of 2568 MWt plus 16 MWt of reactor coolant pump heat results in a subsequent increase in the peak RCS pressure. However, sufficient margin exists to the 4000 psia limit to accommodate the power increase without violating the acceptance criterion. Therefore, reanalysis of the LOFW ATWS event to determine the power level at which AMSAC must be enabled is not required. Based on the current LOFW ATWS analyses, CR-3 operation at a power level of 2568 MWt is supported.

4.12 Containment Mass And Energy Release

4.12.1 Loss of Coolant Accident Mass and Energy Release

A search for the original calculations that provided the mass and energy release data for the reactor building pressure analysis was performed, however, the original calculations were not located. These analyses were performed generically for all of the B&W plants during the initial startup of the Oconee plants in the late 1960's and early 1970's. Therefore a review of the Oconee FSAR was performed. The Oconee FSAR (1984 Update) states that the calculations for peak reactor building pressure were based on 102 percent power. The Oconee rated power level is 2568 MWt. A comparison of the mass and energy release rate data in the Oconee FSAR (1984 Update), Table 15.14-6, for the 70 ft^2 cold leg pump suction (split) break to Table 14-49 of the CR-3 FSAR for the same break confirmed that the same data were in both FSARs. Additionally, this same data is in the TMI-1 FSAR, Table 6.6-3, and the ANO-1 FSAR, Table 14-56. The data for the 14.1 ft^2 hot leg break was also confirmed to be identical between the Oconee, TMI-1 and CR-3 FSARs. The ANO-1 FSAR did not contain this data. Since the rated power levels of the Oconee plants, TMI-1 and ANO-1 are based on 2568 MWt, and the same data exists in the CR-3 FSAR, the mass and energy release rates support a power uprate to 2568 MWt.

4.12.2 Steam Line Break Mass & Energy Release

The Steam Line Break accident analysis for determining mass and energy releases to containment was performed at 102 percent of 2568 MWt. These mass and energy releases were used to calculate the containment pressure response. Therefore, the Steam Line Break analysis for the mass and energy releases for containment pressure and temperature response support CR-3 operation at 2568 MWt.

4.13 Radiological Consequences

4.13.1 Normal Operation Analyses

The proposed power uprate will not cause radiological exposure in excess of the dose criteria (for restricted and unrestricted access) provided in the current 10 CFR 20. From an operations perspective, radiation levels in most areas of the plant are expected to increase no more than the percentage increase in power level. Individual worker exposures will be maintained within acceptable limits by the site As-Low-As-Reasonably-Achievable (ALARA) Program, which controls access to radiation areas. Gaseous and liquid effluent releases are also expected to increase by no more than the percentage increase in power level. Offsite release concentrations and doses will be maintained within the limits of the current 10 CFR 20 and 10 CFR 50, Appendix I by the site radioactive effluent control program.

4.13.2 FSAR Chapter Radiological Accident Analyses

The FSAR Chapter 14 radiological accident analyses are discussed in section 4.11 of this Attachment. All analyses were performed assuming core power of 2619 MWt and support the power uprate. The calculated offsite dose results are unchanged from those approved in License Amendment Number 199, 3N0901-04, dated September 17, 2001.

4.13.3 Other Radiological Analysis

The Radionuclide inventory in the core is also used to calculate the post accident doses to electrical equipment as part of the Equipment Qualification program and for post-accident mission dose to operators per NUREG-0737, Section II.B.1. All EQ and mission dose analyses were performed based on a core inventory that assumed a power level of 2619 MWt. Therefore, these radiological analyses are not affected by the power uprate to 2568 MWt.

4.14 Nuclear Fuel

This section summarizes the evaluations performed (Reference 18) to determine the effect of the power uprate on the nuclear fuel performance. The core design and reload safety evaluations are performed for each specific fuel cycle and vary according to the needs and specifications for each cycle. The nuclear fuel review for the power uprate evaluated the fuel core design, core thermal-hydraulic design, and fuel rod mechanical performance.

4.14.1 Fuel and Core Design

The CR-3 cycle 13 fuel cycle, typical of current designs and fuel management, was modeled at the uprated power level to evaluate the effects of the power uprate conditions on the fuel and core design key parameters. The results were compared to the previous CR-3 cycle 13 design without the uprated power level. Since the power uprate is relatively small, the representative cycle is adequate to demonstrate the sensitivity of reload parameters to the power uprate conditions.

The methods and core models used in the uprate analyses are consistent with those presented in the CR-3 FSAR. No changes to the nuclear design philosophy, methods, or models are necessary due to the uprate. The core analyses for the uprate were performed primarily to determine if the values previously used for the key safety parameters remain applicable prior to the cycle-specific reload design.

The core analyses show that the implementation of the power uprate will continue to meet the current nuclear design basis documented in the FSAR. The impact of the uprate on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is expected to be either well within normal cycle-to-cycle variation of these values or controlled by the core design and will be addressed on a cycle-specific basis consistent with current approved reload methodology.

4.14.2 Core Thermal-Hydraulic Design

The core thermal-hydraulic analyses and evaluations were performed based on the CR-3 Cycle 13 design at an uprated core power level of 2568 MWt. The analyses assumed that the uprated core design will be composed of Mark-B10 fuel assemblies. All fuel assemblies in the Cycle 13 core have compatible thermal-hydraulic characteristics.

The thermal-hydraulic design methods and computer codes used for the 0.9 percent power uprate to meet the DNB design basis are consistent with those presented in CR-3 FSAR. The BWC DNB correlation for Mark-B fuel assemblies with Zircaloy-4 (Zr-4) grid spacers, BAW-10143P-A, is used for the power uprated thermal-hydraulic core protection evaluations. No changes to the thermal-hydraulic design philosophy, methods, or models are necessary due to the power uprate. The results show that the uprated core will meet all required thermal-hydraulic core protection requirements.

4.14.3 Fuel Rod Mechanical Performance

The fuel rod design analysis for fuel rod cladding corrosion was reviewed to assess the impact of the power uprate. Fuel rod cladding corrosion is adversely affected by increases in coolant temperature. An evaluation of the fuel rod cladding corrosion was performed for the uprated conditions. The results show that all the fuel in the core will continue to meet the acceptance criterion of less than or equal to 100 microns. The fuel rod cladding corrosion and other fuel rod design evaluations will continue to be performed on a cycle-by-cycle basis using either enveloping or actual fuel rod power histories. The results of these evaluations are expected to be within normal cycle-to-cycle variation, and will continue to be addressed on a cycle-specific basis consistent with current reload methodology.

4.15 Environmental Qualification (EQ)

The CR-3 Environmental Qualification of Electrical Equipment Important to Safety (10 CFR 50.49) Program was originally developed to FSAR Chapter 14 accident analysis with a conservative assumption of a maximum RTP of 2568 MWt. A re-evaluation of this accident analysis licensing basis for CR-3 demonstrated that all of the current CR-3 analysis continues to support EQ qualification levels up to a maximum error-adjusted power level of 2619 MWt (Reference 17) without program impact. Therefore, no changes to the EQ program are required due to this power uprate.

4.16 Safe Shutdown Fire Analysis

Calculation F97-0017, "Appendix R Cooldown," uses a power level of 2568 MWt, thus it bounds the current power uprate. No plant changes have been made that would affect fire loading, combustibles or fire initiation sources. Therefore the power uprate does not affect these analyses.

4.17 Flooding

The CR-3 flooding analysis is independent of power level. No plant changes are being made that impact flood initiating events or mitigating actions. Therefore the CR-3 flooding analysis is not affected by the power uprate.

4.18 Erosion/Corrosion Program

The increase of 24 MWt will involve a small increase in flow in secondary plant components, approximately one percent. This increase in flow will slightly increase the erosion/corrosion rates of some secondary equipment. The resulting increase in erosion/corrosion rates is small and does not create a significant problem. The increased erosion/corrosion rates will be factored into the CR-3 program and will be addressed by existing trending, tracking and maintenance programs. Therefore, the erosion/corrosion program is not significantly impacted by the power uprate.

4.19 High Energy Line Break (HELB)

The changes to the secondary plant due to increase feedwater and steam flow are very minor. No new piping will be classified as high energy and no areas will change their energy classification. Changes in flow will not significantly increase jet impingement and thrust forces or affect piping stresses that could affect break location designations. Therefore, the power uprate does not impact the HELB analysis.

4.20 Safety-Related Valves

The operation of safety-related valves was reviewed for the proposed power uprate. The primary parameter affecting valve operation is differential pressure across the valve. Changes in secondary system flow may slightly change the differential pressures for some safety-related valves. Some valves may need minor adjustments prior to implementation of the uprated power level. These adjustments can be made with the unit at power. Therefore, the power uprate will not have a significant impact on safety-related valves.

4.21 Plant Operations and Procedures

The proposed power uprate is small, 24 MWt or about 8 MWe. The power range nuclear instrumentation displays reactor power is given as a percentage of RTP. Therefore, these indications will appear the same at 2568 MWt as they did previously at 2544 MWt. The operators will be able to see an increase of approximately 8 MWe on the generator output instrumentation and slight changes on instruments for feedwater flow, steam flow, T_{hot}, T_{cold} and Loop ΔT. The Automated Unit Load Demand (AULD) will also show the increased power level because it displays a digital readout of core thermal power. Operators will be made aware of the uprate and its effects in their normal requalification training and by a required reading (Operations Study Book) at the time of implementation. In addition, numerous normal operating and surveillance procedures will have to be revised to change the value of RTP from 2544 MWt to 2568 MWt. Many procedures reference percentages of RTP. The change to RTP is very small, and therefore, these procedural references will not change except where conversions to MWt or MWe are listed. No changes are required for emergency operating procedures or operator action times.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration Determination

Crystal River Unit 3 (CR-3) proposes to increase the maximum reactor core power level from 2544 Megawatts Thermal (MWt) to 2568 MWt. Operating License Condition 2.C.(1), "Maximum Power Level," Improved Technical Specifications (ITS) Definitions "Effective Full Power Day (EFPD)" and "Rated Thermal Power (RTP)," will be revised to reflect the proposed increase to 2568 MWt.

Florida Power Corporation (FPC) has reviewed the proposed change and the associated revisions to the Operating License and ITS against the requirements of 10 CFR 50.92(c). The proposed changes do not involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

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

The proposed change will increase the maximum core power level from 2544 MWt to 2568 MWt. This increase will only require adjustments and calibrations of existing plant instrumentation and control systems. No hardware upgrades or equipment replacements are needed to implement the proposed change.

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 2568 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 2568 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).

Based on the above, the change will not increase the probability or consequences of an accident previously evaluated.

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

As discussed above, no hardware upgrades or equipment replacements are required to implement the proposed change. 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. The design, physical configuration and operation of the plant will not be changed; as a result, no new equipment failure modes will be introduced. 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 the margin of safety.

Challenges to the fuel, reactor coolant system (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. Mass and energy release to the containment from a loss-of-coolant accident (LOCA) or main steam line break are also bounded by current analyses, which assume an initial power level of 2619 MWt.

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.

5.2 Regulatory Safety Analysis

The proposed power uprate does not adversely affect compliance with any regulatory requirements. No new exemption from any regulation is required. The bases for previous regulatory exemptions remain valid. All the principal architectural and design criteria set out in FSAR Section 1.4 will continue to be met. As discussed in detail in section 4.11 of this Attachment, all Chapter 14 accident analyses remain bounding. Only minor changes to the FSAR will be required following issuance of this amendment (no analyses, results or conclusions will change).

6.0 ENVIRONMENTAL 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 operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this license amendment request 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(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 for this License Amendment Request (LAR).

2. The proposed changes will allow Crystal River Unit 3 (CR-3) to operate at an uprated power level of 2568 Megawatts Thermal (MWt). This represents an increase of approximately 0.9 percent over the current 100 percent power level of 2544 MWt.

The proposed changes do not significantly impact installed equipment performance or require significant changes in system operation. Maintenance and operational practices will not change as a result of the uprate. The specific activity of the primary and secondary coolant are 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.

3. 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.

Non-Radiological Evaluation

With regard to non-radiological impacts, the proposed license amendment involves no significant increase in the amounts or changes in the types of any non-radiological effluents that may be released offsite. The only expected non-radiological impact resulting from the uprate is a slight increase in the discharge temperature of the CR-3 circulating water (CW) system, which will in turn result in a small change in the combined site point of discharge temperature. This temperature is limited to a maximum consecutive three-hour average of 96.5°F. by the National Pollutant Discharge Elimination System (NPDES) permit for the Crystal River Energy Complex. In the event that the point of discharge temperature approaches this limit, a portion of the site discharge water is diverted through mechanical draft "helper" cooling towers and returned to the discharge canal where it mixes with the warmer water to reduce the point of discharge temperature. The heat removal capability of the "helper" cooling towers is more than sufficient to accommodate the small CW temperature increase discussed above and maintain the point of discharge temperature within limits. No changes to the current NPDES permit will be required as a result of the proposed change.

7.0 REFERENCES

1. FRA-ANP 32-5012972-00, "CR-3 Power Uprate Operating Conditions"

2. BAW-10051, Rev. 1, "Design of Reactor Internals and Incore Nozzles for Flow-Induced Vibrations"

3. FRA-ANP 51-5013615-00, "CR-3 Power Uprate (2619 MWt) Systems Review"

4. FRA-ANP 51-5013664-00, "CR-3 Power Uprate RCS Structural Integrity"

5. FRA-ANP 86-1266133-01, "CR-3 PT Fluence Analysis Report - Cycles 7-10"

6. FRA-ANP 32-5013936-00, "Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate"

7. BAW-2325, Revision 1, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," January 1999

8. FRA-ANP 32-5013892-00, "CR-3 Power Uprate PTS Evaluation"

9. FRA-ANP, 51-5013910-00, "Alloy 600 PWSCC Evaluation for CR-3 Power Uprate"

10. FRA-ANP, 32-5013422-00, "CR-3 Power Uprate RV Internals FIV"

11. FRA-ANP, 51-5013985, "Effect of Power Uprate on Current Crystal River unit 3 P-T Curves"

12. FRA-ANP 51-5013980-00, "CR-3 Power Uprate OTSG Tube Integrity"

13. FRA-ANP 51-5000475-01, "CR-3 OTSG FIV Margins"

14. FRA-ANP 32-5013766-00, "CR-3 Power Uprate MSS & FW Evaluation"

15. FPC Document EEM-01-013, Revision 1, "PEPSE Runs for Power Level Uprate"

16. Florida Power Corporation, Crystal River Unit 3, Final Safety Analysis Report, Docket No. 50-302

17. FRA-ANP 51-5012807-01, "CR-3 Safety Analysis Evaluation for Core Power Uprate"

18. BAW-2391, "Crystal River Unit 3 Cycle 13 Reload Report," August 2001, Doc. I.D. 103-2391-00

19. FPC Letter, 3F0199-02, dated January 7, 1999, "Change in Analysis of Record for Small Break Loss of Coolant Accident and 10 CFR 50.46 Notification"

20. FPC Letter 3F1199-01, dated November 10, 1999, "Notification of Change in Peak Clad Temperature for Small Break Loss of Coolant Accident in Accordance with 10 CFR 50.46(a)(3) and Change in the Analysis of Record for Large Break Loss of Coolant Accident"

21. BAW-10193-A, "RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors," January 2000

22. FPC Calculation M-94-0043, Revision 4, "RCS Functional Specification for CR-3"

23. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988

8.0 PRECEDENTS

The NRC has approved over 50 stretch power uprates, many of these were for greater than 5 percent RTP. The proposed CR-3 uprate is less than 1 percent RTP. All B&W 177 FA plants (Davis Besse, Arkansas Nuclear One Unit 1, Three Mile Island Unit 1 and Oconee Units 1, 2 and 3), except CR-3, are currently approved for operation at 2568 MWt or higher. The most similar of these precedents is Three Mile Island (TMI) Unit 1 which was granted a 1.3 percent power uprate to 2568 MWt in 1988. The applicable references are listed below:

Letter from GPU Nuclear Corporation to NRC, Three Mile Island Nuclear Generating Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Technical Specification Change Request No. 184 – Rated Power Upgrade, dated April 18, 1988, C311-88-2036.

Letter from NRC to GPU Nuclear Corporation, Issuance of Amendment (TAC NO. 67903), License Amendment No. 143, dated July 26, 1988.

In the referenced Amendment, the NRC approved a slightly greater power uprate for a very similar plant to the same maximum power level as that requested by CR-3.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #270, REVISION 0
Power Uprate to 2568 MWt

Proposed Revised Improved Technical Specification Pages
and License Condition – Strikeout Version

Crystal River Nuclear Generating Plant
Docket No. 50-302 Facility Operating License

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 ~~2544~~2568 Megawatts (100 percent of rated core power level).

1.1 Definitions

EFFECTIVE FULL POWER DAY (EFPD) (continued)	reactor core at RTP for one full day. (One EFPD is ~~2544~~**2568** MWt times 24 hours or **61,632**~~61,056~~ MWhr.)
EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME	The EFIC RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its EFIC actuation setpoint at the channel sensor until the emergency feedwater equipment is capable of performing its safety function (i.e., valves travel to their required positions, pump discharge pressures reach their required values, etc.) Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

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

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

(continued)

1.1 Definitions

PHYSICS TESTS (continued)

These tests are:

a. Described in Chapter 13, "Initial Tests and Operation" of the FSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.2.19. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature Limits."

QUADRANT POWER TILT (QPT)

QPT shall be defined by the following equation and is expressed as a percentage.

$$\text{QPT} = 100\left(\frac{\text{Power In Any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$$

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2544~~**2568** 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)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT C

LICENSE AMENDMENT REQUEST #270, REVISION 0
Power Uprate to 2568 MWt

Proposed Revised Improved Technical Specification Pages
– Revision Line Version

1.1 Definitions

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