Florida Power
A Progress Energy Company

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

Ref: 10 CFR 50.90

July 1, 2002
3F0702-08

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

Subject: Crystal River Unit 3 - Response to Request for Additional Information LAR #263, Revision 0, Relocation of Reactor Coolant System Parameters to the Core Operating Limits Report and 20 Percent Steam Generator Tube Plugging

References: 1. NRC to FPC letter, 3N0502-02, dated May 9, 2002 "Crystal River Unit 3 Request For Additional Information RE: Proposed License Amendment Request No. 263, Revision 0, Relocation Of Reactor Coolant System Parameters To The Core Operating Limits Report And 20 Percent Steam Generator Tube Plugging" (TAC NO. MB2499)

2. NRC to FPC letter, 3N0602-04, dated June 6, 2002, "Crystal River Unit 3 Request For Additional Information RE: Proposed License Amendment Request No. 263, Revision 0, Relocation Of Reactor Coolant System Parameters To The Core Operating Limits Report And 20 Percent Steam Generator Tube Plugging" (TAC NO. MB2499)

3. FPC to NRC letter, 3F0701-11, dated July 24, 2001, "License Amendment Request #263, Revision 0, Relocation Of Reactor Coolant System Parameters to the Core Operating Limits Report And 20% Steam Generator Tube Plugging"

4. FPC to NRC letter, 3F0602-06, dated June 5, 2002, "Crystal River Unit 3, Response to Request For Additional Information LAR #263, Revision 0, Relocation Of Reactor Coolant System Parameters to the Core Operating Limits Report And 20 Percent Steam Generator Tube Plugging"
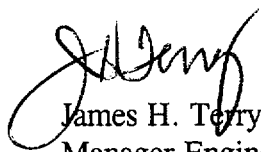
Dear Sir:

Florida Power Corporation (FPC) submits the additional information requested in References 1 and 2 concerning License Amendment Request #263, Revision 0 (Reference 3) and the previous response to Request for Additional Information (Reference 4). The responses to both requests for additional information are included in the attachment to this letter.

This letter makes no new regulatory commitments.

A001

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

Sincerely,

James H. Terry
Manager Engineering
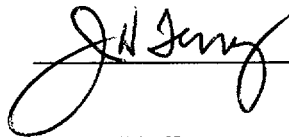Crystal River Nuclear Plant

JHT/pei

Attachment: Response to NRC Request for Additional Information

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

STATE OF FLORIDA

COUNTY OF CITRUS

James H. Terry states that he is the Manager Engineering, 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.

James H. Terry
Manager Engineering
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this _/5ᵗ_ day of _July_____, 2002, by James H. Terry.

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

Signature of Notary Public
State of Florida

_LISA A morris_____

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

Personally _X_   Produced
Known ____ -OR- Identification _____

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT

LICENSE AMENDMENT REQUEST #263, REVISION 0
Relocation Of Reactor Coolant System Parameters To The Core Operating
Limits Report And 20 Percent Steam Generator Tube Plugging

Response to NRC Request for Additional Information

Response To NRC Request For Additional Information

NRC Questions in letter dated May 9, 2002

1. **Explain how the originally licensed feedwater flow rate of 5.4E6 lb/hr was determined. Please include an explanation of the SG tube plugging percentage used in this determination.**

Response:

The originally licensed feedwater flow rate was determined as the feedwater flow rate required to transfer the total Nuclear Steam Supply System (NSSS) power (2568 MegaWatt thermal (MWt) of core thermal power plus 16 MWt for Reactor Coolant Coolant Pump (RCP) heat less makeup/letdown, ambient heat losses, etc.) across the steam generators and is based on the operating steam temperatures of the plant. This type of calculation is shown below, where:

Q = power, BTU/s or MWt; subscript 1 denotes the per Once-Through Steam Generator (OTSG) power
P = pressure, psia
T = temperature, °F
ρ = density, lb_m/ft^3
h = enthalpy, BTU/lb_m; subscripts *stm* and *fw* denote steam and feedwater phases
w = mass flow rate, lb_m/hr; subscripts 0 and 20 denote 0% and 20% plugging cases

$$Q = (2568 + 16) \text{ MWt} * 3413 \text{ Btu/(kW-h)} * 1000 \text{ (kWt/MWt)} * (1/3600) \text{ (hr/s)}$$

$$Q = 2.45E6 \text{ Btu/s}$$

On a per steam generator basis,

$$Q_1 = (2.45E6 \text{ Btu/s})/2 = 1.225E6 \text{ Btu/s}$$

$$\text{At } P = 925 \text{ psia and } T = 592°F, \ h_{stm} = 1250.6 \text{ Btu/lb}_m \text{ and } \rho = 1.79 \text{ lb}_m/ft^3$$

$$\text{At } P = 925 \text{ psia and } T = 450°F, \ h_{fw} = 430.4 \text{ Btu/lb}_m$$

The flow rate for the 0% plugging condition is,

$$w_0 = Q_1 / (h_{stm} - h_{fw})$$

$$w_0 = 1.225E6 \text{ Btu/s} / [1250.6 - 430.4] \text{ Btu/lb}_m = 1493.6 \text{ lb}_m/s$$
$$= 5.38E6 \text{ lb}_m/hr \approx 5.4E6 \text{ lb}_m/hr$$

Note that the original feedwater flow rate did not consider tube plugging. If steam generator tubes are plugged, the steam temperature will decrease resulting in a decrease in steam enthalpy and a required increase in feedwater flow to maintain full power. Thermal-hydraulic calculations show that for tube plugging of 20%, the steam temperature will remain above 582°F. The required feedwater flow thus becomes:

At $P = 925$ psia and $T = 582°F$, $h_{stm} = 1242.0$ Btu/lb$_m$ and $\rho = 1.83$ lb$_m$/ft^3

At $P = 925$ psia and $T = 450°F$, $h_{fw} = 430.4$ Btu/lb$_m$

The flow rate for the 20% plugging condition is,

$$W_{20} = Q_1 / (h_{stm} - h_{fw})$$

$$W_{20} = 1.225E6 \text{ Btu/s} / [1242.0 - 430.4] \text{ Btu/lb}_m = 1509.4 \text{ lb}_m/s = 5.43E6 \text{ lb}_m/hr$$

This represents a 1% increase in feedwater flow rate. However, although feedwater flow increases by 1%, the dynamic pressure (w^2/ρ) is effectively unchanged:

$$(5.38e6)^2/1.79 \approx (5.43e6)^2/1.83$$

2. **The feedwater flow rate to each SG is a safety system setpoint. This number is not specifically stated in the CR-3 Improved Technical Specifications. Explain how this value is controlled on a cycle-specific basis, and how the margin to the original licensed maximum value is determined.**

Response:

Since the current basis for operation did not include a provision for a specified limit on tube plugging, feedwater flow rate has been considered an operational value. However, maximum design limits are specified in the Crystal River Unit 3 (CR-3) design basis documentation. In order to implement these design limits, plant operating procedures impose a limitations on total feedwater flow. OP-103A, Startup Curves, includes the feedwater flow limit on curves for: Outlet Steam Pressure Versus Feedwater Mass Flow rate, Startup Level Versus Loop Feedwater Flow Rate, Loop Feedwater Mass Flow Rate Versus Power, and Feedwater Temperature Versus Loop Feedwater Flow. The limit will be reviewed as plugging levels change from cycle to cycle, as part of the normal reload processes.

3. **Explain in more detail the derivation of the cross-velocity used in the original flow-induced vibration calculations referenced on page 3 of Framatome 51-5000475-01.**

Response:

Flow loads on Once-Through Steam Generator (OTSG) tubes were originally based on tests on a scale model of the OTSG. The secondary side mean velocity flow conditions determined from testing, varied from tube-to-tube over the cross section of the OTSG. The maximum peak factor (ratio) from the mean velocity in each tube was 1.8 and occurred in the span approximately 30 inches above the 15th or top tube support plate. The highest flow load occurred at the 4th tube from the periphery on each side of the open lane, at which the mean cross flow velocity was 35.9 ft/sec. in the top span. The highest predicted mean cross flow velocity for tubes outside of the delta region by the lane or in the periphery of the tube bundle in the top span was 28.0 ft/sec. These values are based on a 5.4E6 lbm/hr OTSG flow rate.

The velocity and density distributions in the top span were based on the following assumptions:

❖ The steam density is 1.79 lbm/ft^3 and is uniform over the top span and over the entire cross section of the OTSG.

❖ The axial velocity distribution follows the same shape for all the tubes in a given span. Thus, the actual cross flow velocity for each tube was assumed to be equal to the same shape function multiplied by a mean velocity.

4. **Explain in detail the correlation between 20-percent tube plugging and the stated corresponding increased value in cross flow-velocity. Provide a description of the methodology used in this correlation.**

Response:

In the initial submittal, it was noted that 20% tube plugging was covered from a flow induced vibration (FIV) viewpoint because tube plugging does not cause an increase in the cross flow dynamic pressure (see response to Question 1). In addition, the FIV analysis in support of the proposed power uprate from 2544 MWt to 2568 MWt provided a 1% increase in feedwater flow rate (and hence cross flow velocity) to allow for 20% tube plugging. Since the cross flow velocities are governed by inertial effects rather than buoyant effects, the tacit assumption that the cross flow velocity will increase 1% as the feedwater flow rate increases by 1% is reasonable. The 1% increase in feedwater flow to account for 20% tube plugging is based on the expected reduction in steam temperature due to steam generator tube plugging (see response to Question 1).

5. **Attachment F, page 15, paragraph 4, to your letter of July 24, 2001, is very confusing. Since there is a specific maximum flow rate per generator previously analyzed, why is the asymmetric condition of 25/10 discussed? The paragraph also states that feedwater**

flow to the lesser plugged generator may need to be limited. This appears not to be relevant to the flow-induced vibration discussion in this section. Please clarify this. If this information is relevant to this section please explain the statement concerning the feedwater flow limitation to the lesser plugged SG for the worst case asymmetric distribution with 20 percent tube plugging.

Response:

This paragraph can be deleted from the submittal. The intent of the discussion was to alert the reader that in the event of asymmetric plugging, or for that matter asymmetric fouling, a situation could arise where one steam generator would be operating at its analyzed feedwater flow rate limit prior to achieving 100% power. If this should occur, then the licensee would have the latitude to revisit the FIV analysis to determine if refinements could be made to increase the feedwater flow rate limit and thus increase plant power.

NRC Questions in letter dated June 6, 2002

1. **RAI 1.d - FPC stated that experimental data from vessel model flow tests were used to correlate inlet nozzle flow asymmetry with core inlet velocity asymmetry. Identify and reference these specific test programs and provide justification for the applicability of the experimental data to CR-3.**

   Response:

   Results from vessel model flow tests, References 1 and 2, were used to define the core inlet velocity asymmetry due to asymmetric tube plugging. Normalized inlet flow distributions were measured for different reactor coolant pump combinations, including four-pump and three-pump flow. With three-pump flow, a large difference between the four inlet nozzle flow rates exists. However, the vessel model flow tests showed that this difference in flow distribution was significantly attenuated at the core inlet. A correlation based on these test data was developed. Then, using the maldistribution of inlet nozzle flow rates resulting from asymmetric tube plugging (based on the CR-3 FSPLIT RCS hydraulics model), the core inlet velocity penalty was determined. Note that the asymmetry in inlet nozzle flow for three-pump operation is much larger than the asymmetry in inlet nozzle flow for asymmetric tube plugging. The test results used in the calculation are valid for a B&W 177 fuel assembly (FA), lowered loop vessel; the CR-3 reactor vessel falls into this design category. Thus, the test results are valid for CR-3.

   References

   1. NPGD-TM-257, Analysis of Core Inlet Flow Factors Obtained From 177-Fuel-Assembly VMFT.

2. ARC Report 4668, Model Tests of a 177 Fuel Assembly – Two Loop PWR Progress Report No. 5 – Oconee Redesign, Vol. 1, February 1974.

2. **RAI 1.e - FPC bases the Reactor Coolant System flow reduction analysis on a generic Babcock & Wilcox (B&W) Owners Group study performed to evaluate 20 percent tube plugging for B&W designed 177 Fuel Assembly plants. Is the generic study referred to in response to this question the same one referred to in responses to questions 4.d, 6 (for Station Blackout (SBO) and Anticipated Transient Without Scram), 10.a and 12.a? FPC relies on several analyses performed as part of this study for justification for CR-3 amendment request. Provide the reference to B&W 51-5009660-01 and describe the information contained and how the specific information pertains to CR-3. Identify if there have been any changes in the design of CR-3 since the time of this generic study which would invalidate the results, and provide adequate justification for continued acceptability from the results of the study?**

    Response:

    Yes, it is the same report. However, although some discussions in the B&WOG report are generic to the B&W-designed plants, certain analyses were performed specifically for CR-3, i.e., the startup accident and the loss of main feedwater accident analyses. The system response to the single locked RC pump rotor transient was, however, a generic analysis that is applicable to CR-3. The RCPs modeled in the analysis are the same as those at CR-3. The initial core power level used in the analysis was 102.1% of 2568 MWt, which bounds the current rated thermal power level of 2544 MWt and the planned power uprate to 2568 MWt submitted in our letter 3F0602-05, dated June 5, 2002. Note that the calculation of the departure from nucleate boiling (DNB) response was based on the CR-3 specific core design.

    These analyses conform to the NRC-approved guidance provided in Appendix A of BAW-10193P-A, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors," Framatome Technologies, Lynchburg, Virginia, February 2000. There have been no changes to the CR-3 plant since these analyses were performed that would invalidate the analyses. As a result, the system analysis is conservative and bounding for CR-3.

3. **RAI 4.b - Regarding the expected increase in clad oxidation levels, FPC states that the highest burnup pins are approaching the clad oxide limit but that acceptable results have been demonstrated by the use of pin-specific power histories. Provide a comparison of clad oxidation results assuming 0 percent and 20 percent tube plugging. Also, provide more detail regarding the methodology used to determine pin specific power histories.**

For the present CR-3 Cycle 13, the fuel rod corrosion predictions are based on the methodology approved in topical BAW-10186P-A Rev. 1, "Extended Burnup Evaluation," for the maximum burn-up pin in each sub-batch. The corrosion predictions for the maximum burn-up pins are in the range of 84 to 99 microns. These predictions use a conservative reduction in flow rate to bound the flow rate resulting from 20% OTSG tube plugging.

If the same analyses were conducted without the flow rate reductions needed to model the 20% OTSG tube plugging, the predictions of end-of-cycle corrosion levels for the highest burn-up pins would be in the range of 74 to 89 microns. The reduced flow rates result in an increase in cladding temperature during the cladding's exposure during the cycle. The increased temperature results in an increased prediction of fuel rod oxide level.

The methodology for the prediction of the corrosion level has been approved in the aforementioned topical report. The end-of-cycle corrosion predictions are made for each of the highest predicted burn-up pins in each sub-batch of fuel within the core. These cycle specific evaluations of predicted fuel rod corrosion use bounding core parameters. If bounding maximum pin power histories for each of the high burn-up pins are available, these are used to provide a conservative end-of-cycle oxide prediction. If these bounding predictions are too conservative, then the methodology allows for the actual pin history for each of the highest burn-up pins to be utilized for the predictions. In this case, the actual pin power history is taken from the nuclear design code and used as an input to the KOROS/COROS2 code. These evaluations are performed early in the core design process and therefore allow for changes to the cycle design, if the fuel rod corrosion limit cannot be met.

4. **RAI 4.d - Provide a reference to the generic analysis used to demonstrate that no saturation occurs in the guide tube assembly hold-down springs, guide tubes and spacer grids. Provide the reference to B&W 51-5009660-01 and describe the information contained and how the specific information pertains to CR-3. Identify if there have been any changes in the design of CR-3 since the time of this generic study which would invalidate the results, and provide adequate justification for continued acceptability from the results of the study?**

Response:

In the RAI response to Question 4, the guide tube boiling methodology was described as being bounding for all B&W 177-FA type plants. This bounding methodology was used to perform a guide tube boiling scoping study which was documented in a generic B&W Owners Group report, 51-5009660-01, "Evaluation of Extended Tube Plugging Limits for the Once-Through Steam Generator," dated March 2001. The bounding nature of the generic study ensured that the results were applicable to CR-3. Conservatisms include the fact that the analysis was performed for 2772 MWt, which bounds a 2568 MWt power

condition. Additionally, core statepoint uncertainties were treated deterministically, and maximum design peaking was assumed. There have not been any changes to the CR-3 plant since these analyses were performed that would invalidate the conclusion of the generic report, which is that long term bulk boiling will not occur in the guide tubes.

5. **RAI 6 - For the Loss of Flow transients (Four and One Reactor Coolant Pump (RCP) Coastdown, Locked Pump Rotor) FPC stated that a new system analysis was not required and only reanalyzed the Departure from Nucleate Boiling (DNB) portion of the event. Justify the assumption that the normalized core inlet flow profile does not change as a result of 20 percent tube plugging. Include an assessment of the impacts of a change in the inlet flow profile on the acceptance criteria for these events as listed in the CR-3 Updated Final Safety Analysis Report. Also, explain the difference between minimum design flow rate and minimum DNB flow rate, and specify the reduction in flow rate used in the analysis. It is not clear which flow rate curve was actually used in the analyses because initially FPC states the minimum design flow rate is used, but later states the minimum DNB flow rate is reduced. This needs clarification.**

Response:

To support increasing the allowable level of SG tube plugging, a study was performed that compared the normalized core inlet flow response for the 4-pump, 2-pump, and 1-pump coastdown transients. For each transient, different levels of SG tube plugging were modeled, including cases from zero to 30 percent as well as an asymmetrical plugging case. The results of this study showed that there was no appreciable difference in the total core inlet flow prediction during the time that the minimum DNB ratio would be reached. Therefore, no new system analyses for the flow coastdown events were required to support higher levels of SG tube plugging.

The minimum design flow rate is based on the rated flow for the RC pumps, 88,000 gpm/pump or a total flow of 352,000 gpm. During initial start-up testing, the actual RCS flow was found to be much greater than the original design conditions. To take advantage of the difference between the actual and the design flow, a larger RCS flow rate was established for use as a minimum limit in the DNB calculations. The minimum DNB flow rate that has been used in the statistical-based DNB analyses is 109% of the rated flow, or 383,680 gpm. To account for 20 percent SG tube plugging, the minimum DNB flow rate is reduced by 4.5% to 367,840 gpm.

6. **RAI 6 - FPC states that the consequences of the loss of all alternating current (ac) power accident are bounded by the loss of main feedwater accident because the net energy addition to the primary coolant during the loss of ac power transient is less due to the RCP's tripping immediately upon a loss of power. Identify the specific consequences referenced? Also, although the energy addition to the primary coolant**

may be less for the loss of ac power transient, the RCP trip produces a countering effect due to loss of forced flow. Quantify the impact of the loss of forced flow on DNB Ratio due to the RCP trip.

Response:

The RCPs are assumed to remain operating following a loss of main feedwater event, increasing the heat contribution to the reactor coolant. As such, RCS pressure, RCS temperature, and the pressurizer level response during the Loss of AC Power (LOAC) transient are bounded by the loss of main feedwater (LOFW) accident. In addition, the emergency feedwater flow requirement for the LOAC is also bounded by the LOFW because of the additional energy from the operating RCPs.

The NRC is correct in that the minimum DNB response for the LOAC power transient is not bounded by the LOFW because the RCPs are operating. However, the minimum DNB ratio for the LOAC would be bounded by (higher than) the DNB ratio for the 4-RCP coastdown transient due to the power and RC flow response from the loss of all RCPs. For the LOAC power event, the reactor and the RC pumps are tripped on loss of power. For the 4-pump coastdown, forced flow is lost immediately but the RPS instrument processing time delays the reactor trip, causing a lower minimum DNB ratio than would be calculated for the LOAC power event.

7. **RAI 6 - Discuss the impacts of 20 percent tube plugging on Condensate Storage Tank inventory required and available for the 4-hour coping period for an SBO event.**

Response:

Emergency Feedwater (EFW) is supplied by the Emergency Feedwater Tank (EFT-2) during a station blackout transient. EFT-2 level is maintained at 150,000 gallons per ITS 3.7.6. The required emergency feedwater flow, and hence inventory, is determined based on the residual core decay heat which is a function of core power, and not SG tube plugging. Although the overall effective heat transfer area will be reduced as the number of SG tubes removed from service increases, there is still adequate heat removal capability through the in-service tubes to ensure that the core is protected throughout the 4-hour coping period. The increase in the amount of SG tube plugging has no effect on Condensate Storage Tank (CST) inventory. The CST is a backup supply of EFW but is not required for the four-hour station blackout coping period.

8. **RAI 8 - Provide a quantitative assessment of 20 percent tube plugging on the Feedwater Line Break accident results. This is not addressed in either the original submittal or the RAI responses.**

Response:

No specific analysis was performed to justify increased levels of SG tube plugging for the feedwater line break accident. This was based on a comparison of analysis results for the loss of main feedwater (LOFW) accident with and without SG tube plugging. Specifically, the LOFW accident is used to establish the minimum emergency feedwater (EFW) flow requirement for CR-3. A comparison of the results of the LOFW transients with and without SG tube plugging demonstrated that no increase in the minimum EFW flow was required to meet the event acceptance criteria of peak RCS pressure and peak pressurizer level. This comparison shows that the reduction in heat transfer area can be easily accommodated with no challenge to the RCS.

The feedwater line break analysis presented in the CR-3 Final Safety Analysis Report (FSAR) was performed with conservative assumptions regarding the time that the "affected" SG blows down and the time that the "unaffected" SG is boiled dry. Increasing the number of SG tubes that are removed from service due to plugging will not be significantly different from the current analysis of record. Increasing levels of tube plugging causes the boiling length and the inventory in the SGs to increase to match core and RC pump heat without changing the nominal operating conditions within the RCS. As a result, the blowdown time for the "affected" SG may increase slightly, but the boil-off time for the unaffected SG will increase due to the higher initial SG inventory. Therefore, the results for the feedwater line break with 20% tube plugging may be somewhat better, but no worse than currently presented in the CR-3 FSAR.

9. **RAI 13 - FPC states that the new Improved Technical Specification minimum Reactor Coolant System (RCS) loop pressure limit of 2064 psig provides assurance that the nominal 2200 psia core exit pressure is maintained. FPC provided a discussion of the methodology used to correlate the 2064 psig measured pressure to the nominal 2200 psia core exit pressure. Additional detail is required:**

   A. **Provide the location of the RCS pressure instruments.**

   B. **Provide the values for the pressure measurement uncertainty and the conservative representation of the delta-P core exit to pressure tap. A brief discussion of the actual calculation methodology would be useful.**

   C. **Provide a more detailed basis for the change in the minimum RCS loop pressure limit from 2061 to 2064 psig. Include a detailed discussion of the calculations / computer programs used to determine this final value.**

Response:

RCS pressure-sensing instruments that provide input to the Reactor Protection System (RPS) are attached to the hot legs. They are approximately 44 feet above the core exit elevation. The table below provides clarifications of the various pressure values found in different CR-3 references.

Reference	Value	Description
FSAR Chapter 14 Table 14-2, "Thermal-Hydraulic Design Conditions"	2185 psig (2200 psia)	This is a minimum core exit pressure value from which the COLR value can be derived. Adjustments are applied to this value for transient DNB analyses, and for deriving the COLR limit (as shown below).
ITS 3.4.1 Bases	2120 psig (2135 psia)	This value is used in the DNB analysis, and reflects a - 65 psid uncertainty adjustment as applied to 2185 psig core exit value. Additionally, the current ITS Bases uses this value, along with the zero plugging RCS flow rate, in a historical calculation of core exit temperature. The discussion of this calculation is proposed for removal from the ITS Bases (in our July 24, 2001 letter), because the value differs from the revised flow rate and could be a source of confusion. In its place is a discussion that better reflects the basis of the minimum DNB pressure limit.
Nominal Hot Leg Pressure	2155 psig (2170 psia)	This value is the generic, B&W 177-FA type plant, nominal operating pressure point as measured at the hot leg, and is applicable to all possible RCP combinations.
COLR	2064 psig (2079 psia)	This is the new minimum RCS pressure limit as sensed at the hot leg for the 20% tube plugging condition. It includes assumed uncertainties and corrections (as shown below). The limit protects both four- and three-RCP operation. As stated in the ITS, when three RCPs are operating, the limit is applied to the loop with two RCPs in operation. By preserving this limit, the accident analysis assumption for RCS pressure is not violated. This limit is to be controlled in the cycle's Core Operating Limits Report (COLR).

The minimum RCS loop pressure limit is derived by reducing the minimum core exit pressure by three components. In summary:

Minimum Core Exit Operating Pressure = 2200 psia

Applied Adjustments:

- ❖ Pressure Measurement Uncertainty = - 65 psid
- ❖ Conversion from psia to psig conversion = -14.7 psi
- ❖ The ΔP Correction for elevation and frictional and form losses from the core exit to the hot leg pressure tap. This term varies with flow (i.e., tube plugging, RCP operation). For consideration of three-RCP operation, its value reflects the OTSG loop with both RCPs operating.

0% Tube Plugging

For 0% tube plugging, the ΔP correction for elevation and frictional and form losses from the core exit to the hot leg pressure tap is 59.9 psid. As such, the existing ITS limit for 0% tube plugging is computed as follows:

2200 psia – 65 – 14.7 – 59.9 psid = 2060.4 psig.

Note that the existing ITS limit is 2061.6 psid. A higher value is more conservative with respect to DNBR.

20% Tube Plugging

As the level of tube plugging in the plant increases, both the flow and the actual value of the ΔP correction will decrease. Therefore, to ensure that a conservative measured pressure limit is set, the ΔP correction term must be based on a minimized flow. The minimized flow associated with 20% tube plugging was used in an evaluation of the existing measured pressure limit. For 20% plugging, a value of 56.6 psid is applicable. Thus, the new ITS limit for 20% tube plugging is:

2200 psia – 65 – 14.7 – 56.6 psid = 2063.7 psig, or 2064 psig (conservative, rounded up)

As these results show, the evaluation indicated that the measurement limit needed to be increased slightly to remain conservative. The minimum core exit pressure and nominal hot leg pressure have not changed.

10. **To show that the referenced generically approved LOCA analysis methodologies apply specifically to CR-3, the staff requests that FPC provide a statement that CR-3 and its vendor have ongoing processes which assure that LOCA analysis input values for peak**

cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

Response:

CR-3 and its vendor have ongoing processes which assure that LOCA analysis input values for peak clad temperature-sensitive parameters bound the as-operated plant values for those parameters.