December 4, 2002

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING POWER UPRATE TO 2568 MWt (TAC NO. MB5289)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 205 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). This amendment consists of changes to the Facility Operating License and Technical Specifications (TS) in response to your application dated June 5, 2002, as supplemented by letters dated August 13, September 30, October 31, November 13, and November 25, 2002.

This amendment approves a revision to the Facility Operating License and TS to reflect an increase in the CR-3 maximum steady-state core power level from 2544 megawatts thermal (MWt) to 2568 MWt, an increase of approximately 0.9 percent.

A copy of the Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/RA by M.McConnell Acting for/

Brenda L. Mozafari, Sr. Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 205 to DPR-72
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page
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OFFICE	PM:PDII-S2	Intern:PDII/S2	LA:PDII-S2	DSSA:SPSB	DIPM	DE:EEIB*	DE:EEIB (I&C)*	DSSA:SRXB*	DSSA:SRXB*
NAME	MmcConnell for BMozafari	MMcConnell	EDunnington	MReinhart	DTrimble	CHolden	EMarinos	FAkstulewicz	FAkstulewicz
DATE	12/04/2002	12/04/2002	12/04/2002	11/25/2002	11/26/2002	09/ 25/2002	11/06/2002	10/31/2002	09/04/2002

OFFICE	DSSA:SRXB*	DE:EMEB*	DSSA:SPLB	DE:EMCB*	DE:EMCB*	OGC	SC:PDII-2	D:PDII-S2	DD/D:DLPM
NAME	RCaruso	KManoly	SWeerakkody	LLund	SCoffin	Weisman	AHowe	HBerkow	TMarsh/JZwolinski
DATE	11/07/2002	11/07/2002	10/18/2002	11/07/2002	11/ 01/2002	12/03 /2002	12/04 /2002	12/04 /2002	12/04 /2002

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AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. DPR-72 - Crystal River Unit 3

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CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION,
CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.205
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated June 5, 2002, as supplemented by letters dated August 13, September 30, October 31, November 13, and November 25, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by replacing paragraph 2.C.(1) on page 4 of Facility Operating License No. DPR-72 with the following:

   (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).

3. The license is also amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 205, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by T.Marsh Acting for/

John A. Zwolinski, Director
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachments:
1. Page 4 of License DPR-72
2. Changes to the Technical Specifications

Date of Issuance: December 4, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 205

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of the Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
Page 4	Page 4

Replace the following pages of the Appendix "A" Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
1.1-4	1.1-4
1.1-6	1.1-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT TO INCREASE THE

AUTHORIZED POWER LEVEL OF

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.90, Florida Power Corporation (FPC), by letter dated June 5, 2002, as supplemented by letters dated August 13, September 30, October 31, November 13, and November 25, 2002, submitted a request for an amendment to the Facility Operating License (FOL), including the Appendix A Technical Specifications (TS) for Crystal River Unit 3 (CR-3). The proposed amendment would increase the authorized reactor core power level from 2544 megawatts thermal (MWt) to 2568 MWt (approximately 0.9 percent increase).

The August 13, September 30, October 31, November 13, and November 25, 2002, supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. 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, CR-3 requested an increase in maximum rated thermal power (RTP) to 2568 MWt. At that time, several transient and accident analyses (moderator dilution accident, letdown line failure, loss of feedwater (FW) event, and small break loss-of-coolant accident (LOCA)) were reevaluated at 2568 MWt. All other analyses had already been performed at 2568 MWt or higher. The request was withdrawn in a letter dated May 1, 1996, stating that 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 FW 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 (approximately 0.9 percent increase).

3.0 EVALUATION

3.1 Reactor Systems

3.1.1 Reactor Systems Evaluation

The uncertainty of the calculated values of reactor core thermal power determines the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. As relevant here, Appendix K of 10 CFR Part 50 requires the licensees to base their LOCA analysis on an assumed power level of at least 102 percent of the licensed thermal power. This required power ratio is to allow for uncertainties in determining thermal power.

In support of the power uprate application, the licensee submitted safety analyses including transient and LOCA analyses, and proposed TS changes for NRC staff review and approval. The licensee evaluated the following: NSSS performance parameters, design transients, systems, components, accidents, and fuel, as well as interfaces between the NSSS and balance-of-plant (BOP) systems. The methodology includes the use of well-defined analysis input assumptions and parameter values, and currently approved analytical techniques, and takes into consideration applicable licensing criteria and standards. The licensee 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 evaluations. The licensee performed a detailed assessment of the LOCA, non-LOCA, steam generator tube rupture (SGTR), and containment accident analyses and fuel performance to determine the ability to perform at the uprated condition. The results show that there are no changes to design-basis or transient analyses required to accommodate the revised NSSS design conditions. Each of the systems and components was evaluated for the uprated conditions. As set forth below, the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met and that CR-3 requires no design changes except setpoint and calibration adjustments to safely operate at the uprated conditions.

The NRC staff review is to verify that the licensee’s analytical results meet the required acceptance criteria, and to ensure that the proposed TS appropriately reflect the results of acceptable safety analyses. The following evaluation is based on the NRC staff review of the licensee’s safety analyses, proposed TS changes, and the responses to the staff’s Requests for Additional Information (RAIs). This review includes non-LOCA and LOCA accident analyses and other events.

3.1.2 Non-LOCA and LOCA Transients Analyses

The licensee discussed the Final Safety Analysis Report (FSAR) Chapter 14 transients and LOCA analyses in Reference 1 for the power uprate conditions. The licensee identified the limiting cases for each event category discussed in FSAR Chapter 14 and evaluated the effects of the proposed power uprate on plant transients and accidents. For those cases that were bounded by the corresponding cases in FSAR Chapter 14, the licensee provided supporting rationales. The majority of them were analyzed at the thermal power level of 2619 MWt, which bounds the current request at 2568 MWt. For those cases with values of plant parameters outside the applicable range of the corresponding FSAR cases, the licensee provided results of reanalyses to show compliance with applicable acceptance criteria used in the corresponding FSAR analysis. The licensee considered a maximum core power of 2568 MWt (increased from the current core power of 2544 MWt; Reference 1).

The NRC staff reviewed the licensee's safety assessment (Reference 1) and the RAI responses and found that the current FSAR analyses for the design-basis transients and accidents remain conservative and, therefore, acceptable for the uprated power. A summary of the NRC staff's conclusions is shown in Table 1. The table includes a description of the events analyzed, the impacted FSAR sections, and a disposition for each of those events. Where the table indicates that the existing analyses "remain valid," the current analyses were performed at 2568 MWt or higher, and bound the uprated power conditions. For the reasons summarized in Table 1, the NRC staff finds all of these transients acceptable for the uprated conditions.

3.1.3 Anticipated Transient Without Scram (ATWS)

The analysis of ATWS events was performed on a generic basis for the B&W plant design. Ten expected operational transients have been analyzed with a failure of the reactor protection system (RPS) to shut down the reactor via rod insertion. The results of the analyses indicated that the loss-of-main-feedwater (MFW) and loss-of-offsite-power (LOOP) events produce limiting responses with respect to the ATWS 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). A generic analysis performed to support the design of the DSS assumed a normal power level of 2772 MWt. It is applicable to all B&W-designed 177-FA plants, including CR-3. Accordingly, the NRC staff concludes that this analysis supports the uprated condition.

3.1.4 Core Thermal-Hydraulic Design

The NRC staff reviewed the licensee's response to its RAI dated September 30, 2002, Item 2.b, with respect to the core thermal-hydraulic design for the CR-3 Cycle 13 operation.

The Cycle 13 core consists entirely of the Mark-B10 fuel assembly design; thus there is no mixed core condition. Therefore, a mixed core penalty was not applied to the departure from nucleate boiling (DNBR) evaluation for CR-3 Cycle 13 operation. In a November 7, 2002, conference call and letter dated November 17, 2002, the licensee confirmed that (1) the applicable test data available for critical heat flux (CHF) calculated with the BWC correlation were obtained in the CHF tests performed at the Alliance Research Center in 1985, and (2) no significant design changes in the grid have occurred since the CHF tests. A minor modification to the outside corner of the spacer grid for the Mark-B10 fuel assembly was insignificant. The change was made to reduce the possibility of interferences during fuel movement. With respect to the fuel assembly design used for testing at the Alliance Research Center, this modification was found to have no impact on the thermal-hydraulic design. Therefore, the NRC staff concludes that the justification for continued use of the previously approved BWC CHF correlation for the CR-3 power uprate is acceptable.

3.1.5 Station Blackout (SBO) (FSAR Section 14.1.2.9)

10 CFR 50.63, "Station Blackout," requires that all light-water-cooled nuclear power plants must have the capability to withstand a loss of all ac power (except from station batteries or alternate ac sources) for an established period of time (4 hours for CR-3) and to recover therefrom.

The SBO event is a complete loss of all unit ac power, except ac power available through inverters from station batteries or alternative ac sources, as defined in 10 CFR 50.2. The loss of such power results in a loss of reactor coolant pump (RCP) seal injection flow. Primary coolant is lost at a specified rate through the RCP seals and reactor coolant system (RCS) inventory decreases throughout the event. The original analysis shows that sufficient RCS inventory is maintained to keep the core covered and maintain adequate core cooling throughout the 4-hour coping period. This event analysis was performed at a power level of 2772 MWt. The NRC staff reviewed the licensee's submittal and determined that the power uprate to 2568 MWt is bounded by the original SBO analyses, the plant continues to meet the requirements of 10 CFR 50.63, and the power uprate is, therefore, acceptable.

3.1.6 Loss of AC Power

Following a loss of ac power, the turbine stop valves close, resulting in an increase in secondary pressure, a decrease in secondary heat removal capacity, reactor coolant heat-up, and primary pressure increase. The RCS pressures and temperatures are controlled by heat removal via steam relief through the main steam safety and atmospheric dump valves. Decay heat removal capability will be maintained via natural circulation. The acceptance criteria, which were established for RCS peak pressure, minimum DNBR, and decay heat removal capability, are more limiting for the loss-of-MFW event, which is analyzed at 102 percent of 2568 MWt. This event was analyzed at 2568 MWt, which is the uprated power, and will support operation at the uprated power condition; therefore, the design is acceptable.

3.1.7 Reactor Systems Summary

The NRC staff has reviewed the licensee's safety analyses in support of operation of CR-3 at the maximum core power level of 2568 MWt. The NRC staff finds that the supporting safety analyses show that the uprated power conditions are bounded by the current FSAR analyses. Therefore, the NRC staff concludes that the current FSAR analysis for the design-basis transients and accidents remains valid and acceptable for the uprated power.

3.2 Electrical Systems

3.2.1 Electrical Systems Evaluation

General Design Criterion (GDC) 17, "Electric Power Systems," of Appendix A to 10 CFR Part 50 requires that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure, among other things, that containment integrity and other vital functions are maintained in the event of postulated accidents.

Section 50.49 of 10 CFR Part 50, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety. Under the rule, each licensee must prepare and maintain a record of qualification to document that each item of equipment subject to the rule (1) is qualified for its application, and (2) meets its specified performance requirements when subjected to the environmental conditions predicted to be present when it must perform its safety function up to the end of qualified life.

The main generator is rated at 989.4 MVA (based on 60 psig hydrogen pressure) at a 0.9 power factor (pf). The station output generated at 22 kV is fed through an isolated phase bus to the primary windings of three single-phase 316.667 MVA units (forced oil and air-cooled) to form a three-phase bank with a nominal rating of 950 MVA at 65^0 C temperature rise. The unit auxiliary transformer (UAT) is sized to carry unit non-safeguard full load auxiliaries. The station distribution system consists of various auxiliary electrical systems to provide electrical power during all modes of operation and shutdown conditions. The electrical distribution system has been previously evaluated to conform to GDC 17. The plant has also been previously evaluated for environmental qualification for electrical equipment in accordance with 10 CFR 50.49.

The following documents the NRC staff's evaluation of grid stability, the main generator, the transformers, the emergency diesel generators, environmental qualification, the motor feeders, and the dc electrical system.

3.2.2 Grid Stability

A review of the generator reactive capability curve confirms that the main generator is capable of operating at a maximum real power output of 989.4 Megawatts electric (MWe) at a 1.0 pf (zero MVAR output). Heat balance studies completed for the uprate identify a gross generator output of 903 MWe at 0.91 pf. The licensee evaluated the impact of the power uprate on the grid stability and determined that it will have no impact. All simulations exhibited damped oscillations with no unusual deviation in voltage or frequency.

The NRC staff reviewed the licensee's submittal and concluded that the power uprate, because it is sufficiently small (less than 1 percent), will have no impact on grid stability. Therefore, the NRC staff concludes that the plant continues to have acceptable grid stability for this power uprate.

3.2.3 Main Generator

The main generator is rated at 989.4 MVA (based on 60 psig hydrogen pressure) at a 0.9 pf. At the current thermal power rating of 2544 MWt, the main generator electrical output is typically 895 MWe. With the power uprate, the gross generator output would be 903 MWe at 0.91 pf. Generator operation at a lower output at a unity pf or less is permissible provided unit operation remains within the real and reactive power limits defined by the reactive capability curve. The main generator and associated cooling equipment are designed to accept the maximum generator output at the uprated condition and no modifications to the main generator will be required for the power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 903 MWe at 0.91 pf (i.e., 992.4 MVA) is approximately the same as the maximum main generator real output of 895 MWe at 0.9 pf (i.e., 994 MVA). Since the proposed increase falls within the real and reactive power limits defined by the capability curve, the NRC staff concluded that operating the main generator at the uprated power condition is acceptable.

3.2.4 Main Transformer

The station output, which is generated at 22 kV, is fed through an isolated phase bus to the primary windings of three single-phase 316.667 MVA units (forced oil and air-cooled) to form a three-phase bank with a nominal rating of 950 MVA at 65^0 C temperature rise.

The NRC staff reviewed the licensee's submittal and concluded that the power uprate of 903.4 MWe at 0.91 pf (992 MVA) minus the plant auxiliary loads that are powered through the UAT is below the maximum main transformer rating of 950 MVA and, therefore, the NRC staff concludes that operating the main transformer at the uprated power condition is acceptable.

3.2.5 Isophase Bus

The isophase bus duct connects the main generator to the primary windings of the main power transformer and the UAT. The power uprate will increase power from 895 MWe at 0.9 pf to 903 MWe at 0.91 pf. The isophase bus duct and associated cooling equipment are designed to accept the maximum generator output for the uprated condition.

The NRC staff reviewed the licensee's submittal and concluded that the rating of the isophase bus bounds the maximum generator output and, therefore, operating the isophase bus at the uprated power condition is acceptable.

3.2.6 UAT

The UAT nameplate rating is 22 kV, 55 MVA (forced oil and air-cooled) at 55°C, three-phase, 60 Hz. The UAT is sized to carry unit non-safeguard full-load auxiliaries.

The NRC staff reviewed the licensee's submittal and concluded that the increase in house loads resulting from power uprate is below the maximum UAT design rating and, therefore, operating the UAT at the uprated power condition is acceptable.

3.2.7 Emergency Diesel Generators (EDGs)

There is no change to the safety-related loads at uprate conditions and, therefore, the EDGs will not be affected by the power uprate and can perform their safety-related functions.

The NRC staff's review determined that the power uprate does not affect the loading on the EDG. Therefore, the NRC staff concludes that the licensee will continue to meet GDC 17 with respect to the EDGs for the power uprate.

3.2.8 Environmental Qualification (EQ) of Electrical Equipment

The licensee analyzed the impact of the power uprate on the environmental qualification of electrical equipment and determined that all the current CR-3 analyses continue to support EQ qualification levels up to a maximum error-adjusted power level of 2619 MWt.

The NRC staff reviewed the licensee's submittal and determined that qualification data for a power level of 2619 MWt are bounding for the proposed uprate to 2568 MWt; therefore, no changes to the EQ program are required for this power uprate, and the plant continues to meet the requirements of 10 CFR 50.49.

3.2.9 Motor Feeders

The licensee reviewed the motor feeder system to determine the impact of the power uprate on the system. The licensee states that all components remain within their design limits and no changes in design are required. For this reason, the NRC staff concludes that the system supports the power uprate.

3.2.10 DC Electrical System

The licensee reviewed the dc electrical systems to identify the major items that may be affected by the uprated conditions and to evaluate the potential impact of an uprate on that equipment. The licensee stated that its reviews confirmed that no dc-powered loads were affected by unit operation at uprated conditions. Furthermore, the reviews confirmed that the dc control power for ac loads remained unchanged. The NRC staff reviewed the licensee's submittal and determined that the dc electrical system is unaffected by the power uprate. Accordingly, the NRC staff concludes that the uprate is acceptable with respect to the dc electrical system.

3.2.11 Electrical Systems Summary

The NRC staff has evaluated the effects of the power uprate on the safety-related electrical systems and environmental qualification of electrical components. Results of these evaluations show that the increase in core thermal power would have a negligible impact on grid stability or the functioning or environmental qualification of electrical components. This is consistent with GDC 17 and 10 CFR 50.49, and the proposed change is, therefore, acceptable.

3.3 <u>Instrumentation and Controls (I&C)</u>

3.3.1 I&C Evaluation

The NRC staff's evaluations of the setpoint changes for the identified instrumentation for the new power level are based on the licensee's use of approved design codes to calculate analytical limits, as indicated.

The reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown system, and control systems for the power uprate condition should be designed such that they continue to meet their safety functions or their failures do not affect the safety functions. The acceptance criteria for this review are based on 10 CFR 50.55a(a)(1) and 10 CFR 50.55a(h) as related to quality and design of protection and control systems. Specific review criteria are contained in the Standard Review Plan (SRP) Chapter 7.

3.3.2 Suitability of Existing Instruments

In its submittal, the licensee stated that for the proposed power uprate, each existing instrument of the affected NSSS and BOP systems was evaluated to determine its suitability for the revised operating range of the affected process parameters. Where operation at the power-uprated condition impacted safety analysis limits, the evaluation verified that an acceptable safety margin continued to exist under all conditions of the power uprate and where necessary, setpoint and uncertainty calculations for the affected instruments were revised. Apart from the few devices that needed change, the licensee's evaluation found most of the existing instrumentation acceptable for the proposed power uprate operation. The evaluation resulted in changes in:

- Software for the automatic unit load demand programmable logic controller and minor adjustments to several integrated control system (ICS) modules required to reflect the increased power output including the increase in FW flow.

- Calibration of various instruments to reflect the increased power output.

These changes will be performed to accommodate the revised process parameters. These changes are established to limit accident initial conditions and preserve accident analysis assumptions and have been reviewed by the NRC staff and documented in other sections of this Safety Evaluation (SE). Therefore, the NRC staff agrees with the licensee's conclusion that when the above-noted modifications and changes are implemented, the CR-3 I&C systems will accommodate the proposed power uprate without compromising safety.

3.3.3 RPS/ESFAS Instrumentation Trip Setpoint and Allowable Values

In its submittal dated June 5, 2002, and in its letter dated October 31, 2002, in response to the NRC staff's RAI, the licensee confirmed that there are no setpoint changes needed to the RPS and ESFAS instrumentation. The licensee did not perform any setpoint calculation for this license amendment because the existing accident analysis was based on the new RTP. Therefore, the licensee did not address instrument setpoint methodology in their submittal of June 5, 2002. Because the existing accident analysis is based on an RTP of 2568 MWt or higher, the NRC staff finds that the existing setpoints will maintain margins between operating conditions and the reactor trip setpoints and do not significantly increase the likelihood of a false trip or failure to trip upon demand. Based on this information, the NRC staff has determined that the proposed power uprate will not result in any significant reduction of margin and the existing licensing basis is not affected by the power uprate.

3.3.4 TS Changes Related to the Power Uprate

As discussed in the previous section, there are no setpoint changes to the safety-related instrumentation and, therefore, no TS changes are required.

3.3.5 I&C Summary

Based on the above review and justification, the NRC staff concludes that the effect of the power uprate on the instrumentation is minimal, and the changes for the power uprate are consistent with CR-3's licensing basis; therefore, CR-3's I&C systems continue to meet 10 CFR 50.55a(a)(1) and 10 CFR 50.55a(h). Thus, the NRC staff finds the licensee's proposed power uprate acceptable with respect to I&C systems.

3.4 RCS

3.4.1 RCS Evaluation

3.4.1.1 Reactor Pressure Vessel (RPV) Integrity Requirements

RPV integrity requirements include provisions for pressure-temperature (P-T) limit curves, upper shelf energy (USE) analyses, RPV material surveillance programs, and evaluation of the susceptibility of the CR-3 RPV to failure during a pressurized thermal shock (PTS) event.

The NRC's regulatory requirements related to the establishment of RPV P-T limit curves for any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, are given in Appendix G to 10 CFR Part 50, which also references, as incorporated in 10 CFR 50.55a, the requirements given in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Appendix G. Additional guidance for the NRC staff's review of RPV P-T limit curves and USE analyses is provided in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," SRP Section 5.3.2, "Pressure-Temperature Limits," and Branch Technical Position (BTP) MTEB 5-2, "Fracture Toughness Requirements." Appendix K to Section XI of the ASME Code and RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-lb" may also be used as guidance when USE equivalent margins analyses are required.

The NRC's regulatory requirements related to the establishment of a facility's RPV surveillance capsule program and withdrawal schedule are given in Appendix H to 10 CFR Part 50, which also references the guidance in American Society for Testing and Materials (ASTM) Standard Practice E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." SRP Section 5.3.1, "Reactor Vessel Materials," provides guidance on this matter.

With regard to the information on protection against PTS, the NRC staff reviewed the requested power uprate against the regulatory requirements established in 10 CFR 50.61. As with P-T limits, BTP MTEB 5-2 (Ref. 11) may also apply with respect to the determination of initial, unirradiated properties of RPV materials.

3.4.1.2 Reactor Vessel Internal Integrity Requirements

Maintenance of the structural integrity of the reactor vessel internals is required in order to demonstrate that the functional requirements of the RPV internals can be met. These functional requirements include core support and emergency core cooling system (ECCS) performance aspects. As such, the structural integrity of the RPV internals is linked to regulatory requirements in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," regarding ECCS performance and maintaining a coolable core geometry. Additional guidance regarding the evaluation of the structural integrity of RPV internals may be found in SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and in ASME Code Sections III and XI, or other standards that were used in the NRC's review of the original licensing basis of a particular facility.

3.4.1.3 Alloy 600 Integrity Issues

In 2001 and 2002, the NRC issued three bulletins concerning Alloy 600 primary water stress corrosion cracking (PWSCC) of the RPV head and vessel head penetrations (VHPs). The first bulletin, Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," was issued in response to the identification of circumferential cracking in control rod drive mechanism (CRDM) nozzles at Oconee and other pressurized-water reactors (PWRs). The purpose of Bulletin 2001-01 was to ascertain the extent of CRDM nozzle cracking in PWRs. The second bulletin, Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," was issued in response to the discovery of significant material degradation of an RPV head at a PWR (Davis-Besse). One of the purposes of Bulletin 2002-01 was to ascertain the extent of material wastage in other PWRs similar to that which occurred at Davis-Besse. During the development of the bulletins and review of the responses to the first two bulletins, the NRC staff determined that it may be necessary for inspection programs for RPV heads and VHP nozzles that rely on visual examinations to be supplemented with additional measures (e.g., volumetric and surface examinations) to demonstrate compliance with applicable requirements. Accordingly, the NRC issued a third bulletin, Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle

Inspection Programs." The purpose of Bulletin 2002-02 was to learn what, if any, changes PWR licensees have made to their RPV head and VHP nozzle inspection programs, and their justification for reliance on visual examinations if that is their primary method to detect degradation.

3.4.1.4 RPV Integrity

In Section 4.7.2 of Attachment A to its June 5, 2002 letter, the licensee indicated that the reactor vessel neutron fluence will increase with the power uprate. To bound the increase in neutron fluence, the current 32 effective full-power year (EFPY) inside surface neutron fluence values for the CR-3 reactor vessel beltline materials were conservatively increased by 7 percent. However, the licensee reported that the increase in neutron fluence resulted in adjusted reference temperatures (ARTs) that were below the values utilized for the current P-T limit curves, which were valid through 32 EFPY. In an RAI, the NRC staff requested that the licensee explain how an increase in neutron fluence resulted in a lower ART for the RPV materials. In the licensee's response to the RAI, the licensee indicated that current neutron fluence was calculated utilizing the NRC-approved fluence methodology documented in Framatome topical report BAW-2241-P, "Acceptance for Referencing of Licensing Topical Report Methodologies," dated February 18, 1999 (Reference 7). Reference 7 contains the methodology, assumptions, and the benchmarking that qualified the method used in the calculations. The neutron fluence values at 32 EFPY for the RPV materials were less than previously reported because the licensee had implemented low leakage cores and the BAW-2241-P methodology resulted in lower neutron fluence at 32 EFPY. Since the neutron fluence decreased, the ARTs also decreased. The NRC staff finds this analysis reasonable and therefore acceptable.

The licensee reported the neutron fluence at critical locations in the RPV. The NRC staff utilized the neutron fluence reported by the licensee to confirm the RT_{PTS} values for the limiting RPV material and to evaluate the RPV's USE.

The NRC staff confirmed that the RT_{PTS} value for the limiting RPV material at 32 EFPY was 206^{o}F, which is significantly below the PTS screening criteria, 270^{o}F. The NRC staff's calculation was performed using the methodology documented in 10 CFR 50.61 and the material data documented in the Reactor Vessel Integrity Database (RVID).

The NRC staff has concluded that the 7-percent increase is a conservative assumption compared to the requested 0.9 percent power uprate, and is acceptable. The NRC staff has confirmed that the licensee has used methods acceptable to the NRC staff and used conservative assumptions to evaluate the requested change. Therefore, the NRC staff concluded that the fluence values proposed for the calculation of the P-T limit curves and RT_{PTS} at 32 EFPY are acceptable.

It should be noted that the original fluence values used for the calculation of the P-T limit curves and RT_{PTS} were higher than those estimated with the power uprate. This is due to the fact that early estimates were based on an out-in refueling scheme, which maximized neutron leakage. Current refueling practice employs neutron economy by minimizing leakage to maximize fuel cycle length. An additional reason is changes in the chemistry of the two beltline critical elements: at 1/4T circumferential weld 1769, and at 3/4T longitudinal welds W-8 and W-18.

For CR-3, the fluence used for the current P-T curves bounds the values estimated for the proposed power uprate; therefore, the P-T limit curves do not need to be revised.

Appendix G to 10 CFR Part 50 requires that the licensee demonstrate that materials with Charpy USE values less than 50 ft-lb contain margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The licensee had provided this equivalent margin analysis prior to the uprate. In the September 30, 2002, RAI response, the licensee indicated that the power uprate has a very small impact on these margins, and the existing equivalent margins analysis in place continues to bound the uprated fluences. The NRC staff confirmed that the CR-3 limiting axial and circumferential beltline welds would meet the margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The NRC staff utilized the copper fluence model documented in NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," (by E. D. Eason, et al.) and the criteria and methodology documented in Appendix K to Section XI of the ASME Code and RG 1.161. This analysis demonstrates that the CR-3 reactor vessel beltline materials will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code, in accordance with the requirements of Section IV.A.1.a of Appendix G of 10 CFR Part 50.

The CR-3 reactor vessel materials surveillance program is part of the B&W Owners Group Integrated Surveillance Program. This program has been approved by the NRC staff. The licensee indicates that if the revised fluence projections do not appreciably exceed the fluence projections used in developing the current capsule withdrawal schedules, then the current withdrawal schedules remain valid. The capsule withdrawal schedules are determined in accordance with 10 CFR Part 50, Appendix H, which references ASTM E-185. There is wide latitude in this standard for when capsules need to be withdrawn to monitor neutron irradiation embrittlement. Therefore, since the power uprate results in a small change in neutron fluence, the withdrawal schedule need not be changed to meet the requirements of 10 CFR Part 50, Appendix H.

3.4.1.5 Reactor Vessel Internals Integrity

The NRC staff reviewed Section 4.7.3 and Table 1 of Attachment A to the licensee's June 5, 2002, letter. Section 4.7.3 discusses the impact of the power uprate on RCS temperature. Table 1 of the letter indicates that the uprate will increase RCS hot leg temperature by less than 1°F. This small change in temperature should not significantly impact the susceptibility of these components to void swelling and irradiation-assisted stress corrosion cracking. Therefore, the NRC staff concludes that the existing reactor vessel internals inspection program does not need to be changed.

3.4.1.6 Alloy 600 PWSCC

Section 4.7.2.5 of Attachment A to the licensee's June 5, 2002, letter provides an evaluation of the impact of the power uprate on Alloy 600 RPV VHPs and other Alloy 600 components and welds. In 2001 and 2002, the NRC issued three bulletins concerning Alloy 600 PWSCC of RPV VHPs. The bulletins requested information on the adequacy of planned inspections for detection of PWSCC. The industry proposed a model to predict the susceptibility of Alloy 600 and its associated weld material to PWSCC. The model is dependent upon time and temperature. The industry utilized the model to rank the RPV VHP into three categories: high, moderate, and low susceptibility. The higher the susceptibility ranking the more stringent the planned inspection. CR-3 RPV VHPs were categorized as high susceptibility. As a result of having a high susceptibility ranking, CR-3 committed to replacing the head at the next outage. Since the power uprate only results in a 0.2°F increase in head temperature and the licensee will install a new head at the next refueling outage, PWSCC will not be significant for RPV VHPs at the uprated condition. The next refueling outage will take place in the fall of 2003.

The licensee evaluated the Framatome Alloy 600 ranking model (the same model as used in the RPV VHP ranking) and determined that the current relative PWSCC ranking of Alloy 600 components will not change after the power uprate. The three components most susceptible to PWSCC are all in the pressurizer, and therefore are not affected by the power uprate. Hence, the impact of the power uprate on Alloy 600 PWSCC is considered very limited and bounded by current B&W Owners Group aging management programs for Alloy 600. Since the power uprate results in a small increase in hot leg temperature, the NRC staff agrees that the relative ranking of susceptible components will not change and the inspection program for these components need not be changed as a result of the power uprate.

3.4.2 RCS Summary

The NRC staff has reviewed information provided in Sections 4.7.2 and 4.7.3 of Attachment A to the June 5, 2002, letter and the licensee's RAI responses. The NRC staff concluded that information provided by the licensee regarding the integrity of the RPV, reactor vessel internals, and Alloy 600 components in CR-3 supports the operation with a 0.9-percent power uprate, as set forth above, and is therefore acceptable.

3.5 <u>Structural and Pressure Boundary Integrity of the NSSS and BOP Systems</u>

3.5.1 Evaluation of Structural and Pressure Boundary Integrity of the NSSS

The review is to assure the structural and functional integrity of piping systems, mechanical components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes. The acceptance criteria for this review are based on 10 CFR Part 50, §50.55a. Specific review criteria are contained in SRP Section 3.9.

The NRC staff reviewed the effects of the proposed CR-3 power uprate as it relates to the structural and pressure boundary integrity of the NSSS and the BOP systems and components. Information reviewed was provided in Sections 4.7, 4.8, and 4.18 of Reference 1. The NRC staff's evaluation concerning the effects of the power uprate on the pertinent components is provided below.

3.5.1.1 Reactor Vessel Structural Evaluation

The licensee stated that 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 the 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. The licensee indicated that, 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 licensee concluded that the existing stress analyses for the reactor vessel remain applicable for the uprated power conditions. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the reactor vessel supports the power uprate.

3.5.1.2 Core Support Structures and Vessel Internals

The licensee stated that 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 is proposed 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 licensee concluded that the existing loads, stresses, and fatigue values remain valid. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the existing loads, stresses, and fatigue values for the core support structure and reactor vessel internals support the power uprate.

3.5.1.3 Reactor Internals Flow-Induced Vibration (FIV)

The licensee stated that the uprate by itself does not result in an appreciable change in RCS mass flow compared to the current operational value (less than 0.1 percent). The design-basis analyses used conservative flow velocities that bound those resulting from the power uprate. The very slight change in mass flow, which is bounded by the existing design-basis analyses, has a negligible impact on the reactor internals components. Thus, core lift and FIV on reactor vessel internals are not significantly affected by the power uprate.

The licensee further indicated that once-through steam generator (OTSG) tube plugging reduces the RCS flow. However, steam generator tube plugging does not adversely affect primary component FIV. FIV 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 FIV analyses remain bounding. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the existing reactor internals FIV analyses support the power uprate.

3.5.1.4 Mechanical Loads Evaluation

The licensee stated that the uprated conditions do not affect the current design bases for seismic and LOCA loads. Thus, it was not necessary to reevaluate the structural effects from seismic loads, and the LOCA hydraulic and dynamic loads. The licensee also indicated that, in regard 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 within the bounds of the current design temperature ranges. Thus, the licensee concluded that the uprated conditions do not affect the mechanical loads. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the mechanical loads support the power uprate.

3.5.1.5 CRDMs Structural Evaluation

The licensee reviewed the impact on the existing CRDM design-basis analyses for the uprated conditions. 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 the design limits in Table 1 of Attachment A to the licensee's June 5, 2002, letter. The design conditions in the existing analyses are based on the RCS functional specification. The licensee further stated that 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 licensee concluded that the existing stress analyses for the CRDMs remain applicable for the uprated power conditions. This applies to both Type “A” and “C” CRDMs. The NRC staff reviewed the licensee’s analysis and, based on the reasons set forth above, concludes that the existing CRDM design-basis analyses support the power uprate.

3.5.1.6 Reactor Coolant Loop Piping and Support Structural Evaluation

The licensee reviewed the impacts of the uprated conditions 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 the design limits. The design conditions in the existing analyses are based on the RCS functional specification. The licensee further stated that 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 licensee concluded that the existing stress analyses for the reactor coolant piping and supports remain applicable for the uprated power conditions. The NRC staff reviewed the licensee’s analysis and, based on the reasons set forth above, concludes that the existing design-basis analyses for the reactor coolant piping and supports support the power uprate.

3.5.1.7 RCP Evaluation

The licensee evaluated the impact of the uprated conditions on the existing design-basis analyses for the RCPs. 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 the design limits. The design conditions in the existing analyses are based on the RCS functional specification. The licensee further stated that the uprate 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 licensee concluded that the existing stress analyses for the RCPs remain applicable for the uprated power conditions. The NRC staff reviewed the licensee’s analysis and, based on the reasons set forth above, concludes that the existing design-basis analyses for the RCPs support the power uprate.

3.5.1.8 RCP Motor Evaluation

The licensee determined that 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.

The licensee also indicated that 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 gallons per minute (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 licensee also evaluated the RCP motors based on the uprated conditions for continuous operation. In addition, the licensee stated that the uprated conditions will have no effect upon motor operation during pump start and cold loop operation and that the power uprate insignificantly changes the RCS mass flow. Thus, the pump head capacity performance and NPSH requirements are unchanged. Therefore, the licensee concluded that the uprated conditions will have a negligible impact on the RCP and RCP motors. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the RCP motors support the power uprate.

3.5.1.9 OTSG Thermal Hydraulics and Flow-Accelerated Corrosion (FAC) Program Evaluation

3.5.1.9.1 OTSG Structural Evaluation

The licensee evaluated the impact of the power uprate conditions on the existing design-basis analyses for the steam generator structures. The RCS design and operating pressure will not be changed under the power uprate. The operating temperatures (T_{hot} and T_{cold}) have slight changes, but the changes are within the design limits and conditions. The operating transients will not change as a result of the power uprate and no additional transients have been proposed; therefore, the existing loads, stresses, and fatigue values for the steam generator structures remain valid. The NRC staff agrees with the licensee that the existing stress analyses for the OTSG structures remain applicable for the power uprate conditions, as set forth above, and that the impact of the power uprate conditions on the existing design-basis analyses for the steam generators is minimal.

3.5.1.9.2 OTSG Tube Integrity

The licensee evaluated the impact of the power uprate on the existing design-basis analysis for the steam generator tubes. The licensee evaluated the steam generator tubes with regard to stress and fatigue usage. The licensee demonstrated that the existing structural and fatigue analyses of the steam generator tubes continue to comply with the ASME Code limits, as incorporated in 10 CFR 50.55a, for the revised design conditions and power uprate conditions.

The licensee evaluated the impact of the power uprate on previously plugged and repaired tubes. Degraded steam generator tubes have been repaired using welded tube plugs, mechanical tube plugs, mechanical sleeve plugs, repair rolls, mechanical sleeves, and tube stabilizers. The licensee reviewed the impact of the power uprate on the existing qualification and design calculations for these tubes. The licensee showed that the temperature changes as a result of the power uprate are bounded by those used in the repair rolls, sleeve and plug qualifications, and analyses. The licensee also showed that the flow increase in the secondary side did not affect the functional integrity of these tubes. The licensee also found that the existing structural and fatigue analyses remain valid. The licensee concluded that the existing mechanical and welded plugs, mechanical sleeves, repair rolls, and tube stabilizers are acceptable for the power uprate conditions. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the existing design-basis analysis for the steam generator tubes supports the power uprate.

The licensee evaluated the impact of the power uprate on the steam generator inspection program. The licensee stated that the current steam generator inspection program at CR-3 follows the inspection guidelines contained in the latest revision of the Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines. The licensee stated that it inspects all active and potential degradation in the CR-3 steam generators. It will expand inspection plans and repair degraded tubes on the basis of condition monitoring and operational assessments of the steam generator tubes. Potential degradation growth rate changes associated with potential effects of the uprate will be incorporated into the operational assessment. The licensee concluded that the modest power uprate will not require a change to the steam generator tube inspection program. The NRC staff agrees with the licensee, based on the reasons set forth above, that the uprated conditions will not significantly impact tube inspection during future outages.

The licensee evaluated the impact of the power uprate on the 40-percent through-wall plugging limits for a degraded tube or sleeve. The RCS pressure and secondary side steam pressure will not change. 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 plugging conditions. The licensee stated that the current tube pressure differentials and tube loads during faulted conditions remain valid. The increase in tube temperature will have an insignificant effect on the tube strength properties. The effect of a 0.2°F temperature increase in T_{hot} may decrease the crack initiation time (i.e., faster crack initiation) by about 1 percent, which is insignificant. The current crack growth estimate remains valid and the current 40-percent tube and sleeve plugging criteria is applicable for the power uprate conditions. The licensee will consider the higher temperature in future growth rate analyses. The licensee concluded that the current RG 1.121 analyses for the 40-percent through-wall plugging limit remains valid under the uprate conditions.

The NRC staff finds the licensee’s evaluation and reasoning to be acceptable because they performed their analysis in accordance with NRC-approved methodology (RG 1.121) and used appropriate uprate condition inputs, thereby satisfying the requirements of 10 CFR 50.55a.

On the basis of the above analysis, the NRC staff concludes that the power uprate will not have a significant impact on the structural integrity of the steam generator tubes at CR-3.

3.5.1.9.3 OTSG FIV

With respect to the FIV of tubes in the OTSG, the licensee stated that rather than perform the subsequent uprate FIV analyses at the best estimate conditions, the analyses in Reference 9 were performed at a greater flow rate. The uprated “design” flow rate was specified as 2 percent greater than the previous FIV-analyzed condition. 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 reassessed with the latest techniques and input parameters in an FIV analysis. The licensee also stated that 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.

The NRC staff reviewed the licensee’s reassessment techniques and input parameters used in the FIV analysis for CR-3 as presented in References 1, 3, 4, and 9. In reviewing References 1 and 9, the NRC staff found the techniques used for reassessment to be acceptable. However, the input parameters used may not be appropriate. In particular, the use of high damping ratios (e.g., 5 percent for virgin tubes) required detailed justification. Since the fluid-elastic stability margin (FSM) of the tube is proportional to the square root of the damping ratio of the tube, the

FSM claimed in Reference 9 may not be accurate. In response to the NRC RAIs, the licensee provided additional information in References 3 and 4.

In Section 6.0 of Attachment C to Reference 3, the licensee provided a summary of damping test results. The results do not support the use of 5-percent damping for a virgin tube. Furthermore, in Attachment A to Reference 3, in response to the NRC staff's RAIs, the licensee stated that since a Connors' constant of 2.4 is considered a lower bound value for fluid-elastic instability analysis and the 5-percent damping is considered <u>an upper bound value</u> for the damping, Framatome ANP believes the combination of a Connors' constant of 3.3 and tube-to-tube support plate interaction damping of 3 percent to be appropriate. The NRC staff concluded that the use of a 5-percent damping ratio for a virgin tube is not acceptable without detailed <u>statistical</u> data justification. Thus, the NRC staff submitted RAIs to obtain further details.

In response to the NRC staff's RAIs, the licensee stated in Reference 3 that the FIV methodologies used by Framatome ANP outlined in Attachment C to Reference 3 have been employed for power uprate FIV analysis at CR-3. Currently, Framatome ANP performs fluid-elastic instability analysis of the OTSG tubes and stabilizers employing a Connors' constant of 3.3 and 3-percent damping value for a loosely supported tube. The 3-percent damping value is used only for the fluid-elastic instability analysis where the tube or stabilized tube experiences large amplitude vibrations. The nominal tube-to-tube support plate diametrical clearance is 18 mils. For random turbulence-induced vibrations where the displacements are smaller, 2-percent viscous damping is used. Based on the input parameters currently used for CR-3 as stated by the licensee, the NRC staff concludes that the CR-3 OTSG FIV analysis for the 0.9-percent power uprate is acceptable.

3.5.1.9.4 FAC Program

The 24 MWt power increase will result in a slight increase in the flow rates in certain plant systems. This increase will affect FAC. The licensee calculated the corresponding increase in corrosion rates using the FAC program. The program is based on EPRI NSAC/202L, "Recommendation for an Effective Flow-Accelerated Corrosion Program," and uses the CHECWORKS computer code for predicting FAC corrosion rates. The licensee has determined that because the highest change of flow velocity occurs in the FW system, the most significant increase in corrosion rates will occur in that system. FPC calculated that the current corrosion rate in the FW system is 5 mil/year and after the power uprate it will increase by 0.6 percent. The increase is, therefore, negligible.

The NRC staff reviewed the methodology used by the licensee and concludes that by using the CHECWORKS predictive code and following the procedure recommended by EPRI, the licensee was able to ascertain that the power uprate will not cause any meaningful changes in FAC and no increase of corrosion wear rates in the FAC-susceptible components will occur.

3.5.1.9.5 OTSG Thermal Hydraulics and FAC Summary

The NRC staff concludes that the power uprate will not adversely impact the FAC-susceptible components managed by the FAC program. Also, based on the information the licensee provided, as discussed above, the NRC staff concludes that the power uprate will not have a significant impact on the SG tube structural and leakage integrity.

3.5.1.10 Pressurizer Structural Evaluation

The licensee evaluated the impact of the power uprate on the existing design-basis analyses for the pressurizer and concluded that 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. 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 licensee concluded that the existing stress analyses for the pressurizer remain applicable for the power uprate conditions. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the existing design-basis analyses for the pressurizer support the power uprate.

The licensee also indicated that the power uprate was reviewed for impact on thermal stratification in the surge and spray lines. The licensee concluded that the evaluation showed that the effects of thermal stratification are either bounded by the existing analysis or negligibly affected. The NRC staff reviewed the licensee's analysis and finds that the impact on thermal stratification in the surge and spray lines are either bounded by the existing analysis or negligibly affected. Therefore, the power uprate is acceptable.

3.5.1.11 RCS Attached Piping and Support Structural Evaluation

The licensee evaluated the impact of the power uprate on the existing design-basis analyses for the RCS attached piping and supports and concluded that 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 the design limits. The design conditions in the existing analyses are based on the RCS functional specification. The licensee further stated that the uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate, and no additional transients have been proposed, the existing loads, stresses, and fatigue values remain valid. Thus, the licensee concluded that the existing stress analyses for the RCS attached piping and supports remain applicable for the power uprate conditions. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the existing design-basis analyses for the RCS attached piping and supports support the power uprate.

3.5.1.12 Leak Before Break (LBB)

The licensee's current LBB evaluation was performed for the RCS primary loops to provide technical justification for eliminating a large primary loop pipe rupture as the structural design basis. The licensee performed an evaluation (Reference 4 of the FPC letter dated June 5, 2002) that determined the impact of the uprate condition on the LBB margins. The results show that the effects of the uprate condition on the LBB margins are negligible, and the LBB conclusions remain unchanged. The NRC staff reviewed the licensee's analysis and, based on the reasons set forth above, concludes that the licensee's current LBB evaluation supports the power uprate.

3.5.2 BOP Systems

The licensee performed a heat balance analysis on the secondary systems for the proposed uprate using the uprated NSSS parameters. 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 are those that are (or could be) directly affected by the uprate. The licensee indicated that these systems include the Main Steam System, Auxiliary Steam System, Condensate System, Condenser Air Removal System, Condensate Polishing Demineralizer System, FW System, Auxiliary Feedwater System, FW Heaters, Main Turbine, Generator Gas System, Moisture Separator/Reheater, Extraction Steam System, Heater Drain System, Circulating Water System, Decay Heat Closed-Cycle Cooling System, Nuclear Services Closed-Cycle Cooling System, Nuclear Services and Decay Heat Seawater System, Secondary Services Closed-Cycle Cooling System, and Main Reactor Building Fan System.

3.5.2.1 Main Steam (MS) System

In Section 4.8.1 of the June 5, 2002, submittal, the licensee stated that the MS system is not impacted by the power uprate. However, the licensee did not explain whether the MS system and its components were originally designed to operate at 2568 MWt. The NRC staff asked the licensee to provide additional information on the maximum steam flow rate and steam pressure corresponding to the power uprate conditions and to explain whether the safety relief valves are capable of supporting plant operation at these conditions.

In its response of August 13, 2002, the licensee stated that the design of the MS system and its components can accommodate power operation at 2568 MWt. The total code safety relief valve capacity is over 13 million pounds mass per hour (Mlbm/hr), which provides more than the capacity required for the design transient of a turbine trip from 112 percent of 2568 MWt core power (which requires 11.1 Mlbm/hr). Accordingly, the NRC staff finds that the MS system and the capacity of the relief valves are designed to accommodate the power uprate conditions, and therefore are acceptable.

3.5.2.2 MS and Main FW Piping

The licensee evaluated the impact of the power uprate on the existing design-basis analyses for the MSmain steam and main FW 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 NRC staff agrees that the existing loads, stresses, and fatigue values remain valid.

3.5.2.3 Nuclear Services and Decay Heat Seawater (RW) System

The RW System provides cooling water to the Nuclear Services Closed-Cycle Cooling (SW) System and Decay Heat Closed-Cycle Cooling System for heat removal during accidents and normal operation. The licensee indicated that the power uprate has no impact on the system based on its accident analysis. However, at the power uprate level, the plant heat discharge to the RW system increases during normal operation and after accidents. The RW system draws seawater from the Gulf of Mexico to cool these safety-related cooling systems. The seawater temperature for the heat sink is normally higher in the Gulf area during summer operation. At the power uprate level, the service water temperature in the SW and Decay Heat Closed-Cycle Cooling systems will be higher than the current level. The NRC staff requested that the licensee explain whether the temperature variation of the heat sink resulting from weather changes was considered in the accident analysis.

In its response, the licensee stated that the accident analyses were performed assuming a maximum ultimate heat sink (UHS) temperature of 95 °F (at the Gulf of Mexico) with reactor power at 102 percent of 2568 MWt. Section 3.7.11 of the CR-3 Improved Technical Specifications (ITS) requires that the UHS temperature must be below 95 °F. If not, the plant must be placed in Mode 3 (hot standby) in 6 hours, and in Mode 5 (cold shutdown) in 36 hours. As long as the UHS remains below 95 °F, the RW and SW systems are capable of rejecting the maximum post-accident heat loads to the UHS. With the UHS temperature below this limit, the accident analyses remain valid for the power uprate conditions. The NRC staff finds that the UHS temperature during the summer months was considered in the original accident analyses, which included operation at the higher SPU power level and, therefore, finds the RW system design acceptable.

3.5.2.4 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 the control complex chillers upon loss of the SW system. However, the SC system will limit the turbine generator output when the heat load is higher than normal (e.g., high reactive load on the generator). Because the heat sink (Gulf of Mexico) temperatures are high during summer operation, the licensee stated that the plant operators will monitor the temperature of the loads served by the SC system. If the equipment approaches its design limit, the generator power may need to be reduced. 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. However, for most operating conditions, the SC system will continue to perform its function at the power uprate conditions. This system is not safety-related. However, the limitations of this system have potential economic consequences due to reduced power generation but do not adversely affect plant safety. Therefore, the NRC staff finds the SC system design acceptable.

3.5.2.5 Remainder of BOP systems

For the remainder of the BOP systems, the licensee concluded from a review of each system that either no changes in design are required and that all parameters remain within design, or the effects of the changes are negligible. The NRC staff reviewed the licensee's analyses and, based on the reasons set forth above, concludes that the remainder of the BOP systems support the power uprate.

3.5.3 Structural and Pressure Boundary Integrity of the NSSS and BOP Systems Summary

On the basis of its evaluations, as described in Sections 3.5.1 and 3.5.2 above, the NRC staff finds that the licensee's evaluations for the NSSS and BOP piping, components, and supports, operating at the proposed 0.9% power uprated conditions, are bounded by the existing analyses and continue to satisfy the licensing codes of record and the original design basis. Therefore, the NRC staff finds the plant components at CR-3 acceptable for the power uprate condition.

3.6 NSSS/BOP Fluid Systems Interface

3.6.1 NSSS Fluid Systems

The NRC staff reviewed the NSSS/BOP Fluid Systems analyses. The NRC's acceptance criteria for reviewing the NSSS/BOP Fluid Systems analyses are based on 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Reactors," and 10 CFR Part 50 Appendix K, "ECCS Evaluation Models." Specific review criteria are contained in SRP Sections 5.2.1.1 and 6.3.

3.6.1.1 RCS

The licensee stated that the revised operating conditions all remain within the design basis of CR-3. Furthermore, the accident analyses have been performed at power levels that bound the uprated conditions. For this reason, the NRC staff finds that the RCS is unaffected by the power uprate.

3.6.1.2 ECCS

The licensee stated that since the accident analyses are unchanged by the power uprate, the uprated conditions have no direct effect on the overall performance capability of the ECCS. For this reason, the NRC staff finds the licensee's conclusion acceptable.

3.6.1.3 Core Flood System

The licensee stated that this system is not affected by the power uprate, since the accident analysis remains unchanged by the power uprate. For this reason, the NRC staff finds the licensee's conclusion acceptable.

3.6.1.4 High Pressure Injection (HPI) System

The licensee stated that this system is not affected by the power uprate, since the accident analysis remains unchanged by the power uprate. For this reason, the NRC staff finds the licensee's conclusion acceptable.

3.6.1.5 Makeup Function of HPI/Makeup System

The uprated conditions are bounded by the design conditions in the functional specification. Therefore, the NRC staff finds that operation of the makeup/HPI system is unaffected by the power uprate.

3.6.1.6 Low Pressure Injection (LPI) System

The licensee stated that this system is not affected by the power uprate, since the accident analysis remains unchanged by the power uprate. For this reason, the NRC staff finds the licensee's conclusion acceptable.

3.6.1.7 Decay Heat Function of LPI/Decay Heat (DH) System

The power uprate will cause a slight increase in DH. The licensee stated that the increase is less than 1 percent, and well within the design capability of the LPI/DH system. For this reason, the NRC staff finds the licensee's conclusion acceptable.

3.6.2 BOP Fluid Systems Interface Evaluation

3.6.2.1 Spent Fuel Cooling (SF) System

The SF system removes decay heat from the spent fuel pool water by the use of heat exchangers. The SF system also removes radioactive materials and particulates through filters and ion exchange units. The SF system is designed with a cooling capacity based on the maximum heat generation rate that would result from the maximum inventory of spent fuel assemblies and accounting for the burn-up and post-irradiation decay time of the fuel to be stored. When operating at the power uprate level, the decay heat load will increase in proportion to the power increase. The licensee stated that the decay heat of the fuel discharged to the spent fuel pool may slightly increase (less than 1 percent), but is still well within the design limits of the cooling system. The NRC staff's review found that the forced cooling capacity of the SF system would be capable of supporting operation at the power uprate level because the reactor core was originally designed for 102 percent of the proposed 2568 MWt limit. Therefore, the NRC staff finds the SF system design acceptable.

3.6.2.2 Reactor Building Spray (RBS) System

The RBS system was designed with a sufficient heat removal capability to rapidly reduce containment pressure and temperature following a design-basis LOCA. The reactor building integrity analysis was performed at 102 percent of the proposed 2568 MWt limit. Therefore, the NRC staff finds that the RBS system is not affected by the power uprate.

3.6.2.3 Emergency Feedwater (EFW) System

In the event of a loss of the MFW system during accident conditions, the EFW system provides FW to the steam generators for heat removal. The EFW initiation and control system will automatically control the steam generator level to prevent reactor core damage during emergencies. The EFW system was analyzed assuming a maximum power level of 2619 MWt (i.e., 102 percent of the proposed limit) and the resulting decay heat level. Therefore, the NRC staff finds that the EFW system is not affected by the power uprate level.

3.6.3 NSSS/BOP Fluid Systems Interface Summary

The NRC staff has completed its review of the licensee's submittals and concludes that the licensee's safety analyses with respect to NSSS and BOP system performance at the proposed power uprate conditions are in accordance with the guidelines of the SRP. Since the original plant design and construction bounds the power uprate level (102 percent of 2568 MWt), the proposed change is found to be acceptable relative to NSSS and BOP system performance.

3.7 Containment/Fire Protection Systems

3.7.1 Containment/Fire Protection

The NRC staff reviewed the containment analysis. The NRC's acceptance criteria for reviewing the containment analysis are based on 10 CFR 50.55a, "Codes and standards." Other relevant parameters are listed in CR-3 FSAR Tables 14-44, 14-45, and 14-46. Specific review criteria are contained in SRP Section 3.8.1.

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent performance of necessary plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure structures, systems, and components required for the safe shutdown of the plant are protected from the effects of a fire and continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50 for the development of a fire protection plan to ensure the capability to safely shut down the plant. Specific review criteria are provided in SRP Section 9.5.1.

The licensee's calculation for the safe shutdown fire analysis uses a power level of 2568 MWt, thus bounding the proposed power uprate. The licensee states that no plant changes have been made that would affect fire loading, combustibles, or fire initiation sources. The NRC staff's review determined that the power uprate does not affect CR-3's safe shutdown fire analysis since the original analysis bounds the proposed power uprate.

3.7.2 Containment Integrity Analysis

In its submittal of June 5, 2002, the licensee did not discuss the containment integrity analysis for CR-3 or explain whether a LOCA analysis for the power uprate condition was performed. The NRC staff was unable to determine the acceptability of the containment safeguards systems for mitigating the consequences of a hypothetical large-break LOCA at the power uprate level. In a letter dated July 16, 2002, the NRC staff requested the licensee to provide additional information for the reactor building integrity analysis.

In a letter dated August 13, 2002, the licensee stated that the existing containment analysis for a large-break LOCA was performed based on 102 percent of the proposed 2568 MWt limit using the CONTEMPT computer code (BAW-10095A, Revision 1). The containment was designed to withstand an internal pressure of 55 psig with a 1.5 safety factor. The peak containment temperature and pressure were determined to be 278.4 $^{\circ}$F and 54.2 psig. These values were used in the EQ analysis. Therefore, no new containment integrity analysis was required for the power uprate application.

The NRC staff finds that the licensee's containment integrity analysis and the computer code used for the analysis were previously evaluated and found to be acceptable by the NRC staff during original plant licensing. The impact of the power uprate is negligible because the power uprate level is bounded by the original containment integrity analysis. Since the original plant design and construction bound the power uprate level (102 percent of 2568 MWt), the NRC staff finds the proposed change acceptable relative to containment integrity.

3.7.3 Containment/Fire Protection Systems Summary

Based on the review of the licensee's analysis as set forth above, the NRC staff finds that the results are reasonable, conservative, and therefore acceptable with respect to the containment and fire protection systems.

3.8 Radiological Analysis

3.8.1 Radiological Consequences Regulatory Basis

The NRC staff reviewed the design-basis accident radiological consequences analyses. The NRC's acceptance criteria for radiological consequences analyses using the alternate source term are based on (1) GDC 19 for control room habitability and (2) 10 CFR 50.67 for radiological consequences of a postulated accident. Specific review criteria are contained in SRP Section 15.0.1.

3.8.2 Assessment of Radiological Consequences

The licensee performed an assessment of the impact of the power uprate to 2568 MWt on the safety analyses in Chapter 14 of the FSAR. This assessment is described in a letter dated June 5, 2002, which states 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. In addition, all analyses use previously approved NRC methods and codes. All Chapter 14 radiological analyses were performed assuming a core power of 2619 MWt using the alternative source term in accordance with 10 CFR 50.67 as permitted in CR-3 License Amendment 199, dated September 17, 2001. In its October 31, 2002, RAI response, the licensee states that the power uprate will not impact CR-3's compliance with GDC 19, "Control Room." The licensee concluded that for all areas reviewed, the doses expected due to the proposed changes will be bounded by or do not increase currently approved values.

The NRC staff reviewed the CR-3 FSAR and the licensee's amendment request describing the proposed change. A review of Chapter 14 of the FSAR, which describes radiological consequence analyses, indicates that the analyzed power level is 2619 MWt, or approximately a 2.9-percent increase from the current RTP of 2544 MWt, and is 2 percent greater than the requested power. According to the licensee, neither the assumed reactor power of 2619 MWt nor the licensing basis methodologies have been changed in support of the proposed amendment to increase the RTP to 2568 MWt (approximately 0.9 percent increase). The radiological consequences presented in the FSAR continue to remain bounding upon implementation of the proposed amendment to increase the RTP to 2568 MWt for CR-3.

The results of the NRC staff's assessment were used to confirm the acceptability of the licensee's analysis methodology and conclusion. Based on the considerations above, the NRC staff concluded that the licensee's analyses remain acceptable.

3.8.3 Radiological Consequences Summary

The NRC staff concluded, based upon the information provided by the licensee, as described above, that the proposed changes to the TS are acceptable with regard to the radiological consequences of design-basis accidents. The NRC staff has determined that reasonable assurance exists that the radiological consequences of the proposed change will continue to meet the regulatory requirements of 10 CFR 50.67, 10 CFR Part 100, and GDC 19.

3.9 Human Factors

3.9.1 Human Factors Regulatory Basis

This evaluation is limited to the operator performance considerations resulting from the increased allowable maximum power level. It includes required changes to operator actions, human-system interface changes, and changes to procedures and training resulting from the change in maximum power level.

The NRC staff's guidance for this review includes Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," and NUREG-0800, SRP, Chapter 18 (draft), "Human Factors Engineering."

3.9.2 Human Factors Evaluation

The proposed power uprate is on the order of 24 MWt, or approximately 8 MWe. The power range nuclear instrumentation displays reactor power as a percentage of RTP, thus indications will appear the same at 2568 MWt as they did at 2544 MWt. The licensee states that the operators will be able to view the approximate 8 MWe increase on the generator output instrumentation as well as slight changes on instruments for FW flow, steam flow, T_{hot}, T_{cold}, and loop delta temperature. The automated unit load demand will also show the increased power level because it displays a digital readout of core thermal power.

The licensee stated that 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. Furthermore, numerous normal operating and surveillance procedures will 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.

3.9.3 Human Factors Summary

In view of the above, the NRC staff finds that the licensee has satisfactorily addressed the human factors areas associated with the proposed power uprate. The NRC staff further finds that the power uprate should not adversely affect operator performance or operator reliability. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to human factors issues.

3.10 Flooding

The NRC staff conducts its review in the area of flood protection to ensure that structures, systems, and components important to safety are protected from flooding. The NRC staff's review covers flooding of structures, systems, and components important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focuses on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. Specific review criteria are provided in SRP Section 3.4.1.

The licensee stated that the CR-3 flooding analysis is independent of power level. Therefore, no plant changes are being made that would impact flood-initiating events or mitigating actions.

The NRC staff agrees, based on the reasons set forth above, that the CR-3 flooding analysis is not affected by the power uprate.

3.11 <u>High-Energy Line Break</u>

The NRC staff reviewed the high-energy line break analysis. The NRC's acceptance criteria for reviewing the containment analysis are based on BTP SPLB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and BTP EMEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." Specific review criteria are contained in SRP Sections 6.2.1.4, 3.6.1, and 3.6.2.

In its letter dated June 5, 2002, the licensee stated that changes to the secondary plant due to increased FW and steam flow are very minor. Additionally, no new piping will be classified as high energy and no areas will change their energy classification. The licensee also stated that changes in flow will not significantly increase jet impingement and thrust forces or affect piping stresses that could affect break location designations. Therefore, the NRC staff concludes that the high-energy line break analysis is not impacted by the power uprate.

3.12 <u>Safety-Related Valves</u>

The NRC staff reviewed the safety-related valves analysis. The NRC's acceptance criteria for reviewing the safety-related valves analysis are based on 10 CFR 50.55a, "Codes and standards." Additional information is also provided by the plant-specific evaluations of Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," GL 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

The licensee reviewed safety-related valves with respect to 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 safety-related valves may need minor adjustments prior to implementation of the power uprate level. The licensee stated that these adjustments can be made with the unit at power. Therefore, the NRC staff concludes that the power uprate will not have a significant impact on safety-related valves.

3.1.5 <u>License and TS Changes</u>

The NRC staff reviewed the proposed license and TS changes. The NRC's acceptance criteria for reviewing the proposed TS changes are based on 10 CFR 50.36, "Technical Specifications." Specific review criteria are contained in SRP Section 2.4.14.

The proposed change will revise the CR-3 FOL as follows:

1. The plant operating license paragraph 2.C.(1), Maximum Power Level will be revised to read:

   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).

2. The Improved TS definitions will be revised to read:

   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.

The NRC staff has reviewed the associated license and TS changes in support of operation of CR-3 at the maximum core power level of 2568 MWt. The NRC staff finds that the supporting safety analyses show that the uprated power conditions are bounded by the current FSAR analyses. The NRC staff finds that the proposed license and TS changes adequately reflect the results of the acceptable safety analyses and, therefore, concludes that the proposed license and TS changes are acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 42826). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: December 4, 2002

7.0 REFERENCES

1. Letter from Daniel L. Roderick, (FPC) to NRC, “Crystal River Unit 3 - License Amendment Request #270, Revision 0, Power Uprate to 2568 MWt,“ dated June 5, 2002.

2. Letter from Dale E. Young, (FPC) to NRC, “Crystal River Unit 3 - Response to Request for Additional Information Re: Proposed License Amendment Request No. 270, Revision 0, Power Uprate to 2568 MWt” (TAC No. MB5289), dated August 13, 2002.

3. Letter from Dale E. Young, (FPC) to NRC, “Crystal River Unit 3 - Response to Request for Additional Information Re: Proposed License Amendment Request No. 270, Revision 0, Power Uprate to 2568 MWt“ (TAC No. MB5289), dated September 30, 2002.

4. Letter from Dale E. Young, (FPC) to NRC, “Crystal River Unit 3 - Response to Request for Additional Information Re: Proposed License Amendment Request No. 270, Revision 0, Power Uprate to 2568 MWt“ (TAC No. MB5289), dated October 31, 2002.

5. Letter from Dale E. Young, (FPC) to NRC, “Crystal River Unit 3 - Response to Request for Additional Information Re: Proposed License Amendment Request No. 270, Revision 0, Power Uprate to 2568 MWt” (TAC No. MB5289), dated November 13, 2002.

6. FRA-ANP 32-5013936-01 “Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate,” by L. Yu, Framatome, dated August 7, 2002.

7. FRA-ANP 86-1266133-01, “CR-3 PT Fluence Analysis Report - Cycles 7-10" by M. J. Devan, et al., Framatome, dated August 28, 1998.

8. Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” US NRC, March 2001.

9. FRA-ANP 51-5000475-01, “CR-3 OTSG FIV Margins.”

Table 1 - Summary - Non-LOCA and LOCA Transients Analyses

Event Description	FSAR Bounding Event Reference	Does the power uprate affect the previous analysis?	Disposition of the Accident/Transient Analysis	Staff Conclusion
Uncompensated Operating Reactivity Changes	14.1.2.1	No	Analysis in the FSAR remains valid	Accept
Startup Accident	14.1.2.2	No	Startup Accident is initiated from zero power and is not affected by the power uprate	Accept
Rod Withdrawal at Power	14.1.2.3	No	Analysis in the FSAR remains valid	Accept
Moderator Dilution from Full Power	14.1.2.4	No	Analysis in the FSAR remains valid	Accept
Moderator Dilution Accident during Refueling	14.1.2.4	No	Analysis in the FSAR remains valid	Accept
Cold Water Accident	14.1.2.5	No	A licensing restriction limits the consequences of the cold water accident to the startup of one reactor coolant pump	Accept
Single Pump Coastdown	14.1.2.6	No	Flow coastdown is a function of the reactor coolant pump inertia. This event is analyzed each reload. Analysis in the FSAR remains valid	Accept
Locked Rotor	14.1.2.6	No	The results of the analysis show that DNB will not occur during a locked rotor event	Accept
Four-Pump Coastdown	14.1.2.6	No	The results of the analysis show that DNB will not occur during a four-pump coastdown event	Accept
Stuck-Out, Stuck-In, or Dropped Control Rod Accident	14.1.2.7	No	Analysis in the FSAR remains valid	Accept
Load Rejection/Turbine Trip	14.1.2.8	No	Analysis in the FSAR remains valid	Accept
Station Blackout	14.1.2.9	No	Bounded by the original station blackout analyses and the plant continues to meet requirements of 10 CFR 50.63	Accept
Loss of AC Power	14.1.2.9	No	Analysis in the FSAR remains valid	Accept

Event Description	FSAR Bounding Event Reference	Does the power uprate affect the previous analysis?	Disposition of the Accident/Transient Analysis	Staff Conclusion
Steam Line Failure Accident	14.2.2.1	No	Analysis in the FSAR remains valid	Accept
Steam Generator Tube Rupture	14.2.2.2	No	Analysis in the FSAR remains valid	Accept
Fuel Handling Accident	14.2.2.3	No	Analysis in the FSAR remains valid	Accept
Hot Zero Power Rod Ejection	14.2.2.4	No	Analyzed from zero power and it is not directly affected by the power uprate	Accept
Full Power Rod Ejection	14.2.2.4	No	Analysis in the FSAR remains valid	Accept
Loss-of-Coolant Accidents	14.2.2.5	No	Analysis in the FSAR remains valid	Accept
Makeup System Letdown Line Failure	14.2.2.6	No	Analysis in the FSAR remains valid	Accept
Maximum Hypothetical Accident	14.2.2.7	No	Analysis in the FSAR remains valid	Accept
Waste Gas Decay Tank Rupture Accident	14.2.2.8	No	The dose assessment for the waste gas decay tank rupture accident is based on waste gas decay tank inventories of radioactive nuclides and is independent of power level. Analysis not affected by operation of CR-3 at the uprated power level.	Accept
Loss of Main Feedwater	14.2.2.9	No	Analysis in the FSAR remains valid	Accept
Total Loss of Feedwater Accident	14.2.2.9	No	Analysis in the FSAR remains valid	Accept
Feedwater Line Break	14.2.2.9	Yes	Feedwater Line Break was reanalyzed for the power uprate. All acceptance criteria, such as RCS pressure, minimum DNBR, and offsite doses are within acceptable limits.	Accept

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GENERATING PLANT

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