SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
TOPICAL REPORT ANP-10286P, REVISION 0
"U.S. EPR ROD EJECTION ACCIDENT METHODOLOGY TOPICAL REPORT"
AREVA NP, INC.
DOCKET NO. 52-020

# 1.0 INTRODUCTION

## 1.1. SUMMARY

By letter dated November 20, 2007 (Agencywide Documents Access and Management System [ADAMS] Accession Number ML073310620), as supplemented by letters dated July 10, 2008 (ADAMS ML081970349), October 3, 2008 (ADAMS ML082880500), and March 26, 2009 (ADAMS ML090890175), AREVA NP, Inc., (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval Topical Report (TR) ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report" ADAMS ML073310629 (proprietary), ADAMS ML073310622 (nonproprietary) [1]. This Safety Evaluation Report (SER) is based on the submitted Licensing Topical Report, the information obtained during a number of meetings and conference calls with the applicant, and formal requests for additional information (RAIs).

In the document ANP-10286P [1], the applicant describes a method of analyzing the consequences of a control rod ejection accident (REA) for the U.S. EPR. The methodology is based on a 3-D nodal kinetics solution with both thermal-hydraulic and fuel temperature feedback. In addition, there is a separate peak rod thermal evaluation using an open channel thermal-hydraulic model. This methodology includes updated models for determining the trip detection, control rod insertion and effective fuel temperature.

The review is carried out in conformance with the regulatory guidance as summarized in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) [2] Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," and SRP Section 4.2, "Fuel System Design," Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents."

The rod ejection accident is analyzed using two primary codes, NEMO-K [3] and LYNXT [3]. The former is used to carry out the 3-D kinetic calculation while the control rod is being ejected from the core, and LYNXT (coupled with COPERNIC [3]) is primarily used to determine the number of failed rods. In addition, it is necessary under certain conditions that a system level code, in this case S-RELAP [3], be used to determine the state points in the core, which are used to determine the number of failed rods. Rod failure is defined by either the addition of, at least, 110 cal/g (197.4 BTU/lb) during the initial transient, or if the specified acceptable fuel design limit (SAFDL) limits are violated on the value of the minimum departure from nucleate boiling ratio (MDNBR).

The staff's evaluations presented in this report are divided into several sections that cover: (1) Code and Code Updates Descriptions, (2) Selected Transient Responses, and (3) Independent Validation. Summaries of these sections are provided as follows:

1) Code and Code Updates Descriptions – Sections 3.3 and 3.4 of this report include short descriptions of the codes used in the applicant's analysis as well as a description of the updates (if applicable). These include CASMO-3G, COPERNIC, NEMO-K, and LYNXT. These descriptions are followed by a discussion of the changes made to NEMO-K and LYNXT since the submittal of ANP-10263P, "Codes and Methods Applicability Report for the U.S. EPR" [3]. NEMO-K has an improved trip detection and control rod insertion model that mimics the behavior of the actual trip signal processing closely. In addition, the rod insertion model allows for rod acceleration and slowing down at either end of its stroke. An improved determination of the effective fuel temperature is included in NEMO-K. This is particularly important since this temperature controls Doppler feedback, which is the most important shutdown mechanism in the near term. The improved effective temperature model is validated against calculations carried out using APOLLO2 and MCNP. The changes introduced in LYNXT involve an improved gap thermal property model.

2) Selected Transient Responses – Section 3.5 of this report describes the categories of transient responses analyzed by the applicant and provides the staff's evaluation. There are two transient categories in the REA scenarios. The first are those transients that are terminated within approximately the first 5 seconds (sec.) by the ex-core detectors; the second are those transients that carry on until they are terminated by a system level trip signal. One of each of these transients is evaluated by the staff. In addition, the method of determining the number of failed rods is reviewed by the staff.

3) Independent Confirmatory Analysis – Section 3.6 of this report presents two sets of confirmatory analyses used by the staff to support the safety evaluation. The first set of confirmatory analyses involved the staff using an independent code package to analyze the limiting cases presented by the applicant. The second set of confirmatory analyses involved a comparison of the staff's analysis of the Special Power Excursion Reactor Test (SPERT) using TRITON/TRACE [4,5] to the applicant's NEMO-K based analysis.

## 1.2. DESCRIPTION OF A GENERIC ROD EJECTION ACCIDENT TRANSIENT EVENT

Rod Ejection Accidents are a class of accident transients that pressurized water reactor (PWR) vendors are required to analyze in order to meet the requirements of General Design Criterion (GDC) 28, "Reactivity Limits," (as described in SRP Section 15.4.8) in order to obtain an NRC license for a particular reactor design. The following discussion is based on descriptions presented in the Topical Report.

The accident is initiated by the sudden ejection of a control rod from the core of a critical reactor. Initially, the reactor can be at hot full power (HFP), or hot zero power (HZP), and, in addition, the core could be at beginning of cycle (BOC) or end of cycle (EOC). Thus, a total of at least four different combinations exist to be analyzed. Partial power situations might be considered in particular cases to explore bounding conditions. Furthermore, the possibility that simultaneous depressurizations of the primary coolant system occurs must be considered, since the ejection of the rod could damage the pressure vessel and create a coolant leakage path. In general, there are a large number of initial conditions that can affect the transient response, and its ultimate termination.

In a typical REA, a control rod is rapidly ejected and accelerated by the system pressure, resulting in a step change in reactivity. The sudden addition of reactivity results in a corresponding increase in power and fuel temperature. The only feedback mechanism that can counter this power increase is the Doppler Effect associated with the fertile component of the fuel ($^{238}U$). As a result of the increase in power, the fuel temperature increases and the Doppler feedback becomes progressively more negative until it reverses the power increase, resulting in a typical power pulse. Finally, the ex-core power detectors trip the scram system and the transient is terminated. The duration of the transient is approximately 5 sec., which is short enough to ignore all system related changes to the coolant temperature and pressure.

However, if the time frame is long enough that system level thermal hydraulic changes are significant, a second transient type results. This can occur at HFP when the control rods are mostly withdrawn. In this case, the power increase is comparatively small, causing a small amount of negative Doppler feedback and, thus, a small pulse followed by a slow increase in reactor power. In addition, the primary system boundary may be compromised due to the ejected rod creating a small break loss of coolant. In this case, a system level response is necessary, since transient termination will be due to activation of reactor trips associated with system response.

Two important results from a REA analysis are of primary interest. First, the number of fuel rods that have failed as a result of the transient, and second, whether or not any of the regulatory requirements have been violated (see Section 2 of this report). Two rod failure mechanisms are important in REA transients:

1) Those that occur during the initial power pulse, caused by pellet clad mechanical interaction (PCMI).

2) Those due to fuel clad failure caused by violation of the Minimum Departure from Nucleate Boiling Ratio (MDNBR) limits. There is an accepted limit (to be discussed below) known as the specified acceptable fuel design limit, which will be referred to in connection with the MDNBR.

If none of the regulatory requirements are violated, the transient analysis is considered complete. However, if there is any violation of these regulatory requirements, it is necessary to re-configure the core design or reactor system to ensure compliance.

## 2.0 <u>REGULATORY CRITERIA</u>

### 2.1. REQUIREMENTS

The applicant submitted ANP-10286P [1] in order to support the rod ejection analysis summarized within the U.S. EPR Final Safety Analysis Report (FSAR). As such, the regulatory requirements and guidance outlined by SRP Section 15.4.8, “Spectrum of Rod Ejection Accidents (PWR),” and SRP Section 4.2, “Fuel System Design,” Appendix B, “Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents,” were used by the staff in the review of this topical report. These requirements concern cladding failure, coolability, and radiological release. Radiological release was not addressed explicitly in ANP-10286P [1], so this requirement must be addressed when an applicant applies the methodology presented in ANP-10286P [1] to the review of a rod ejection event. The following is a summary of the applicable criteria.

GDC 13, “Instrumentation and Control,” requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

GDC 28, "Reactivity Limits," assures that the effects of postulated reactivity accidents can neither damage the reactor coolant pressure boundary nor result in sufficient disturbance to impair the core cooling capability.

## 2.2. RELEVANT GUIDANCE

SRP Section 4.2, Appendix B provides the interim acceptance criteria and guidance for Reactivity Initiated Accidents, of which the Rod Ejection Accident is a subset. By following the provided guidance, an applicant demonstrates compliance with GDC 28. A brief description of the guidance from SRP Section 4.2, Appendix B is provided below:

1) Cladding failure

   The PCMI due to the sudden rise in power during the pulse phase of a REA requires a limit regarding energy (cal/g) as a function of clad thickness (clad thickness change due to oxidation). The oxide thickness increases with burnup. The applicant has estimated that for a maximum burnup of 62 GWD/MT, the oxide thickness/wall thickness is 0.061, which implies an energy deposition limit of 110 cal/g (198 BTU/lb) for PCMI failures to occur. Additionally, clad failures can occur in cases which the pin internal pressure is below system pressure when the total enthalpy exceeds 170 cal/g (306 BTU/lb), and for cases that the pin internal pressure is above system pressure when the total energy exceeds 150 cal/g (270 BTU/lb). Both of these limits apply for core power levels below five percent. Finally, violating the thermal design limits for all power levels above five percent is assumed to lead to clad failure. These requirements, as they relate to SAFDLs are captured in GDC 10, "Reactor Design."

2) Coolability

   Pin cooling is assumed failed for all pins with a total enthalpy of 230 cal/g (414 BTU/lb). In addition, pin cooling is assumed to fail in cases in which there is incipient fuel melting. Furthermore, cooling failure will occur for all cases in which there is a failure to preserve the reactor pressure boundary, reactor internals, and fuel assembly structural integrity. Finally, a loss of coolable geometry will result following clad and fuel fragmentation, and clad ballooning. These requirements are captured in GDC 27, "Combined Reactivity Control Systems Capability," and GDC 35, "Emergency Core Cooling," as they relate to control rod insertability and core coolability.

3) Radiological Impact

   Radiological impact will not be explicitly discussed in this report. However, SRP Section 4.2 Appendix B provides guidance related to the calculation of fission product inventory that would be available after an event. This inventory is to include both the steady-state gap inventory (further guidance is provided in RG 1.183 and RG 1.195) as well as fission gas released during the event. SRP Section 4.2 Appendix B provides a correlation between gas release and maximum fuel enthalpy increase that can be used to calculate the transient fission gas release.

The above limits summarize the guidance used to demonstrate compliance with GDC 28, which must be met in carrying out the analyses outlined in Section 3.2, Figures 2 and 3 of this report. These limits are used at the decision points regarding fuel temperature and cal/g determinations, and the number of failed rods that imply unacceptable radiological release.

## 3.0 SUMMARY OF TECHNICAL INFORMATION

In this chapter, the applicant's methodology is summarized and the codes used by the applicant in the methodology are briefly described, including their input, output, and analytic modeling.

### 3.1. OUTLINE OF ROD EJECTION ACCIDENT PHYSICAL PHENOMENA, MODELING, AND OVERALL METHODOLOGY

In this section, an outline is given of the various physical phenomena that govern the progression of an REA transient, and the implied requirements that are placed on the numerical algorithms to be used. Broadly, the initial response of the core to an REA is generally a skewed increase in the power, which severely impacts the fuel temperature and cooling of the core in selected assemblies. In all these analyses, temperature-dependent cross-section data and temperature- and pressure-dependent thermo-physical properties are necessary in order to model the event accurately.

The core power shape is most accurately determined by using a 3-D space-time kinetic calculation. In the applicant's model, the cross-section data is a function of temperature, burnup, and composition.

The fluid dynamics and heat transfer calculations used by the applicant were carried out on the most highly challenged fuel assemblies recognizing every fuel rod, and allowing for both axial and transverse flow. This calculation takes input from the 3-D kinetics calculation or the system level calculation, and the variable thermo-physical data used in the above calculation is also used in this analysis. The primary output from this analysis is the number of failed rods, either due to PCMI or due to violation of the SAFDL limits.

Finally, for those transients that go on for more than approximately 5 sec., a system level code is required to determine the coolant temperature and pressure, and to determine any phase change in the cases where the pressure is dropping. This code takes input from the 3-D kinetics code NEMO-K, and the variable thermo-physical data used in the above calculations. The system level code traditionally uses a point kinetic model, but in this application this model is turned off, and a power versus (vs.) time characteristic is obtained from the 3-D kinetic calculation and used as a driving function.

Rod failure due to PCMI is determined by a threshold value of enthalpy deposited per gram of fuel material (cal/g) in conformance with SRP Section 4.2, Appendix B. All rods that exceed this limit are assumed failed and occur during the initial power pulse phase of the transient. In the longer term, additional rod failure may occur due to violation of the SAFDL limit (MDNBR < SAFDL). Rods that fail and have an energy deposition above 31.2 cal/gm are ascribed a higher radiological worth than rods that fail below this value. This enhancement in worth is recognized by multiplying the number of rods by a transient fission gas release (TFGR) factor, which is a function of the enthalpy rise (threshold of 31.2 cal/gm), to arrive at an equivalent number of rods failed. Thus, to obtain the total number of rods failed for radiological release consideration, those that fail and have a prompt enthalpy rise above 31.2 cal/gm are multiplied by a TFGR factor and added to the number of rods failed below this value. The radiological consequences of this total number of failed rods is compared to the regulatory requirements of 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," or 10 CFR 50.67, "Accident Source Term," as indicated in SRP Section 15.4.8 to determine acceptability. Figure 1 of this report illustrates the methodology used by the applicant to analyze the REA event.

The starting point is determined by the design for fuel pins, which are assembled into a fuel assembly, and which finally are combined to form the reactor core. Following this step, the state of the reactor before the rod ejection accident is defined; this includes burnup, power level, location of rod to be ejected, system state points, etc. In addition, the regulatory limits as listed in Section 2 of this report are defined, and the code suite to be used is selected based on the expected transient. 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 above 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 initiated. The details of this step are discussed in Section 3.2 of this report, in which the detailed inter-linkage of the various codes and associated information are described. Following the analysis step, a decision is made regarding whether or not the core design under consideration can successfully survive the imposed transient. If any of the regulatory requirements listed in Section 2 of this report are violated, the reactor core needs to be re-configured and the process repeated until all requirements are satisfied. If the transient is completed without violating any regulatory requirements, the analysis is considered complete (see Section 2 of this report).

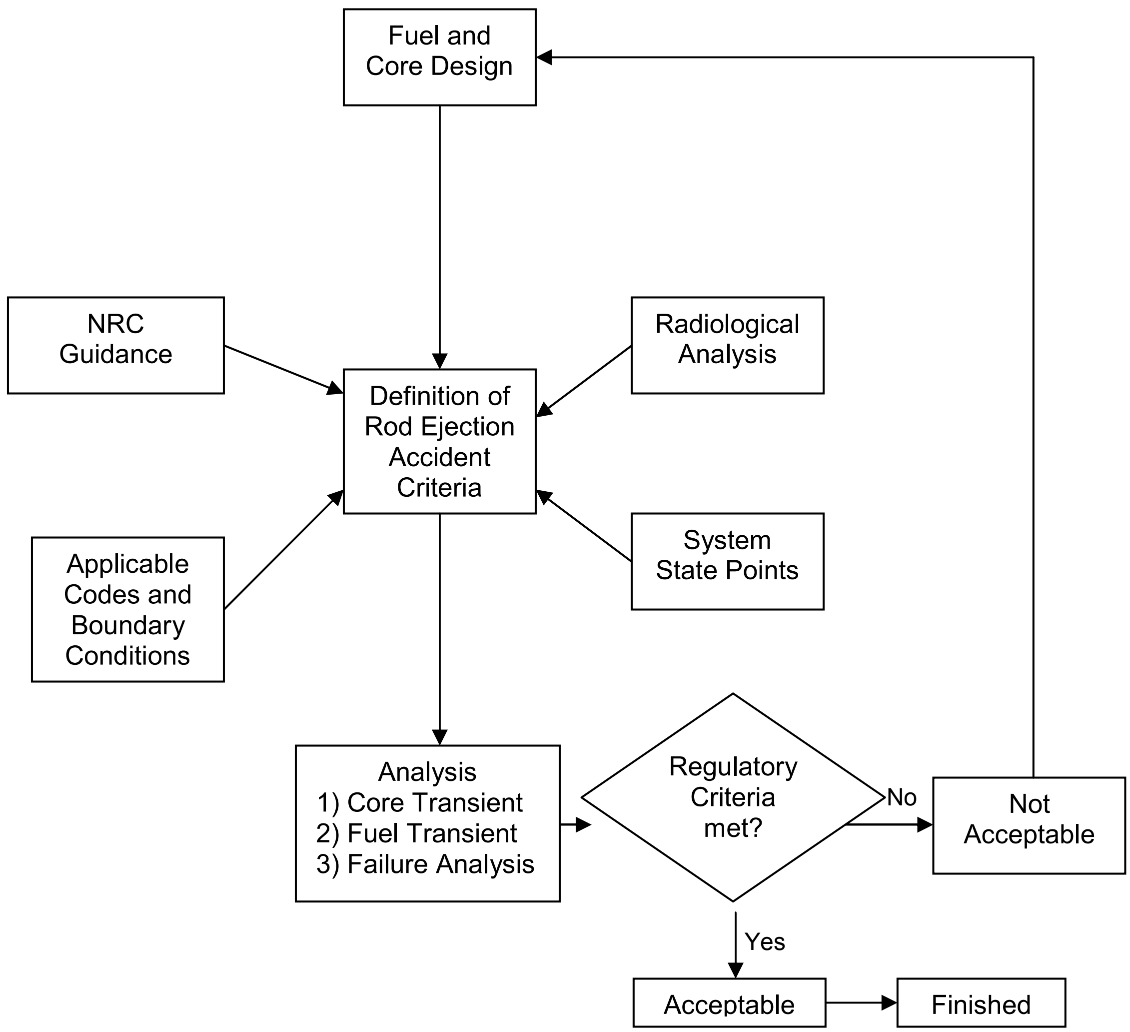


**Figure 1 – Overall Core-Fuel Rod Ejection Accident Analysis Methodology.**

## 3.2. DETAILS OF ANALYSIS METHODOLOGY

This section summarizes the linkages of the codes, the information being transmitted along the links, and the sequence of codes used by the applicant in ANP-10286P [1]. The code linkages are shown in Figures 2 and 3 of this report. The primary codes are NEMO-K, LYNXT, and S-RELAP, which cover the initial power pulse, thermal-hydraulic response of the most highly challenged fuel assemblies and rods, and the system response, respectively. More detailed descriptions of the codes, particularly NEMO-K are given in Section 3.3.3 of this report. The codes CASMO-3G and COPERNIC supply temperature, pressure, and composition dependent input to the above three codes. Furthermore, as outlined above, REA transients are divided into two broad groups: (1) Those that are terminated by ex-core detectors within approximately 5 sec.; and (2) those that need to be terminated by system level trip signals and extend beyond 5 sec. The linkages shown in Figures 2 and 3 of this report for these two time frames are:

1) The first time frame is defined as 0 – 5 sec., Figure 2 of this report. NEMO-K is used to calculate the core power pulse, with input from CASMO-3G and COPERNIC.

CASMO-3G is used to supply temperature-dependent cross-section data, and COPERNIC calculates thermo-physical data and pin power shape. If the scram system is activated, the transient is over and the analysis proceeds to the LYNXT step to determine the number of failed rods. LYNXT takes input from COPERNIC and NEMO-K to determine the fuel temperature and the cal/g added to the fuel. If these parameters are outside acceptable limits, then the design needs to be modified. However, if they are within limits, the MDNBR needs to be checked by comparing to the SAFDL limits. If the MDNBR is greater than the SAFDL limit, then no fuel failure is implied for the assumed transient. However, if the MDNBR is below the SAFDL limit, an estimate of the number of failed fuel rods needs to be made. The most accurate manner of determining the number of failed pins would be to obtain a rod-by-rod power distribution vs. time from a 3-D neutronics and thermal-hydraulic calculation to determine the MDNBR, which can then be compared to the SAFDL limit. However, this option is not realistic and, thus, an acceptable practical approximate method is necessary. In Section 3.3 of this report, the method used by the applicant is outlined. 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 as discussed in Section 2 of this report. 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 re-designed.

2) The second time frame (greater than 5 sec. in Figure 3 of this report) begins after the coolant has made a complete circuit of the primary system. Thus, the system code S-RELAP5 is used to generate input for LYNXT in order to determine the number of failed rods. The S-RELAP5 code is driven by a total power input derived from NEMO-K. S-RELAP5 has the major system scram functions built in and, thus, eventually 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 short transient 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 of this report. 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 analysis is considered acceptable and complete; if it exceeds the limit, then the core needs to be re-designed.

Figures 2 and 3 of this report show that there are three points following in which the transient is terminated in an acceptable manner, and two terminations that imply that the reactor needs to be re-designed because it failed one of the regulatory requirements.

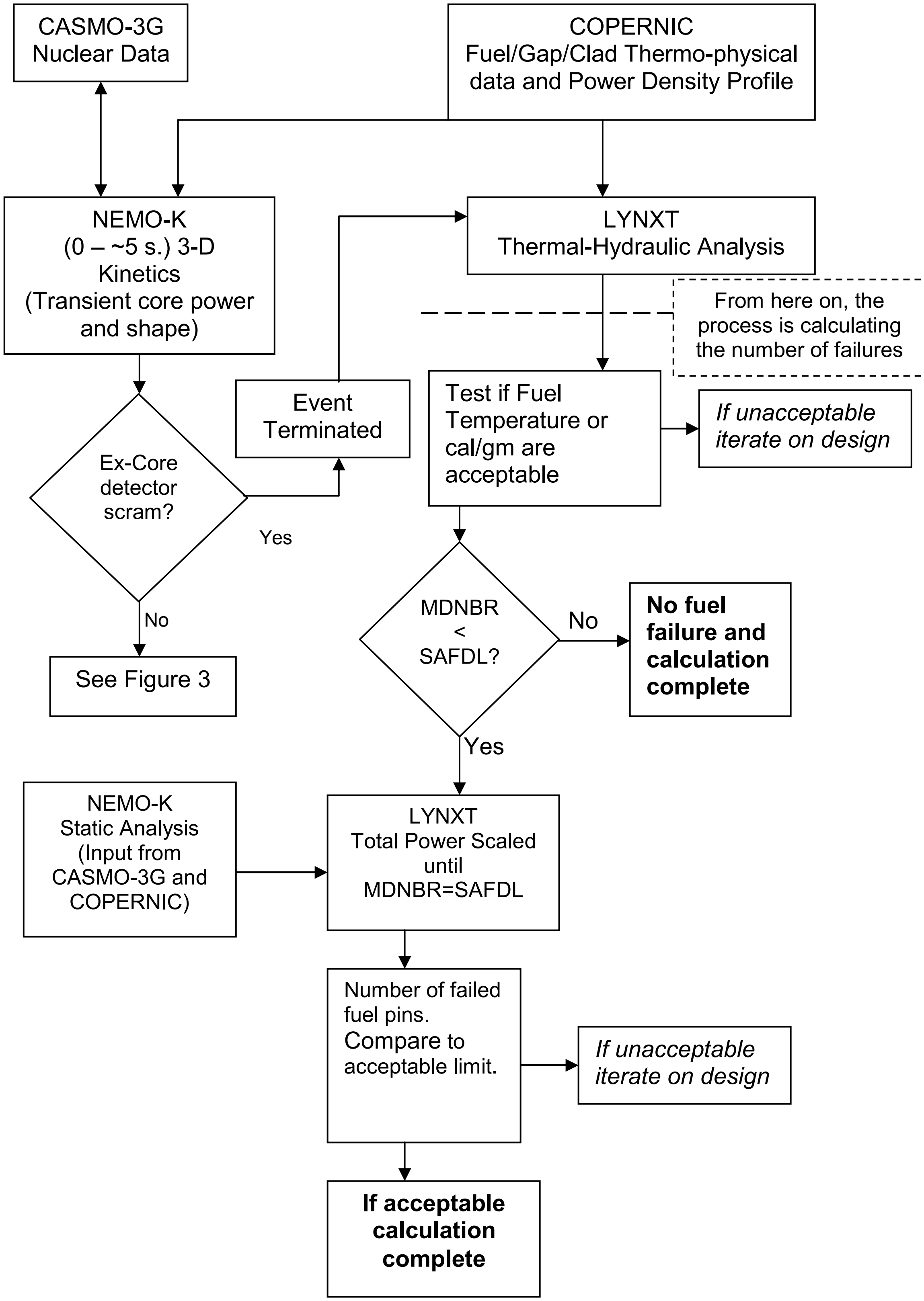


**Figure 2 – Details of Analysis Methodology (0 - ~ 5 sec.)**

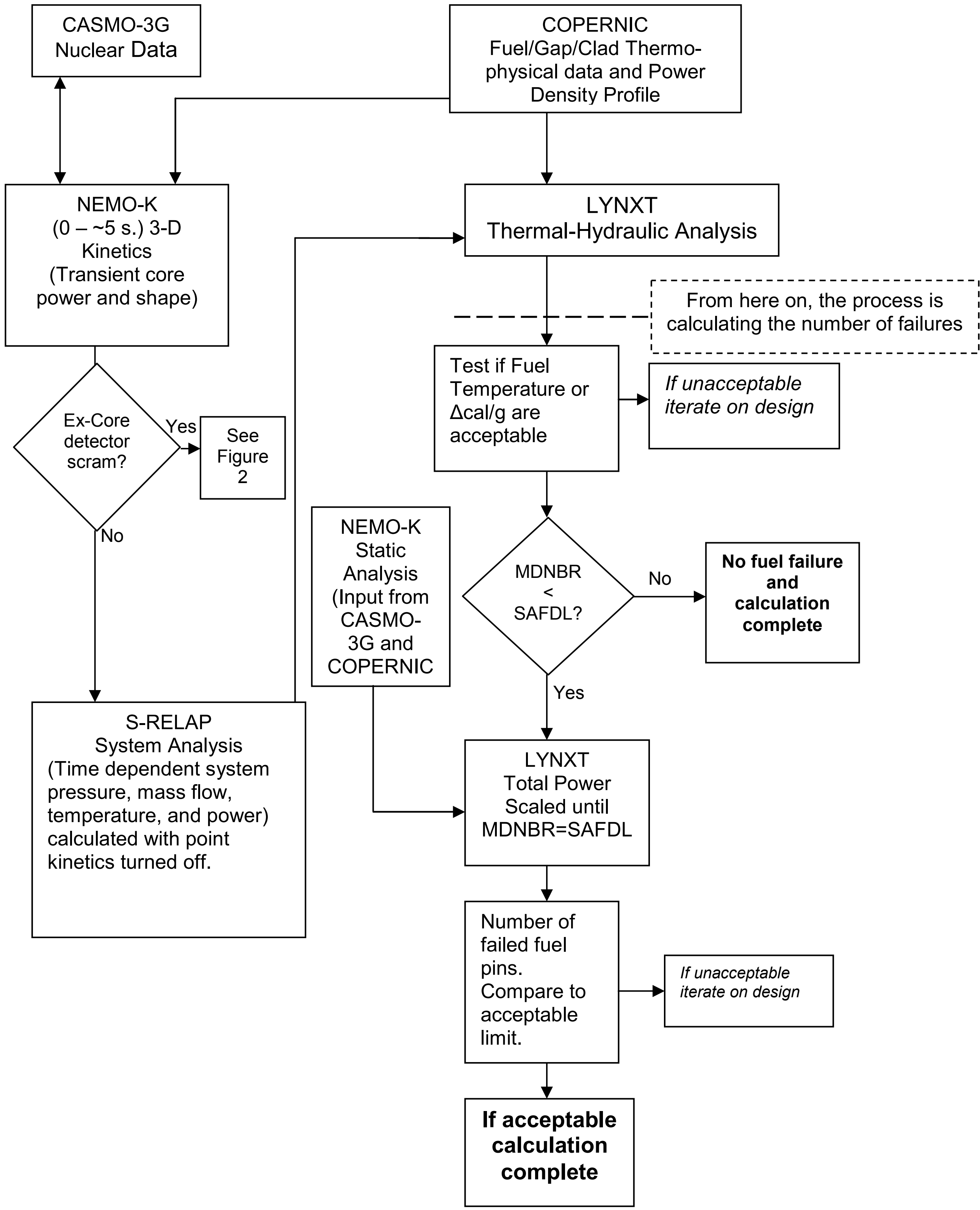


**Figure 3 – Details of Analysis Methodology (greater than approximately 5 sec.)**

### 3.3. CODE DESCRIPTIONS

#### 3.3.1. CASMO-3G

CASMO-3G [3] is used to generate the temperature-dependent few-group nuclear data libraries for use in NEMO-K. CASMO-3G takes input from a processing code, which makes the Evaluated Nuclear Data File (ENDF/B) compilations compatible with the input to CASMO-3G. CASMO-3G models a unit assembly that consists of a two dimensional slice through the assembly, in which individual rods are recognized in all their detail (each rod has a pellet, gap, and clad). This calculation uses reflective boundary conditions, and leakage effects are included by assuming that the leakage can be approximated by a fundamental mode. In this case, spatial flux varies as approximately $e^{iB.r}$ using the $B_1$ approximation, which implies $P_1$ scattering. In this manner, the global leakage effect can be accounted for in determining the neutron energy spectrum over which the cross-sections will be averaged. These calculations are repeated for each assembly type (loading and burnup), and for various temperatures, resulting in a multi-dimensional table of few group cross-section as a function of these variables.

The CASMO-3G table is accessed by NEMO-K as the transient progresses. Due to variation of the fuel temperature, the resonance range cross-section changes as a result of the Doppler effect, and these cross-section changes play a pivotal role in the progression of the transient. In general, it is, thus, important to generate cross-section input for NEMO-K on a sufficiently fine mesh in order to capture all temperature peaks.

#### 3.3.2. COPERNIC

COPERNIC [3] 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. The properties for the fuel 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 are determined and tabulated. The thermal properties are a complex function of burnup that affects the composition of gap gas, gap size, surface temperatures of the pellet and the clad, and contact pressure once the gap closes due to creep. It is desirable to use a constant gap size model in NEMO-K and LYNXT and, thus, an appropriate multi-dimensional table must be created that preserves the functional dependence described above, while not varying the gap size. **[**

**]**. These three parameters capture all the dependencies implied above.

In order to create the desired table, the applicant runs **[**

**]**. These calculations are repeated for various burnup levels, which finally result in a complete table of gap thermal properties that capture all the complex interactions in the desired format.

#### 3.3.3. NEMO-K

NEMO-K [3] solves the three dimensional space-time kinetic multi-group neutron diffusion equations, and it is possible to run NEMO-K in both the static and time-dependent modes. Generally, core transients require that the calculations recognize simultaneous variations in neutron flux, fuel pin temperature, and core coolant conditions.

The neutron flux algorithm evaluates the requirements for specific cross-sections, based on fuel temperature, coolant temperature and density, etc., and determines the three dimensional neutron flux shape and associated power. The cross-section input is obtained from specially prepared tables created by CASMO-3G calculations. Output from this step includes the power generated in the fuel and the power directly deposited in the coolant.

The fuel pin model uses thermo-physical properties determined by COPERNIC to calculate the fuel pin temperature. Inputs to this model are the power shape within the pellet, the clad wall temperature, and the fuel model time step. Outputs from the model are the Doppler effective temperature, which is used in the neutron model to select the appropriate cross-section, and heat flux at the clad surface.

The coolant model conserves mass and energy to determine the coolant properties, including temperature and density. Input to the model are the heat flux at the clad surface, power directly deposited in the coolant, inlet coolant temperature and volumetric flow rate, the system pressure, and the thermal-hydraulic time step.

As approved in ANP-10263P-A [3], NEMO-K is suitable for both rapid REA transients and slow rod drop transients. In addition, with appropriate bounding parameters and assumptions, it can be used for safety related transient simulations.

#### 3.3.3.1. Modifications to NEMO-K

In this section, the two most important modifications to the original NEMO-K code are outlined. They include a more realistic trip detection-control rod insertion, and fuel effective temperature determination models. In addition to these models, there are other changes considered minor by the staff. These concern the editing of fuel enthalpy changes and application of adjustment factors to selected parameters. These parameters include: Fuel and gap conductivity; cross-section changes due to temperature; and cross-section changes due to control rod position. These adjustment factors are included by the applicant to account for additional conservatism in the calculation.

##### 3.3.3.1.1. Trip detection and control rod insertion model

The U.S. EPR uses an ex-core rate lagged power trip signal to sense if an REA has occurred, and to subsequently scram the core. This trip function has three components, including an ex-core detector signal, a rate lagging processor for the signal, and a control rod insertion model. NEMO-K has models that simulate the functions implied by these actions and/or processes. The ex-core detectors are located in each quadrant surrounding the pressure vessel, which causes the ex-core response to differ from the core average when an off center rod is ejected. In addition, they measure the rate of power change, and if this is outside of threshold values, a trip signal is sent. These signals are processed with a suitable rate lagging function that is compared to trip values. Once the trip signal is exceeded, a time delay is employed before the control rods are moved to scram the core. The activation of the signal requires two detectors out of four total detectors to respond. This is known as 2/4 logic.

The ex-core detectors measure the fast flux exiting the core, and are calibrated to the actual conditions within the core. The in-core assembly powers are multiplied by weighting factors to correlate in-core conditions to the ex-core signals. The weighting factors are determined by transport calculations. In addition, the overall ex-core response is calibrated against measured thermal power.

Ex-core instantaneous read out, $E_n$ for radial detector can be summarized by the equation listed on page 6-2 of ANP-10286P [1]. This equation applies to detectors located at the top or bottom ex-core detector locations. The measured signals are provided by fast neutron leakage from the core, and the weighting factors are generated by using transport theory calculations. As a validation of this technique, comparisons between calculated and measured ex-core detector response results [3] for a rod drop transient show agreement to within one percent. The calibration factors are generally determined from plant specific data, and are periodically updated as per technical specification requirements.

ANP-10286P [1] did not contain sufficient information regarding the lead and lagged signal time constants, and, thus, in RAI-10 and RAI-29 [7], the staff requested that the applicant provide additional information to justify the constants used. In an October 3, 2008, response to RAI-10 and RAI-29 [7], the applicant provided the values used for these time constants. Additionally, in response to RAI-29, the applicant outlined a sensitivity analysis for power changes occurring in less than 1 second. The staff concludes that the response adequately addressed the staff's concern. Portions of these responses are included in the following discussion. The relationship between the rate lagged signal, used to scram the reactor, and the unfiltered ex-core detector response is simulated by a coupled set of first order differential equations that act as a filter. These equations are solved numerically, with the time step synchronous with the sampling rate. The time to start to move the scram rods is set based on an input delay time between the measured trip and the start of rod movement.

Rate Lagged Signal $F_l$ (t) is given by the equation on page 6-4 of ANP-10286P [1]. The gain and the lagged signal time constant are both set at **[ ]**.

The control rod drop model, used to determine the reactivity insertion during the scram phase of the transient, has five different steps. **[**

**]**. Because rods could be at different positions at the beginning of the transient, each rod bank is treated separately.

3.3.3.1.2. Effective fuel temperature model

ANP-10286P [1] did not contain sufficient information regarding the effective fuel temperature and power profile change during the transient, and, thus, in RAI-12 and RAI-13 [7], the staff requested that the applicant provide clarification regarding effective fuel temperature and power profile change during the transient. In an October 3, 2008, response to RAI-12 and RAI-13 [7], the applicant provided justification for the revised method of calculating the effective fuel temperature during a transient event. In addition, the new method was validated against known independent methods. Portions of these responses are included in the following discussions and evaluations. The staff concludes that the responses adequately address the staff's concern.

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. The staff notes that the determination of a representative fuel temperature as the transient progresses is, therefore, of great importance. In ANP-10286P [1], the cross-sections are generated assuming a flat temperature distribution in CASMO-3G, 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 needs to be determined. A simple solution to this problem used by the applicant was to use a volume averaged temperature. The original model (the Rowland model) used in NEMO-K for determining a time dependent fuel temperature was a model that combined the fuel surface and centerline temperatures given below and found on page 6-6 of ANP-10286P [1]. The following equation is known as the Rowland model:

$$T_{eff} = T_s \cdot wt_{sc} + T_{cl}\,(1 - wt_{sc})$$

Where:

- $T_{eff}$ = Effective flat profile fuel temperature
- $T_s$ = Fuel surface temperature
- $T_{cl}$ = Fuel centerline temperature
- $wt_{sc}$ = Weight factor for Rowland model

NEMO-K was modified to include an improved weighted effective flat fuel temperature that **[**

**]**. This relationship is given below:

**[ ]**

Where:

**[ ]**
**[ ]**

The weight factor used in the Rowland formulation of $T_{eff}$ ($wt_{sc}$) is obtained by assuming that the pellet temperature varies as function of radius squared, which is characteristic of fresh fuel. Based on this assumption the value of $wt_{sc}$ is 0.7, which is used in all subsequent calculations regardless of burnup. The improved formulation of $T_{eff}$ includes an average temperature term and a weighted temperature difference across the pellet. **[**

**]** Typically, the value of **[ ]**, which is used in all subsequent calculations regardless of burnup.

In RAI-12 [7], the staff requested that the applicant provide additional validation of the improved methodology to determine the effective temperature. In an October 3, 2008, response to RAI-12 [7], the applicant validated the improved determination of $T_{eff}$ by comparing calculated values, using the methods outlined above, to independent values determined using the APOLLO2 code [6]. The staff notes that the APOLLO2 code has not been validated or approved for U.S. EPR use. However, based on the staff's approval of APOLLO2 for use in the operating fleet and a review of the applicable limitations and parameters, the staff concluded that for the purposes of providing additional assurance regarding the validity of the Rowland model, the use of APOLLO2 is reasonable. The staff did not review or approve APOLLO2 for any other use for the U.S. EPR design.

The validation was carried out by comparing $^{238}U$ capture rate and reactivity for variable fuel temperatures within the pin to a constant temperature distribution within a pin. The value of the constant temperature that yielded the same $^{238}U$ capture rate and reactivity as the variable temperature case was considered $T_{eff}$ as determined by the APOLLO2 code. This value of $T_{eff}$ was then 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 were used in the validation exercise. It was found that for steady state fresh fuel conditions all three values of $T_{eff}$ were in good agreement. However, under all other conditions investigated it was found that the new $T_{eff}$ formulation agreed with APOLLO2 determinations of $T_{eff}$ to within ± 2K, while the Rowland formulation deviated from the APOLLO2 value by 23K – 65K 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 transients 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 of reactivity could be inserted upon ejection of the rod.

### 3.3.4. LYNXT

LYNXT is an open channel thermal-hydraulic code that includes a fuel thermal model. The open channel aspects of the code allow both axial and cross-channel flow, thus recognizing flow around the fuel pins. This flow model allows for the determination of more accurate fuel pin surface temperatures, which are important in the current application to determine the MDNBR value for the fuel pin of interest and whether or not it meets the SAFDL limits.

The fuel thermal model allows for both axial and radial heat conduction. The following two fuel rod models are included as options in the code:

1) The Constant Gap/Constant Properties (CG/CP) model is the simplest option. As the name implies, the pellet/clad gap and the thermo-physical properties remain invariant, with the exception of the fuel conductivity that can be represented by a third order polynomial.

2) The Variable Gap/Temperature Dependent Properties (VG/TDP) model is the most realistic. In this case, all the thermo-physical properties and the pellet gap are permitted to vary with temperature. In the case of the pellet/gap variation, both the temperature and pressure difference between the coolant and the gap gas pressure are accounted for. The fuel rod model in this case is based on fuel performance codes such as COPERNIC.

In this analysis, the LYNXT code is used to determine:

1) The MDNBR value of the fuel rod being analyzed, and to determine if it meets the SAFDL limits

2) The fuel centerline temperature and margin to incipient fuel melting

3) The enthalpy added to the fuel during the transient, and to determine if it meets acceptable limits

#### 3.3.4.1. Modifications to LYNXT

Modifications to LYNXT were minimal in nature, and did not involve changes to the fuel rod modeling. This model is based on a solution to the two dimensional conduction equation, with radial and axial dependence. Briefly, in the approved version of the code the fuel rod model either uses the constant gap/constant properties (CG/CP), or variable gap/temperature dependent properties (VG/TDP) representation. The later model being the most elaborate since it allows for the gap between the pellet and clad to change during the transient.

ANP-10286P [1] did not contain sufficient information regarding the thermo-physical properties used in LYNXT, the use of the CG/TDP model, and its validation against the COPERNIC code. In RAI-14, RAI-15, RAI-16, RAI-17, RAI-24, RAI-27, and RAI-31 [7], the staff requested that the applicant provide additional information to improve understanding of LYNXT. In an October 3, 2008, response to RAI-14, RAI-15, RAI-16, RAI-24, RAI-27, and RAI-31 [7] the applicant provided justification for the use of the property tables for the fuel, clad, and gap. In addition, the CG/TDP model was justified and validated against comparisons with the COPERNIC code, which more accurately represents the test problem. The staff concludes that the responses provide adequate supporting documentation for the LYNXT modification. Portions of these responses are included in the discussion below.

There are two enhancements to the approved LYNXT code. First, the number of solution locations is increased. Second, the modified LYNXT code implements a combination of the CG/CP and VG/TDP models, resulting in a CG/TDP model. The increased number of solution locations allows for a more accurate representation of the radial power profile, including those that peak on the outside, which can occur at EOC. The CG/TDP model requires that the following parameters stay invariant during the transient:

1) Fuel pellet dimension

2) Pellet/clad gap

3) Gas inventory in gap

4) Radial power profile

However, it does allow the following properties to be entered in tabular form:

1) Thermal properties for fuel and clad as a function of temperature.

2) Specific heat for fuel and clad as a function of temperature.

3) Gap conductance **[** **]**. 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 staff reviewed these model changes based on the information provided in ANP-10286P [1] and responses to RAI-14, RAI-15, RAI-16, RAI-17, RAI-24, RAI-27, and RAI-31 [7]. The staff concludes that the model changes more accurately represent, and remain bounding of, the thermal phenomena occurring during a rod ejection accident (particularly at EOC) and are, therefore, acceptable for use.

### 3.3.5. S-RELAP

S-RELAP5 uses a volume and junction model to solve the conservation of mass, momentum, and energy equations, similar to all other RELAP based codes. S-RELAP5 is used for both loss-of-coolant-accident (LOCA) and non-LOCA transients that involve changes to the thermal-hydraulic state points of the reactor. S-RELAP5 takes thermo-physical input from the steam tables for the coolant, and the properties determined by COPERNIC for the fuel. In addition, many of the reactor trip setpoints specific to the U.S. EPR system are included in the code. Thus, any condition that may change pressure, inlet temperature and/or flow rate during an REA can be modeled using this code.

It is traditional in the system level codes such as RELAP to use a point kinetics model to determine the transient reactor power. In this application (REA), the applicant disabled the point kinetics capability and instead used a power vs. time characteristic for transients that need a system level calculation. In these cases, the reactor power versus time response is either obtained directly from NEMO-K or considered to be conservatively defined higher than the NEMO-K results with time following the completion of the initial transient pulse of the event. In the former case where NEMO-K powers are used directly, iterations would be required between NEMO-K and S-RELAP5 until the power/thermal conditions are converged between NEMO-K and S-RELAP5. The staff considers the approximation in the latter case to be conservative with respect to power, since increasing fuel and coolant temperatures and decreasing system pressure would generally lead to a decrease in power.

### 3.3.6. Conclusions

The code suite used by the applicant for analyzing the rod ejection event for the U.S. EPR design includes codes previously approved for the applicant's operating fleet that were subsequently approved for use in the U.S. EPR design in ANP-10263P-A [3]. Based on the approval of ANP-10263P-A [3], the staff's review of the application of these codes to the analysis of the U.S. EPR rod ejection analysis, and the similarities to the operating fleet rod ejection analysis code suite, the staff concludes that the code suite detailed in ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report" [1], is applicable for analyzing the U.S. EPR response to an REA and conforms to the guidance provided in SRP Section 15.4.8.

Based on the evaluations provided in Section 3.3.3.1 of this report, the staff concludes that the changes presented in NEMO-K as compared with the previously approved version in ANP-10263P-A [3] 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 REA analyses for the U.S. EPR. The staff's review only applies to REA analyses for U.S. EPR and any other use of the modified version of NEMO-K presented in ANP-10286P [1] would require further NRC review and approval.

The staff also concludes that the changes presented in LYNXT as compared with the previously approved version in ANP-10263P-A [3] more accurately represent heat conduction by creating a constant gap / temperature dependent properties model and increasing the number of solution locations. Based on the evaluations provided in Section 3.3.4.1 of this report, these modifications are approved for use in REA analyses for the U.S. EPR. The staff's review only applies to REA analyses for U.S. EPR and any other use of this modified version of LYNXT would require further NRC review and approval.

The applicant verified the CG/TDP model by comparison with independently determined parameters (fuel temperatures, etc.) using higher fidelity codes and benchmark problem results in ANP-10286P [1]. These parameters include:

1) Analytic and fuel performance benchmark problems

2) Comparisons to COPERNIC, which used a form of a VG/TDP model

The code comparisons indicate that the CG/TDP model based on input gap conductance tables agrees with the benchmark solutions to **[ ]**. Comparisons to the dynamic COPERNIC calculations, which used a more accurate form of a VG/TDP model, indicate that the CG/TDP model in LYNXT consistently over-predicts the fuel temperature. The over-prediction is highest during the peak of the transient pulse, and then reverts to an over-prediction of **[ ]**. The duration of the pulse is very small and, thus, the overall over-prediction of the energy deposited is expected to be **[ ]**. The applicant concluded from these comparisons that the LYNXT CG/TDP model combination conservatively predicts the fuel temperature and, thus, the gap conductance fitting tables used in this model are an acceptable method to predict the fuel melt and MDNBR conditions for REA event analysis. The staff reviewed the applicant's comparisons and concurs with the applicant's conclusion.

## 3.4. SELECTED TRANSIENT RESPONSE

In this section, applications of the code system outlined in Section 2 of this report to two different types of REA transients are described. Topical Report ANP-10286P [1] did not include enough detail for the staff to conclude if the spectrum of rod ejection events was accurately modeled. In RAI-26 [7], the staff requested that the applicant provide an event timeline and description of the various REA events. In an October 3, 2008, response to RAI-26 [7], the applicant responded with a detailed explanation of the various scenarios that were analyzed along with respective timelines. The staff concludes that the response adequately demonstrates the ability of the methodology to handle the spectrum of REA events. The staff also used select portions of this response for comparison in the independent confirmatory analyses [13]. The transients discussed follow two distinct trajectories in time; the first is initiated by a significant reactivity input and is terminated by an ex-core detector trip signal, while the second experiences a smaller reactivity input and is terminated by a system level scram. In the case of the U.S. EPR, these transients occur at EOC and BOC respectively, and are discussed in ANP-10286P [1], Chapter 8. Because of the buildup of transuranic elements at EOC, the value of the effective total delayed neutron fraction ($\beta_{eff}$) is lower than the value at BOC. This fact, together with the fact that the control rods are worth more at EOC, ensures that the added reactivity exceeds $1.0 for transients up to 40 percent of full power. At BOC, the added reactivity is always below $1.0, and consequently, the transient power is not expected to be as sudden and as peaked. At EOC, the moderator and Doppler feedback are both negative, while at BOC only, the Doppler feedback is negative and the moderator feedback is small but positive as listed in ANP-10286P [1], Table 7-1. Moderator feedback plays a secondary role in the initial part of the transient. Only if it continues beyond the initial power pulse, and the remainder of the system is involved, does the moderator effect become important. In the following two sections, the progression through these transient types is outlined. These sections should be read while consulting Figures 2 and 3 of this report.

The objective of an REA event analysis is to determine the number of failed fuel rods following the termination of the event. In order to assure that this number is as conservative as possible while still remaining realistic, the input parameters and boundary conditions to the analysis are chosen to yield conservative results. These conservative limits are summarized in ANP-10286P [1], Table 7-3. ANP-10286P [1] did not contain sufficient information regarding the sensitivity of the number of fuel rods to the assumptions made in the analysis. In RAI-34 and RAI-35 [8], the staff requested that the applicant provide additional information related to these topics. In a March 26, 2009, response to RAI-34 and RAI-35 [8], the applicant provided the primary conservatisms in the REA methodology that result in the maximum defensible number of failed rods. These are listed below:

1) The deterministic choice of ejecting the highest worth control rod

2) Adding a 15 percent uncertainty to the ejected control rod

3) Adding uncertainties to the Doppler and moderator temperature coefficients, and $\beta_{eff}$

4) The reference cycle has an additional bias on the ejected rod worth, Doppler and moderator temperature coefficients, and $\beta_{eff}$ that allow for possible variations introduced by further cycles

5) Peaking factor uncertainties of **[ ]** for the peak rod power ($F_{\Delta H}$) and **[ ]** for the peak local power ($F_Q$) are added

Based on the staff's review of these conservatisms, the staff concludes that the number of failed fuel rods determined by the applicant's methodology is a conservative upper limit.

3.4.1. Transient involving approximately $1.0 of reactivity

The analysis is started by preparing the appropriate nuclear data library using CASMO-3G, and the thermo-physical property input data and pellet power profile using COPERNIC. These two data files and appropriate NEMO-K input, which involves initial conditions, spatial grid configuration, and transient termination conditions completes the input preparation. At this stage, the NEMO-K calculation can be carried out, which results in a reactor trip following approximately 1.0 sec., since an ex-core overpower trip signal is sent to the scram system at this time. An illustration of these power pulses is presented in Figure 26-2 of the October 3, 2008, response to RAI-26 [7] for several EOC starting points. It is seen that they are very sharp, and with peak values almost double the reactor full power. However, the Doppler effect is very efficient at reversing the power rise and the ex-core detectors are responsible for ensuring a reactor scram after approximately 1.0 sec. It takes approximately 3 sec. to fully insert the control rods and shut the reactor down. Output from NEMO-K in the form of transient core power, axial power profiles, and rod power profiles, which includes allowances to account for other possible reactor designs, forms the input to LYNXT. The LYNXT code also uses input from COPERNIC. At this stage, the fluid dynamic and heat transfer analysis can be carried out for the transient. The primary output from this analysis is the level of compliance with the relevant regulatory guidelines concerning maximum fuel temperature, maximum enthalpy rise in fuel, DNBR limits, and the number of rods failed. The minimum DNBR values, as a ratio of MDNBR to SAFDL, vary as a function of time during the transient. If the ratio dips below unity, rods are potentially failed.

### 3.4.2. Transient involving less than $1.0 of reactivity

In RAI-26 [7], the staff requested that the applicant provide an event timeline and description for various REA events, including transients involving less than $1.0 of reactivity. In an October 3, 2008, response to RAI-26 [7], the applicant provided examples of transients that do not reach a trip signal in the traditional manner for an REA occur at BOC and EOC HFP. In these cases, a system analysis is required, since the duration of the transient is longer than the coolant transit time around the primary circuit. In ANP-10286P [1], S-RELAP is used to carry out the system calculations, which take input from NEMO-K. The S-RELAP5 calculation continues the analysis into the regime where NEMO-K does not apply, since it does not recognize the remainder of the system. In addition, in these transient calculations the primary system has been compromised by a hole caused by the control rod ejection. The primary system boundary break is assumed to be 2.95 in. (7.5 cm) in diameter, and conforms to the control rod flange size. This added boundary condition will affect the system pressure, since there will be a slow leak out of the break. The simulation continues until a trip in the S-RELAP5 model is reached. Results for this type of transient are shown in Table 26-3 and 26-4 of the response to RAI-26 [7]. It is seen in the October 3, 2008, response to RAI-26 [7] that the pressure decreases, essentially monotonically, while the inlet temperature increases almost linearly. The transient would eventually trip, either because of low system pressure or high secondary steam pressure. In these cases, the transient is slow enough that the core is essentially in equilibrium with the thermal parameters. Thus, a quasi-static approach is appropriate for this analysis and the complication of coupled space-time analysis is not needed. Thus, using various thermal boundary conditions obtained from S-RELAP5, a series of static calculations are run using NEMO-K in its static mode to determine power shapes and magnitudes following a rod removal. The bounding conditions, for example maximum power, are evaluated using LYNXT to determine the response of the fuel rods. The staff reviewed the applicant’s approach as presented in ANP-10286P [1] and supplemented by the October 3, 2008, response to RAI-26 [7] and agrees with the assumptions used. Additionally, the staff concludes that the methodology presented is acceptable for modeling transients involving less than $1.0 of reactivity.

The above section outlines the reactor response to a typical rod ejection transient. Two transients are considered: the first one involves a rapid scram of the reactor due to ex-core detectors, and the second one has a longer duration and is terminated by a system level trip. ANP-10286P [1] did not contain sufficient information regarding the application of the LYNXT code in these analyses, and thus the staff requested additional information. In an October 3, 2008, response to RAI-30, RAI-31, and RAI-32 [7], the applicant provided justification as to why the pin power distribution is conservative, explained differences between the COPERNIC and LYNXT results, and explained the analysis of the application of LYNXT in the BOC hot zero power transient. The primary difference between LYNXT and COPERNIC regarding fuel temperature is the coarseness of the grid used to determine the thermo-physical properties in LYNXT. This leads to abrupt changes in the calculated temperature, which are not seen in the COPERNIC calculations. The staff concludes that these discussions adequately address the issues raised by the staff.

### 3.4.3. Determination of the number of failed rods

ANP-10286P [1] does not contain sufficient information regarding the calculation of the number of failed fuel rods following a REA transient. In RAI-28 [7], the staff requested that the applicant provide additional information to explain this calculation methodology. In an October 3, 2008, response to RAI-28 [7], the applicant provided an explanation of the methodology used in determining the number of failed rods regardless of the length of the transient event.

As discussed below, the staff concludes that the response satisfactorily provides the information requested in the RAI. Portions of the response to this RAI are included in the following evaluation discussion.

In the case of each transient, two different rod failures need to be considered in order to conform to the guidance provided in SRP Section 4.2, Appendix B regarding cladding failure and radiological impacts: (1) Rods that fail during the initial transient event, which is controlled by the energy deposition (cal/g) added during the transient; and (2) rods that fail to meet the MDNBR value defined by the SAFDL. Therefore, the MDNBR/SAFDL ratio must be determined and those rods below unity are considered failed. Fuel rod failure during the initial transient did not occur in the transients considered in ANP-10286P [1], since the failure criterion, or addition of 110 cal/g (197.4 BTU/lb), was never met. Thus, all rod failures for this reactor are due to the second mechanism described above. To quantify this mechanism, the values of $F_{\Delta H}$ and $F_Q$ as determined by LYNXT with input from NEMO-K, corresponding to the conditions for a fuel rod when the MDNBR/SAFDL ratio is unity, are used as the failure criterion. Thus, any rod for which $F_{\Delta H}$ or $F_Q$ exceed these values is considered failed. The most accurate manner of determining the number of failed pins would be to obtain a rod-by-rod power distribution vs. time from a three dimensional neutronics/thermal-hydraulic calculation to determine the MDNBR. However, this option is not realistic and the staff agrees that a practical approximate method that captures the salient phenomena is acceptable.

The applicant has determined that the **[**

**]**.

The information presented by the applicant is contained in Figures 28-1 to 28-3 in the October 3, 2008, response to RAI-28 [7]. **[**

**]**. This fact is used to simplify the determination of fuel rod or assembly power at which the MDNBR calculation is carried out.

In order to determine if the MDNBR condition is reached for an arbitrary fuel rod, the following procedure is used: (1) Selected assemblies that have been analyzed using NEMO-K in full transient mode are analyzed using LYNXT to determine the power vs. time response at which DNB occurs; (2) this power vs. time response is determined by scaling the power profile until the DNB conditions are satisfied; (3) the scaling factor that corresponds to this power level is known as the "multiplier," and **[**

**]** the transient values can also be determined.

Representative samples of different fuel assembly responses are analyzed for which the full space-time kinetic and thermal hydraulic method was used. The fuel assemblies for which a full 3-D analysis has been carried out will be referred to as FA3-D. These assemblies are chosen in such a manner as to characterize the bulk of the core response to the transient. Two pieces of information are obtained from this analysis for each of the fuel assemblies: **[**

1)

2)

3)

4)

].

For those cases that are terminated by a system trip, and determined by analysis using S-RELAP5, the transient vs. static correlation is not necessary since the time dependence is relatively weak, and a quasi-static approach is sufficient. The coolant pressure and temperature change with time, but the power is essentially constant. In order to find the values of $F_{\Delta H}$ and $F_Q$ at which the DNBR limits are exceeded, LYNXT calculations are repeated using input from S-RELAP5 and NEMO-K. Generally, the peak assembly power is scaled until it reaches the MDNBR design limits. The scaled values of $F_{\Delta H}$ and $F_Q$ for this fuel rod then become the failure criterion for each fuel rod in the core. Any fuel rod that exceeds the $F_{\Delta H}$ and $F_Q$ failure criteria is assumed to be failed.

The implication of conservatisms and contingencies that are assumed in the estimate of rod failures, which covers uncertainties and future fuel cycle designs, was made by removing them and re-evaluating the rod failure census. The primary sources of conservatism are:

1) The highest worth rod is chosen to be ejected. The worth is further increased by 15 percent to account for uncertainties.

2) Uncertainties are applied to the Doppler temperature coefficient, moderator temperature coefficient, beta-effective, and the peaking factors.

3) The reference cycle configured for the analysis has additional biases concerning the rod worth, Doppler temperature coefficient, moderator temperature coefficient, beta-effective, and the peaking factors.

4) The most extreme assemblies are chosen for the **[**
**]**.

An analysis by the applicant, reported in ANP-10286P [1], used this methodology with the full complement of uncertainties for the BOC HFP case and predicted rod failures in both the dynamic and the quasi-static phases of the transient. ANP-10286P [1] did not contain sufficient information regarding the sensitivity of the number of failed rods to uncertainties that were included in the analysis. In RAI-34 and RAI-35 [8], the staff requested that the applicant provide additional information related to this sensitivity. In a March 26, 2009, response to RAI-34 and RAI-35 [8], the applicant removed the uncertainties included in the ANP-10286P [1] analysis and re-estimated the number of failed rods. This analysis was carried out for both the rapid transients, characterized by an ex-core detector scram, and the longer transients that rely on a system level scram. If the transient is re-analyzed with the uncertainties removed the results shown in Table 1 of this report are obtained. The results shown in this table are based on ANP-10286P [1], Table 8-3, and the March 26, 2009, responses to RAI-34 and RAI-35 [8].

**Table 1 – Percent of rods failed with and without uncertainties**

| | Dynamic phase | Quasi-static phase |
|---|---|---|
| With uncertainties | 0.3 | 7.2 |
| Without uncertainties | - | 0.6 |

It is seen that there is a significant reduction in the percentage of rods failed by removing the uncertainties. The staff concludes from this result that the analysis presented in ANP-10286P [1] would cover a wide range of fuel cycle designs.

#### 3.4.4. Conclusions

The staff reviewed the application of the code system outlined in Section 2 of this report for large and small REA transients, (i.e., transients that are terminated by either ex-core detectors or system trips, respectively). Based on the applicant's analysis of the events, RAI responses, and comparison to similar confirmatory analyses performed by the staff as detailed in Section 3.4 of this report, the staff concludes that the applicant's U.S. EPR REA methodology as presented in ANP-10286P [1] conforms to the appropriate guidelines in SRP Section 4.2, Appendix B, and the criteria listed in GDC 13 and GDC 28. The staff concludes that the methodology presented in ANP-10286P [1] can be used to analyze the REA event for the U.S. EPR design.

## 3.5. CONFIRMATORY ANALYSES

### 3.5.1. Introduction

Confirmatory calculations of reactivity insertion accidents and related phenomena were carried out as a guide in the understanding of the various physical phenomena involved in the transient, and as an independent check of the applicant's analysis techniques. Specifically, two confirmatory analyses were performed: (1) An independent confirmatory analysis by the staff of the most limiting case provided in Reference 1 using the TRITON/TRACE code package, and (2) a new analysis performed by the applicant of select SPERT [12] experimental rod ejection data. This SPERT data came from a reactor test conducted in 1965 [12], and provides a benchmark with which to validate analytical methods.

### 3.5.2. Independent Confirmatory Analyses

The TRITON/TRACE code package was used to assist in the review of the applicant's code package [10 and 13]. Specifically, confirmatory analyses were used to investigate the analysis results for the most limiting case and the NEMO-K model changes as presented in ANP-10286P [1]. The independent confirmatory analyses showed nearly identical energy deposition calculations for the most limiting case.

The staff modified the internal confirmatory runs to include the original and modified temperature models from NEMO-K to investigate the fuel temperature model changes introduced into NEMO-K. In order to perform the confirmatory calculations, the staff requested detailed design information. In a July 10, 2008, response to RAI-1 and RAI-2 [9], the applicant provided this information. The staff developed models for each fuel type and generated cross-sections in which all the relevant physical variables were parameterized. The most limiting event proposed in ANP-10286P [1] was used in this confirmatory calculation to assist in the staff's review of the applicant's calculations. The results of the rod ejection simulations are shown in Figures 4, 5 and 6 of this report. The staff's TRACE model predicts a rod worth of $1.33 which, compared to the applicant's prediction of $1.35, further indicates that the models are consistent with one another. The maximum power predicted by TRACE of 240 percent compares well to the applicant's prediction of approximately 210 percent. Figure 6 of this report summarizes the fuel temperature calculation results using three different effective fuel temperature models referred to as the Rowland's, volume, and AREVA models. These models represent, respectively: (1) A linear combination of the fuel centerline and surface temperatures; (2) a simple volumetric average of the radial temperature profile in the fuel; and (3) a combination of both the linear and volumetric approach. As shown in the results, TRACE predicts very little impact of these different models on the predicted fission power, but a noticeable, yet relatively insignificant, effect on the fuel temperature can be seen. TRACE predicts a maximum fuel temperature of 850 Kelvin which is consistent with the applicant's prediction of 785 Kelvin. Most importantly, however, the TRACE-predicted enthalpy deposition is approximately 30 cal/g which is well below the acceptance criterion of 110 cal/g as specified by SRP Section 4.2, Appendix B. For completeness, the components of reactivity are shown in Figure 7 of this report.

The changes did not lead to any unexpected anomalies and the results compared well with the results reported in ANP-10286P [1].

No additional RAIs were generated as a result of the confirmatory analyses.

### 3.5.3. SPERT Analysis

In this section, results of an independent analysis of the applicant's REA transient by the NRC, and validating analyses of SPERT experimental data by both NRC and the applicant are presented. In order to provide additional validation of the applicant's rod ejection accident methodology, the applicant was asked to model test cases from the SPERT reactivity accident experiments [11]. The SPERT-III E-Core reactor is a small PWR designed to investigate power excursions. The maximum number of assemblies that can be loaded in the core is 68, which results in a high-leakage core design when compared with the U.S. EPR core with 241 assemblies. The applicant's REA code package was designed to analyze large PWR cores instead of small reactor cores, and it is expected by the staff for some differences to occur due to modeling limitations. Regardless, expected differences due to modeling limitations would be relatively small and the overall predicted reactor response would reasonably match experimental results.

As detailed in a March 26, 2009, response to RAI-36 [8], the applicant validated the COPERNIC-NEMO-LYNXT REA analysis methodology by analyzing two rod ejection transients carried out at the SPERT facility and documented in IDO-17036, "SPERT III Reactor Facility, E-Core Revision" [13]. Test 60 (HZP) and Test 86 (HFP) from IDO-17036 [12] were chosen as the two candidate experiments for comparison. The Test 60 power vs. time experimental results were compared to analytical results obtained using the methodology and code package outlined in ANP-10286P [1]. The calculated peak power is 439 MW and the measured peak power is reported as 410 ±41 MW. The calculated integrated power at the peak is 8.6 MW-sec.

and the measured integrated power is reported as 8.5 ±1.1 MW-sec. The staff notes that these calculated values are within the measurement uncertainty of the test. This resolves the question raised by RAI-36 [8], and acts as a further validation of the methodology outlined in ANP-10286P [1].

The Test 86 power vs. time experimental results were compared to analytical results obtained using the methodology and code package outlined in ANP-10286P [1]. The calculated peak power response is in agreement with the measured response, yielding a calculated peak power of 604 MW compared to a measured value of 610 ±60 MW. The calculated time integrated power, or the time integrated power above the initial power, at the peak is 13.7 MW-sec. and the measured is reported as 17 ±2 MW-sec. The calculated result for the integrated power for this test is outside of the quoted measured uncertainty. A possible cause of the difference between the measured and calculated integrated power for Test 86 could be a higher uncertainty in the measured results attributed to the at power conditions than reported, or a lack of refinement in the model to represent the small SPERT III-E core configuration which is not typical of a PWR.

In the first comparison between NRC and the applicant's calculations of an REA the agreement is very good, particularly considering that completely different codes were used in the two analyses. In the second comparison both NRC and the applicant's analyses agreed well with the SPERT experimental data, essentially validating both methods for carrying out rod ejection accidents.

### 3.5.4. Conclusions

Three NRC confirmatory analyses of the U.S. EPR, using TRACE/PARCS, were performed to assist in the review of ANP-10286P [1]. The TRACE/PARCS models were consistent with the applicant's models while maintaining as much analytical independence as possible. The only calculation by the applicant used to develop the staff's models was the equilibrium BOC exposure. The staff's REA simulations served to independently support the staff's review of the applicant's predictions and guide the development of RAIs. The resulting agreement between the staff's independent confirmatory calculations and the applicant's U.S. EPR REA analysis adds further support to the staff's review that shows that the U.S. EPR REA methodology acceptably models the REA event and that the U.S. EPR reactor response should meet regulatory requirements as outlined in Section 2 of this report.

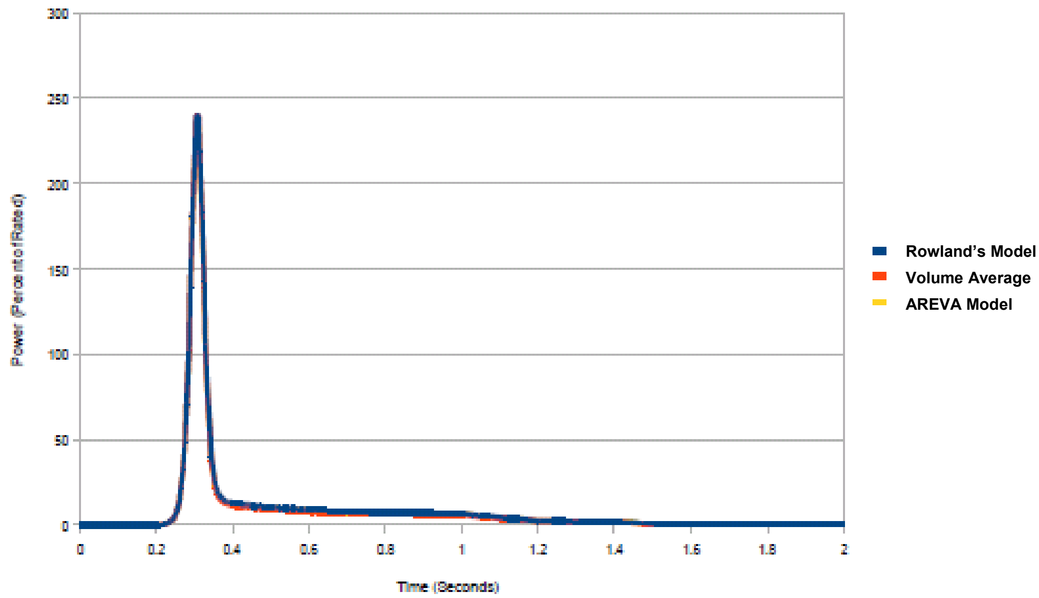


**Figure 4 – Fission Power Following Rod Ejection**

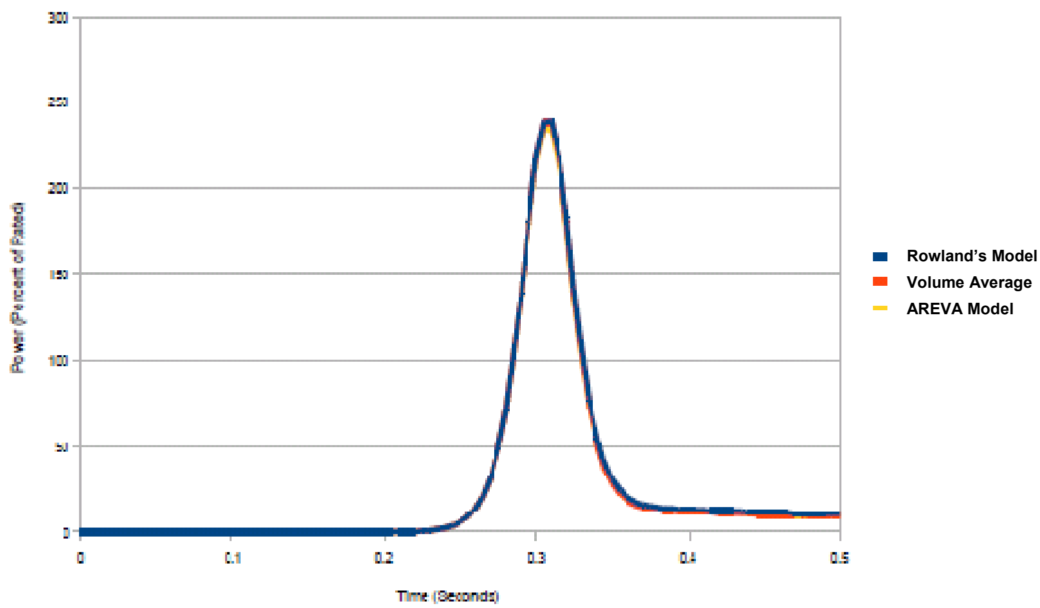


**Figure 5 – Fission Power Following Rod Ejection (Zoom)**

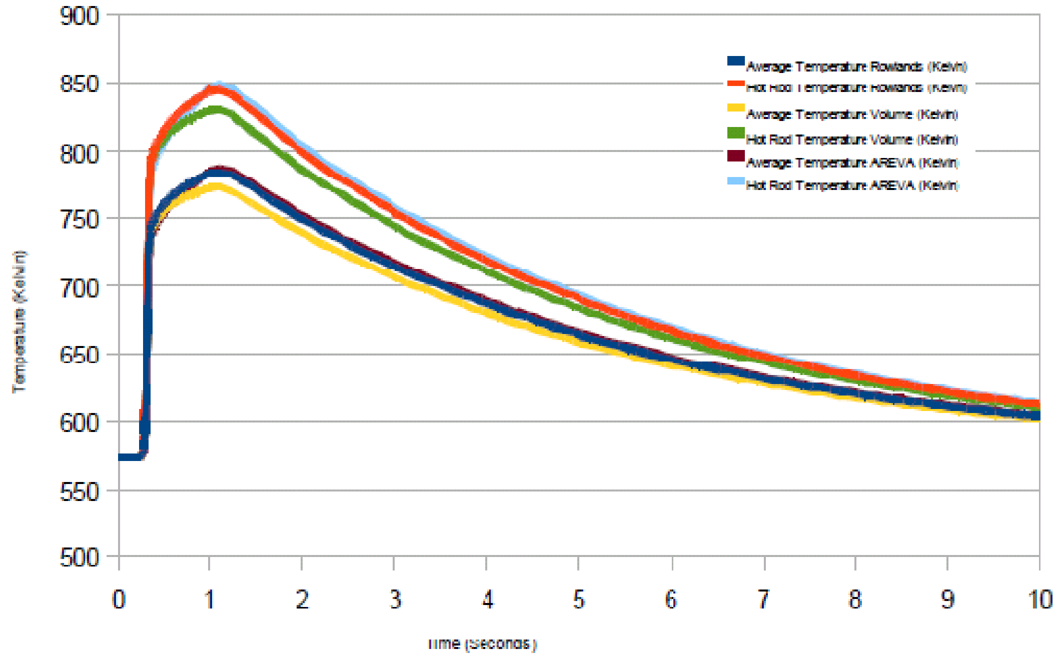


**Figure 6 – Fuel Temperature Calculation for U.S. EPR Rod Ejection Comparing Different Effective Fuel Temperature Models**

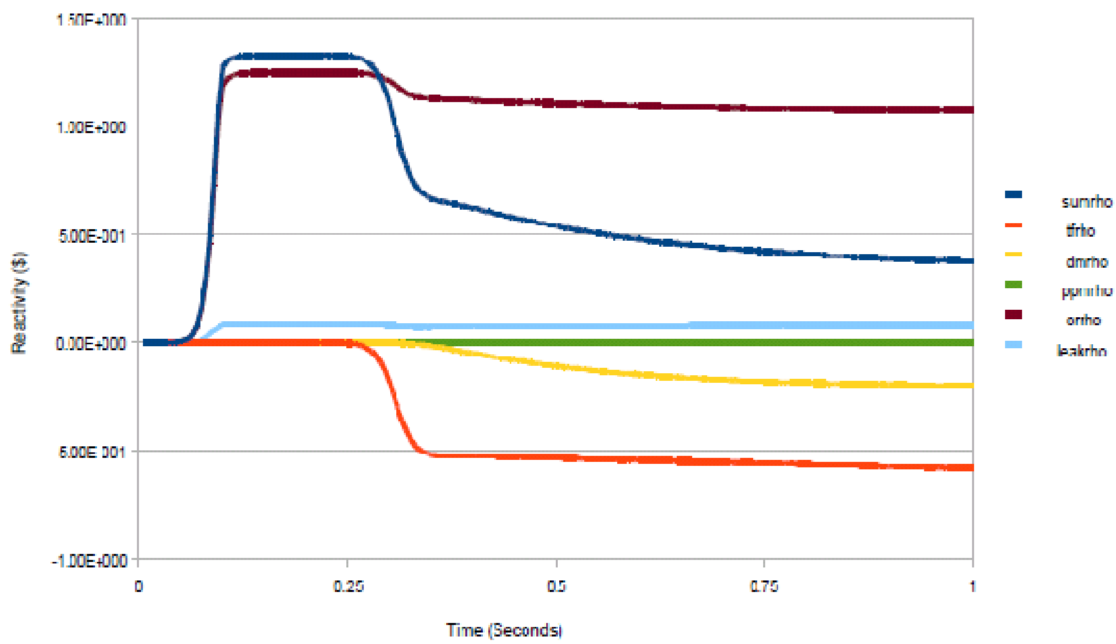


**Figure 7 – Reactivity Balance Following Rod Ejection (Rowland's Effective Fuel Temperature Model)**

## 4.0 CONCLUSIONS

The staff has reviewed the applicant's rod ejection analysis methodology documented in ANP-10286P [1]. From this review, the staff concludes from this review that this methodology is acceptable for performing rod ejection analyses for the U.S. EPR nuclear power plant design, and that results generated by the methodology presented in ANP-10286P [1] can accurately determine whether or not the regulatory requirements outlined in Section 2 of this report are satisfied. Using this methodology, the applicant has analyzed the U.S. EPR rod ejection accident based on inputs listed in ANP-10286P [1] and RAI responses [7, 8, and 9]. The staff concludes that the use of the methodology described in ANP-10286P [1], for the U.S. EPR design and within the boundaries and limitations set forth within ANP-10286P [1] and Section 5 of this report, will result in the satisfaction of GDC 13 and GDC 18, due to the resulting conformance with the guidance provided in SRP Sections 15.4.8 and 4.2, Appendix B.

Additionally, based on the present review, the staff reaches the following conclusions regarding the updates concerning the NEMO-K and LYNXT codes:

1. The revised effective temperature model and the trip detection and control rod activation model used in NEMO-K are approved for use in rod ejection accident (REA) analyses.
2. The modified constant gap/temperature dependent properties (CG/TDP) model used in LYNXT is approved for use in REA analyses.
3. 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 REA analyses.

## 5.0 RESTRICTIONS AND LIMITIATIONS

Based on the review of TR ANP-10286P [1], the staff imposes the following restrictions and limitations:

### 5.1. LIMITATION NO. 1 – APPROVAL OF NEW CODE MODELS

The models of NEMO-K and LYNXT as submitted in ANP-10286P [1] are different than the code models previously approved in ANP-10263P-A [3]. The staff's review of the code modifications found in ANP-10286P [1] only pertains to their use in reactivity initiated accident analyses as described by ANP-10286P [1]. The approval of this topical report does not in any way imply an approval of these new models of NEMO-K and LYNXT for any other purpose.

### 5.2. LIMITATION NO. 2 - DESIGN APPLICABILITY

Per the methodology and analysis detailed in ANP-10286P [1], the staff's review and approval is limited to the U.S. EPR reactor design.

### 5.3. LIMITATION NO. 3 - FUEL DESIGN

The REA analysis presented in ANP-10286P [1] is based on the fuel and nuclear design of the U.S. EPR as presented in Revision 0 of the U.S. EPR Final Safety Analysis Report. This analysis is applicable to core designs bounded by the parameters listed in Table 9-1 of ANP-10286P [1]. Any core designs that are not bounded by ANP-10286P [1] will need to be addressed separately following the methodology presented in ANP-10286P [1].

## 6.0 REFERENCES


1. ANP-10286P, “U.S. EPR Rod Ejection Accident Methodology Topical Report,” November 2007 (ML073310622).

2. NUREG-0800, Revision 3, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Section 15.4.8, Spectrum of Rod Ejection Accidents (PWR),” March 2007 (ML070550014).

3. ANP-10263P-A, “Codes and Methods Applicability Report for the U.S. EPR,” August 2006, (ML062270395).

4. “SCALE 5.1,” ORNL/TM-2005/39, Oak Ridge National Laboratory, November 2006.

5. “TRACE 5.0 Theory Manual: Field Equations, Solution Methods and Physical Models,” U.S. NRC, (ML071000097).

6. BAW-10228PA, “Science,” Framatome Cogema Fuels, December 2000 (ML010110422).

7. “Response to Second Request for Additional Information - ANP-10286P ‘U.S. EPR Rod Ejection Accident Methodology Topical Report’," October 3, 2008 (ML082880502).

8. “Response to a Third Request for Additional Information Regarding ANP-10286P, ‘U.S. EPR Rod Ejection Accident Methodology Topical Report’,” March 26, 2009 (ML090890175).

9. “Response to Request for Additional Information Regarding ANP-10286P, ‘U.S. EPR Rod Ejection Accident Methodology Topical Report’,” July 10, 2008 (ML081970402).

10. Ulses, A. P, “TRACE Calculations of U.S. EPR using a 3-D Kinetics Model,” January 29, 2009 (ML090300045).

11. McCardell, R. K., et al., “Reactivity Accident Test Results and Analyses for the SPERT III E-Core – A Small, Oxide-Fueled, Pressurized-Water Reactor,” March 1969 (ML080320431).

12. Dugone J., “SPERT III Reactor Facility, E-Core Revision,” IDO-17036, U.S. Atomic Energy Commission, November 1965.

13. Ulses, A.P. “Final Results of US EPR RIA and Setpoint Confirmatory Calculations,” May 27, 2009 (ML091420248)

## 7.0 ACRONYMS

| | |
|---|---|
| BOC | beginning of cycle |
| CFR | *Code of Federal Regulations* |
| CG | constant gap |
| CP | constant properties |
| DNBR | departure from nucleate boiling ratio |
| ENDF | Evaluated Nuclear Data Base File |
| EOC | end of cycle |
| FSAR | Final Safety Analysis Report |
| GDC | General Design Criterion |
| HFP | hot full power |
| HZP | hot zero power |
| LOCA | loss-of-coolant-accident |
| MDNBR | minimum departure from nucleate boiling ratio |
| NRC | Nuclear Regulatory Commission |
| PCMI | pellet clad mechanical interaction |
| PWR | pressurized water reactor |
| RAI | request for additional information |
| REA | rod ejection accident |
| SER | Safety Evaluation Report |
| SAFDL | specified acceptable fuel design limit |
| sec. | seconds |
| SPERT | special power excursion reactor test |
| SRP | Standard Review Plan |
| TDP | temperature dependent properties |
| TFGR | transient fission gas release |
| TR | Topical Report |
| VG | variable gap |

**Attachment**
**The Staffs Disposition of AREVA's comments on the Draft SE**

**COMMENT 1**
AREVA commented in Section 1.0 that 31.2 cal/gm is the value above which increase fission gas release must be considered, not the failure point.

DISPOSITION
The value was corrected to 110 cal/g.

**COMMENT 2**
AREVA noted in item 3 of Section 1.1 that LYNXT was not used for the SPERT comparison.

DISPOSITION
Staff corrected text to delete reference to LYNXT.

**COMMENT 3**
AREVA noted in bullet 2 of Section 2.2 that the requirements are that the pressure boundary integrity, the reactor internals integrity, and the fuel assembly integrity need to be evaluated if the fuel rod bursts, not the reverse. The evaluation of this is eliminated in the method by not allowing the fuel rod to burst.

DISPOSITION
NRC staff note that this section only deals with the description of the relevent NRC guidance regarding coolability and does not make any judgement on the whether or not the topical report meets the guidance or requirements. No changes were made to the text.

**COMMENT 4**
AREVA noted in bullet 3 of Section 2.2 that the step was performed regardless of the failure mechanism.

DISPOSITION
This section was re-written to focus on describing the relevent NRC guidance.

**COMMENT 5**
AREVA noted on page 5 that the methodology accounts for increase fission gas release whether the failure is due to DNB or PCMI as long as the failure occurs above 31.2 cal/gm.

DISPOSITION
NRC staff updated the text in the last paragraph of page 5 for clarity and accuracy.

**COMMENT 6**
AREVA noted on page 6 that there is no Section 1.4 in the topical report, and inquired which Section was referred to in the text of the SER.

DISPOSITION
NRC staff updated the reference to point to Section 3.2 of the SER.

**COMMENT 7**
AREVA noted on bullet 2 of Section 3.2 that the input to S-RELAP5 is either form NEMO-K of a conservative value relative to the NEMO-K value, and that the sample problem characteristics are an example instead of a method.

DISPOSITION
NRC staff deleted text referring to the input power variation used for S-RELAP5.

**COMMENT 8**
AREVA noted on Figure 2 that the box with “Ex-core detecto” is missing some text.

DISPOSITION
NRC staff corrected the figure.

**COMMENT 9**
AREVA noted on Figure 3 that S-REALP should be S-RELAP5 in boxes and that the words “No scram” in the NEMO-K box were incorrect.

DISPOSITION
NRC staff corrected the figure.

**COMMENT 10**
AREVA noted in Section 3.3.2 that the COPERNIC calculation was not described correctly.

DISPOSITION
NRC staff removed the incorrect statement.

**COMMENT 11**
AREVA noted in Section 3.3.5 that the COPERNIC the SER language suggested that only a conservative flat response with time is allowed to be used, and that this statement could be interpreted that NEMO-K powers cannot be used in SRELAP5.

DISPOSITION
NRC staff updated the text in the final paragraph of Section 3.3.5 to address the above concern.

**COMMENT 12**
AREVA noted in Section 3.3.6 that the two statements regarding the staff’s review were very broad and outside the context of this topical report.

DISPOSITION
NRC staff note that this statement is also captured in the conclusions contained in Section 4.0. No changes were made to the text.

**COMMENT 13**
AREVA noted in Section 3.3.6 that COPERNIC does not have “the VG/TDP” model but has a similar model.

DISPOSITION
NRC staff corrected the text to state that a form of a VG/TDP model was used.

**COMMENT 14**
AREVA noted in Section 3.4.1 that the statement "It takes approximately 1.5 seconds . . ." should be consistent with Figure 6-1.

DISPOSITION
NRC staff corrected the text to 3 seconds.

**COMMENT 15**
AREVA noted in the fourth paragraph of Section 3.4.3 that in this context the term DNB is more appropriate than DNBR.

DISPOSITION
NRC staff corrected the text to use DNB instead of DNBR.

**COMMENT 16**
AREVA noted that Figures 4 through 7 were difficult to read, and better figures may be desirable.

DISPOSITION
NRC staff were unable to obtain better figures for use in the SER. No changes were made to the SER.

**COMMENT 17**
AREVA noted in Section 4.0 that while AREVA would prefer that these changes be approved for all reactivity insertion accidents it is clear that the NRC wants to be more restrictive in its approval, and that the language in the numbered list was not consistent with Section 5.0 of the SER.

DISPOSITION
NRC staff changed the text to be consistent with Section 5.0 of the SER.

**COMMENT 18**
AREVA noted in Section 5.0 that the topical report requested approval of a set of models, not a set of coding.

DISPOSITION
NRC staff changed the text of Limitation No. 1 to reference models instead of codes.