UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 28, 2010

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

SUBJECT: CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT REGARDING ADOPTION OF A NEW METHODOLOGY FOR ROD EJECTION ACCIDENT ANALYSIS UNDER EXTENDED POWER UPRATE CONDITIONS (TAC NO. ME0730)

Dear Mr. Franke:

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-72 for Crystal River, Unit 3 (CR-3) in response to your application dated February 26, 2009, as supplemented by letter dated May 29, 2009. The amendment approves a new methodology, developed by AREVA NP for CR-3, to analyze the rod ejection accident under extended power uprate conditions. The adoption of the new methodology is reflected in a change to the CR-3 Operating License and Improved Technical Specifications (ITS). The CR-3 ITS Section 5.6.2.18.b is being revised to add this new methodology to the list of approved methods used in developing the Core Operating Limits Report. Additionally, Operating License Condition 2.C.(12), which was a one-cycle license condition, is being deleted.

A copy of the safety evaluation is enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 237 to Facility
   Operating License No. DPR-72
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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FLORIDA POWER CORPORATION

CITY OF ALACHUA

CITY OF BUSHNELL

CITY OF GAINESVILLE

CITY OF KISSIMMEE

CITY OF LEESBURG

CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION,

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ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO

SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated February 26, 2009, as supplemented by letter dated May 29, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 237 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented during Refuel 17 that is scheduled for fall of 2011.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: January 28, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 237

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of Facility Operating License DPR-72 with the attached revised pages.

Remove	Insert
4	4
5	5

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

Remove	Insert
5.0-23	5.0-23

of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

2.C.(1) Maximum Power Level

Florida Power Corporation is authorized to operate the facility at a steady state reactor core power level not in excess of 2609 Megawatts (100 percent of rated core power level).

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 237 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

The Surveillance Requirements contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment 149. The Surveillance Requirements shall be successfully demonstrated prior to the time and condition specified below for each.

a) SR 3.3.8.2.b shall be successfully demonstrated prior to entering MODE 4 on the first plant start-up following Refuel Outage 9.

b) SR 3.3.11.2, Function 2, shall be successfully demonstrated no later than 31 days following the implementation date of the ITS.

c) SR 3.3.17.1, Functions 1, 2, 6, 10, 14, & 17 shall be successfully demonstrated no later than 31 days following the implementation date of the ITS.

d) SR 3.3.17.2, Function 10 shall be successfully demonstrated prior to entering MODE 3 on the first plant start-up following Refuel Outage 9.

e) SR 3.6.1.2 shall be successfully demonstrated prior to entering MODE 2 on the first plant start-up following Refuel Outage 9.

f) SR 3.7.12.2 shall be successfully demonstrated prior to entering MODE 2 on the first plant start-up following Refuel Outage 9.

g) SR 3.8.1.10 shall be successfully demonstrated prior to entering MODE 2 on the first plant start-up following Refuel Outage 9.

h) SR 3.8.3.3 shall be successfully demonstrated prior to entering MODE 4 on the first plant start-up following Refuel Outage 9.

2.C.(6) Deleted per Amendment No. 21, 7-3-79

2.C.(7) Prior to startup following the first regularly scheduled refueling outage, Florida Power Corporation shall modify to the satisfaction of the Commission, the reactor coolant system flow indication to meet the single failure criterion with regard to pressure sensing lines to the flow differential pressure transmitters.

2.C.(8) Within three months of issuance of this license, Florida Power Corporation shall submit to the Commission a proposed surveillance program for monitoring the containment for the purpose of determining any future delamination of the dome.

2.C.(9) Fire Protection

Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports, dated July 27, 1979, January 22, 1981, January 6, 1983, July 18, 1985, and March 16, 1988, subject to the following provisions:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. *{Amdt. #147, 1-22-93}*

2.C.(10) The design of the reactor coolant pump supports need not include consideration of the effects of postulated ruptures of the primary reactor coolant loop piping and may be revised in accordance with Florida Power Corporation's amendment request of April 24, 1986. *{Added per Amdt. #89, 5-23-86}*

2.C.(11) A system of thermocouples added to the decay heat (DH) drop and Auxiliary Pressurizer Spray (APS) lines, capable of detecting flow initiation, shall be operable for Modes 4 through 1. Channel checks of the thermocouples shall be performed on a monthly basis to demonstrate operability. If either the DH or APS system thermocouples become inoperable, operability shall be restored within 30 days or the NRC shall be informed, in a Special Report within the following fourteen (14) days, of the inoperability and the plans to restore operability. *{Amdt. #164, 1-27-98}*

2.C.(12) Deleted per Amendment No. 237

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.

ANP-2788P, "Crystal River 3 Rod Ejection Accident Methodology Report," Revision 0, and License Amendment 237 dated January 28 , 2010.

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)

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated February 26. 2009, (Reference 1), as supplemented by letter dated May 29, 2009 (Reference 2), Florida Power Corporation, the licensee for Crystal River Nuclear Plant, Unit 3, submitted a license amendment requesting approval of a new methodology, which was developed by AREVA NP, to analyze the rod ejection accident (REA) under extended power uprate (EPU) conditions at Crystal River Unit 3 (CR-3). The adoption of the new methodology is reflected in a change to the CR-3 Operating License and Improved Technical Specifications (ITS). The CR-3 ITS Section 5.6.2.18.b is being revised to add this new methodology to the list of approved methods used in developing the Core Operating Limits Report. Additionally, the amendment would delete Operating License Condition 2.C.(12), which was a one-cycle license condition.

The supplement dated May 29, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* June 23, 2009 (74 FR 29732).

2.0 REGULATORY EVALUATION

The methodology used to analyze the REA for the CR-3 reactor is based on a 3-dimensional (3-D) space-time kinetics solution to the diffusion equations with both thermal-hydraulic and fuel temperature feedback and a separate peak rod evaluation with open channel thermal-hydraulic and fuel thermal model. These models provide more realistic neutronic and thermal-hydraulic conditions than the previous analysis was capable of, and are able to show compliance with the interim reactivity initiated accident criteria of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 3 (SRP).

The proposed amendment is evaluated against the applicable regulatory requirements in the following regulations, guidance, or criteria:

1. SRP, Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR [Pressurized Power Reactor])," March 2007

2. SRP, Section 4.2, "Fuel System Design," March 2007

3. NUREG/CR-6742, LA-UR-99-6810, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," Los Alamos National Laboratory, September 2001

Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," of SRP, Section 4.2 provides the interim acceptance criteria and guidance for the reactivity-initiated accidents that involve a sudden and rapid insertion of positive reactivity, including a control rod ejection for pressurized water reactors.

In addition, the regulatory positions and specific guidelines necessary to meet the relevant requirements are provided in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

The results computed using the methodology outlined in the following sections of this SE have to meet regulatory requirements regarding fuel cladding failure and core cooling and are discussed in Sections 2.1 and 2.2 of this safety evaluation (SE). The impact of the proposed change on radiological consequences analysis is evaluated in Section 2.3 of this SE.

2.1 Fuel Cladding Failure Criteria

Cladding failure can occur as a result of three different mechanisms that include pellet clad mechanical interaction (PCMI), total energy deposited, and DNBR. Each rod is examined following a transient to ascertain if it has exceeded any of the limiting criteria for these mechanisms that are outlined in SRP, Section 4.2, Appendix B, as listed below.

- For zero power conditions for fuel rods with an internal rod pressure at or below system pressure, the high cladding temperature failure criteria is a peak radial average fuel enthalpy greater that 170 cal/g

- For zero power conditions for fuel rods with an internal rod pressure exceeding system pressure, the high cladding temperature failure criteria is a peak radial average fuel enthalpy greater that 150 cal/g

- For greater than 5 percent and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits, e.g., departure from nucleate boiling ratio (DNBR)

- Further, the pellet cladding mechanical interaction (PMCI) criteria, as depicted Figure B-1 of SRP, Section 4.2, Appendix B, is a change in radial average fuel enthalpy greater than corrosion-dependent limit (oxide/wall thickness)

In Reference 1, the licensee stated that the PCMI failure takes place during the initial power pulse, and is a function of the increase in fuel enthalpy and the oxide film thickness on the clad. The limiting rod at the end of life sets the requirements for this failure mechanism. A burnup of 62 gigawatt-day per ton (GWD/T) is used to determine the requirements, based on this limit the oxide to wall thickness ratio is 0.052. This implies a PCMI failure limit of 125 cal/g. Regarding cladding failure due to total energy deposition, the licensee stated that at the reactor power less or equal to 5 percent, an increase in enthalpy of 150 cal/g is assumed to lead to rod failure for REA simulations. This is more conservative than the value of 170 cal/g limit in the SRP. Further, for REA simulation at CR-3 with core power above 5 percent, cladding failure is assumed to occur if the surface heat flux exceeds the thermal design limits for minimum DNBR (MDNBR).

2.2 Core Coolability

The SRP, Section 4.2, Appendix B requirements regarding core coolability are:

- Peak average enthalpy must remain below 230 cal/g
- Peak fuel temperature must remain below incipient fuel melting conditions
- Mechanical energy generated by possible fuel-coolant interactions and fuel rod burst must be considered when assessing the reactor pressure boundary, reactor internals, and structural integrity
- No loss of coolable geometry due to fuel pellet and cladding fragmentation and fuel rod ballooning

In Reference 1, the licensee stated that the coolability for fuel rods undergoing DNBR failure is established by limiting rod heat up post critical heat flux. If the rods remain below the limits established above for PCMI and total energy added, then energetic interaction of fuel or fragmented rod pieces with the coolant is prevented. In the case where the rod pressure is above system pressure, rupture or significant ballooning is unlikely by limiting the maximum cladding temperature.

2.3 Radiological Consequences

The number of failed rods is an input to the radiological consequences calculation for postulated REA events. The proposed amendment provides a simple equation for determining an equivalent number of failed rods using the number of rods failed due to DNBR and the equivalent number of additional rods failed due to transient fission gas released, but does not propose a new methodology for calculating the radiological consequences for REA events. Because there is no change to the existing REA radiological consequences analysis methodology, the proposed change does not impact the existing methodology and does not impact the results of the current CR-3 radiological consequences analyses. Since the results of the REA remain unchanged, the NRC staff finds the proposed change acceptable with respect to the radiological consequences of a design basis REA.

3.0 TECHNICAL EVALUATION

The overall methodology required in the calculations for the CR-3 REA is evaluated in this section. The salient steps involved in the methodology are illustrated in Figure 1 of Reference 3.

The starting point is determined by the design for fuel pins that are assembled into a fuel assembly and combined to form the reactor core. Following this step, the state of the reactor before the REA are defined; these include burnup, power level, location of rod to be ejected, system state points, etc. In addition, the regulatory and radiological limits are recognized, and the code suite to be used is selected. The later step is important, since the inter-linkage of the various codes that supply the solution as the transient progresses is dependent on the expected outcome of the transient. This point has been made in connection with the reactor scram and transient termination depending on the amount of reactivity insertion requiring either an ex-core detector or a system level scram mode.

Once all the above preparatory work has been carried out, the analysis step can be carried out. The details of this step will be discussed in the section below in which the detailed inter-linkage of the various codes and associated information will be described. Following the analysis step, a decision is made regarding the ability of the core design under consideration to successfully survive the imposed transient or not. If any of the regulatory requirements are violated, the reactor core needs to be reconfigured and the process repeated until all requirements are satisfied. If the transient is completed without violating any regulatory requirements the analysis is considered complete.

3.1 Model Boundary Conditions and Uncertainties Requirements

The modeling boundary conditions and the uncertainty analysis discussion are divided into two sections, first addressing the plant transient analysis, and second addressing the fuel rod transient model. The first section is further divided into the initial transient caused by the rod ejection (lasting about 5 seconds), and the remainder of the time until the reactor is tripped. The fuel rod transient model discusses the fuel and cladding temperature limits and variation, and the onset of DNBR. Finally, there is a discussion on failure analysis and failed rod census conditions.

The licensee's discussion regarding the boundary conditions and uncertainty requirements is based on the results of PIRT studies carried out for the REA event. During the PIRT process, all major phenomena contributing to an REA are identified and their level of importance and knowledge are ranked. In general, results of the PIRT identify all the important parameters to be considered in the general analysis, and indicate the current level of knowledge required to determine them.

3.1.1 Description of Plant Transient Analysis

The plant transient analysis is dominated for the first 5 to 10 seconds of the event by core kinetics, nodal fuel temperature, and nodal thermal-hydraulic conditions. The time frame is set by the coolant loop transport time of the plant and the very rapid nature of the rod ejection event. During this time frame the inlet thermal hydraulic conditions are relatively constant, which allows the 3-D core kinetic analysis to be carried out independently of the system response. The PIRTs study is carried out assuming a point kinetics model, but since the differences between the point model and the 3-D model manifest themselves only in the local weighting of changes that occur, the overall importance of phenomena is very similar, and thus the conclusions apply equally to both kinetic models. The local weighting of changes is very important in the case of an REA. The parameters that could affect the results are reviewed below. The specific values for the parameters that apply to the CR-3 reactor are discussed in detail in Section 3.4 of this SE.

- Maximum Ejected Rod Worth: The worth of the maximum ejected rod is an extremely important value, since it drives the event and can be a limiting parameter. The worth is not a direct input, but is represented by the control rod cross sections and pre-ejection location in the core. The worth is a function of fuel cycle design, cycle lifetime, and initial xenon conditions. An uncertainty is applied that is greater than the approved uncertainty values. In addition, conservatisms are applied to bound future fuel cycle designs.

- Rate of Reactivity Insertion: This is not considered to be an important parameter for a prompt critical event. Sensitivity calculations are carried out to estimate the effect.

- Moderator Feedback: Moderator temperature coefficient (MTC) feedback is not considered an important effect during the power pulse of an REA. However, power variation after the initial pulse is affected by the MTC, which affects DNBR. The MTC is not directly input to the code, since this is a 3-D kinetics calculation, but the moderator cross sections reflect the variation with temperature and pressure.

- Fuel Temperature Feedback: Fuel temperature feedback in the form of a doppler temperature coefficient (DTC) is an important parameter in the event progression. It is the DTC that terminates the initial power pulse following the rod ejection. The DTC is not directly input to the code, since this is a 3-D kinetics calculation, but the fuel cross sections reflect the variation with temperature.

- Delayed Neutron Fraction: The delayed neutron fraction (β_{eff}) determines the rate of neutron flux change from its initial state. The higher the reactivity input relative to β_{eff}, the faster the flux increase.

  For reactivity insertions greater than β_{eff} the core becomes prompt critical and an extremely fast doubling time results. Low values of β_{eff}, as those that exist at end of cycle (EOC), result in much higher power pulses than at beginning of cycle (BOC) when the values are relatively higher.

- Reactor Trip Reactivity: The initial prompt critical power excursion is terminated by the DTC. Following the power excursion, the power settles to a lower value. The ex-core detectors react to the power excursion by inserting the scram rods and the reactor power is reduced to shutdown levels. During the power pulse the importance of the reactor trip reactivity is essentially zero, but following the pulse the timing and worth of the trip reactivity is important since it determines the post-pulse power level and thus the possibility of DNBR. This is particularly important for at power REA, when the ex-core detectors do not trip the reactor but system level trip signals are required. In a 3-D simulation reactivity effects are dynamically calculated by following the rod motion and appropriately adjusting the nodal cross sections.

- Fuel Cycle Design: The effects of the fuel cycle design are captured by examining the BOC and EOC β_{eff}, DTC, MTC, and peaking factors. In addition, the fuel cycle effects can also manifest themselves by the proximity of the ejected rod to end of life fuel assemblies. These effects are captured in the 3-D nature of the analysis.

In addition, thermal parameters that impact the analysis include:

1) Heat resistances and transient cladding to coolant heat transfer
2) Heat capacities
3) Fractional heat deposited in coolant
4) Pellet radial power distribution
5) Rod power peaking factors
6) Neutron velocities
7) System T-H conditions

The PIRTs analysis is extended to cover the specifics of the phenomena associated with the fuel and cladding, and in particular their failure. There is a certain amount of overlap between this discussion and that outlined in the previous section. Some of the parameters that could affect the results are:

1) Pellet and Cladding Dimensions
2) Burnup Distribution
3) Cladding Oxidation
4) Power Distribution
5) Initial Coolant Conditions
6) Transient Power Specification

3.1.2 Transient Model beyond 5 Seconds

The second time frame (greater than 5.0 seconds) requires the recognition that the coolant has made a complete circuit of the primary system. Thus, the system code S-RELAP is used to generate input for LYNXT, in order to determine the number of failed rods. The S-RELAP code is driven by a total power input derived from NEMO-K, which is generally a slowly rising function of time determined by extrapolating the last part of the initial transient calculation that did not end in a scram. S-RELAP has all the system scram functions built in, and thus sooner or later one of these is tripped and the system will scram, ending the transient. The determination of the number of failed rods in this case is simpler than in the above case, since the transient progresses in a quasi-static manner at this stage. The details of the method used to determine the number of failed fuel rods is outlined in section 3.3 below. A final tally of the number of failed rods is made and then a determination is made whether or not the number of failed rods is within regulatory limits. If the number of failed rods is within the regulatory limit the transient is over; if it exceeds the limit then the core needs to be redesigned.

3.1.3 Description of Fuel Rod Transient Model

The fuel rod model is dominated by the initial transient temperatures and the energy deposited in the rod versus time. As discussed in the previous section, thermal parameters such as inlet temperature, core pressure, and flow are relatively constant. The discussion in this section is a review of the parameters listed in Table 4-2 of the licensee's February 26, 2009 (Reference 1), submittal.

Pellet and cladding dimensions are considered important and well known. Nominal dimensions and application of the uncertainty for manufacturing allowances are appropriate. Approximations of the full core geometry model surrounding the limiting rod can affect the results. These approximations are shown to be appropriate for the REA analysis.

The local rod radial burnup distribution is rated as a relatively low importance parameter and a homogenized pellet is acceptable.

The cladding oxidation is rated as a relatively low importance parameter and can be modeled on a best estimate basis or ignored.

The power distribution is assumed to be the radial pellet power distribution and is weighted as an important parameter. The fuel model considers the conditions that do change during a REA transient in relation to affecting the radial pellet power profile. The radial pellet power profile is a strong function of pellet burnup and uranium enrichment.

Fuel burnup determines the amount of plutonium created in the rim of the pellet from ^{238}U resonance absorptions. At high burnups, the rim power can be twice as high as the average pellet power. As the plutonium is created on the rim, the plutonium power fraction is less in a higher enrichment pellet, and the surface power is smaller than a lower enriched pellet at the same burnup.

The initial enrichment and burnup for the pellet are initial conditions for the transient and the pellet radial power profile remains fixed during the transient. A typical or bounding fuel performance power history from an approved fuel performance code can provide this information and is acceptable for the REA. Sensitivity calculations are used to define the impact of this parameter.

Initial coolant conditions for inlet temperature, flow and pressure are defined by the initial power level and operational mode. These parameters are already defined conservatively for other safety analyses. Existing methods are applicable.

The transient core power and peaking factors are defined by the results generated from the plant transient analysis, which also includes the initial power distributions. Uncertainties applied to the REA power distributions are consistent with the current uncertainties applied for axial and radial peaking factors for other accidents.

Bounding fuel performance power history is obtained from an approved fuel performance code, and provides the heat resistances in fuel, gap, and cladding. Sensitivity calculations are performed to define the bounding conditions.

Cladding to coolant heat transfer coefficient for prompt critical power excursions is not considered a major contributor to the REA event. However, because the present methodology treats DNBR as a fuel failure criterion, transient cladding-to-coolant heat transfer becomes an important parameter.

The heat capacity of uranium oxide is primarily dependent upon temperature. Therefore, the local rod model requirement for heat capacity is the same as that used in the plant transient model. Section 4.1.9 of Reference 1 addresses the heat capacity as a noncritical parameter for REA when predicting temperatures and no uncertainty is needed. The transient water temperatures, local flows, and pressure are important to estimate fuel and cladding temperatures and DNBR of the fuel rods. An NRC-approved thermal-hydraulic computer code with time dependent capability is used with the approved uncertainties defined for licensing. The inlet temperature, core flow, and system pressure can affect the fuel rod transient analysis. The longer the transient is modeled (greater than 5 seconds) the more the system thermal-hydraulic conditions can impact the transient fuel rod model. Prompt critical excursions will not be impacted by the system thermal-hydraulic conditions because the maximum power deposition and maximum fuel temperatures are reached in less than a second. Nonprompt excursions may require modeling for more than a few seconds and the impact of plant conditions on the overall results is evaluated.

3.1.4 Fuel Rod Failure Analysis

Fuel rod failure in this analysis is based on the number of rods that violate the MDNBR safety limit. Thus, every rod that has a MDNBR less than some specified acceptable fuel design limit (SAFDL) is considered failed. In addition, by correlating the SAFDL with a particular value of peak rod power ($F_{\Delta H}$) and peak local power (F_Q) makes it possible to use these parameters to determine rod failure criteria. Since the CR-3 peak enthalpy rise is less than the 150 cal/g and the maximum clad temperature is below ballooning failure (SRP, Section 4.2, Appendix B), clad failure due to these mechanisms is precluded. The limit of the number of rods allowed to fail by these criteria is limited to the amount allowed by the radiological analysis, which is assumed to be 4.3 percent for the sample problem.

3.1.5 Evaluation and Conclusion

The discussion and analysis provided in this section covered the intended subject matter introduced in the opening paragraph of the same section. The NRC staff reviewed the applicant submitted analyses in accordance with the appropriate sections of the SRP, as stated above, and deemed the respective sections of the applicant submittal as acceptable.

3.2 Crystal River REA Methodology Description

The PIRT methodology described in Section 3.1 of this SE is applied to the CR-3 REA in this section. The primary computer codes utilized for the analysis of the CR-3 REA are all NRC-approved codes and cover fuel performance, 3-D space-time kinetics, and an open channel thermal-hydraulic capability. In particular the codes are COPERNIC, NEMO-K, and LYNXT, respectively. In addition, a system level code, RELAP5/MOD2 is used for those transients that last longer than 5 seconds. In the following sections the computer codes and any modifications to them are described and validated, the particular boundary conditions and uncertainty analysis specific to the CR-3 case is outlined, and bounding sample problems are presented and evaluated against the requirements of the current analysis, as well as bound future cycle design applications, respectively.

3.2.1 Overall Code Calculational Flow for the Ejected Rod Accident Evaluation

In this section the details of the linkages between the computer codes, the information being transmitted along the links, and the sequence of computer codes usage are discussed. The primary codes are NEMO-K, LYNXT, and RELAP5/MOD2 that cover the initial power pulse, thermal-hydraulic response of the most highly challenged fuel assemblies and rods, and the system response respectively. The computer codes CASMO and COPERNIC supply temperature, pressure, and composition dependent input to the above three codes. Before any transients are analyzed the static option of NEMO-K is used to determine initial boundary conditions. These conditions include ejected rod worth, DTC, MTC, β_{eff}, time-in-cycle, and power level.

Furthermore, REA transients are divided into two broad groups; those that are terminated by ex-core detector signals exceeding the high flux trip, and those that need to be terminated by system level trip signals.

3.2.2 Computer Codes

In this section those changes that have been introduced into the accepted codes to either improve their ability to more accurately reflect the phenomena that take place during an REA event, or that are specific to the CR-3 reactor are outlined. The three primary codes that have new applications are COPERNIC, NEMO-K, and LYNXT. These will be discussed below.

3.2.3 COPERNIC

COPERNIC is used to prepare input for both NEMO-K and LYNXT. It defines thermo-physical properties for the fuel, gap and clad material, including conductivity and specific heat. Fuel properties are a function of temperature and burnup. Clad properties are a function of the oxide film buildup on the surface, which is also determined by COPERNIC. Finally, the gap thermal properties, which are a complex function of burnup (composition of gap gas), gap size, surface temperatures of pellet and clad, and contact pressure (once the gap closes due to creep) are also determined.

NEMO-K and LYNXT use the constant gap geometry model, since it results in a more efficient numerical algorithm for solving the heat transfer equations. Thus, an appropriate multi-dimensional table must be created that preserves the functional dependence described above, while not varying the gap size. The complex functional dependence is reduced to three pertinent variables that capture all the dependencies implied above.

In order to create the multi-dimensional table as discussed in the above paragraph, COPERNIC is run statically, at different conditions to simulate different clad and fuel temperatures. These calculations are repeated for various burnup levels, which finally result in a complete table of gap conductivity that captures all the complex interactions in a desired format, for the entire cycle.

The above described code is typical code used by the licensees and vendors to analyze rod ejection events. The code mentioned in the above section is an NRC-approved code and was preapproved of use in analyzing the rod ejection event.

3.2.4 NEMO-K Plant Transient Model

The plant transient analysis is carried out by the NRC-approved NEMO-K computer code. It has been validated by comparing its solutions to six benchmark problems specifically designed to test REA code capabilities (Topical Report BAW-10221 PA, September 1998 (Reference 4)). These benchmark problems include three at hot zero power (HZP) and three at hot full power (HFP) conditions. Changes made to NEMO-K to either improve it or make it applicable to CR-3 are outlined below.

3.2.5 Trip Function

Ex-core power trip signals are used at CR-3 to sense severe reactivity insertion accidents and then subsequently scram the reactor. The trip function consists of two overall systems, the ex-core signals and the control rod drop models. The ex-core signals are generated by neutron detectors located outside the vessel in such a manner that the signal response differs when an asymmetric power shape exists. The trip criteria are set at 2/4 logic when trip signal is reached, and the rods are inserted. The rod insertion model used in NEMO-K is defined by safety analysis control rod position versus time from an input table.

- Ex-core detector model description: In the ex-core detector model, NEMO-K generates simulated signals by using assembly powers multiplied by an appropriate weighting factor to translate the in-core conditions to the ex-core signal. The weighting factors are determined by separate NEMO-K calculations or input from measured data. The weight factors include the effects of fuel assembly position relative to the detector, axial position along the assembly, and the position of the particular detector of interest (top or the bottom detector). In addition, the overall ex-core response is calibrated against measured thermal power. The ex-core detectors measure the fast flux exiting the core, and are calibrated to the actual conditions within the core. Briefly, the in-core assembly powers are multiplied by weighting factors to correlate in-core conditions to the ex-core signals.

- Control Rod Drop Model Description: The control rod drop model, used to determine the reactivity insertion during the scram phase of the transient, has five different steps. These steps include, initial acceleration, free fall, deceleration due to flow restriction, free fall within flow restriction, and finally stop at the bottom. Because rods could be at different positions at the beginning of the transient, each rod bank is treated separately.

3.2.6 Adiabatic Heat transfer Edit

An edit is added to NEMO-K to indicate the change in energy deposited over each time step. This method conservatively estimates the cal/g as defined for the reactivity initiated accident criteria since it neglects losses due to heat transfer.

3.2.6.1 Pellet Weighted Temperature for DTC

The most significant feedback mechanism to counter the positive reactivity input caused by a control rod ejection is the doppler feedback due to fuel heat-up and subsequent broadening of the fertile material absorption resonances. Thus, the determination of a representative fuel temperature as the transient progresses is of paramount importance. The cross sections are generated assuming a flat temperature distribution in CASMO, and the libraries are tabulated with a single temperature in mind. However, since the fuel temperature has a distribution that depends on burnup, an effective temperature is determined. The original model used in NEMO-K for determining a time dependent fuel temperature was a model that combined the fuel surface and centerline temperatures.

An improved weighted effective burnup dependent fuel temperature has been proposed that includes the volume-averaged temperature with the centerline and surface temperatures. In this manner information regarding the fuel temperature distribution is included in the effective temperature determination. An improved weighted effective burnup dependent fuel temperature has been implemented into the code proposed that includes the volume averaged temperature with the centerline and surface temperatures. In this manner information regarding the fuel temperature distribution with burnup is included in the effective temperature determination. The NRC staff considers this modification to the NEMO-K code as an enhancement, because it improves the code by making it more realistic.

3.2.7 NEMO-K Validation

The improved determination of T_{eff} has been validated by comparing results determined using both above outlined methods outlined to calculate an effective temperature to independent values determined using the APOLLO2 code. The APPOLLO2 code is capable of calculating neutronically reactivity behavior, fuel inventory and burnup in PWRs and BWRs. It was used by the licensee as another comparison check for the NEMO-K results. Briefly, the validation is carried out by comparing ^{238}U capture rate and reactivity for variable fuel temperatures to a constant temperature distribution within a pin. The value of the constant temperature that yields the same ^{238}U capture rate and reactivity as the variable temperature case is considered "T_{eff}" as determined by the APOLLO2 code. This value of T_{eff} is compared to values determined by the above two methods based on the variable temperature distribution used in the pin. A series of variable temperature distributions characteristic of steady state and transient conditions for a variety of burn-ups are used in the validation exercise. It has been found that for steady state fresh conditions all three values of T_{eff} are in good agreement. However, under all other conditions investigated it has been found that the new T_{eff} formulation agreed with APOLLO2 determinations of T_{eff}, while the Rowland formulation (Reference 7) deviated from the APOLLO2 value for transient cases.

Application of the above study to the transient being investigated indicates that for the case of fresh fuel, when the temperature distribution is expected to be parabolic, these two formulations give similar results, provided that the appropriate values are chosen for the respective weight factors. However, under transient conditions the original method underestimates the doppler effect. This deviation increases for transients starting with low initial reactor power. Thus, the largest difference in doppler effect is expected for transient initiated when the reactor is at hot zero power. Under these conditions the control rods are at or close to their maximum insertion, and over approximately 1.0 dollar of reactivity could be inserted upon ejection of the rod.

3.2.8 NEMO-K Summary

The licensee carried out the overall validation of the NEMO-K code by comparing results from determined by NEMO-K for benchmark problems to fine mesh reference solutions of the same problems. In all six cases including HZP (three) and HFP (three) were analyzed with very good agreement between NEMO-K and the reference solutions. Graphical comparisons provided in the licensee submittal (Reference 1), indicate very good agreement between NEMO-K and the reference solutions. The reference solutions are presented in the NEMO-K Topical Report BAW-10221 PA (Reference 4). In addition, spatial power distributions were compared on a detailed basis at selected times during the transient, to solutions obtained using the PANTHER code (Reference 4). Again the agreement between NEMO-K and the reference solution is very good. These comparisons show that NEMO-K is validated for carrying out REA analyses.

The NRC staff reviewed the modifications (improvements) made to NEMO-K and found them acceptable.

3.3 LYNXT Transient Fuel Rod Model

The transient fuel rod analysis is carried out by the NRC-approved LYNXT code. In this section the approved code is reviewed, and the improvements used to facilitate the analysis of REA events is outlined.

3.3.1 Overview of Existing LYNXT Fuel Rod Model

The approved version of the LYNXT code is based on a solution to the 2-D conduction equation, with radial and axial nodalization. Briefly, the code fuel rod model either uses the constant gap/constant properties (CG/CP), or the variable gap/temperature dependent properties (VG/TDP) representation. The latter model allows for the gap between the fuel pellet and clad to change during the transient, thus permitting conductivity of the gas to vary with burn-up and gap size.

3.3.2 Enhancements to the Fuel Rod Models

The enhancements to the NRC generically approved LYNXT code are two folds: first the number of solution locations is increased, and second implementation of a combination of the above two models, resulting in a constant gap/temperature dependent properties (CG/TDP) model. The increased number of solution locations allows for a more accurate representation of the radial power profile that can occur at higher burnup. The AREVA CG/TDP model, as applied to the CR-3 REA (Reference 7), requires that the following parameters be entered for in each burnup analyzed:

1. Fuel and clad behavior as a function of temperature
2. Specific heat for fuel and clad as a function of temperature
3. Gap conductance for the varying gap dimensions during the transient are accounted for by suitably varying the gap thermal properties in the tabular input
4. Fuel enthalpy as a function of temperature

The above model enhancements are improvements to the original code and should be more representative of the phenomena occurring during an REA, particularly with higher burnup.

3.3.3 LYNXT Benchmark Review

The LYNXT equations have not changed, only the input and input format. Thus, the previous code validation is still applicable. However, the changes indicated above need to be validated. In the following discussion the past validation are touched on and those analyses repeated with the new version of LYNXT.

- Past Qualification: The past validation of the LYNXT code as reviewed by the staff and accepted (Reference 5) was based on the constant gap/constant properties (CG/CP) and variable gap/temperature dependent properties (VG/TDP) models. It was found that LYNXT based predictions of agreed well with analytic solutions for the CG/CP models. The VG/TDP model could be compared to independent analyses for temperature distributions and Departure from Nucleate Boiling Ratio (DNBR) predictions. Comparisons between the VG/TDP model and TACO (Reference 7) for temperature distributions and RADAR for DNBR predictions agreed very well. These comparisons essentially validated the LYNXT code.

- The licensee repeated these analyses for the new CG/TDP model introduced in the code. It was found that the agreement was equally good for all parameters considered. The conclusion drawn by the licensee from this exercise is that the CG/TDP model is a validated addition to LYNXT.

- LYNXT-to-COPERNIC Example Cases: In this section the modified LYNXT will be compared to COPERNIC models. COPERNIC has much more comprehensive fuel, gap, and clad property models and thus the tabular form of the LYNXT model can be validated by this comparison. However, it should be pointed out the COPERNIC is not approved for the rapid transients characteristic of REA events, but the comparison will still be useful as a secondary check of the LYNXT CG/TDP model. Both codes used the same input for dimensions, imposed time variation of power, spatial power distribution, boundary conditions, and burnup profile. Four examples were analyzed that included HZP/BOL, HZP/EOL, HFP/BOL, and HFP/EOL, where BOL is for fresh fuel and EOL is for fuel conditions near discharge burnup. Comparisons were made for fuel surface, average fuel, fuel centerline, fuel maximum, and cladding maximum temperatures. Data provided in the licensee's submittal (Reference 1) demonstrates that the LYNXT consistently predict higher temperatures and, thus, yielding conservative results suitable for analyzing REA events and to predict fuel temperatures, clad temperatures and DNBR conditions.

3.3.4 LYNXT Conclusions

The revised CG/TDP model used in the LYNXT code has the advantage that its input requirements are more compatible with any fuel performance code. In summary the model allows specification of temperature dependent thermal properties for the fuel and clad, radial power profile across the fuel pellet, and gap conductance. The NRC staff reviewed the technical changes to the LYNXT code and the resulting consequences as they were presented in Reference 1, and found them acceptable.

3.3.4.1 System Thermal-Hydraulic Model

Typically, a thermal hydraulic system model is required for all those transients that are not terminated by a scram signal initiated by the ex-core detectors. In this case the temperature, pressure, and flow rate vary and are a function of the system response to the event. RELAP5/MOD2 is used for this purpose. For the CR-3 REA analysis, the RELAP5/MOD2 computer code (References 9 and 10) is used for this purpose. The only significant change being introduced in executing the code is to switch off the point kinetics option and use a power versus time history extrapolated from the NEMO-K calculation.

3.3.5 Conclusion

In this section of the SE, the changes that have been introduced into the previously NRC-accepted codes to either improve their ability to more accurately reflect the phenomena that take place during an REA event, or that are specific to the CR-3 reactor are outlined. The three primary codes with new applications are COPERNIC, NEMO-K, and LYNXT.

The new application in COPERNIC involves obtaining the gap thermal properties. The output is stored in a tabular form that can be readily accessed by the codes NEMO-K, LYNXT, and RELAP5/MOD2. The plant transient code NEMO-K has had the trip function improved and made compatible with the CR-3 reactor.

Finally, there were additional edits added concerning energy deposited in the fuel, and adjustment factors were added to facilitate sensitivity studies and the inclusion of contingency factors. Both NEMO-K and LYNXT have been validated against more accurate stand-alone codes. In the case of NEMO-K the validation for the effective temperature model was compared to the APOLLO and MCNP computer codes, and the power shapes were compared to the PANTHER computer code (Reference 4), to further validate the NEMO-K code results. In the case of LYNXT, the revised fuel rod model was validated against COPERNIC calculations. In addition, excluding the changes made to these codes as discussed in this section, these changes are consistent with the code version approved in Reference 10, and thus the staff deemed them acceptable.

The NRC staff concludes that the application and changes presented in the NEMO-K code as compared with the previously approved version in Topical Report BAW-10156A (Reference 5) more accurately represent the fuel temperature and provide an improved trip detection and control rod insertion model. Therefore, these modifications are approved for use in the REA analyses for the CR-3. This NRC staff's review only applies to the REA CR-3 analyses.

3.4 Application of Boundary Conditions and Uncertainties

Boundary conditions and uncertainties are required for all rod ejection analysis. These conditions and uncertainties requirement are necessary to analyze and bound the operating limits defined by a power range from HZP to HFP, and a cycle range from BOC to EOC. The CR-3 reactor average temperature versus power level has a transition at 20 percent power, thus at BOC zero (0) percent, 20 percent, and 100 percent power will be considered. There is no discontinuous behavior with respect to cycle time.

3.4.1 NEMO-K Boundary Conditions and Uncertainties

The application of conservatisms and uncertainties to the ejected rod worth, MTC, DTC, β_{eff}, fuel cycle burnup, and rod power peaking is addressed in the following sections. In general these parameters are set in the limiting direction to reflect the bound of a range of best estimate values adjusted by uncertainties, and an allowance for future cycles.

3.4.1.1 Ejected Rod Worth

The uncertainty of the ejected rod worth used in NEMO-K is 15 percent. In addition to this increase the rod worth is increased from nominal values to those used in the REA analysis by various factors depending on burnup and power level. At BOC and HZP and at HFP the worth is increased by approximately 70 percent, while at EOC and HZP the worth is increased by approximately 130 percent and at HFP the increase is approximately 60 percent. These increases were chosen to cover future cycles.

3.4.1.2 Moderator Temperature Coefficient (MTC)

The uncertainty of the MTC used in NEMO-K is 2 (per cent mil/degrees Fahrenheit) pcm/°F. In addition to this increase the MTC is increased from nominal values to those used in the REA analysis by various factors depending on burnup and power level. At BOC and HZP the MTC not only increases from the nominal value but changes sign from negative to positive. The increase in this case ranges from minus 4.0 pcm/°F to plus 2.5 pcm/°F depending on the equilibrium cycle chosen. At BOC HFP the MTC is increased from approximately minus 7.85 pcm/°F to minus 2.0 pcm/°F. At EOC and HZP the MTC changes from approximately minus 20.8 pcm/°F to minus 14.5 pcm/°F and at HFP the increase is from minus 34.5 pcm/°F to minus 26.0 pcm/°F. These increases are chosen to cover future cycles.

3.4.1.3 Doppler Temperature Coefficient (DTC)

The uncertainty of the DTC used in NEMO-K is 10 percent. However, measurements of the power coefficient, which include the doppler coefficient, have indicated a variation of approximately 20 percent. These measurements were carried out at BOC (fresh core) at HZP, and it is felt that this variation is due primarily to the DTC. The 20 percent difference is an under prediction of the measured coefficient. Thus, all REA analyses will be carried *will* be performed using DTC values that are approximately 20 percent more positive than the nominal values.

Other increases were chosen to cover future cycles.

3.4.1.4 Beta Effective

The uncertainty of the β_{eff} used in NEMO-K is 5 percent. However, in the REA analyses the nominal values are decreased by approximately 12 percent for all conditions. These decreases were chosen to cover future cycles.

3.4.1.5 Fuel Cycle Design

The bounding conditions are defined by cycles 18, 19, and 20, with each cycle being 24 months long. The nominal values for cycle 20 are used for the REA event analysis. The limits on enthalpy rise and MDNBR are not burnup dependent, and contingencies used for the key parameters are discussed in the above sections. Thus, parameters for future cycles can be compared to these parameters to verify their applicability.

3.4.1.6 Transient Power and Rod Power Peaking

There are a variety of components that contribute to the uncertainty of peaking factors. These include uncertainties due to the following:

1) Nuclear data
2) Hot Channel Factor (HCF)
3) Fuel Rod Bowing
4) Fuel Assembly Bowing
5) Burnable poison (Gadolinia)
6) Core Power
7) Grid Depression (F_Q only)
8) Quadrant Power Tilt

These factors are combined statistically (1 through 6) and deterministically (7 and 8) to yield uncertainties for $F_{\Delta H}$ and F_Q respectively. Additional penalties are included to account for future cycles.

3.4.1.7 Base Analysis Conditions

The base analysis conditions include the following:

1. Rate of reactivity insertion (100ms from full insertion to full ejection)
2. Reactor trip reactivity
3. Heat resistance of fuel, gap, and clad
4. Transient clad-to-coolant heat transfer coefficient
5. Heat capacity of fuel and clad
6. Fractional energy deposited in pellet
7. Pellet radial power distribution
8. Effective fuel temperature model

3.4.2 Sensitivity Calculations for Plant Transient Calculations

The licensee has presented two sensitivity analyses in its submittal dated February 26, 2009 (Reference 1). The first covers an analysis that was carried out for the US-EPR, and the second more abbreviated study is specific to the CR-3 reactor. In both cases the variation introduced in a parameter was to remove the uncertainties and repeat the analysis to determine the sensitivity to the change. In the first sensitivity change, the uncertainties are removed from the ejected rod (minus 15 percent), DTC (10 percent), β_{eff} (5 percent), and MTC (minus 2.0 pcm/°F) and the analysis was repeated. It was found that the largest effect was for those transients that started at the lowest power, and the effect got progressively smaller as the power increased. This trend is due to the fact that the conservatism is biased toward reactivity insertion phenomena, and thus since the reactivity insertion decreases with increasing power, the conservatism due to uncertainties decreases. The remaining parameters, which include rate of reactivity insertion, reactor trip reactivity, power peaking, coolant heat transfer, fractional heat deposition in fuel, pellet radial power profile, neutron velocity, time step, number of fuel rod nodes, and effective fuel temperature model are either conservative or not significant contributors to the prompt power response of an ejected rod event. The general conclusions from this study also apply to the CR-3 reactor analysis.

The licensee has repeated this analysis for the CR-3 reactor for the HZP cases at BOC and EOC. The same trend in the results was observed for these results as for the results discussed in the above section. If future cycles differ significantly from either of the two reactor studies outlined above, then the HZP cases will need to be repeated and reviewed by the NRC.

3.4.3 LYNXT Boundary Conditions and Uncertainties

In this section the boundary conditions and uncertainties of parameters applicable to LYNXT are discussed.

3.4.3.1 Pellet and Cladding Dimensions

The LYNXT model of the core uses an $1/8^{th}$ symmetry inherent in the core design. The model allows for twelve explicitly modeled fuel rods, the remaining nodes are modeled with decreasing level of explicitness. The geometry used for the temperature and enthalpy determinations is based on nominal dimensions for all cases. The uncertainties introduced by manufacturing etc. are included by an engineering hot channel factor. Axially the length of the cladding is extended to include the gas plenum located at either end of the core.

3.4.3.2 Radial Pellet Power Distribution

The pellet radial power profile is primarily a function of the burnup and initial enrichment. As the fuel is depleted the power profile increases the power peaking at the outer edge due to the build up of plutonium at this location. This dished power shape can be quite pronounced, with a peak/average of approximately two. The power profile remains invariant during the transient and is treated as an input quantity determined by burnup and initial enrichment.

3.4.3.3 Coolant Conditions

The coolant boundary conditions used in LYNXT are the inlet mass flux, and coolant temperature, and the system pressure. The system pressure is set at the core outlet condition, and then reduced by 65 pounds per square inch absolute (psia) to account for uncertainties. The minimum mass flux is reduced to allow for core by-pass flow, and then reduced by 2.5 percent to allow for uncertainties. In addition to these reductions the mass flux is reduced for individual fuel assemblies depending in location. The inlet temperature is determined by a heat balance performed in conjunction with the coolant average temperature as a function of power level. The inlet temperature is increased by 2 °F to allow for uncertainties. For transients that last for less than 5 seconds, these boundary conditions are held constant, and for longer transients they vary with time, as determined by the system level code RELAP5/MOD2.

3.4.3.4 Transient Power

Time dependent normalized axial power shapes and radial peaking factors are determined by the NEMO-K calculation and used in LYNXT. The DNBR performance is determined for one assembly that is analyzed in detail and is designated the "assembly of interest." The assembly of interest is assumed to have 13 heated zones surrounded by coarser volumes to represent the neighboring assemblies and the rest of the core.

3.4.3.5 Heat Resistance in Fuel Gap and Cladding

The determination of MDNBR depends on the heat transfer from the fuel, through the gap and clad to the coolant. It is also a strong function of the radial power shape in the fuel pellet. In general the MDNBR is lower for high burnup fuel since the gap conductance is higher and the "rim effect" biases the radial power shape to the outside of the pellet. This is illustrated by calculations that were performed at 5.0 w/o enrichment at 2.5 and 50 GWD/MT for BOC and 20 and 70 GWD/MU for EOC cases in order to bound the potential burnup and thermal property states of the fuel rods.

It was determined that rods containing gadolinia have a lower maximum temperature than uranium oxide rods, since they contain less uranium and the remaining gadolinium depresses the power. Thus, only uranium oxide properties will be necessary to define the bounding fuel rods.

3.4.3.6 Coolant Heat Transfer Coefficient and Transient Coolant Conditions

The minimum flow conditions as described above are used, and if the local DNBR is less than the design limit, the heat transfer coefficient switches from Dittus-Boelter to a correlation that includes the inception of film boiling and post-critical heat flux (CHF) conditions (BAW-10179 PA Revision 7, Reference 11). The SAFDL for DNBR used is the correlation limit with allowances applied.

3.4.4 Fuel Melt Limit

The uranium oxide melt temperature is a function of burnup. In cases were the maximum temperature is close to the outside rim of the pellet, the melting temperature limit must also account for local burnup within the pellet, which might be higher than the pellet average. During irradiation the pellet power distribution shifts from a peak to average of 1.06 for fresh fuel to maximally 2.5 for fully irradiated fuel. Thus, it is assumed that at the time of maximum pellet average burnup, the ratio will be no higher than 2.2. Therefore, using 70 GWD/MT as the average maximum burnup, the maximum rim burnup can be no higher than 155 GWD/MT, and the corresponding fuel melt temperature from COPERNIC minus uncertainties is used.

3.4.5 Failure Boundary Conditions

Any rod in the core (for core powers greater than 5 percent power), that has a MDNBR of less than the SAFDL, is considered failed. To quantify this mechanism the values of peak rod power ($F_{\Delta H}$) and the values of peak local power (F_Q) is subjected to a failure criterion defined by the ratio MDNBR/SAFDL equal to one. Thus, any rod for which $F_{\Delta H}$ or F_Q exceed these values is considered failed. The NRC staff reviewed the licensee's results presented in Reference 1 and found them acceptable because these results meet the SAFDL limit, which is conservative.

3.4.6 Evaluation and Conclusion

In section 3.4 and the subsequent sub-sections, the staff reviewed the licensee's submittal containing the uncertainties and the boundary conditions discussed above as they apply to the CR-3 reactor. The aim of the boundary conditions and applied uncertainties requirement is to analyze and bound the operating limits defined by a power range from HZP to HFP, and a cycle range from BOC to EOC. The CR-3 reactor average temperature versus power level has a transition at 20 percent power, thus 0 percent, 20 percent, and 100 percent power will be considered. Boundary conditions and uncertainties applied by the licensee to this analysis cover all the pertinent input parameters. These include ejected rod worth, MTC, DTC, β_{eff}, fuel cycle, and power peaking. In addition, uncertainties relevant to the fuel rod analysis are discussed, which include rod dimensions clad oxidation, pellet power distribution, heat resistance in fuel, clad, and gap, fuel melt variation with burnup, and the fuel failure method due to violation of DNBR conditions.

In letter dated May 29, 2009 (Reference 2), the licensee responded to the staff request for additional information (RAI)-01 regarding the change in the melt temperature of gadolinia containing fuel rods, which is summarized in the following paragraph.

The licensee stated that the melt temperature of gadolinia containing fuel is essentially the same as that of pure uranium dioxide based fuel, up to a gadolinia content of 8 weight percent. However, the thermal conductivity drops with increasing gadolinia content. This decrease in thermal conductivity is compensated for by proportionately decreasing the uranium enrichment in those fuel pins containing gadolinia. Thus, for the example presented in the submittal (Reference 1), (BOC HFP) the maximum fuel temperature in the gadolinia containing fuel pins is further from the melt limit than the pure uranium dioxide containing fuel pins. Further, the licensee's response pointed out that the gadolinia additions are to reduce the thermal conductivity of the fuel pellets. This reduction in thermal conductivity is accounted for in the COPERNIC code (section 4.3.3.2 of the NRC approved COPERNIC topical report) and is countered by reduced ^{235}U enrichments that lower the power level capability in the gadolinia fuel rods. The NRC staff reviewed the licensee's response to RAI-01, in particular Table RAI-01-1 that contained fuel temperature differences from fuel melt limit temperature, and found it acceptable.

An RAI concerning the relationship between post- and pre-ejection determined both dynamically and statically was the subject of RAI-02. The licensee's response to this RAI (Reference 2) is summarized in the following paragraphs.

The relationships of the pre-ejection to post-ejection curves for the dynamic vs. the static calculation presented in this section are benchmark examples, and are not specific to the CR-3 reactor. This type of relationship is used to determine the number of failed pins during the dynamic stage of the transient (0 seconds < time < 5 seconds), if the SAFDL limits for DNBR are violated. The need for this relationship for the CR-3 reactor will be discussed in the "Evaluation and Conclusion" section at the end of the next section.

The licensee, in response to RAI-02, addressed pre-ejection and post ejection curves as they pertained to CR-3. The licensee stated that such curves were not generated for CR-3 because none of the cases analyzed for CR-3 went into DNB. If analyses of future cycles for CR-3 show that DNB is possible, then pre-ejection and post-ejection curves will be generated to count the number of failed fuel rods in the bundles. The NRC staff reviewed the licensee's response to RAI-02 and found the result of the analyzed cases acceptable.

3.5 Description of CR-3 Sample Problem Results

In this section the application of the methodology described above is applied to the CR-3 reactor. Five trip functions are applied to the transients analyzed in this case. Each trip function has its own delay time, which is used to determine when the scram rods are inserted. These are given below (Reference 1):

1. Ex-core high flux, activated at 112 percent of reactor thermal power, delay 0.42 seconds
2. Low reactor cooling system pressure, activated at 1894.95 psia, delay 0.61 seconds
3. High reactor cooling system pressure, activated at 2400.0 psia, delay 0.61 seconds
4. High reactor coolant temperature, activated at 620 °F, delay 5.67 seconds
5. Variable low reactor cooling system pressure, function of hot leg temperature, delay 5.67 seconds

The overall series of calculations are shown in Table 1 below.

Table 1 – Series of calculations carried out for CR-3 Sample Cases

Time in cycle	Power level
BOC	HZP
BOC	20 percent
BOC	HFP
EOC	HZP
EOC	20 percent
EOC	HFP

3.5.1 NEMO-K Results

The above six transient cases were analyzed using NEMO-K, and results are shown for total core power, $F_{\Delta H}$ and F_Q. During these simulations core inlet conditions are held constant, and this is acceptable for those transients that are terminated by ex-core trip signals. However, those that are not tripped by ex-core signals need to be continued using the system code RELAP5/MOD2. Both HZP transients are tripped by ex-core signals, and both HFP cases need a system calculation to be terminated. In the case of 20 percent power, both the BOC and the EOC case are continued with a system calculation. Thus, in summary the two transients at HFP and 20 percent power are continued using the system code RELAP5/MOD2.

3.5.2 RELAP5/MOD2 Evaluation

The licensee conducted system level analyses to determine the effect of using constant inlet conditions, and operation without a trip due to ex-core detectors. In addition, the effect of a sudden increase in system pressure due to the sudden increase in reactor power will be estimated. Using the EOC HFP case as an example it was found that the sudden increase in pressure due to the power pulse was a small change in pressure, which had a negligible effect on the power and transient progression.

The conditions following no trip were examined using the system code RELAP5/MOD2. The licensee has assumed that the rod ejection generates a hole in the pressure vessel. Two different leak sizes are evaluated. The maximum size is assumed to be 2.765 inches in diameter (corresponding to the control rod flange), and a partial leak due to a smaller sized hole. The transients to be considered in this study are the 20 percent power case, and the HFP EOC case. The trip signals to terminate these transients were based on either high or low system pressure, high hot leg temperature or the variable low system pressure trip (VLPT).

The system level calculations are either performed with a bounding power vs. time variation, or carried out iteratively, iterating between NEMO-K and RELAP5/MOD2. The bounding power input to the system code is determined by carrying out several static power distributions using NEMO-K and thermal conditions from RELAP5/MOD2 to determine the limiting power that may be reached following the rod ejection. This value is used to carry out the system calculation. In the iterative approach the power history as calculated from NEMO-K is passed to RELAP5/MOD2, which determines the corresponding thermal-hydraulic conditions, which in turn are input to NEMO-K were the next power history is determined, and so on. This iterative procedure is continued until a converged neutronic and plant thermal hydraulic transient is reached and trip conditions are reached for the transient is terminated.

3.5.3 Description of LYNXT Results

The licensee used the results determined by both NEMO-K and RELAP5/MOD2 in LYNXT to determine the number of failed rods for each of the six transients considered. The LYNXT code also takes thermal-hydraulic input from the RELAP5/MOD2 calculation, and determines the variation of DNBR with time.

Results for the BOC HZP case indicates that MDNBR/SAFDL ratio is always well above unity, the relevant temperatures for fuel and clad are acceptable and the enthalpy added is also acceptable. In the 20 percent power BOC case, the MDNBR/SAFDL ratio drops below one at 8.3 seconds for the full leak and remains above unity for the partial leak. The 20 percent power EOC case did not violate the MDNBR/SAFDL ratio limit.

Results for the BOC and EOC HFP cases, the power level stabilizes at 106.4 percent and 104.0 percent, respectively. Under these power conditions coupled with a decreasing pressure and increasing temperature the MDNBR/SAFDL ratio drops below unity quite rapidly (before ~ 10 seconds into the transient). In this case the VLPT setpoint terminates the transient before the clad temperature reaches the ballooning failure limit. Based on the VLPT function (applied to the RELAP5/MOD2 output), the HFP transients would have tripped at 19 seconds for the full leak and 25 seconds for the partial leak.

The EOC HZP transient fuel pins reach 34 differential (Δ)cal/g, which is well below the failure criterion outlined in Section 3.4 of this SE. In addition, the peak radial average enthalpy is less than 55 cal/g. This event is terminated by rods full in at 3.5 seconds. This event is the limiting event. The 34 differential (Δ) cal/g, and the 55 cal/g values are well below the values stipulated in Appendix B of SRP, Section 4.2. Thus, this analysis is acceptable to the NRC staff.

In all cases the peak fuel temperature and clad temperature stayed within acceptable limits. The maximum fuel temperature was determined for the HFP BOC case with a partial leak at 25 seconds, which is the estimated time of the VLPT. The maximum clad temperature was determined to be 1436 °F, which occurred in the HFP EOC case at 25 seconds, which is the estimated time of the VLPT.

3.5.4 Rod Census

None of the assemblies experienced an enthalpy rise of more than 23 Δcal/g, only the delayed response was found to contribute to the failed rod census.

LYNXT was used to determine the power at which the limiting fuel rod meets the SAFDL limit. The $F_{\Delta H}$ and F_Q that correspond to this condition are used as failure criteria. The method described above in "Failure Boundary Conditions" section of this SE is used.

3.5.5 Summary Evaluation and Conclusion

In summarizing the CR-3 rod ejection event, the NRC staff concurs with the licensee that no rods failed during the power pulse, and failures that occur were due to violating the MDNBR limits after the pulse. In all cases there was no fuel melting, and the clad temperature remained below the incipient fuel melting condition ballooning temperature limit (SRP, Section 4.2, Appendix B). Hence, there are no coolability issues as outlined in Section 4.2 Appendix B of the SRP.

In this section the application of the methodology described in section 3.5,the above sections is applied to the CR-3 reactor. The six starting conditions for the six transients considered in this section of the SER are analyzed subject to five trip functions. The first of these analyses indicated that the HZP at BOC and EOC both are terminated within the first 5 seconds, and resulted in no fuel rod failures. The remaining four transients all extended into the time range that requires a system level scram signal, since the ex-core detector either did not reach the limits or they were ignored. Rod failures were determined to occur only for the BOC conditions starting at 20 percent power and HFP. No failures were determined at EOC.

Additional information was requested concerning the transient that started at 20 percent power at EOC. In this case a scram should have been initiated during the initial power peak. However, it was decided to treat it as a transient requiring system level scram initiation (RAI-03). The response of the licensee indicated that the late scram resulted in more conservative values for rod temperatures and enthalpy rise. Upon review, the NRC has accepted this conclusion. Furthermore, the fact that there are no rod failures during the initial power peak makes the need for the relationship, specific to CR-3, between pre- and post-ejection parameters for the dynamic and static calculations superfluous (RAI-02).
The response to an RAI (Reference 2) pertaining to ignoring the ex-core detector trip in the case of the BOC 20 percent power case were reviewed and analyzed by the NRC staff. Chapter 8 of the submittal alludes to the 20 percent power EOC case. The question was raised to help the staff interpreted the results of Table 8-1 of the submittal regarding the high flux trip point. The licensee pointed out that core power listed in the reference case, Table 8-6 of Attachment E, "ANP-2788P, Revision 0," in Reference 1 is the core power and is not the indicated power of the ex-core detectors. The CR-3 plant trip employs the indicated power from the measured signals of the ex-core detectors. In addition, the trip function depends on the ex-core signal response to the ejected rod event (Reference 2). The staff accepted the clarification.

3.6 Conclusions and Cycle Specific Checks

The February 26, 2009, submittal and associated attachments (Reference 1), outlines a methodology for carrying out reactivity insertion accident analysis, and illustrates it by carrying out several transients for the CR-3 reactor. The analysis in this submittal is considered a bounding analysis by CR-3 for the upcoming cycle and future cycles. Operating cycle 20 at CR-3 was used as the limiting case for the basis of the analysis presented above. The corresponding best estimate values for these parameters, before uncertainties and contingencies are added, are all within the values in Table 2. Consequently, if there should be a change in the cycle design one of three options are possible: (1) portions of the example analysis presented here can be repeated for each cycle, (2) the current analysis can be shown to be relevant to the new design, or (3) a complete reanalysis is carried out.

A check list given in the report is included here that new cycles or designs must meet to apply the above analysis.

Table 2– Ejected Rod Analysis Check-List

Parameter	Acceptability	BOC HZP	BOC 20 percent	BOC HFP	EOC HZP	EOC 20 percent	EOC HFP
Rod Worth (pcm)	Less than	715	556	60	741	535	73
β_{eff}	Greater than	.0058	.0058	.0058	.0048	.0048	.0048
MTC (pcm/°F)	Less than	2.5	0.0	-2.0	-14.5	-25.0	-26.0

DTC (pcm/°F)	Less than	-1.30	-1.24	-1.0	-1.40	-1.36	-1.20
Initial F_Q	Less than	N/A	3.48	2.53	N/A	5.37	2.25
Static F_Q After	Less than	14.84	8.88	3.07	27.23	12.70	3.73
Initial $F_{\Delta H}$	Less than	N/A	2.27	1.71	N/A	2.27	1.71
Static $F_{\Delta H}$ After	Less than	8.15	5.51	2.20	7.59	4.85	2.31
Equivalent rods failed (percent)	Less than	0	4.3	4.3	0	0	4.3

Operating cycle 20 at CR-3 was used as the limiting case for the basis of the analysis presented above. The corresponding best estimate values for these parameters, before uncertainties and contingencies are added, are all within the values in table 2.

3.6.1 Evaluation and Conclusion

In this section the applicability of the analysis presented in this report to other cycle designs is discussed. A checklist is presented that covers all the major parameters involved in an REA and their upper limits in order that the analysis presented in this report can be carried over directly to other cycles.

Important parameters that contribute to the behavior of the reactor following an REA vary monotonically between BOC and EOC, with the exception of the MTC. In general, it can be stated that an analysis of the core at BOC and EOC conditions should cover all possible variations. However, due to the fact that the MTC could vary in a non-linear fashion for some core design the MTC could be at a maximum between the BOC and EOC burnup states. The current design process requires that the analysis be repeated at the point of maximum MTC and compared to the analysis at BOC conditions. If the design were found to be not acceptable (compared to the check list of parameters mentioned above), it would be rejected. It is also possible to carry out a re-analysis with new limiting conditions.

In RAI-04 the staff raised the issue of ensuring that the most severe transient response is included in the BOC and EOC cases studied in the submittal. Information in the licensee's response to RAI-04 indicated that the input parameters vary monotonically between these two limits, and thus no intermediate burnup level would result in a more severe condition. The response has been reviewed and accepted by the staff and included in the evaluation, thus resolving the issues raised by this RAI.

The licensee also pointed out that the MTC is the only coefficient that does not change monotonically with burnup. In addition, the current core design process does exam the cycle lifetime for maximum MTC, and if the maximum MTC occurs later than the beginning of cycle (BOC), the ejected rod checks would be performed at that burnup and compared with BOC conditions (Reference 2). The staff concurs with response information in the applicant's response to RAI-04 has been included in the evaluation, and resolves the issues raised by this RAI.

3.7 Overall Conclusion

The licensee's methodology used in the CR-3 rod ejection accident analysis is based on a suite of computer codes, namely: COPERNIC, NEMO-K, LYNXT, and RELAP5/MOD2. The linkage between these computer codes and the failed rod census are based on a methodology developed by AREVA and ANP (Reference 3). These NRC-approved codes have been modified to either improve their solution algorithm, or make them more consistent with the CR-3 reactor. These modifications include:

- The new application of COPERNIC involve generating gap conductance at various temperature conditions and fuel burnup. The output is converted to a tabular form that can be used in the codes NEMO-K and LYNXT.
- The revised effective temperature model and the trip detection and control rod activation model used in NEMO-K are approved for use in reactivity initiated accident analyses,
- The plant transient code NEMO-K has had the trip function improved and made compatible with the CR-3 reactor. In addition, the effective fuel temperature model in NEMO-K has been improved. The improved effective temperature model is validated by comparing it to the APOLLO and MCNP computer codes.
- In the case of LYNXT the revised fuel rod model is validated against COPERNIC calculations.
- The modified CG/TDP model used in LYNXT is approved for use in reactivity initiated accident analyses,
- The transient phase fuel rod failure census model, based on a linear relationship for the ratio of post-to-pre-ejection fuel rod power determined by transient calculations and static calculations, is acceptable for rod ejection accident analyses.

The NRC staff has reviewed the CR-3 submittals, dated February 26, and May 29, 2009, and has found acceptable the CR-3 analysis of the rod ejection event where the licensee demonstrated compliance with SRP requirements. Compliance with the SRP, Section 4.2, Appendix B requirements will ensure preventing of postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary. Based on this review, the NRC staff concluded that the CR-3 analysis of the rod ejection is acceptable.

Finally, the approval of the application of the methodology as described in the above submittals (References 1 and 2) is plant-specific. That is, the approval of this methodology is specific to the CR-3 reactor. In addition, the use of the reviewed and approved changes to the code suite mentioned in these submittals is specific to the CR-3 reactor.

Additionally, the licensee in its submittal dated February 26, 2009, proposed deletion of the CR-3 Operating License (OL) Condition 2.C.(12), which identified specific vendor documents that were used in developing the Cycle 14 Core Operating Limits Report. The licensee stated that this one cycle condition had become obsolete, since those specific documents were merged into BAW-10179PA, "Safety Criteria Methodology for Acceptable Cycle Reload Analyses," that was approved by the NRC in Amendment 211 (ADAMS Accession No. ML032930435). The NRC staff reviewed this amendment and concluded that the OL condition is no longer required.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (74 FR 22179). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Crystal River Unit 3 - License Amendment Request #307, Revision 0, Methodology for Rod Ejection Accident Analysis Under Extended Power Uprate Conditions, February 26, 2009, Agencywide Document Access and Management Systems (ADAMS) Accession No. ML090700533)

2. Crystal River Unit 3 - License Amendment Request # 307, Revision 0: Methodology for Rod Ejection Accident Analysis Under Extended Power Uprate Conditions - Response to Request for Additional Information, May 29, 2009 (ADAMS Accession No. ML091530015).

3. "U.S. EPR Rod Ejection Accident Methodology Topical Report", AREVA NP Inc., ANP-10286P Revision 0, November 2007.

4. "NEMO-K A Kinetics Solution in NEMO," Framatome Cogema Fuels, BAW-10221-PA, September 1996.

5. "LYNXT Core Transient Thermal-Hydraulic Program," B&W Fuel Company, BAW-10156A, Revision 1, August 1993

6. BAW-10228PA, "Science," Framatome Cogema Fuels, December 2000.

7. NEACRP-L-335 (Revision 1), "NEACRP 3-D LWR Core Transient Benchmark," Final Specifications, H. Finnemann and A. G. Galati, October 1991 (January 1992).

8. BAW-10087P, "TACO – Fuel Pin Performance Analysis," Babcock & Wilcox, December 1975.

9. NP 43-10193PA-00, RELAP5/MOD2-B&W For Safety Analysis of B&W Designed Pressurized Water Reactors.

10 NP 43-10164PA-06, RELAP5/MOD2-B&W An Advanced Computer Program For Light Water Reactor LOCA and Non-LOCA Transient Analysis.

11. BAW-10179 PA, Revision 7, "Safety Criteria and Methodology for Acceptable cycle Reload," January 2008.

Principal Contributor: Anthony Attard

Date: January 28, 2010

Mr. Jon A. Franke, Vice President
Crystal River Nuclear Plant (NA1B)
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15760 W. Power Line Street
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SUBJECT: CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT REGARDING ADOPTION OF A NEW METHODOLOGY FOR ROD EJECTION ACCIDENT ANALYSIS UNDER EXTENDED POWER UPRATE CONDITIONS (TAC NO. ME0730)

Dear Mr. Franke:

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-72 for Crystal River, Unit 3 (CR-3) in response to your application dated February 26, 2009, as supplemented by letter dated May 29, 2009. The amendment approves a new methodology, developed by AREVA NP for CR-3, to analyze the rod ejection accident under extended power uprate conditions. The adoption of the new methodology is reflected in a change to the CR-3 Operating License and Improved Technical Specifications (ITS). The CR-3 ITS Section 5.6.2.18.b is being revised to add this new methodology to the list of approved methods used in developing the Core Operating Limits Report. Additionally, Operating License Condition 2.C.(12), which was a one-cycle license condition, is being deleted.

A copy of the safety evaluation is enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 237 to Facility Operating License No. DPR-72
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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