July 16, 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 RELOCATION OF REACTOR COOLANT SYSTEM PARAMETERS TO THE CORE OPERATING LIMITS REPORT AND 20-PERCENT STEAM GENERATOR TUBE PLUGGING (TAC NO. MB2499)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 204 to Facility Operating License No. DPR-72 for the Crystal River Unit 3. The amendment consists of changes to the existing Improved Technical Specifications (ITS) in response to your letter dated July 24, 2001, as supplemented June 5, and July 1, 2002.

The amendment revises future changes in plant design, including increased levels of once-through steam generator tube plugging. The changes are categorized into two sets. The first set of changes relocate parameters from the ITS to the cycle-specific Core Operating Limits Report (COLR). These parameters are the Variable Low Pressure Trip equation specified in ITS Table 3.3.1-1, and Reactor Coolant System (RCS) pressure limit within Surveillance Requirement (SR) 3.4.1.1. The second set of changes is directly related to tube plugging equivalent to up to 20 percent of all tubes, and addresses its impact. These changes include the revision of the hot leg maximum temperature limit, and the revision of the RCS minimum flow limits for four- and three-reactor coolant pump operation. The RCS limits associated with 20-percent plugging will be maintained in the ITS; however, cycle-specific values for these limits will be relocated to the COLR. The hot leg temperature and RCS flow limit values within SR 3.4.1.2 and 3.4.1.3 "RCS Pressure, Temperature, and Flow DNB [departure from nucleate boiling] Limits," are relocated to reflect their location in the COLR. For both sets of changes, ITS 5.6.2.18(a) is modified to reflect the relocation of cycle-specific values from the ITS to the COLR.

If increased feedwater flow exceeding the licensed limit is needed for any reason with up to 20-percent tube plugging, Florida Power Corporation should provide the flow-induced vibration analysis with sufficient details for staff review to demonstrate the functional integrity of the steam generator tubes for up to 20-percent tube plugging under symmetric and worst-case asymmetric plugging distributions.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

John M. Goshen, 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. 204 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

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ADAMS ACCESSION NO. ML021980259 (Letter) *see previous concurrence

OFFICE	PDII-2/PM	PDII-2/LA	OGC	PDII-2/ Acting SC
NAME	JGoshen	BClayton	STurk*	BMozafari for KJabbour
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SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 204
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 July 24, 2001, as supplemented June 5, 2002, and July 1, 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 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:

    Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 204, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by B. Mozafari Acting for/

Kahtan N. Jabbour, Acting Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: July 16, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 204

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

Remove	Insert
3.3-5	3.3-5
3.4-2	3.4-2
5.0-23	5.0-23
B 3.4-2	B 3.4-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By application dated July 24, 2001, as supplemented June 5, and July 1, 2002, Florida Power Corporation, et al. (FPC, the licensee), requested an amendment to the Facility Operating License for Crystal River Unit No. 3 (CR-3). The amendment would relocate certain reactor coolant system (RCS) parameters to the core operating limits report (COLR) and allow up to 20-percent tube plugging in each Once-Through Steam Generator (OTSG). The changes are categorized into two sets:

- The first set of changes facilitates the direct relocation of parameters from the Improved Technical Specifications (ITS) to the cycle-specific COLR. These parameters are the Variable Low Pressure Trip (VLPT) equation specified in ITS Table 3.3.1-1, and RCS pressure departure from nucleate boiling (DNB) limit within Surveillance Requirement (SR) 3.4.1.1.

- The second set of changes is directly related to an OTSG tube plugging equivalent of 20 percent of all tubes. These changes include the revision of the hot leg maximum temperature limit (SR 3.4.1.2), and the RCS minimum flow limits for three and four Reactor Coolant Pump (RCP) operation (SR 3.4.1.3). The limits for the RCS parameters associated with 20-percent tube plugging will be maintained in the ITS; however, the licensee will relocate cycle-specific values for these limits to the COLR. As such, the licensee has revised the hot leg temperature and RCS flow limit values within SR 3.4.1.2 and 3.4.1.3, "RCS Pressure, Temperature, and Flow DNB Limits," to reflect their location in the COLR. ITS 5.6.2.18(a) is revised to reflect the relocation of cycle-specific values from the ITS to the COLR and is required for both sets of changes.

To summarize, the licensee proposed the following ITS revisions:

Table 3.3.1-1 Function 5, Allowable Value is the RCS Variable Low Pressure equation in the COLR

ENCLOSURE

SR 3.4.1.1 Verify RCS loop pressure meets the RCS loop pressure limits specified in the COLR.

SR 3.4.1.2 Verify RCS hot leg temperature meets the RCS hot leg temperature limits specified in the COLR, AND is $\leq$ 605.8°F.

SR 3.4.1.3 Verify RCS total flow rate meets the RCS total flow rate limits specified in the COLR, AND is $\geq$133.5E6 lb/hr with four RCPs operating or $\geq$ 99.7E6 lb/hr with three RCPs operating.

5.6.2.18(a) Adds the above parameters to the COLR Administrative Section of ITS.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's July 24, 2001, submittal regarding the OTSG Flow-Induced Vibration (FIV) described in Section D.4, "Secondary System Performance and Integrity," of Attachment F to the submittal. By letters dated December 27, 2001, April 23, 2002, May 3, 2002, and June 6, 2002, the staff requested additional information (RAI) from FPC. FPC responded with letters dated June 5, and July 1, 2002. These letters provided clarifying information only and did not expand the scope of the proposed action or change the proposed no significant hazards consideration determination.

The staff has reviewed the information provided by the licensee as described above. The staff's evaluation is provided in section 3. The staff did not review and evaluate Attachments B and D to the licensee's June 5, 2002, letter, because these attachments pertain to the evaluations of power uprate, which are not the subject of this review, and these attachments did not respond to the staff's specific RAI.

2.0 BACKGROUND AND REGULATORY EVALUATION

CR-3 operates with two OTSG, with each OTSG currently limited to 7-percent equivalent tube plugging (combined plugging and sleeving). In anticipation of further tube plugging in the future, FPC requested the CR-3 ITS be amended to incorporate DNB parameters consistent with a 20-percent OTSG tube plugging limit. Additionally, CR-3 requested that they be able to relocate cycle-specific values for these parameters to the COLR. These changes will allow evaluation and operation of CR-3 considering actual OTSG plugging levels using the limits in the COLR. The licensee stated that the resulting DNB parameter limits are acceptable but are overly restrictive while actual plugging values are much lower than 20 percent. The amendment would allow up to an equivalent of 20 percent of the tubes in either or both OTSG to be plugged, including the asymmetric tube case of 0-percent plugging in one OTSG and 20-percent plugging in the other OTSG (0 percent/20 percent). FPC proposed to relocate the VLPT function and minimum RCS pressure DNB limit from the ITS to the COLR to ensure that the cycle-specific variations of these parameters are re-evaluated during the reload analysis process. While plugging an equivalent of 20 percent of OTSG tubes may not require changes to these limits, relocating these values to the COLR will allow the flexibility to utilize the available margins to increase cycle operating margins without the requirement of cycle-specific license amendments.

Additionally, FPC proposed relocating the RCS coolant hot leg temperature and RCS total flow cycle-specific limit values from the ITS to the COLR. Relocating the DNB parameters limit values to the COLR will allow the flexibility to utilize the available margins to increase cycle operating margins without the requirement of cycle-specific license amendments. However, the licensee proposed that the limits set by plugging up to an equivalent 20 percent of OTSG tubes will remain in the ITS. Therefore, cycle-specific limits cannot exceed the ITS limits. The available margins associated with the relocated parameters are those resulting from a 20-percent OTSG tube plugging limit. In response to RAI 14.d (Reference 2), the licensee stated that utilizing the parameter limits for 20-percent OTSG plugging would be limiting for core reload designs when actual plugging is less than 20 percent. For example, if actual plugging was 5 percent of OTSG tubes, a larger value for RCS flow could be used in the safety analyses, rather than the RCS flow value for 20-percent tube plugging. The ability to utilize the associated margins until the proposed 20-percent OTSG tube plugging limit is reached allows for more economical core designs.

NRC Generic Letter (GL) 88-16 (Reference 4) provides guidance for relocating Technical Specification parameters to a COLR. This guidance allows a licensee to use a COLR to include cycle-specific parameter limits that a licensee established using an NRC-approved methodology as long as the applicable limits are consistent with the safety analysis. In evaluating this approach the NRC staff concluded that it is essential to safety that the licensee operate the plant within the bounds of cycle-specific parameter limits, and that the licensee retain a requirement to maintain the plant within the appropriate bounds in the TS. The staff will evaluate the licensee’s proposed relocation of cycle-specific parameters from the ITS to the COLR against the guidance in GL 88-16.

The licensee’s proposed revision of ITS limits directly associated with 20-percent OTSG plugging (RCS Temperature and Flow DNB limits) were also evaluated. Increasing the OTSG tube plugging limit from 7 percent to 20 percent impacts several design and performance characteristics of the plant. As such, the licensee provided a detailed evaluation regarding the impacts of a 20-percent OTSG tube plugging limit on component integrity and performance, and transient and accident analyses. The most significant effects of increasing to a 20-percent OTSG tube plugging limit are reductions in RCS flow, RCS volume, and the primary to secondary heat transfer rate and a slight increase in RCS pressure. The revised ITS limits for these RCS parameters are as follows: a minimum RCS flow of at least 133.5E6 lb/hr with four RCPs running, at least 99.7E6 lb/hr with three RCPs running, and a maximum hot leg temperature of 605.8°F. The licensee also proposes raising the RCS pressure DNB limit by approximately 3 psi to 2064 psig. This pressure value was relocated to the COLR. The impacts of these changes on component integrity and performance, and transient and accident analyses are evaluated in Section 3 of this Safety Evaluation (SE). For each element evaluated the staff will discuss the current licensing basis and the staff’s basis for finding the proposed change acceptable.

The NRC staff has previously approved similar changes for other plants. The staff approved transfers of the requested parameters to the COLR for Arkansas Nuclear One, Unit One (ANO-1) (Reference 5), and for both Byron and Braidwood (Reference 6). Oconee, a similarly designed Babcock and Wilcox (B&W) Pressurized Water Reactor (PWR), has also relocated these parameters to their COLRs. The staff approved new limits to accommodate 20-percent tube plugging for Three Mile Island-1 (Reference 7).

3.0 TECHNICAL EVALUATION

This section documents the NRC staff's evaluation of the licensee's proposed ITS changes listed in the Introduction section of this SE. Section 3.1 evaluates the relocation of the proposed ITS parameters to the COLRs. Section 3.2 evaluates the proposed changes to the ITS limits as a result of increasing to a 20-percent OTSG tube plugging limit. Section 3.3 evaluates the OTSG tube FIV due to the increase in steam generator tube plugging up to 20 percent.

3.1 RELOCATION OF PARAMETERS TO THE COLR

Section 182A of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to state the TS to be included as part of the license. The regulations regarding the content of plant TS are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical Specifications." This section of 10 CFR requires that each licensee authorized operation of a production facility include TS derived from the analyses and evaluations included in the licensee's safety analysis report. This regulation requires the TS to include items in five specific categories, including (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) SRs, (4) design features, and (5) administrative controls. However, the regulation does not specify any particular requirement to be included in a plant's TS.

Section 50.36 of 10 CFR defines four criteria for determining whether a particular limiting condition for operation (LCO) and related SR is required to be included in the TS:

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4. A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

NRC GL 88-16 (Reference 4) was issued to all power reactor licensees and applicants and discusses removal of cycle-specific parameter limits from TS to a COLR. This GL allows licensees to modify their existing TS in this way provided that the following three conditions are satisfied:

1. The licensee establishes a named formal report (COLR) that includes values of the cycle-specific parameter limits. The cycle specific limits must be determined using an NRC-approved methodology and must be consistent with all applicable limits of the safety analysis.
2. The licensee establishes an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information.
3. The licensee modifies individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the COLR.

Using these criteria, the proposed TS to be relocated to the CR-3 COLR are evaluated as follows:

<u>Relocation of the RCS Variable Low Pressure Trip Equation to the COLR (ITS Table 3.3.1-1)</u>

CR-3 ITS Table 3.3.1-1, “Reactor Protection System Instrumentation,” lists reactor trip setpoints and SRs. The RCS VLPT is one of the reactor trip functions. As described in the CR-3 TS Bases, the RCS VLPT in conjunction with the RCS High Outlet Temperature and RCS Low Pressure Trips provide protection for the DNBR Safety Limit. The RPS uses the VLPT equation to initiate a reactor trip whenever RCS pressure and temperature approach the conditions necessary for DNB.

The licensee proposed to relocate this TS parameter to the COLR to ensure that cycle-dependent variations due to future plant modifications continue to be properly evaluated during the reload analysis process. While plugging an equivalent of 20 percent of OTSG tubes may not require changes to this limit, relocating this equation to the COLR will allow the flexibility to utilize available margins to increase cycle operating margins without the requirement of cycle-specific license amendments. The licensee for ANO-1, a similarly designed B&W plant (177 Fuel Assembly (FA)), expected frequent changes to their VLPT setpoint due to future changes in core design, and relocated this parameter to the COLR. As such, the NRC staff considers the VLPT setpoint an appropriate cycle-specific parameter, and has previously approved relocation of this parameter to a COLR for Oconee and ANO-1 Unit 1 (Reference 5).

The staff reviewed the proposed TS change and determined that the cycle-specific VLPT may be modified by the licensee, without affecting nuclear safety, provided that such changes are determined using the NRC-approved methodology specified in CR-3 ITS 5.6.2.18(b). The NRC-approved calculational basis for the VLPT setpoint exists in BAW-10179P-A, “Safety Criteria and Methodology for Acceptable Cycle Reload Analysis” (Reference 15). With regard to the Variable Low Pressure Setpoint equation, Section 6.4 of BAW-10179P-A addresses the generation of the RCS DNB Safety Limits (pressure-temperature limits), while Section 7.6 addresses the methods used to set the VLPT. These sections refer to other approved topical reports, as appropriate, to identify the computer codes and procedures used to analyze these parameters (Reference 2, Question 14.b). BAW-10179P-A is already referenced as an NRC-approved methodology in CR-3 ITS 5.6.2.18(b) and, therefore this section of the CR-3 ITS does not need to be revised. NRC approval and a license amendment would be required prior to using a methodology other than one approved and specified in ITS 5.6.2.18(b). Additionally, the VLPT function does not need to be specifically listed in ITS 5.6.2.18(a) (COLR administrative section). CR-3 ITS 5.6.2.18(a) already lists LCO 3.3.1, “Reactor Protection System Instrumentation,” having been added in a previous license amendment.

Because plant operation will continue to be limited in accordance with the values of the cycle-specific VLPT setpoint, calculated using NRC-approved methodologies, the NRC staff finds the proposed change acceptable and consistent with NRC guidance contained in GL 88-16 on modifying cycle-specific parameters.

Relocation of the RCS DNB Parameter Values to the COLR (ITS 3.4.1)

CR-3 ITS 3.4.1 provides the requirements for RCS DNB parameters. The licensee proposed to relocate the values associated with RCS Pressure, Temperature and Flow to the COLR. Specifically, the following revised TS are proposed:

SR 3.4.1.1 Verify RCS loop pressure meets the RCS loop pressure limits specified in the COLR.

SR 3.4.1.2 Verify RCS hot leg temperature meets the RCS hot leg temperature limits specified in the COLR, AND is ≤605.8°F.

SR 3.4.1.3 Verify RCS total flow rate meets the RCS total flow rate limits specified in the COLR, AND is ≥ 133.5E6 lb/hr with four RCPs operating or ≥ 99.7E6 lb/hr with three RCPs operating.

The licensee proposed a direct relocation of the RCS Pressure limit value from the ITS to the COLR. The limits for the RCS parameters associated with 20-percent tube plugging will be maintained in the ITS; however, cycle-specific values for these limits will be relocated to the COLR. In addition, ITS 5.6.2.18(a) of the COLR administrative section is revised to reflect the relocation of cycle-specific values from the ITS to the COLR.

As described in the CR-3 TS Bases, the TS limits on these DNB parameters ensure that RCS pressure, temperature, and flow will not be less conservative than were assumed in the safety analyses, and therefore provide assurance that the minimum DNBR will meet the required criteria for each of the transients. The RCS pressure limit is consistent with operation within the nominal operating envelope and is above the value used as the initial pressure in the safety analyses. The RCS coolant hot leg temperature limit is consistent with full-power operation within the nominal operating envelope and is lower than the initial hot leg temperature in the safety analyses. The minimum RCS flow rate limit corresponds to that assumed for the DNBR analyses.

The staff previously approved relocation of these DNB parameters to a COLR for Oconee and for Byron and Braidwood Nuclear Power Plants (Reference 6), and also generically approved this change for Westinghouse plants in WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report” (Reference 8). Although CR-3 is not a Westinghouse PWR, the NRC staff has evaluated the licensee’s proposed ITS changes considering the guidance in this WCAP. B&W-designed PWRs do not have an existing Topical Report that can be referenced for relocating TS parameters to a COLR. The proposed CR-3 ITS changes and the reasons for the changes are consistent with WCAP-14483-A.

As discussed in the NRC staff SER for WCAP-14483, a number of licensees have implemented steam generator tube plugging programs. As stated in that SER:

> A number of WOG licensees have implemented T-hot reduction and steam generator tube plugging programs. In these cases, additional margin has been allocated to support the TS and to minimize any licensing impacts associated with cycle-to-cycle changes in RCS T-avg and RCS flow rate. In addition, some licensees have preformed safety analyses which support plant operation at different nominal operating pressures. In these cases, additional margin must be allocated for the pressurizer pressure TS to reflect the most limiting value assumed in the safety analyses to avoid cycle-specific TS changes. Therefore, although these plants may operate with a full power T-avg that is lower than the licensed upper T-avg limit, with higher RCS flow rates than assumed in the tube plugging analysis (due to actual lower steam generator tube plugging levels), or with lower operating pressures, the reactor protection system setpoints must be based on the limiting TS values since the safety analyses were based on these conservative TS values. By relocating these DNB TS parameters to the COLR, the COLR values would reflect the cycle-specific operating conditions and allow reactor trip setpoints to be consistent with actual operating conditions, thereby avoiding the necessity of overly conservative TS limits.

For WCAP-14483, the NRC staff concluded that relocation of the RCS DNB limits to the COLR is acceptable. However, the staff recommended that if RCS flow rate were to be relocated to the COLR, the minimum limit for RCS total flow based on a staff-approved analysis (e.g., maximum tube plugging) should be retained in the TS to assure that a lower flow rate than reviewed by the staff would not be used. The proposed CR-3 ITS change retains the RCS flow and the RCS temperature limits in the ITS. Only the cycle-specific values will be relocated to the COLR.

The staff has reviewed the proposed TS change for CR-3 and has determined that the licensee may modify the cycle-specific values for RCS pressure, temperature, and flow DNB limits without affecting nuclear safety, provided that such changes are determined using the NRC-approved methodology specified in CR-3 ITS 5.6.2.18(b). The design basis for “Transient Core Thermal-Hydraulic Analysis” is discussed in Section 6.7 of the NRC-approved topical report BAW-10179P-A, “Safety Criteria and Methodology for Acceptable Cycle Reload Analysis.” This section discusses the criteria used to establish the LCOs on RCS pressure, RCS hot leg temperature and RCS flowrate and refers to other approved topical reports, as appropriate to identify the computer codes and procedures used to analyze these parameters (Reference 2, Question 14.b). BAW-10179P-A is already referenced as the NRC-approved methodology in CR-3 ITS 5.6.2.18(b), and therefore this section of the CR-3 ITS does not need to be revised. NRC approval and a license amendment would be required prior to using a methodology other than one approved and specified in ITS 5.6.2.18(b). Additionally, the licensee revises ITS 5.6.2.18(a) to add these DNB parameters to the list of items contained in the CR-3 COLR. The NRC staff has judged these proposed changes to be consistent with WCAP-14483-A, and therefore finds this acceptable.

ITS SR 3.4.1.1, RCS pressure DNB limit currently includes minimum RCS loop pressure limits for both four and three RCP operation. In relocating the RCS pressure DNB limit value to the COLR, the licensee proposed a single RCS pressure limit. The licensee developed limits for both four and three RCP operation, and consistent with the current ITS values, for both conditions the limits were nearly identical. The licensee concluded that the small difference between the two values did not warrant maintaining separate limits (Reference 2, question 14.c). The licensee proposed maintaining the limiting value (higher pressure) as the limit for both conditions. The licensee proposed adding the following note regarding this value to the COLR:

> (Assumes 20-percent OTSG tube plugging and bounds either four or three RCPs operating).

The staff finds this to be acceptable because the higher value is chosen, which is conservative with respect to DNBR.

Based on the above discussions, the NRC staff has concluded that the relocation of the VLPT, RCS pressure, RCS temperature and RCS flow DNB parameters to the COLR is acceptable based as follows:

1. These parameters are cycle-specific, and therefore, meet the intent of GL 88-16.

2. Reference to and the requirement for conformance to these limits remains in the CR-3 ITS, assuring conformance with 10 CFR 50.36.

3. Plant operation continues to be limited in accordance with the values of these parameter limits that are established using NRC-approved methodologies specified in CR-3 ITS 5.6.2.18(b) and will ensure that operation will be consistent with applicable limits of the safety analyses.

3.2 20-PERCENT OTSG TUBE PLUGGING LIMIT REVISIONS

CR-3 has two OTSGs, each with 15,531 tubes initially in service to form the mass and energy transport loops that facilitate primary to secondary side heat transfer. Over time, tube degradation mechanisms have warranted repairs such as by sleeving, or removal from service, principally by plugging an affected tube. Presently, CR-3 has a total of 906 tubes plugged and 322 tubes sleeved. This corresponds to an equivalent plugging level of approximately 954 tubes, or 3.1 percent of all tubes. Per OTSG, the equivalent plugs are approximately 227 (1.5 percent) in OTSG-A, and 727 (4.7 percent) in OTSG-B (Reference 1). CR-3 is currently limited to 7-percent OTSG tube plugging in any one OTSG. The licensee's proposed amendment request would allow up to an equivalent of 20 percent of the tubes in either or both OTSG to be plugged, including 0-percent/20-percent asymmetric tube plugging. The licensee evaluated both symmetric and asymmetric tube plugging in establishing the 20-percent OTSG tube plugging limit.

The most significant effects of 20-percent OTSG tube plugging limit are reductions in RCS flow, RCS volume, and the primary to secondary heat transfer rate, and a slight increase in RCS pressure. The proposed revisions to the limits for these RCS parameters are as follows: a minimum RCS flow of at least 133.5E6 lb/hr with four RCPs running, at least 99.7E6 lb/hr with three RCPs running, a maximum hot leg temperature of 605.8°F, and a minimum RCS pressure of 2064 psig.

RCS flow resistance increases due to the reduction in flow area through the OTSG tube primary side. The overall effect is a decrease in system flow, particularly for significant levels of tube plugging. The licensee evaluated the impact on nominal RCS flow, conservatively assuming a high resistance core with a debris filter plate. The licensee determined that the bounding RCS flow penalty, considering both symmetric and asymmetric OTSG tube plugging, is 4.5 percent (Reference 2, question 2.e and Reference 3, question 2). The proposed RCS flow ITS limit values incorporate the bounding 4.5-percent penalty.

The reduction in heat transfer area between the primary and secondary side results in an increased OTSG boiling length, and a decrease in the steam superheat temperature. The reduction in steam superheat temperature requires an increase in feedwater flow to maintain the rated power condition. This and the increased boiling length also lead to a slight increase in the secondary inventory. Further, if the same RCS average temperature is maintained, then the cold leg temperature will decrease slightly, while the hot leg temperature will increase slightly. The licensee quantified this change and found that the hot leg temperature will increase by approximately 1°F, while the cold leg temperature will decrease by approximately 1°F. These changes in temperature are not significant.

Plugging an OTSG also reduces the available RCS volume slightly. As such, with 20 percent of all OTSG tubes plugged, RCS volume will be reduced by approximately 600 ft^3. While this reduction has minimal impact on normal operations, accident and transient responses may be impacted and will be evaluated.

The licensee has also proposed that the RCS pressure limit be increased slightly, by < 3 psi, to 2064 psig. The licensee used the minimum RCS flow associated with 20-percent OTSG tube plugging and a plant-specific hydraulics model to calculate this value (Reference 2, question 13 and Reference 3, question 9).

Increasing the OTSG tube plugging limit from 7 percent to 20 percent and the associated changes in RCS DNB parameter values impacts several design and performance characteristics of the plant. As such, the licensee provided a detailed evaluation regarding the impacts of a 20-percent OTSG tube plugging limit on component integrity and performance, and transient and accident analyses (References 1-3). To evaluate the impacts of 20-percent tube plugging, the NRC staff independently evaluated the licensee's submittals and responses to staff questions, including the impacts of the revised RCS DNB limits on component integrity and performance, and transient and accident analyses. The staff review verified that appropriate regulatory requirements and acceptance criteria are satisfied assuming the effects of 20-percent OTSG tube plugging. The staff's evaluation of related issues follows:

3.2.1 Fuel Component Integrity

NUREG-0800, "Standard Review Plan," (SRP) Section 4.2 (Reference 9) provides the basis for the staff's requirements regarding fuel system design. The objectives of the fuel system review are to provide assurance that:

(a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences,
(b) fuel system damage is never so severe as to prevent control rod insertion when it is required,
(c) the number of fuel rod failures is not underestimated for postulated accidents, and
(d) coolability is always maintained.

For objective (a), “not damaged” means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10, and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits.

As a result of 20-percent OTSG tube plugging, the reduction and potential asymmetry in system flow and the corresponding responses in core inlet flow can impact fuel components. The affected aspects of fuel system design are evaluated here:

Clad Corrosion

Fuel rod mechanical analyses performed for each fuel rod design include analyses for cladding fatigue, transient strain, stress, creep collapse, and corrosion. Using NRC-approved methods (Reference 10), the licensee determined that the most limiting of these is corrosion, a phenomenon sensitive to flow rate. Corrosion concerns are more important for fuel assemblies that have undergone at least one cycle of operation. In order to ensure that clad oxidation limits are not exceeded, the licensee performs cycle-specific checks of clad oxidation levels as part of Framatome’s standard reload licensing analysis activities, following NRC-approved methods outlined in Reference 10. This analysis is performed using conservative values for core flow rate and core power, which bound the 20-percent OTSG tube plugging values (Reference 3, question 3). The licensee performs these analyses early in the reload design process, thus ensuring that clad oxidation limits will not be exceeded. Based on acceptable analyses, the staff concludes that the clad corrosion performance will remain acceptable for CR-3.

Fuel Temperature, Rod Internal Pressure, and Clad Lift-Off

The licensee evaluated the impact of reduced RCS flow on fuel-to-coolant heat transfer using the TACO3 Code (Reference 11). The evaluation demonstrated that a reduction of up to 4.5 percent in RCS flow, consistent with 20-percent OTSG tube plugging, resulted in an increase of less than 10°F in average fuel temperature. The staff finds the increased level of tube plugging to be acceptable because the fuel temperature remains below the melt temperature.

The impact on fuel rod end-of-life internal pressure was also determined using TACO3. The licensee determined that the maximum internal pressure was not limiting. However, the clad lift-off could become limiting due to higher pressure. Clad lift-off is the process wherein, with increasing burnup, the clad expands more rapidly than the fuel. The loss of pellet-clad contact presents a heat transfer challenge. Therefore, precluding clad lift-off is a fuel cycle design requirement. The licensee performs cycle-specific checks of fuel thermal performance, including inception of pellet/clad lift-off, as part of Framatome’s standard reload licensing analysis activities following methods outlined in NRC-approved topical reports (References 11-13). CR-3 Cycle 13 was the first reload cycle to consider the effects of 20-percent OTSG tube plugging. For Cycle 13, the reduced flow rate associated with the higher level of tube plugging was incorporated into all fuel performance analyses. The analysis demonstrated that all acceptance criteria are met (Reference 2, question 4c). The results show that clad lift-off would not occur for higher internal pressure. Based on the acceptable results, the staff finds that the clad lift-off analysis is acceptable for CR-3.

Guide Tube Boiling

The licensee evaluated the impacts of 20-percent OTSG tube plugging on guide tube boiling. The acceptance criterion is that there shall be no saturation in the guide tube Assembly Hold-down Springs, Guide Tubes, and Spacer Grids. To ensure that this criterion will be satisfied, the licensee credited a generic B&W Owners Group analysis (Reference 14). The generic analysis, which used design assumptions that were bounding for all B&W 177 FA-type plants, conservatively bounded the CR-3 specific design conditions. In response to RAIs regarding this analysis, the licensee confirmed the applicability of this generic analysis to CR-3 (Reference 3, questions 2 and 4). This analysis demonstrated adequate positive margin to saturation under the conservative assumptions. Therefore, the staff finds that CR-3 operation with 20-percent tube plugging is acceptable with respect to guide tube boiling.

3.2.2 Core Physics Impact

SRP Section 4.3 (Reference 9) provides the basis for the staff's requirements regarding nuclear design. The review of the nuclear design includes the fuel assemblies, control systems, and reactor core, and is carried out to confirm that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core.

This review assures the requirements of General Design Criteria (GDC) 10, 11, 12, 13, 20, 25, 26, 27, and 28 are satisfied. These criteria ensure that acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11), and power oscillation is addressed in (GDC 12). A control and monitoring system is required (GDC 13) that automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients (GDC 20). The control system is required to be designed so that a single malfunction or a single operator error will cause no violation of fuel limits (GDC 25). A reactor coolant boration system is provided, which is capable of bringing the reactor to cold shutdown conditions (GDC 26), and the control system is required to control reactivity changes during accident conditions when combined with the engineered safety features (GDC 27). Reactivity accident conditions are required to be limited so that no damage to the RCS boundary occurs (GDC 28).

Significant levels of OTSG tube plugging have the potential to affect the core power distribution and reactivity due to perturbations imposed on RCS flow rate and coolant temperature distribution. To determine the effects of 20-percent tube plugging, the licensee performed evaluations to determine appropriate modeling techniques, procedures, restrictions, and augmentation factors that might be required to accommodate 20-percent OTSG tube plugging in reload licensing analyses.

The licensee evaluated the impact of 20-percent OTSG tube plugging on nuclear design using NRC-approved methods described in Section 5 of BAW-10179P-A (Reference 15). Framatome ANP's standard nuclear design code (NEMO) used for cycle-specific reload safety evaluations (Reference 16) was used for the power distribution simulations. The reload analysis process involves cycle-specific three-dimensional core power distribution NEMO analysis (maneuvering analysis) to determine the core design's sensitivity to control and power-shaping rod positions, power level, fuel burnup, and Xenon distribution. Additionally, nuclear parameter analyses are performed to calculate reactivity coefficients, rod worths, boron requirements, and other parameters necessary to ensure that the safety analysis remains valid for the reload core. The licensee verified that the NEMO code was used within its range of applicability considering 20-percent OTSG tube plugging (Reference 2, Question 5.c).

To determine the effects of plugging on steady-state core physics, the licensee evaluated four plant configurations, simulated over a 24-month depletion cycle. The four cases were designed to provide a bounding evaluation of core power distributions and margins to power peaking limits for both symmetric and asymmetric power distributions with 20-percent OTSG tube plugging (Reference 2, question 5.e). The four cases are as follows:

(a) 0/0 no tube plugging with the nominal core flow rate (base case)
(b) 20/0 with a 5-percent global flow reduction
(c) 20/0 with no flow reduction
(d) 20/20 with the 5-percent global flow reduction

The licensee's evaluation determined that Burnup, Radial Pin Peaking (2-D) and Total (3-D) Peaking were impacted by the increased level of OTSG tube plugging.

The licensee performed assembly-specific comparisons for pin burnups in the core at various times in life. The maximum cumulative burnup increases and decreases were on the order of 50 Mwd/mtU (approximately 1.5 Effective Full Power Days) (Reference 2, question 5.b). The licensee determined that these changes are not large enough to necessitate changes to the application of burnup-dependent peaking limits in licensing analyses and that simulation of the core power distribution using quarter-core symmetry could be continued in reload cycle licensing evaluations. The staff evaluated that these potential changes in fuel rod burnup are not significant, and that current reload methods remain acceptable.

The licensee performed assembly-specific comparisons for 2-D and 3-D peaking factors in the core at various times in life. The largest increases occurred for the 20/0 asymmetric tube plugging cases and occur at or shortly after Beginning of Cycle. As the fuel depletes, the changes decrease such that they are almost insignificant at End of Cycle. The maximum 2-D peaking factor increase is 0.6 percent and the maximum 3-D peaking factor increase is 1.0 percent. The licensee will apply these peaking penalties in future reload analyses.

The NRC staff has reviewed the impacts of 20-percent OTSG tube plugging on core physics and finds them to be acceptable. NRC-approved methods were used to perform these analyses and the impacts of 20-percent tube plugging were determined to be insignificant. Therefore, the acceptance criteria for nuclear design continue to be satisfied. Future CR-3 core reload designs and analyses will continue to be performed using NRC-approved methods, thus ensuring that all acceptance criteria will continue to be satisfied.

3.2.3 Final Safety Analysis Report (FSAR) Chapter 14 Transient and Accident Analyses

The licensee evaluated the impacts of 20-percent OTSG tube plugging on the FSAR Chapter 14 transients and accidents. The NRC staff evaluation, which follows, is organized by event classification, consistent with the licensee's submittal (Reference 1).

3.2.3.1 Primary Flow Events

Primary flow events are those initiated by changes in RCS flow rate due to some equipment failure or inadvertent operation. A concern for these events is the possibility of challenging the DNB acceptance criteria. For CR-3, this class of events sets the limiting value for DNB-sensitive parameters, such as minimum RCS flow. The primary flow events for CR-3 include the Cold Water Event (idle RCP start), Four RCP and One RCP Coastdowns, and Locked Rotor Accident.

The SRP Sections 15.3.1 - 15.3.4 and Section 15.4.4 (Reference 9) provide the basis for the staff's review of these transients and accidents. The acceptance criteria for this class of events are based on meeting the requirements of GDC 10, related to the RCS being designed with adequate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences (AOOs), and GDC 15, as it relates to the RCS being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operation, including AOOs. Specific criteria necessary to meet the relevant requirements of GDC 10 and 15 for incidents of moderate frequency are:

- minimum DNBR (MDNBR) remains above the 95/95 DNBR limit, and
- pressure in the RCS should be maintained below 110 percent of the design value.

For the cold water event, which may result in an increase in core reactivity due to decreased moderator temperature, the licensee determined that tube plugging will have a slight beneficial effect on this event due to increased loop resistance and slightly smaller primary-side tube inventory. These factors would tend to slow the flow rate and limit the amount of cold water introduced into the active RCS. Therefore, tube plugging will not challenge the acceptance criteria for this event. The staff has evaluated this event and finds the licensee's conclusion acceptable.

For the four RCP and one RCP coastdown events, the licensee reanalyzed only the DNBR portions of the transient. An update to the RCS system response analysis, which provides the inputs for normalized RCS flow and power vs. time, was not required. The licensee performed sensitivity studies to confirm that the normalized profiles for these parameters remain valid as a result of 20-percent tube plugging (Reference 3, question 5). The licensee performed the DNB analyses in accordance with the approved methods outlined in BAW-10179P-A (Reference 15), and used the LYNXT computer code (Reference 17). The licensee also provided verification that all restrictions and limitations of the approved computer codes and methods were satisfied. The analysis results demonstrate that the DNBR remains above the thermal design limit throughout the event. The current statistically based thermal design limit for CR-3 is 1.40 for the BWC correlation. The four RCP coastdown MDNBR was approximately 1.75, the one RCP coastdown MDNBR was approximately 1.95, and the locked rotor MDNBR value was approximately 1.75. The licensee reanalyzed the RCS system response analysis for the locked rotor accident, which is the limiting event for this class, and demonstrated that the RCS pressure remains below the acceptance criteria.

The staff has reviewed the licensee's primary flow events considering 20-percent OTSG tube plugging and finds them acceptable.

3.2.3.2 Overheating Events

Overheating events occur due to a postulated mismatch in power production and rejection or removal. Reductions in RCS inventory and primary-to-secondary heat transfer area affect the outcome of these events. The acceptance criteria for these events involve limiting RCS or secondary side pressure, as well as DNBR criteria. RCS flow and resistance have a small impact on the ability to meet the acceptance criteria for this class of events. The licensee did not reanalyze all events within this category, but did evaluate the impacts of 20-percent OTSG tube plugging on all events.

The startup event involves a control rod assembly bank withdrawal from subcritical or low-power conditions. The positive reactivity insertion causes power, temperature, and pressure excursions. This event is the limiting primary overpressure event for the B&W 177-FA plant design. The licensing basis acceptance criteria for this event for CR-3, as discussed in the FSAR, are that the peak RCS pressure shall remain below 110 percent of the design pressure of the RCS (2500psig x 1.10 = 2750psig) and that the reactor thermal power shall remain below 112 percent of rated power. By maintaining thermal power below the 112-percent limit, the core is assured to remain below DNB and fuel centerline melt (kW/ft) limits. These acceptance criteria are consistent with NUREG-0800, “Standard Review Plan” (Reference 9), and are in place to ensure the requirements of GDC 10 are satisfied.

In order to support 20-percent OTSG tube plugging, the licensee reanalyzed the startup event using the RELAP5/MOD2-B&W computer code (References 18 and 19). The licensee stated that all restrictions and limitations of the approved topical reports were satisfied (Reference 2, question 6). The analysis conservatively assumed up to 30-percent steam generator tube plugging and a pressurizer safety valve lift tolerance of ± 3 percent. The results of the reanalysis showed a peak RCS pressure of 2479.6 psia and a peak thermal power of 72.52 percent. The staff finds that the proposed level of 20-percent OTSG tube plugging is acceptable for this transient because the licensee used approved analytical methods, conservatively assumed 30-percent OTSG tube plugging, and the analysis demonstrated that the acceptance criteria for this event are satisfied.

Other reactivity-related overheating events that the licensee evaluated, but did not reanalyze, include rod withdrawal at rated power, moderator dilution event and the dropped rod event. For these events, the licensee concluded that the reactivity insertion and subsequent effects are bounded by the startup event. The acceptance criteria for these events are the same as for the startup event. The staff has reviewed the licensee’s current analyses of record for these events as described in the CR-3 FSAR and finds the acceptance criteria for these events will continue to be satisfied.

The licensee evaluated the impacts of 20-percent tube plugging on the control rod ejection accident. The acceptance criteria for this event are that peak RCS pressure remain below 110 percent of RCS design pressure, and that the maximum fuel enthalpy remain below 280 cal./gm. The licensee concluded that the acceptance criteria for this event will continue to be met because the duration of the transient is so short that the primary-to-secondary heat transfer has no significant role in the sequence of events. The staff finds the licensee’s assessment that the impacts of the increased levels of OTSG tube plugging on RCS operating pressure and temperature are so small that the acceptance criteria continue to be satisfied, to be acceptable.

The licensee evaluated the impacts of 20-percent tube plugging on the loss of electric load / turbine trip event. This event occurs when the external transmission system deteriorates significantly, causing the unit to automatically disconnect from the transmission grid. This is the limiting event for peak secondary pressure, and the acceptance criterion is that the secondary system pressure remain below 110 percent of rated pressure. The licensee concluded that considering the reduction in primary-to-secondary heat transfer due to increased OTSG tube plugging, the secondary side pressurization rate will actually be reduced by increased tube plugging. Therefore, the acceptance criteria for this event will continue to be satisfied. The staff finds this acceptable.

The licensee's current analysis of record for the loss of feedwater event, as described in the CR-3 FSAR, was performed assuming 20-percent OTSG tube plugging. The acceptance criteria for this event for CR-3 are that the peak RCS pressure shall remain below 110 percent of the design pressure of the RCS and that the reactor thermal power shall remain below 112 percent of rated power. Additionally, at no time is liquid to be passed through the pressurizer safety or relief valves. The licensee analyzed this event using the RELAP5/MOD2-B&W computer code (Reference 22). The licensee's analysis found that all acceptance criteria are satisfied with 20-percent OTSG tube plugging. The staff has reviewed the event as described in the CR-3 FSAR and finds it acceptable.

The licensee evaluated the feedwater line break accident and concluded that the existing analysis of record remains valid for an increased level of 20-percent tube plugging. The licensee's conclusion is based on conservative OTSG blowdown and boiloff assumptions in the existing analysis (Reference 3, question 8). The staff finds the licensee's conclusions acceptable.

The licensee evaluated the loss of all ac power event considering the effects of 20-percent OTSG tube plugging and concluded that the RCS pressure consequences are bounded by the loss of main feedwater flow event and that the DNBR consequences are bounded by the four RCP coastdown event. With respect to RCS pressure, the net energy addition to the primary coolant during this transient is less than that during a loss of feedwater flow event because the RCPs trip immediately upon a loss of power (Reference 2, question 6). With respect to DNBR, the reactor trip occurs slightly faster in the loss of ac power event as compared with the four RCP coastdown event, resulting in a higher MDNBR value (Reference 3, question 7). The staff has reviewed the licensee's submittals and finds the licensee's conclusions acceptable.

The licensee credited the B&W Owners Group project, FRA ANP Document 51-5009660-01, "Evaluation of Extended Tube Plugging Limits for Once-Through Steam Generators" (Reference 14), for its Station Blackout (SBO) evaluation. This generic analysis assumed a conservative power level of 2772 MWt as compared to the CR-3 rated power level of 2544 MWt. This analysis concluded that extended OTSG tube plugging will not adversely affect the required 4-hour coping duration. The licensee confirmed the applicability of this generic analysis to CR-3 and that NRC-approved methods were applied (Reference 3, question 2). Based on the results of this analysis, the staff finds that CR-3 will continue to meet the 4-hour coping duration for an SBO with the increased levels of OTSG tube plugging.

The licensee credited the same generic analysis (Reference 14) for the anticipated transient without scram (ATWS) evaluation. For the B&W-designed PWRs, the loss of feedwater represented the most severe ATWS transient. To meet the requirements of 10 CFR 50.62, CR-3 has installed a diverse scram system (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC). These systems provide independent reactor trip, turbine trip, and emergency feed water actuation signals. As such, this analysis is similar to the loss of feedwater event discussed previously, which was performed assuming 20-percent OTSG tube plugging. The staff finds that the 20-percent tube plugging does not impact the ATWS analysis of record for CR-3 because the licensee meets the requirements of 10 CFR 50.62, and the loss of feedwater analysis assumed 20-percent OTSG tube plugging.

3.2.3.3 Overcooling Events

The limiting event in this category is the main steam line break (MSLB). This event results in a rapid depressurization and loss of secondary side inventory. The increased steam flow results in a cooldown and depressurization of the RCS, which coupled with a negative moderator temperature coefficient, can cause positive reactivity addition and a subsequent return to power after the reactor has tripped. The acceptance criteria for this event require that the core shall remain intact for effective core cooling, the resultant doses do not exceed 10 CFR Part 100 limits, and OTSG tube failure due to pressure and temperature gradients shall not occur.

The licensee evaluated this event considering 20-percent OTSG tube plugging and concluded that the current analysis of record is bounding for the MSLB event. Relative to increased OTSG tube plugging, the key influence on the outcome of the MSLB is the initial OTSG secondary side inventory. A larger initial inventory will result in the greatest overcooling of the RCS. The current CR-3 analysis of record, as described in FSAR Section 14.2.2.1, assumes a conservatively high initial OTSG secondary side inventory, which bounds the 20-percent tube plugging inventory. Additionally, the current MSLB analysis assumes that no OTSG tubes are plugged. This assumption maximizes the cooldown because it maximizes the heat transfer area between the primary and secondary side. Because a bounding secondary side inventory is modeled in the analysis and the full primary-to-secondary side heat transfer area is modeled, the overcooling consequences are more severe than if OTSG tube plugging was modeled. The staff finds the licensee’s assessment that the current analysis of record remains bounding for 20-percent OTSG tube plugging acceptable.

Other events in this category include the OTSG overfeed and the excessive load increase. The licensee stated in its submittal (Reference 1) that these events are not part of CR-3 FSAR Chapter 14 analyses, but included an assessment for completeness. The licensee concluded that these events are affected in the same way as the MSLB event, and are bounded by the consequences of the MSLB event. Therefore, any limitations on OTSG tube plugging with respect to overcooling events will be determined by the MSLB event.

3.2.3.4 Loss of Primary Coolant Events

This class of events involves postulated failures of the RCS pressure boundary. For this class of events, the licensee reanalyzed the large- and small-break loss-of-coolant accidents (LOCA) assuming 20-percent OTSG tube plugging. Additionally, the licensee evaluated the letdown line failure and steam generator tube rupture events, and concluded that these two events are not significantly impacted by increased levels of OTSG tube plugging.

CR-3's large- and small-break LOCA analyses include 20-percent OTSG tube plugging. The acceptance criteria for these events are specified in 10 CFR 50.46, and include limits on peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, maintaining a coolable core geometry, and ensuring long-term core cooling. The LOCA analyses were performed in accordance with the NRC-approved BWNT LOCA Evaluation Model, BAW-10192P-A (Reference 20), and the results of the analyses were reported to the NRC in the Reference 21 letter and are already incorporated into the CR-3 FSAR. In the Reference 21 letter, the licensee discussed how all restrictions and limitations of the approved methods are satisfied for these analyses. Additionally, the licensee showed that the BWNT LOCA Evaluation Model applies to CR-3 by confirming that CR-3 and its vendor have ongoing processes that ensure LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters (Reference 3, question 10). The LOCA analyses were performed assuming a limiting asymmetric OTSG tube plugging of 25 percent in the broken loop OTSG and 15 percent in the intact OTSG, which is the limiting case (Reference 2, questions 6 and 9a). Additionally, the small-break LOCA analysis was performed assuming that no more than 75 percent of the tubes in the wetted region of each OTSG were plugged. The results of these large- and small-break LOCA analyses, as shown in the CR-3 FSAR and in Reference 21, show that all acceptance criteria of 10 CFR 50.46 are satisfied. Based on this discussion, the staff finds that 20-percent OTSG tube plugging is acceptable with respect to the LOCA analyses and the acceptance criteria of 10 CFR 50.46.

3.2.4 Impact on Safety Limits, Reactor Protective System, and Engineered Safeguards

3.2.4.1 Impact on DNB Safety Limits

CR-3 ITS Figure 2.1.1-1, “Reactor Coolant System Departure from Nucleate Boiling Safety Limits,” identifies the regions of pressure and temperature that ensure protection against the DNB phenomenon. The licensee has determined that the existing curve is conservative and has chosen to retain the existing curve in the ITS. The existing CR-3 limits curve is based on deterministic methods, not on Statistical Core Design (SCD) methods. As such, the existing curve is more conservative than SCD-based limits calculated with and without 20-percent OTSG tube plugging considerations. The SCD-based DNB safety limits are generated with the LYNXT computer code (Reference 17), using methods described in Section 6.4 of the NRC-approved Topical Report BAW-10179P-A (Reference 15). CR-3 has fully converted to the SCD analysis method and has analyzed all DNB limited transients using SCD (Reference 2, question 11). As such, the existing (deterministic based) DNB safety limit curve is conservative and more restrictive than it needs to be. Because maintaining the existing DNB safety limits curve in the ITS is conservative, the staff finds this to be acceptable.

3.2.4.2 Impact on Reactor Protective System and Engineered Safeguards

The licensee evaluated the impact of 20-percent OTSG tube plugging on CR-3's Reactor Protective System (RPS) and Engineered Safeguards (ES) setpoints. The functions were evaluated based on the acceptability CR-3 FSAR Chapter 14 transients and accidents. The licensee determined that because all acceptance criteria for transients and accidents remain acceptable, no RPS or ES setpoint changes are required. Based on the staff’s evaluation of transients and accidents, discussed in Section 3.2.3 of this SE, the staff finds the licensee’s conclusion that there are no RPS or ES ITS setpoint changes required, to be acceptable.

3.3 OTSG FLOW-INDUCED VIBRATION EVALUATION

In its July 24, 2001, submittal, FPC concluded, in Attachment F, Section D.4, “Secondary System Performance and Integrity,” that the FIV of the OTSG tubes will not be significantly affected by a symmetric tube plugging distribution because the tube plugging would result in an insignificant increase in dynamic pressure. For an asymmetric plugging distribution situation, FPC stated that the refinement of FIV analyses could allow for increased feedwater flow under limited power operations. However, the staff noted that it is not clear what effect increased FIV would have, under the asymmetric plugging distribution and full-power operations, on steam generator tube integrity. The staff noted also that potential FIV of OTSG tubes could be subjected to 2-phase flow, vortex shedding, fluid-elastic instability, and turbulence-induced vibration. Therefore, the staff requested in the RAI on FIV analysis that FPC provide the results of FIV reassessment calculations or analyses to demonstrate the functional integrity of the steam generator tubes due to increase in steam generator tube plugging up to 20 percent under full-power operations for symmetric and worst-case asymmetric plugging distributions. In response to the staff’s RAIs, FPC stated that the analysis performed for the 20-percent tube plugging did not assume an increase in feedwater flow from that previously analyzed. FPC also indicated that no other input parameters to the flow-induced vibration model were changed. Therefore, there was no increase in the calculated flow-induced vibration as part of this effort. FPC further stated that if 20 percent of steam generator tubes were actually plugged, CR-3 may not be able to maintain 100-percent electrical power output with the current feedwater flow. At that time, feedwater flow could be increased to attempt to restore 100-percent power capability. The increased feedwater flow would change input parameters to the analysis and a revision to the FIV analysis would be required. FPC also indicated in the July 1, 2002, letter that plant operating procedures already impose a limit on total feedwater flow, to be consistent with the design limit. The staff finds that there is no need to review CR-3's FIV analysis until increased feedwater flow exceeds the licensed feedwater flow limit.

Although Attachments B and D to the licensee’s June 5, 2002, letter are not the subject of this review, the staff noticed that some parameters used in the FIV analysis of the OTSG tubes may not be appropriate and may require detailed justification in future reviews. In particular, if the high damping ratios (> 3 percent) of the steam generator tubes continue to be used in any future analysis, the staff will require detailed justification and review of the test data.

Based on its review of the amendment request and the followup responses to the staff’s RAIs, the staff finds that increased tube plugging up to 20 percent for CR-3 steam generators, with respect to FIV of OTSG tubes, is acceptable considering the licensed limit placed on the total feedwater flow in plant operating procedures. The staff also finds that CR-3 can continue to operate under previously analyzed condition with up to 20-percent tube plugging without exceeding the currently licensed feedwater flow, although it may not be able to maintain 100-percent electrical power output. However, if increased feedwater flow exceeding the licensed limit is needed for any reason with up to 20-percent tube plugging, the licensee should provide the FIV analysis with sufficient details for staff review to demonstrate the functional integrity of the steam generator tubes for up to 20-percent tube plugging under symmetric and worst-case asymmetric plugging distributions.

3.4 TECHNICAL EVALUATION CONCLUSION

The staff has reviewed the proposed changes, which relocate CR-3 ITS SR 3.4.1.1, SR 3.4.1.2, SR 3.4.1.3 and Table 3.3.1-1 Function 5 (VLPT) to the COLR and revise the parameter limits of SR 3.4.1.1, SR 3.4.1.2, and SR 3.4.1.3 based on 20-percent OTSG tube plugging, and finds these changes to be 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 and changes surveillance requirements. 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 (66 FR 44173). 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.

7.0 REFERENCES

1. Letter from D. E. Young, Florida Power Corporation to USNRC, "Crystal River Unit 3, License Amendment Request #263, Revision 0 - Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20 percent Steam Generator Tube Plugging," Docket No. 50-302, License No. DPR-72, July 24, 2001.

2. Letter from D. E. Young, Florida Power Corporation to USNRC, "Crystal River Unit 3, Response to Request for Additional Information, LAR #263, Revision 0, Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20 percent Steam Generator Tube Plugging," Docket No. 50-302, License No. DPR-72, June 5, 2002.

3. Letter from D. E. Young, Florida Power Corporation to USNRC, “Crystal River Unit 3, Response to Request for Additional Information, LAR #263, Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20 percent Steam Generator Tube Plugging,” Docket No. 50-302, License No. DPR-72, July 1, 2002.

4. United States Nuclear Regulatory Commission Generic Letter 88-16, “Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits,” October 3, 1988.

5. Letter from T. Alexion, USNRC, to J. Yelverton, Entergy, “Issuance of Amendment No. 186 to Facility Operating License No. DPR-51 Arkansas Nuclear One, Unit No. 1 (TAC No. M95368),” dated October 3, 1996.

6. Letter from G. Dick, USNRC, to O. Kingsley, Commonwealth Edison Company, “Byron and Braidwood - Issuance of Amendments Regarding Relocation of Reactor Coolant System Limits (TAC Nos. MA7809, MA7810, MA7807 and MA7808),” dated May 15, 2000.

7. Letter from T. Colburn, USNRC, to J. Langenbach, GPU Nuclear, “Three Mile Island - Issuance of Amendment RE: Technical Specification Change Request No. 279 - Core Protection Safety Limit with an average of 20 percent Steam Generator Tubes Plugged (TAC No. MA4301),” dated August 19, 1999.

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15. BAW-10179P-A, “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.”

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17. BAW-10156A, Revision 1, “LYNXT - Core Thermal Hydraulic Program,” B&W Fuel Company, March 1993.

18. BAW-10164P-A, “RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis,” Revision 3, dated October 1992.

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