Progress Energy

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MAR 23 2010

Serial: HNP-10-029
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
APPLICATION FOR REVISION TO TECHNICAL SPECIFICATION
CORE OPERATING LIMITS REPORT (COLR) REFERENCES
FOR REALISTIC LARGE BREAK LOCA ANALYSIS

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) doing business as Progress Energy Carolinas, Inc. (PEC), requests an amendment to Appendix A, Technical Specifications (TS), of Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No. 1 (HNP).

The proposed amendment would modify HNP TS 6.9.1.6 to add NRC-approved Topical Report (TR) EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," to the Core Operating Limits Report methodologies list. The addition of TR EMF-2103(P)(A), Revision 0, as approved by the NRC on April 9, 2003, will allow the use of the S-RELAP5 thermal-hydraulic analysis code methodology for Chapter 15 realistic large break loss-of-coolant accident (RLBLOCA) in the HNP safety analyses.

The NRC Safety Evaluation on the use of the S-RELAP5 code contained a restriction that specific plant submittals include justification of the nodalization, chosen parameters and conservative nature of input parameters and calculated results. Accordingly, this submittal contains HNP's plant-specific analysis ANP-2853, Revision 0, "Harris Nuclear Plant Realistic Large Break LOCA Summary Report."

HNP requests approval of the proposed License Amendment by March 2011, with implementation to occur within 60 days of approval. This requested approval date has been administratively selected to accommodate a normal NRC review time.

In accordance with 10 CFR 50.91(b), PEC is providing the State of North Carolina with a copy of this proposed license amendment.

P.O. Box 165
New Hill, NC 27562

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ADD1
NRR

This document contains no new Regulatory Commitment.

Please refer any question regarding this submittal to Mr. Dave Corlett, Supervisor – HNP Licensing/Regulatory Programs, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on [MAR 23 2010].

Sincerely,

CLB/kms

Enclosures:
1. Evaluation of the Proposed Change
2. Affidavit for Withholding of Proprietary Information
3. AREVA Report No. ANP-2853(P), Revision 0 (Proprietary)
4. AREVA Report No. ANP-2853(NP), Revision 0 (Non-Proprietary Version)

cc:
Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, N.C. DENR Section Chief
Mr. L. A. Reyes, NRC Regional Administrator, Region II
Ms. M. G. Vaaler. NRC Project Manager, HNP

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APPLICATION FOR REVISION TO TECHNICAL SPECIFICATION
CORE OPERATING LIMITS REPORT (COLR) REFERENCES

Subject: *Request for License Amendment to add new analytical method to the Core Operating Limits Report (COLR) list of approved reports in Technical Specification 6.9.1.6.2.p.*

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ATTACHMENTS:

1. Technical Specification Page Markups
2. Retyped Technical Specification Pages

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1. SUMMARY DESCRIPTION

Carolina Power & Light Company (CP&L), doing business as Progress Energy Carolinas, Inc. (PEC), is proposing a change to Appendix A, Technical Specifications (TS), of Renewed Facility Operating License No. NPF-63, for the Shearon Harris Nuclear Power Plant, Unit No. 1 (HNP).

HNP TS 6.9.1.6.2 requires that, "The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR." The proposed change will revise TS 6.9.1.6, "Core Operating Limits Report," to add an NRC- approved topical report to TS 6.9.1.6.2, the listing of analytical methods used to determine the core operating limits.

2. DETAILED DESCRIPTION

The method HNP currently uses for determination of core operating limits and analysis of large break loss-of-coolant accidents (LBLOCA) is identified as EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," in HNP TS 6.9.1.6.2.f. This methodology complies with 10 CFR 50 Appendix K requirements and is used to demonstrate HNP's compliance with 10 CFR 50.46 requirements. The loss-of-coolant accident (LOCA) analysis is one input in determining the core peaking factors ($F_{\Delta h}$ and F_Q) specified in the core operating limits report (COLR).

HNP's current LOCA analysis methodology, EMF-2087(P)(A), is mechanistic and, since it is limited to material properties for industry legacy fuel cladding material Zircaloy-4, not retrofitted with M5 ™ properties.

PEC is submitting this License Amendment Request (LAR) for approval of the NRC-accepted Topical Report (TR) EMF-2103(P)(A), Revision 0, "Realistic Large Break Loss-of-Coolant Methodology for Pressurized Water Reactors," as a COLR reference to HNP TS. This methodology complies with the LOCA emergency core cooling system (ECCS) rule which allows the use of realistic LOCA evaluation models in place of the prescribed conservative evaluation models as specified by 10 CFR 50 Appendix K, provided that it can be established with a high probability that the criteria of 10 CFR 50.46 are not violated (Reference 1).

EMF-2103(P)(A) utilizes a best estimate methodology in the application of the S-RELAP5 thermal-hydraulic analysis computer code to realistic large break loss-of-coolant accidents (RLBLOCA) in Westinghouse and Combustion Engineering pressurized water reactors (PWRs). EMF-2103(P)(A) has been accepted by the NRC for

referencing in licensing applications to the extent specified and under the report limitations. The NRC Safety Evaluation (SE) for EMF-2103(P)(A) (Reference 2) approves application of the S-RELAP5 code in a realistic manner in which the uncertainties in estimating the necessary parameters to satisfy the requirements of 10 CFR 50.46(b) are determined for the LBLOCA.

The addition of EMF-2103(P)(A), Revision 0 as an authorized COLR reference for HNP will allow the use of the S-RELAP5 thermal-hydraulic analysis code methodology for Chapter 15 RLBLOCA in the HNP safety analysis. TR EMF-2103(P)(A), "Realistic Large Break Loss-of-Coolant Accident Methodology for Pressurized Water Reactors," will be added to HNP TS as 6.9.1.6.2.p. The core operating limits will be established in accordance with the applicable limitations as documented in the referenced NRC SE.

This proposed change in methodology does not result in a configuration change to plant structures, systems and components. The HNP-specific analysis, as presented in ANP-2853(P), Revision 0, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Summary Report," is based on inputs that are nominally aligned with the requirements of HNP's current (Cycle 16) Plant Parameters Document.

Since EMF-2087(P)(A) must be retained until the analysis based on EMF-2087(P)(A) are replaced with the newer EMF-2103 analysis, this LAR does not delete EMF-2087(P)(A) from HNP's TS 6.9.1.6.2 COLR reference list.

2.1 EMF-2103(P)(A) "Realistic Large Break Loss-of-Coolant Methodology for Pressurized Water Reactors"

EMF-2103(P)(A) applies the S-RELAP5 code in a realistic manner in which the uncertainties in estimating the necessary parameters to satisfy 10 CFR 50.46(b) requirements are determined for the LBLOCA. The use of a realistic code allows the replacement of many of the prescriptive evaluation models by more realistic models and requires evaluating the uncertainty in the calculated results. The NRC determined that the AREVA methodology for the statistical results of a RLBLOCA PWR analysis meets the 10 CFR 50.46 and Regulatory Guide 1.157 acceptance criteria.

In the EMF-2103(P)(A) analysis, LOCA simulations, performed with the S-RELAP5 computer code, are run with several different sets of plant input. Each input that changes between runs is randomly chosen from a distribution of possible configurations, with other inputs conservatively biased to produce more limiting LBLOCA results such as higher peak clad temperature (PCT). The RLBLOCA methodology determines values of PCT at the 95 percent probability level. Total oxidation and total hydrogen are based on the limiting PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the applicable 10 CFR Part 50.46 acceptance criteria.

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The proposed change to HNP's TS 6.9.1.6.2 adds a methodology that has been previously approved by the NRC. A number of licensees, including H.B. Robinson, Ft. Calhoun, Palisades, North Anna and Sequoyah Nuclear Station, have obtained NRC approval to utilize this methodology. This LAR and the accompanying HNP-specific ANP-2853 incorporate lessons learned from these submittals. No significant generic licensing questions are outstanding.

NRC's review and approval of EMF-2103(P)(A), Revision 0 for use in determining core operating limits authorizes use of the S-RELAP5 code and realistic methods described above in the analysis of LBLOCA events in PWRs. Specifically, the NRC concluded in its letter and SE (Reference 2), for TR EMF-2103(P)(A), Revision 0, that:

> The NRC staff finds that the Framatome ANP methodology for the statistical results of an analysis of a RLBLOCA of a PWR meets the acceptance criteria stated in 10 CFR 50.46 and Regulatory Guide 1.157...
>
> The NRC staff concludes from its review of the documentation, code and input models submitted that the S-RELAP5 RLBLOCA methodology is structured consistent with the CSAU [Code Scaling, Applicability, and Uncertainty] methodological process, and satisfactorily reflects the intended use of the methodology to address licensing requirement for a variety of similarly designed nuclear power plants.

The NRC's acceptance of EMF-2103(P)(A) noted that since a generic TR describing a code such as S-RELAP5 cannot provide a detailed justification for each plant application, each applicant must provide justification for its specific application of the S-RELAP5 code. The results of a plant specific analysis are to be submitted with the LAR for approval of the S-RELAP5 code. This plant specific analysis is expected to include the nodalization, chosen parameters and conservative nature of input parameters and calculated results (Reference 2).

In accordance with the above requirement, HNP's submittal includes proprietary and non-proprietary, designated as P (proprietary) and NP (non-proprietary), versions of AREVA NP Inc. report ANP-2853, Revision 0, "Harris Nuclear Plant Realistic Large Break LOCA Summary Report." ANP-2853(P) provides HNP's plant specific LBLOCA analysis using the EMF-2103(P) methodology, demonstrating that the applicable acceptance criteria are met when using the EMF-2103(P) methodology.

The NRC SE (Reference 2) for EMF-2103(P)(A) also contains restrictions for consideration when the AREVA methodology is used for analysis of RLBLOCA. Table 3-4 of ANP-2853(P) contains the RLBLOCA conditions and limitations identified in the NRC SE and the corresponding HNP site-specific response.

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3. TECHNICAL EVALUATION

The purpose of the submitted analysis is to verify typical TS peaking factor limits and ECCS adequacy by demonstrating that 10 CFR 50.46(b)(1) through (3) criteria are met.

3.1 Identification of Event

A LBLOCA is initiated by a postulated rupture of the reactor coolant system (RCS) primary piping. The rupture is of sufficient size that there is rapid depressurization of the RCS. The accident is characterized by four phases: blowdown, refill, reflood, and long term core cooling. Blowdown is the initial depressurization of the RCS, defined as the time period from initiation of the break until accumulator or safety injection tank flow begins. The LOCA refill phase begins with the injection of water from the Safety Injection Accumulators and the pumped flow from the ECCS and ends when the reactor vessel downcomer and lower plenum are refilled. During the reflood phase, the fuel is rewet as liquid water level is restored in the core region. This reflood phase continues until the clad is quenched and stable long term cooling is established.

3.2 LOCA Long-term Cooling

Since the addition of EMF-2103 as an evaluation methodology for LBLOCA does not impact the current analysis of long term cooling, it is therefore not addressed in the submitted AREVA report ANP-2853, Revision 0. The current analysis presented in FSAR Sections 6.3 and 15.6.5 verifies the ability of HNP's ECCS to prevent core heatup following a LOCA, meeting 10 CFR 50.46(b)(5) criteria for long-term cooling.

The current long-term cooling calculations demonstrate that the first, and subsequent, ECCS switchovers to hot leg injection and back to cold leg injection will prevent boron precipitation. Critical parameters to this analysis include the volumes and boron concentrations of tanks that inject into the reactor following a LOCA. Analysis also confirms that ECCS flow exceeds the reactor boil-off rate following the LOCA PCT, based on a conservative core decay heat assumption.

3.3 Justification of Nodalization

Figures 3-1 to 3-5 of the submitted AREVA report contain the reactor vessel, primary system, secondary system core and upper plenum nodalization details based on a HNP plant model. This plant configuration is represented by an S-RELAP5 model which nodalizes the primary and secondary sides into control volumes representing reasonable homogeneous regions, interconnected by flow paths.

The RCS is modeled by multi-node representations for the reactor vessel, which is comprised of an active core region, inlet and outlet plena, a downcomer, barrel-baffle region and reactor vessel upper head. Specifically, the loop configuration for HNP consists of three loops, each with one hot leg, a U-tube steam generator, a cold leg, accumulator, reactor coolant pump and nodes for the injection of ECCS water. All three individual reactor coolant loops are modeled and include connections to the three steam generators, with one loop connected to the pressurizer.

The HNP steam generator models contain inlet and outlet plena, multi-node U-tubes for the primary side and multi-node downcomers, U-tube boiling regions, separators and steam domes in the secondary side. Steam lines, steam safety valves and steam line isolation valves are also represented.

3.4 Parameter Selection

Consistent with the approved EMF-2103(P)(A) methodology, several parameters were "sampled" or varied from case to case over a distribution. Others were conservatively biased to increase PCT. The distribution of each "sampled" parameter was chosen based on an uncertainty assessment of plant data, plant operating limits and/or plant data. Sampled parameters include break size, break type, core burnup, core power, F_Q, axial offset, pressurizer pressure, pressurizer level, reactor coolant system T_{avg}, coolant flow rate, accumulator volume, accumulator pressure, containment volume and containment temperature. Lists and descriptions of the sampled parameters are provided in Tables 3-1 and 3-3 of the AREVA report ANP-2853, Revision 0. The generated plant data used in each analysis is contained in Figure 3-6. Table 3-2 provides the operating parameters supported by the analysis.

Reactor power was conservatively biased to the TS rated thermal power of 2900 MWt plus a power uncertainty of 2%, for a total power of 2958 MWt.

In concurrence with the NRC's interpretation of General Design Criteria (GDC) 35, a set of 59 cases was run with loss-of-offsite power (LOOP) and 59 cases were run without LOOP. The set of 59 cases which resulted in the highest PCT is reported in Sections 2 and 3 of AREVA report ANP-2853, Revision 0.

Containment pressure has been biased low for the LBLOCA analysis. As stated in AREVA report ANP-2853, Revision 0, all pressure reducing systems are assumed to function.

3.5 Safety Evaluation Report Conditions and Limitations

A description of HNP's responses to the conditions and limitations identified in the NRC SE are provided in Table 3-4 of AREVA report ANP-2853, Revision 0. Incorporated in Section 4.0 of ANP-2853, Revision 0 are responses to NRC Requests for Additional Information pertaining to the generic application of EMF-2103(P)(A), Revision 0 in other licensee applications.

3.6 Results

The input data used and supported by the AREVA analysis are provided in Tables 3-2, 3-8 and 3-9 and in Figure 3-6 of ANP-2853. Two case sets of 59 transient calculations were performed sampling the parameters listed in ANP-2853 Table 3-1.

Results are reported for the case with the highest PCT. The sequence of events for this case is provided in Table 3-6. Figures 3-11 to 3-21 provide responses of key system variables. Table 3-5 lists results for the limiting PCT case, demonstrating compliance with the applicable criteria for PCT and metal-water reaction.

Application of the NRC-approved RLBLOCA analysis EMF-2103(P)(A), Revision 0 to HNP results in a PCT of 1930°F for the limiting LOOP case. Maximum oxidation thickness and hydrogen generation are within regulatory requirements.

Supported by this analysis are a Rated Thermal Power operation of 2900 MWt, including a measurement uncertainty of 2 percent, steam generator tube plugging level of up to 3 percent in all steam generators, a total peaking factor (FQ) of 2.52 (including uncertainty), a nuclear enthalpy rise factor (FDH) of 1.73 (including 4 percent uncertainty) with no axial or burnup dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. For LBLOCA, the three 10 CFR50.46(b) criteria presented in Section 3.0 are met and operation of HNP with AREVA NP-supplied 17x17 Zr-4 clad fuel is justified.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

TR EMF-2103(P)(A) pertains to LBLOCA analyses that are part of the HNP licensing basis. The regulatory bases for these analyses are found in the GDC (Reference 4). The GDCs that pertain to each of the analyses are listed in the Standard Review Plan (SRP) (Reference 5).

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The definition of evaluation models of LOCA events per 10 CFR 50.46 is:

> "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."

10 CFR 50, Appendix K, Section II contains the documentation requirements for evaluation models as follows:

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 or 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 NRC 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)(ii), compliance with required features of section I of this Appendix K; and, 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.

10 CFR 50, Appendix B, Section III, which governs references to design control measures in the COLR 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 in-service inspection, maintenance and repair; and delineation of acceptance criteria for inspections and tests."

4.2 Precedent

This application is submitted in accordance with the restrictions regarding the use of EMF-2103(P)(A) as provided in the NRC SE on the topical report.

The AREVA analysis submitted (Enclosures 3 and 4) satisfy the NRC limitations contained in the SE regarding site-specific application for approval of EMF-2103(P)(A). Therefore, the addition of EMF-2103(P)(A) to the HNP TS 6.9.1.6.2 is acceptable.

4.3 Significant Hazards Consideration

Carolina Power & Light Company (CP&L), doing business as Progress Energy Carolinas, Inc. (PEC), has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below. This evaluation is in conformance with the guidance provided in NRC Regulatory Issue Summary (RIS) 2001-22.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The topical report has been reviewed and approved by the NRC for use in determining core operating limits and for evaluation of large break loss-of-coolant accidents. The core operating limits to be developed using the new methodologies for HNP will be established in accordance with the applicable limitations as documented in the NRC Safety Evaluation Report. In the April 9, 2003, NRC SE, the NRC concluded that the S-RELAP5 RLBLOCA methodology is acceptable for referencing in licensing applications in accordance with the stated limitations.

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The proposed change enables the use of new methodology to re-analyze a large break loss-of-coolant accident. It does not, by itself, impact the current design bases. Revised analysis may either result in continued conformance with design bases or may change the design bases. If design basis changes result from a revised analysis, the specific design changes will be evaluated in accordance with HNP design change procedures and 10 CFR 50.59.

The proposed change does not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fission product barriers during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated.

The proposed methodologies will ensure that the plant continues to meet applicable design and safety analyses acceptance criteria. The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are impacted and there are no adverse effects on the factors that contribute to offsite or onsite dose as a result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, and components are maintained consistent with the safety analysis and licensing bases.

Therefore, this amendment does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

Response: No.

The proposed change does not involve any physical alteration of plant systems, structures, or components. No new or different equipment is being installed and no installed equipment is being operated in a different manner. There is no change to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced.

Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

There is no impact on any margin of safety resulting from the incorporation of this new topical report into the Technical Specifications. If design basis changes result from a revised analysis that uses these new methodologies, the specific design changes will be evaluated in accordance with HNP design change procedures and 10 CFR 50.59. Any potential reduction in the margin of safety would be evaluated for that specific design change.

Therefore, this amendment does not involve a significant reduction in the margin of safety.

4.4 Conclusions

Based on the considerations discussed above, (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.

5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Paragraph (c)(9).

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Therefore, pursuant to 10 CFR 51.22, Paragraph (b), an Environmental Impact Statement or Environmental Assessment is not required in connection with the proposed amendment.

6. REFERENCES

1. Realistic Large Break LOCA Methodology for Pressurized Water Reactors, EMF-2103(NP), Revision 0, dated August 23, 2001 (ML012400019).

2. Letter from NRC to Framatome ANP, Safety Evaluation of Licensing Topical Report EMF-2103(P), Revision 0, "Realistic Large Break LOCA methodology for Pressurized Water Reactors" (TAC NO. MB7554), April 9, 2003 (ML030760312).

3. NRC Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989.

4. Title 10 of the *Code of Federal Regulations*, Appendix A, Part 50, General Design Criteria for Nuclear Power Plants.

5. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.

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ATTACHMENT 1
TECHNICAL SPECIFICATION PAGE MARKUPS
(2 Pages)

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

Delete

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

Insert "A"

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.1. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

Insert "A":

p. EMF-2103 (P)(A), Realistic Large Break LOCA Methodology approved version as specified in the COLR

(Methodology for Specification 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
APPLICATION FOR REVISION TO TECHNICAL SPECIFICATION
CORE OPERATING LIMITS REPORT (COLR) REFERENCES

ATTACHMENT 2
RETYPED TECHNICAL SPECIFICATION PAGES
(2 Pages)

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

p. EMF-2103 (P)(A), Realistic Large Break LOCA Methodology approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.1. The report shall include:

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
APPLICATION FOR REVISION TO TECHNICAL SPECIFICATION
CORE OPERATING LIMITS REPORT (COLR) REFERENCES

AREVA
AFFIDAVIT PURSUANT TO 10 CFR 2.390
For AREVA Report No. ANP-2853(P), Revision 0 (Proprietary)
(3 Pages)

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the report ANP-2853(P), Revision 000, entitled "Harris Nuclear Plant, Unit 1, Realistic Large Break LOCA Summary Report," dated January 2010 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 28th

day of January 2010.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010