

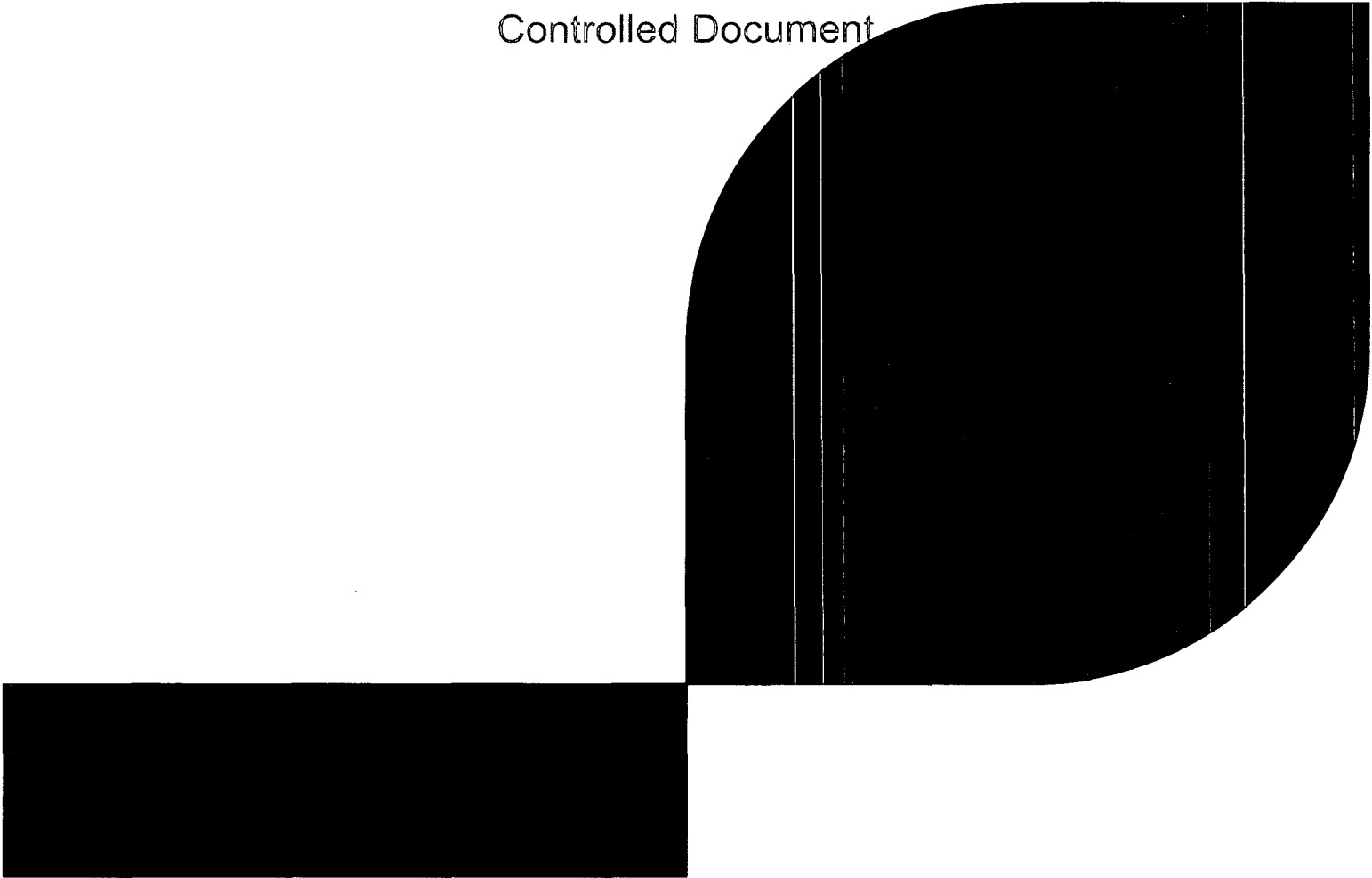
FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT D

**ANP-3195(NP), REVISION 0,
RESPONSES FOR CRYSTAL RIVER UNIT 3 EPU LICENSING
AMENDMENT REPORT NRC REACTOR SYSTEMS BRANCH
REQUESTS FOR ADDITIONAL INFORMATION
(NON-PROPRIETARY)**



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Report NRC Reactor Systems Branch Requests for Additional
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Nomenclature

ANO-1	Arkansas Nuclear One Unit 1
AOO	Anticipated Operational Occurrence
B&W	Babcock and Wilcox
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CR-3	Crystal River Unit 3
DB-1	Davis Besse Unit 1
DHRS	Decay Heat Removal System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EFIC	Emergency Feedwater Initiation and Control
EFPY	Effective Full Power Years
EFW	Emergency Feedwater System
EPU	Extended Power Uprate
FSAR	Final Safety Analysis Report
FWLB	Feedwater Line Break
HZP	Hot Zero Power
ICS	Integrated Control System
ITS	Improved Technical Specifications
HPI	High Pressure Injection [System]
LAR	Licensing Amendment Request
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LONF	Loss of Normal Feedwater
LTOP	Low-Temperature Overpressure Protection
MAP	Maximum Allowable Peaking
MFW	Main Feedwater
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NRC	United States Nuclear Regulatory Commission
OTSG	Once Through Steam Generator
P/T	Pressure / Temperature
PCT	Peak Clad Temperature
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RIR	Reactivity Insertion Rate

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Nomenclature (continued)

ROTSG	Replacement Once Through Steam Generator
RPS	Reactor Protection System
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss of Coolant Accident
SRXB	NRC Reactor Systems Branch
TMI-1	Three Mile Island, Unit 1
TR	Technical Report
TS	Technical Specification



1.0 INTRODUCTION

Duke Energy's Licensing Amendment Request (LAR) submittal for the Crystal River 3 (CR-3) extended power uprate (EPU) has resulted in requests for additional information (RAI) from the United States Nuclear Regulatory Commissions (NRC) Reactor Systems Branch (SRXB) (Reference [1]).

The NRC has transmitted a total of 36 RAIs in Reference [1]. Of these, six will require additional analysis in order to respond and eight are the scope of CR-3. This document has been prepared to record the response of AREVA NP Inc (hereafter AREVA) to the remaining 22 RAIs.

2.0 REQUEST FOR ADDITIONAL INFORMATION RESPONSES

The following sections provide a listing of the NRC SRXB RAIs and AREVA's response to those RAIs. The numbering of the third tier sub-sections and below is consistent with section numbering of the CR-3 EPU LAR submittal.

2.1 RAIS Related to LAR Section 2.8.4.3

2.8.4.3 Overpressure Protection During Low Temperature Operation

2.8.4.3.2 Please verify that there were no changes to lower mode mass and energy input sources that would require revisiting the LTOP relief system capacity.

Response:

The limiting mass and energy input source is a failed open make-up valve. There have been no changes that would impact this being the limiting LTOP challenge.

The LTOP events that are considered are as follows:

- Erroneous actuation of the high pressure injection (HPI) system

This would be the most limiting mass addition, but per Improved Technical Specifications (ITS) & procedures HPI is deactivated in the lower modes where LTOP is a concern – unchanged for EPU.

- Erroneous opening of the core flood tank discharge valve

Per ITS & procedures core flood tanks are isolated in the lower modes where LTOP is a concern – unchanged for EPU.

- Erroneous addition of nitrogen to the pressurizer



Limited by plant equipment which limits nitrogen pressure to 150 psig.

- Makeup control valve (makeup to the reactor coolant system [RCS]) fails full open

Limiting event – no modifications made that would affect the mass addition due to a failed open makeup valve.

- All pressurizer heaters erroneously energized

Slow transient that is bounded by the makeup control valve failure – unchanged for EPU.

- Temporary loss of the Decay Heat Removal System's (DHRS) capability to remove decay heat from the RCS

Slow transient that is bounded by the makeup control valve failure – unchanged for EPU.

- Thermal expansion of the RCS after starting an reactor coolant pump (RCP) due to stored thermal energy in the steam generator.

This event results in a finite increase in pressure that is less than the margin between the Appendix G and LTOP limits and is not power-level dependent. Because of the presence of a pressurizer bubble, this event is much less severe than at other pressurized water reactors.

2.2 RAIS Related to LAR Section 2.8.5.1

2.8.5 Accident and Transient Analyses

2.8.5.1 Increase in Heat Removal by the Secondary System

2.8.5.1.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

2.8.5.1.1.1 Provide an evaluation of events in this category relative to modifications to secondary heat removal capability. Include modifications to normal operational systems, such as main feedwater and main steam systems, as well as engineered safety systems, such as the proposed emergency feedwater initiation and control system.



Response:

As discussed in the LAR, a number of design modifications are proposed to support the EPU. The details of the major plant modifications are provided in Appendix E of the Technical Report (TR). The modifications to be installed in Phase 3 of the EPU project include:

1. Turbine (High Pressure and Low Pressure) Replacements (TR Appendix E Section 1.2.1)
2. Deaerator Re-Rate and Bypass Line Installation (TR Appendix E Section 1.2.2)
3. Condensate and Feedwater System Enhancements (TR Appendix E Section 1.2.3)
 - a. Condensate Pump Motor and Control Valves (TR Appendix E Section 1.2.3 A)
 - b. Feedwater Booster Pumps and Main Feedwater Pumps (TR Appendix E Section 1.2.3 B)
 - c. Feedwater Heater Replacement (TR Appendix E Section 1.2.3 C)
 - d. Replacement of Motor Operated Valves (TR Appendix E Section 1.2.3 D)
4. ICS [Integrated Control System] Scaling & Function Curves and Other Values Exiting 17R (TR Appendix E Section 1.2.4)
5. Makeup Tank Bypass (TR Appendix E Section 1.2.5)
6. Structural Support Improvements (TR Appendix E Section 1.2.6)
7. Emergency Feedwater Flow Increase Implementation (TR Appendix E Section 1.2.7)
8. Reconciliation of Replacement Steam Generator (TR Appendix E Section 1.2.8)
9. Low Pressure Injection Cross-Tie/Hot Leg Injection Implementation (TR Appendix E Section 1.3.1)
10. Enhanced Secondary Cooldown Capability Implementation (TR Appendix E Section 1.3.2)
11. Inadequate Core Cooling Mitigation System (TR Appendix E Section 1.3.3)
12. High Pressure Injection System Resistance (TR Appendix E Section 1.3.4)

The majority of the plant modifications noted above result in improved operating and analytical margins. Each modification has been developed in accordance with the CR-3 design change process. This includes the requirement to evaluate expected performance and assess the potential for introducing new failure modes and unintended consequences.

Specifically, modification Items 1, 2, 3, 4, 5, 6 and 8 are not associated with response to any events of this category and do not have any adverse impact on the Increase in Heat Removal by the Secondary System event consequences. All of these modifications have been developed per the CR-3 engineering change procedure and are being implemented under the 50.59 evaluation process.



The plant modification discussed in Item 7 increases the minimum required emergency feedwater (EFW) flow rate to ensure adequate margins associated with the loss of normal feedwater (LONF) event. However, the maximum flow rate is not impacted and does not change. Also, the logic within emergency feedwater initiation and control (EFIC) that establishes the level setpoint and the fill rate does not change. Consequently, increasing the minimum flow rate does not adversely impact the Increase in Heat Removal by the Secondary System event consequences. This plant modification, like the other items listed above, is being implemented under the 50.59 evaluation process.

The plant modifications discussed in Items 9 and 10 are specifically designed to increase the margin to the Peak Clad Temperature (PCT) limit for certain small break loss of coolant accident (SBLOCA) events. With respect to the Increase in Heat Removal by the Secondary System event consequences, only the plant modifications proposed in Item 10 have the potential to adversely impact the event consequences. The impact of this plant modification is encompassed by the limiting overcooling event discussed in the second paragraph of the response to RAI 2.8.5.1.1.2. CR-3 LAR #309 (EPU) provides the licensing basis for implementing the plant modifications associated with Items 9 and 10.

The plant modification discussed in Item 11 provides the initiation signal for the plant modifications listed for Items 9 and 10. This plant modification also automates an existing manual operator action to trip the reactor coolant pumps. An initiation signal will not be generated during an Increase in Heat Removal by the Secondary System event because a sustained (~8 minutes) loss of subcooling margin in conjunction with inadequate HPI flow will not exist. CR3 LAR #309 (EPU) provides the licensing basis for implementing the plant modification associated with Item 11.

The plant modification discussed in Item 12 allows more HPI flow to the RCS. For Increase in Heat Removal by the Secondary System events that have HPI initiation, a reactor trip on Low RCS Pressure will occur. This protective action ensures that the event consequences are bounded by the limiting overcooling event discussed in the response to RAI 2.8.5.1.1.2. CR3 LAR #309 (EPU) provides the licensing basis for implementing the plant modification associated with Item 12.

2.8.5.1.1.2 Provide information to demonstrate that, at planned EPU conditions, events of this category remain non-limiting, such that planned EPU modifications would not, for example, create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report or result in more than a minimal increase in the consequences of a malfunction of a system, structure, or component important to safety, two criteria that, if satisfied, would cause otherwise cause the proposed modifications to require a license amendment.

**Response:**

The Decrease in Feedwater Temperature and Increase in Feedwater Flow events were evaluated using conservative boundary conditions relative to the expected values for the EPU after factoring in the modifications to secondary heat removal capability. As discussed in the TR, the acceptance criteria for RCS pressure and departure from nucleate boiling ratio (DNBR) were met for all transients. In fact, the peak RCS pressure remained below the reactor protection system (RPS) High RCS Pressure trip setpoint in all instances. Since the reactivity insertion rates (RIRs) associated with the feedwater malfunction transients fall within the range of RIRs for other moderate frequency events such as the uncontrolled rod withdrawal events described in Sections 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, and 2.8.5.4.2, Uncontrolled Control Rod Assembly Withdrawal at Power, the RCS pressure responses for the feedwater malfunction transients are bounded by the uncontrolled rod withdrawal events. With respect to DNBR, the feedwater malfunction transients are bounded by the limiting overcooling event, as described in Section 2.8.2, Nuclear Design, and RAI 2.8.5.1.1.3.

The Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve events are not specifically analyzed because the consequences for these events are encompassed by the evaluation of the limiting overcooling event. Hence, the consequences for the Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve transients are bounded by the consequences for the limiting overcooling event, as described in Section 2.8.2, Nuclear Design, and RAI 2.8.5.1.1.3. The technique employed to define the limiting overcooling event ensures that the proposed modifications to the secondary heat removal capability are factored into the solution.

The above response in combination with RAI 2.8.5.1.1.1 demonstrates that at the planned EPU conditions, the Increase in Heat Removal by the Secondary System events remain non-limiting. Furthermore, each modification will be finalized and implemented in accordance with the CR-3 design change process to meet the applicable design and licensing basis requirements unless specifically addressed in the LAR. This process ensures that the planned EPU modifications do not create the possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report (FSAR) or result in more than a minimal increase in the consequences of a malfunction of a system, structure, or component important to safety.

- 2.8.5.1.1.3** Relative to the information contained in the TR, provide additional and more specific information concerning axial offset limit determination. Describe the limiting overcooling events that are specifically analyzed, and provide information to demonstrate that the increase in secondary heat removal events remain bounded by other anticipated operational occurrences (AOOs) that are within the CR-3 current licensing basis.



Response:

The limiting overcooling events that are specifically analyzed are discussed first, followed by presentation of the additional information concerning related axial offset limit determination.

Discussion of Overcooling Transients

The overcooling transient represented by a small steam line break is evaluated as part of the licensing basis for the CR-3 reload process. This event maximizes the secondary side overcooling to the point where the indicated core power, as measured by the de-calibrated ex-core detectors, stabilizes just below the high neutron flux reactor trip setpoint. Actual core power at these conditions is calculated to be higher than the high neutron flux reactor trip setpoint value. The high core power level of this event bounds the AOOs of the Increased Heat Removal by the Secondary System category.

The evaluation described above addresses the potential for an increased neutron measurement error during overcooling transients. This transient-induced measurement error is in addition to the normal instrumentation error used for the accident analyses. The transient-induced neutron measurement error is caused by an increase in neutron attenuation between the core and the power range out-of-core neutron detectors due to the presence of colder, i.e., higher density, coolant in the downcomer region. Since the detector calibration is affected by the downcomer fluid density during normal operation, the transient-induced increase in neutron attenuation results in a measured power lower than the actual core power. For transients that rely upon the nuclear instrumentation overpower trip for core protection, the neutron power measurement error may delay a reactor trip at a time when core power levels are in fact elevated.

The reactor coolant system response evaluation for the CR-3 EPU included a spectrum of cases run with varied reactivity parameters (moderator and Doppler coefficients, beta effective). For each set of reactivity parameters the break size was systematically reduced until the break size was small enough to avoid RPS or EFIC system trips. This produced limiting results for each set of reactivity parameters. In addition to varying the reactivity parameters, the impact of modeling control systems such as pressurizer heaters and makeup was included. The result of the numerous analyses performed was a matrix of different cases, each with its own set of reactor coolant system statepoints (core power, core exit pressure, core inlet temperature, core inlet flow). From this matrix, a bounding set of statepoints was developed for use in evaluating the limiting condition for operation (LCO) offset limits for acceptability. The bounding statepoint set is used in the development of the LCO axial offset limits, as discussed below.



Discussion of Axial Offset Limits

NRC-approved topical report BAW-10179P-A, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses* (Ref [2], Section 5.3.6) describes how the effect of overcooling transients on core power distribution is evaluated to ensure that the core safety limits are protected. Since the overcooling transient may cause reactor thermal power to exceed design overpower without a reactor trip, margins to centerline fuel melt, transient cladding strain, and steady-state departure from nucleate boiling (DNB) peaking limits are evaluated.

In the TR, Table 2.8.2-3 of Section 2.8.2, Nuclear Design, provides a subset of the LCO axial offset limits for the CR-3 EPU that are referred to in the *Increase in Feedwater Flow* section and *Increase in Steam Flow* section of Section 2.8.5.1.1.2 of the TR. For the EPU LAR evaluation, offset limits were calculated only at overpower conditions because those conditions produce the smallest margins to the power peaking acceptance criteria and are sufficient to gauge the acceptability of the EPU core designs by comparing them to the offset limits at similar overpower conditions for previous CR-3 cycles.

The LCO offset limits are calculated on a cycle-specific basis using the methodology described in BAW-10179P-A (Ref [2]). The initial (or base) LCO offset limits at the EPU conditions were calculated based on peaking margins to LOCA and Initial-Condition DNBR acceptance criteria. Then, additional evaluations were performed to determine whether the initial LCO offset limits needed to be restricted to accommodate the overcooling transient, which results in temperature-induced neutron flux errors due to reactor vessel downcomer cooling described in Section 2.8.5.1.1.2 of the TR. As noted above, these temperature-induced neutron flux errors potentially allow the reactor power to increase above the design RPS overpower power level without an automatic RPS trip. In order to ensure the initial conditions for a potential overcooling transient are restricted so that adequate margins to centerline fuel melt kW/ft limits, transient cladding strain kW/ft limits, and steady-state DNBR maximum allowable peaking (MAP) limits are preserved for the event, the LCO offset limits are adjusted based on this process.

The smallest LCO offset limits for the EPU cycles that were analyzed as described in Table 2.8.2-3 of TR Section 2.8.2 are for Cycle 18: -14.3 percent and +20.0 percent



offset. The initial LCO offset limits based on loss of coolant accident (LOCA) and Initial Condition DNBR criteria are not shown in Table 2.8.2-3 but were calculated to be [] The LCO positive offset limit was restricted to +20.0 percent offset to limit the axial peaks used in the AREVA Thermal-Mechanical analyses to meet the acceptance criteria for end-of-life pin pressure. The LCO negative offset limit was restricted to -14.3 percent offset in order to provide acceptable peaking margins for transient cladding strain for the overcooling transient.

Additional information on power distribution analysis and axial offset limits was provided in the responses to RAI-8 and RAI-9 in ANP-3120(P) (Ref [3]).

2.8.5.1.1.4 Provide an evaluation of the effects of an untripped overpower transient relative to fuel cladding strain acceptance criteria.

Response:

For the evaluations described in the EPU TR, cladding strain linear heat rate (kW/ft) limits for the CR-3 EPU LAR were calculated using the methodology described in the NRC-approved topical reports BAW-10162P-A, *TACO3-Fuel Pin Thermal Analysis Computer Code* (Ref [4]) and BAW 10184P-A, *GDTACO, Urania-Gadolinia Fuel Pin Thermal Analysis Code* (Ref [5]). Application of these kW/ft limits ensures that the 1 percent cladding strain acceptance criterion is not exceeded. The cladding strain kW/ft limits (and the centerline fuel melt kW/ft limits) used in the CR-3 EPU analysis include reductions necessary to account for thermal conductivity degradation with burnup. The COPERNIC code, as described in the NRC-approved topical report BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code* (Ref [6]) has been approved by the NRC for use in cycle-specific licensing analyses and will be used for the reload licensing for the CR-3 EPU cores. The COPERNIC code includes the effects of fuel thermal conductivity degradation with burnup. Additional information on how the TACO3 code was used in the CR-3 EPU analysis for the LAR and how the COPERNIC codes will be used was provided in the responses to RAI-1 and RAI-14 in ANP-3120(P) (Ref [3]).

The overcooling transient that was discussed in Section 2.8.5.1.1.2 of the TR and in the response to RAI 2.8.5.1.1.3 may produce an untripped overpower transient. As described in the response to RAI 2.8.5.1.1.3, margins to transient cladding strain kW/ft limits, as well as margins to centerline fuel melt kW/ft limits and steady-state DNBR MAP limits, are evaluated on a cycle-specific basis to ensure that the core operating limits specified in the Core Operating Limits Report (COLR) provide core safety limit protection. The response to RAI 2.8.5.1.1.3 describes how the transient cladding strain acceptance criterion is protected for the overcooling transient that is analyzed as part of the cycle-specific reload safety evaluations. The cladding strain kW/ft limits used in the overcooling transient analysis ensure that the 1 percent cladding strain criterion is not exceeded during the overcooling transient.



2.3 RAIS Related to LAR Section 2.8.5.2

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulator Failure

2.8.5.2.1.1 The TR, in this and other sections, describes the reactor trip on high pressure as follows: "Reactor trip was modeled to occur on a nominal high RCS pressure setpoint plus uncertainty (2400 pounds per square inch absolute (psia))." Please clarify whether 2400 psia is the nominal value or the model value. Also, please discuss the relationship between the nominal value, the modeled value, and the setpoint values contained in the TSs.

Response:

The RELAP5/B&W analysis high pressure reactor trip setpoint value of 2385.3 psig (2400 psia) represents the combination of nominal trip setpoint value (2355 psig) plus uncertainty (20.75 psi) plus margin (9.55 psi).

The actual plant field high pressure reactor trip setpoint value of 2325.44 psig represents the nominal trip setpoint value (2355 psig) minus uncertainty (12.51 psi) minus margin (17.05 psi).

Note that the nominal setpoint value and the setpoint value contained in the ITS are the same – 2355 psig.

2.8.5.2.1.2 Provide information concerning the basis for the assumed trip delay times for both high neutron flux and high pressure.

Response:

The 0.61 second high pressure reactor trip delay time represents the sensor & processing delay time (0.375 seconds) plus breaker open & control rod drive mechanism (CRDM) release time (0.230 seconds) plus assumed margin (0.005 seconds).

The 0.42 second high neutron flux reactor trip delay time represents the sensor & processing delay time (0.185 seconds) plus breaker open & CRDM release time (0.230 seconds) plus assumed margin (0.005 seconds)

2.8.5.2.1.3 The TR states that 0 percent steam generator tube plugging was modeled. It is apparent to the reviewer that this assumption would maximize the primary-to-secondary heat transfer, and therefore result in the most prompt delivery of core energy to the main steam system. It is not, however, apparent that such an



assumption is conservative and appropriate for modeling to determine the peak RCS pressure. Please justify this assumption, or quantify the effect of changing the input to a more limiting value.

Response:

Although steam generator tube plugging results in a reduction in available steam generator tube heat transfer area, for once-through steam generators (OTSGs) it also results in an increase in secondary steam generator inventory / level that effectively maintains heat transfer area for steam generator tube plugging below 10 percent. As a result, tube plugging levels at or below 10 percent do not appreciably bias the analysis results for primary or secondary pressure response for Babcock and Wilcox (B&W) OTSG plants.

The turbine trip event is the limiting secondary system pressure event with parameters biased to maximize the secondary system pressure. The event is not limiting for primary system pressure, with a calculated peak RCS pressure of 2569.8 psia compared to the overpressure limit of 2764.7 psia. Steam generator tube plugging of five percent will cause a reduction in the primary system volume of approximately 150 ft³ representing 1.4 percent of the total RCS volume. This small decrease has the potential to increase the rate of RCS temperature and pressure increases during the event. However, increased pressurization rates would lead to an earlier reactor trip. Since the calculated RCS peak pressure is significantly below the overpressure limit, the limit would not be challenged even if the slight reduction in RCS volume due to tube plugging were included in the calculations.

2.8.5.2.3 Loss of Normal Feedwater Flow

2.8.5.2.3.2 Pressurizer safety valves, for this and other transients, are modeled with 3 percent lift tolerance, with 0 percent accumulation and 4 percent blowdown. Please justify this modeling assumption relative to valve design and observed performance capabilities.

Response:

The NRC approved topical report, BAW-10193(PA) (Ref [7], Section A.3.2.3) provides the following justification for the pressurizer safety valve modeling assumption:



2.8.5.2.3.3 Provide similar justification as in 2.8.5.2.3.2 relative to the main steam safety valves modeling assumptions.

Response:

The NRC approved topical report, BAW-10193(PA) (Ref [7], Section A.3.2.3) provides the following justification for the main steam safety valve modeling assumption:

CR-3 has 8 MSSVs for each steam generator. Seven of the MSSVs on each steam generator have a capacity of 845,759 lbm/hr at 1159.2 psia. The remaining MSSV on each steam generator has a capacity of 583,574 lbm/hr at 1159.2 psia. The nominal lift setpoints are shown below. Note that the MSSV setpoints are analyzed at the inlet to the valve.

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Nominal Lift Setpoint Psig	Number per steam generator
1050	2
1070	2
1090	2
1100	2 ⁽¹⁾

(1) The 1 MSSV with a capacity of 583,574 lbm/hr has a nominal setpoint of 1100 psig.

Figure 2.8.5.2.3.3-1 Normalized MSSV Area vs. Steam Pressure Logic Diagram



2.8.5.2.3.5 Please provide additional information to justify the chosen steam generator tube plugging modeling selection.

Response:



2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment

By letter dated July 17, 2012, the licensee provided ANP-3114(P), which discussed sensitivity studies performed on the initial conditions. The sensitivity studies identified a new set of limiting initial conditions. It is this analysis, and the associated initial conditions, that the NRC staff evaluated in support of the proposed EPU.

2.8.5.2.4.1 The limiting results from ANP-3114(P) are slightly less severe than those presented in the TR. The base case evaluated in ANP-3114(P) is significantly less severe than the analysis reported in the TR. Although ANP-3114 indicates that the TS minimum value for EFW was used in the analysis, the peak pressure occurs before EFW flow initiates, both in the TR analysis and in ANP-3114(P). Please identify the modeling assumptions that differed between the TR analysis and ANP-3114(P), which caused the ANP-3114(P) base case results to be significantly less severe.



Response:

The NRC approved standard non-LOCA methodology (BAW-10193P-A, Ref [7]) uses nominal inputs coupled with some conservative/bounding values. Prior to fully exploring the sensitivity of the analysis results to fully biased inputs we removed some of the conservative/bounding inputs. Thus, the base case was less severe than the original analysis. The results demonstrate that the approved methods' combination of nominal values and some conservative bounding ones as opposed to a full spectrum of biased inputs produced very similar results indicating that the methods were and remain sound.

A summary of modeling assumptions that differ between the base cases presented in ANP-3114(P) (Ref [8]) and the TR is shown below:





2.8.5.2.4.2 Explain why the feedwater line break (FWLB) results are significantly more severe than the loss of normal feedwater (LONF) results.

Response:

A LONF is an anticipated operational occurrence or an ANS-57.5 Condition II event. The FWLB is limiting fault or ANS-57.5 Condition IV event that is not expected to occur.

Table 2.8.5.2.4.2-1 below provides a comparison of differences between LONF and FWLB in terms of key input parameters and initial conditions, sequence of events and results. It should be noted that, related to the severity and the timing of peak RCS pressure response, the input parameters that have the potential to increase the severity of results for FWLB as compared to the LONF are MFW flow reduction and the RCS pressure setpoint for reactor trip.

Isolating the MFW flow at the beginning of the transient for FWLB completely reduces the inflow of MFW into the steam generators; this approach represents additional conservatism to produce a more limiting peak RCS pressure. In contrast, the MFW flow is linearly reduced over the first 3.2 seconds for LONF, allowing additional mass to supplement the steam generator secondary inventory early into the transient. Thus, early into the transient, more steam generator inventory is available for LONF event to ensure an increased heat transfer rate from the primary to the secondary than that available for FWLB event.

As the transient progresses towards the time of peak RCS pressure, steam generator secondary side is depleted at different rates for LONF compared to FWLB. The former releases steam only through the MSSV, while the latter expels both steam and liquid through the break at a much higher rate. The faster depletion of steam generator secondary side inventory for the FWLB results in a decrease in the primary to secondary heat transfer rate which translates into a faster and earlier RCS pressure excursion compared to LONF. The table below shows that the total steam generator mass available at the time of peak RCS pressure to lessen the RCS pressure increase for FWLB is significantly less than total steam generator mass for LONF.

Aside from the arguments presented above, an additional factor that should be noted is that the high RCS pressure setpoint for reactor trip for FWLB is higher by 45.45 psi compared to LONF. The associated consequence is a delayed reactor trip for FWLB compared to the scenario that would have considered a high RCS pressure setpoint of 2400.0 psia for FWLB (same as LONF). The additional delay for the high RCS pressure trip allows for more energy buildup in the RCS, augmented by the rapid reduction in the SG secondary side inventory, which results into an increased RCS pressurization rate.



Table 2.8.5.2.4.2-1. Comparison of LONF and FWLB Parameters

	Loss of Normal Feedwater – Condition A, Overpressure Event (CR-3 EPU TR - Section 2.8.5.2.3.2)	Feedwater Line Break – Overpressure Event No PZR Spray (CR-3 EPU TR - Section 2.8.5.2.4.2)
Key Input Parameters and Initial Conditions		
MFW flow reduction ¹	Linearly reduced to zero over 3.2 seconds	Reduced to zero over 1.0E-6 seconds
High RCS pressure setpoint for reactor trip	2400.0 psia	2445.45 psia
Sequence of Events		
Transient initiation, s	0.0	0.0
MFW to both steam generator interrupted, s	3.2	1.0E-6
RPS high RCS pressure trip actuated, s	15.47	8.802
Control rods begin to insert, turbine stop valve begin to close, s	16.08	9.415
Initial pressurizer safety valve lift occurs, s	18.86	~12
Peak RCS pressure occurs, s	19.49	13
Results		
Peak RCS pressure, psia	2750.63	2896.2
SG-A / SG-B Total Mass at time of peak RCS pressure, lbm	24877.6 / 24889.1	28681.5 / 6482.48

Table notes:

1. The complete isolation of the MFW flow to both steam generators at the beginning of the FWLB overpressure event with no pressurizer spray represents additional conservatism to produce a more limiting peak RCS pressure.



2.8.5.2.4.3 Discuss any additional available analytic operating experience with the FWLB for other Babcock & Wilcox plants. How do the CR-3 results differ from results for other plants, and why are they different?

Response:

FWLB analyses performed for other B&W plants indicate similar results to those calculated for CR-3 prior to EPU conditions. Peak RCS pressure is calculated to remain below 110 percent of design pressure at hot full power conditions given a double ended guillotine break of the feedwater pipe.

The peak RCS pressure for the post-EPU FWLB event is greater than 110 percent of design pressure due to the higher RCS stored energy and greater integrated power from event initiation to the time of the peak. Table 2.8.5.2.4.3-1 below provides a comparison of peak RCS pressure for several B&W plants, including CR-3 at pre-EPU conditions, for the FWLB event.

Table 2.8.5.2.4.3-1 Comparison of B&W Plant FWLB Peak Pressure Results

Plant	Peak RCS Pressure, psia	Acceptance Criteria Limit, psia	Acceptance Criteria (% of limit)
CR-3 Pre-EPU	2731.21	2764.7 (110% of 2500 psig)	98.79
CR-3 Post-EPU	2878.38	3014.7 (120% of 2500 psig)	95.48
Plant A (TMI-1)	2755.15	2764.7 (110% of 2500 psig)	99.65
Plant B (DB-1)	< 2600.0	2764.7 (110% of 2500 psig)	< 94.04

2.4 RAIS Related to LAR Section 2.8.5.3

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

2.8.5.3.2.1 The Locked Rotor Analysis in the CR-3 EPU TR does not clearly show the acceptance criteria and the analytical basis to demonstrate that the criteria are met. TR section 2.8.5.3.2 states that the thermal design limit is not met, therefore indicating that fuel failure may occur. However, the locked rotor maximum allowable peaking analyses demonstrate that fuel failure does not occur. The alternate source term dose acceptance criteria are used to back-calculate the amount of pins allowed to fail without exceeding the 90 percent limit in TR section 2.9.2. It is not clear what

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acceptance criteria are being proposed, nor is it clear how the acceptance criteria are being met for the locked rotor event. Please clarify.

Response:

The locked rotor event is a Condition IV event which has the following acceptance criterion: a maximum allowable failed fuel fraction criterion such that the dose from the event does not exceed 90% of the dose at the limiting receptor as defined by the LOCA analysis.

The number of fuel pins that violate the DNB criterion for failed fuel—the Thermal Design Limit (TDL)—is a key input to the dose analysis. The maximum allowable number of failed fuel pins for the locked rotor event is determined such that the dose from the locked rotor event does not exceed 90% of the dose at the limiting receptor defined by the LOCA analysis. This maximum allowable number of failed fuel pins can be used in conjunction with a cycle-specific pin census to show that the maximum allowable failed fuel fraction criterion is met for that cycle. Alternatively, a DNB analysis could be performed that shows the event does not violate the DNB criterion for failed fuel determination, therefore indicating that no failed fuel occurs (which proves the cycle meets the allowable failed fuel fraction criterion).

When evaluating the plant at the EPU power level, the locked rotor analysis was initially evaluated using the design peaking distribution and found to have a minimum DNBR of 1.41 which is slightly below the TDL of 1.45. A reduction in the design radial peak (from the design peaking distribution) was imposed to produce a minimum DNBR equal to the TDL, thereby demonstrating that no fuel pin census calculation is necessary. This reduced radial peak was imposed in the form of locked rotor MAP limits.

For the conceptual cycles that were analyzed for the EPU LAR, the above process led to an imposed small radial peak reduction which avoided the need for a fuel pin census calculation for the event. Alternatively, violation of the TDL could be addressed by conducting a fuel pin census for the event and verifying that the number of failed fuel pins meets the maximum allowable failed fuel fraction criterion. For future cycles, either analytical basis may be utilized.



2.5 RAIS Related to LAR Section 2.8.5.4

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

2.8.5.4.1.2 If the transient were terminated instead by a neutron flux trip, please explain what sensitivity the trip timing, and corresponding results, would have to the selected initial conditions.

Response:

If the transient were terminated instead by a neutron flux trip, the time at which the RPS High Neutron Flux trip setpoint is reached would not vary as a result of changing the initial RCS pressure. The Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition was analyzed over a range of reactivity insertion rates from 12.3 pcm/sec to 25.0 pcm/sec. None of these insertion rates resulted in a reactor trip on high neutron flux because these events are initiated at low power while the high neutron flux reactor trip setpoint is established at 112 percent full power. At 25.0 pcm/sec, the peak total power reached is 99.65 percent full power. However, the maximum reactivity insertion rate is capped at 20.0 pcm/sec due to primary system overpressure considerations. Therefore, higher reactivity insertion rates that could potentially result in a high neutron flux reactor trip are precluded.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

2.8.5.4.2.2 Justify the selection of 4.62 percent millinile [pcm]/sec as a conservative reactivity insertion limit.

Response:

The Uncontrolled Control Rod Assembly Withdrawal at Power event was evaluated for a spectrum of RIRs extending from 0.10 pcm/sec to 92.8 pcm/sec. This spectrum study represents control rod worths from 0.0139% $\Delta k/k$ to 12.9% $\Delta k/k$. The nominal single group worth is typically about 1.71% $\Delta k/k$ and an all group worth is about 12.9% $\Delta k/k$. The RIRs are increased by a factor of two to account for variations in RIR along the travel of the rod due to non-linear differential worth. The range is analyzed since one of two reactor trip signals, high RCS pressure and high neutron power, may be used to terminate the event. The RIR which produces the highest peak pressure and peak thermal power is the RIR that would trip on high RCS pressure and high neutron flux at the same time. RIRs at the lower end of the analyzed range result in a reactor trip on high RCS pressure, whereas RIRs at the higher end trip on high flux. Near the RIR value resulting in the peak RCS pressure, additional cases are run to identify the breakpoint between the two trip functions. At



RIR rates higher than this transition point, the reactor will trip earlier and the peak RCS pressure will be lower. For RIRs below this point, the rate of pressurization is slower such that the pressure overshoot following reactor trip is less. This is shown in Figure 2.8.5.4.2.2-1 below. Thus, the 4.62 pcm/sec RIR represents a point of maximum peak pressure, due to the trip function transition and is not a maximum or minimum insertion rate limit. The LAR presented data for this case since it produced the worst peak RCS pressure results for this event at full power.

Figure 2.8.5.4.2.2-1 Peak RCS Pressure versus Reactivity Insertion Rate for the Rod Withdrawal at Power Event





2.8.5.4.3 Control Rod Misoperation

2.8.5.4.3.1 Please explain how the stuck-out control rod assembly and stuck-in control rod assembly events are dispositioned through analyses. How is it confirmed that the dropped rod is the most limiting control rod mis-operation?

Response:

A stuck-out control rod event occurs if one control rod fails to insert or remains stuck in the withdrawn position when a reactor trip occurs. The concern associated with a stuck-out control rod during a reactor trip is the reduction in total available control rod worth. To address this concern, the shutdown margin requirements in ITS 3.1.1 ensure that there is sufficient negative reactivity insertion to achieve the hot shutdown condition when considering the maximum worth rod stuck out of the core.

A stuck-out or stuck-in control rod misalignment could also occur if one rod becomes stuck at some position while the other control rods are moved within their insertion limits. Control rod misalignment at power can result in excessive power peaking. The EPU Core Power Distribution Analysis described in Section 2.8.2 of the TR considers the impact of misaligned control rods when determining the LCO axial offset limits.

For the CR-3 EPU, explicit evaluations of the peaking associated with control rod misalignment were performed to verify that the LCO axial offset limits provide margin to the specific acceptable fuel design limits (SAFDLs) (centerline fuel melt kW/ft limits, transient cladding strain kW/ft limits and misaligned rod DNB MAP limits) at full power steady-state conditions.

A dropped control rod event occurs when a single control rod rapidly falls into the core. The core power initially decreases due to the negative reactivity associated with the control rod. Moderator and Doppler feedback provide positive reactivity insertion which results in an increase in power. The system analysis described in Section 2.8.5.4.3 of the TR is used to determine the magnitude of the increase in power and the remaining statepoint conditions (core pressure, RCS flow, and core inlet temperature) associated with the dropped rod event. The statepoints from a dropped rod event are then used in the Core Power Distribution Analysis (Section 2.8.2 of the TR) to verify that the LCO axial offset limits would provide margin to the SAFDLs if a dropped rod event were to occur.





Thus, as explained in Section 2.8.2 of the TR, the core power distribution analysis evaluates both rod misalignment and dropped rod conditions. Therefore, it is not necessary to confirm that the dropped rod event is the most limiting control rod misoperation. Nevertheless, evaluations of both events have typically shown that the dropped rod has less margin to the SAFDLs than rod misalignment. The CR-3 EPU evaluations determined that the minimum DNB peaking margin associated with control rod misalignment is [] compared to [] margin for the dropped control rod event.

2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature

2.8.5.4.4.1 Justify the selected initial conditions in light of the predicted response. How could other permissible initial conditions make the results of this event more severe?

Response:

The current licensing basis of the plant is the startup of two RCPs from an initial two pump condition, with one pump operating in each steam generator loop. This operating condition is precluded by the RPS which allows only three and four pump operation based upon the Reactor Coolant Pump Power Monitor trip function. The initial power level chosen is higher than that allowed by operating procedures and control system interlocks. In addition, two pumps are modeled to start simultaneously to maximize the reactivity and power response during the event even though this would not be allowed at the plant.

The startup of an inactive RCP is characterized by an increase in core flow over time. Due to the nature of the CR-3 design, during partial pump operation the cold leg temperatures of the "idle" pump loops remain very close to that of the operating pump loops due to the reverse flow through the idle pumps. The actual temperatures in the loops are modeled in RELAP5 prior to the pumps starting. Starting the idle pumps leads to a reduction in the average temperature in the core as well as a drop in the temperature rise across the core. The maximum reactivity insertion rate occurs at end of cycle when starting two pumps from an initial two pump operating condition.

To understand the behavior, consider the following:

Assuming no power feedback, the increased flow (at the same inlet temperature) will lead to a decrease in the temperature rise across the core equivalent to the inverse of the flow increase. Thus, the temperature rise will be cut in half for a start of two pumps from an initial two pump operating condition. In contrast, the temperature rise will decrease by 33 percent for a start of one pump from an initial three pump operating condition.

Since T_{cold} is initially unchanged, the average temperature in the active core region drops, leading to positive feedback from a negative moderator



temperature coefficient (MTC). The magnitude of the temperature change in the active core region can be estimated by equating the energy removal before and after the pumps start.

$$\Delta T_{\text{start}} \times \omega_{\text{start}} \approx \Delta T_{\text{end}} \times \omega_{\text{end}}$$

The ratio of the beginning and ending flows (~50 percent full flow [ω_{start}] going to 100 percent full flow [ω_{end}], respectively) provides the largest change in temperature which will lead to the largest power feedback. Thus, starting a single pump is a less severe event because it leads to a smaller reactivity feedback from the flow increase.

The temperature rise across the core is also a function of the initial power level, i.e., the higher the initial power level the higher the initial core ΔT . Hence, the event is chosen to start from a higher initial power level than allowed by operating procedures and control system interlocks.

The MTC was chosen as a bounding value of -50 pcm/ $^{\circ}$ F. This maximum value results in the largest power increase during the event and bounds all times in life. The Doppler coefficient has been chosen as the least negative value to provide the least feedback to the power increase.

In summary, the Startup of an Inactive Loop at an Incorrect Temperature event is analyzed at conditions prohibited by plant operations. As discussed in the TR, the RPS precludes power operation with less than three pumps operating. The ICS also acts to prevent starting an RCP when the initial power level is greater than the associated interlock (30 percent FP). However, the analysis modeled two RCPs starting simultaneously to maximize the flow and power increase. For conservatism, the MTC is maximized while the Doppler was minimized, again to maximize the power increase during the event. Since the transient evolution is based upon changes in the core average temperature from the initial condition, a change in the initial temperature or pressure would not affect the results because the same initial core ΔT would be present. Thus, the initial conditions are very conservative relative to those expected at the plant. Based upon the items described above, a conservative set of initial conditions were analyzed for this event.

2.8.5.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

- 2.8.5.4.5.1** Describe the effects of steam generator tube plugging on this transient. Why is 0 percent steam generator tube plugging conservative? Does this conservatism exceed the reduction in RCS volume that would cause an increase in the dilution rate?



Response:

Steam generator tube plugging reduces the RCS volume available for mixing during a moderator dilution event. For a constant dilution volume and flow rate, higher tube plugging results in lower RCS volume and larger reduction in boron concentration.

The Moderator Dilution Accident (MDA) is comprised of a RELAP5 analysis and hand calculation. The RELAP5 analysis determines the primary peak pressure and timing of the reactor trip. The timing of the reactor trip was used in the hand calculation to determine when the makeup system deborated water isolation valve [MUV 541] closes. At the time of valve closure, the only dilution source is the total volume of the makeup tank. The MDA event is not a limiting event for primary peak pressure; thus, zero percent steam generator tube plugging was assumed to delay the reactor trip in order to maximize the time of [MUV 541] closing which maximizes the integrated dilution volume introduced into the RCS.

The MDA hand calculation is conservatively analyzed with a range of constant dilution flow rates to verify that, although dilution of the moderator would continue until the makeup tank empties, sufficient shutdown margin would exist such that the reactor would always remain subcritical. The total reactivity insertion due to the reduction in boron concentration is calculated using the minimum active RCS volume in conjunction with the maximum initial boron concentration. Since the active RCS volume decreases as the level of steam generator tube plugging increases, the volume change associated with five percent plugged steam generator tubes is factored into the calculation of the total reactivity inserted.

2.8.5.4.5.2 Describe the extent of operator action required to terminate the event.

Response:

During a postulated moderator dilution accident, the dilution valve [MUV 541] from the deboration water source to the makeup tank would close after the control rods reach a preset position following reactor trip. While [MUV 541] would close post reactor trip; thus terminating the source of deborated water, it was assumed that the makeup tank volume of 600 ft³ was injected into the RCS. The plant design and analysis approach does not require any operator actions to terminate the event.



2.8.5.4.6 Spectrum of Rod Ejection Accidents

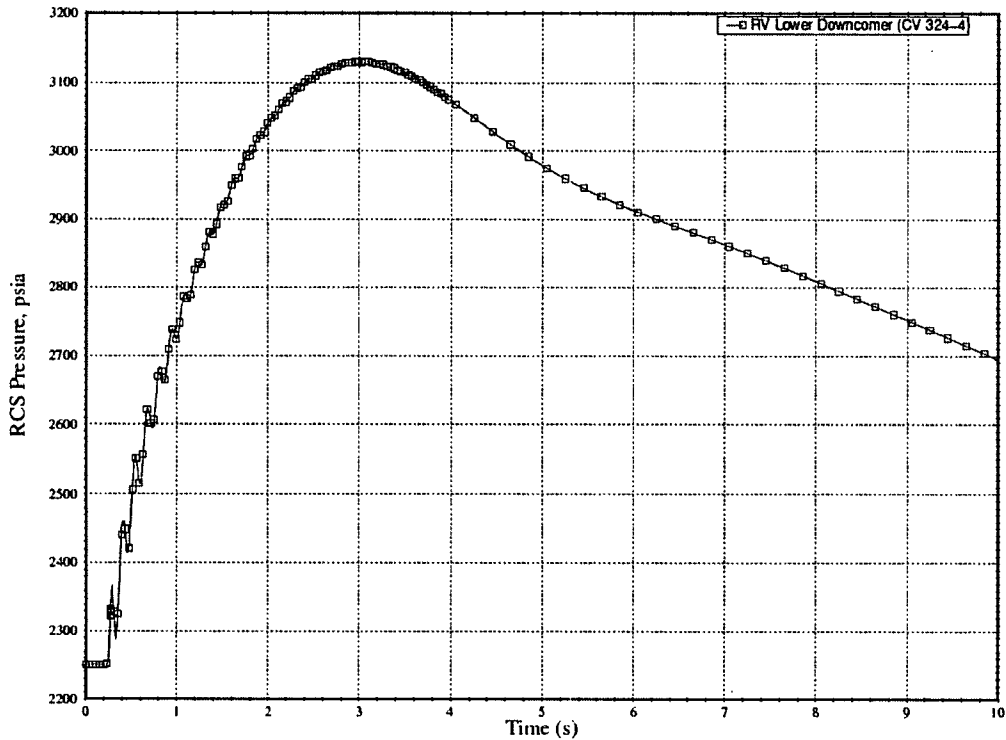
2.8.5.4.6.1 Please provide a plot of pressure vs. time for the pressure-limiting rod ejection event, and explain why there is such a significant pressure excursion.

Response:

The plot of pressure versus time for the pressure-limiting event (HZP, BOC, with direct moderator heating) is provided in Figure 2.8.5.4.6.1-1 below.

Due to the rod ejection, the neutron power rises, the power increases and, as the energy in the fuel is transferred to the coolant, the RCS pressure is increased. As discussed in the TR, the over-pressurization analysis assumes no breach of RCS integrity occurs as a direct result of the rod ejection, thereby maximizing the RCS pressure excursion. Moreover, this pressure increase is compounded by the pressurizer surge line diameter.

Figure 2.8.5.4.6.1-1 CR-3 HZP BOC w/ Direct Moderator Heating – RCS Pressure





3.0 REFERENCES

1. NRC Letter from Siva P. Lingam to Mr. Jon A. Franke (ML12333A089), Subject: CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT -REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST (TAC NO. ME6527), Dated December 19, 2012
2. BAW-10179P-A, Revision 8, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses*, May 2010
3. BAW-10162P-A, Revision 00, *TACO3 - Fuel Pin Thermal Analysis Computer Code*, October 1989
4. BAW 10184P-A, Revision 00, *GDTACO, Urania-Gadolinia Fuel Pin Thermal Analysis Code*, February 1995
5. BAW-10231P-A, Revision 1, *COPERNIC Fuel Rod Design Computer Code*, January 2004
6. ANP-3120(P), Revision 0, *Crystal River Unit 3 EPU - NRC RAI and Responses (LAR Sections 2.8.1, 2.8.2, 2.8.3)*, June 2012
7. BAW-10193P-A, Revision 0, *RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors*, January 2000
8. ANP-3114(P), Revision 0, *CR-3 EPU - Feedwater Line Break Analysis Sensitivity Studies*, May 2012

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT E

LOCATION OF REACTOR SYSTEMS RAI RESPONSES

LOCATION OF REACTOR SYSTEMS RAI RESPONSES

Attachment A	Attachments B & D
2.8.4.3.1	2.8.4.3.2
2.8.4.4.1	2.8.5.1.1.1
2.8.4.4.2	2.8.5.1.1.2
2.8.5.2.3.1	2.8.5.1.1.3
2.8.5.2.3.6	2.8.5.1.1.4
2.8.5.2.4.4	2.8.5.2.1.1
2.8.5.2.4.5	2.8.5.2.1.2
2.8.5.6.1.1	2.8.5.2.1.3
	2.8.5.2.3.2
	2.8.5.2.3.3
	2.8.5.2.3.5
	2.8.5.2.4.1
	2.8.5.2.4.2
	2.8.5.2.4.3
	2.8.5.3.2.1
	2.8.5.4.1.2
	2.8.5.4.2.2
	2.8.5.4.3.1
	2.8.5.4.4.1
	2.8.5.4.5.1
	2.8.5.4.5.2
	2.8.5.4.6.1