Safety Evaluation Report

Transition to AREVA NP Fuel
and Safety Analysis Methodology

Calvert Cliffs Nuclear Power Plant

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 297 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 273 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 Introduction

By letter dated November 23, 2009, (Agencywide Document Access and Management System (ADAMS) Accession No. ML093350189), as supplemented by letters dated January 26 (ML100261541), April 22 (ML101160081), July 23 (ML102070551), August 9 (ML102230034), October 29 (ML103080024), November 19 (ML103280081), and December 30, 2010 (ML110040369), and January 14 (ML110180606), January 18 (ML110200065), January 28 (ML110320243), February 11, 2011 (ML110450537), and February 15, 2011 (ML110470403), Calvert Cliffs Nuclear Power Plant, LLC, the licensee for the Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2 requested to modify the CCNPP licensing basis and Technical Specifications (TSs) as necessary to transition from Westinghouse-supplied fuel and licensing safety analysis to AREVA NP-supplied products[1]. The amendments would revise the licensing basis and the TSs to allow the use of AREVA Advanced CE-14 High Thermal Performance (HTP) fuel in the Calvert Cliffs reactors. The AREVA Advanced CE-14 HTP fuel design consists of standard uranium dioxide ($U0_2$) fuel pellets with gadolinium oxide (Gd_20_3) burnable poison and M5 cladding. The requested licensing action also included a request for exemption as necessary to use the M5 fuel cladding product. The exemption was granted by letter dated January 13, 2011 (ML103070113).

The requested changes to the TSs include several types of revisions. The licensee proposed to revise TS 2.1.1, “Reactor Core SLs [Safety Limits],” to replace a linear heat rate-based safety limit with one that is based on fuel centerline temperature. The TS changes also include a request to delete power distribution limits and surveillance requirements associated with the total planar radial peaking factor (F^T_{xy}). The licensee proposed to revise TS 4.2, “Reactor Core,” of TS 4.0, “Design Features,” to eliminate language related to advanced cladding material for lead test assemblies, and to include the proprietary M5 cladding material in the reactor matrix description. Finally, the licensee is proposing to add AREVA methodology references to TS

[1] Trepanier, T. E., Constellation Energy, letter to U.S. NRC, “License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel,” Dockets 50-317 and 50-318, November 23, 2009. ML093350099.

5.6.5, "Core Operating Limits Report (COLR)," References list, and to remove outdated, legacy references that are no longer in use.

The letters dated July 23, August 9, October 29, November 19, December 30, 2010, and January 14, January 18, January 28, February 11, and February 15, 2011, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 Fuel System Design

CCNPP currently uses Westinghouse Turbo 14x14 fuel assemblies in both Unit Nos. 1 and 2. The licensee has experienced fuel failures in numerous operating cycles, including recent operating cycles, related to grid-to-rod fretting wear of the fuel cladding. As a result, the licensee has chosen to replace the existing Westinghouse Turbo fuel with AREVA Advanced CE-14 HTP fuel for use in the CCNPP reactors. AREVA Advanced CE-14 HTP fuel consists of dimensionally similar fuel as the current Westinghouse Turbo fuel. In addition to changing reactor fuel, the licensee is changing their licensing basis by transitioning from Westinghouse fuel design and evaluation methods to AREVA fuel design and evaluation methods.

The Nuclear Regulatory Commission (NRC) staff's review of the AREVA CE14HTP fuel design and its introduction into CCNPP is described below. For the co-resident TURBO fuel bundles, Westinghouse will continue to evaluate its performance using approved methods to ensure that established fuel thermal-mechanical design criteria (e.g., dimensional clearances, rod internal pressure, fatigue) are satisfied. Westinghouse will continue this practice as long as TURBO fuel bundles are loaded in CCNPP cores.

2.1 Regulatory Evaluation

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to General Design Criteria (GDC)-10, GDC-27, and GDC-35 is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design". In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

The proposed TS changes including a list of the previously approved AREVA fuel design requirements, fuel thermal-mechanical design methodology, and reload design methodologies are provided in Reference 1. The staff's review of these topical reports (TRs) is to ensure that the fuel design criteria, thermal-mechanical design methodology, and reload design methodologies remain valid and that the AREVA CE14HTP fuel assembly design adequately addresses the regulatory requirements identified in Standard Review Plan (SRP) 4.2.

As discussed in Reference 1, CCNPP intends to transition from the Westinghouse Turbo 14x14 fuel assembly design to the AREVA Advanced CE-14 HTP fuel assembly design beginning in 2011 for Unit 2 and 2012 for Unit 1. The staff's review of the requested CCNPP fuel transition is summarized below:

- Verify that the previously approved topical reports are being applied consistent with the staff's review and approval and that all limitations and conditions are satisfied.
- Verify that the fuel assembly components and fuel rod design criteria are consistent with regulatory criteria identified in SRP Section 4.2.
- Verify that the fuel thermal-mechanical design methodology is capable of accurately or conservatively evaluating each component with respect to its applicable design criteria.
- Verify that the reference CE14HTP fuel assembly design satisfies the regulatory requirements.
- Verify that AREVA's experience database supports the operating limits being requested.

As part of its review of the CCNPP fuel transition, the NRC staff conducted a desk audit at NRC headquarters of the supporting AREVA engineering calculations. Issues identified via this audit were discussed with the licensee and AREVA at two separate meetings (August and December 2010). Where necessary, a request for additional information (RAI) was generated to capture information important for the staff's safety findings. These items are described below.

2.2 CE14 HTP Fuel Assembly Design

The CE14HTP fuel assembly design features are described in Section 1 and Section 2 of the CCNPP Reload Transition Report (Attachment 4 of Reference 1). CE14HTP design specifications are listed in Table 2-1 and illustrations are provided in Figures 1-1 through 1-6 of the Reload Transition Report.

Section 2.3 of Reference 1 describes the fuel thermal-mechanical design methodology used to assess the performance of the CE14HTP fuel design. Specifically, the generic pressurized-water reactor (PWR) fuel mechanical design criteria contained in EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," were employed along with the approved fuel thermal-mechanical methodology in BAW-10227(P)(A)[2] and BAW-10240(P)(A)[3]. AREVA's generic fuel mechanical design criteria were developed to provide the following assurances:

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[2] Mitchell, D. B. and B. M. Dunn, Framatome Cogema Fuels, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227-A, February 2000. ML003686365.
[3] Meginnis, A. B., et al., Framatome ANP, "Incorporation of M5^{TM} Properties in Framatome ANP Approved Methods," BAW-10240(NP)-A, May, 2004. ML042800314.

- The fuel assembly shall not fail as a result of normal operation and abnormal operational occurrences (AOOs). The fuel assembly dimensions shall be designed to remain within operational tolerances and the functional capabilities of the fuels shall be established to either meet, or exceed those assumed in the safety analysis.
- Fuel assembly damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from in-plant handling and shipping.

Table 2-2 of Reference 1 provides the specific thermal-mechanical design criteria used to assess the performance of the CE14HTP fuel design. These design criteria are consistent with Section 4 of the SRP and, therefore, acceptable.

It is required that previously approved models and methodologies be applied in a manner consistent with their approval. The licensee provided a list of all of the staff's conditions and limitations imposed on the 15 AREVA (formally Exxon Nuclear Company and Framatome Cogema Fuels) topical reports[4]. EMF-92-116(P)(A) contained two safety evaluation (SE) conditions and limitations. The first restriction limits the application of the generic design criteria to a fuel rod burnup of 62 gigawatt days/metric ton uranium (GWd/MTU). This item is discussed further below. The second restriction requires a submittal to the NRC documenting the design evaluation process and conformance. In their response, the licensee stated that this restriction did not apply since the CCNPP CE14HTP fuel design did not constitute a new fuel design. The NRC staff rejects this assertion. Any first batch application of a new or modified assembly component design or material or first batch application of a prior assembly design or material to a new reactor meets the staff's interpretation of a new fuel design and necessitates formal notification in accordance with this SE restriction. The CCNPP Reload Transition Report (Attachment 4 of Reference 1) satisfies this SE restriction.

The NRC staff's review and approval of BAW-10227(P)(A) was limited to the application of M5 alloy cladding to Mark-B (B&W type reactors, 15x15 array) up to 62 GWd/MTU and Mark-BW (Westinghouse type reactors, 15x15 and 17x17 arrays) up to 62 GWd/MTU. Whereas, the staff's review and approval of BAW-10240(P)(A) states that the methodology is applicable for use in support of licensing actions for Westinghouse and Combustion Engineering plants for fuel burnup to 62 GWd/MTU. Constellation is currently limited to a peak rod average burnup of 60 GWd/MTU for the resident Westinghouse fuel designs.

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[4] Gellrich, G. H., Constellation Energy, letter to U.S. NRC, "Supplement to the License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," Dockets 50-317 and 50-318, January 26, 2010. ML100261521.

2.2.1 CE14 HTP Fuel Assembly Design Analysis

Section 2 of the Reload Transition Report (Attachment 4, Reference 1) documents the fuel mechanical design analysis. Evaluations were completed to confirm the mechanical compatibility of the CE14HTP fuel assembly design to the CCNPP reactor core internals, fuel handling equipment, storage racks, and co-resident Westinghouse Turbo fuel. With the exception of the MONOBLOC™ guide tubes and lower Alloy 718 grid strap, the CE14HTP fuel design is identical to the AREVA lead test assemblies already being irradiated at CCNPP. Pool-side examinations have been performed on the 4 lead test assemblies (LTAs) with no anomalous results. Two of the LTAs have completed a third cycle of irradiation and were discharged with a peak rod burnup of approximately 70 GWd/MTU. These high burnup LTAs were inspected prior to loading the full batch of CE14HTP fuel.

The guide tubes material for the CE14HTP fuel assembly design is Zircaloy-4. The growth characteristics of Zircaloy-4 are well established. As such, the previous operating experience associated with excessive assembly growth in M5 guide tubes is not applicable to this fuel assembly design.

Based upon the information presented in the Reload Transition Report and the LTA operating experience, the NRC staff finds the CE14HTP fuel assembly design acceptable.

2.2.2 CE14 HTP Fuel Rod Design Analysis

As part of a desk audit, the NRC staff reviewed the AREVA fuel rod thermal-mechanical design calculation (32-9135500-001). AREVA's RODEX2 code, documented in XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Model," was used to simulate a fuel rod's performance under normal steady-state irradiation and during AOOs. Recognizing that this legacy code lacked a fuel thermal conductivity model which accurately captures its degradation with increasing exposure, the staff had concerns with the application of RODEX2 for the CCNPP fuel transition. Lack of a burnup dependent fuel thermal conductivity model would impact (in a non-conservative manner) several fuel design analyses, including the fuel rod internal pressure, fuel centerline melt, and fuel rod cladding strain design analyses, as well as downstream inputs to safety analyses. (The downstream inputs are discussed in Section 5.0 of this SE)

In response to an RAI regarding the development and application of penalty factors to offset RODEX2's deficient fuel thermal conductivity model[5,6], the licensee described the basis for each penalty factor. With respect to fuel centerline melt, the licensee performed a code-to-code comparison between COPERNIC[7] and RODEX2 (1984) to generate burnup, and fuel type (e.g., UO_2, Gd-2%, Gd-8%) dependent temperature penalties. For example, the UO_2 fuel code-to-

[5] Gellrich, G. H., Constellation Energy, letter to U.S. NRC, "Supplement to the License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," Dockets 50-317 and 50-318, November 19, 2010. ML103280082. RAIs 21 and 22.
[6] Gellrich, G. H., Constellation Energy, letter to U.S. NRC, "Supplement to the License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," Dockets 50-317 and 50-318, December 30, 2010. ML110040374. RAI 29.
[7] Framatome ANP, "COPERNIC Fuel Rod Design Computer Code," BAW-10231(NP)-A, January 2004. ML042930240.

code comparison yielded a **[[ ]]** penalty at 18 GWd/MTU which increased to a **[[ ]]** at 35 GWd/MTU. With respect to cladding fatigue and AOO cladding strain, a fuel diameter adjustment was calculated based upon the burnup dependent fuel temperature penalty established by comparing RODEX2 to the extended empirical database. For example, the application of the fuel diameter adjustment yields an additive penalty of **[[ ]]** on calculated AOO strain. With respect to rod internal pressure, **[[ ]]**

Prompted by concerns with respect to the impact of the deficient fuel thermal conductivity on the fuel design analyses, the NRC staff performed independent calculations using the FRAPCON-3.4 fuel thermal-mechanical performance code[8]. A recently released FRAPCON statistical package was utilized for this examination. This pre- and post-processor randomly samples manufacturing tolerances (e.g., pellet diameter, cladding thickness, stack height) and modeling uncertainties (e.g., fission gas release, fuel thermal expansion) in a Monte-Carlo type simulation. For this examination, the average and 95/95 upper tolerance limit (UTL) for each of the important parameters were calculated based upon a set of 500 FRAPCON cases.

2.2.2.1 Rod Internal Pressure Specified Acceptable Fuel Design Limit (SAFDL)

The AREVA rod internal pressure limit is **[[ ]]** psia above system pressure (i.e., **[[ ]]** psia). This criterion ensures that the fuel-to-cladding gap does not reopen as a result of cladding creep in excess of fuel irradiation swelling. This no clad liftoff (NCLO) criterion is consistent with SRP-4.2 and, therefore, acceptable.

Table 1 provides the results of the NRC staff's independent calculations of rod internal pressure along with the limiting values from the AREVA design analysis. For the limiting UO_2 fuel rod power history shown in Figure 1 (with the addition of power adjustments and AOO power ramps in accordance with XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"), RODEX2 calculated a maximum rod internal pressure of **[[ ]]** psia. Neglecting the power adjustments and power ramps and following the cycle-specific rod depletion power history depicted in Figure 1, a best-estimate, nominal FRAPCON-3.4 calculation yielded a peak rod internal pressure of **[[ ]]** psia. With the application of manufacturing tolerances and fission gas release model uncertainty, FRAPCON-3.4 predicted a 95/95 UTL rod internal pressure of **[[ ]]** psia. At **[[ ]]** psia, the difference in predicted rod internal pressure contributed to the staff's concerns.

In response to an RAI regarding the difference in predicted rod internal pressure, the licensee provided RODEX2-to-FRAPCON-3 comparisons for three different rod power histories[9]. AREVA executed the publically available FRAPCON-3 code as part of this benchmark. For each RODEX2 case, the rod power history was adjusted and power ramps were added in accordance with XN-NF-82-06(P)(A). None of these power adjustments were included for the FRAPCON-3 calculations. Figure 28-1 of Reference 9 illustrates the various rod power histories. Based upon this benchmark, the license concludes that the RODEX2 methodology

[8] U.S. Nuclear Regulatory Commission, "FRAPCON-3.4: AComputer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," NUREG/CR-7022, Vol. 1, August 31, 2010. ML102930541.
[9] Trepanier, T. E., Constellation Energy, letter to U.S. NRC, "Supplement to the License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," Dockets 50-317 and 50-318, January 14, 2011. ML110180621. RAI 28.

becomes significantly more conservative at higher duty fuel rod power histories and that the methodology is appropriate and conservative.

The licensee's conclusion with respect to the RODEX2-to-FRAPCON-3 benchmark relies upon the difference in rod power history, which is a direct user input to the two codes. Based upon review of Section 6 of XN-NF-82-06(P)(A), the NRC staff does not believe the intent of these power adjustments was to compensate for a non-conservative fuel thermal conductivity or to compensate for modeling uncertainties in general. Cycle-specific rod power histories are extracted from the core physics reload depletion calculations, which are based upon projected core power profiles for the upcoming cycle. The adjustments in XN-NF-82-06(P)(A) bridge the gap between cycle-specific reload depletion calculations and Core Operating Limits Report (COLR) allowable power operations. Without adjustments, the results of the fuel thermal-mechanical design analysis would no longer be valid when any plant operations or maneuvering went beyond the projected core reload depletion.

Recognizing the need to provide operational flexibility and solidify the basis of TS core operating limits, most approved fuel rod thermal-mechanical design methodologies explicitly account for the difference between projected core reload depletions and COLR allowable power operations. Furthermore, model uncertainties need to be properly addressed as best-estimate predictions are not adequate to demonstrate compliance to SAFDLs.

If the RODEX2-to-FRAPCON-3 benchmark (documented in response to RAI #28 of Reference 9) had used identical rod power histories, the difference between the two code predictions decreases (and potentially reverses) and the basis for concluding that RODEX2 rod internal pressure predictions are adequate and conservative becomes questionable. For example, with the application of a 5% uncertainty on the Cycle 20 core reload depletion rod power history (consistent with XN-NF-82-06(P)(A) for less limiting cycles), FRAPCON-3.4 predicted a 95/95 UTL rod internal pressure of **[[ ]]** psia. The application of the remaining two adjustments **[[ ]]** would certainly increase the FRAPCON-3.4 predicted rod internal pressure and exacerbate the difference between the two codes.

The results of the NRC staff's FRAPCON-3.4 calculations and the licensee's response to RAI#28 did not resolve concerns regarding the application of the legacy RODEX2 fuel performance code to CCNPP. However, the results of the FRAPCON-3.4 calculations do provide a degree of assurance that the CE14HTP fuel rod design will not fail as a result of rod internal pressure. Examination of Table 1 reveals that the rod internal pressure criteria are satisfied. Specifically, the maximum calculated rod internal pressure (UO_2+Tol+FGR+CY20Pow, **[[ ]]** psia) is less than the limit of **[[ ]]** psia and the gadolinia fuel rods are less limiting that the UO_2 fuel rods. This conclusion is based on the relatively benign rod power histories shown in Figure 1. As described in Appendix F of 32-9135500-001, the licensee intends to employ more aggressive fuel utilization in future cycles with a full core of AREVA CE14HTP fuel. As such, the staff's calculations and conclusions may not be applicable to future cycles.

The rod internal pressure criterion is an end-of-life concern, as fission gas is released and rod internal pressure builds with increasing exposure. In addition, the impact of the deficient fuel thermal conductivity model increases with exposure. As such, irradiation of CE14HTP fuel up to a burnup of 40 GWd/MTU (bounds first cycle of operation) is acceptable.

For future CCNPP cycles, a reduction of the rod internal pressure limit is required to compensate for the RODEX2 methodology which does not explicitly model degraded fuel thermal conductivity nor adequately account for modeling uncertainties. Based upon the large differences in predicted rod internal pressure relative to FRAPCON-3.4 (see Table 1), future CCNPP cycles which rely on the RODEX2 methodology must ensure that predicted maximum rod internal pressure remains below system pressure.

The NRC staff has concluded that a license condition is necessary to capture the more restrictive design criteria for CCNPP reload designs. License Condition 1 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. It should be noted that License Conditions 1 and 2, which will be discussed later in this SE, are linked in Section 7.0 and Appendix C, License Conditions, of the CCNPP operating license. The following summarizes the license condition.

License Condition 1:

Predicted rod internal pressure shall remain below the steady state system pressure.

TABLE 1: FRAPCON-3.4 Rod Internal Pressure Calculations

Case	**Nominal** (psia)	**95/95 UTL** (psia)
UO2 Fuel Rod (power history shown on Figure 1)		
AREVA RODEX2	**[[ ]]**	
UO2 Fuel Rod	**[[ ]]**	- -
UO2+Tol	- -	**[[ ]]**
UO2+Tol+FGR	- -	**[[ ]]**
UO2+Tol+FGR+CY19Pow	- -	**[[ ]]**
UO2+Tol+FGR+CY20Pow	- -	**[[ ]]**
Gadolinia Fuel Rod (power history shown on Figure 1)		
AREVA RODEX2	**[[ ]]**	
Gd2% Fuel Rod	**[[ ]]**	- -
Gd2%+Tol	- -	**[[ ]]**
Gd2%+Tol+FGR	- -	**[[ ]]**
Gd2%+Tol+FGR+CY20Pow	- -	**[[ ]]**

Notes:

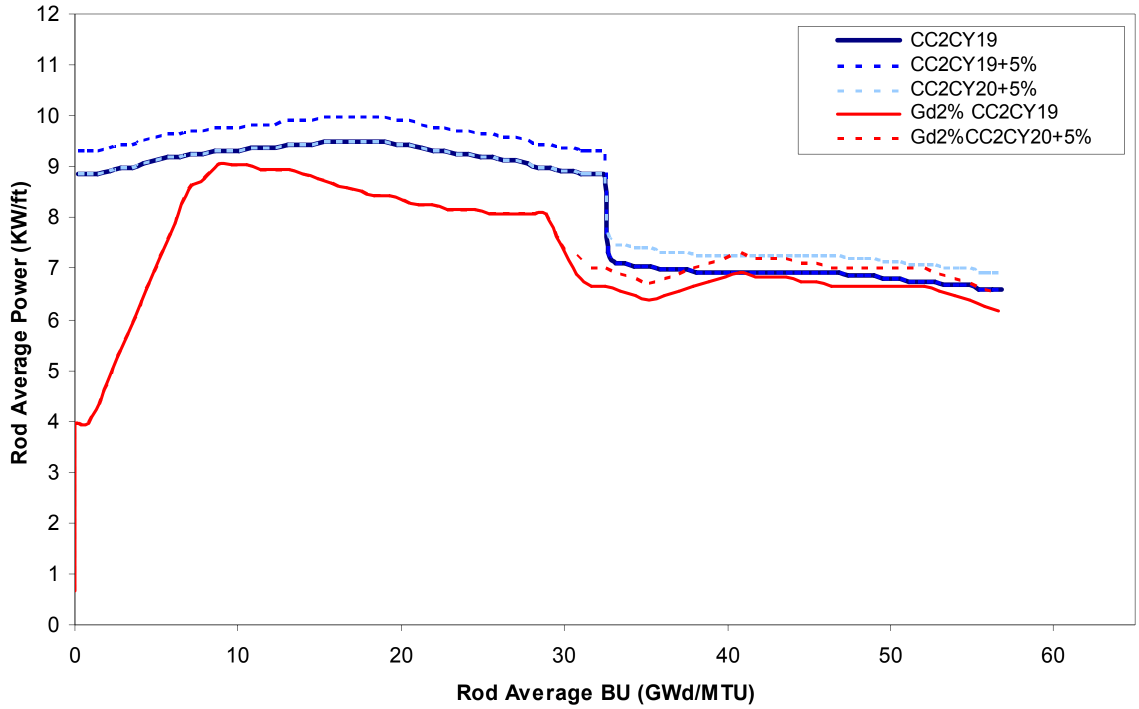
a) Tol = random sampling of manufacturing tolerances on cladding thickness, pellet diameter, and stack height (i.e., plenum volume).

b) FGR = random sampling of FGR model uncertainty.

c) CY19Pow = 5% multiplier on Cycle 19 fuel rod power history.

d) CY20Pow = 5% multiplier on Cycle 20 fuel rod power history.

FIGURE 1: Calvert Cliffs Unit 2 Cycle 19 Rod Power Histories



2.2.2.2 Fuel Centerline Melt SAFDL

The AREVA design criteria preclude fuel centerline melting during normal operation and AOOs. This criterion ensures fuel cladding integrity by avoiding the large volumetric expansion within the fuel which occurs during the phase transition and the resulting cladding strain. This no fuel centerline melting criterion is consistent with SRP-4.2 and therefore acceptable.

For events which experience a rapid power excursion (e.g., control rod ejection), the S-RELAP5 hot spot model was used to predict fuel centerline temperature. These events are described below. For the slower events, RODEX2 is used, in conjunction with COPERNIC-based penalties (discussed above), to define a peak linear heat generation rate (LHGR) limit corresponding to incipient centerline melt (referred to as LHGR fuel centerline melt (FCM) safety limit). For CCNPP, the LHGR FCM safety limit is **[[ ]]** kW/ft. The limiting event with respect to challenging the LHGR FCM safety limit is the control element assembly (CEA) Drop event which experiences a peak LHGR of **[[ ]]** kW/ft.

Table 2 provides the results of the NRC staff's independent calculations of fuel centerline temperature. Starting with the limiting UO_2 power history in Figure 1, an AOO power ramp was simulated at time step 45 (end-of-cycle (EOC) 19, 33 GWd/MTU) up to a peak LGHR of **[[ ]]** kW/ft (CEA Drop event). A second case was run with a peak LHGR of **[[ ]]** kW/ft (FCM safety limit). Recognizing that AREVA's transient analysis methodology combined with a steady-state fuel performance code contains significant conservatism (relative to dynamic fuel centerline temperature calculations with local reactivity feedback), the staff elected to randomly sample manufacturing tolerances, but not to apply a fuel temperature model uncertainty in their FRAPCON-3.4 calculations. At the LHGR FCM safety limit **[[( )]]**, FRAPCON-3.4 predicts a nominal fuel centerline temperature of 5068 °F which is above the incipient centerline melting temperature of 4890 °F [(5080 °F – (58 °F/10 GWd/MTU)), based on 33 GWd/MTU local burnup].

The results of the NRC staff's FRAPCON-3.4 calculations did not resolve concerns regarding the application of the legacy RODEX2 fuel performance code to CCNPP. However, the results of the FRAPCON-3.4 calculations do provide a degree of assurance that the CE14HTP fuel rod design will not fail as a result of fuel centerline melting for CC2CY19. Specifically, the FRAPCON predicted 95/95 UTL peak fuel centerline temperature of 4727°F (at CEA Drop peak LHGR of **[[ ]]** kW/ft) remained below the melting temperature. However, the CC2CY19 CEA Drop radial peaking augmentation factors may not be applicable to future reloads.

Figure 2 provides the results of a FRAPCON sensitivity study where peak LHGR was varied between 19.0 and 22.0 kW/ft. Examination of the figure reveals that fuel centerline melting is reached with a peak LHGR of approximately 21.0 kW/ft. This study is based upon a "knee" in the radial falloff curve (rod power versus burnup) at 33 GWd/MTU. For future CCNPP cycles, a reduction of the LHGR FCM safety limit is required to compensate for the RODEX2 methodology, which does not explicitly model degraded fuel thermal conductivity.

The NRC staff has concluded that a license condition is necessary to capture the more restrictive design criteria for CCNPP reload designs. License Condition 2 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. As previously stated, License Conditions 1 and 2 are linked in Section 7.0 and Appendix C, License Conditions, of the CCNPP operating license. The following summarizes the license condition.

> <u>License Condition 2:</u>
>
> The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 KW/ft.

The FRAPCON-3.4 independent calculations performed by the NRC staff demonstrate that the thermal conductivity penalty factor applied to UO_2 fuel rods, which was derived based on the COPERNIC FCM limit in Figure 29-1 of Reference 6, was not sufficient. To evaluate the thermal conductivity penalty factors for the Gadolinia bearing fuel rods, the staff ran additional FRAPCON-3.4 cases. Figure 3 provides the results of a FRAPCON sensitivity study where peak LHGR was varied between 18.0 and 21.0 kW/ft for fuel designs containing 2%, 4%, 6%, and 8% Gadolinia. A comparison of FRAPCON's FCM threshold (from Figure 3) to COPERNIC's FCM limit in Figures 29-2 through 29-5 of Reference 6 (basis of penalties factors) reveals that the thermal conductivity penalty factors for the Gadolinia bearing fuel rods are conservative.

TABLE 2: FRAPCON-3.4 Fuel Centerline Melt Calculations

Case	**Average** (°F)	**Std. Dev.** (°F)	**95/95 UTL** (°F)
AOO assumed at EOC 19 (~ 33 GWd/MTU local)			
UO_2+Tol - **[[ ]]** kW/ft	**[[ ]]**	**[[ ]]**	**[[ ]]**
UO_2+Tol - **[[ ]]** kW/ft	**[[ ]]**	**[[ ]]**	**[[ ]]**

Notes:

a) Tol = random sampling of manufacturing tolerances on cladding thickness, and pellet diameter.

FIGURE 2: FRAPCON-3.4 Fuel Centerline Melt Study – UO2 Fuel
(AOO Power Ramp at 33 GWd/MTU)

FIGURE 3: FRAPCON-3.4 Fuel Centerline Melt Study – Gd Fuel
(AOO Power Ramp at 33 GWd/MTU)

2.2.2.3 Fuel Cladding Strain

AREVA's M5 alloy cladding design criterion limits predicted cladding strain during an AOO to less than 1.0%. This limit is consistent with SRP-4.2 and, therefore, acceptable. Unlike the fuel centerline melting analysis where the target is peak LHGR, the cladding strain analysis investigates the change in local LHGR (ΔF_q). In response to an RAI regarding calculated cladding strains[10], the licensee provided a table of local power peaking (pre- and post-event) and RODEX2 calculated cladding strain for the hot full-power (HFP) Excess Load, HFP CEA Drop, and HFP CEA Bank Withdrawal event. As described above, AREVA generated an additive penalty of **[[ ]]** on calculated AOO strain to account for the lack of a burnup-dependent fuel thermal conductivity model. The maximum RODEX2 calculated cladding strain was **[[ ]]** (including additive penalty) based up the CEA Drop event with a power factor **[[( )]]** of **[[ ]].**

Table 3 provides the results of the NRC staff's independent calculations of fuel cladding strain. Starting with the limiting UO2 power history in Figure 1, an AOO power ramp was simulated at

[10] Reference 9 op. cit., RAI 3, ML110180621.

time step 45 (33 GWd/MTU) up to a peak LHGR of **[[ ]]** kW/ft **[[ ]]** = **[[ ]]** kW/ft). A second case was run with an AOO power ramp at time step 75 (52 GWd/MTU) up to a peak LHGR of **[[ ]]** kW/ft **[[( ]]** = **[[ ]]** kW/ft). Following the AREVA methodology, a **[[ ]]** was used during the AOO. Recognizing that AREVA's transient analysis methodology combined with a steady-state fuel performance code contains significant conservatism (relative to dynamic fuel temperature calculations with local reactivity feedback), the staff elected not to apply a fuel temperature model uncertainty. The statistical FRAPCON-3.4 calculations randomly sample manufacturing tolerances along with an uncertainty on fuel thermal expansion.

For the CEA Drop analysis, FRAPCON-3.4 predicted a nominal cladding strain of 0.742% and a 95/95 UTL of 1.075% at 33 GWd/MTU and a nominal cladding strain of 0.528% and a 95/95 UTL of 0.740% at 52 GWd/MTU. The RODEX2 predicted cladding strain (including the thermal conductivity correction), **[[ ]],** is approximately equal to the nominal FRAPCON-3.4 prediction. These benchmark results are similar to the trend seen in predicted rod internal pressure. The results of the NRC staff's FRAPCON-3.4 calculations did not resolve concerns regarding the application of the legacy RODEX2 fuel performance code to CCNPP. However, the results of the FRAPCON-3.4 calculations do provide a degree of assurance that the CE14HTP fuel rod design will not fail as a result of cladding strain for CC2CY19. While the predicted FRAPCON-3.4 cladding strain is slightly above the 1.0% design criterion at 33 GWd/MTU, the actual failure threshold for cladding with no excess hydrogen (i.e., no precipitated zirconium hydrides) is at least 1.5% total strain. With the beneficial corrosion properties of M5 alloy cladding, the CE14HTP fuel rods are unlikely to exceed the hydrogen solubility at operating temperatures (90-100 weight parts per million) for high power, low-to-middle burnup fuel rods. By the time absorbed hydrogen exceeds solubility and forms zirconium hydride precipitates, rod power will have diminished (with U_{235} depletion) and predicted cladding strain will remain below 1.0%. Thus, the beneficial corrosion properties of M5 alloy coupled with the 1.0% design criterion provides adequate margin to compensate for any potential non-conservatism with the corrected RODEX2 cladding strain predictions.

Based upon independent calculations and the M5 alloy corrosion properties, the NRC staff finds that the CE14HTP satisfies the cladding strain criterion and the use of RODEX2 (with the thermal conductivity penalty) up to a predicted cladding strain of 1.0% is acceptable.

TABLE 3: FRAPCON-3.4 Fuel Cladding Strain Calculations

Case	**Nominal** (%)	**95/95 UTL** (%)
CEA Drop **[[ ]]** Power Factor at 33 GWd/MTU		
AREVA RODEX2	**[[ ]]**	
Nominal	**[[ ]]**	- -
Tol	- -	**[[ ]]**
Tol+Exp	- -	**[[ ]]**
CEA Drop **[[ ]]** Power Factor at 52 GWd/MTU		
Nominal	**[[ ]]**	- -
Tol	- -	**[[ ]]**
Tol+Exp	- -	**[[ ]]**

Notes:

a) Tol = random sampling of manufacturing tolerances on cladding thickness, and pellet diameter.

b) Exp = random sampling of fuel thermal expansion model uncertainty.

2.2.3 Operating Experience

CCNPP CE14HTP fuel assembly features Zircolay (Zry)-4 corner and center guide tubes, seven Zry-4 HTP grid spacers, one Alloy 718 High Mechanical Performance (HMP) grid spacer at the lowest position, M5 alloy fuel rod cladding, FUELGUARD lower tie plate, and a reconstitutable upper tie plate. Each of these assembly components and materials has considerable operating experience.

Section 2.4 of Attachment 4 to Reference 1 describes the operating experience for the HTP assembly components and M5 alloy cladding. Included in the vast experience database of over 10,000 HTP fuel assemblies in 45 nuclear power plants are over 1,000 CE 14x14 HTP fuel assemblies in 5 different CE reactors. HMP grid spacer designs have been included in over 4,000 of the HTP assemblies. With respect to M5 alloy cladding, over two million fuel rods have been manufactured and operated. Over 2,500 HTP fuel assemblies combined with M5 alloy cladding have been irradiated up to a maximum assembly average burnup of 61 GWd/MTU. Significant operating experience has also been achieved for the MONOBLOC guide tubes, FUELGUARD lower tie plate, and reconstitutable upper tie plate.

In addition to this operating experience, CCNPP has irradiated 4 AREVA HTP LTAs. Pool-side examinations have been performed on the 4 LTAs with no anomalous results. Two of the LTAs are currently in a third cycle of irradiation and are expected to be discharged with a peak rod burnup of approximately 70 GWd/MTU. These high burnup LTAs will be inspected prior to loading the full batch of CE14HTP fuel. A comparison between the LTAs and the batch

CE14HTP fuel assemblies is presented in Table 2-1 of Reference 1. Based upon relevant operating experience at Millstone Unit 2, the lowest HTP grid spacer is being replaced by an Alloy 718 HMP grid spacer.

Based on the NRC staff's review in Section 3.1 and Section 3.2, along with the extensive operating experience detailed in Section 2.4 of Attachment 4 to Reference 1, the staff finds the CE14HTP fuel assembly design and design methodology (with the RODEX2 provisions described in Section 3.2) acceptable up to a rod average burnup of 62 GWd/MTU.

3.0 Nuclear Design

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core. The staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation.

3.1 Regulatory Evaluation

The NRC's acceptance criteria are based on the GDC of Appendix A to 10 CFR 50:

1. GDC 10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

2. GDC 11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity;

3. GDC 12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed;

4. GDC 20, insofar as it requires that the reactor core be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions;

5. GDC 25, insofar as it requires that the protection system be designed to assure that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems;

6. GDC 26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes from planned, normal power changes;

7. GDC 27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and

8. GDC 28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Specific review criteria are contained in SRP Section 4.3.

3.2 Technical Evaluation

The licensee evaluated the effects of transitioning from Westinghouse Turbo fuel to AREVA CE14HTP fuel on the nuclear design bases and the methodologies for Calvert Cliffs.

The licensee ensures that the acceptance criteria listed above are met as follows. The licensee adheres to GDC 10 by establishing TSs and core operating limits for the power distributions and the linear heat rate. The licensee ensures that the Doppler coefficient of reactivity is negative at all operating conditions, that the power coefficient is negative at all operating power levels relative to hot zero power, and that the moderator temperature coefficient remains within limits established in the TSs. Conformance to these requirements for the reactivity coefficients ensures compliance with GDC 11. The fuel is designed and loaded such that uncharacteristic power oscillations due to fuel design and loading do not occur, and the TSs limit the allowable azimuthal power tilt. These items ensure compliance with GDC 12. Finally, the core is designed with margin to the TS value for minimum shutdown margin, with an allowance for the most reactive rod to be stuck out of the core. While the licensee referred to GDC 28 in this statement, the NRC staff believes instead that this design requirement ensures compliance with GDC 27.

The licensee performs the nuclear design analysis to establish the operating limits described above using the following methods:

- EMF-96-029(P)(A) Volumes 1 and 2, “Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results,” Siemens Power Corporation, January 1997,

- XN-75-27(A) and supplements 1 – 5, “Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors,” Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981, Supplement 4 dated 1986 and Supplement 5 dated February 1987, and

- XN-NF-78-44(NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” Exxon Nuclear Company, October 1983.

EMF-96-029 has three limitations associated with it:

1. SAV95 (the code that supports EMF-96-029) application will be supported by additional code validation to ensure that the methodology and uncertainties are applicable:

    a. For designs differing from the Westinghouse reactors with 157 fuel assemblies with either 15x15 or 17x17 fuel rod arrays, and CE reactors with 217 fuel assemblies with 14x14 fuel rod arrays

    b. When using incore monitoring systems differing from the INPAX-W and INPAX-2 systems contained in this SE when Siemens Power Corporation (SPC) provides input from SAV95

2. Modifications to the code and methodology will be validated using the criteria approved in EMF-96-029

3. The validation will be maintained by SPC and be available for NRC audit

Since the CCNPP units are 217-fuel assembly CE reactors with 14x14 fuel rod arrays, the NRC staff finds that Item 1.a. is satisfied. The licensee also stated that CCNPP will be utilizing INPAX-2. The staff finds that this statement satisfies Item 1.b. Limitations 2 and 3 are tied to vendor oversight and are not directly applicable to the CCNPP-specific implementation of this methodology; however, the staff notes that the licensee stated that neither methodology nor code modifications were expected for CCNPP.

Although the licensee did not address any conditions or limitations on the approval of XN-75-27, the NRC staff performed a cursory review of the licensing topical report. It describes the nuclear design methodology, making reference to legacy codes that have been replaced by the SAV95 code system described in EMF-96-029. The staff determined that the licensee is implementing the SAV95 code system using the methodology described in XN-75-27. The safety evaluation approving XN-75-27 contained no conditions or limitations. Based on the staff's cursory review of both topical reports, the fact that the licensee has satisfied the implementation condition associated with EMF-96-029, and is using the SAV95 code system in accordance with the NRC-approved methodology described in XN-75-27, the staff finds the implementation of both licensing topical reports acceptable for CCNPP.

XN-NF-78-44 is a CEA accident analytic methodology. The CEA accident, along with the implementation of this methodology, is discussed in Section 5.2 of this SE.

Compliance with GDC 20, 25, 26, and 28 are demonstrated by performing acceptable safety analyses of postulated transients and accidents, and showing that the results meet the applicable acceptance criteria. Accident and transient analysis is addressed Section 5 of this SE.

3.3 Conclusion

The NRC staff has reviewed the licensee's evaluation and proposed methods implementation related to the requested fuel transition. The staff concludes that the licensee has adequately accounted for the effects of the proposed fuel transition on the nuclear design and has provided reasonable assurance that the fuel design limits will not be exceeded during normal or anticipated transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDC 10, 11, 12, 20, 25, 26, 27 and 28. Therefore, the staff finds the proposed fuel transition acceptable with respect to the nuclear design.

4.0 Thermal Hydraulic Design

4.1 Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design

1. Has been accomplished using acceptable analytical methods;

2. Is equivalent to, or a justified extrapolation from proven designs; and

3. Provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs;

The review also covered hydraulic loads on the core and RCS components during normal operation and the design basis accident (DBA). The NRC's acceptance criteria are based on GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. Specific review criteria are contained in SRP Section 4.4.

4.2 General Information

The thermal-hydraulic design of the core is evaluated using the XCOBRA-IIIC computer code, documented in XN-NF-75-21(P)(A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation," to calculate departure from nucleate boiling performance, crossflow velocity, core pressure drop, and guide tube flow parameters. The staff position concerning XN-NF-75-21(P)(A) is as follows:

- The XCOBRA-IIIC code is valid for predicting PWR hydraulic conditions in conjunction with the departure from nucleate boiling (DNB) critical heat flux correlation.

- The use of XCOBRA-IIIC is limited to the "snapshot" mode when used for transients and is restricted from use in loss of coolant accidents (LOCAs) and other calculations with flow reversal and recirculation. This mode is based on a series of steady-state calculations for input over a series of time steps.

The licensee is using the HTP DNB correlation described in licensing topical report EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel." Although this is an apparent inconsistency with the NRC staff's position identified above, the HTP correlation was developed and approved after the staff approved

XCOBRA-IIIC, and the staff considered its use with the XCOBRA-IIIC code when approving the HTP correlation. It appears, therefore, that the use of the HTP correlation is appropriate for use with the XCOBRA-IIIC on that basis.

During a desk audit, the NRC staff reviewed certain accident and transient analyses, and did not identify any uses of the XCOBRA-IIIC code that were inconsistent with the second staff position identified above.

The HTP correlation, with a limit of 1.141, was generically approved by the NRC for use with the CE14HTP fuel design. On that basis, the NRC staff finds its use acceptable for thermal-hydraulic analysis at CCNPP.

The NRC staff's review of the transient analytical methods used to confirm the thermal-hydraulic acceptability of the CE14HTP fuel system is discussed in Chapter 5 of this SE. Additional topics considered by the staff pertinent to the fuel system thermal-hydraulic design include the licensee's evaluation of the mixed core (co-resident Westinghouse and AREVA fuel assemblies) and the effect of fuel rod bowing on the departure from nucleate boiling ratio (DNBR).

4.3 Mixed Core Evaluation

Section 4.0 of the Reload Transition Report (Attachment 4, Reference 1) details the thermal-hydraulics analysis supporting the CCNPP fuel transition. AREVA's portable Hydraulic Test Facility performed pressure drop testing to derive loss coefficients for the AREVA CE14HTP and Westinghouse TURBO fuel assembly components. Due to lower overall flow resistance, flow will be diverted from the TURBO fuel to the CE14HTP fuel. This promotes an increase in thermal performance and DNB margin for the CE14HTP fuel.

The strategy for addressing the presence of both TURBO and CE14HTP fuel assemblies relies on limiting the relative power (via fuel management) in the TURBO fuel bundles to compensate for reduced flow. Using currently approved methods, Westinghouse and AREVA independently assessed the impact of several mixed core configurations to confirm the adequacy of the power reduction.

In response to an RAI regarding the mixed core rod power limits[11], the licensee provided predicted radial peaking factors (F^T_r) for the fresh CE14HTP fuel and co-resident TURBO fuel at each reload depletion time step for CC2CY19. This comparison of integral rod power (F_r) confirms that the targeted **[[ ]]** power margin is maintained throughout Cycle 19 operation. In response to an RAI regarding the preservation of the **[[ ]]** power margin during off-nominal plant operations[12], the licensee stated this margin will be preserved under all conditions allowed by COLR power dependent insertion limits (PDILs) and at all power levels. In addition, the **[[ ]]** margin was verified for the limiting transient events that result in power redistribution, including main steam line break (MSLB), CEA withdrawal, and CEA drop events.

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[11] Reference 5 op. cit., RAI 2.c. ML103280082.
[12] Gellrich, G. H., Constellation Energy, letter to U.S. NRC, "Supplement to License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," Dockets 50-317 and 50-318, October 29, 2010. ML103080025. RAI 2.b.

Due to differences in critical heat flux (CHF) correlations, the two fuel types will preserve a different amount of DNB margin at identical local conditions. To assess the relative worth of the **[[ ]]** power reduction on predicted DNB thermal margin, the NRC staff asked the licensee to provide a calculation of overpower margin for the two CHF correlations. In response to RAI #2c[13], the licensee provided this comparison which demonstrates that the TURBO power reduction will have the desired result of ensuring that the TURBO fuel is less limiting.

Based on the information presented in Section 4.0 of the Reload Transition Report (Attachment 4, Reference 1) and in response to RAIs, the NRC staff finds the mixed core evaluation and the imposed power restriction on the co-resident TURBO fuel bundles acceptable.

4.4 Rod Bow

The CCNPP application to use AREVA fuels and methods represents an apparent first-time application of CE HTP fuel to a 24-month fuel cycle with peak rod exposure up to 62 GWd/MTU; thus, rod bow evaluations are of interest.

Fuel rod bow penalties in the DNBR analyses are assessed based on generic evaluations documented in XN-NF-75-32, Supplements 1-4, “Computational Procedure for Evaluating Fuel Rod Bowing.” This report discusses several evaluations that are performed to establish that existing methods are sufficiently conservative, or to determine appropriate penalties to apply to safety analysis initial conditions to assure sufficiently conservative results.

The evaluations are based on empirical data collected from several reactors and extrapolated to other bundle designs using engineering factors. The empirical database includes data taken from 18 fuel assemblies with burnups ranging from zero to 32,800 MWd/MTU, from the DC Cook and HB Robinson plants. Engineering factors were used to adjust for different bundle designs and variation in hot-to-cold and batch-to-batch conditions. AREVA evaluated the effect of the observed bowing on power peaking by using a physics code to study fuel rod arrays with displaced rods. AREVA demonstrated that peaking factor limits included sufficient conservatism to account for the perturbed peaking that resulted from the analyzed fuel rod displacement. AREVA also determined the effect of fuel rod bow on several transients.

The generic evaluations performed by AREVA demonstrated that fuel rod displacement in amounts below a threshold value did not cause significant effects on the safety analysis, and concluded that specific penalties were generally not necessary.

The NRC staff, in its SER approving XN-NF-75-32, Supplements 1-4, specifically noted that the evaluations described therein were not supported by any direct thermal-hydraulic testing.

During the audit, the NRC staff questioned the applicability of these data and analyses in light of CCNPP’s more modern operating strategies, including longer cycles and higher burnup. The licensee asserted that higher-burnup bundles would not be of concern, since they would likely be at the core periphery and would be sufficiently depleted as not to be DNBR-limiting bundles.

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[13] Reference 5 op. cit., RAI 2.c. ML103280082.

In light of the operational differences between current practice at CCNPP and general practice at the time XN-NF-75-32s1-4 was approved, the NRC staff requested additional confirmatory data to demonstrate that the generic rod bowing analysis remains applicable.

The licensee's response indicated, based on poolside measurements, including M5 clad fuel, taken at several different facilities, provided no indication of significant distortions in the channel size attributable to fuel rod bowing. The database included data that exceeded 60 GWd/MTU.

Although the staff noted above that its SE approving XN-NF-75-32s1-4 cited no support for the fuel rod bow treatment via thermal hydraulic testing, the licensee provided references to two journal articles presenting experimental testing of the effect of fuel rod bowing on critical heat flux in PWR fuel assemblies. The testing showed that DNB results were not adversely impacted for rod-to-rod gap closures up to 50-percent.

Based on the information provided by the licensee, the NRC staff concludes the following:

1. Fuel rod bowing was explicitly modeled in the fuel rod bow topical report discussed above.
2. Despite that this modeling was not supported by testing for CHF effects, the licensee provided reference to such testing, which was not included in the original topical report.
3. The CHF effects testing demonstrated that channel closures as much as 50 percent resulted in no adverse CHF effects, and recent data supports that there would not be such significant channel distortion in the fuel life.

Since the licensee provided information to demonstrate that the fuel rod bow would not become so significant during the fuel operating life as to cause adverse CHF effects, the NRC staff finds that the licensee has appropriately addressed the potential for adverse CHF effects due to fuel rod bow. The staff further finds that the implementation of XN-NF-75-32 is acceptable for implementation and listing in the COLR – References section of the CCNPP TS.

4.5 Conclusion

The NRC staff has reviewed the licensee's evaluation and proposed methods implementation related to the requested fuel transition. The staff concludes that the licensee has adequately accounted for the effects of the proposed fuel transition on the thermal hydraulic design and has provided reasonable assurance that the fuel design limits will not be exceeded during normal or anticipated transients. Based on this evaluation and in coordination with the reviews of the fuel system design, nuclear design, and transient and accident analyses, the staff concludes that the thermal hydraulic design of the fuel assemblies and reactor core will continue to meet the applicable requirements of GDC 10. Therefore, the staff finds the proposed fuel transition acceptable with respect to the thermal hydraulic design.

5.0 Safety Analysis

The NRC staff reviewed the licensee's safety analyses for two classes of transients: AOOs and accidents. For AOOs, the staff reviewed the licensee's analyses to ensure that reactor protective functions served to mitigate AOOs in such fashion that the postulated events are terminated and mitigated without exceeding fuel centerline melt, DNB, or RCS or main steam system pressure limits. The staff reviewed five postulated accidents – the MSLB, the locked reactor coolant pump rotor, CEA ejection, small break LOCA analysis, and the large break LOCA analysis to ensure that the analyses acceptably demonstrated that the predicted results met the acceptance criteria for each of the events. For the CEA ejection, steam line break, and seized rotor, the acceptance criteria found in SRP Chapter 15, that incorporate all applicable GDC, are related to gross fuel failure and peak fuel enthalpy. For the LOCA analyses, the acceptance criteria are contained in Title 10 to the *Code of Federal Regulations* (10 CFR) 50.46(b)(1) – (b)(3).

5.1 Anticipated Operational Occurrences

The licensee is implementing the method described by AREVA NP licensing topical report EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," to analyze AOOs. The NRC staff reviewed the methodology, the CCNPP-specific implementation of the methodology, and the modeling assumptions and results of specific postulated transients, as described in the following sections.

5.1.1 Methods Implementation

Topical Report EMF-2310(P)(A) pertains to non-LOCA accident and transient analyses that are part of the CCNPP licensing basis. Since CCNPP was designed and constructed to meet the intent of the draft GDC that were proposed by the Atomic Energy Commission (AEC) in 1967, the regulatory bases for these analyses are based on the draft GDC proposed by the AEC. The specific review guidance and acceptance criteria are discussed in Chapter 15 of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*. The specific acceptance criteria applicable to CCNPP are identified in Chapter 14 of the CCNPP Updated Final Safety Analysis Report (UFSAR).

5.1.1.1 Regulatory Evaluation

Evaluation models for LOCA events are defined in 10 CFR 50.46. This definition, which the NRC staff considers applicable to non-LOCA analyses, states that:

> An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

Section II of Appendix K to 10 CFR Part 50, also written for LOCA analyses, and considered by the NRC staff to be applicable to non-LOCA analyses, contains the documentation requirements for evaluation models. It states:

1.a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters of the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability – Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including:……for models covered by 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of 50.46(b) would not be exceeded.

Section III of Appendix B to 10 CFR 50 governs references to design control measures in the Core Operating Limits Report (COLR). It states, "Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests."

5.1.1.2 General Methodology

The system analysis is performed with S-RELAP5, as described in EMF-2100(P), "S-RELAP5 Models and Correlations Code Manual." The reactor vessel nodalization provides modeling of the key components in the reactor vessel using junctions, volumes, and heat structures. The secondary side includes the tube bundles, feedwater system, separators, steamlines, and turbine simulator.

A complete reactor point kinetics model simulates the production of nuclear power in the core. The model computes both the immediate fission power and the power generated from decay of fission fragments and actinides. The model provides capability to include feedback due to moderator density and fuel temperature changes.

Fuel modeling contributes to the determination of the power and heat flux for the core. The heat flux determines the core coolant heating rate and, ultimately, the temperature response of the RCS to power changes. The power is affected by changes in fuel temperature, that determines the Doppler feedback, and by the change in the core coolant temperature, that determines the moderator feedback. The fuel modeling is based on the RODEX2 code which is addressed in subsequent sections of this SE.

Fuel parameters that are considered are reflective of both beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions. BOC cases consider the maximum reactivity feedback and a low gap conductance, whereas EOC cases consider the minimum reactivity feedback and a high gap conductance.

The EMF-2310 method divides transients into two categories: fast and slow transients.

To account for the fact that the reactor trip system is not compensated at CCNPP, transient shifts are evaluated to ensure that DNBR transients include sufficient margin based on the reactor trip system performance.

Not all events are analyzed on a cycle-specific basis. A disposition of events is prepared, which categorizes each of the events into one of the following four categories:

- Event is reanalyzed
- Event is bounded by another event
- Event is bounded by a previous analysis, or
- Event is outside the licensing basis of the plant

5.1.1.3 DNBR Analysis

Detailed DNBR studies are performed on the hot rod using the XCOBRA-IIIC subchannel code. The code performs a quasi-steady state evaluation of the DNBR using statepoint parameters obtained from the S-RELAP5 code. Some of the S-RELAP5 parameters are biased for the DNBR evaluation in order to ensure that the DNBR result is limiting. The implementation of XCOBRA-IIIC is discussed in Chapter 4.2 of this SE.

5.1.1.4 Pressurization Transient Analysis

Reductions in secondary heat removal, those transients described in Chapter 15.2 of NUREG-0800, usually result in a pressurization of both the primary and secondary coolant systems. Typically, reactor trips are required to terminate these transients, and they present a limiting challenge to the integrity of the RCPB.

Acceptable transient analysis must demonstrate that applicable reactor protective features serve to terminate the event and mitigate its consequences without exceeding the RCPB safety limit. The transient analysis should account for the entire range of permissible, full-power operation and identify the limiting transient scenario, including the limiting initiating event and limiting initial conditions. This range is defined by the facility's TSs.

During its review, the NRC staff observed that the analytic assumptions in EMF-2310(P)(A), which the licensee would use to analyze these postulated events, may not directly lead to results consistent with regulatory requirements.

10 CFR 50.36 contains requirements for limiting safety system settings (LSSS), which are settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. 10 CFR 50.36 also requires the establishment of limiting conditions for operation (LCOs) of a nuclear reactor, which include, among other things, process variables, design features, and operating restrictions that are initial conditions of design basis accident or transient analyses that either assume the failure of, or present a challenge to the integrity of a fission product barrier.

Since EMF-2310(P)(A) has been approved by the NRC, the staff expects that acceptable use of the methodology would generate safety analysis results that validate facility-specific TSs with respect to the above regulatory requirements. This would mean that transient analyses are initiated at an appropriately conservative initial condition, including consideration of the full range of permissible plant operation with applicable uncertainties included.

CCNPP TS 2.1.2 limits RCS pressure to less than 2750 psia. TS Table 3.3.1-1, Item 4, "Pressurizer Pressure – High" specifies a limiting allowable value for protective system actuation due to an increase in reactor coolant system pressure. The allowable value is less than or equal to 2400 psia. In accordance with 10 CFR 50.36, the LSSS has been chosen on the basis that it corrects an abnormal situation – increasing RCS pressure – before the safety limit is exceeded. Additionally, TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," further constrains RCS pressure to a minimum value of 2200 psia. At CCNPP, therefore, the allowable steady-state operating range spans 2200 psia to 2400 psia, because below 2200 psia, the DNBR limits are invalid, and above 2400 psia, a reactor trip signal would be generated.

The abnormal situation referenced by 10 CFR 50.36 ties to Chapter 14.5 of the CCNPP UFSAR, which presents a limiting transient analysis for increasing RCS pressure. The current licensing basis specifically states that the assumed initial condition corresponds to the Tminimum indicated pressure, and includes an uncertainty on indicated pressurizer pressure.

A loss of load event results in an abrupt reduction in secondary heat removal, disabling the primary means of energy removal from the RCS. The selection of a low pressure allows the RCS to continue operating the longest amount of time prior to receipt of a high pressurizer pressure, or other, trip. The low-pressure initial condition, therefore, maximizes the energy delivery to the RCS and is hence appropriately selected. Based on this, the results of the safety analysis demonstrate that the LSSS was chosen to correct an abnormal situation before the safety limit is exceeded because the peak pressure is predicted to be less than 2750 psia.

As previously stated, CCNPP was designed and constructed to meet the intent of the draft GDC that were proposed by the AEC in 1967. Of those criteria, the following apply to pressurization transients, such as the loss of load event:

Criterion 9, Reactor Coolant Pressure Boundary – The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 33, Reactor Coolant Pressure Boundary Capability – The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy into the coolant.

Section 14.5 of the CCNPP UFSAR documents the plant's adherence to these criteria via a safety analysis that is initiated at a limiting condition for operation with appropriate allowance for uncertainty. Contrary to these requirements, and in a manner inconsistent with 10 CFR 50.36, however, the method described in Chapter 5 of EMF-2310 requires the selection of a different value for initial pressure that neither considers the full permissible range of operation nor instrumentation uncertainty.

The staff's SER approving EMF-2310(P)(A) states that the analyst is responsible for choosing correct input parameters that are consistent with facility licensing basis and TS requirements, and with NRC regulatory guidance. Chapter 5 of EMF-2310(P)(A) is, however, prescriptive regarding the selection of certain input parameters. In some cases, parameters taken at the initial condition prescribed by the approved methodology may not appropriately consider uncertainties, TS LSSSs, or TS LCOs.

In light of this concern, the NRC staff requested that the licensee provide plant analyses to demonstrate the effects of initiating pressurization transients at pressure-limiting initial conditions, including, for example, main steam system pressure, RCS pressure, and steam generator initial level. The licensee responded, stating that overpressure aspects of transients are not being analyzed in support of the requested fuel transition[14]. The supplemental information was non-responsive to the staff's regulatory concern, and as a result, the staff does not consider EMF-2310(P)(A), Revisions 0 or 1, to contain an NRC-accepted, generic basis for the use of S-RELAP5 to demonstrate RCPB integrity that is applicable to CCNPP.

The NRC staff has concluded that a license condition is necessary to capture the more restrictive design criteria for CCNPP reload designs. The licensee proposed a license condition restricting its implementation of EMF-2310(P)(A) as described in TS 5.6.5.b[15]. Section 7 of this SE lists the exact license condition for both CCNPP units 1 and 2.

<u>License Condition 3</u>:

Approval of the use of S-RELAP5 (Technical Specification 5.6.5.b.8) is restricted only to those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits. Prior transient specific NRC approval is required to analyze transient performance relative to reactor coolant pressure boundary integrity

[14] Reference 5 op. cit., RAI 4. ML103280082.
[15] Reference 6 op. cit., RAI 9. ML110040374.

> until NRC-approval is obtained for a generic or Calvert Cliffs-specific basis for the use of the methodology in Technical Specification 5.6.5.b.8 to demonstrate reactor coolant pressure boundary integrity.

The NRC staff finds that this license condition addresses the staff's regulatory concern with the use of the prescriptive EMF-2310(P)(A) methodology relative to consistency with 10 CFR 50.36 requirements, since it assures that the methodology will not be used to analyze pressurization transients at CCNPP without prior review and approval, or NRC-approved revision to EMF-2310.

5.1.1.5 Power Level Sensitive Transient Analysis

In accordance with the EMF-2310 method, events are analyzed either at hot zero power or hot full power conditions. Some of the postulated accidents and transients that are analyzed and described in the CCNPP UFSAR are sensitive to the initial power level. This is of concern, but may not be limited to, reactivity and power distribution anomalies.

While the current licensing basis and the proposed safety analysis methodology include prospects for analyzing these events at zero- and full-power conditions, the NRC staff did not locate documentation describing further analyses, data, or sensitivity studies to indicate that the consequences of these events, if initiated at a power level between zero- and full-power, would be less severe than the two power levels analyzed. Further, allowable operating ranges in the COLR LCOs often vary as a function of power level. The basis for these power-dependent state points must be grounded in safety analysis.

The NRC staff requested that the licensee provide information to demonstrate appropriate consideration of combinations of power-dependent variations in permissible state points and demonstrate that, given transients initiated at power-dependent limiting initial conditions, the results would remain within the applicable acceptance criteria.

In response, the licensee provided a detailed description of each aspect of the safety analysis intended to demonstrate the conservatism[16]. The description included conservatisms in the setpoint verification process, the span of analyzed power shapes, perturbations applied to the analyzed initial conditions, and consideration of trips not credited in the safety analysis,

The licensee stated that, in addition to the variable high power trip function, the Combustion Engineering (CE) plant is equipped with a rate of change of power-high function, which would preclude the RCS from reaching the power level-high trip credited for the hot full power analyses. In the process of generating or validating linear power density setpoints, the licensee stated that **[[**

]] This demonstrates that the licensee has included a large set of initial conditions **[[** **]]**, because it enables the licensee to conclude certain analyzed power shapes extend beyond the range permitted by core operating limits and instrumentation setpoints.

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16 Reference 5 op. cit., RAI 3. ML103280082.

The licensee also stated that **[[**

]] This information also demonstrates that, in addition to conservatively treating reactor plant and time-in-life phenomena[17], the licensee is considering a large range of initial conditions **[[**

]], because a wider range, along with more severe power shapes, are permitted at lower power levels than 100-percent.

The licensee discussed a setpoint verification process[18], which involves examining the post-trip transient progression to ensure that **[[**

]] appropriately accounts for the fact that the reactor trip system at CCNPP uses uncompensated functions, and could otherwise leave the plant susceptible to undershoots or overshoots in the parameter of interest.

The staff's concern is related to the fact that, at lower power levels, steady-state conditions include a set of achievable power distributions that can potentially be more challenging for two reasons (1) a wider array of axial and radial peaking factors is permissible, and (2) the core physics behavior at these lower power levels could generate more severe changes in peaking during the reactivity insertion portion of the transient.

As discussed above, the licensee provided information to demonstrate that the set of analyzed, full-power initial conditions sufficiently bounds the initial permissible core physics parameters for power levels above 60-percent. The licensee stated that **[[**

]]. Based on these considerations, the NRC staff concluded that Item (1), above, is addressed by the generic AREVA transient analysis methodology. This is because the peak heat generation is a product of the average linear heat generation rate and local peaking factor.

Concerning Item (2), however, the information provided by the licensee demonstrates that the analyses consider a large array of various initial conditions that span operating power level and time in core life. Based on the information provided by the licensee, however, it appears that the transient analyses inherently assume that there is no time-dependent change in the initial power shape, and that if there is, the associated margin degradation is bounded by the fact that a large number of other power shapes, including some shapes that aren't permissible under normal operating conditions, is also analyzed.

Based on its review of the current CCNPP licensing basis, which includes a proprietary CE-generated analysis of the CEA withdrawal at various power levels, the NRC staff does not have sufficient information to conclude that the wide array of analyzed initial conditions is sufficiently conservative to account for possible transient variations in the core power distribution that may lead to more limiting DNBR conditions at lower power levels than are presently analyzed.

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[17] Note that this discussion is limited to analyses performed for DNBR and fuel melt.
[18] Reference 12 op. cit., RAI 5. ML103080025.

The licensee addressed this concern by providing explicit analyses of part-power transients, which are discussed in other sections of this SE. And since the current licensing basis incorporates a generic disposition for the part-power CEA withdrawal event based on existing core operating limits, the licensee proposed a license condition to restrict itself from changing certain core operating limits without prior NRC review and approval.

The NRC staff has concluded that a license condition is necessary to capture the more restrictive design criteria for CCNPP reload designs. Section 7.0 of this SE lists the exact license condition for CCNPP units 1 and 2.

> License Condition 4:
>
> Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 shall not be changed without prior NRC review and approval until an NRC-accepted generic, or Calvert Cliffs-specific, basis is developed for analyzing the Control Element Assembly Rod Bank Withdrawal Event, the Control Element Assembly Drop, and the Control Element Assembly Ejection (power level-sensitive transients) at full power conditions only.

The NRC staff finds that this combination of additional information and implementation conditions provides an acceptable means to assure the NRC staff that the currently used analytic methodology provides acceptable results with respect to DNBR and fuel centerline melt. Additional discussion concerning License Condition 4 appears in Section 5.1.5 of this SE.

5.1.2 Increases in Secondary Heat Removal

5.1.2.1 Excess Load Event

The excess load event is described in the CCNPP UFSAR Section 14.4 as follows:

> An Excess Load event is defined as any rapid, uncontrolled increase in SG [steam generator] steam flow other than an SLB [steam line break]. The full opening of the turbine control valves, atmospheric dump valves, or turbine bypass valves during steady-state operation would result in an Excess Load event.

Classified as an AOO, this event is analyzed to the following acceptance criteria:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110% of design.

These acceptance criteria are consistent with SRP Chapter 15.1 and are, therefore, acceptable for application to CCNPP. As a cooldown event, the excess load does not challenge the peak pressure criteria.

As part of a desk audit, the NRC staff reviewed the AREVA S-RELAP5 and XCOBRA-IIIC calculations documenting the CCNPP excess load event (32-9117786-000, 32-9131007-000).

Both the AREVA analysis and UFSAR identified the most limiting hot full power (HFP) excess load scenario as the inadvertent opening of all turbine bypass valves and atmospheric dump valves. This scenario results in an increase in main steam flow of 45% rated design flow and corresponding increase in primary to secondary heat transfer.

The AREVA HFP excess load analysis included a parametric study varying moderator temperature coefficient (MTC) (up to -35 percent millirho (pcm)/°F) and turbine control modes. A comparison of the initial conditions and assumptions of the AREVA calculation to the CCNPP UFSAR Section 14.4 reveals differences. The AREVA calculation identified the limiting scenario as **[[**

]]. The UFSAR analysis evaluated a single case at 145% steam demand and the most negative MTC of -33.0 pcm/°F.

A comparison of the credited reactor protection system (RPS) and engineered safety feature actuation system (ESFAS) actuations of the AREVA calculation to the CCNPP UFSAR Section 14.4 reveals no significant differences. In both analyses, the variable high power trip (VHPT) function is credited to initiate a reactor trip at a ceiling of 110.33% rated thermal power (RTP) with a response time of 0.4 seconds. The AREVA analysis properly accounts for excore detector decalibration (i.e., temperature shadowing) and resistance temperature detector (RTD) thermal lag.

A comparison of the nuclear steam supply system (NSSS) response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.4 reveals reasonable agreement. Steam flow rapidly increases which promotes an increase in primary to secondary heat transfer which in turn decreases core inlet temperature. With an assumed negative MTC, core power increases past the VHPT setpoint. A reactor scram terminates the power excursion.

XCOBRA-IIIC was used to calculate the minimum DNBR for several of the limiting HFP excess load scenarios. In these cases, primary pressure and inlet temperature were biased to account for monitoring uncertainty and core flow rate was set to the TS minimum of 370,000 gpm. In all cases, the minimum DNBR remained greater than the SAFDL for both the transition core and a full core of CE14HTP fuel.

In response to an RAI regarding the axial power distribution used in the XCOBRA-IIIC DNBR calculations, the licensee provided a figure illustrating the limiting design axial power profile, -0.30 axial shape index (ASI), used in the DNBR calculations[19]. This conservative top peaked shape bounds the COLR operating range and, therefore, is acceptable.

The peak LHGR was calculated at **[[ ]]** kW/ft based upon the S-RELAP5 power excursion of **[[ ]]** with adjustments to account for uncertainties and an engineering factor. This peak LHGR remains below the safety limit of **[[ ]]** kW/ft. The NRC staff questioned the accuracy of the RODEX2 calculated fuel centerline melt safety limit of **[[ ]]** kW/ft due to the lack of a burnup dependent degraded fuel thermal conductivity model. (In Section 2.2.2.2, the licensee proposed License Condition 2 to resolve the staff's concern.)

[19] Reference 12 op. cit., RAI 20. ML103080025.

Both the AREVA analysis and UFSAR identified the most limiting hot zero power (HZP) excess load scenario as the inadvertent full opening of the turbine control valves. Cases initiated at Mode 2 and Mode 3 conditions were investigated to evaluate approach to DNB and FCM limits.

A comparison of the initial conditions and assumptions of the AREVA calculation to the CCNPP UFSAR Section 14.4 reveals some differences. Most notable are differences in initial power level and scram worth. Both analyses target the most negative MTC of -33.0 pcm/°F.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.4 reveals no significant differences. Both analyses credit the VHPT function to initiate a reactor trip at a floor of 40.0% RTP with a response time of 0.4 seconds. The AREVA analysis properly accounts for excore detector decalibration (i.e., temperature shadowing).

A comparison of the NSSS response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.4 reveals reasonable agreement. Steam flow rapidly increases which promotes an increase in primary to secondary heat transfer which in turn decreases core inlet temperature. With an assumed negative MTC, core power increases past the VHPT setpoint. A reactor scram, along with Doppler feedback, terminates the power excursion.

XCOBRA-IIIC was used to calculate the minimum DNBR for several time steps in the limiting HZP excess load scenarios. In these cases, primary pressure and inlet temperature were biased to account for monitoring uncertainty and core flow rate was set to the TS minimum value of 370,000 gpm. In all cases, the minimum DNBR remained greater than the SAFDL for both the transition core and a full core of CE14HTP fuel.

The S-RELAP5 hot spot model is used to calculate peak centerline temperature for short-duration power excursions such as HZP Excess Load. While the S-RELAP5 hot spot calculation uses fuel thermal conductivity that is based on RODEX2 without explicit correction for thermal conductivity degradation with increasing exposure, the conservative nature of the hot spot fuel centerline temperature response provides adequate means to offset the effect of thermal conductivity degradation. For example, the core power excursion is calculated based on maximum gap conductivity which minimizes Doppler reactivity feedback. This approach is conservative when applied with a minimum gap conductivity within the hot spot model which promotes an increase in calculated fuel centerline temperature. Furthermore, the hot spot calculation does not have any local reactivity feedback mechanisms. As a result, conservative local power peaking factors are applied without the benefit of the increased Doppler reactivity feedback (which would accompany the increased local fuel temperature). In addition, the hot spot thermal conductivity model is based on the thermal-conductivity for an 8 weight percent (w/o) Gd_2O_3 rod without taking credit for reduced power in an 8 w/o Gd_2O_3 rod relative to the highest powered UO_2 rod. Consequently, the resulting fuel centerline temperature from the hot spot model conservatively bounds Gd_2O_3 bearing rods as well as UO_2 rods.

Calculated hot spot centerline fuel temperatures remain below the corrected melting point. Melting temperature is decreased for both burnup and gadolinia content. Hence, CC2CY19 satisfies the fuel melting criterion.

In response to an RAI regarding cladding strain, the licensee provided the results of corrected RODEX2 calculations for the Excess Load event[20]. For both HFP and HZP scenarios, calculated strain remained below the criterion of 1.0%. See Section 3.1.2 for more discussion of cladding strain.

Neither HFP nor HZP excess load events predict a violation of the DNB, fuel centerline melt, or cladding strain SAFDLs. Therefore, a radiological assessment was not performed.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the CCNPP UFSAR, and responses to RAIs, the NRC staff finds the excess load analysis acceptable.

5.1.2.2 Excess Feedwater Heat Removal

The excess feedwater heat removal event is described in the CCNPP UFSAR Section 14.7 as follows:

> An Excess Feedwater Heat Removal event is defined as a reduction in SG feedwater temperature without a corresponding reduction in steam flow from the SGs. This could be caused by the loss of one or more of the feedwater heaters, or due to a feedwater controller malfunction at steady-state power that causes an increase in feedwater flow.

Classified as an AOO, this event is analyzed to the following acceptance criteria:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110% of design.

These acceptance criteria are consistent with SRP Chapter 15.1 and are, therefore acceptable for application to CCNPP. As a cooldown event, the excess feedwater heat removal does not challenge the peak pressure criteria.

As part of a desk audit, the NRC staff reviewed the AREVA S-RELAP5 calculation documenting the CCNPP excess feedwater heat removal event (32-9120860-000). Both the AREVA analysis and UFSAR identified the most limiting excess feedwater heat removal scenario as a loss of both high pressure feedwater heaters at HFP conditions. This event initiator results in a rapid decrease in main feedwater (MFW) temperature from 461°F to 361°F and a corresponding increase in primary to secondary heat transfer.

A comparison of the initial conditions and assumptions of the AREVA calculation to the CCNPP UFSAR Section 14.7 reveals insignificant differences. Both analyses target the most negative MTC of -33.0 pcm/°F and assume a decrease in feedwater enthalpy of approximately 105 British Thermal Units per pound-mass (BTU/lbm) resulting from the loss of both high pressure feedwater heaters.

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[20] Reference 9 op. cit., RAI 3. ML110180621.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.7 reveals no differences. No RPS or ESFAS actuations are credited for the excess feedwater heat removal event. The transient reaches a new quasi-steady state operating condition until plant operators manually trip the reactor at 1800 seconds. The AREVA analysis properly accounts for excore detector decalibration (i.e., temperature shadowing) and resistance temperature detector (RTD) thermal lag.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.7 reveals reasonable agreement. Feedwater temperature rapidly decreases which promotes an increase in primary to secondary heat transfer which in turn decreases core inlet temperature. With an assumed negative MTC, core power increases to a quasi-steady state value just below the VHPT setpoint. The excess feedwater heat removal event is relatively benign and does not challenge the DNB, fuel centerline melt, or cladding strain SAFDLs. Therefore, no radiological assessment is necessary.

Based upon its review of the supporting AREVA calculation and comparison to the current AOR, the staff finds the excess feedwater heat removal event acceptable.

5.1.3 Loss of Forced Coolant Flow

5.1.3.1 Single Reactor Coolant Pump Seized Rotor

The single reactor coolant pump (RCP) seized rotor event is described in the CCNPP UFSAR Section 14.16 as follows:

> Seized Rotor event is defined as a complete seizure (i.e., binding) of a single RCP shaft. The seizure is postulated to occur due to a mechanical failure or a loss of component cooling to the pump shaft seals. The most limiting Seized Rotor event is an instantaneous RCP shaft seizure at HFP. The reactor coolant flow through the core would be asymmetrically reduced to three pump flow as the result of a shaft seizure on one pump.

Classified as a Condition IV postulated accident, this event is analyzed to the following acceptance criteria:

1. If fuel failure is predicted, the radiological consequences must not exceed the limits defined in RG 1.183, Table 6.

The limits on radiological consequences (2.5 REM TEDE) are consistent with the Alternate Source Term (AST) guidance provided in RG 1.183 and, therefore, acceptable. Section 6.3.16 of the Reload Transition Report (Reference 1) states that the AST radiological consequences are based on the assumption that 5.0% of the fuel will experience cladding failure. No partial fuel melting is considered in the AST dose assessment.

As part of a desk audit, the NRC staff reviewed the AREVA S-RELAP5 and XCOBRA-IIIC calculations documenting the CCNPP single RCP seized rotor event (32-9124142-001, 32-9128093-000).

A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals differences. For example, the AREVA methods apply monitoring uncertainties on core inlet temperature and primary pressure in the DNB analysis (as opposed to within the NSSS simulation). Both analyses targeted the BOC for the most positive MTC.

In response to an RAI regarding a modeling assumption which stated that no significant flow asymmetry exists at the core inlet, the licensee stated that inlet flow asymmetries are modeled using a **[[ ]]** flow penalty applied on the hot channel and four face adjacent channels[21]. The response also stated that inlet flow asymmetries quickly become more uniform in a PWR open lattice core. As described in response to RAI #1, the UFSAR assumes an instantaneous degradation to 3-pump asymmetric core inlet flow distribution[22]. The staff was not satisfied that the **[[ ]]** flow penalty, which is applied to all DNB calculations as part of the approved AREVA methodology, may be sufficiently conservative to account for the degraded, asymmetric 3-pump flow distribution experienced during a seized rotor event.

In response to RAI #7, the staff requested that the event be re-analyzed with a XCOBRA-IIIC model which captures the 3-pump asymmetric core inlet flow distribution consistent with the current licensing basis[23]. As a result, the licensee completed a XCOBRA-IIIC sensitivity study which simulated different asymmetric core inlet flow distributions. The core inlet flow distribution depicted in Figure 2-1 of Reference 6 is more representative of the USFAR asymmetric flow distribution than simply reducing the inlet flow factor from **[[ ]]** in the limiting assembly and four face adjacent assemblies.

Based upon the XCOBRA-IIIC study, the licensee concludes that “the analysis positively demonstrates that the flow distributions are ‘washed out’ by the time of minimum DNBR” and therefore “AREVA’s modeling assumption on the flow distribution is validated.” The NRC staff does not accept this conclusion. Results from the sensitivity study in Table 2-1 of Reference 6 show significant differences between the AREVA methodology and the more realistic representation of the asymmetric 3-pump core inlet distribution. In addition, the sensitivity study calculated DNBR using a top skewed axial power distribution (AXPD). It is reasonable to estimate that DNBR differences between the two methods would be exacerbated at AXPDs with a higher ASI (e.g., cosine, bottom peaked). While DNBR is more limiting at top peaked AXPDs, AREVA’s method would not capture axial dependent fuel design features which may influence the location of minimum DNBR (e.g., mixing vanes, enrichment cutback zones).

Based upon the modified methodology with the core inlet flow distribution depicted in Figure 2-1 of Reference 6, the NRC staff’s concerns regarding Seized Rotor asymmetric core inlet flow distribution are resolved.

The NRC staff has concluded that a license condition is necessary to capture the modified Seized Rotor methodology criteria for CCNPP reload designs. License Condition 5 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. The following summarizes the license condition.

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[21] Reference 5 op. cit., RAI 7. ML103280082.
[22] Reference 12 op. cit. ML103080025.
[23] Reference 6 op. cit., RAI 2. ML110040374.

License Condition 5:

For the Seized Rotor Event analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet flow distribution.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.16 reveals no significant differences. Both analyses credit the Low Coolant Flow trip function with an analytical setpoint of 90% of initial flow and a response time of 0.5 seconds. The response times of the credited RPS and ESFAS functions were consistent with UFSAR Tables 7-2 and 7-4. One minor difference is that the AREVA analysis specifically models actuation of pressurizer sprays and power operated relief valves. Due to their timing, these actuations have no impact on thermal margin degradation and minimum DNBR.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.16 reveals reasonable agreement. However, the AREVA analysis credits a turbine trip coincident with reactor trip signal. In general, the timing of the turbine trip (if it is part of the licensing basis) is selected to maximize the consequences of the particular event. The staff had concerns that a coincident turbine trip would promote a decrease in primary to secondary heat transfer and an associated increase in primary pressure prior to both CEA insertion and minimum DNBR. Any additional primary pressure decrease due to the assumed coincident turbine trip would be non-conservative with respect to DNB degradation. As part of its response to RAI #7, the licensee stated that delaying the turbine trip was found to have a negligible effect on minimum DNBR[24].

Section 6.3.16 of the Reload Transition Report (Reference 1) states that the AST radiological consequences are based on the assumption that 5.0% of the fuel will experience cladding failure. No partial fuel melting is considered in the AST dose assessment. XCOBRA-IIIC was used to calculate the minimum DNBR at several points during the transient. In these cases, primary pressure and inlet temperature were biased to account for monitoring uncertainty. In all cases, the minimum DNBR remained greater than the SAFDL for both the transition core and a full core of CE14HTP fuel. Therefore, the radiological source term, based on 5.0% fuel failure due to DNB, remains conservative for CC2CY19.

An evaluation of DNB propagation was not used for this event. This is acceptable since no fuel rods are predicted to experience DNBR.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the CCNPP UFSAR, and responses to RAIs, the NRC staff finds the Seized Rotor analysis acceptable with the aforementioned license condition.

[24] Reference 5 op. cit., RAI 7. ML103280082

5.1.4 Asymmetric Events

5.1.4.1 Asymmetric Steam Generator

The asymmetric steam generator transient (ASGT) is described in the CCNPP UFSAR Section 14.12 as follows:

> Asymmetric SG events, which are the result of a malfunction of one SG, cause a nonuniform core inlet temperature distribution. The non-uniform core inlet temperature distribution in conjunction with the moderator temperature reactivity feedback produces asymmetric local power peaking in the core.

Classified as an AOO, this event is analyzed to the following acceptance criteria:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110% of design.

These acceptance criteria are consistent with SRP Chapter 15.4 and are, therefore, acceptable for application to CCNPP. The ASGT does not challenge the peak pressure criteria.

As part of a desk audit, the NRC staff reviewed the AREVA S-RELAP5 calculation documenting the CCNPP asymmetric steam generator transient (ASGT) (32-9124813-000). An ASGT is defined as any initiator that affects only one of the two SGs (e.g., single main steam isolation valve (MSIV) closure, single atmospheric dump valve (ADV) opening). Both the AREVA analysis and UFSAR identified the most limiting ASGT scenario as a single MSIV closure at HFP conditions.

A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals minor differences. For example, the AREVA methods apply monitoring uncertainties on **[[**
]]. Both analyses targeted an end-of-cycle as the most negative MTC of -33 pcm/°F.

In response to an RAI regarding the simulation of an asymmetric core inlet temperature distribution, the licensee stated that the S-RELAP5 model is based on a uniform, non-segmented core nodalization[25]. The licensee stated that this approach was acceptable since the reactor trips prior to any significant asymmetry at the core inlet and that a conservative moderator feedback model offsets any augmentation due to power redistribution effects. The NRC staff was concerned that the unique aspects of this asymmetric transient were not properly modeled and the basis of the ASGT trip function (i.e., setpoint and response time) was not being maintained.

In response to RAI #1b, the NRC staff requested that the event be re-analyzed with an S-RELAP5 model which captures the asymmetric core inlet temperature distribution consistent

[25] Reference 5 op. cit., RAI 1.b. ML103280082.

with the current licensing basis[26]. As a result, the licensee completed a re-analysis of the limiting ASGT scenario (i.e., single MSIV closure) using the pre-trip MSLB S-RELAP5 model. **[[**

]] Neutronic calculations were completed to develop local power peaking augmentation factors based upon the asymmetric core inlet flow distribution and resulting power tilt. Initial conditions and assumptions were conservatively selected to maximize the asymmetric core tilt and delay the ASGT trip. The re-analysis included two cases: one case with an instantaneous MSIV closure and a second case with a maximum MSIV closure time of 7 seconds.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.12 reveals one significant difference[27]. The AREVA analysis models the actuation (lifting) of a main steam safety valve (MSSV) on **[[ ]]**. This modeling approach is conservative since the early opening of a MSSV on **[[ ]]** delays the ASGT trip. Both analyses credit the ASGT SG ΔP trip function with an analytical setpoint of 186 psid and a response time of 0.9 seconds. The response times of the credited RPS and ESFAS functions were consistent with UFSAR Tables 7-2 and 7-4.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.16 reveals reasonable agreement. Based upon the modified methodology with the split core S-RELAP5 model, the NRC staff's concerns regarding ASGT asymmetric core inlet temperature distribution are resolved.

The NRC staff has concluded that a license condition is necessary to capture the modified ASGT methodology for CCNPP reload designs. License Condition 6 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. The following summarizes the license condition.

> <u>License Condition 6</u>:
>
> For the Asymmetric Steam Generator Transient analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet temperature distribution and application of local peaking augmentation factors.

XCOBRA-IIIC calculations including the asymmetric inlet boundary conditions and augmented peaking factors demonstrate that the minimum DNBR remained greater than the SAFDL for both the transition core and a full core of CE14HTP fuel. In addition, the peak LHGR remained below the fuel centerline melt condition.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the CCNPP UFSAR, and responses to RAIs, the NRC staff finds the ASGT analysis acceptable with the aforementioned license condition.

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[26] Reference 9 op. cit., RAI 1. ML110180621.
[27] Reference 9 op. cit., RAI 1. ML110180621.

5.1.4.2 Pre-Trip Main Steam Line Break

The pre-trip portion of the MSLB event is described in the CCNPP UFSAR Section 14.14 as follows:

> The SLB transient was analyzed in two distinct portions, referred to as pre-trip and post-trip. For the pre-trip portion of the event, the main concern is the power excursion seen due to the cooldown in combination with a negative MTC. A loss of power on reactor trip is also assumed. A parametric analysis of break size and MTC was performed to determine the limiting case with respect to the DNB SAFDL. Cases run for an inside Containment break credited high containment pressure and high power trips. Cases run for an outside Containment break credited high power and delta-T power trips.

Classified as a Condition IV postulated accident, this event is analyzed to the following acceptance criteria:

- If fuel failure is predicted, the radiological consequences must not exceed the limits defined in RG 1.183, Table 6.

The limits on radiological consequences (25 REM TEDE for fuel failure or pre-existing iodine spike (PIS), 2.5 REM TEDE generated iodine spike (GIS)) are consistent with the AST guidance provided in RG 1.183 and therefore acceptable. Section 6.3.14 of the Reload Transition Report (Reference 1) states that the AST radiological consequences are based on the assumption that 0.8% of the fuel will experience cladding failure. No partial fuel melting is considered in the AST dose assessment.

As part of a desk audit, the NRC staff reviewed the AREVA S-RELAP5 and XCOBRA-IIIC calculations documenting the CCNPP pre-trip main steam line break (MSLB) event (32-9115243-001, 32-9125451-000). Similar to the CCNPP UFSAR Section 14.14, the AREVA calculation includes a parametric analysis of break size, break location, and MTC to identify the limiting scenario. A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals differences. For example, the AREVA methods employ cycle-specific physics parameters (e.g., FTC, B_{eff}) as opposed to bounding values. In addition, monitoring uncertainties on core inlet temperature and primary pressure are applied in the DNB analysis. Both analysis included cases at the most negative MTC of -33.0 pcm/°F.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.14 reveals one significant difference. In addition to crediting the VHPT, thermal margin / low pressure (TM/LP), containment high pressure (CHP), and low SG pressure (LSGP) trip functions, the AREVA simulations credit the asymmetric SG ΔP (ASGPT) trip function. In response to an RAI regarding the environmental qualification of the ASGPT, the licensee reanalyzed the pre-trip MSLB scenarios without credit for this trip function[28]. For all cases where this trip function had initiated a reactor trip, the power excursion became slightly more severe. However, these cases eventually tripped on a different trip function and their resulting peak power remained less limiting than the previous bounding scenario.

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[28] Reference 12 op. cit., RAI 20. ML103080025.

The response times of the credited RPS and ESFAS functions were consistent with UFSAR Tables 7-2 and 7-4. In addition, the AREVA analysis properly accounted for excore detector decalibration (i.e., temperature shadowing) and resistance temperature detector (RTD) thermal lag.

The UFSAR identified the limiting scenario as a 1.0 ft^2 outside containment break which prompted a VHPT (ΔT) at 19.7 seconds and achieved a maximum core power of 123.8% RTP. The AREVA calculation identified the limiting DNBR scenario as a 1.5 ft^2 outside containment break with a -16 pcm/°F MTC which prompted a VHPT (NI) and achieved a maximum core power of 126.4% RTP. With respect to fuel centerline melt, AREVA identified the limiting scenario as a 2.0 ft^2 outside containment break with a -12 pcm/°F MTC which prompted a LSGP trip and achieved a maximum core power of 127.2% RTP.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.14 reveals reasonable agreement. However, the timing of the RCP coastdown shifted from concurrent with the reactor trip signal to concurrent with reactor scram. In response to an RAI regarding this departure from the current UFSAR, the licensee stated that the change was an error[29]. The limiting cases were re-run with the correct sequence of events and the impact was found to be negligible.

In response to an RAI regarding the asymmetric steam flow between the faulted and intact SG prior to MSIV closure, the licensee provided plots of steam line flow rate for various break sizes[30]. In a follow-up RAI, the licensee provided further information regarding their S-RELAP5 secondary model nodalization[31]. Based upon the information provided in response to these RAIs, the staff finds the MSLB model and its predictions acceptable.

XCOBRA-IIIC was used to calculate the minimum DNBR for several of the limiting pre-trip MSLB scenarios. In these cases, primary pressure and inlet temperature were biased to account for monitoring uncertainty and core flow rate was set to the TS minimum value of 370,000 gpm. In all cases, the minimum DNBR remained greater than the SAFDL for both the transition core and a full core of CE14HTP fuel.

The peak LHGR was calculated at **[[ ]]** kW/ft based upon the S-RELAP5 power excursion of **[[ ]]** with adjustments to account for uncertainties and an engineering factor. This peak LHGR remains below the safety limit of **[[ ]]** kW/ft. The NRC staff questioned the accuracy of the RODEX2 calculated fuel centerline melt safety limit of **[[ ]]** kW/ft due to the lack of a burnup dependent degraded fuel thermal conductivity model. (In Section 2.2.2.2, the licensee proposed License Condition 2 to resolve the staff's concern)

The CCNPP UFSAR currently states that the radiological assessment is based upon an outside containment break with an assumed 1.35% fuel failure. Section 6.3.14 of the Reload Transition Report (Reference 1) states that the AST radiological consequences are based on the assumption that 0.8% of the fuel will experience cladding failure. No partial fuel melting is considered in the AST dose assessment.

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[29] Reference 12 op. cit., RAI 7. ML103080025.
[30] Reference 5 op. cit., RAI 9. ML103280082.
[31] Reference 6 op. cit., RAI 19. ML110040374.

Sufficient thermal margin is preserved such that recent CCNPP reload cores do not predict fuel failure. The AREVA analyses also conclude that fuel failure is not predicted for the pre-trip MSLB event. As such, the radiological assessment was not revisited for this fuel transition. Therefore, the radiological source term, based on 0.8% fuel failure due to DNB, remains conservative for CC2CY19.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the CCNPP UFSAR, and responses to RAIs, the NRC staff finds the pre-trip MSLB analysis acceptable.

5.1.4.3 Post-Trip Main Steam Line Break

The post-trip portion of the MSLB event is described in the CCNPP UFSAR Section 14.14 as follows:

> For the post-trip portion of the event, the concern is a return to power in the vicinity of an assumed stuck rod. Full load and no load cases, with and without a loss of power, were analyzed to find the limiting cases with respect to the DNB and LHR [linear heat rate] SAFDL. A guillotine break of the main steam line was assumed, as this provides the largest cooldown of the RCS and the greatest potential for a post-trip return to power.

Classified as a Condition IV postulated accident, this event is analyzed to the following acceptance criteria:

1. If fuel failure is predicted, the radiological consequences must not exceed the limits defined in RG 1.183, Table 6.

The limits on radiological consequences (25 REM TEDE for fuel failure or PIS, 2.5 REM TEDE GIS) are consistent with the AST guidance provided in RG 1.183 and therefore acceptable. Section 6.3.14 of the Reload Transition Report (Reference 1) states that the AST radiological consequences are based on the assumption that 0.8% of the fuel will experience cladding failure. No partial fuel melting is considered in the AST dose assessment.

As part of a desk audit, the NRC staff reviewed the AREVA S-RELAP5 and XCOBRA-IIIC calculations documenting the CCNPP post-trip MSLB event (32-9118282-001, 32-9122215-000). Both the AREVA analyses and UFSAR evaluate HFP and HZP cases each with and without a loss of offsite power. Both analyses evaluate the largest break size (equivalent to the SG flow restrictor) located inside containment (requires harsh condition uncertainties). In addition, both analyses assume the largest worth CEA stuck out of the core, an active single failure of one train of high pressure safety injection (HPSI), and target the most negative MTC.

A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals minor differences. For example, the AREVA methods apply monitoring uncertainties on core inlet temperature and primary pressure in the **[[ ]]** (as opposed to within the NSSS simulation).

In response to an RAI regarding the impact of applying the monitoring uncertainties on the overall MSLB simulation, the licensee demonstrated that the influence of a small change in initial pressure or core inlet temperature on the transient simulation would be insignificant[32]. The staff accepts this position.

In response to an RAI regarding lower operating mode MSLB events, the licensee describes the various lower operating mode scenarios including the availability of RPS and ESFAS actuations[33]. To bound Mode 3 and below MSLB scenarios with reduced ECCS operational requirements, a Mode 2 event was analyzed assuming no HPSI flow to mitigate the event. The results of this MSLB analysis, with and without a loss of offsite power, demonstrate that the return to power is relatively small and the DNB and fuel centerline melt SAFDLs are not violated. Based upon the response to RAI #12, the NRC staff finds the lower Mode MSLB events acceptable.

In response to an RAI regarding differences in the moderator reactivity versus moderator density curve used in the S-RELAP5 cases and the current USFAR, the licensee stated that the UFSAR moderator reactivity curves are based on a bounding stuck rod worth[34]. The AREVA moderator reactivity curves are based on cycle-specific neutronics calculations. While both curves target the most negative MTC of -33 pcm/°F, cycle-to-cycle changes in stuck rod location and rod worth slightly affects the moderator reactivity curve. The licensee stated that changes will be evaluated for future reloads. Based upon the response to RAI #13, the NRC staff finds the moderator reactivity curve acceptable.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.14 reveals no significant difference. Both analyses credit a manual reactor trip at the start of the event, a Low SG Pressure isolation signal with an analytical setpoint of 600 psia and a MSIV closure time of 7.9 seconds, a SG ΔP AFW lockout signal with an analytical setpoint of 250 psid and a response time of 20 seconds, and a Low Pressurizer Pressure SIAS with an analytical setpoint of 1640 psia and a 30.9 second delay to full pump speed. The response times of the credited RPS and ESFAS functions were consistent with UFSAR Tables 7-2 and 7-4.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the CCNPP UFSAR Section 14.14 reveals reasonable agreement. The CCNPP UFSAR states:

> The limiting case among full load and no load, with and without LOOP, depends upon cycle-specific core physics parameters. In many instances, the limiting scenario for approach to the DNBR SAFDL is different from the limiting scenario for approach to the LHGR [linear heat generation rate] SAFDL.

The CCNPP UFSAR presents the results of the HFP MSLB with loss of offsite power event. For this case, the UFSAR reports a peak return to power of approximately 91 MWt (Unit 2) compared with a return to power of 68 MWt in the AREVA calculation.

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[32] Reference 12 op. cit., RAI 11. ML103080025.
[33] Reference 5 op. cit., RAI 12. ML103280082.
[34] Reference 12 op. cit., RAI 13. ML103080025.

XCOBRA-IIIC was used to calculate the minimum DNBR at the time of peak return to power for the post-trip MSLB scenarios. In all cases, the minimum DNBR remained greater than the SAFDL. Therefore, the radiological source term, based on 0.8% fuel failure due to DNB, remains conservative for CC2CY19.

The peak LHGR was calculated at **[[ ]]** kW/ft based upon the HFP MSLB with no loss of offsite power case and included cycle-specific local peaking factors (F_q) and an engineering factor. This peak LHGR remains below the safety limit of **[[ ]]** kW/ft. The NRC staff questioned the accuracy of the RODEX2 calculated fuel centerline melt safety limit of **[[ ]]** kW/ft due to the lack of a burnup dependent degraded fuel thermal conductivity model. (In Section 2.2.2.2, the licensee proposed License Condition 2 to resolve the staff's concern)

In recent reload cores, sufficient shutdown margin has been preserved such that fuel rod failure due to DNB or fuel centerline melt is avoided. The AREVA analyses also conclude that fuel failure is not predicted for the post-trip MSLB event. As such, the radiological assessment was not revisited for this fuel transition.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the CCNPP UFSAR, and responses to RAIs, the NRC staff finds the post-trip MSLB analysis acceptable.

5.1.5 Reactivity and Power Distribution Anomalies

The NRC staff reviewed the effects and consequences of postulated reactivity and power distribution anomalies to assure conformance to the applicable design criteria. The staff reviewed the description of the causes of AOOs, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the AOOs as compared with the acceptance criteria.

5.1.5.1 Regulatory Evaluation

The NRC's acceptance criteria are based on meeting the relevant requirements of the following GDC, or their appropriate equivalents in the context discussed in Appendix 1C of the CCNPP UFSAR:

1. GDC 10, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not to be exceeded during any condition of normal operation, including the effects of AOOs.

2. GDC 13, which requires that the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC17, which requires, in part, that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety.

4. GDC 20, which requires, in part, that the protection system shall be designed to initiate automatically the operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs.

5. GDC 25, which requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded for a single malfunction of the reactivity control systems, such as accidental withdrawal of control rods.

The licensee demonstrates conformance to the above acceptance criteria by analyzing each transient, crediting only safety-related mitigating systems and control-grade systems to the extent they exacerbate the consequences of the transient, to demonstrate the following:

- Fuel cladding integrity is ensured by confirming that the DNB ratio remains above its safety limit;

- Fuel centerline melt does not occur by confirming that the peak linear heat rate remains below the fuel centerline melt limit; and

- The reactor coolant and main steam system pressure boundaries remain intact by confirming that their respective pressures remain less than 110-percent of their design values.

The acceptance criteria are consistent with SRP Chapter 15.4.

5.1.5.2 CEA Withdrawal at Zero-Power Conditions

The NRC staff reviewed the licensee's analysis of the CEA withdrawal from subcritical or low-power conditions as described in audit documents 32-9123010-000 and 32-9133635-000. The licensee analyzed this event using the EMF-2310 method. The system analysis was performed using the NRC-approved S-RELAP5 code. A detailed DNBR calculation was performed for the time frame surrounding the time of peak heat flux using the NRC-approved XCOBRA-IIIC code.

The licensee stated that this transient is self-limited by the reactivity feedback effect of the negative Doppler coefficient. The limiting time-in-cycle for this transient is BOC, because the least negative value of the Doppler coefficient occurs at the BOC.

The system response analysis credits the VHPT function; however the high rate of change of power trip can also serve to terminate this event.

The event is assumed to occur at HZP conditions because the hot initial condition provides the least DNB margin when compared to a cool, shutdown condition.

The system performance for this event is analyzed with S-RELAP5 using a combination of the **[[**

]].

Initial conditions included [[

]].

For this transient, the use of the [[ ]] conditions discussed above is acceptable because the transient is terminated by the VHPT function. Therefore, the trip timing is a function of power escalation, and since the lowest achievable initial power level is assumed, and the trip setpoint is assumed to be at its TS-required setpoint with uncertainty, the transient termination timing is conservatively modeled, and this provides for the greatest ascension in power. The remaining initial conditions are acceptable because they will also tend to drive the greatest ascension in power prior to the reactor trip, which would in turn maximize the clad surface heat flux and fuel temperature transients.

The DNBR evaluation is performed using boundary conditions from the S-RELAP5 analysis described above at the time of peak heat flux. The boundary conditions are biased [[

]]. The biases are appropriate to DNBR, because they conservatively account for uncertainty and variability in the conditions applied to the S-RELAP5 analysis.

[[

]]

The NRC staff reviewed the licensee's analyses of the uncontrolled CEA withdrawal from subcritical or low power conditions. The staff reviewed the initial conditions and the reactor parameters used in the analysis, and found them acceptable as described above. The staff notes that the licensee is using the NRC-approved analytical methods and computer codes described in EMF-2310 to analyze the event. The staff confirmed that the consequences of the AOO meet the applicable acceptance criteria for fuel cladding integrity, fuel centerline melt, and RCS pressure. Based on these considerations, the staff concludes that the licensee has demonstrated acceptable reactor system performance for CCNPP for the requested fuel transition with respect to the uncontrolled CEA withdrawal from a subcritical or low-power startup condition.

5.1.5.3 CEA Withdrawal at Power

The NRC staff reviewed the licensee's analysis of the CEA withdrawal at power as described in audit documents 32-9134290-000 and 32-9125101-000. The licensee analyzed this event using the EMF-2310 method. The system analysis was performed using the NRC-approved S-RELAP5 code. A detailed DNBR calculation was performed for the time frame surrounding the time of peak heat flux using the NRC-approved XCOBRA-IIIC code.

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The DNBR evaluation is performed using boundary conditions from the S-RELAP5 analysis described above at the time of peak heat flux. The boundary conditions are biased **[[**

]].

The licensee stated that, because the VHPT setpoint is automatically reset to track the current operating power level, the range of initial reactor power levels is bounded by analyzing only full-power cases[36]. The range of possible reactivity insertion rates and core peaking is not, however, bounded by full-power conditions and therefore the NRC staff questioned the basis of this assertion. The assertion is evaluated on a generic basis with respect to the EMF-2310 method in Section 5.1.1.5 of this SE; the results of sensitivity studies performed explicitly for this transient are evaluated herewith.

The licensee provided sensitivity studies, based on a reference plant, that considered the CEA withdrawal at 25-, 50-, 75-, and 100-percent power[37]. This range of intermediate power levels includes the broader spectrum of reactivity insertion rates.

The reference plant had, in most cases, a more flexible analyzed operating envelope than CCNPP, making the analysis mostly conservative relative to the permissible operating conditions at CCNPP. The reference plant had a slightly lower variable high power trip setpoint uncertainty and slightly higher core flow; however, the NRC staff believes that the wider range of power distributions analyzed for the reference plant compensates these slightly non-

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[35] Proprietary
[36] Reference 5 op. cit., RAI 3. ML103280082.
[37] Reference 9 op. cit., RAI 9. ML110180621.

conservative initial conditions. Another noteworthy difference is the fact that CCNPP has a higher floor on the VHPT than the reference plant. At CCNPP, the VHPT floor, or its lowest setpoint, is 40-percent. The licensee accounted for this design difference by repeating the affected analyses using conditions more reflective of the CCNPP design.

The results of the sensitivity studies showed similar results at all power levels. There was a slight indication that the lower power cases ascended in power level slightly higher prior to completion of the reactor trip, as compared to the full-power result, and the 25-percent power case had a longer power ascension, due apparently to the higher floor of the VHPT. At the other power levels, the VHPT is approximately ten-percent higher than the assumed power level. All transients exhibited similar behavior, except for the 25-percent power case, for which the RCS flow and core inlet temperature transients were slightly more benign than in the other cases. The DNBR and peak linear heat rates were bounded by the full-power case.

The full-power MDNBR results for CCNPP were **[[ ]]** well above the HTP DNBR correlation limit, and hence acceptable.

Because the CCNPP operating envelope is more restrictive than that of the reference plant, the NRC staff finds that the part-power sensitivity results are applicable to CCNPP with its current core operating limits. The staff's finding is, however, based on the fact that the reference plant analysis is conservative when applied to CCNPP, because the CCNPP operating limits are more restrictive than those for the reference plant. The results do not, therefore, provide a CCNPP-specific basis to conclude that changes in the core operating limits are acceptable with respect to the part-power transients. The staff does not find EMF-2310 acceptable for implementation as a methodology to generate core operating limits for this and other power level sensitive transients. This is consistent with License Condition 4, discussed in Section 5.1.1.5 of this SE.

5.1.5.4 CEA Drop

The NRC staff reviewed the licensee's analysis of the CEA drop as described in the licensee's calculations 32-9116892-001 and 32-9131108-000. The licensee analyzed this event using the EMF-2310 method. The system analysis was performed using the NRC-approved S-RELAP5 code. This event presents a limiting challenge to the fuel melt acceptance criterion, so a detailed evaluation of fuel centerline melt was performed to demonstrate that this event does not impose the risk of fuel centerline melt.

The CEA drop event is defined as the inadvertent release of a CEA, causing it to drop into the core. A dropped CEA could be detected by either a position limit switch on each CEDM or by a reduction in power as sensed by the ex-core detectors.

The negative reactivity insertion when the CEA drops into the core causes a reduction in the core power and reactor coolant temperatures. Increased cladding heat fluxes and fuel temperatures in the hot assembly result in a challenge to the DNB and fuel centerline melt SAFDLs. For a given dropped CEA worth, there is a tradeoff between the resultant return-to-power and the radial peaking augmentation factor. Therefore, a range of dropped CEA worths, from a minimum of 10 pcm to a maximum of 200 pcm, is analyzed. This event is characterized by a return-to-steady-state, and is not usually terminated by a reactor trip.

The calculations are performed using the S-RELAP5 code at EOC HFP conditions, maximum TS core inlet temperature, and minimum TS RCS flow rate, which results in the minimum margin to the DNB limit. The event is analyzed with the most negative HFP MTC, which results in the most positive moderator reactivity feedback as the RCS cools down due to the dropped CEA.

The fuel centerline melt evaluation is performed using a simple calculation of the peak achievable linear heat rate based on peaking factor limits and an augmentation factor to account for the dropped rod reactivity insertion and power redistribution. The peak linear heat rate is compared to a limit to demonstrate that the fuel does not melt. The calculations showed that the linear heat rate did not exceed the fuel melt limit.

Given the NRC staff's concerns identified in Section 5.1.1.5, the staff requested that the licensee analyze the hot full-power sensitivities to dropped rod worth, initial and final power level, and limiting power distributions and redistributions at lower power levels, to confirm that the sensitivity study results remain bounded by the return-to-power and radial peaking augmentation factors obtained from full-power neutronic analyses.

The licensee responded, providing a power-level-dependent study of the dropped rod augmentation factor, along with information comparing allowable peaking factors as a function of reactor power, compared to the power-level-dependent local peaking factor required to cause fuel centerline melting[38].

The licensee re-evaluated the dropped rod augmentation factors at 20- and 80-percent power. The evaluation showed that the augmentation factor could increase as much as 10 percent at lower power levels.

The licensee also characterized the fuel centerline melt limit as a function of power level by illustrating the total peaking required to achieve fuel centerline melt at any given power level. This plot was accompanied by a trace of the permissible total peaking factors based on the power level and the PDIL.

Considering this information together, it was possible to observe that the margin between achievable peaking factors and the peaking factor required for fuel melt increased as power decreased, and that the broader margins at lower power levels more than compensated for the increased peaking augmentation. Based on this information, the NRC staff accepts the licensee's disposition regarding the power level sensitivity, and agrees that the limiting dropped CEA transient is that which is initiated from HFP conditions.

The licensee demonstrated that the selected initial conditions were appropriate, the event was analyzed using NRC-approved methods, and the results were less than the peak linear heat rate SAFDL. Therefore, the NRC staff finds the requested fuel transition acceptable with respect to the dropped CEA event. This finding is based on an analyzed set of power-

[38] Reference 9 op. cit., RAI 4. ML110180621.

dependent peaking factor operating limits, and is subject to License Condition 4 as discussed in Section 5.1.1.5 of this SER.

5.2 CEA Ejection

The control rod ejection (CRE) event is described in the CCNPP UFSAR Section 14.13 as follows:

> A CEA Ejection Event is defined as a rapid, uncontrolled, total withdrawal of a single or dual CEA. A dual CEA is two CEAs connected to a single CEA extension shaft. The event is postulated to occur as a result of a complete instantaneous circumferential rupture of either the CEDM pressure housing or the CEDM nozzle from the reactor vessel closure head. The pressure of the RCS causes the ejection of the extension shaft through the rupture and the movement of the CEA to a fully-withdrawn position. The most limiting CEA Ejection Event is a rapid total withdrawal of the highest worth CEA.

Classified as a Condition IV postulated accident, this event is analyzed to the following acceptance criteria:

1. The radial average fuel pellet enthalpy at the hot spot must be < 280 cal/g.

2. Maximum RCS pressure must remain below the ASME emergency condition stress limit of 120% of design pressure.

3. If fuel failure is predicted, the radiological consequences must not exceed the limits defined in RG 1.183, Table 6.

These acceptance criteria were evaluated by the NRC staff as discussed below.

The first criterion, 280 cal/g, relates to maintaining core geometry amenable to cooling and limiting fuel rod fragmentation and fuel dispersion in accordance with GDC 27. Dispersed fuel particles will interact with the coolant (referred to as fuel-to-coolant interaction, FCI), quickly dissipate heat, and may generate large volumes of steam. Molten FCI is especially volatile. The rapid pressure increase resulting from the FCI generated steam may challenge the integrity of the reactor pressure boundary. The 280 cal/g criterion is consistent with the CCNPP UFSAR. However, a re-examination of the reactivity initiated accident (RIA) empirical database (e.g., SPERT, TREAT, and PBF tests) reveals that test rods exposed to a radial average fuel enthalpy above 220-230 cal/g exhibited significant fragmentation (and a loss of coolable geometry). Recognizing that this older empirical database was comprised of low burnup fuel rod specimens, a further reduction in the coolability criterion is necessary to account for the effects of higher fuel burnup (e.g., high burnup structure, edge peaking, gaseous swelling). To preserve a coolable geometry and avoid molten FCI, the modified criterion for peak radial average fuel enthalpy is 200 cal/g.

For the purpose of calculating a conservative source term for radiological assessment, the S-RELAP5 hot spot model is used to predict fuel centerline temperature. This information is used to assess the number of fuel rods experiencing fuel centerline melting. Section 6.3.13 of the Reload Transition Report (Reference 1) states that the AST radiological consequences are based on the assumption that 8% of the fuel will reach incipient centerline melt and 2% will experience cladding failure (for a combined 10% fuel failure). In this report the licensee states the following:

> Therefore, further radiological analyses will not be performed provided that the thermal hydraulic analysis demonstrates that the combined percentage of fuel experiencing centerline melt and fuel experiencing cladding damage remains below 10% for the CEA Ejection event.

This statement implies that that the percentage of fuel from each failure mechanism may be traded off provided that the total remains below 10%. Based upon the guidance in RG 1.183 Appendix H item #1, the NRC staff does not agree with the logic of trading off the percentage of fuel failure since a fuel rod with partially molten fuel will emit a significantly larger source term than a fuel rod which fails due to DNB. For example, the existing radiological source term would remain conservative for a combined fuel failure of 9.5% DNB and 0.5% centerline melt. However, the existing radiological source term would be non-conservative for a combined fuel failure of 0.5% DNB and 9.5% centerline melt.

The NRC staff has concluded that a license condition is necessary to capture the more restrictive CEA ejection design criteria for CCNPP reload designs. License Condition 7 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. The following summarizes the license condition.

License Condition 7:

a. Predicted peak radial average fuel enthalpy when calculated in accordance with the methodology of Technical Specification 5.6.5.b.11 shall remain below 200 cal/g.

b. For the purpose of evaluating radiological consequences, should the SRELAP-5 hot spot model predict fuel temperature above incipient centerline melt conditions when calculated in accordance with the methodology of Technical Specification 5.6.5.b.8, a conservative radiological source term (in accordance with RG 1.183, Revision 0) shall be applied to the portion of fuel beyond incipient melt conditions (and combined with existing gap source term), and cladding failure shall be presumed.

The limits on radiological consequences (6.3 REM TEDE) are consistent with the AST guidance provided in RG 1.183 and the regulations found in 10 CFR 50.67 and are, therefore, acceptable.

The limit on peak RCS pressure is consistent with SRP Chapter 15.4.8 and GDC 14 and are, therefore, acceptable. No new peak pressure case was performed for the CCNPP fuel transition. The licensee will maintain the existing UFSAR RCS pressure evaluation.

As part of a desk audit, the NRC staff reviewed the AREVA core physics calculation, S-RELAP5 calculation, and XCOBRA-IIIC calculation documenting the CCNPP control rod ejection (CRE) event (32-9111033-001, 32-9122310-001, 32-9124269-000). The CRE analysis is broken up into three components: (1) fuel enthalpy calculation, (2) S-RELAP5 transient simulation, and (3) XCOBRA-III DNB calculation and fuel centerline melt calculation. Each component is discussed below.

Unlike the AREVA calculation, which is based on cycle-specific physics parameters, the CCNPP UFSAR CRE is based on bounding core physics parameters. A comparison of these bounding physics parameters (from UFSAR Table 14.13-1) to the CC2CY19 values is provided in Table 4 below.

Table 4 Comparison of Key Physics Parameters

Parameter	CC2CY19		UFSAR
	BOC	EOC	BOC/EOC
Ejected Rod Worth (%Δρ)			
- HFP	[[ ]]	[[ ]]	0.310
- HZP	[[ ]]	[[ ]]	0.870
Post-Ejected Fq			
- HFP	[[ ]]	[[ ]]	5.04
- HZP	[[ ]]	[[ ]]	24.7
Doppler Reactivity (%Δρ/° F)	[[ ]]	[[ ]]	-0.00060
Delayed Neutron Fraction (βeff)	[[ ]]	[[ ]]	0.00440

Due to the burnup dependence of delayed neutron fraction (smaller with exposure, prompts higher ejected worth) and Doppler coefficient (more negative with exposure, prompts more negative reactivity feedback), the NRC staff questions the selection of separate BOC and EOC cases. Note that the UFSAR combined worst-case physics parameters, regardless of burnup (e.g., BOC Doppler with EOC β_{eff}). In response to an RAI regarding the use of burnup dependent physics parameters, the licensee stated that CRE scenarios at intermediate burnup were bounded by the BOC and EOC cases[39]. Further justification was provided in response to RAI #26 where the licensee presented plots of ejected rod worth, delayed neutron fraction, and Doppler reactivity as a function of burnup[40]. In addition, sensitivity cases were run to investigate whether the conservatism in the analysis is adequate to cover the variability in key physics parameters. Based upon the response to RAI #26, the staff finds the evaluation of BOC and EOC acceptable to cover intermediate burnup points.

In response to an RAI regarding the significant differences in calculated physics parameters between AREVA and the UFSAR, the licensee stated that calculations are performed every reload.[41] Further justification was provided in response to RAI #24a where the licensee presented a comparison of AREVA calculated CC2CY19 physics parameters to Westinghouse

[39] Reference 5 op. cit., RAI 14. ML103280082.
[40] Reference 9 op. cit., RAI 26. ML110180621.
[41] Reference 12 op. cit., RAI 18. ML103080025.

calculated CC1CY15 and CC1CY16 values[42]. One key difference is that the AREVA methods initiate the event from the COLR power dependent insertion limits (PDIL) whereas the Westinghouse methods assume full rod insertion at the start of the event. Unlike the UFSAR bounding physics parameters, the Westinghouse cycle-specific values exhibit reasonable agreement with the AREVA cycle-specific values. Based upon the response to RAI #24a, the NRC staff's concerns regarding differences in key physics parameters are resolved.

AREVA's approved CRE methodology is based on a parametric study of ejected rod worth, post-ejected power peaking, Doppler reactivity coefficient, and delayed neutron fraction performed with the computer code XTRAN, described in licensing topical report XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors." The XTRAN computer code is a two dimensional, space and time dependent neutron diffusion model. A set of curves based on these parametric studies was developed and is used to calculate peak fuel enthalpy (radial average, cal/g). In response to an RAI regarding the applicability of the generic XTRAN curves to CCNPP, the licensee provided a comparison of the UFSAR reported CRE fuel enthalpy calculated with the STRIKIN-II code against values calculated with the generic XTRAN curves[43]. To further study this issue, the NRC staff prepared a similar comparison using information reported in the Waterford Unit 3 (WSES-3) and Arkansas Nuclear One Unit 2 (ANO-2) UFSARs. Table 5 contains the results of the staff's review along with the calculated values for CC2CY19. Based upon this comparison, the staff's concerns regarding the applicability of the generic XTRAN curves to CCNPP are resolved.

In response to an RAI regarding the basis for power dependent limiting conditions of operation (LCOs) such as the PDIL, the licensee stated that the CRE analysis is performed from the PDIL and that, in general, deeper control rod insertion promotes higher ejected rod worth and power peaking[44]. Recognizing the 20% power breakpoint in the CCNPP PDIL, the NRC staff requested further evidence that the HZP case bounded a scenario initiated at intermediate power levels. In response to RAI #8, the licensee calculated ejected rod worth at the 20% power PDIL and discovered that the HZP case was not bounding[45]. In addition to a potentially higher ejected rod worth, the 20% power case has a higher initial fuel enthalpy. The licensee agreed to modify their methodology whereby cycle-specific HZP physics parameters would be calculated based on a modified CCNPP PDIL with an artificially extended Bank 3 full insertion (in analytical space, not actual plant operations), as depicted in Figure 8-2 of the RAI response. Key physics parameters were calculated for CC2CY19 based upon the extended PDIL. These results demonstrate that the HZP case with the modified methodology bounds the 20% power case. Based upon the modified methodology with the extended PDIL, the staff's concerns regarding CRE at intermediate power levels are resolved.

The NRC staff has concluded that a license condition is necessary to capture the modified CRE methodology for CCNPP reload designs. License Condition 8 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. The following summarizes the license condition.

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[42] Reference 9 op. cit., RAI 24a. ML110180621.
[43] Reference 9 op. cit., RAI 24b. ML110180621.
[44] Reference 5 op. cit., RAI 3. ML103280082.
[45] Reference 6 op. cit., RAI 8. ML110040374.

License Condition 8:

For the Control Element Assembly Ejection analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.11, the cycle-specific hot zero power peak average radial fuel enthalpy is calculated based on a modified power dependent insertion limit with Control Element Assembly Bank 3 assumed to be fully inserted (only in the analysis, not in actual plant operations).

As described above, the modified acceptance criterion to ensure a coolable geometry is that calculated peak radial average fuel enthalpy remain below 200 cal/g. For CC2CY19 the peak calculated radial average fuel enthalpy was 155 cal/g. Based on a peak radial average fuel enthalpy less than 200 cal/g, the staff finds the CRE analysis acceptable.

A comparison of the initial conditions and assumptions of the AREVA S-RELAP5 CRE calculation to the UFSAR reveals some important differences. As a reactivity driven event, differences in key physics parameters (e.g., ejected worth, Doppler, β_{eff}) have a first order effect. These differences are discussed above.

In response to RAI #15 regarding the ejection time, the licensee performed a sensitivity study to quantify the impact of a 0.05 second ejection time relative to 0.10 second ejection time[46]. The resulting change on peak centerline temperature was insignificant. Furthermore, the AREVA calculation artificially increases ejected rod worth until Doppler reactivity feedback is responsible for turning the event around. Based upon the response to RAI #15, the staff's concern regarding the change in ejection time is resolved.

In response to RAI#16 regarding the use of an N-1 scram reactivity (RAI #16, Reference 4), the licensee stated that control rod scram reactivity is not a factor in the calculation of deposited enthalpy and that Doppler reactivity is responsible for arresting the power excursion[47]. Based upon the response to RAI #16, the staff's concerns regarding the change in scram reactivity is resolved.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the CCNPP UFSAR Section 14.13 reveals no significant differences. Both analyses credit the VHPT function with an analytical setpoint of 110.33% and 40% for the HFP and HZP cases respectively and a response time of 0.4 seconds. The response times of the credited RPS and ESFAS functions were consistent with UFSAR Tables 7-2 and 7-4.

Calvert Cliffs RPS design criteria (UFSAR Chapter 1) dictate that the RPS be capable of performing its function is the event of a single failure. In addition, CCNPP TSs allow a single excore safety channel to be inoperable. The CRE event exhibits a rapid, localized power excursion (especially in heavily rodded HZP conditions). The NRC staff had concerns that the timing and response of an excore detector driven VHPT would be influenced by the core position of the ejected rod and core locations of inserted rods relative to each operable excore safety channel. In response to RAI #17 regarding excore detector availability and response, the

[46] Reference 12 op. cit., RAI 15. ML103080025.
[47] Reference 5 op. cit., RAI 16. ML103280082.

licensee re-iterated that the CRE power excursion is arrested by Doppler reactivity feedback prior to reactor trip and that any delay in RPS response due to inoperable excore safety channels or harsh environment conditions would not significantly affect the predicted results[48]. This is true for the AREVA methodology where the ejected rod worth is increased to ensure that Doppler reactivity drives the event. Based upon the response to RAI #17, the staff's concerns regarding the excore detector response is resolved.

The NSSS response and sequence of events of the AREVA calculation seem reasonable. A direct comparison to the CCNPP UFSAR is not possible since it does not provide a sequence of events table and only provides a plot of core power versus time.

The SRELAP-5 hot spot model is used to calculate peak centerline temperature for short-duration power excursions such as CRE. While the S-RELAP5 hot spot calculation uses fuel thermal conductivity that is based on RODEX2 without explicit correction for thermal conductivity degradation with increasing exposure, the conservative nature of the hot spot fuel centerline temperature response provides adequate means to offset the effect of thermal conductivity degradation. For example, the core power excursion is calculated based on maximum gap conductivity, which minimizes Doppler reactivity feedback. This approach is conservative when applied with a minimum gap conductivity within the hot spot model which promotes an increase in calculated fuel centerline temperature. Furthermore, the hot spot calculation does not have any local reactivity feedback mechanisms. As a result, conservative local power peaking factors are applied without the benefit of the increased Doppler reactivity feedback (which would accompany the increased local fuel temperature).

Calculated hot spot centerline fuel temperatures remain below the fuel melting point (corrected for burnup and gadolinia content). Therefore, the radiological source term, based on 8% of the fuel reaching centerline melt, remains conservative for CC2CY19.

XCOBRA-IIIC was used to calculate the minimum DNBR for several time steps for the HZP-BOC, HZP-EOC, HFP-BOC, and HFP-EOC CRE scenarios. The RCS pressure used in these DNBR calculations is held constant (i.e., initial core exit pressure used). For the HFP cases, a design axial power shape was assumed that bounds the LCO and LSSS ASI limits[49]. In all cases, the minimum DNBR remained greater than the SAFDL for both the transition core and a full core of CE14HTP fuel. Therefore, the radiological source term, based on 2% fuel failure due to DNB, remains conservative for CC2CY19.

The radiological assessment requires a conservative estimate of the number of fuel rod failures predicted during the CRE event. As described above, fuel rod failures are estimated based upon conservative DNB and centerline fuel melt predictions. SRP Section 4.2 Appendix B (Reference 7) provides interim criteria and guidance for the CRE accident. With respect to cladding failure, the interim criteria list the failure threshold as (1) a total radial average fuel enthalpy greater than 150 cal/g (assuming rod internal pressure above system pressure) for HZP cases, (2) departure from nucleate boiling for at-power cases, and (3) a change in radial average fuel enthalpy greater than the PCMI failure threshold in Figure B-1 of SRP-4.2, Appendix B, for all power levels. Based upon the corrosion properties (i.e., absorbed hydrogen)

[48] Reference 12 op. cit. RAI 17. ML103080025.
[49] Reference 12 op. cit. RAI 19. ML103080025.

for M5 alloy cladding, the PCMI failure threshold should remain above 125 Δcal/g. For CC2CY19, the HZP peak total radial average enthalpy (29 cal/g) was well below 150 cal/g, minimum DNB is being used to assess clad failure (although none predicted), and the change in radial average fuel enthalpy was well below 125 Δcal/g (approximately 60 Δcal/g, based on a peak of 154 cal/g and an initial enthalpy around 90 cal/g). Hence, CC2CY19 meets the stricter interim cladding failure criteria.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the CCNPP UFSAR, and responses to RAIs, the NRC staff finds the CRE analysis acceptable with the aforementioned license conditions.

TABLE 5: Comparison of STRIKIN-II and XTRAN Calculations

	Ejected Rod Worth (%Δρ)	Post-Ejected Fq	Delayed Neutron Fraction	Doppler Coefficient (%Δρ/°F)	XTRAN Fuel Enthalpy (cal/g)	STRIKIN-II Fuel Enthalpy (cal/g)
HZP						
CC2CY19, BOC	[[ ]]*	[[ ]]*	[[ ]]	[[ ]]	[[ ]]	- -
CC2CY19, EOC	[[ ]]*	[[ ]]*	[[ ]]	[[ ]]	[[ ]]	- -
CCNPP UFSAR	0.870	24.70	0.00440	-0.00060**	304***	180
ANO-2 Cycle 2	0.820	20.58	0.00482	-0.00060**	288	164
ANO-2 Cycle 10	0.820	12.71	0.00536	-0.00060**	168	95
WSES-3 , BOC	0.799	24.02	0.00723	-0.00060**	189	193
WSES-3, EOC	0.825	21.90	0.00530	-0.00060**	264	155
HFP						
CC2CY19, BOC	[[ ]]*	[[ ]]*	[[ ]]	[[ ]]	[[ ]]	- -
CC2CY19, EOC	[[ ]]*	[[ ]]*	[[ ]]	[[ ]]	[[ ]]	- -
CCNPP UFSAR	0.310	5.04	0.00440	-0.00060**	258***	185
ANO-2 Cycle 2	0.300	5.15	0.00482	-0.00060**	223	156
ANO-2 Cycle 10	0.280	5.21	0.00536	-0.00060**	220	157
WSES-3 , BOC	0.164	4.73	0.00723	-0.00060**	195	154
WSES-3, EOC	0.146	4.87	0.00530	-0.00060**	209	147

* Proprietary

** Generic least negative Doppler curve used in UFSAR analyses. Asymptotic curve approaches -0.00060 %Δρ/°F at 4000 °F.

*** Differences in calculated fuel enthalpy relative to RAI #24b due to assumed Doppler (-0.001 in RAI #24b, -0.0006 in Table 3

5.3 Small Break LOCA

CCNPP consists of 2x4-loop, Pressurized Water Reactors (PWR) designed by Combustion Engineering, Inc. enclosed within a large, dry containment. The emergency core cooling system (ECCS) consists of two high pressure safety injection pumps, two low pressure safety injection pumps and four safety injection tanks (SITs). These high and low pressure pumps deliver coolant from the refueling water storage tank to the cold legs. The safety injection tanks also deliver coolant to the cold legs and are pressurized with a nitrogen cover gas with a pressure of 194.7 psia. No credit for the charging pumps was assumed in the evaluation of the small break LOCA spectrum.

5.3.1 Method of NRC Staff Review

The NRC staff reviewed the licensee's application containing the results of the small break LOCA analyses to ensure 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) activities proposed will be conducted in compliance with the Commission's regulations as stated in 10 CFR 50.46, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the staff's review is, therefore, to evaluate the licensee's assessment of ECCS performance following a small break LOCA.

The NRC staff's evaluation also included an audit of the AREVA calculations pertaining to small break LOCA ECCS response. The staff also performed independent calculations, using the RELAP5/MOD3 code, to study a spectrum of small breaks.

5.3.2 Evaluation

The staff evaluation consisted of reviewing the results of the licensee evaluation of the small break LOCA spectrum using the S-RELAP5 code, performed at 2754 mega-watt-thermal (MWt) (including a 1.02% uncertainty) and a peak linear heat generation rate of 15.0 kW/ft for CCNPP. The evaluation of the small break LOCA will be discussed below.

The staff analysis employed a RELAP5/MOD3 model which included 24 axial cells to better track the two-phase level in the core, which also included a hot bundle parallel channel containing the hot rod with the same level of axial detail. The most limiting top skewed power shape was also used in the evaluation. All four loops in the RELAP5 model were represented explicitly in the nodalization of the CCNPP plants. The ECCS was also modeled as well as the ADVs and power-operated relief valves (PORVs) to assess the plant cool down

5.3.2.1 Small Break Short Term Behavior

The submittal for small breaks submitted by the licensee using the NRC-approved S-RELAP5 code included a detailed break spectrum analysis which included the **[[**

]]

ft^2 cold leg breaks in the reactor coolant discharge leg. The peak clad temperature (PCT) for the limiting break was calculated to be 1352°F for the 0.15 ft^2 cold leg break. The NRC staff RELAP5/MOD3 analysis showed that the limiting break was the 0.075 ft^2 cold leg break at the reactor coolant discharge with a PCT of 1875°F. It should be noted that the current design basis small break LOCA analyses performed by CE using staff approved small break LOCA methods was calculated to be 1855°F for the 0.08 ft^2 cold leg break, which is consistent with the staff calculated PCT of 1875°F for the 0.075 ft^2 break using the RELAP5/MOD3 code. The slightly different break area for the limiting break for the CE and staff analysis is due to differences in the critical break flow model. However, because of the large discrepancy between the licensees PCT and the staff/ CE small break LOCA analyses of record (i.e. a difference in PCT of 523°F), the staff performed an audit and detailed review of the licensee's small break LOCA methods and input to the AREVA S-RELAP5 small break LOCA methodology previously approved by the staff.

Review of the licensee's small break LOCA analysis revealed the following errors and omissions:

- The licensee incorrectly assumed **[[**

  ]] the core steaming need not vent through the higher resistance loop piping.

- The kinetics model determining for core power prior to reactor trip failed to include the moderator–density feedback effects. Since the MTC is positive for CCNPP, this omission precludes an over-power transient in the early phase of the small break LOCA.

- The upper core barrel leakage paths **[[**

  ]] maximized the two-phase level in the core.

- The SIT temperature was assumed to be 90ºF instead of the maximum value of 120ºF.

- The analysis also failed to locate the largest small break that results in the minimum RCS pressure remaining slightly above the SIT actuation pressure of 194.7 psia.

The NRC staff questioned the potential in the S-RELAP5 code to allow top down cooling of liquid from the upper plenum to drain into the core providing cooling during periods of core uncover. AREVA modified the S-RELAP5 inputs for CCNPP to include a large reverse flow K-factor at the exit to the core to assure top down cooling of liquid from the upper plenum is prevented in the hot bundle during the LOCA.

The licensee re-analyzed the small break spectrum with the above corrections **[[**

]] and found the limiting break to now be the 0.16 ft^2 break with a PCT of 1454ºF[50].

Since the new PCT was still well below the staffs calculated PCT of 1875ºF and that for the CE analysis of record, further evaluations of the licensee's small break model were under taken to

[50] Reference 5 op. cit., RAI 23. ML103280082.

understand the large discrepancy. **[[**

]] a PCT of 1626ºF for the 0.09 ft^2 break[51]. Moreover, since the PCT was terminated by SIT injection for this break, the licensee was asked to confirm that this was the limiting break, as a slightly smaller break which does not actuates SITs may be more limiting. Additional analyses demonstrated that the 0.089 ft^2 break produced a PCT of 1605ºF or 130ºF higher than the 0.08 ft^2 break but not greater than the limiting 0.09 ft^2 break.

Review of the limiting 0.09 ft^2 break revealed that the two-phase level displayed no increases or decreases from 100 to 1500 seconds in the transient. This was found to be due to the fact that S-RELAP5 homogeneously mixes the liquid and vapor in a volume or cell region. The nodalization produces cell heights in the core of 6 inches. As such, all of the liquid must drain from the cell until heat-up begins in the core above the two-phase surface. This model behavior reduces the heat-up of the vapor above the two-phase surface in the core during uncover and acts to reduce the PCT. Since there is considerable margin for this break relative to the 2200ºF criterion limit, the staff finds that this result can be accepted for this application. However, should the PCT increase above the current limiting temperature of 1626ºF for this break, additional sensitivity studies on cell height or modifications to S-RELAP5 to compute the two-phase level within a cell will be required to assure the limiting PCT is identified. This condition, as stated in License Condition 9, will be included as a restriction on the S-RELAP5 analysis for determining PCT for the limiting SBLOCAs for CCNPP .

With the 0.09 ft^2 PCT of 1626ºF, with modifications to allow only a single loop suction leg piping to clear of liquid, this break is identified as the limiting break size. It should also be noted that the NRC staff analysis with the higher PCT of 1875ºF is due, in part, to the use of the CE HPSI flow curve that is reduced by an additional 5% margin from the surveillance flow acceptance criteria and other differences in the heat transfer correlations affecting the clad temperature heat-up methods.[52] The additional flow reduction was included in the CE and staff calculations after the HPSI flow curve was developed from the surveillance testing, which included the flow measurement uncertainty. The licensee did not include the additional 5% reduction in flow in the S-RELAP5 analysis and stated that the head-flow curve used in their analysis would be confirmed with the new surveillance testing to be conducted during the refueling of the replacement AREVA nuclear fuel.

Since the generic break spectrum model was shown to predict a non-conservative peak cladding temperature, the NRC staff concludes that a license condition is necessary to capture the more restrictive design criteria for CCNPP reload designs. License Condition 9 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact

[51] Reference 9 op. cit., RAI 12, ML110180621.
[52] Reference 12 op. cit. RAI 23, ML103080025.

license condition for both CCNPP Units 1 and 2. The following summarizes the license condition.

> <u>License Condition 9</u>:
>
> The small break loss of coolant accident performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size.

5.3.2.2 Severed Injection Line

The NRC staff questioned whether a severed emergency core cooling (ECC) injection line analysis was performed. Since this analysis was not included, it could potentially represent the limiting small break LOCA size since the severed injection line results in more than 25% of the broken ECC injection spillage to containment. Likewise, this condition results in much less than 75% injection flow into the RCS through the other three intact injection lines, since with the broken injection line pumping against containment pressure and the intact lines feeding the much higher pressure RCS, more flow is pumped through the broken ECC line.

Based on these considerations, the licensee performed an analysis of a severed injection line **[[ ]]**, which is less than the limiting 0.09 ft^2 break with the PCT of 1626ºF, discussed above[53].

In the analysis of the severed injection line, the licensee did not credit HPSI flow to the cold legs during the injection phase and only included SIT flow. This bounding assumption maximizes the PCT for the severed injection line event. The NRC staff finds this evaluation conservative and bounding for injection line breaks.

5.3.2.3 RCP Trip Based on the More Limiting Hot Leg Breaks

The licensee submitted an analysis of the effect of RCP operation on small breaks in the cold leg, only. The NRC staff posed that the limiting break is expected to be a hot leg break because during the early portion of the LOCA, the operating RCP acts to push the two-phase level well above the bottom elevation of the hot leg, causing a large amount of liquid to be lost through the break during the flow coast-down period of the LOCA. Once the RCPs void and cavitate, thus losing their driving head, the liquid levels will equilibrate in the vessel between the downcomer and core regions, producing a deep and prolonged core uncovery. The core uncovery that follows the loss of the RCPs driving head is more severe than that for the limiting cold leg break when the RCPs tripped very early in the event. As a consequence, an analysis of hot leg breaks is required to determine (1) the potential limiting nature of hot leg breaks with the RCPs running, and (2) identification of the possible more restricted timing for tripping the RCPs for input to the emergency operating procedures (EOPs) so that the operators may terminate RCP operation following the most break location. The timely tripping of the RCPs following a break in the most limiting location is necessary to assure 10 CFR 50.46 acceptance criteria for ECC performance are not exceeded.

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[53] Reference 9 op. cit., RAI 13. ML110180621.

The licensee performed a spectrum of hot leg breaks and found breaks in this location to be more limiting than the previously analyzed cold leg breaks[54]. The limiting cold leg break of 0.12 ft^2 in the pump discharge leg requires an operator delay time for tripping the RCPs of no more than 7 minutes following the loss of RCS subcooling of 20ºF. The limiting hot leg break was found to be **[[ ]]**. And for breaks in this location, the analysis demonstrated that the operators must trip the RCPs within no more than 4 minutes following loss of subcooling in the RCS. Thus, operator action described in the EOPs is necessary to maintain the PCT below 2200ºF for small hot leg breaks. In summary, the licensee performed a detailed spectrum of both cold and hot leg breaks in the range 0.5 ft^2 down to and including 0.06 ft^2 to identify the more restrictive 4 minute operator action to trip RCPs following a small break LOCA in the hot leg.

The licensee stated that the 4 minute RCP trip time is input to EOP-0 for CCNPP. When questioned as to how this trip time would be met and confirmed as part of the operator training program, the licensee stated that they are developing a formal program, or a job/simulator performance measure for establishing this Time Critical Action for tripping all RCPs based on loss of 20ºF subcooling. The simulator/job performance measure will be implemented independently of the formal Time Critical Actions program. The simulator performance measure includes pass/fail criteria for training operators on the simulator. This simulator/job performance measure will be incorporated into the training program prior to the loading of AREVA fuel. Operators will be trained on all associated procedure changes prior to plant operation with AREVA fuel loaded in the core. The licensee also confirmed that during three separate training simulations the operators during a LOCA simulation tripped all four RCPs in 2 minutes and 10 seconds or less. Section 5.5 of the SE describes the NRC staff's review of the proposed operator actions.

5.3.3 Small Break LOCA Conclusion

The NRC staff reviewed the licensee's vendor small break LOCA analyses and performed audit calculations using RELAP5/MOD3 applicable to CCNPP, operating at 2754 MWt and 15.0 kW/ft. The staff's review confirmed that the licensee and its vendor have processes to assure that the Calvert Cliffs specific input parameter values and operator action times (where appropriate) that were used to conduct the analyses will assure that 10 CFR.50.46 limits are not exceeded following small break LOCAs. Furthermore, the staff finds that the analyses were conducted within the conditions and limitations of the NRC-approved AREVA S-RELAP5 small break LOCA methodology with several changes listed below, and that the results satisfied the requirements of 10 CFR 50.46(b), at a power level of 2754 MWt and a peak linear heat generation rate of 15.0 kW/ft.

The staff notes that, to support the acceptability of CCNPP operation at 2754 MWt and 15.0 kW/ft, the licensee included the following staff recommendations, modifications to the S-RELAP5 input modeling and changes to the EOPs to assure successful ECCS performance for small breaks that include the following:

[54] Reference 9 op. cit., RAI 14. ML110180621.

1. Modifications to the S-RELAP5 modeling to allow one cold leg suction piping to clear of liquid for all small breaks with diameters of 4 inches or less.

2. Removal of credit for the hot leg nozzle gaps and the upper core barrel flange.

3. Leakage paths that represent communicate paths for fluid flow between the upper plenum and upper head directly into the upper downcomer region.

4. Inclusion of a large reverse flow K-factor at the outlet of the core to prevent the downflow of liquid from above to cool the core hot bundle during periods of core uncover.

5. The HPSI head-flow curve input to the S-RELAP5 code will be verified against the surveillance testing to be conducted prior to power operation with the AREVA fuel loaded in the core. The head flow curve should include adjustments for all measurement uncertainties associated with the surveillance test.

6. The simulator operator training and qualification should be conducted periodically to ensure the operators can trip the RCPs following the limiting small break LOCA within 4 minutes following loss of 20ºF subcooling.

The following restriction, as described in License Condition 9, is imposed on the S-RELAP5 small break LOCA methodology for CCNPP:

> Should the PCT increase above the current limiting break PCT of 1626ºF in any subsequent evaluation, the licensee will be expected to correct the ability of the S-RELAP5 code to more accurately compute the two-phase level and resultant heat-up of the fuel cladding in the core. Currently, the core nodalization produces core cells with a height of 6 inches. Because of the homogeneous assumption regarding vapor and liquid mixing in a control volume (cell) in S-RELAP5, saturated conditions are imposed on the entire volume regardless of the amount of liquid contained in the volume. As such, no heat-up occurs in the cell containing the two-phase surface in the core until all of the liquid in the cell drains to the cell below. To correct this deficiency, more cells in the core region are required or modifications to compute the two-phase surface within the cell containing the level are necessary to properly account for the vapor superheat and cladding heat-up in this region.

In areas where the licensee and its contractors used NRC approved methods in performing analyses related to the proposed nuclear fuel transition from Westinghouse to AREVA nuclear fuel, the NRC staff reviewed relevant material to ensure that the licensee used the methods consistent with the limitations and restrictions placed on these methods. In addition, the staff considered the effects of the changes to the S-RELAP5 input modeling to conservatively bound ECCS performance for small breaks on the use of these modified methods to ensure that the they are appropriate for use at 2754 Mwt and 15.0 kW/ft.

Based on the review of the licensee's small break LOCA analyses, the NRC staff concludes that the AREVA S-RELAP5 small break LOCA methodology, including the implemented modeling changes and modifications, is acceptable for CCNPP to demonstrate acceptable ECCS performance and compliance with the temperature and oxidation criteria requirements of 10 CFR 50.46 at 2754 MWt and 15.0 kW/ft.

5.4 Realistic Large Break LOCA (RLBLOCA)

The licensee requested to implement AREVA NP licensing topical report EMF-2103(P)(A), "Realistic Large Break LOCA Methodology," for use in demonstrating compliance with 10 CFR 50.46 requirements for postulated large break loss of coolant accidents.

5.4.1 Regulatory Evaluation

The RLBLOCA analyses were performed to demonstrate that the ECCS design would provide sufficient ECCS flow to transfer the heat from the reactor core following an LBLOCA at a rate such that: (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than the amounts that would compromise cladding ductility and result in excessive hydrogen generation in accordance with 10 CFR 50.46.

The NRC staff reviewed the analyses to assure that the safety functions could be accomplished, assuming a single failure, for large break LOCAs (LBLOCAs) and considering the availability of only onsite or offsite electric power (i.e., assuming offsite electric power is not available with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available).

10 CFR 50.46 requires that the ECCS must be designed such that, when analyzed using the guidance set forth in 10 CFR 50.46, it demonstrates acceptable performance subject to the criteria contained in 10 CFR 50.46(b)(1) through (b)(5). Other evaluations unaffected by this LAR demonstrate acceptable performance relative to 10 CFR 50.46(b)(4) and (b)(5); therefore, this SE is concerned with the licensee's demonstration of compliance with 10 CFR 50.46(b)(1) – (b)(3).

The NRC staff used the acceptance criteria for ECCS performance provided in 10 CFR 50.46(b) in assessing the application of the AREVA RLBLOCA methodology for CCNPP. In its assessment of the application of the methodology for CCNPP, the staff also reviewed the limitations and conditions stated in its SE supporting generic approval of the AREVA RLBLOCA methodology and the range of parameters described in EMF-2103(P)(A), Revision 0.

5.4.2 Technical Evaluation

The NRC staff review of the licensee's request to implement the EMF-2103 methodology is comprised of three elements. The staff reviewed (1) the results to ensure they comply with 10 CFR 50.46 requirements, (2) the implementation to ensure the methodology is correctly applied, and (3) the input assumptions to ensure they are reflective of the plant licensing basis.

5.4.2.1 Results

The licensee provided the results for the CCNPP RLBLOCA analyses, operating at the rated power of 2737 MWt (performed in accordance with the AREVA RLBLOCA methodology). The licensee's results for the calculated PCTs, the maximum cladding oxidations (local), and the maximum core-wide cladding oxidations for CCNPP are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

Calvert Cliffs Nuclear Power Plant
LARGE BREAK LOCA ANALYSIS RESULTS

Parameter	Results	10 CFR 50.46 Limits
Peak Clad Temperature	1670°F	2200°F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	0.907%	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	0.011%	1.0% (10 CFR 50.46(b)(3))

The results demonstrate compliance with 10 CFR 50.46 acceptance criteria and on that basis the NRC staff finds that they are acceptable.

The NRC staff observed a difference between the limiting LBLOCA scenario described in the UFSAR and that described in ANP-2834(P)[55]. Specifically, the limiting scenario in the UFSAR was based on conditions that minimized containment pressure, whereas the limiting scenario described in ANP-2834(P) included a single failure of a safety injection (SI) train instead. The staff issued two RAIs concerning this. One RAI concerned the low-pressure containment scenario, and the other concerned AREVA's limiting single failure evaluation.

In response to the RAI concerning the containment, the licensee stated that the AREVA evaluation model assumed the use of full containment sprays without a time delay at the minimum TS temperature and containment volume sampling from nominal to empty volume[56]. The NRC staff finds the licensee's response acceptable because the containment spray assumptions will serve to minimize the containment pressure, while the licensee also stated that the containment volume assumptions, which affect the containment pressure, show little sensitivity to the PCT result. Therefore, the overall assumptions are slightly conservative with respect to containment pressure, and this is consistent with the current licensing basis analysis.

In response to the RAI concerning the single failure evaluation, the licensee stated that AREVA had performed a generic single failure evaluation during development of the EMF-2103 methodology, and that the generic evaluation demonstrated that the limiting single failure was the loss of a single SI train[57]. The NRC staff reviewed the documentation concerning this evaluation, which is included as RAI responses appended to EMF-2103(P)(A). Revision 0, and finds that although it considered sufficient single failures to demonstrate that the PCT-limiting single failure is the loss of a SI train for a Westinghouse nuclear steam supply system (NSSS) design, the applicability of this sensitivity study to the CE NSSS design was not clear.

[55] Reference 1 op. cit., Enclosure 1 ML093350099
[56] Gellrich, G. H., Constellation Energy, letter to U.S. NRC, "Response to Request for Additional Information – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," Dockets 50-317 and 50-318, August 9, 2010. ML102230340. RAI 2.
[57] Reference 55 op. cit., RAI 3. ML102230340.

The licensee provided additional information clarifying that the loss of a SI train assumption excludes the containment spray system, which remains fully operational[58]. This has the effect of lowering containment pressure, which is the driving phenomenon for the CE-analyzed case, in which maximum SI spillage into containment reduces the containment pressure and minimizes the primary coolant inventory. The licensee also provided the results of a sensitivity study of the RLBLOCA case with the maximum PCT, comparing it to a similar run with maximum ECCS flow. The maximum ECCS case had a slightly lower PCT – 1656°F compared to 1670°F – than the reported case. Although the licensee stated that this sensitivity study demonstrated the conservatism of the AREVA modeling assumption, the NRC staff does not agree with this characterization, because the predicted PCTs are so similar. The staff finds instead that the difference between the maximum ECCS and loss of one train with containment spray available cases is sufficiently small compared to the predicted margin to the PCT acceptance criterion in 10 CFR 50.46(b) – 14°F compared to 530°F – that the chosen single failure assumption is satisfactory.

The licensee stated in Section 3 of ANP-2834(P) that the RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water for ECCS pumped injection need not be considered. The NRC staff observed, however, that parametric plots of reactor vessel water level did not appear to be stable or increasing at the time of transient termination. The staff requested that the licensee extend the liquid mass plot to show that vessel inventory was, in fact, stable or increasing. The licensee provided a figure demonstrating that the vessel inventory began to increase at about 100 sec, and continued to do so through the termination of the plot at 350 seconds[59]. The staff accepts the response because it provided the requested demonstration.

5.4.2.2 EMF-2103(P)(A) Implementation

<u>Adherence to Conditions and Limitations</u>

The NRC SE approving EMF-2103(P)(A), Revision 0, included 13 conditions and limitations restricting its use. The licensee provided Table 3-4 in ANP-2834(P), which listed the conditions and limitations and included information to demonstrate that adherence to each had been attained. The NRC staff reviewed this information and concluded that the licensee's implementation of EMF-2103 is acceptable because it was adhering to each of the conditions and limitations.

The licensee's adherence to each of the conditions and limitations listed in Table 3-4 of ANP-2834(P) is apparent, with the exception of Item 3, concerning the re-evaluation of a need for a blowdown cladding rupture model. During its review, the NRC staff requested additional information concerning AREVA's basis for not including a cladding ballooning and rupture model[60]. In addition to a CCNPP-specific disposition regarding the need for a cladding ballooning/rupture model, the licensee also provided a generic, phenomenological discussion suggesting that cladding rupture need not be modeled.

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[58] Reference 6 op. cit., RAI 17. ML110040374.
[59] Reference 55 op. cit., RAI 4. ML102230340.
[60] Reference 6 op. cit., RAI 31. ML110040374.

The licensee considered a number of cooling effects attributable to a postulated cladding rupture, based largely on the geometric change associated with the rupture of a single pin, and compared these to three heating effects related to fuel pellet relocation and cladding oxidation. While the licensee cited experimental data to show that the cooling effects are greater than the possible heating effects, the licensee also acknowledged a body of experimental evidence showing, albeit in extreme conditions, behavior to the contrary. Based on the licensee's discussion, and the NRC staff's awareness of increasing experimental data, and ongoing research, showing fuel relocation into the region of cladding swell as high as 80-percent fill fraction, which has been shown to significantly increase the heat load in this region, the staff currently believes that AREVA's omission of a model representing fuel rod swell, rupture, and pellet relocation is unjustified and possibly non-conservative. Calculations have shown that, with 70-percent fill fractions attributable to pellet relocation in the balloon/burst region, peak cladding temperatures increase as much as 400°F.

The CCNPP-specific argument provided by the licensee, however, stated that **[[**

]]. In consideration of this, and the fact that the predicted peak cladding temperatures at CCNPP do not exceed 1800°F, the NRC staff finds that there is reasonable assurance that blowdown cladding ruptures would not occur during a postulated LOCA at CCNPP, and thus the model is acceptable because it provides an acceptable representation of the LBLOCA progression without modeling a blowdown cladding rupture.

Departures from EMF-2103(P)(A), Revision 0

Subsequent to its approval of Revision 0 to EMF-2103(P)(A), the NRC staff has found that certain modeling assumptions and constitutive relationships contained in the EMF-2103 methodology are not suitable for demonstrating compliance with the 10 CFR 50.46(b) acceptance criteria, as described in a draft safety evaluation dated April 3, 2007 (ML070940125[61]). Therefore, the staff also considered the conditions and limitations in this draft safety evaluation, and the corresponding departures from NRC-approved EMF-2103(P)(A), Revision 0, required to adhere to these conditions and limitations.

The power assumed in the analyses, 2754 MWt, is 0.62-percent higher than the operating power of CCNPP to account deterministically for measurement uncertainties. This departure from the previously approved methodology is acceptable because it is conservative in that the previously approved methodology permitted ranging the assumed power level, meaning that some cases could have initiated at a power level less than 2754 MWt, had the analysis been performed using the previously approved methodology. It is also acceptable because it is consistent with the NRC staff's position that parametrically ranging the assumed initial power level is inconsistent with 10 CFR 50.46 requirements, whereas deterministically including uncertainty in the assumed initial power level is acceptable.

[61] This safety evaluation was never formally issued, and the vendor withdrew the topical report revision that it supported. Therefore, there is no publicly available copy of this report. Nonetheless, the NRC staff has continued to use adherence to the conditions and limitations listed in this safety evaluation as a basis for its approval of requests to implement EMF-2103, Revision 0, as an interim review approach until a second revision to EMF-2103 is submitted to the NRC for review and approval. While EMF-2103(P)(A) is an acceptable evaluation model as described in 10 CFR 50.46, the NRC staff requires these deviations for plant-specific application of the methodology.

The RLBLOCA analysis was performed with a version of S-RELAP5 that requires both the void fraction to be less than 0.95 and the clad temperature to be less than 900°F before the rod is allowed to quench. During its review of EMF-2103(P), Revision 1, the NRC staff determined that the S-RELAP5 evaluation model could allow rod quench to occur once the temperature drops below the minimum film boiling temperature regardless of the void fraction in the channel. Contrarily, NUREG-0915 demonstrated that the void fraction must also be less than 0.95 for rod rewet to occur. To address this concern for CCNPP, the CCNPP-specific analyses includes this departure from the approved methodology, and the staff finds this acceptable because the departure provides for analytic predictions that are not only more consistent with observed data, but also more conservative than predictions obtained using the previously NRC-approved methodology.

The RLBLOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than 15-percent of the total heat transfer at and above a void fraction of 0.9. This departure from the approved methodology accounts for experimentally observed phenomena that appear to inhibit droplet contact with heated fuel rods at high void fractions. Thus, this departure is conservative relative to the approved methodology because it corrects for any potential to over-predict heat transfer through conduction to entrained droplets, which experimental observations have shown not to come in contact with the fuel rods at such high void fractions. The net effect of this conservatism would serve to increase the overall predicted PCT compared to evaluations performed using the previously NRC-approved methodology.

The analyses ranged in area between the minimum break area and an area of twice the size of the broken pipe. The licensee stated that A_{min} was calculated to be 28.7-percent of the double-ended guillotine break area. This information demonstrates that the total number of sampled cases is appropriate because the phenomenology dominating the limiting aspects of the event for all sampled break areas is consistent. That is, a certain number of sampled cases is appropriate, because the limiting results of the accident for pipe ruptures ranging from about 20- to 100-percent of the double-ended pipe rupture size are all limited by dispersed flow film boiling ahead of the quench front. If the sampled break area included a greater range, i.e., break sizes less than 20-percent of the double-ended guillotine rupture, additional phenomenology would govern the limiting events, and additional cases would be required to provide the necessary high level of statistical confidence that a bounding upper tolerance limit had been identified.

The analyses addressed the availability of offsite power correctly by ranging each case separately. This is acceptable because it satisfies General Design Criterion 35 of 10 CFR 50 Appendix A, in that each distribution type has been accounted for separately with its own set of cases, thereby addressing possible concerns associated with the mixing of two separate statistical spectra. The NRC staff finds this treatment acceptable because it is consistent with the staff's position regarding compliance with GDC 35 as noted in ANP-2834(P).

The NRC staff has historically identified differences in results between staff confirmatory calculations and those produced using the AREVA RLBLOCA evaluation model, attributable to downcomer boiling modeling, that cause significant differences in peak cladding temperature results. The NRC staff's confirmatory models predict peak cladding temperatures on the order of 400°F higher than those obtained using the AREVA model. As discussed in Section 1.0 of

ANP-2834(P), AREVA attributes this to an underprediction of cold leg condensation efficiency. To correct for this, AREVA has identified appropriate multipliers to force fluid entering the downcomer to saturated conditions following the deployment of the SI tanks. The staff finds this departure from the previously approved methodology acceptable because (1) the artificially saturated fluid conditions will conservatively reduce both the downcomer driving head and the core flooding rate, which becomes conducive to portions of the fuel remaining in a vapor-cooled environment, thus presenting a greater challenge to clad surface cooling, and (2) conditions in the downcomer following safety injection tank discharge are expected to be slightly subcooled, meaning that assuming fully saturated conditions is conservative[62].

Additional Issues with EMF-2103(P)(A), Revision 0

On August 3, 1998, the NRC issued IN 98-29: "Predicted Increase in Fuel Rod Cladding Oxidation," expressing concern that predicted cladding total oxidation (including both pre-accident and accident oxidation) resulting from a postulated LOCA could for some plants exceed the 17 percent limit, and that LOCA methodologies were not addressing that concern.

In letters dated March 31 and November 8, 1999, to the Nuclear Energy Institute, NRC provided its position that both pre-accident and accident oxidation must be estimated, citing several references, including the Opinion of the Commission dated December 28, 1973, that demonstrate that the NRC position regarding oxidation predates the present LOCA acceptance criteria and the first accepted LOCA evaluation models under those criteria.

The licensee's estimate of oxidation resulting from the postulated LBLOCA alone for M-5 cladding is 0.907 percent, without considering pre-LOCA oxidation. EMF-2103(P)(A) provides a generic disposition concluding that considering transient-only oxidation is acceptable. Given the very low values of cladding oxidation reported for CCNPP, the NRC staff did not evaluate the applicability of this generic disposition to CCNPP.

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the RCS throughout the operating cycle, the NRC staff noted that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA and therefore it does not need to be addressed further when determining whether the calculated total core-wide oxidation meets the 1.0 percent criterion of 10 CFR 50.46(b)(3).

While 50.46 requires consideration of oxidation on both inner and outer cladding surfaces, EMF-2103(P)(A), Revision 0, provides a generic disposition regarding the EMF-2103 treatment of two-sided cladding oxidation. The disposition defers to generating a limiting peak cladding temperature, since AREVA argues that a clad burst would cool the fuel and result in a lower PCT result[63]. The NRC staff does not agree with this argument (see Section 5.4.2.2, "EMF-2103(P)(A) Implementation," subsection *Adherence to Conditions and Limitations* for additional discussion).

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[62] The NRC staff questions the validity of the AREVA downcomer nodalization, but did not question the CCNPP-specific application any further because the predicted PCT is below 1800°F with a containment pressure in excess of 30 psia, conditions which indicate that substantial downcomer boiling is unlikely.
[63] Reference 6 op. cit., RAI 32. ML110040374.

The NRC staff notes that the predicted cladding oxidation result is very low. Doubling it would still be within the 10 CFR 50.46 acceptance criteria. The staff also notes that the licensee did not predict conditions which would result in cladding rupture of fuel through the second cycle[64]. Based on these considerations, and in light of the fact that the predicted PCT is 1670°F, which is below the threshold temperature at which cladding oxidation becomes significant, the staff finds that the omission of consideration of double-sided oxidation is acceptable for this application of the EMF-2103(P)(A) method.

NRC IN 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," describes a recently identified issue concerning the ability of legacy thermal-mechanical fuel modeling codes to predict the exposure-dependent degradation of fuel thermal conductivity accurately. Some legacy codes, including RODEX-3A, non-conservatively over-predict fuel thermal conductivity at higher burnups. To correct for this issue, AREVA has applied a polynomial transformation, which is used to bias the fuel pellet centerline temperature based on empirical data collected supporting the more recent RODEX4 fuel performance code.

Information provided by the licensee explained the polynomial transformation applied to the fuel centerline temperature, but did not explain how the temperature augmentation propagated out to the cladding surface[65]. A safety concern with fuel thermal conductivity degradation in a LOCA would be that fuel temperatures modeled incorrectly would affect the heat transfer to the cladding surface, causing the LOCA evaluation model to predict potentially erroneous PCTs.

In its December 30, 2010, supplemental letter, the licensee provided information stating that the fuel pellet centerline temperature (FCT) correction has a minor impact on the cladding surface temperature. Additional information provided by the licensee also demonstrated that, for the PCT-limiting case, the pellet centerline and cladding surface temperatures tend to converge rapidly – on the order of seconds – during the transient. The burnup for the limiting case was 25.7 GWd/MTU, however. It is not clear that the FCT and cladding surface temperature would converge so quickly, or to the same low temperature, if the initial FCT were higher.

The staff remains concerned that the centerline temperature correction does not provide a sufficiently detailed representation of the impact of fuel thermal conductivity degradation as a function of burnup. Moreover, since the AREVA evaluation model only evaluates fresh fuel, it was not clear to the staff that an assembly with greater burnup – and therefore a more significant fuel thermal conductivity degradation – would not have a higher PCT.

The licensee stated that, through the second cycle, compensating for the burnup-dependent fuel thermal-conductivity degradation would impose a roughly 7% fuel centerline temperature penalty on the once burnt fuel. But the co-resident Westinghouse fuel has a significantly lower assembly peaking factor in its second cycle. The Westinghouse fuel is about 14% less peaked, which apparently compensates for any deficiencies in the capability of RODEX-3A to provide an accurate representation of fuel temperature profile, and the fact that AREVA only models fresh fuel in its RLBLOCA EM.

[64] Reference 6 op. cit., RAI 31. ML110040374.
[65] Reference 55 op. cit., RAI 1. ML102230340.

The information provided by the licensee demonstrates that the once-burnt Westinghouse fuel is significantly less peaked than the first-cycle AREVA fuel. This, in turn, assures the staff that the limiting PCT case, based on a fresh rod with 25 GWD/MTU burnup, is bounding of second-cycle Westinghouse fuel. However, it is still not clear that similar conclusions could be drawn regarding a second campaign of AREVA fuel.

The NRC staff has concluded that a license condition is necessary to restrict plant operation to a single cycle under the current LBLOCA analysis of record, and to obtain NRC review and approval of a generic disposition concerning the analysis of only first-cycle fuel in light of the fuel thermal conductivity degradation issue. License Condition 10 is unit-specific at CCNPP and linked to an operating cycle. Section 7.0 of this SE lists the exact license condition for both CCNPP Units 1 and 2. The following summarizes the license condition.

> License Condition 10:
>
> The approval of the emergency core cooling system evaluation performed in accordance with the methodology of Technical Specification 5.6.5.b.7 shall be valid only for Calvert Cliffs' current cycle. To remove this condition, Calvert Cliffs shall obtain NRC approval of the analysis of once- and twice-burned fuel for core designs following the current cycle.

In light of the fact that the once-burnt, co-resident fuel will be significantly less peaked than the first-cycle AREVA fuel, and the licensee stated that this power peaking tradeoff compensates for the increased stored energy and possible fuel centerline temperature increases in the older fuel, the NRC staff finds that the proposed realistic large break LOCA method is acceptable for implementation with respect to the issues identified in IN 2009-23. This finding is valid for a single cycle, consistent with the license condition presented above.

The NRC staff questioned the licensee's application of the decay heat sampling technique prescribed by EMF-2103. The licensee's PCT-limiting case had a decay heat multiplier that was less than one.

Consistent with NRC regulatory guidance for realistic LOCA evaluation models, the licensee uses the ANS-1979 standard. The licensee samples a normal distribution of decay heat uncertainty, characterized by a one-sigma uncertainty. Although the NRC staff finds the use of the 1979 decay heat standard acceptable for realistic evaluation models, the staff does not find that this sampling technique is appropriate for the standard, because the standard bounds stochastic processes over time, and sampling with a single chosen decay heat multiplier for the duration of the transient does not provide an appropriate representation of the possible decay heat behavior. While the EMF-2103 realistic model uses a constant exponential decay relationship to model the decay heat in each analyzed case, the actual decay heat behavior is not expected to follow a single exponential curve.

This problem is best described by comparing the decay heat mechanistic phenomena to other input parameters that are also ranged in the EMF. Consider instead the initial SI tank water volume and cover pressure. In the case of an actual plant, the actual cover pressure and water volume may vary, but once the LOCA sequence initiates, the water volume is driven into the core by the nitrogen cover pressure, and once the volume of water stored in the accumulator is delivered to the core, the accumulator is empty. There are no other sources to increase cover pressure, or create additional water mass, in the SI tank. Therefore, ranging the initial condition is appropriate, because once it is established, it remains fixed at that value as a boundary condition for the transient.

By contrast, the decay heat standard provides representation of combinations of numerous nuclear processes that provide the total decay heat load. Therefore, at any given point in time, the actual decay heat may fall above or below the chosen decay heat curve. This is not an initial condition, but rather a time-varying parameter during the transient. The mechanistic difference is that the SI tank water is a fixed mass, but the decay heat generation is a stochastic process that varies erratically with time. The selection of a single decay heat multiplier for an entire transient, therefore, does not appropriately characterize the uncertainty and variability inherent in the decay heat process. For this reason, the NRC staff finds that the use of a non-bounding decay heat multiplier that is less than one is neither conservative, nor a realistic representation of the decay heat phenomena.

The licensee provided a discussion of several assumptions concerning the chosen decay heat model that may make it conservative[66]. Namely, the licensee assumes hot assembly operation near the peak linear heat rate, which is an operationally undesirable condition, and the licensee assumes decay energy from only U-235, with no decay energy from Pu-239, which has a lower decay energy. The licensee believes that this provides the methodology with a means to build in conservatism over the life of the core.

The staff does not accept these modeling approaches as quantifiable conservatisms in the evaluation model for several reasons. First, there is nothing to limit the licensee from operating its hot assembly near the peak linear heat rate. Indeed, the evaluation model needs to account for this possibility. Second, the use of the U-235 is not as conservative at the beginning of fuel life, and this assumption may not build in a significant amount of conservatism in the decay heat model, especially since the EMF-2103 model only models fresh fuel. The staff believes that additional studies and discussion are necessary in order to resolve this issue with the decay heat model chosen by AREVA.

The NRC staff has brought this issue, which the staff considers adverse to the quality of the ECCS performance evaluation, to the licensee's attention. The staff is aware that the licensee is evaluating the issue and will determine the net PCT impact and report it to the NRC in accordance with 10 CFR 50.46(a)(3). In the meantime, the staff notes that there is ample margin to the 2200°F peak cladding temperature acceptance criterion, which is considered sufficient to provide the margin to account for this, and the other, issues identified by the NRC staff with EMF-2103, Revision 0.

Input Assumptions

The NRC staff reviewed the information contained in Table 3-2 of ANP-2834(P) to ensure that the initial operating conditions supported by the LOCA analysis aligned with the CCNPP current licensing basis, or that any changes were appropriately justified. The staff did not observe any

[66] Reference 6 op. cit., RAI 33. ML110040374.

significant discrepancies between TS-limited parameters and the assumed initial operating conditions; however, clarification was required in some areas, as discussed below.

Pressurizer level was expressed in percent and the TS limit the level by inches. In response to an NRC RAI, the licensee stated that the sampled range for the pressurizer liquid level was 115.9 to 241.9 inches, which slightly exceeds the 133 to 225 inch range permitted by TSs. The NRC staff finds the initial conditions acceptable because they are consistent with and support the plant licensing basis requirements.

The NRC staff did not identify a licensing basis requirement regarding the minimum containment temperature; however, the licensee's analysis assumed a containment temperature range of 60°F to 125°F. The NRC staff requested that the licensee provide information to demonstrate that 60°F was an appropriate minimum. The licensee responded by stating that plant data from four previous years of operation was reviewed and minimum containment temperatures of 68°F and 65.7°F had been observed at Units 1 and 2, respectively. TS 3.6.5, "Containment Air Temperature," limits the containment temperature to a maximum of 120°F. The NRC staff finds that the containment temperature assumptions are acceptable, therefore, because they are bounding of achievable plant operation.

5.5 Human Performance

As discussed in Section 5.3.2.3 of this SE, the licensee proposed to credit an existing manual operator action to secure Reactor Coolant Pumps (RCP's) within 4 minutes of subcooling margin falling to < 20 degrees F. The NRC staff reviewed the licensee's operator training program to verify that the proposed operator actions are credible.

5.5.1 Regulatory Evaluation

The regulatory requirements and guidance which the NRC staff considered in its review are as follows:

1 Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria (GDC)," *Criterion 19 - Control room. "*A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents…. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures."

2 10 CFR 50.120, "Training and qualification of nuclear power plant personnel"

3 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." Chapter 13 addresses "Conduct of Operation", specific sub-chapters considered in this review were Chapters 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training", and 13.5.2.1, "Operating and

Emergency Operating Procedures". Chapter 18, provides review guidance for "Human Factors Engineering".

4 NUREG-1764,"Guidance for the Review of Changes to Human Actions;"

5 GL 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability";

6 NUREG-0700, "Human-System Interface Design Review Guidelines" Revision 2;

7 NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2;

8 NRC Information Notice (IN) 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times;"

5.5.2 Technical Evaluation

In accordance with the generic risk categories established in Appendix A to NUREG-1764, the task sequence, tripping RCPs, reviewed herein, is considered "risk-important" due to the fact that it is required to prevent core uncovery. Because of its risk importance, the NRC staff performed a "Level One" review, i.e., the most stringent of the graded reviews possible under the guidance of NUREG-1764.

Description of Operator Action(s) Added/Changed/Deleted

As a result of its review of operator actions, the licensee has identified a reduction in time available for tripping RCPs while executing EOP-0, "Post Trip Immediate Actions". The licensee identified no other operator actions that are required, including any effects on the time available for, or the time required to, perform operator actions.

This operator action, tripping the RCPs, is not new; however, the proposed change will require timely tripping of the RCPs, i.e., the action will become time-critical. In order to protect the AREVA fuel, the action must be performed in four minutes or less, once subcooling margin falls to < 20 degrees F. Response to a small break LOCA scenario was identified by the licensee as the worst case because reactor pressure will trend down slowly, extending the time required to reach the safety injection tank pressure and recover core two-phase level with HPSI flow. This scenario "produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b)(1-4) criteria."

Operating Experience Review

The licensee stated, "Although similar AREVA fuel designs are licensed for other Combustion Engineering plants, there is not (sic) precedent that covers all aspects of the fuel design proposed for this request." Regarding M-5 cladding, the licensee identified an SE for Fort Calhoun dated August 30, 2006 (ML0617105380), as a precedent approving an M-5 cladding exemption. No precedent for the RCP trip response time reduction or any other human performance issue was identified. Based on a search of the ADAMS database for issues with timely tripping of RCPs, including license amendment requests, SEs, and licensee event reports over the last five years, the NRC staff agrees with the licensee's assessment and finds the licensee's position on operating experience acceptable

Functional Requirements Analysis and Function Allocation

Because this operator action is not a new action, a re-analysis of the functional requirements analysis and function allocation was not necessary. If the licensee's engineering analysis had shown that the required tasks could not be done within the time constraints established, the NRC staff would have expected a reallocation of this function from the operator to an automatic system. However, this was not the case, and so, there was no need for either a new or revised functional requirements analysis or a reallocation of function. The staff finds the CCNPP approach acceptable based on their validation of the action as feasible and reliable.

Task Analysis

Because this operator action is not a new action, the only aspect requiring reanalysis was the establishment of time constraints for the action sequence. The licensee established the design values for the time to trip all RCPs in accordance with the S-RELAP5 system analysis. The design values for the timing of the action sequence were later validated (see Human Factors Verification and Validation, below). The design value established for the operator action to trip all RCPs is 4 minutes after loss of 20 degrees of subcooling margin. The simulator testing using a small break LOCA scenario demonstrated sufficient margin to these design times. Specifically, the testing determined that actual operator response times are 2 minutes and ten seconds or less (Supplement dated December 30, 2010, ML110040374). The NRC staff finds the licensee revision of task success criteria acceptable.

Staffing

The licensee found that staffing and qualification are not affected by the proposed license amendment request (LAR). No new or additional staff is required, nor are there any new or additional qualifications required to perform the action sequence within the time constraints established. The NRC staff agrees that no additional staffing or qualifications, or changes thereto, are required, and finds this human performance aspect of the LAR acceptable.

Human-System Interface Design

The licensee stated in its January 28, 2011 submittal (ML1103202430) that Human-System Interface (HSI) design of the control room and the simulator, including the design of the Safety Parameter Display System (SPDS) will not be affected by the proposed LAR. The same controls, displays, and alarms that have been successfully used in the past will continue to be used under the proposed LAR. Based on the fact that no changes are needed to the HSI design, the NRC staff finds the CCNPP proposal acceptable.

Procedure Design

In its submittal of November 19, 2010 (ML1032800820), the licensee stated that the only changes to the procedures are those being made to EOP-0, "Post Trip Immediate Actions". The

EOP will be revised to add the following criterion for tripping the RCPs: "IF RCS subcooling drops below 20 degrees F, THEN trip ALL RCPs."

In addition, the EOP basis documents will be revised to highlight the step as time-critical. The NRC staff finds these changes acceptable based on successful operating experience during past training with the existing EOPs, and on the time-testing that was done in the CCNPP simulator using the revised procedures and a sample of CCNPP operators (documented in the licensee's December 30, 2010 and January 14, 2011 submittals, ML1100403740 and ML1101806210 respectively).

Training Program Design

Because the EOPs are an integral part of the licensed operator qualification and requalification training programs, training on the proposed action sequence will be included in both initial and continuing operator training. The licensee determined that the simulator is capable of modeling the task sequence and will, therefore, be used in training. Training on the time-critical aspect of the task sequence will be completed prior to amending the TS. The licensee has established a job/simulator performance measure for this action including a pass/fail criterion. This will ensure that all CCNPP operators will have been time-tested during training to further validate the operator response times. Based on the facts that the revised action sequence will be included in the training program and that the training changes will be implemented prior to amending the TS, the NRC staff finds that the training to be provided is acceptable.

Human Factors Verification and Validation

Per the licensee's December 30, 2010, supplement to the LAR, time testing at the CCNPP simulator was performed to demonstrate sufficient margin to the licensee-established design values. The simulator testing, using a small break LOCA scenario, demonstrated that a value of two minutes and ten seconds could be reasonably established as the maximum time after subcooling margin falls to less than 20 degrees F. to trip all RCPs. Based on this testing, the NRC staff agrees that there is sufficient margin to the design value of four minutes for this task.

Human Performance Monitoring Strategy

The actions proposed by this LAR, tripping RCPs when subcooling margin falls below 20 degrees F., will be included in the licensee's "Time Critical Action (TCA) Program", which provides a means to: a) ensure that the time-critical actions within the scope of the procedure can be accomplished by plant personnel, b) document periodic validation of credited action times, and c) ensure that subsequent changes to the plant, procedures, or programs will not invalidate the credited action times (January 14, 2011 submittal, ML1101806210). Based on the administrative protection against inadvertent change and the periodic re-validation provided by the licensee, the NRC staff finds the CCNPP long-term monitoring strategy acceptable.

5.5.3 Conclusion

Based on the evidence provided by CCNPP, i.e., that pilot-testing in the simulator demonstrated sufficient margin to design, as well as the appropriate administrative controls being applied to procedures, training, and human interface design, the NRC staff concludes that the proposed LAR is acceptable from the human performance point of view.

5.6 Radiological Consequences

The NRC staff reviewed the impact of the proposed transition to AREVA fuel on previously analyzed design basis accident (DBA) radiological consequences.

5.6.1 Regulatory Evaluation

The regulatory requirements and guidance documents on which the staff based its review include:

- Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.67, "Accident source term";
- 10 CFR Part 50 Appendix A, General Design Criterion 19 (GDC-19), "Control Room";
- Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000;
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000; and
- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Rev. 3, March 2007.

The NRC staff evaluated the licensee's proposed change against the requirements specified in 10 CFR 50.67(b)(2). 10 CFR 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE).
- An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE.
- Adequate radiation protection is provided to permit access to and occupancy of the control room (CR) under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

The regulatory requirements from which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, and the accident specific guideline values in Regulatory Position 4.4 of RG 1.183 and Table 1 of Standard Review Plan (SRP) Section 15.0.1. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

5.6.2 Technical Evaluation

By letter dated August 29, 2007 (ML072130521), the NRC approved a license amendment to fully implement an AST methodology at CCNPP. To support this license amendment, an analysis was done by the licensee to determine CCNPP's bounding source term, which is used for the radiological DBAs that assume fuel failures. This analysis was performed based on the release fractions given in RG 1.183. The core isotopic activity was calculated using SAS2H/ORIGEN-S, which is a NRC-approved computer code. In that analysis, the core inventory for each core power distribution scenario analyzed was used to calculate an effective TEDE dose and dose rate, which was then compared against a reference TEDE dose and dose rate calculated for the default pressurized water reactor core inventory file supplied with the RADTRAD 3.03 code, which is a NRC-approved code. In the supplement letter dated January 18, 2011, the licensee stated that the bounding case in the currently approved AST was determined to yield doses and dose rates that were a factor of 1.7222 and 1.3416 greater, respectively, than those calculated using the default RADTRAD pressurized water reactor inventory file.

A modification to the licensing basis fuel type can have the potential to change the core isotopic activity distribution assumed for post-accident conditions. In the supplement letter dated, July 23, 2010, the licensee stated that an additional analysis was performed to demonstrate the impact of AREVA fuel on the AST due to the transition from the Westinghouse fuel. The same methodology and core power distribution scenarios were used to evaluate the AREVA fuel design as discussed in the above paragraph. In the supplement letter dated, January 18, 2011, the licensee stated that the bounding case for the AREVA fuel was found to be the same core power distribution as for the bounding Westinghouse case, and yielded doses and dose rates that were factors of 1.7038 and 1.3254 greater, respectively, than those for the default RADTRAD pressurized water reactor inventory file.

The NRC staff has reviewed the licensee's analysis that demonstrated the impact of AREVA fuel on the AST source term. The staff finds that the licensee used approved methodology consistent with the CCNPP current licensing basis and NRC regulatory guidance. The staff finds that the current Westinghouse fuel-based source term input for the radiological design basis accidents does not need to be updated since it yielded greater doses than an AREVA fuel-based source term. Therefore, CCNPP DBA regulatory dose limits are unaffected and still meet the regulatory requirement in 10 CFR 50.67.

5.6.3 Conclusion

The NRC staff reviewed the analysis used by the licensee to assess the radiological impacts of the transition from Westinghouse fuel to AREVA fuel at CCNPP. The staff finds that the licensee used methods consistent with regulatory requirements and guidance identified above. The staff finds, with reasonable assurance that the licensee's estimates of the exclusion area boundary, low-population zone, and control room doses will continue to comply with these criteria. Therefore, the proposed change is acceptable with regard to the radiological consequences of postulated DBAs.

5.7 Conclusion

Based on its review, and on acceptable comparison between the CCNPP licensing basis and the analytic assumptions listed in Tables 3-2 and 3-3 of ANP-2834(P), the NRC staff finds the requested implementation of the AREVA RLBLOCA methodology acceptable.

The ECCS evaluation was performed using an acceptable evaluation model. The PCT result is 1670ºF, which falls within the acceptance criterion of 2200ºF. The oxidation results also meet the 10 CFR 50.46 acceptance criteria with significant margin.

Although the NRC staff has noted numerous issues with the acceptable evaluation model, the licensee has deviated from the approved model to compensate for a majority of those issues. Others, as discussed in the preceding sections of this SE, are either insignificant in light of the CCNPP-specific results, or are acceptable for ongoing evaluation in accordance with 10 CFR 50.46(a)(3) ECCS evaluation model change and error evaluation and reporting requirements. This model change/error evaluation requirement is further described in NRC IN 1997-015, "Reporting of Errors and Changes in Large-Break Loss-of-Coolant Accident Evaluation Models of Fuel Vendors And Compliance With 10 CFR 50.46a(3)," and IN 1997-015, Supplement 1.

The NRC collects and trends ECCS performance analysis data. Surveying these data, which includes the results of numerous CE-designed ECCS analyses performed using a variety of LOCA evaluation models, the NRC staff did not identify any results to suggest that CCNPP would exceed the 10 CFR 50.46(b) acceptance criteria, had either (1) the evaluation model been corrected to address all of the issues identified by the NRC staff, or (2) the ECCS been analyzed using a different, more acceptable ECCS evaluation model.

Based on the pilot-testing in the simulator that demonstrated sufficient margin to design, as well as the appropriate administrative controls being applied to procedures, training, and human interface design, the NRC staff concludes that the proposed LAR is acceptable from the human performance point of view.

The NRC staff reviewed the analysis used by the licensee to assess the radiological impacts of the transition from Westinghouse fuel to AREVA fuel at CCNPP. The staff finds that the licensee used methods consistent with regulatory requirements and guidance. The staff finds, with reasonable assurance that the licensee's estimates of the exclusion area boundary, low-population zone, and control room doses will continue to comply with these criteria. Therefore, the proposed change is acceptable with regard to the radiological consequences of postulated DBAs.

Based on the considerations discussed above, the NRC accepts the licensee's ECCS evaluation of the large break loss of coolant accident in accordance with the transition package of EMF-2103(P)(A), Revision 0, subject to License Condition 10, discussed above.

6.0 TS Changes

In support of the fuel transition, Constellation has proposed changes to the following TSs:

- TS SL 2.1.1.2, Reactor Core Safety Limits
- TS LCO 3.2.2, Total Planar Radial Peaking Factor (FTxy)
- TS LCO 3.1.8, Special Test Exception (STE) – MODES 1 and 2
- TS SR 3.2.1.1 Linear Heat Rate
- TS SR 3.2.2.1 – Total Planar Radial Peaking Factor
- TS LCO 3.2.4, Azimuthal Power Tilt (Tq)
- TS Design Feature 4.2.1, Fuel Assemblies
- TS Reporting Requirements 5.6.5, Core Operating Limits Report (COLR)

The basis for each proposed TS change is discussed below.

TS SL 2.1.1.2, Reactor Core Safety Limits

The purpose of this proposed change to the safety limit is as follows:

1. In accordance with Technical Specification Task Force (TSTF)-445-A, replace the steady-state peak LHGR threshold corresponding to incipient fuel centerline melting to an actual fuel centerline temperature threshold corresponding to incipient fuel melting, and

2. Specifically account for burnup effects and burnable poison effects on the fuel centerline temperature corresponding to incipient fuel melting.

Two fuel centerline temperature limits are proposed, one for the AREVA CE14HTP fuel design and the other for the co-resident Westinghouse TURBO fuel design. In both equations, the fuel centerline temperature limit decreases from a maximum of 5080°F (Westinghouse) and 5081°F (AREVA) with local burnup and burnable poison content. The proposed safety limit refers to proprietary topical reports for the specific adjustment for the burnable poison.

Future changes in fuel pellet design or fuel pellet composition (e.g., fuel additives, uranium-plutonium mixed oxide, change in burnable poison) may invalidate the equations provided in this proposed SL.

Based on the justification provided in Attachment 1 of Reference 1, the NRC staff finds the proposed change to TS SL 2.1.1.2, Reactor Core Safety Limits, acceptable.

TS LCO 3.2.2, Total Planar Radial Peaking Factor (F^T_{xy})

In accordance with 10 CFR 50.36(c)(2), LCO 3.2.2 limits the total planar radial peaking factor (i.e., limits the core power distribution) to the initial values assumed in accident analyses. The approved AREVA methodologies do not use total planar radial peaking factor as an initial value in the accident analyses. As such, the proposed change removes not only LCO 3.2.2 in its entirety, but also reference to this core power distribution limit in LCO 3.1.8, SR 3.2.1.1, SR 3.2.2.1, and LCO 3.2.4. The basis for removing reference to the planar radial peaking factor is described in Section 2, Attachment 1 of Reference 1.

In response to an RAI regarding the removal of SR 3.2.1.1, the licensee stated that the surveillance on LCO 3.2.1, Linear Heat Rate (LHR), using total planar radial peaking factor (when using excore detectors) was inconsistent with the approved AREVA methodology[67]. Further, this surveillance requirement was not included in the Standard Improved Technical Specifications for Combustion Engineering Plants (NUREG-1432). The licensee also noted that other CE plants have removed this surveillance requirement.

In response to an RAI reading the technical basis for surveillance of LCO 3.2.1 in the absence of total planar radial peaking factor, the licensee stated that, when using excore detectors, this surveillance would be performed by monitoring axial shape index (ASI) alarms[68]. Specifically, the PRISM three dimensional neutronics code is used to calculate limiting LHR peaking factors based upon core power maneuvers that bound the allowed power versus peripheral ASI operating space. Provided the plant maneuvers within the peripheral ASI operating band, the cycle-specific setpoint analysis ensures that LCO 3.2.1 is satisfied.

Based upon the information provided in Section 2, Attachment 1 of Reference 1 and in response to RAIs, the NRC staff finds the proposed changes to LCO 3.2.2, as well as the related changes to LCO 3.1.8, SR 3.2.1.1, SR 3.2.2.1, and LCO 3.2.4, acceptable.

TS Design Feature 4.2.1, Fuel Assemblies

The proposed revision to TS 4.2.1 accomplishes two goals; (1) removes the description of a prior lead test assembly program at CCNPP and (2) adds M5 alloy cladding material to the description of the fuel design. While AREVA's M5 alloy is an NRC approved cladding material, it requires an exemption to 10 CFR 50.46 and 10 CFR 50 Appendix K since it is outside of the definition (alloy composition) of the two cladding materials specifically identified in the regulation, Zircaloy or ZIRLO. CCNPP was granted this exemption by letter dated January 13, 2011 (ML103070113).

The NRC staff has reviewed the proposed change to TS 4.2.1, Fuel Assemblies, and finds it acceptable.

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[67] Reference 5, op. cit., RAI 24. ML103280082.
[68] Reference 6, op. cit., RAI 16. ML110040374.

TS Reporting Requirements 5.6.5, Core Operating Limits Report (COLR)

The licensee proposed to eliminate Line Item 3.2.2, "Total Planar Radial Peaking Factor," from the TS. This elimination was to bring TS 5.6.5.a in alignment with the new TS Surveillance Requirements, LCOs, and core operating limits, which eliminate limitations to and surveillances for the total planar radial peaking factor (F^T_{xy}), on the basis that this parameter is not used in the AREVA safety analysis methodology. The NRC staff finds this proposed change acceptable because it makes the COLR section of the TSs consistent with the proposed TSs. The technical basis is discussed in further detail in the subsection entitled "TS LCO 3.2.2, Total Planar Radial Peaking Factor (F^T_{xh})," above.

The licensee proposed to add the following list of references to TS 5.6.5.b, "Core Operating Limits Report – References":

1. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnup of 62 GWd/MTU"

2. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods"

3. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs"

4. EMF-92-153(P)(A), "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel"

5. EMF-96-029(P)(A), "Reactor Analysis System for PWRs"

6. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"

7. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors"

8. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

9. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"

10. XN-NF-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

11. XN-NF-78-44A, "Generic Analysis of the Control Rod Ejection Transient for PWRs"

12. XN-NF-79-56(P)(A), "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation"

13. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

14. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"

15. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

Existing COLR references were renumbered if still necessary to generate core operating limits, or deleted as necessary. The licensee performed a review of the existing references and proposed the deletion of unused references and retained those which would still be used to generate core operating limits after the proposed fuel transition.

The use of these methodologies is evaluated in the Sections 2-5 of this safety evaluation. On that basis, and consistent with the License Conditions proposed by the licensee and incorporated herein, the NRC staff finds the implementation of this list of references into TS 5.6.5.b acceptable.

7.0 License Conditions

As previously discussed throughout this SE, the NRC staff has concluded that a number of license conditions are necessary to capture the more restrictive criteria for CCNPP reload designs. Furthermore, the licensee has agreed with these license conditions. Many of the license conditions are cycle-specific for each operating unit at CCNPP. They also have implementation dates that, by necessity, are linked to future refueling outages. For the purpose of clarity, the cycle-specific aspects of the license conditions were not previously addressed in this SE and those license conditions were presented as summaries. The actual, unit-specific license conditions that were proposed and found acceptable by the NRC staff are stated below. Finally, as previously described in this SE, License Conditions 1 and 2 have been combined for clarity.

Calvert Cliffs Unit No, 1

License Conditions 1 and 2:

For the RODEX2-based fuel thermal-mechanical design analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.3, Calvert Cliffs Unit 1 core reload designs (starting with Cycle 21) shall satisfy the following criteria:

a. Predicted rod internal pressure shall remain below the steady state system pressure.

b. The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 KW/ft.

License Condition 3:

Approval of the use of S-RELAP5 (Technical Specification 5.6.5.b.8) is restricted to only those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits. Prior transient specific NRC approval is required to analyze transient performance relative to reactor coolant pressure boundary integrity until NRC-approval is obtained for a generic or Calvert Cliffs-specific basis for the use of the methodology in Technical Specification 5.6.5.b.8 to demonstrate reactor coolant pressure boundary integrity.

License Condition 4:

Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 shall not be changed without prior NRC review and approval until an NRC-accepted generic, or Calvert Cliffs-specific, basis is developed for analyzing the Control Element Assembly Rod Bank Withdrawal Event, the Control Element Assembly Drop, and the Control Element Assembly Ejection (power level-sensitive transients) at full power conditions only.

License Condition 5:

For the Seized Rotor Event analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet flow distribution. The revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 21.

License Condition 6:

For the Asymmetric Steam Generator Transient analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet temperature distribution and application of local peaking augmentation factors. The revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 21.

License Condition 7:

For the Control Element Assembly Ejection analysis, Calvert Cliffs Unit 1 core reloads (starting with Cycle 21) shall satisfy the following criteria:

a. Predicted peak radial average fuel enthalpy when calculated in accordance with the methodology of Technical Specification 5.6.5.b.11 shall remain below 200 cal/g.

b. For the purpose of evaluating radiological consequences, should the S-RELAP5 hot spot model predict fuel temperature above incipient centerline melt conditions when calculated in accordance with the methodology of Technical Specification 5.6.5.b.8, a conservative radiological source term (in accordance with RG 1.183, Revision 0) shall be applied to the portion of fuel beyond incipient melt conditions (and combined with existing gap source term), and cladding failure shall be presumed.

License Condition 8:

For the Control Element Assembly Ejection analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.11, the cycle-specific hot zero power peak average radial fuel enthalpy is calculated based on a modified power dependent insertion limit with Control Element Assembly Bank 3 assumed to be fully inserted (only in the analysis, not in actual plant operations). This revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 21.

License Condition 9:

The small break loss of coolant accident performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to Calvert Cliffs Unit 1 core reload designs starting with Cycle 21.

License Condition 10:

The approval of the emergency core cooling system evaluation performed in accordance with the methodology of Technical Specification 5.6.5.b.7 shall be valid only for Calvert Cliffs Unit 1, Cycle 21. To remove this condition, Calvert Cliffs shall obtain NRC approval of the analysis of once- and twice-burned fuel for core designs following Unit 1 Cycle 21.

Implementation Date (For all the above Unit 1 license conditions)

This amendment is effective immediately and shall be implemented within 60 days of completion of the Unit 1 2012 refueling outage.

Calvert Cliffs Unit No. 2

License Conditions 1 and 2

For the RODEX2-based fuel thermal-mechanical design analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.3, Calvert Cliffs Unit 2 core reload designs (starting with Cycle 19) shall satisfy the following criteria:

a. Predicted rod internal pressure shall remain below the steady state system pressure.
b. The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 KW/ft.

License Condition 3

Approval of the use of S-RELAP5 (Technical Specification 5.6.5.b.8) is restricted to only those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits. Prior transient specific NRC approval is required to analyze transient performance relative to reactor coolant pressure boundary integrity until NRC-approval is obtained for a generic or Calvert Cliffs-specific basis for the use of the methodology in Technical Specification 5.6.5.b.8 to demonstrate reactor coolant pressure boundary integrity.

<u>License Condition 4</u>

Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 shall not be changed without prior NRC review and approval until an NRC-accepted generic, or Calvert Cliffs-specific, basis is developed for analyzing the Control Element Assembly Rod Bank Withdrawal Event, the Control Element Assembly Drop, and the Control Element Assembly Ejection (power level-sensitive transients) at full power conditions only.

<u>License Condition 5</u>

For the Seized Rotor Event analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet flow distribution. The revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

<u>License Condition 6</u>

For the Asymmetric Steam Generator Transient analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet temperature distribution and application of local peaking augmentation factors. The revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

<u>License Condition 7</u>

For the Control Element Assembly Ejection analysis, Calvert Cliffs Unit 2 core reloads (starting with Cycle 19) shall satisfy the following criteria:

a. Predicted peak radial average fuel enthalpy when calculated in accordance with the methodology of Technical Specification 5.6.5.b.11 shall remain below 200 cal/g.
b. For the purpose of evaluating radiological consequences, should the S-RELAP5 hot spot model predict fuel temperature above incipient centerline melt conditions when calculated in accordance with the methodology of Technical Specification 5.6.5.b.8, a conservative radiological source term (in accordance with RG 1.183, Revision 0) shall be applied to the portion of fuel beyond incipient melt conditions (and combined with existing gap source term), and cladding failure shall be presumed.

<u>License Condition 8</u>

For the Control Element Assembly Ejection analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.11, the cycle-specific hot zero power peak average radial fuel enthalpy is calculated based on a modified power dependent insertion limit with Control Element Assembly Bank 3 assumed to be fully inserted (only in the analysis, not in actual plant operations). This revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

License Condition 9

The small break loss of coolant accident performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

License Condition 10

The approval of the emergency core cooling system evaluation performed in accordance with the methodology of Technical Specification 5.6.5.b.7 shall be valid only for Calvert Cliffs Unit 2, Cycle 19. To remove this condition, Calvert Cliffs shall obtain NRC approval of the analysis of once- and twice-burned fuel for core designs following Unit 2 Cycle 19.

Implementation Date (For all the above Unit 2 license conditions)

This amendment is effective immediately and shall be implemented within 60 days of issuance.

Correction to Previous Amendments to Appendix C to CCNPP

The NRC staff has recently recognized an existing error in CCNPP Appendix C. By letter dated October 30, 2009 (ML092880805), the NRC staff issued conforming license amendments (i.e., Amendment Nos. 295 and 271 for CCNPP Unit Nos. 1 and 2, respectively) reflecting a corporate restructuring and indirect license transfers. As part of those license amendments, license conditions were introduced regarding decommissioning funding assurance. In those license amendments, the staff issued new pages to Appendix C that inadvertently deleted previous license conditions associated with Amendment Nos. 287 and 264 regarding control room habitability. As a result, the staff is correcting this error by reissuing the appropriate pages to the CCNPP Appendix C.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Maryland State official was notified of the proposed issuance of the amendments. The State official had no comments.

9.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 [and changes Surveillance Requirements]. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (75 FR 23810). Accordingly, the amendments meet 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 amendments.

10.0 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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