November 3, 2006

Mr. R. T. Ridenoure
Vice President - Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: USE OF AREVA NP, INC. REALISTIC LARGE BREAK LOSS-OF-COOLANT ACCIDENT METHODOLOGY (TAC NO. MC8946)

Dear Mr. Ridenoure:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 245 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1 (FCS). The amendment consists of changes to the Core Operating Limit Report in response to your application dated September 30, 2005, as supplemented by letters dated May 23 and August 16, 2006.

The amendment changes the FCS methodology for large break (LB) loss-of-coolant accident (LOCA) analyses. The NRC staff has approved the use by Omaha Public Power District of the AREVA NP, Inc. (formerly Framatome ANP) best estimate LBLOCA methodology described in EMF-2103 (P)(A),"Realistic Large Break LOCA Methodology," dated April 2003, Revision 0, at the FCS. This topical report will replace EMF-2087(P)(A), Revision 0, "SEM/PWR-98: ECCS [Emergency Core Cooling System] Evaluation Model for PWR [Pressurized-Water Reactor] LBLOCA Applications," in the FCS Core Operating Limits Report. This change is necessary because the EMF-2087(P)(A) methodology is not approved for analyzing M5 clad fuel, which will be used in the FCS reactor core starting in Cycle 24.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/
Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 245 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 245
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee), dated September 30, 2005, as supplemented by letters dated May 23 and August 16, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, by Amendment No. 245 , changes to the Core Operating Limits Report reflect the approved use by Omaha Public Power District of the AREVA NP, Inc. (formerly Framatome ANP) best estimate large break loss-of-coolant accident methodology described in EMF-2103(P)(A), “Realistic Large Break LOCA Methodology,” dated April 2003, Revision 0, at the Fort Calhoun Station, Unit No. 1, as authorized.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: November 3, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 245 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated September 30, 2005, as supplemented by letters dated May 23 and August 16, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052770174, ML061460190, and ML062290086, respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Core Operating Limits Report for the Fort Calhoun Station, Unit No. 1 (FCS).

The proposed amendment changes the FCS methodology for large break (LB) loss-of-coolant accident (LOCA) analyses. Specifically, the proposed change approves the use of the AREVA NP, Inc. (formerly Framatome ANP) best estimate (BE) LBLOCA methodology described in AREVA NP, Inc. topical report EMF-2103(P)(A), "Realistic Large Break LOCA Methodology," dated April 2003, Revision 0 (Reference 3), by OPPD for use at FCS. This topical report will replace EMF-2087(P)(A), Revision 0, "SEM/PWR-98: ECCS [Emergency Core Cooling System] Evaluation Model for PWR [Pressurized-Water Reactor] LBLOCA Applications," in the FCS Core Operating Limits Report. This change is necessary because EMF-2087(P)(A) methodology is not approved for analyzing M5 clad fuel, which will be used in the FCS reactor core starting in Cycle 24. As part of this review, the NRC staff reviewed the licensee's evaluations of the ECCS performance analyses for FCS, which were done in accordance with the EMF-2103(P)(A), "Realistic Large Break LOCA Methodology," dated April 2003, Revision 0.

The supplemental letters dated May 23 and August 16, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2006 (71 FR 152).

2.0 REGULATORY EVALUATION

The applicable regulations and guidance for this review are the following:

Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specifies requirements for the acceptability of the emergency core cooling system. Paragraphs 50.46(a)(1)(i) and 50.46(a)(1)(ii) of 10 CFR specify alternative approaches to show compliance with the acceptance criteria of 10 CFR 50.46(b). Part 50 of 10 CFR, Appendix K, provides requirements for calculating whether those acceptance criteria are satisfied. A licensee's compliance with these criteria demonstrates the acceptability, following a LOCA, of (1) the peak calculated cladding temperature, (2) the maximum cladding oxidation, (3) the maximum hydrogen generation, and (4) the capability to maintain a coolable geometry, and (5) the capability to maintain long-term core cooling. The LBLOCA analyses were performed by AREVA NP, Inc. for OPPD to demonstrate that the FCS system design would provide sufficient emergency core cooling system (ECCS) flow to transfer the heat from the reactor core following a LOCA, 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 not compromise cladding ductility and would not result in excessive hydrogen generation. The Nuclear Regulatory Commission (NRC) staff reviewed the analyses to ensure that the analyses reflected suitable redundancy in components and features; and that suitable interconnections, leak detection, isolation, and containment capabilities are available such that the safety functions could be accomplished, assuming a single failure, for LOCAs, considering the availability of onsite power (i.e., either 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). As noted above, the acceptance criteria for ECCS performance are provided in Section 50.46 of 10 CFR, and were used by the NRC staff in assessing the acceptability of the EMF-2103(P)(A), Revision 0, methodology (Reference 3) for use at the FCS.

The licensee has proposed to comply with 10 CFR 50.46(a)(1)(i), which requires modeling the realistic behavior of the reactor system following a LOCA and estimating and accounting for the uncertainty in the calculated results. Previous calculations were performed under 10 CFR 50.46(a)(1)(ii).

An important aspect of the LOCA calculations is estimating the containment pressure during the LOCA. The containment pressure has a significant effect on the rate at which the core is reflooded by the ECCS and, therefore, on the five criteria of 10 CFR 50.46(b). The licensee states the following:

> For the FANP [Framatome ANP] realistic LBLOCA evaluation model, significant containment parameters, as well as NSSS [nuclear steam supply system] parameters, were established via a PIRT process. Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to [peak cladding temperature]) containment parameters—containment pressure and temperature.

Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989, provides guidance on methods acceptable to the NRC staff for realistic or best-estimate calculations of ECCS performance during a LOCA. In particular, Section 3.12.1, "Containment Pressure," states the following:

> The containment pressure used for evaluating cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of all pressure-reducing equipment

> assumed to be available. Best-estimate models will be considered to be acceptable provided their technical basis is demonstrated with appropriate data and analyses.

Appendix K to 10 CFR Part 50, Section I.D.2, Containment Pressure, requires that "[t]he containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure reducing systems and processes."

Technical Branch Position CSB 6-1, "Minimum Containment Pressure Model for PWR [Pressurized-Water Reactor] ECCS Performance Evaluation," of NUREG-0800, the Standard Review Plan, provides guidance for complying with Appendix K, Section I.D.2.

The licensee has proposed a combination of the above two approaches for calculating the containment pressure during a LOCA- i.e., using both the statistical calculation and the requirements of 10 CFR Part 50, Appendix K, and the guidance of CSB 6-1. In this approach, some of the input parameters are treated statistically, while most are chosen in a manner which complies with the guidance of Technical Branch Position CSB 6-1 so as to underestimate the containment pressure. As explained below, the NRC staff finds this approach to be acceptable since it complies with 10 CFR 50.46 and results in an overall conservative calculation of containment pressure relative to the guidance of Regulatory Guide 1.157.

As part of its assessment of the acceptability of the methodology for the FCS, the NRC staff also reviewed the limitations and conditions stated in its previously issued safety evaluation (SE) supporting approval of the methodology and the range of parameters described in the topical report EMF-2103(P)(A), Revision 0. Furthermore, on April 1, 2004, the NRC approved a license amendment for the Virginia Electric and Power Company for a similar request referencing EMF-2103(P)(A), Revision 0, methodology for the North Anna Power Station, Unit 2, LBLOCA analysis.

3.0 TECHNICAL EVALUATION

FCS is a PWR of the Combustion Engineering design, enclosed within a large, dry containment. The ECCS consists of low-pressure safety injection (LPSI) flow and high-head safety injection (HHSI) flow delivered to the cold legs, and two accumulators with a cover gas pressure of 271.7 $\pm$ 17.5 pounds per square inch absolute (psia), also injecting into the cold legs. The shut-off head of the high-pressure safety injection (HPSI) pumps is about 1250 psia.

The NRC staff reviewed the licensee's evaluations of the ECCS performance analyses for FCS, which were done in accordance with the EMF-2103(P)(A), "Realistic Large Break LOCA Methodology," dated April 2003, Revision 0. The licensee's analysis used a core power of 102 percent of the licensed core power of 1,525 mega-watts thermal (MWt) or 1,555.5 MWt. The licensee conducted its LOCA analyses assuming the use of AREVA NP, Inc. M5 fuel assemblies at FCS. This change is necessary because EMF-2087(P)(A) methodology is not approved for analyzing M5 clad fuel, which will be used in the FCS reactor core starting in Cycle 24.

3.1 LBLOCA Analysis

As explained above, the LBLOCA analyses were performed by AREVA NP, Inc. for OPPD to demonstrate that the FCS system design would provide sufficient ECCS flow to transfer the heat from the reactor core following a LOCA, 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 not compromise cladding ductility and would not result in excessive hydrogen generation. The Nuclear Regulatory Commission (NRC) staff reviewed the analyses to ensure that the analyses reflected suitable redundancy in components and features; and that suitable interconnections, leak detection, isolation, and containment capabilities are available such that the safety functions could be accomplished, assuming a single failure, for LOCAs, considering the availability of onsite power (i.e., either 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). As noted above, the acceptance criteria for ECCS performance are provided in Section 50.46 of 10 CFR, and were used by the NRC staff in assessing the acceptability of the EMF-2103(P)(A), Revision 0, methodology (Reference 3) for use at the FCS.

In its May 23, 2006, submittal (Reference 2), the licensee stated, "OPPD and its LBLOCA analysis vendor (AREVA NP, Inc.) have ongoing processes that assure that the values and ranges of parameters for the FCS LBLOCA analyses conservatively bound the values and ranges of those parameters for the as-operated plant parameters. . ." The NRC staff finds that this statement, combined with the generic acceptance of use of EMF-2103(P)(A), Revision 0, methodology, provides assurance that the LBLOCA analyses performed using that methodology can be applied to the FCS operating at the current licensed power level of 1,525 MWt.

In its submittal, the licensee provided the results for the FCS BE LBLOCA analyses at 1,555.5 MWt (about 102 percent of the operating power of 1,525 MWt) performed in accordance with the EMF-2103(P)(A), Revision 0, methodology. The licensee's results for the calculated peak cladding temperatures (PCTs), the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 1: LARGE BREAK LOCA ANALYSIS RESULTS

Parameter	FCS EMF-2103 Results M5	10 CFR 50.46 Limits
Limiting Break Size/Location	DEG/PD	N/A
Cladding Material	M5	(Cylindrical) Zircaloy or Zirlo
Peak Clad Temperature	1695° F	2200° F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	0.82% *	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	0.02 % *	1.0% (10 CFR 50.46(b)(3))

DEG/PD is a double-ended guillotine break at the pump discharge.
*see Section 3.2.4, below.

In analyses for the FCS, the licensee considered the concern that present fuel (M5) may have pre-existing oxidation that must be considered in the LOCA analyses. In the review of this issue, AREVA NP, Inc. on behalf of its licensee responded to NRC staff's requests for additional information by stating that it considered in the analyses that the M5 clad fuel has both pre-existing oxidation and oxidation resulting from the LOCA (pre- and post-LOCA oxidation both on the inside and outside cladding surfaces). AREVA NP, Inc. and its licensee have also noted that the fuel with the highest LOCA oxidation will likely not be the same fuel that has the highest pre-LOCA oxidation. AREVA NP, Inc. and the licensee indicated that when the calculated pre-LOCA oxidation was factored into the licensee's BE LBLOCA analyses for the fuel consistent with the previous LOCA methodology for FCS, the sum of the calculated pre- and post-LOCA oxidation was sufficiently small - even during a fuel pin's final cycle in the core - that the total local oxidation remained less than the 17 percent acceptance criterion of 10 CFR 50.46(b)(2) as noted above. The NRC staff finds that this analysis appropriately addresses the issue of pre-LOCA oxidation because the FCS fuel is AREVA NP, Inc. designed, and the computer code (RELAP) used in the previous methodology is the same code used in the EMF-2103 (P)(A), Revision 0, methodology. The NRC staff also considered that the calculated LOCA oxidation is sufficiently low (less than 1 percent) that the pre-accident oxidation would have to be extremely high (greater than 16 percent) for any power-producing rod in the core to exceed the 10 CFR 50.46(b)(2) total oxidation limit of 17 percent.

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 reactor coolant system throughout the operating cycle, the NRC staff notes that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA and, therefore, it does not need to be addressed when determining whether the calculated total core-wide oxidation meets the 1.0 percent criterion of 10 CFR 50.46(b)(3).

As discussed previously, OPPD requested that AREVA NP, Inc. conduct the BE LBLOCA analyses for FCS at about 102 percent of the current licensed power level of 1,525 MWt using EMF-2103(P)(A), Revision 0, methodology that was previously approved for the North Anna Power Station, Unit 2. The NRC staff concluded that the results of these analyses (see Table 1) demonstrated compliance with 10 CFR 50.46(b)(1) through (b)(3) for licensed power levels of up to 1,525 MWt. Meeting these criteria provides reasonable assurance that at the current licensed power level the FCS core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). The NRC staff finds that the capability of FCS to satisfy the long-term cooling requirements of 10 CFR 50.46(b)(5) is assured by satisfaction of 10 CFR 50.46 (b)(1) through (b)(4) and the approved ECCS design.

3.2 Containment Analysis

The containment pressure is used for evaluating the cooling effectiveness during the post-blowdown phase of an LOCA. The containment pressure should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of all pressure-reducing equipment assumed to be available. The NRC will consider best-estimate models to be acceptable provided their technical basis is demonstrated with appropriate data and analyses.

AREVA NP, Inc. Topical Report EMF-2103(P)(A) describes AREVA NP, Inc.'s methodology for performing containment pressure calculations are performed. The S-RELAP5 computer code, which performs the LOCA calculations, incorporates a containment analysis computer code called ICECON, which was originally developed by Exxon Nuclear Company, Inc. (Reference

4). ICECON is based on the CONTEMPT/LT-022 computer code (References 5 and 6). The NRC approved ICECON in a safety evaluation, dated June 30, 1978, which stated that "[t]he NRC staff [previously] reviewed the Exxon ECCS evaluation model and published a safety evaluation dated September 11, 1975. We concluded that Exxon's containment pressure model for dry containments was acceptable for ECCS evaluation."

The June 30, 1978, safety evaluation approved the extension of this model to ice condenser containments as the ICECON code. The ICECON topical report states that ICECON can be used to model PWR dry, ice condenser, and subatmospheric containments. FCS has a large, dry containment.

ICECON compared well against the GOTHIC 6.1 (References 7 and 8) containment computer code for a Westinghouse three-loop reactor with a large, dry containment. GOTHIC 6.1 is a containment analysis computer code developed by Numerical Applications, Inc., for the Electric Power Research Institute and has undergone extensive validation against data and analytic solutions.

The licensee's May 23 and August 16, 2006, letters describe how the input for the containment model was selected. Two parameters—containment volume and the initial containment air temperature—are uniformly sampled and treated statistically. The containment volume is sampled over a range from nominal volume to the containment empty volume. Since the pressure decreases with increasing volume, this range ensures that the selection of the volume is either best estimate (if the nominal volume is chosen) or conservative. The licensee stated that in many instances it used the guidance in Standard Review Plan, Section, 6.2.1.5, and Branch Technical Position CSB 6-1 in setting other containment model input parameters. The NRC staff agrees that the input for the containment model treatment provides a conservative bias to these parameters.

The initial containment air temperature was uniformly sampled over its normal operating conditions, ranging from a minimum of 83.44 °F to a maximum of 120 °F. Containment pressure was set at its nominal value of 14.2 psia. Containment volume was also uniformly sampled from an empty containment to its nominal value (1.16E6 to 1.02E6 ft^3). The spray flow (801.2 lb_m/s) was based on two pumps operating at their maximum flow rates (a conservatism to minimize containment pressure). The spray temperature (40 °F) was a conservative representation of the 50 °F minimum TS temperature for the safety injection and refueling water tank to minimize containment pressure. The licensee assumed a relative humidity of 100%, which is conservative. The licensee noted that the minimum containment accident pressure is not sensitive to this parameter.

Branch Technical Position CSB 6-1 classifies two types of heat sinks, passive and active. The licensee used containment sprays and fan coolers as active heat sinks. The licensee conservatively used the maximum spray flow rate without assuming any active failures and assuming that the spray pumps would immediately start without a time delay following a LOCA. The licensee stated that it used passive heat sinks as modeled in the current LBLOCA deterministic analysis of record (a 10 CFR Part 50, Appendix K, compliant calculation). These assumptions comply with the guidance of Branch Technical Position CSB-6-1.

An NRC staff study has shown that the CONTEMPT code containment spray model "tends to reduce pressure more rapidly than the data indicates" (Reference 9). Therefore, the NRC staff agrees that the CONTEMPT containment spray model is conservative for minimum pressure calculations. CONTEMPT, as described above, is the basis for the ICECON code used by the licensee.

An important consideration in determining the containment pressure is the heat transfer coefficient between the containment atmosphere and the passive heat sinks. The licensee uses the Uchida correlation (Reference 10) with a proprietary, conservative multiplier for the blowdown and post-blowdown phases of the LOCA. The licensee stated that it reestablished this multiplier following a realistic LBLOCA (RLBLOCA) evaluation model process. The guidance of Branch Technical Position CSB 6-1 recommends the use of four times the value obtained from the Tagami condensation heat transfer correlation (Reference 11) for the blowdown phase of the accident and the Uchida correlation for the post-blowdown phase with a multiplier of 1.2. The Uchida correlation models free convective condensation heat transfer. This mode of condensation heat transfer produces a lower bound to the expected heat transfer since air currents caused by the sprays and general turbulence of the containment atmosphere will cause the heat transfer to be greater than that from natural circulation alone. The multipliers- e.g., the value proposed by the licensee or the 1.2 value proposed in Branch Technical Position CSB 6-1- compensate for the use of the Uchida correlation for the case of minimum pressure calculations where the heat transfer should be maximized for conservatism.

Following the blowdown phase, ECCS water injected into the reactor vessel will be discharged from the break. The mixing of this water with the containment atmosphere can have an important effect on the containment pressure and the PCT. The licensee's May 23, 2006, letter noted that it used approved code models to model this mixing. The NRC staff requested that the licensee describe the models used. By letter dated August 16, 2006, the licensee responded to the above request with the following:

> The statement refers to the use of the ICECON containment code in the RLBLOCA evaluation model (EM). ICECON is NRC-approved for use within the context of the RLBLOCA EM. ICECON is a CONTEMPT-type containment pressure code and uses the normal instantaneous equilibrium steam-water mixing model. All ECCS fluid is injected into the reactor coolant system (RCS). There is no direct spillage of ECCS to the containment. The RCS resistance network determines what ECCS is processed through the break and into the containment. Then, in ICECON, the break discharge of steam and liquid components are instantaneously equilibrated in the containment atmosphere.

The instantaneous equilibrium occurs because the containment atmosphere is modeled as one volume. The staff agrees with this assumption and finds the licensee's treatment of ECCS spillage to be acceptable.

Containment pressure has a significant effect on the PCT following a LOCA. Minimizing containment pressure is conservative for LOCA PCT calculations. As described above, the licensee's method of calculating containment pressure is a combination of a realistic approach considering uncertainty and a deterministic approach. The licensee treated some input parameters statistically (the containment volume and the initial temperature of the containment atmosphere), while others were treated in a conservative manner relative to the guidance of Regulatory Guide 1.157 (containment heat sinks, both passive and active). Therefore, the NRC staff finds that the licensee's proposed method for calculating containment pressure for use in the RLBLOCA analysis will be conservative, and therefore, is acceptable.

3.3 Other Technical Issues

Although the NRC staff finds the licensee's proposed use of this methodology to be acceptable, there are a number of associated technical issues that the NRC staff evaluated that could have potentially affected the results of the analysis. This included the following issues:

3.3.1 Breaks at the Top and Side of Cold-Leg Pump Discharge Piping

For postulated split breaks at the top and side of cold-leg pump discharge piping, scenarios with indefinitely extended core uncovery could result without appropriate operator action to depressurize and establish low pressure recirculation cooling. This scenario applies to plants with deep reactor coolant system pump (suction) loop seals. In its September 30, 2005, letter (Reference 1), OPPD stated that FCS does not have "deep loop seals." Therefore, the NRC staff agrees with the licensee that FCS is not vulnerable to the postulated scenario of concern.

3.3.2 Post-LOCA Boron Precipitation

The issue of post-LOCA boron precipitation was considered in the original licensing of FCS. The licensee requested, and the NRC staff has approved by Amendment No. 241, the use of M5, an advanced alloy for fuel rod cladding and other assembly, in the FCS reactor core. Post-LOCA boron precipitation was evaluated in connection with the approval of that amendment, and so the NRC staff considers that the status of this issue is unchanged by the use of M5 fuel. The NRC staff may reconsider this if and when an FCS action is proposed that could affect boron precipitation findings, such as a power uprate.

3.3.3 Downcomer Boiling

The issue of downcomer boiling is associated with a concern that during the reflood stage of post-LOCA ECCS operation latent heat from the reactor vessel, the core barrel, and other vessel internals would be transferred to the water in the downcomer of the vessel. The head of water in the downcomer provides the driving force for the reflooding of the core. The heat transferred to the downcomer water would reduce the density of the water, and boil away water. Both of these effects would reduce the driving head for reflooding of the core. Depending on the magnitude and timing of this heat addition to the downcomer water, the core reflood rate could be adversely affected, with adverse consequences to the fuel in the core. In short, if the LOCA analysis methodology does not correctly model heat conduction in the reactor vessel wall, the magnitude and timing of vessel wall heat deposition to the fluid in the downcomer could be non-conservatively timed, such that oxidation and hydrogen generation would be underestimated due to the downcomer boiling effect.

This issue is a generic safety issue whose generic resolution is ongoing. When the Commission determines the appropriate approach, FCS will be required to implement the generic resolution applicable to its class of plants. Prior to that resolution, the EMF-2103(P)(A), Revision 0, methodology may not be used to support power uprates for FCS.

3.3.4 Oxidation and Hydrogen Generation

The NRC staff has identified a generic concern with the BELOCA methodology described in EMF-2103 with respect to how to determine the limiting LBLOCA oxidation and hydrogen generation. The NRC staff considered it necessary to evaluate whether the AREVA NP, Inc. BE LBLOCA methodology is consistent with Regulatory Guide 1.157, Section 4.4, which states, "[t]he revised paragraph 50.46(a)(1)(i) requires that it be shown with high probability that none of the criteria of paragraph 10 CFR 50.46(b) will be exceeded, and is not limited to the peak cladding temperature criterion. However since the other criteria are strongly dependent on peak cladding temperature, explicit consideration of the probability of exceeding the other criteria may not be required if it can be demonstrated that meeting the temperature criterion at the 95 percent probability level ensures with an equal or greater probability that the other criteria will not be exceeded." The NRC has referenced this guidance with regard to criteria of acceptance in BE LOCA reviews. However, the NRC staff has continued to examine whether the statistical approach used in the methodology provides the necessary assurance that the local oxidation and core wide hydrogen generation criteria of 10 CFR 50.46(b)(2) and (b)(3), respectively, would not be exceeded.

In assessing whether the local oxidation and core wide hydrogen generation results of the EMF-2103 (P)(A), Revision 0, BE LBLOCA methodology analysis for FCS provide the NRC staff assurance that these criteria will not be exceeded, the NRC staff considered the margin between the results and the acceptance criteria, and the results of the previous 10 CFR Part 50, Appendix K, LBLOCA analyses. As noted in Table 1, the results of the BE LBLOCA analyses are significantly below the limits established in 10 CFR 50.46(b)(2) and (b)(3). Despite the NRC staff's ongoing review of the statistical approach used in the AREVA NP, Inc. TR EMF-2103 BE LBLOCA methodology, the NRC staff concluded, given the margin between the calculated results and the acceptance criteria, that there is a high probability that at the current licensed power level for FCS, the oxidation and hydrogen generation acceptance criteria of 10 CFR 50.46(b)(2) and (b)(3) would not be exceeded. Further, the NRC staff finds that the previous LBLOCA analysis results described in the FCS Updated Safety Analysis Report (USAR) for oxidation (3.13 percent) and hydrogen generation (less than 1.0 percent) bound the results of the EMF-2103(P)(A), Revision 0, methodology, for FCS operating at its current licensed power level. Because the results in the FCS USAR bound the results using the EMF-2103(P)(A), Revision 0, methodology, and because of the significant margin between the EMF-2103(P)(A), Revision 0, results and the acceptance criteria of 10 CFR 50.46(b)(2) (more than 13 percent margin) and (b)(3), the NRC staff finds the submitted LBLOCA oxidation and hydrogen generation results acceptable for FCS operating at its current licensed power level.

3.4 LBLOCA Conclusions

The NRC staff's review of the acceptability of the EMF-2103(P)(A), Revision 0, methodology for FCS focused on assuring that the FCS-specific input parameters or bounding values and ranges (where appropriate) were used to conduct the analyses, that the analyses were conducted within the conditions and limitations of the NRC-approved AREVA NP, Inc. EMF-2103(P)(A), Revision 0, methodology, and that the results satisfied the requirement of 10 CFR 50.46(b) based on a licensed power level of up to 1,525 MWt.

This Safety Evaluation documents the NRC Staff review and the bases of acceptance of the EMF-2103(P)(A), Revision 0, BE LBLOCA analysis methodology for application to the FCS, and of the results of the BE LBLOCA analyses discussed above that were performed with the EMF-2103(P)(A), Revision 0, methodology for reference at FCS.

Based on its review as discussed above, the NRC staff concluded that the AREVA NP, Inc. BE LBLOCA methodology, as described in EMF-2103(P)(A), Revision 0, is acceptable for use at FCS. Further, the NRC staff concluded that the results demonstrate, with a high-probability level, that the acceptance criteria of 10 CFR 50.46(b)(1), (b)(2), and (b)(3), would not be exceeded during an LBLOCA at the current licensed power level. The NRC staff's conclusion was based on the assumed core power up to 1,525 MWt (plus 2.0 percent measurement uncertainty or 1,555.5 MWt).

4.0 FCS TECHNICAL SPECIFICATIONS

4.1 TS 5.9.5.b Core Operating Limits Report (COLR)

TS 5.9.5.b contains a list of documents that describe the analytical methods that may be used to determine the core operating limits. TS 5.9.5.b requires that these methods shall be reviewed and approved by the NRC and the approved version shall be identified in the COLR. There were no revisions to TS 5.9.5.b. However, TS 5.9.5.b Items 1, 2, and 3 reference [separate controlled documents that currently references] EMF-2087(P)(A), Revision 0, which will now be replaced by EMF-2103(P)(A), Revision 0, (LBLOCA PCT only). As such, this change is required to be submitted for review and approval by the NRC staff and, after issuance of this amendment, the COLR will be changed to reflect this new methodology.

The NRC staff has concluded that EMF-2103(P)(A) is an acceptable methodology to be applied by FCS for determination of PCT as discussed in Section 3 of this safety evaluation. Therefore it is an appropriate reference for the FCS LBLOCA analyses for core power up to 1,525 MWt (plus 2.0 percent measurement uncertainty or 1,555.5 MWt).

4.2 Summary

In summary, the licensee has performed LBLOCA analyses for FCS using an NRC-approved AREVA NP, Inc. methodology. The NRC staff concluded that the licensee's LBLOCA analyses were performed using an approved AREVA NP, Inc. methodology that applies to FCS for core power up to 1,525 MWt (plus 2.0 percent measurement uncertainty or 1,555.5 MWt). Based on the review of the licensee's LBLOCA calculations, the NRC staff has concluded the following for the operation of the FCS at the current licensed power level:

- The calculated LBLOCA values for PCT, oxidation, and core-wide hydrogen generation demonstrate with a high probability that the acceptance criteria of 2,200 °F, 17 percent, and 1.0 percent specified in 10 CFR 50.46(b)(1), (2), and (3), respectively, would not be exceeded during an LBLOCA.

- Compliance with 10 CFR 50.46(b)(1) through (3) and (5) ensures that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4).

- The proposed method for calculating containment pressure for use in the RLBLOCA analysis is conservative and, therefore, is acceptable.

- The licensee's LBLOCA analyses for FCS are acceptable for core power up to 1,525 MWt (plus 2.0 percent measurement uncertainty or 1,555.5 MWt).

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 152; published January 3, 2006). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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

8.0 REFERENCES

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