Florida Power
A **Progress Energy** Company

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

Ref: 10 CFR 50.90

July 24, 2001
3F0701-11

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

Subject: Crystal River Unit 3, License Amendment Request #263, Revision 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Dear Sir:

Florida Power Corporation (FPC) hereby submits License Amendment Request (LAR) #263, Revision 0, requesting a change to the Crystal River Unit 3 (CR-3) Facility Operating License No. DPR-72 in accordance with 10 CFR 50.90. LAR #263 revises Improved Technical Specifications (ITS) Table 3.3.1-1, 3.4.1 and 5.6.2.18.

The changes proposed in this LAR accommodate future changes in plant design, including increased levels of Once-Through Steam Generator (OTSG) tube plugging. The changes are categorized into two sets. The first set of changes facilitate the direct relocation of parameters from the ITS to the cycle-specific Core Operating Limits Report (COLR). These parameters are the Variable Low Pressure Trip (VLPT) 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% 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 (RCP) operation.

The RCS limits associated with 20% plugging will be maintained in the ITS, however, cycle-specific values for these limits will be relocated to the COLR. As such, 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," are revised to reflect their location in the COLR.

For both sets of changes, FPC is proposing to modify ITS 5.6.2.18(a) to reflect the relocation of cycle-specific values from the ITS to the COLR.

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FPC requests approval of LAR #263 by March 31, 2002 with a 60-day implementation period. This implementation period is needed in order to revise and implement the COLR. The revised COLR will be submitted per ITS 5.6.2.18.d. This requested approval schedule is contingent upon approval of LAR #252, Once-Through Steam Generator Tube Surveillance Program, Tube Repair Roll (Re-Roll) Process, submitted March 21, 2001. If LAR #252 is not approved prior to the fall 2001 outage, LAR #263 will be needed prior to plant startup, currently scheduled for October 2001.

The CR-3 Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

This submittal establishes no new regulatory commitments.

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

Sincerely,

Dale E. Young

Dale E. Young
Vice President, Crystal River Nuclear Plant

DEY/pei

Attachments:

A. Description of Proposed Changes, Background, Reason for Request, and Evaluation of Request
B. No Significant Hazards Consideration Determination
C. Environmental Impact Evaluation
D. Proposed Revised Improved Technical Specifications and Bases Change Pages - Strikeout / Shadow Format
E. Proposed Revised Improved Technical Specifications and Bases Change Pages - Revision Bar Format
F. Detailed Evaluation
G. Draft Revised COLR Pages

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

STATE OF FLORIDA

COUNTY OF CITRUS

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

Dale E Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 24th day of July, 2001, by Dale E. Young.

Lisa A Morris

Signature of Notary Public
State of Florida

LISA A. MORRIS
Notary Public, State of Florida
My Comm. Exp. Oct. 25, 2003
Comm. No. CC 879691

LISA A MORRIS

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

Personally Known X -OR- Produced Identification _____

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Description of Proposed Changes, Background, Reason for Request, and Evaluation of Request

Description of Proposed Changes

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 Core Operating Limits Report (COLR). These parameters are the Variable Low Pressure Trip (VLPT) 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 Once-Through Steam Generator (OTSG) tube plugging equivalent to up to 20% of all tubes, and addresses its impact. These changes include revision to the hot leg maximum temperature limit, and the RCS minimum flow limits for four- and three- Reactor Coolant Pump (RCP) operation.

While the limits for RCS parameters associated with 20% OTSG plugging will be maintained in the ITS, cycle-specific values for these limits will be located in the COLR. As such, 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," are revised to reflect their location in the COLR. In addition, ITS 5.6.2.18(a) is revised to reflect the relocation of cycle-specific values from the ITS to the COLR.

No changes are being proposed to the existing pressure-temperature Departure from Nucleate Boiling (DNB) Safety Limits in ITS Figure 2.1.1-1. The existing limits are based on a previous deterministic methodology, not on the Statistical Core Design (SCD) methods. Recalculation of these limits using SCD shows that the original methodology contains significant conservatism. Therefore, the existing limits remain bounding and applicable. Florida Power Corporation has no need to reduce this conservatism and therefore the safety limits are not being changed.

The following ITS and ITS Bases are affected by this proposed change:

ITS Table 3.3.1-1 Reactor Protection System Instrumentation, Function 5, RCS Variable Low Pressure (equation).

ITS SR 3.4.1.1 RCS Pressure, Temperature, and Flow DNB Limits (RCS Pressure)

ITS SR 3.4.1.2 RCS Pressure, Temperature, and Flow DNB Limits (RCS Temperature)

ITS SR 3.4.1.3 RCS Pressure, Temperature, and Flow DNB Limits (RCS Flow)

ITS Bases 3.4.1 RCS Pressure, Temperature, and Flow DNB Limits

ITS 5.6.2.18(a) Core Operating Limits Report (COLR)

The proposed change would revise the text of ITS Table 3.3.1-11 to read as follows:

Function	Allowable Value
5. RCS Variable Low Pressure	**RCS Variable Low Pressure equation in COLR**

The change also revises the text of ITS 3.4.1.1, 3.4.1.2 and 3.4.1.3 to read as follows:

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 ≥ 133.5 E6 lb/hr with four RCPs operating or ≥ 99.7 E6 lb/hr with three RCPs operating.**

Draft revised COLR pages, including the relocated values, are provided in Attachment G. The ITS Bases associated with these RCS parameters are revised to reflect the relocated limit values and revision of ITS 5.6.2.18(a), Core Operating Limits Report. The ITS Bases pages are included in Attachments D and E.

Background

Crystal River Unit 3 (CR-3) has two OTSGs each containing 15,531 tubes. Currently, OTSG "A" has approximately 1.5% and "B" has approximately 4.7% equivalent plugging (combined plugging and sleeving). The CR-3 limiting analysis can currently accommodate up to 7% equivalent plugging in each generator. In anticipation of further tube plugging, CR-3 has revised the analyses to account for up to 20% equivalent tube plugging. The resulting DNB parameters are acceptable but are overly restrictive while actual plugging values are much lower than the 20% limit.

The guidance in Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," concludes that it is essential to safety that the plant is operated within the bounds of cycle-specific parameter limits, and that a requirement to maintain the plant with these limits must be retained in the Technical Specifications. However, the GL concludes that specific values of these limits may be modified by the licensees, without affecting nuclear safety, provided that these changes are determined using an approved methodology and are consistent with all applicable limits of the plant safety analysis. In this proposed change, the requirements for meeting and monitoring these parameters and the surveillance frequency for monitoring these parameters are retained in ITS. Only the cycle-dependent value of these parameters, which may change due to modification activities,

are being relocated to the COLR. As part of the COLR, they will be properly evaluated during the reload analysis process.

Reason For Request

Florida Power Corporation requests that the ITS be amended to include DNB parameters that result from a incorporation of 20% OTSG plugging. This change will permit OTSG plugging greater than the current analyzed limit of 7%. The second group of changes permits the cycle specific values for the DNB parameters to be located in the COLR. These changes allow evaluation and operation of CR-3 considering current OTSG plugging levels using the limits in the COLR. These limits will ensure conservative operation of CR-3 without imposing the restrictions as if 20% OTSG tube plugging had already occurred.

The relocation of the RCS VLPT function and minimum RCS loop pressure from the ITS to the COLR is being proposed to ensure that cycle-dependent variations of these parameters due to plant modification activities continue to be properly evaluated during the reload analysis process. While plugging an equivalent of 20% of tubes would 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.

Similarly, the relocation of RCS coolant hot leg temperature and RCS total flow cycle-specific limit values from the ITS to the COLR is being proposed. 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 limits set by plugging up to an equivalent of 20% of the OTSG tubes will remain in ITS. Therefore, cycle-specific limits can never exceed these values.

Evaluation of Request

This section summarizes the justification for the acceptability of the proposed changes. The detailed evaluation of the proposed changes is included in Attachment F. The detailed evaluation provides general information on the current state of CR-3's tube plugging along with the methods used, impacts on component integrity and performance, safety analyses, and technical specifications. The evaluation demonstrates that while plugging does result in a reduction of the minimum RCS flow limit and small increases in RCS pressure and the maximum hot leg temperature limits, analyses and evaluations have shown that these changes are not adverse to the health and safety of the public.

Plugging OTSG tubes affects several design and performance characteristics of the plant. The most significant effects are reductions in RCS flow, RCS volume, and the primary to secondary heat transfer rate and a slight increase in RCS pressure. The revised values for these RCS parameters are as follows: a minimum RCS flow of at least 133.5 E6 lb/hr with four RCPs running, at least 99.7 E6 lb/hr with three RCPs running, a maximum hot leg

temperature of 605.8°F, and a minimum RCS pressure of 2064 psig. These effects were evaluated and found to be acceptable in terms of steady-state performance, as well as in the transient analyses. Further, the protective features of the plant were found to be adequate for accommodating this level of tube plugging. Moreover, while the cycle-specific values will reside in the COLR, the revised RCS flow and temperature limits discussed above will be retained in the ITS. Therefore, it is concluded that the proposed changes to accommodate an equivalent tube plugging of 20% are acceptable for CR-3.

The relocation of the parameters to the COLR does not alter the methodologies used for any parameter limit calculation. Additionally, the relocation does not conflict with the plant safety analysis and does not change the analytical methods described in ITS 5.6.2.18(b). The relocated portions of the proposed ITS change will be included in the COLR and submitted to the NRC in accordance with ITS 5.6.2.18(d). Changes to the COLR, including those resulting from the proposed change, are made in accordance with 10 CFR 50.59. Since changes to the COLR are made under the provisions of 10 CFR 50.59, auditable and appropriate control over the relocated limit values is assured.

Precedents

Precedent to the request of approval for new limits to accommodate 20% tube plugging has been established by the issuance of a license amendment for Three Mile Island-1 (TMI-1) (Attachment F, Reference 23). As a result of the license amendment, the TMI-1 Technical Specifications and COLR reflect limits associated with 20% tube plugging.

Precedent to the request of parameter transfer to the COLR has been established by the NRC issuance of license amendments for Arkansas Nuclear One, Unit One (ANO-1, Unit 1) (Attachment F, References 17 to 20). As a result of these license amendments, the ANO-1, Unit 1 COLR contains the relocated parameters.

The amendments cited above are similar to the proposed changes contained in this submittal. CR-3 is similar in design and analysis to both TMI-1 and ANO-1 and, therefore, these precedents are applicable to the proposed changes.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT B

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

No Significant Hazards Consideration Determination

No Significant Hazards Consideration Determination

Florida Power Corporation (FPC) has reviewed the proposed revisions to Improved Technical Specifications (ITS) Table 3.3.1-1, 3.4.1 and 5.6.2.18 against the requirements of 10 CFR 50.92(c). The proposed changes do not involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

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

The proposed change relocates several Reactor Coolant System (RCS) parameters from the ITS to the Core Operating Limits Report (COLR). The purpose for this relocation is to permit the values of these parameters to be changed under the 10 CFR 50.59 change process for cycle-specific analyses. In addition, these changes will allow increased Once-Through Steam Generator (OTSG) tube plugging. The increased plugging limit is in accordance with the analysis and will support continued proper maintenance of the OTSGs. The increased OTSG plugging will result in a small decrease in RCS flow and primary to secondary heat transfer. The difference in heat transfer results in small changes to primary and secondary operational parameters but will not result in any challenges to plant equipment. The change in RCS parameters will have no impact on the probability of accident initiators or precursors. Increased OTSG plugging will slightly reduce mass release to the containment following some loss of primary coolant accidents. Previously analyzed accidents were reevaluated considering the proposed changes and were found to be within established limits. Therefore, the change will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not introduce any new operating methods or configurations. The revised RCS parameters have been analyzed and have been determined to be within established limits. No new failure modes or limiting single failures were identified. All safety and design criteria continue to be met. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes affect RCS parameters, which are inputs to the plant's safety limits. The changes have been evaluated and the resultant plant analysis and configuration remain within the existing safety limits. The safety limits themselves are not being altered. The accident analysis was reevaluated and it has been determined that there is no significant impact on the fuel cladding, reactor coolant system or the containment structure. Therefore, this change does not involve a significant reduction in the margin of safety.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT C

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Environmental Impact Evaluation

Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this license amendment request and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the no significant hazards evaluation for this License Amendment Request.

2. The proposed changes relocate Reactor Coolant System Parameters from the Improved Technical Specifications to the Core Operating Limits Report. The change will permit increased steam generator tube plugging and will not impact generation or processing of radioactive fluids. Therefore, the proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site.

3. The proposed change will permit increased steam generator tube plugging. The change will not impact how steam generator tube plugging will be performed or affect how many tubes are actually plugged. Steam generator tube plugging activities will continue to be performed per the plant radiation protection procedures and in accordance with the ALARA program. The proposed changes do not require operator or other actions that could increase occupational radiation exposure. Therefore, the proposed license amendment will not result in a significant increase to the individual or cumulative occupational radiation exposure.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT D

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Proposed Revised Improved Technical Specifications and Bases Change Pages

Strikeout / Shadow Format

~~Strikeout Text~~	Indicates deleted text
Shadowed text	Indicates added text

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower -				
a. High Setpoint	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.7	≤ 104.9% RTP
b. Low Setpoint	2[b],3[b] 4[b],5[b]	G	SR 3.3.1.1 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 618°F
3. RCS High Pressure	1,2	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig
4. RCS Low Pressure	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≥ 1900 psig
5. RCS Variable Low Pressure	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	~~≥ (11.59 *T$_{hot}$ - 5037.8) psig~~ RCS Variable Low Pressure equation in COLR
6. Reactor Building High Pressure	1,2,3[c]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 4 psig
7. Reactor Coolant Pump Power Monitor (RCPPM)	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	More than one pump drawing ≤ 1152 or ≥ 14,400 kW
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope in COLR
9. Main Turbine Trip (Control Oil Pressure)	≥ 45% RTP	H	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 45 psig
10. Loss of Both Main Feedwater Pumps (Control Oil Pressure)	≥ 20% RTP	I	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 55 psig
11. Shutdown Bypass RCS High Pressure	2[b],3[b] 4[b],5[b]	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 1820 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD Control System (CRDCS) capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRDCS capable of rod withdrawal.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- With three RCPs operating, the limit is applied to the loop with two RCPs in operation. ---------- Verify RCS loop pressure ~~≥ 2061.6 psig with four RCPs operating or ≥ 2057.2 psig with three RCPs operating.~~meets the RCS loop pressure limits specified in the COLR.	12 hours
SR 3.4.1.2	----------NOTE---------- With three RCPs operating, the limit is applied to the loop with two RCPs in operation. ---------- Verify RCS hot leg temperature ~~≤ 604.6°F.~~ meets the RCS hot leg temperature limit specified in the COLR, AND is ≤ 605.8°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate ~~≥ 139.7 E6 lb/hr with four RCPs operating or ≥ 104.4 E6 lb/hr with three RCPs operating.~~ meets the RCS total flow rate limits specified in the COLR, AND is ≥133.5 E6 lb/hr with four RCPs operating or ≥ 99.7 E6 lb/hr with three RCPs operating.	12 hours
SR 3.4.1.4	----------NOTE---------- Only required to be performed when stable thermal conditions are established > 90% of ALLOWABLE THERMAL POWER. ---------- Verify RCS total flow rate is within limit by measurement.	24 months

5.6 Procedures, Programs and Manuals

5.6.2.18 COLR (continued)

LCO 3.2.3 AXIAL POWER IMBALANCE Operating Limits
LCO 3.2.4 QUADRANT POWER TILT
LCO 3.2.5 Power Peaking Factors
LCO 3.3.1 Reactor Protection System (RPS) Instrumentation
SR 3.4.1.1 Reactor Coolant System Pressure DNB Limits
SR 3.4.1.2 Reactor Coolant System Temperature DNB Limits
SR 3.4.1.3 Reactor Coolant System Flow DNB Limits
LCO 3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC:

BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed) and License Amendment 144, SER dated June 25, 1992. The approved revision number for BAW-10179P-A shall be identified in the COLR.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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

a. Other Applicable ITS:

3.4.3 RCS P/T Limits
3.4.11 Low Temperature Overpressure Protection

b. RCS pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. The analytical methods used to determine the pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and approved by the NRC in BAW-10046A, Rev. 2, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," June 1986. The analytical method used to determine vessel fluence shall be those reviewed by the NRC and documented in BAW-2241P, May 1997. The analytical method used to determine LTOP limits shall be those previously reviewed by the NRC based on ASME Code Case N-514. The Materials Program is in accordance with BAW-1543A, "Integrated Reactor Vessel Surveillance Program."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

that transients initiated from the limits of this LCO will meet the event-specific DNBR acceptance criterion. This is the acceptance limit for the RCS DNBR parameters. The transients analyzed for include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "APSR Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE OPERATING LIMITS," and LCO 3.2.4, "QUADRANT POWER TILT."

~~The core outlet pressure assumed in the safety analyses is 2135 psia. The minimum pressure specified in LCO 3.4.1 is the limit value in the reactor coolant loop as measured at the hot leg pressure tap.~~

~~The safety analyses are performed with an assumed RCS coolant average temperature of 581°F (579°F plus 2°F allowance for calculational uncertainty). The corresponding hot leg temperature of 604.6°F is calculated by assuming an RCS core outlet pressure of 2135 psia and an RCS flow rate of 374,880 gpm. The maximum temperature specified is the limit value at the hot leg resistance temperature detector.~~

~~The safety analyses are performed with an assumed RCS flow rate of ≤ 374,880 gpm. The flow rate specified in SR 3.4.1.3 is the corresponding mass flow rate for full-power conditions.~~

For full power conditions, limits are set on loop parameters to meet the DNBR criterion. The minimum core outlet pressure is the limit value in the reactor coolant loop as measured at the hot leg pressure tap. The maximum hot leg temperature specified is the limit value as measured at the hot leg resistance temperature detector. The minimum volumetric flow rate corresponds to the mass flow rate for four RCP operation at full power.

Additionally, a~~A~~nalyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump operation~~ as well~~. The flow limits for three pump operation are approximately 25% lower than the four pump limits. To meet the DNBR criterion, a corresponding maximum power limit of less than the top of the "doghouse" (Power-Imbalance-Flow trip setpoint contained in the COLR) is required in combination with the 3 pump limits (see Bases for LCO 3.3.1, "RPS Instrumentation").

RCS DNB limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

Limits on RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate ensure that the core

(continued)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT E

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Proposed Revised Improved Technical Specifications and Bases Change Pages

Revision Bar Format

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower -				
a. High Setpoint	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.7	≤ 104.9% RTP
b. Low Setpoint	2[b],3[b] 4[b],5[b]	G	SR 3.3.1.1 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 618°F
3. RCS High Pressure	1,2	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig
4. RCS Low Pressure	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≥ 1900 psig
5. RCS Variable Low Pressure	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	RCS Variable Low Pressure equation in COLR
6. Reactor Building High Pressure	1,2,3[c]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 4 psig
7. Reactor Coolant Pump Power Monitor (RCPPM)	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	More than one pump drawing ≤ 1152 or ≥ 14,400 kW
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope in COLR
9. Main Turbine Trip (Control Oil Pressure)	≥ 45% RTP	H	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 45 psig
10. Loss of Both Main Feedwater Pumps (Control Oil Pressure)	≥ 20% RTP	I	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 55 psig
11. Shutdown Bypass RCS High Pressure	2[b],3[b] 4[b],5[b]	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 1820 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD Control System (CRDCS) capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRDCS capable of rod withdrawal.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- With three RCPs operating, the limit is applied to the loop with two RCPs in operation. ---------- Verify RCS loop pressure meets the RCS loop pressure limits specified in the COLR.	12 hours
SR 3.4.1.2	----------NOTE---------- With three RCPs operating, the limit is applied to the loop with two RCPs in operation. ---------- Verify RCS hot leg temperature meets the RCS hot leg temperature limit specified in the COLR, AND is ≤ 605.8°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate meets the RCS total flow rate limits specified in the COLR, AND is ≥133.5 E6 lb/hr with four RCPs operating or ≥ 99.7 E6 lb/hr with three RCPs operating.	12 hours
SR 3.4.1.4	----------NOTE---------- Only required to be performed when stable thermal conditions are established > 90% of ALLOWABLE THERMAL POWER. ---------- Verify RCS total flow rate is within limit by measurement.	24 months

5.7 Procedures, Programs and Manuals

5.6.2.18 COLR (continued)

LCO 3.2.3 AXIAL POWER IMBALANCE Operating Limits
LCO 3.2.4 QUADRANT POWER TILT
LCO 3.2.5 Power Peaking Factors
LCO 3.3.1 Reactor Protection System (RPS) Instrumentation
SR 3.4.1.1 Reactor Coolant System Pressure DNB Limits
SR 3.4.1.2 Reactor Coolant System Temperature DNB Limits
SR 3.4.1.3 Reactor Coolant System Flow DNB Limits
LCO 3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC:

BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed) and License Amendment 144, SER dated June 25, 1992. The approved revision number for BAW-10179P-A shall be identified in the COLR.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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

a. Other Applicable ITS:

3.4.4 RCS P/T Limits
3.4.11 Low Temperature Overpressure Protection

b. RCS pressure and temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. The analytical methods used to determine the pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and approved by the NRC in BAW-10046A, Rev. 2, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," June 1986. The analytical method used to determine vessel fluence shall be those reviewed by the NRC and documented in BAW-2241P, May 1997. The analytical method used to determine LTOP limits shall be those previously reviewed by the NRC based on ASME Code Case N-514. The Materials Program is in accordance with BAW-1543A, "Integrated Reactor Vessel Surveillance Program."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

that transients initiated from the limits of this LCO will meet the event-specific DNBR acceptance criterion. This is the acceptance limit for the RCS DNBR parameters. The transients analyzed for include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "APSR Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE OPERATING LIMITS," and LCO 3.2.4, "QUADRANT POWER TILT."

For full power conditions, limits are set on loop parameters to meet the DNBR criterion. The minimum core outlet pressure is the limit value in the reactor coolant loop as measured at the hot leg pressure tap. The maximum hot leg temperature specified is the limit value as measured at the hot leg resistance temperature detector. The minimum volumetric flow rate corresponds to the mass flow rate for four RCP operation at full power.

Additionally, analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump operation. The flow limits for three pump operation are approximately 25% lower than the four pump limits. To meet the DNBR criterion, a corresponding maximum power limit of less than the top of the "doghouse" (Power-Imbalance-Flow trip setpoint contained in the COLR) is required in combination with the 3 pump limits (see Bases for LCO 3.3.1, "RPS Instrumentation").

RCS DNB limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

Limits on RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate ensure that the core

(continued)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR - 72

ATTACHMENT F

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Detailed Evaluation

quantified and analyzed. More detailed discussions on these effects are provided in Section C. The impact on component integrity and performance is discussed in Section D. The impact of the RCS flow reduction on transient and accident conditions will be discussed in depth in Section E.

B. METHODS DESCRIPTION

The methods used to analyze the effects of 20% tube plugging are grouped into three categories: thermal-hydraulic performance-related, fuel performance-related, and safety analysis-related. The codes or methods used help characterize the impact of tube plugging on these general categories of plant operation.

1. OTSG Thermal-Hydraulic Performance Related Codes

Tube plugging affects OTSG thermal-hydraulic performance. As will be discussed in Section C, increases in RCS hydraulic resistance, reduction in secondary heat transfer and its related secondary-side responses, are performance impacts of tube plugging.

- FSPLIT — This is a vendor-proprietary code [Reference 2] which determines RCS flow and temperature distributions as a result of tube plugging. The code was also used in conjunction with vessel model flow test data, to determine the effects of plugging asymmetry. FSPLIT models were benchmarked against plant operation with accurate feedwater flowrates. These benchmark values were typically obtained at plant startup, prior to any feedwater venturi fouling, or after the venturis have been cleaned.

  FSPLIT has also been used in support of the 20% OTSG tube plugging license amendment at Three Mile Island-1 (References 22 and 23).

- VAGEN — This is a vendor-proprietary one-dimensional, homogeneous equilibrium code [Reference 3] that predicts mean steam temperatures for OTSGs. The VAGEN model was benchmarked to plant data. Tube plugging is modeled by adjusting the primary-to-secondary heat transfer area.

- PORTHOS — This is an industry-recognized three-dimensional, non-homogeneous, non-equilibrium tubed heat exchanger program. A vendor-proprietary version of PORTHOS [Reference 4] customized for OTSG applications was used for this effort. PORTHOS allows localized effects to be simulated, such as those due to the following OTSG design features:

  - Inspection lane — A tube row-wide section within the tube bundle array with no tubes.
  - Tube-to-shroud gap — The annular region between the tube bundle and the shroud which separates the boiling region and the downcomer and upper steam space.

- Localized plugging — Such as in the peripheral region of the tube bundle, where plugging limitations exist to ensure the continued effectiveness of emergency feedwater cooling.

- Unique OTSG features — Such as an internal auxiliary feedwater header, or a three tube-wide inspection lane.

2. Fuel Performance Codes

Tube plugging or sleeving result in global changes, such as the reduction in RCS flow. On a local scale, examples include variations in core inlet flowrate and temperature perturbations due to asymmetric plugging. The combination of global and local effects are analyzed with respect to fuel or core performance. The major codes used to perform these analyses are as follows:

- TACO3 — The reduced flowrates also affect the analysis for predicting the end-of-life fuel rod internal pressure. TACO3, an approved code [Reference 5], is the principal tool for performing this analysis.

- KOROS — Reduced flowrates affect the analysis for clad corrosion prediction. The KOROS code utilizes the approved COROS02 method [Reference 6] for corrosion analyses for specific fuel batches. The analysis predicts the thickness of a corrosion layer given the highest pin burnup.

- NEMO — This is an approved vendor-proprietary code [Reference 7] for three-dimensional core nodal analysis, which yields solutions for neutron flux, power, and reactivity. The code was used to analyze the impact of plugging and its effects, principally flow and temperature asymmetries, on neutronics and overall core performance.

3. Safety Analysis-Related Codes/Methods

- RELAP5 — A vendor-proprietary version, RELAP5MOD2-B&W, has been approved for use for both LOCA [Reference 8] and non-LOCA [Reference 9] applications. The model can mimic the presence of plugged OTSG tubes. The code calculates the RCS and secondary responses to transient and accident conditions.

- LYNXT — This is an approved vendor-proprietary code [Reference 10] for calculating the Departure from Nucleate Boiling Ratio (DNBR). As part of the Statistical Core Design (SCD) method [Reference 11], the code is used to assess the DNB impact of tube plugging, particularly with respect to the reductions and/or redistributions in system and core inlet flow.

Where appropriate, if none of the methods above apply to areas which need to be addressed, evaluations were performed to determine the impact of tube plugging.

C. GENERAL TUBE PLUGGING EFFECTS

The general effects of plugging 20% of all OTSG tubes are increased RCS flow resistance, decreased primary-to-secondary heat transfer, and decreased primary side volume. These effects are detailed below, along with the impact of asymmetric plugging distribution.

1. Increase in RCS Flow Resistance

RCS flow resistance increases due to the reduction in flow area through the tube primary side. Sleeving slightly increases flow resistance. Approximately 6.7 sleeves have the same hydraulic resistance as one plug. The overall effect is a decrease in system flow, particularly for significant levels of plugging. Table 1 below shows the impact on nominal RCS flow, conservatively assuming a high resistance core with a debris filter plate. The case of symmetric plugging is reflected in the table.

Table 1 Impact of Plugging on Nominal RCS Flow (352,000 gpm Design flow)

Percent Plugged (Symmetric)	Nominal Flow, gpm
0%	386,290 (109.7% of Design)
20%	372,190 (105.7% of Design)

The table indicates a reduction of 4% in RCS flow, if symmetric plugging is assumed. In the case of asymmetric plugging, an FSPLIT case reflecting a maximum asymmetry of 25% in one OTSG and 0% in the other (25/0) determined a minimum flow penalty of 0.4% for four-RCP operation. However, no penalty was found to be required for three-RCP operation. For the purpose of core design and DNB determinations, a 0.5% local flow penalty is imposed.

2. Decrease in Primary-to-Secondary Heat Transfer Area

Sleeves physically do not significantly reduce a tube's heat transfer capability. However, plugging a tube removes that tube's primary and secondary surface areas from the primary-to-secondary heat transfer path. The reduction in heat transfer area results in an increased steam generator boiling length, and a decrease in the steam superheat temperature. The reduction in steam temperature requires an increase in feedwater flow to maintain the rated power production. 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 be lower, while the hot leg temperature will be higher. Using FSPLIT, it was found that hot leg temperature will increase by 1°F, while cold leg temperature will decrease by 1°F, for a total change of 2°F in the hot-to-cold leg temperature difference.

To address the effects of asymmetric plugging, B&W units are equipped with the Integrated Control System (ICS), which automatically modifies main feedwater flow to adjust the cold leg temperature difference (ΔT_C) between the loops. However, if the

asymmetry is sufficiently large, it may challenge the plant-specific feedwater flowrate limit on feedwater-induced vibration (FIV), more refined FIV analyses or a power reduction may be required.

3. Decrease in RCS Free Volume

Plugging a tube also reduces the available RCS volume slightly, by about 0.095 ft^3 per tube. As such, with 20% of tubes plugged, RCS volume would be reduced by approximately 600 ft^3. While the reduction has minimal direct impact on normal operations, transient responses may be altered slightly, as discussed in Section E.

The above discussions establish the global limitations or system impact of tube plugging. The impact on the integrity and steady-state performance of specific components or sub-systems are addressed in the next section.

D. Steady-State Component Integrity And Performance Evaluations

Global effects cascade to more local effects on specific components or sub-systems. It is on this more local scale that potentially deleterious effects of tube plugging can be of concern. The components or sub-systems examined fall into three broad categories: fuel, primary (or RCS) side, and secondary side.

1. Fuel Component Integrity

Fuel components are primarily affected by the reduction and potential asymmetry in system flow and the corresponding responses in core inlet flow. The affected aspects of fuel design are discussed below.

- Clad Corrosion

  Fuel rod mechanical analyses performed for each fuel rod design include analyses for cladding fatigue, transient strain, stress, creep collapse, and corrosion. The most limiting of these is corrosion, a phenomenon sensitive to flowrate. Corrosion concerns are more important for fuel assemblies which have undergone at least one cycle of operation.

  Using the KOROS/COROS02 method, clad oxidation has been determined to be the limiting constraint while CR-3 utilizes Zircalloy-4 cladding. Employing advanced clad material, such as Framatome Cogema Fuels' M5 alloy, avoids this problem. Until such time as advanced cladding is deployed, limitations on burnup and peaking will be considerations of cycle-specific reload designs.

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

  TACO3 was used to determine the impact of the reduced flow on fuel-to-coolant heat transfer. This initial condition is important for the LOCA analysis initialization (Section E). The evaluations found that a reduction of up to 4.5% in RCS flow

resulted in an increase of less than 10°F in average fuel temperature. Additionally, in terms of centerline fuel melt calculations, the impact is expected to be comparably small.

The impact on fuel rod end-of-life internal pressure was also determined using TACO3. While the maximum internal pressure limit was determined to be not limiting, the more limiting phenomenon was revealed to be clad lift-off. 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. Thus, precluding clad lift-off will be a consideration for fuel cycle designs.

- Guide Tube Boiling

Elevated core exit temperatures, particularly the potential for saturated conditions within a guide tube, poses a corrosion concern in that region. For the guide tube boiling analysis, the presence of a burnable poison rod assembly (BPRA) in the tube was conservatively assumed, since the hydraulic resistance in the tube is increased, and the BPRA has the highest heat flux than any other type of control element. The evaluation of this control element concludes that at 2568 MWt, margin exists to the acceptance criterion of no saturation in the guide tube Assembly Hold-down Springs, Guide Tubes, and Spacer Grids.

The pertinent effects of tube plugging on these components are the reduction in RCS flow and up to a 2°F (or less) increase in core outlet temperature. For reduced RCS flow, there is a 7.5% decrease in the forces that contribute to fuel assembly lift and spacer grid lift. This represents a net increase in margin to assembly lift at either power or fourth pump startup temperature.

On the other hand, from the perspective of assembly compression, there is a net increase of 50 lbs of compressive force on the fuel assembly guide tubes over the design life. However, this load, which is evaluated in terms of fuel assembly growth, bow, and guide tube structural integrity, does not impose any specific limitation on fuel assembly performance.

As far as the increased coolant exit temperature, the corrosion impact on these particular components is well bounded by that which is predicted to occur on an active fuel rod surface. The predicted oxide thickness bounds the data from the Post-Irradiation Examination oxide measurements from other B&W units operating at higher power levels. In summary, so long as bulk boiling is precluded, as described in the Guide Tube Boiling section above, tube plugging poses no limitations on these fuel assembly components.

2. Core Physics Impact

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

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 methods employed in this analysis have been approved, and are described in Reference 12.

The results of the maneuvering analysis determine the RPS imbalance trip limits and the core operating limits report provided in the cycle-specific Core Operating Limits Report (COLR). The impact on the RPS is discussed in Section F.

To determine the effects of plugging on steady-state core physics, four 177-FA LL plant configurations were simulated for a 24-month depletion:

a. 0/0 no tube plugging with the nominal core flow rate (base case).

   This is the base case, with no plugging, no asymmetry induced in the input conditions, and with the nominal core flowrate of 109%.

b. 20/0 with a 5% global flow reduction

   This case represents the maximum asymmetry condition, with 20% of all tubes plugged in one OTSG, and 0% in the other. The RCS flowrate was also reduced by 5%, a reduction which bounds (is greater than) the 4% identified in Section C.1. To represent the core asymmetry, the NEMO core model was divided in half, with a ±1.1°F inlet temperature asymmetry to induce a moderator temperature tilt, and a ±0.5% flow penalty to simulate the flow asymmetry.

c. 20/0 with no flow reduction

   This case represents the same condition as (b), but with no RCS flow reduction. This is used to establish the sensitivity of the core to flow and temperature asymmetries only. The same asymmetry penalties as (b) were applied.

d. 20/20 with the 5% global flow reduction

   This case represents the maximum but equal amount of plugging in both OTSGs. This is used as a flow sensitivity case, for comparison with the 0/0 case.

The four cases encompass symmetric and asymmetric plugging conditions. The effects on the different aspects of core physics are as follows:

- Burnup

  Assembly-specific comparisons for maximum and minimum pin burnups in the core at carious times-in-life were made. The comparisons were generated by calculating the differences in pin burnup for the four cases above. It was found that some rods accumulated increased burnups while others accumulated decreased end-of-cycle burnups. A conservative equation to determine an adjusted burnup was developed for use in core designs.

- Radial Pin Peaking

  Assembly-specific comparisons for 2-D radial peaking factors in the core at various times-in-life were made. The comparisons were generated by calculating the percent differences in radial peak for the four cases above. The largest increases in radial occurred at or shortly after BOC. As the fuel depletes, the changes decreased such that they were almost insignificant at EOC.

  The largest increases occurred for the two 20//0 cases (5% penalized flow and nominal flow). Table 2 below summarizes the peaking penalties to be applied for reload analyses:

Table 2 2-D Peaking Factor Increase

	BOC	EOC
All Assemblies	0.6%	0.3%
Peak Assembly	0.5%	0.0%

- Total (3D) Peaking

  Assembly-specific comparisons were made for 3-D total peaking factors in the core at various times-in-life. The comparisons were generated by calculating the percent differences in radial peak for the four cases above. The largest increases in radial occurred at or shortly after BOC. As the fuel depletes, the changes decreased such that they were almost insignificant at EOC.

  The largest increases occurred for the 20//0 case (5% penalized flow). Table 3 below summarizes the peaking penalties to be applied for reload analyses:

Table 3 3-D Peaking Factor Increase

	BOC	100 EFPD	EOC
All Assemblies	1.0%	0.4%	0.3%
Peak Assembly	0.7%	0.4%	0.0%

- Axial Power Imbalance

  Axial power imbalance comparisons were made for different steady-state depletions. For all plant configurations, the imbalance change was found to be small (< 0.5% at BOC) relative to the base case. By MOC, the differences became insignificant. Therefore, the axial imbalance changes due to tube plugging will not affect steady-state operation.

- Quadrant Power Tilt

  Quadrant Power Tilt was calculated at various times in the steady-state depletions. The maximum increase in tilt was 0.30%, which is insignificant with respect to reload analyses because they are well within the quadrant tilt peaking allowance implemented in the maneuvering analyses. The radial and total peaking adjustments implicitly include the effects of power redistribution, but the magnitude of tilt itself is not significant. Therefore, there is not a penalty required for quadrant power tilt.

- Quadrant Burnup Tilt

  Quadrant Burnup Tilt was calculated at various times in the steady-state depletions based on maximum pin burnups in each assembly in the core. The maximum increase in burnup tilt was found to be less than 0.10%, which is insignificant with respect to reload analysis impact. Therefore, there is not a penalty required for quadrant burnup tilt.

- Transient Xenon

  The peaking margin evaluations performed in the maneuvering analyses include the effects of Xenon spatial redistribution during power level changes. These effects are modeled in NEMO using simulated Xenon transients. The results of this analysis indicate that tube plugging does not significantly alter the transient Xenon power distributions.

- Boron Concentrations and Boric Acid Storage Requirements

  The impact of 20% tube plugging on RCS boron is only noticeable at power conditions, therefore, no instantaneous impact exists on shutdown or refueling concentrations. Moreover, no changes in boric acid storage requirements are necessary.

- Rod Worth

  The various aspects of rod worth in terms of total rod worth (TRW), ejected rod worth (ERW), stuck rod worth (SRW), and dropped rod worth (DRW) were examined in light of tube plugging. The impacts of the RCS flow reduction and temperature and flow imbalances on these worths was found to be insignificant.

Therefore, those evaluations which use rod worths remain valid, and no penalty or special considerations are required for reload analyses.

- Temperature Reactivity Coefficients

  Moderator temperature coefficient (MTC) and isothermal temperature coefficient (ITC) are conservatively calculated. The BOC MTC and ITC measurements are made during Zero Power Physics Testing. HZP conditions are not affected by tube plugging, and therefore, the measurements are similarly unaffected.

  An EOC HFP MTC of -40 pcm/°F is evaluated with control rods partially inserted at EOC with no soluble boron for a conservative MTC estimate. The asymmetric condition effects on power distribution are near zero at EOC relative to the base case, and the impact on asymmetric inlet temperature is minimal and does not necessitate additional penalties.

- Power Coefficients and Deficits

  The total power defect and power coefficient calculation methods remain valid since NEMO provides an adequate representation of the core average moderator temperature. The asymmetry effect on deficits and coefficients is very minor. The axial and radial core power distributions, core isotopic concentrations, and Xenon distributions are not significantly affected, as discussed above. Therefore, the total power deficit used in shutdown margin calculations and power coefficients will remain nearly unchanged.

- Shutdown Margin

  The calculation of shutdown margin requires using total and stuck rod worths, as well as total power deficit, off-nominal flux and rod insertion allowances, and a penalty on the insertable rod worth. All of these have been discussed except for the off-nominal flux allowance and reactivity insertion allowance. The former does not need to be re-evaluated, based on the results of the power distribution discussion (see Radial Pin Peaking, Total (3D) Peaking, Quadrant Power Tilt sections). Similarly, the reactivity insertion allowance is calculated on a cycle-specific basis with a severely skewed power distribution that is more severe than allowed by operating limits. Since no significant impact is expected for either, shutdown margin is not adversely affected by tube plugging.

3. Primary System Performance and Integrity

The thermal hydraulic inputs used in the primary component structural analyses (including the steam generator) are defined in the RCS functional specifications. The RCS functional specification provides system and component pressure, temperature, and flow conditions for the normal, upset, emergency, and faulted design events. It did not originally address tube plugging or its effect on system parameters. The effects of plugging on these values are discussed below.

- Temperature and Flow

The RCS functional specification uses the full power T_{hot} and T_{cold} as either the starting or completion points for most of the plant transients analyzed for fatigue cycles. Exceptions include plant heatups and cooldowns, which end or start at hot shutdown or approximately 8% power conditions.

The original RCS functional specification values of T_{hot} and T_{cold} are based on the design flow rate. CR-3's design flow rate is 88,000 gpm per RCP (or 352,000 gpm total RCS flow). Thus, for a 2568 MWt plant with an additional 16 MWt (to approximate RCP heat, makeup/letdown, and ambient losses) at 579°F, T_{cold} and T_{hot} would be 554°F and 604°F, respectively.

Initial startup flows (no tube plugging) were typically 110% of the design flow so that the primary hot leg-to-cold leg temperature difference was less than the design specification value. Operating T_{hot} values were less than the specification value and T_{cold} was greater than the specification value. Tube plugging reduces the RCS flow, increases T_{hot}, and decreases T_{cold}. However, since the actual flow rates at 20% tube plugging will remain greater than the design flow rate, the T_{hot} and T_{cold} values at 20% plugging will be bounded by those in the RCS functional specification as shown on Table 4.

Table 4 Comparison of Primary System Values

Parameter	RCS Functional Specification	0% Tube Plugging	20% Tube Plugging
Thot	604°F	601.7°F	602.7°F
Tcold	554°F	556.3°F	555.3°F
RCS Flow	352,000 gpm	386,290 gpm	372,190 gpm

- Erosion

While tube plugging reduces the overall RCS flowrate, it also increases the velocity through individual tubes. For example, while 20% tube plugging reduces the RCS flow from approximately 110% to 106%, it increases the velocity from 17.7 ft/s to 21.4 ft/s. While there is not an established limit on tube velocities, however, pipe flow for water is usually limited to 60 ft/s to avoid erosion problems. To achieve tube velocities of this magnitude, an incredible amount of tube plugging would be required

(>50%). Thus, the increase in steam generator tube fluid velocity due to plugging is not a driving factor in establishing tube plugging limits.

- Primary Side Flow-Induced Vibration

  Symmetric steam generator tube plugging does not adversely affect primary component flow-induced-vibration. Flow induced vibration (FIV) is a function of dynamic pressure, which is in turn proportional to mass flow rate squared divided by the density. An increase in mass flow rate or a decrease in density tends to increase FIV.

  Plugging introduces competing effects. As steam generator tubes are plugged, primary flow decreases, but increases the primary hot leg-to-cold leg temperature difference: cold leg temperature decreases and hot leg temperature increases. Thus, hot leg density will decrease (unfavorable) and cold leg density will increase (favorable).

  For 20% symmetric plugging, flow is reduced by approximately 4% and the primary temperatures will change less than 2°F in both the hot leg and cold leg. A 2°F change in hot leg temperature from say, 602°F to 604°F represents an approximately 0.4% decrease in density. In the hot leg, this density decrease is more than offset by the mass flow rate decrease, and the dynamic pressure will decrease with tube plugging. In the cold leg, both the flow rate decrease and density increase will reduce the dynamic pressure. Thus, the susceptibility of primary components to flow induced vibration will not be increased due to symmetric steam generator tube plugging.

  For asymmetric plugging, the RC flow in the loop with the least plugging will not be significantly different from that of an unplugged system. For example, 15/0 plugging would result in only a 0.2% increase in RC flow in one loop (over an unplugged case), while for 25/15, the flow rates in each loop will be bounded by the unplugged flow rate. Since CR-3 does not have an unplugged OTSG, the susceptibility of primary components to flow induced vibration will not be increased due to asymmetric steam generator tube plugging.

- Tube-to-Shell Delta-T

As previously discussed, tube plugging reduces the steam temperature. As a result, the steam annulus and adjacent shell temperature will decrease as well. The lower shell temperature will remain constant as long as steam pressure remains unchanged. Relative to tube-to-shell ΔT limits, tube plugging improves the tensile (shell hotter than tubes) conditions and exacerbates the compressive (tubes hotter than shell).

During zero-power situations, the steam temperature will be the saturation temperature corresponding to the steam pressure, which is governed by operator actions (in the cases of heatups and cooldowns, for example) or by secondary relief valve setpoints (in the case of plant transients). Thus, tube plugging should not affect the shell temperature for zero-power conditions.

For power conditions, superheated steam is produced when the steam generator pressure reaches its normal power operating value. The full power (no plugging) tube-to-shell temperature difference is approximately 20°F, while the tube-to-shell ΔT for 20% plugging is approximately 24 °F. Although the tube-to-shell ΔT is increased by plugging, remains within the 60°F compressive limit.

E. FSAR Chapter 14 Transient and Accident Analyses

This section describes the impact of OTSG plugging on the transient and accident analyses identified in Chapter 14 of CR-3's Final Safety Analysis Report. The discussion is divided into the different classes of events, with each class reflecting a type or types of acceptance criteria.

1. Normal Operation: Uncompensated Operating Reactivity Changes

This event determines the rate of change in core reactivity and RCS temperature during a fuel cycle. The acceptance criterion is whether or not sufficient time exists for automatic control systems to respond to and compensate for those changes.

This event was not reanalyzed. The increase in the number of plugged tubes will slightly affect the initial conditions at the beginning of the cycle, but will not significantly alter the rate of change of reactivity or temperature as the fuel cycle progresses. Therefore, tube plugging does not affect the results of this analysis.

2. Radiological Release Events

These events involve postulated radiological releases to the environment, and are non-mechanistic in general. The acceptance criteria therefore involve either 10 CFR Part 100 or 10 CFR Part 20. The events include:

- Fuel Handling Accident

  This accident is postulated to occur during refueling, when the OTSGs are not in service. Therefore, an increase in tube plugging will not affect this event.

- Waste Gas Decay Tank Rupture

  A postulated rupture of this tank, which is assumed to contain a conservatively high activity, is analyzed for dose consequences only. The RCS volume is an input to the analysis. However, since the RCS volume will be reduced by tube plugging, it is conservative to consider unplugged OTSGs. Therefore, the Waste Gas Decay Tank Rupture did not require reanalysis as a result of tube plugging.

- Maximum Hypothetical Accident

  The Maximum Hypothetical Accident analysis determines the dose from a postulated Large Break Loss Of Coolant Accident (LBLOCA) which involves fuel damage. Unlike 10 CFR 50.46-compliant LBLOCA analyses, non-mechanistic assumptions regarding the release are made. Therefore, the results of this analysis do not depend on OTSG performance or tube plugging.

3. Primary Flow Events

Primary flow events are those initiated by changes in RCS flowrate due to some equipment failure or inadvertent operation. Of concern in these events is the possibility of challenge to DNB acceptance criteria. As such, this class of events sets the limiting value for DNB-sensitive parameters, such as minimum flow. CR-3 utilizes the Statistical Core Design (SCD) methods described and approved in Reference 11. The events analyzed for CR-3 include the following:

- Cold Water Event

  The event is a mild overcooling transient induced by an increase in flow from an idle loop, such as from a reactor coolant pump (RCP) start. The potential exists for a reactivity transient due to the combination of an existing negative moderator temperature coefficient and the cooler incoming water.

  This event was not reanalyzed. Tube plugging will actually have a slight beneficial effect on this event due to the increased loop resistance and slightly smaller primary-side tube inventory. These would tend to slow the injection rate and limit the amount of cold water introduced into the active RCS. Therefore, tube plugging will not pose a challenge to the acceptance criteria of this event.

- Four RCP and One RCP Coastdowns

  These events involve the inadvertent tripping of either all four or only one RCP. In the case of the four-RCP coastdown, system flow decreases with time. For the one-RCP coastdown case, system flow decreases slowly, but is maintained by the three remaining RCPs. Both events are normally analyzed as part of the reload process.

  These events were reanalyzed. The effects of tube plugging (symmetric and asymmetric) on the RCS flow coastdown events were determined using RELAP5/MOD2-B&W. RC pump torque coefficients were adjusted so that zero-plugging model predictions matched or bounded measured flow data. Tube plugging scenarios of 20% symmetric and up to 30/0 asymmetric were analyzed to develop transient RCS flow values.

  To account for a 20% plugging level and the potential of having a 20/0 plugging asymmetry, the RCS flow used in SCD analyses, 109% of design, was penalized with a 4.5% reduction. The reanalysis of these events showed that the four-RCP coastdown is more limiting. However, the DNB ratio remained greater than the thermal design limit criterion using SCD methods. This establishes the flowrate of 104.5% of design as the new ITS minimum RCS flow limit.

- Locked Rotor

  The locked rotor event involves a mechanical failure of one RCP such that RCS flow is suddenly decreased by the equivalent of that RCP's contribution. A calculation to

- Rod Withdrawal at Power

  This event is initiated by an inadvertent withdrawal of a control rod during power operations. As with the Startup event, the reactivity insertion induces a power, pressure, and temperature excursions.

  This event was not reanalyzed. On the primary side, the sequence of events may occur slightly faster due to the reduced heat transfer area and lower RCS inventory. However, the reactivity insertion and subsequent effects are bounded by the Startup Event. Moreover, the presence of forced flow ameliorates the impact of tube plugging. Therefore, the acceptance criteria for this event will continue to be met.

- Moderator Dilution

  Two scenarios are investigated for this event: one at full power, and one in a refueling condition. For the former, the moderation dilution event is a slow power, pressure, and temperature excursion based on a postulated uncontrolled but limited dilution of the RCS boron concentration. In the latter scenario, an uncontrolled dilution occurs which requires operator intervention. The dilution causes a reactivity transient which has the potential to cause power, pressure, and temperature excursions.

  This event was not reanalyzed. For the at-power scenario, as with the rod withdrawal at power, the sequence of events may occur slightly faster due to the reduced heat transfer area and lower RCS inventory. However, the rate of reactivity insertion will be bounded by the Startup event. Moreover, the presence of forced flow ameliorates the impact of tube plugging. For the refueling scenario, the OTSGs are not relied upon for core cooling. Thus, tube plugging will not have a role in the progression of the event. Therefore, the acceptance criteria for this event will continue to be met.

- Dropped Rod Event

  The dropped rod event is initiated by a control rod that inadvertently drops while the reactor is at power. The event is characterized by a local power suppression, and an initial cooldown. However, with a negative moderator temperature coefficient, the cooldown causes a power excursion, which returns the core to a new equilibrium power, and an RCS heatup and pressurization.

  This event was not reanalyzed. The event is best characterized as a rapid reactivity change which evolves into a relatively slow power excursion. Thus, similar to the rod withdrawal at power and moderator dilution events, the reduced RCS volume and primary-to-secondary heat transfer area will cause a slight increase in the rate of response of primary system parameters. Again however, the continued presence of forced RCS flow ameliorates the impact of tube plugging. Therefore, the acceptance criteria for this event will continue to be met.

- Control Rod Ejection

  This event is initiated by a break in a control rod drive motor housing, which causes a control rod to be ejected from the core. The positive reactivity insertion is rapid but is immediately curtailed by fuel temperature feedback effects, such that the maximum fuel enthalpy acceptance criterion will be met.

  This event was not reanalyzed. Because of the short duration of the transient, the primary-to-secondary heat transfer has no significant role in the sequence of events. Therefore, the acceptance criteria for this event will continue to be met.

- Loss of Electric Load / Turbine Trip

  The loss of electric load occurs when the external transmission system deteriorates significantly, causing the unit to automatically disconnect from the transmission grid. The result is a sudden mismatch of heat production and heat removal. This is the limiting event for peak secondary pressure.

  However, this event was not reanalyzed. Considering the reduction in primary-to-secondary heat transfer due to plugging, the secondary side pressurization rate would actually be reduced, albeit slightly, by plugging. Thus, the results are not expected to be significantly affected. Therefore, the acceptance criteria for this event will continue to be met.

- Loss of Main Feedwater / Feedwater Line Break

  The loss of feedwater (LOFW) event simulates a main feedwater pump trip due to an event such as a loss of power, loss of suction, or malfunction of valves in the system. Following MFW pump trip, the heat removal capacity of the secondary side drops significantly and the primary temperature and pressure start increasing. The LOFW transient is performed with the RCPs operating to maximize the energy addition to the RCS. The feedwater line break postulates the failure of a main feedwater line to an OTSG, coupled with a blowdown of that OTSG. The result is a similar loss of heat sink removal. For both events, the controlling parameters are initial core power level (decay heat), the EFW actuation time, and the minimum EFW flow rate.

  The LOFW event, although not the limiting overheating event with respect to peak RCS pressure, does set the minimum EFW rate. The event described in FSAR Chapter 14 already incorporates 20% tube plugging, following a reanalysis with RELAP5/MOD2-B&W. The analysis concluded that the acceptance criteria will be met with 20% tube plugging and the existing total EFW flow requirement of 550 gpm.

- Loss of All AC Power / Station Blackout (SBO)

The loss of all AC power is an event in which all unit AC power is postulated lost. The reactor and RCPs will trip promptly upon loss of power. The unit batteries are available to supply DC power and redundant quick-starting emergency diesel generators are available to supply AC power to vital loads. Even if the emergency diesel generators are not available, steam- and diesel-driven emergency feedwater pumps can supply cooling water to the steam generators to remove core decay heat and the residual stored energy in the coolant and metal.

This event was not reanalyzed. Each OTSG was sized to remove approximately 60% of the core power, so while the OTSG heat transfer area is reduced with tube plugging, sufficient tube surface area exists to remove energy from the RCS. Emergency feedwater flow or the relief capability of the MSSVs will remain the same. Therefore, the acceptance criteria for the loss of AC power transient will continue to be met.

A special case was considered wherein a specified leak rate through the RCP seals was postulated for a total loss of station AC power, or a station blackout (SBO). The RCS would experience a net loss of inventory through the leaking RCP seals and other sources of leakage until power is restored to makeup or injection pumps to provide mass addition. The net loss of reactor coolant would be sufficient for steam voids to form in the hot legs and interrupt natural circulation flow in the primary loops. Continued loss of inventory could result in core uncovering and core damage if power is not restored soon enough to start injection to the RCS.

This SBO event was not reanalyzed, but was re-evaluated in light of tube plugging. Plugging will reduce the initial RCS inventory, reducing the inventory available to maintain core covering during the time required to restore power to makeup or injection pumps (coping period). Analysis of the SBO event, assuming 111 gpm of combined RCP seal, identified, and unidentified leakage, shows that loop flow interruption occurs after the pressurizer empties and the U-bend regions of the hot legs void.

Following loop flow interruption, the loss in inventory results in voiding of the reactor vessel down to the hot leg elevation and subsequent voiding of the cold leg discharge piping. Once the RV head region and cold leg discharge piping are voided, hot leg and steam generator levels continue to decrease, and the level of the reactor coolant in the steam generators falls below the tube sheet.

At this point, steam in the primary system is condensed on tubes cooled by emergency feedwater flow. Although the OTSG heat transfer area for condensation is reduced with tube plugging, there will still be sufficient tube surface area to boil emergency feedwater supplied by the steam- or diesel-driven EFW pumps. Consequently, the steam condensation rate via boiler-condenser cooling is sufficient to keep the core protected throughout the coping time. This condensation provides removal of core decay heat from the primary system and allows the primary system

pressure to decrease. Shortly after the onset of this boiler-condenser mode (BCM) of cooling, the required coping duration (four hours) is met, with the core still covered.

- Anticipated Transients without Scram (ATWS)

  This event is not a part of CR-3's FSAR Chapter 14 analyses. However, it is included in the discussion for completeness. An ATWS event is initiated by a design basis transient. Because of a postulated failure of the reactor protection system (RPS), control rod insertion does not occur following receipt of the high RCS pressure reactor trip signal. The primary ATWS safety concern is the pressurization of the reactor coolant system and the potential for a failure of the reactor coolant pressure boundary. Therefore, those ATWS events initiated by a reduction in heat removal (e.g. loss of feedwater for the B&W-designed plants) are the most limiting events. A maximum pressure of 3200 psig is the acceptance criterion.

  To combat the ATWS, CR-3 relies on the diverse scram system (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC). The DSS initiates an independent and redundant trip signal on high RCS pressure (2500 psig). On the other hand, AMSAC ensures a turbine trip and emergency feedwater actuation. Consequently, the peak RCS pressure predicted for the ATWS event is significantly below the maximum pressure criterion.

  This event was not reanalyzed, but was evaluated with consideration for tube plugging. For the limiting ATWS event, removing OTSG tubes from service does not have an effect on the magnitude of emergency feedwater flow or the energy relief capability of the MSSVs. Even with 20% tube plugging, sufficient tube surface area exists to boil the emergency feedwater completely. And while the RCS temperature and pressure may increase at a slightly faster rate due to the reduction in initial RCS inventory, resulting in a small increase in peak pressure, significant margin exists in the analysis to accommodate any small change in the peak pressure prediction. Therefore, the acceptance criterion will continue to be met.

5. Overcooling Events

The general acceptance criteria for these events are: (1) the core shall remain intact for effective core cooling, (2) the resultant doses shall not exceed 10CFR100 limits, and (3) OTSG tube failure due to pressure and temperature gradients shall not occur.

- Main Steam Line Break (MSLB)

The limiting transient in this category is the steam line break event. An MSLB is the double-ended rupture of a main steam line at power that results in a rapid depressurization and loss of the secondary inventory. The increased steam flow results in cooldown and depressurization of the RCS, which coupled with a negative moderator temperature coefficient (MTC), can cause positive reactivity addition and a subsequent resurgence of reactor power due to subcritical multiplication after the reactor has tripped.

This event was not reanalyzed. However, it is reevaluated with consideration for tube plugging. An increase in tube plugging produces a decrease in the total volume of the RCS and a decrease in RCS flow. Additionally, to maintain the same power production while accommodating tube plugging, initial main feedwater flow is increased due to the decreased OTSG heat transfer surface area and reduced RCS flow. The main feedwater flowrate increase leads to an increase in the boiling length. The increase in boiling length is proportional to the amount of tube plugging. The additional inventory related to the increase in boiling length is added in the high void fraction region such that the total mass increase in the OTSG is relatively small: approximately 1% of the nominal full power inventory of 45,000 lbs. As such, the RCS overcooling event and subsequent return to power in a MSLB event may be more severe.

The key parameters for the MSLB event are the initial OTSG inventory and the main feedwater flow prior to isolation. The current analysis assumes a conservatively high inventory of approximately 55,000 lb per OTSG. When modeling this OTSG mass for the B&W plants using the RELAP5/MOD2 code, this inventory is consistent with a OTSG operate range level of approximately 95 to 99 percent. The MSLB design base analyses will remain bounding for the B&W plants with extended tube plugging as long as the error adjusted OTSG level remains below 95% on the operate range, and the plant is not being normally operated with the OTSG aspirator port flooded. The conservative MSLB analysis OTSG inventories will also cover the slight increase in feedwater flow to the OTSGs prior to MFW pump trip or isolation valve closure. Therefore, the current analyses remain valid for increased levels of OTSG tube plugging.

The following cooldown events are not part of CR-3's FSAR Chapter 14 analyses, but are included only for completeness. They were not reanalyzed.

- OTSG Overfeed

  This is an event where steam generator overfill occurs due to feedwater system malfunction, step change in feedwater temperature, or flow. By removing OTSG tubes from service, the OTSG levels in the boiling region will generally be higher, but the overall increase in total OTSG inventory will be small, approximately 1%. This may require slightly earlier action times for mitigation. However, sufficient operator action time is available to recognize the event and to take action. The core response to these events is bounded by the MSLB, therefore, any limitations on extended tube plugging limits with respect to overcooling events will be determined by the MSLB evaluation.

- Excessive Load

  The excessive load increase event is the result of an inadvertent opening of a steam relief or turbine bypass valve by the operator. This event results in a sudden decrease in steam pressure accompanied with an increase in steam flow rate, which decreases the RCS pressure and temperature similar to, but slower than, the MSLB. The consequences of the MSLB are more limiting than the excessive load increase. Therefore the excessive load increase transient does not limit the amount of OTSG tube plugging.

6. Loss of Primary Coolant Events

These events are characterized by a postulated gross failure of a pressure boundary containing primary coolant. The failures are such that releases to the containment atmosphere, secondary side, and environment are encompassed.

- Letdown Line Failure

  This accident is considered a subset of the spectrum of loss of coolant accidents (LOCAs). However in this case, primary coolant is lost through a break in the letdown line outside the containment; specifically, the break occurs in the Auxiliary Building. The break can be isolated, which terminates the release, but an offsite dose consequence is incurred.

  For this accident, the key factor is isolating the break. The break will be isolated if either engineered safeguards (ES) actuates on low RCS pressure or manual operator action is taken. Depending on the break size and other boundary conditions, the initial decrease in RCS pressure during the subcooled blowdown may not be sufficient to automatically actuate ES, and operator action may be required to terminate the release. This can be true regardless of tube plugging.

  This event was not reanalyzed. The break flow remains choked throughout the transient. The release of primary coolant is controlled by the break area and RCS pressure/temperature. Since OTSG tube plugging has no effect on the break size and there is sufficient heat transfer area in the OTSG to remove RCS heat, RCS pressure and temperature will not be affected by tube plugging. Therefore, OTSG tube plugging will not have an effect on this event.

- Steam Generator Tube Rupture

  The steam generator tube rupture accident considers the double-ended rupture of a single steam generator tube. The event is primarily a dose-related transient since the core cooling aspects are bounded by the SBLOCA transients. The dose release is a function of the primary-to-secondary leak rate and the time required for the operators to depressurize the RCS to the point where the affected steam generator can be isolated. The design basis scenario assumes a continuous non-mechanistic break flow of 435 gpm from the affected OTSG, and 1 gpm from the intact OTSG. The condenser and off-site power are assumed available.

  This event was not reanalyzed. Plugging or sleeving tubes does not have a significant effect on the leakage rate from the broken tube. The initial OTSG levels will increase to compensate for the reduction in heat transfer area. The OTSG initial pressure is controlled by the turbine throttle valves and will remain the same. Only a slight reduction in the amount of superheat will result.

  For this transient, the operator is instructed to steam both steam generators until one of several isolation criteria is met. Since adequate steaming capacity exists through

the atmospheric dump valves and/or to the condenser, the duration of the blowdown will not change significantly and thus the off-site dose release from the event will not be significantly affected by tube plugging or sleeving. Therefore, the SGTR is not limiting relative to OTSG plugging and/or sleeving.

- Small Break LOCA

The SBLOCA is a break ranging in size from 0.01 ft^2 to 0.75 ft^2 in the RCS and attached piping. Mitigation of this event involves Engineered Safeguards (ES) systems to ensure core cooling. As such, the acceptance criteria are those specified in 10 CFR 50.46. In addition to Emergency Core Cooling System (ECCS) injection, OTSG cooling via EFW is required to mitigate the smaller break sizes. The EFW spray "wets" the outer rows of tubes at the EFW injection elevation and gradually spreads inward as the water falls and runs across the tube support plates. The heat transfer from the EFW removes energy from the RCS.

For the limiting size breaks, natural circulation will be interrupted due to steam accumulation in the hot leg piping. Boiler condenser mode (BCM) cooling will be required. The EFW flow will condense steam inside the OTSG tubes, which reduces RCS pressure, thus allowing more ECCS injection into the vessel. Condensate formed by the BCM process also supplements the ECCS injection rate, and depending on the break size and location, may represent a significant fraction of that flow. Removing OTSG tubes from this "wetted" region effectively reduces the heat transfer between the primary and secondary regions, which reduces the cooling potential of the OTSGs.

The Chapter 14 FSAR SBLOCA analysis already allows for 20% tube plugging. Small break sensitivity studies performed with 20% tubes plugged analyses with the RELAP5-based EM for the 177-FA lowered loop plant show that the most limiting break location is in the bottom of the cold leg piping between the reactor vessel inlet nozzle and the HPI nozzle. Two special break cases, an HPI line break and a CFT line break, were also analyzed. The limiting SBLOCA (cold leg pump discharge or CLPD break) was demonstrated to meet all the acceptance criteria of 10 CFR 50.46, as described in Reference 14.

Since OTSG cooling is essential for mitigation of the SBLOCA transient, a limit must be established for how many tubes in the "wetted" region, which is composed of a minimum of 10% of the tubes, can be plugged. The analyses were performed with a maximum of 75% of the tubes in the "wetted" region plugged in each OTSG.

- Large Break LOCA

  The LBLOCA event is a break anywhere in the RCS that is larger than the SBLOCA, up to a double-ended guillotine of a hot leg. The analysis and results of this event set limits on linear heat rate, as well as determine the adequacy of containment design.

  FSAR Chapter 14 analysis already assumes 20% tube plugging. LBLOCA analyses were performed using the RELAP5-based evaluation model (Reference 8). The results, summarized in Reference 14, demonstrate compliance with the five criteria of 10CFR50.46 for breaks up to and including the double-ended severance of the largest primary coolant pipe.

  The analyses considered a plant average tube plugging level of 20 percent, with 15 percent in the intact loop and 25 percent in the broken loop. This plugging distribution was arrived at via sensitivity studies that demonstrated that an increased resistance in the broken loop has a more significant effect on the predicted peak clad temperature (PCT) over a similar increase in the intact loop. The LBLOCA analyses support a unit average tube-plugging limit of 20 percent with a single OTSG not exceeding 25 percent plugged. The sensitivity studies also support a maximum plugging ratio with respect to LOCA of 25% equivalent tube plugging in the broken loop OTSG and any level of plugging in the intact OTSG up to 15%.

- LBLOCA Containment Analysis

  The mass and energy release data were not recalculated since the current data would conservatively bound the effects of OTSG tube plugging. The mass and energy release during blowdown is a function of the initial RCS inventory and the average temperature. The RCS average temperature will not be changed with increased levels of tube plugging. Tube plugging also results in a small reduction in the initial inventory, which reduces the available mass and energy addition to the containment building. Therefore, the existing blowdown mass and energy release data is conservative with respect to OTSG tube plugging. The post-blowdown mass and energy release is function of core power and the ECCS flow. Since these parameters are not affected by OTSG tube plugging, the data will also be conservative.

- Post-LOCA Boron Precipitation

  A challenge to the post-accident long-term core cooling capability is the potential for post-accident boron precipitation. Reference 15 reviewed and approved the measures in place at CR-3 to address this challenge. It is evaluated in light of increased tube plugging.

  Tube plugging 20% of all tubes reduces RCS volume by approximately 600 ft^3. The reduced volume introduces the potential for a smaller mixing volume or higher boron concentration in the sump. However, the total volumes considered in the analyses which established the post-LOCA boron precipitation control measures include the Borated Water Storage Tank (approximately 47,000 ft^3) and the Core Flood Tanks (1070 ft^3/tank). The decrease in RCS volume is small (approximately 1%) relative to

the total boron sources considered. Further, the key parameter in the analysis is the mass of boron available for concentration. A reduction of volume in the RCS is accompanied by a reduction in available boron mass. Moreover, the measures in-place remain based on increasing sump concentration indications. Tube plugging does not affect the method by which these concentrations are obtained, nor does it alter the established guidance. Therefore, tube plugging will not have a significant impact on post-LOCA boron precipitation controls.

The above section provides information that describes the impact and acceptability of up to 20% tube plugging. This information forms the bases for changes to the Improved Technical Specifications.

2. Impact on Reactor Protective System

CR-3's RPS instrumentation and functions are specified on ITS Table 3.3.1-1. The impact of tube plugging on RPS is assessed against the impact on the various accident analyses. The different functions of RPS potentially impacted by tube plugging are addressed below.

- High Flux Trip

  Inadvertent reactivity additions, which lead to power excursions, are the events protected against by this trip function. This class of events can be induced by a direct reactivity addition, or by overcooling. The most limiting of the former, which relies upon the RPS, is the Startup Event. The analysis for this event actually assumes the coincidence of the high flux and high RCS pressure trips. Since all the acceptance criteria for this event were met, no change to the high flux setpoint was found to be required.

  For overcooling, the MSLB is the most limiting. The high flux reactor trip setpoint is modeled for this event, but is dependent on the positive reactivity feedback due to the initial overcooling. A conservatively low setpoint is modeled such that the MSLB is not limiting for this setpoint. Therefore, no change to the high flux setpoint was found to be required.

- RCS High Outlet Temperature

  This trip function is not credited in the accident analyses. Increased levels of tube plugging result in a lower steady-state RCS flowrate and higher hot leg temperatures. No change to this setpoint is proposed. Thus, a postulated heatup event will result in an earlier trip time.

- RCS High Pressure

  The limiting RCS overpressure transient is the Startup Event. It was reanalyzed for CR-3 using the RELAP5/MOD2-B&W computer code with a pressurizer safety valve (PSV) lift tolerance of +3 percent and 30 percent OTSG tube plugging modeled. The results demonstrated that all acceptance criteria were met when tube plugging (greater than 20%) was considered, and no change to the setpoint was required.

- RCS Low Pressure

  The class of events pertinent to this trip function are the overcooling and loss of primary coolant events, both of which result in RCS depressurization. The limiting overcooling event, the MSLB, does not account for plugging of OTSG tubes. However, the initial inventory in the analyses is conservatively high and bounds operation with 20% tube plugging. Since the potential for overcooling the RCS has not changed appreciably, similar rates and magnitudes of RCS and OTSG

depressurization will be expected. Therefore, the RCS low pressure setpoint will not require modification for the MSLB and 20% tube plugging.

It must be noted that the Large Break LOCA does not directly credit a reactor trip. Therefore, it does not depend on the RPS functions. On the other hand, the Small Break LOCA relies upon an automatic reactor trip on RCS low pressure. The classical SBLOCA analysis models a loss of off-site power on reactor trip. Forced RCS and main feedwater flow will be available until the reactor trip so that the actual RCS low pressure trip setpoint is not sensitive to OTSG tube plugging. Reference 16 describes the review and approval of the current CR-3 RPS low RCS pressure setpoint, which is based on the SBLOCA analysis that accommodates 20% tube plugging.

- Variable Low Pressure Trip (VLPT) Function

  The VLPT function provides protection against the degradation of thermal margin, or margin to DNB. None of the accident analyses credit the VLPT. However, in order to verify that the plant can operate safely within the pressure and temperature limits during steady state operation with either four or three RCPs operating, a calculation of the VLPT setpoint was required. A calculation of minimum DNBR for different combinations of RCS pressure and temperature, with three and four pumps operating, was performed for CR-3. The results of the evaluation concluded that the current CR-3 trip function is conservative and could accommodate up to 20% OTSG tube plugging with no changes to the technical specifications required.

- Flux-to-Flow

  The pertinent events for this trip function are those involving a loss of forced RCS flow. These were reanalyzed as described previously, with the analyses demonstrating that the coastdown and locked RCP events confirmed that the trip function remains valid. No further evaluation was performed for the function, since the setpoint is evaluated and/or revised for each new fuel reload. Therefore, no additional justifications are required.

- Main Turbine, Loss of Both Main Feedwater, and Shutdown Bypass Trips

  The anticipatory trips for the turbine trip and loss of both main feedwater pumps are not affected by tube plugging. Therefore, they will remain unaltered. The bypass trips are linked to the normal high-pressure setpoint and will only require modification if the setpoint is changed. As described above, the high RCS pressure trip is not changed. Therefore, the bypass trip is not changed.

3. Impact on Engineered Safeguards (ES)

The automatic ES functions are found in ITS Table 3.3.5-1. The impact of tube plugging on these functions are addressed below, particularly in terms of the key events which involve a loss of primary coolant (i.e., SBLOCA and LBLOCA).

- RCS Pressure - Low

  The SBLOCA relies upon both the RCS low pressure trip and the ES low RCS pressure, with the latter condition initiating High Pressure Injection (HPI). For the limiting break size, the time of reactor trip (RPS) will not be changed significantly with increased levels of tube plugging. Reduced primary-to-secondary heat transfer can however, affect the time that the ES is actuated. As the level of tube plugging increases, depending on the break size and location, the minimum expected RCS pressure during the initial period of subcooled blowdown could be greater than the ES setpoint, and ES may not automatically be actuated at this point in the transient.

  If no operator action is assumed, ES will be delayed until the boiler-condenser cooling is established. The overall consequences of the transient become highly sensitive to the capacity of the ECCS once flow is initiated. Therefore, analyses were required to justify increased level of OTSG tube plugging. CR-3's analysis, which incorporates the revised ES (and RPS) setpoints described in Reference 16, was shown to successfully accommodate 20% tube plugging.

  In addition, new SBLOCA analyses were performed specifically for the BWOG to address localized tube plugging limits. The analysis modeled 20% overall tube plugging, with 25% in the broken loop and 15% in the intact loop. The analyses also limit the localized tube plugging in the EFW "wetted" region by allowing no more than 75% in this region to be plugged. These analyses also demonstrated acceptable results.

- RCS Pressure – Low-Low

  The LBLOCA analyses do not credit reactor trip during the initial blowdown. The only other trip function that is modeled is ES. In particular, the low-low RCS pressure signal actuates Low Pressure Injection (LPI) to provide cooling. If a significant asymmetry in plugging between the two OTSGs exists, the increase in hydraulic resistance will have an effect on the blowdown and resulting stored energy in the fuel pin. As a result, new LBLOCA analyses were performed.

  The analyses assumed 20% overall tube plugging, with 25% in the broken loop and 15% in the intact loop. The effect of the increased OTSG tube plugging was not reflected in a change to the plant setpoints; rather, the effect of plugging tubes was absorbed in the calculated peak clad temperature and maximum allowed linear heat rate limit. No change to the ES setpoints was found to be required.

- Reactor Building Pressure - High

  This function is not credited in the LOCA analyses. Therefore, there is not a need to revise this setpoint.

- RB Pressure – High-High

  This function is responsible for initiating Building Spray. The applicable Design Basis Accident for this function is the LBLOCA. However, the mass and energy release data were not recalculated for this project as the current data would conservatively bound the effects of OTSG tube plugging. The mass and energy release during blowdown is a function of the initial RCS inventory and the average temperature. The RCS average temperature will not be changed with increased levels of tube plugging. Additionally, tube plugging causes a small reduction in the initial inventory, which reduces the available mass and energy addition to the containment building. The existing blowdown mass and energy release data will remain conservative with respect to OTSG tube plugging, and this setpoint will therefore not require modification.

4. Impact on Emergency Feedwater Initiation and Control (EFIC) System

Emergency Feedwater controls (e.g., level, isolation) are provided by the EFIC system, whose functions are specified in ITS Table 3.3.11-1. The pertinent events are those which require secondary side cooling, such as a loss of feedwater and SBLOCA, and the limiting overcooling event, the MSLB.

- EFW Initiation

  The most pertinent events for this function are the loss of feedwater event and the SBLOCA. The loss of feedwater event relies upon timely actuation and adequate delivery of EFW for successful mitigation. The analysis sets the minimum required EFW flowrate, and does not as such credit the specific controls provided by EFIC. The only feature of EFIC credited in the analysis is the low OTSG level initiation of EFW. In conjunction with the 550 gpm minimum requirement, the analyses concluded that the acceptance criteria were met with 20 percent tube plugging.

  For the SBLOCA, the OTSG secondary level control is important for mitigation of the smaller size SBLOCA transients. EFW is initiated on loss of forced RCS flow or HPI actuation. Subcooling margin is lost early in the transient and it is assumed that, typically within 20 minutes, the operators raise the OTSG level control to the loss of subcooling margin setpoint. These actions were modeled in the SBLOCA analyses with 20% OTSG tube plugging, and therefore no changes in the control setpoints are required.

- EFW Vector Valve Control

  The most pertinent event for this function is the MSLB. Vector valve control or Feed-Only-Good-Generator, or FOGG, logic ensures that only the intact OTSG is being supplied with EFW. As stated previously, tube plugging is not accounted for in the MSLB. However, since FOGG is a function of the sensed pressure difference between the faulted and intact steam lines, and given the magnitude of secondary side depressurization, FOGG logic will continue to perform its function, regardless of the amount of tube plugging. Therefore, there is not a need to modify this function.

- Main Steam Line Isolation

  The most pertinent event for this function is the MSLB. However, Main Stream Isolation Valve (MSIV) actuation is a function of the sensed pressure in the faulted main steam line, and is thus is not dependent on tube plugging. Therefore, there is not a need to revise this function.

- Main Feedwater Isolation

  The most pertinent event for this function is the MSLB. However, like For the MSLB, the key EFIC features credited in the analysis are Main Steam Isolation Valve (MSIV) actuation and the Feed-Only-Good-Generator, or FOGG, logic. The FOGG logic ensures that only the intact OTSG is being supplied with EFW. As stated previously, tube plugging is not accounted for in the MSLB. However, given the secondary side depressurization magnitude of a MSLB, FOGG logic is expected to perform its function, regardless of the amount of tube plugging.

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15. 3N1118-3, "Crystal River Unit 3 - Staff Evaluation and Issuance of Amendment Re: Post-LOCA Boron Dilution Precipitation Prevention (TAC No. M99892)." November 30, 1998.

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18. Letter from the NRC to Entergy Operations Inc., "Issuance of Amendment NO. 159 To Facility Operating License NO. DPR-51 – Arkansas Nuclear One, Unit NO. 1 (TAC NO. M82123).

19. Letter from Entergy Operations Inc. to the NRC, "Technical Specification Change Request Concerning Relocation of the Variable Low Reactor Coolant System Pressure Trip Setpoint and Its Associated Protective Limits to the Core Operating Limits Report," dated April 29, 1996.

20. Letter from the NRC to Entergy Operations, Inc., 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.

21. Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." U.S. Nuclear Regulatory Commission. October 4, 1998.

22. General Public Utilities (GPU) Nuclear Letter 1920-99-20088, "Three Mile Island Nuclear Station Unit 1 (TMI-1), Operating License DPR-50 Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279, Core Protection Safety Limit." March 26, 1999.

23. Letter from NRC to GPU Nuclear, Inc., "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." August 19, 1999.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT G

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20% Steam Generator Tube Plugging

Draft Revised COLR Pages

DRAFT

Crystal River Unit 3, Cycle 13

1.0 Core Operating Limits

This Core Operating Limits Report for CR-3 Cycle 13 has been prepared in accordance with the requirements of Technical Specification Section 1.1 and 5.6.2.18. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. These methods are documented in BAW-10179PA, Rev. 3 "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses". The Cycle 13 limits generated using this methodology above are documented in BAW-XXXX *[TBD]*, "Crystal River Unit 3 Cycle 13 Reload Report."

The following limits are included in this report.

SL 2.1.1.1	AXIAL POWER IMBALANCE PROTECTIVE LIMITS
SL 2.1.1.2	AXIAL POWER IMBALANCE PROTECTIVE LIMITS
LCO 3.1.1	SHUTDOWN MARGIN
LCO 3.1.3	MODERATOR TEMPERATURE COEFFICIENT
SR 3.1.7.1	API/RPI POSITION INDICATION AGREEMENT
LCO 3.2.1	REGULATING ROD INSERTION LIMITS
LCO 3.2.2	AXIAL POWER SHAPING ROD INSERTION LIMITS
LCO 3.2.3	AXIAL POWER IMBALANCE OPERATING LIMITS
LCO 3.2.4	QUADRANT POWER TILT
LCO 3.2.5	POWER PEAKING FACTORS
LCO 3.3.1	REACTOR PROTECTION SYSTEM INSTRUMENTATION
SR 3.4.1.1	RCS PRESSURE DNB LIMITS
SR 3.4.1.2	RCS TEMPERATURE DNB LIMITS
SR 3.4.1.3	RCS FLOW RATE DNB LIMITS
LCO 3.9.1	REFUELING BORON CONCENTRATION

Crystal River Unit 3, Cycle 13

RCS DNB Pressure Limits

RCS loop pressure ≥ 2064 psig

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

These limits are referred to by SR 3.4.1.1

Crystal River Unit 3, Cycle 13

RCS DNB Temperature Limit

RCS Hot Leg Temperature ≤ 604.6°F (Cycle 13 limit).

To accommodate up to 20% equivalent OTSG tube plugging, RCS Hot Leg Temperature shall be ≤ 605.8°F (ITS limit).

These limits are referred to by SR 3.4.1.2

Crystal River Unit 3, Cycle 13

RCS DNB Flow Rate Limits

For Cycle 13, RCS total flow rate ≥ 139.7 E6 lb/hr with four RCPs operating, or ≥ 104.4 E6 lb/hr with three RCPs operating.

To accommodate up to 20% equivalent OTSG tube plugging, RCS total flowrate shall be ≥ 133.5 E6 lb/hr with four RCPs operating, or ≥ 99.7 E6 lb/hr, with three RCPs operating (ITS limit).

These limits are referred to by SR 3.4.1.3

Crystal River Unit 3, Cycle 13

RCS Variable Low Pressure Setpoint Equation

$P_{Trip} \geq (11.59 * T_{HOT} - 5037.8)$ psig

These limits are referred to by ITS Table 3.3.1-1, Item 5