Progress Energy

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10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

Ladies and Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations (10 CFR), Part 50.90, Carolina Power and 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 1 (HNP).

The proposed amendment would revise TS 6.9.1.6, "Core Operating Limits Report," to add plant specific methodology ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, that implements AREVA's NRC-approved topical report EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," and add EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 2 or higher upon approval of the specific revision by the NRC, to the TS 6.9.1.6.2 listing of analytical methods used to determine the core operating limits, and eliminates extraneous detail in TS 6.9.1.6 that cross references each method to the applicable TS Section 3.0 specifications and parameters.

The Enclosure to this letter provides an evaluation of the proposed changes. The evaluation presents both deterministic and risk-informed justifications for the acceptability of the proposed change.

CP&L requests approval of the proposed License Amendment by March 21, 2012, with implementation to occur within 60 days of approval or upon Cycle 18 startup.

Enclosure 3 contains proprietary information. Per the affidavit for withholding proprietary information (Enclosure 2), CP&L requests that the NRC withhold the information in accordance with 10 CFR 2.390. Upon removal of Enclosure 3, this letter and Enclosure 1 are decontrolled.

In accordance with 10 CFR 50.91(a)(1), "Notice for Public Comment," the analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is being provided to the Commission in accordance with the distribution requirements in 10 CFR 50.4. In

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

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

This document contains no Regulatory Commitment.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on
[8-19-11].

Sincerely,

Keith Holbrook
Manager – Support Services
Harris Nuclear Plant

RKH/kab

Enclosures:
1. Evaluation of Proposed Change
2. AREVA Affidavit for Withholding of Proprietary Data
3. AREVA Report ANP-3011(P) Revision 1 (Proprietary)
4. AREVA Report ANP-3011(NP), Revision 1 (Non-Proprietary)

cc: Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, N.C. DENR Section Chief
Mr. V. M. McCree, NRC Regional Administrator, Region II
Mrs. B. L. Mozafari, NRC Project Manager, HNP

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Subject: *Request for License Amendment to add new analytical method to the Core Operating Limits Report (COLR) List of Approved Methodologies in Technical Specification 6.9.1.6.2.*

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ATTACHMENTS:

1. Technical Specification Page Markups
2. Retyped Technical Specification Pages
3. Cross-reference of RAIs and ANP-3011(P)

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1.0 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 plant-specific methodology ANP-3011(P), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," that implements AREVA's NRC-approved topical report (TR) EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 0, and also adds EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 2 or higher upon approval of the specific revision by the NRC, to the TS 6.9.1.6.2 listing of analytical methods used to determine the core operating limits.

Specifically, TS 6.9.1.6.2.f is revised by replacing "EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR," with "ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, or EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 2 or higher, approved version as specified in the COLR."

The proposed change also eliminates narrative in TS 6.9.1.6 that cross-references each method to the applicable TS Section 3.0 specifications and parameters.

2.0 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 the requirements of Appendix K to 10 CFR 50 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, is not retrofitted with M5™ properties.

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CP&L is submitting this License Amendment Request (LAR) for approval of ANP-3011(P) as a plant-specific implementation methodology of the NRC-accepted TR EMF-2103(P)(A), Revision 0, (Reference 1) and for the addition of ANP-3011(P) and EMF-2103(P)(A), Revision 2 or higher upon approval of the specific revision by the NRC, to the TS 6.9.1.6.2 as a COLR reference for HNP. 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 Appendix K to 10 CFR 50, provided that it can be established with a high probability that the criteria of 10 CFR 50.46 are not violated.

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 ANP-3011(P) and EMF-2103(P)(A), Revision 2 or higher upon approval of the specific revision by the NRC, as an authorized COLR methodology for HNP will allow the use of the S-RELAP5 thermal-hydraulic analysis code methodology for HNP Final Safety Analysis (FSAR) Chapter 15 RLBLOCA in the HNP safety analysis. ANP-3011(P) and EMF-2103(P)(A), Revision 2 or higher upon approval of the specific revision by the NRC, will be added to HNP TS as 6.9.1.6.2.f. The core operating limits will be established in accordance with the applicable limitations as documented in the referenced NRC SE for EMF-2103(P)(A) and the conservative application of EMF-2103(P)(A) as described in ANP-3011(P).

Through review of several recent submittals, the NRC staff identified some issues related to AREVA methodologies, some of which were employed in the development of the HNP-specific RLBLOCA analysis. These issues, and the path forward for NRC review of EMF-2103, Revision 2, were discussed with the NRC in a meeting on March 1, 2011. Upon NRC approval of EMF-2103, Revision 2, the plant-specific methodology of ANP-3011(P), and the deviations from EMF-2103(P)(A), will no longer be required as a COLR methodology and will be replaced by the use of EMF-2103(P)(A), Revision 2 or higher upon NRC approval of the specific revision by the NRC. Notification to the NRC of such a change will be via updates to the COLR required by TS 6.9.1.6.4.

This proposed change in methodology does not result in a configuration change to plant structures, systems and components (SSCs). The HNP-specific analysis, as presented in ANP-3011(P), is based on inputs that are aligned with the requirements of HNP's Cycle 18

Plant Parameters Document. Cycle 18 is currently scheduled to begin in the second quarter of 2012.

EMF-2087(P)(A) is deleted from the TS 6.9.1.6.2 listing of COLR methodologies with the implementation of this TS amendment as it will no longer be needed.

2.1 ANP-3011(P), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis"

The HNP COLR methodology list is being revised to reference the plant-specific implementation of EMF-2103(P)(A) because of several exceptions taken to the methodology. These exceptions are generally in the conservative direction and are in response to NRC questions on the implementation of EMF-2103(P)(A) or other issues that have arisen since the issuance of the SE on EMF-2103(P)(A). These items are summarized in the noted sections of ANP-3011(P) and listed below:

a. Treatment of cold leg condensation efficiency is revised to conservatively address the impacts of potential downcomer boiling on peak clad temperature (PCT) (Section 1.0).

b. The core thermal power is not sampled. The value used is the core rated thermal power plus power measurement uncertainty (Sections 1.0 and 5.1)

c. The RLBLOCA analysis requires both the void fraction to be less than 0.95 and the clad temperature to be less than 900 degrees F before the rod is allowed to quench (Section 5.2).

d. The split verses double ended break type is no longer related to break area. Both the split and the double ended break will range in area between the minimum break area (A_{min}) and are of twice the size of the broken pipe (Section 5.6).

e. The contribution of the Forslund-Rohsenow heat transfer model is limited to no more than 15 percent of the total heat transfer at and above a void fraction of 0.9 (Section 5.4).

f. Thermal conductivity degradation is accounted for as a function of burnup and once-burned rods are included in the modeling (Section 6.1).

g. The analysis performs 59 cases with Loss of Offsite Power (LOOP) and 59 cases with offsite power available (Section 6.3).

h. One HNP-specific sensitivity study has been performed to address the effect of rod swelling, rod rupture and pellet material relocation (Section 6.9).

i. Decay heat generation uses an alternate approach to that contained in EMF-2103(P)(A) (Section 6.11).

The following sections describe the implementation of EMF-2103(P)(A) via ANP-3011(P).

2.2 EMF-2103(P)(A), Revision 2, "Realistic Large Break LOCA 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 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 50.46 acceptance criteria.

The proposed change to HNP's TS 6.9.1.6.2 is based on an NRC-approved methodology. A number of licensees, including H.B. Robinson, Fort Calhoun, Palisades, North Anna, Calvert Cliffs and Sequoyah Nuclear Station (References 12 through 17), have obtained NRC approval to utilize EMF-2103(P)(A). This LAR and the accompanying HNP-specific ANP-3011(P) incorporate lessons learned from these submittals. With the constraints identified in Section 2.1 above, no significant generic licensing questions are outstanding.

NRC's review and approval of EMF-2103(P)(A) 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 made the following conclusion in its letter and SE, for TR EMF-2103(P)(A):

> 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 versions designated as P and NP, respectively, of AREVA NP Inc. ANP-3011(P), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis." ANP-3011(P) provides HNP's plant-specific LBLOCA analysis using the EMF-2103(P)(A) methodology, demonstrating that the applicable acceptance criteria are met when using the EMF-2103(P)(A) methodology.

The NRC SE 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-3011(P) contains the RLBLOCA conditions and limitations identified in the NRC SE and the corresponding HNP site-specific response.

2.3 Deletion of Extraneous COLR Detail

Currently, the HNP TS 6.9.1.6.2 listing of methodologies includes details cross-referencing each methodology to the applicable specifications and parameters. NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 3 (Reference 11), provides a model of the expected content of the Administrative Controls description of the COLR. CP&L has compared the required contents of the NUREG-1431 COLR description with the current HNP description and has determined that cross-referencing each methodology to the associated specifications and parameters is not required. Therefore, this detail is deleted.

3.0 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(P)(A) as an evaluation methodology for LBLOCA does not impact the current analysis of long-term cooling, it is not addressed in ANP-3011(P). 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 through 3-5 of ANP-3011(P) 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, 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, and/or plant operating limits. Sampled parameters include break size, break type, core burnup, F_Q, axial offset, pressurizer pressure, pressurizer level, RCS average temperature (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 ANP-3011(P). 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 core thermal power of 2900 MWt plus a power uncertainty of 2 percent, for a total analyzed core power of 2958 MWt. This approach matches the conditions for the proposed Measurement Uncertainty Recapture (MUR) uprate which uses a total core power of 2948 MWt and a power uncertainty of 0.34 percent.

In concurrence with the NRC's interpretation of General Design Criteria (GDC) 35, a set of 59 cases was run with 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 ANP-3011(P).

Containment pressure has been biased low for the LBLOCA analysis. As stated in ANP-3011(P), all pressure reducing systems are assumed to function.

3.5 EMF-2103(P)(A) Safety Evaluation 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 ANP-3011(P). Incorporated in Sections 5.0 and 6.0 of ANP-3011(P) are responses to NRC RAIs pertaining to the generic application of EMF-2103(P)(A) 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-3011(P). Two case sets of 59 transient calculations were performed sampling the parameters listed in Table 3-1 of ANP-3011(P). Results are reported for the case with the highest PCT. The sequence of events for this case is provided in Table 3-6. Figures 3-12 through 3-22 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.

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Application of the NRC-approved RLBLOCA analysis EMF-2103(P)(A) to HNP results in a PCT of 1919 degrees F for the limiting non-LOOP case. Maximum oxidation and hydrogen generation are within regulatory requirements.

Supported by this analysis are:

- Current Rated Thermal Power operation of 2900 MWt, including a measurement uncertainty of 2 percent or the proposed MUR power of 2948 MWt with a measurement uncertainty of 0.34 percent;
- Steam generator tube plugging level of up to 3 percent in all steam generators;
- Total peaking factor (F_Q) of 2.52 (including uncertainty);
- Nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.73 (including 4 percent uncertainty) with no axial or burnup dependent power peaking limit.

For LBLOCA, the 10 CFR 50.46(b)(1) through (3) criteria are met and operation of HNP with AREVA NP-supplied 17x17 M5™ or Zircaloy-4 clad fuel is justified.

3.7 Resolution of NRC Requests for Additional Information (RAI)

A license amendment was previously proposed to add EMF-2103(P)(A) directly to the HNP list of COLR references (Reference 8). Two transmittals of RAIs were received; the response to the first set (Reference 9) was docketed and the response to the second set (Reference 10) led to the reanalysis that is provided in ANP-3011(P).

Attachment 3 provides a cross-reference between the respective RAI and the analysis method contained in ANP-3011(P).

3.8 Deletion of Extraneous COLR Detail

Currently, the HNP TS 6.9.1.6.2 listing of methodologies includes details cross-referencing each methodology to the applicable TS specifications and parameters. NUREG-1431 provides a model of the expected content of the Administrative Controls description of the COLR. CP&L has compared the required contents of the NUREG-1431 COLR description with the current HNP description and has determined that cross-referencing each methodology to the associated specifications and parameters is not required. NUREG-1431 requires that the COLR description contains: 1) a list of the individual specifications that address core operating limits and 2) identification of the topical reports used to determine the core operating limits by name and title or identify the staff Safety Evaluation for a plant-specific methodology by NRC letter and date. NUREG-1431 does not require that each methodology be cross-referenced to each applicable specification and parameter. Therefore, these cross-referencing details are deleted. Because there is no technical change to the TS, this change is administrative in nature.

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4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

TR EMF-2103(P)(A) pertains to LBLOCA analyses, and the regulatory bases for these analyses are found in the GDC (Reference 6). The GDCs that pertain to each of the analyses are listed in the Standard Review Plan (SRP) (Reference 7).

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."

Section II of Appendix K to 10 CFR 50 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.

Section III of Appendix B to 10 CFR 50, 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

The use of EMF-2103(P)(A) in RLBLOCA analyses has been previously approved by the NRC in the following SEs:

- "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 – Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)," dated February 18, 2011 (ADAMS Accession No. ML110390224),
- "Sequoyah Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Core Operating Limits Report References for Realistic Large Break Loss-of-Coolant Accident Methodology (TAC No. MD8532)," dated September 24, 2008 (ADAMS Accession No. ML073601002),
- "Palisades Nuclear Plant – Issuance of Amendment Re: Realistic Large Break Loss-of-Coolant Accident (TAC No. MD3492)," dated January 31, 2008, (ADAMS Accession No. ML080110060),
- "H. B. Robinson Steam Electric Plant, Unit 2 – Issuance of Amendment Regarding Methodology for Large Break Loss-of-Coolant Accident Analyses (TAC No. MC6630)," dated September 20, 2006 (ADAMS Accession No. ML062330018),
- "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)," dated November 3, 2006 (ADAMS Accession No. ML062900184), and
- "North Anna Power Station, Unit 2 – Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel (TAC No. MB4715)," dated April 1, 2004 (ADAMS Accession No. ML040960040),

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

The HNP methodology and illustrative analysis ANP-3011(P) incorporates RAI responses that resulted from the NRC review of license amendment dated March 23, 2010, as well as the NRC limitations contained in the SE regarding site-specific application for approval of EMF-2103(P)(A). Therefore, the addition of ANP-3011(P) and EMF-2103(P)(A), Revision 2 or higher upon approval of the specific revision by the NRC, to the HNP TS 6.9.1.6.2 listing of methodologies is acceptable.

4.3 Significant Hazards Consideration

The proposed change would revise the Shearon Harris Nuclear Power Plant, Unit No. 1 (HNP), Renewed Facility Operating License No. NPF-63, Technical Specification (TS) 6.9.1.6, "Core Operating Limits Report," to add plant-specific methodology ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, that implements AREVA's NRC-approved Topical Report (TR) EMF-2103(P)(A), "Realistic Large Break Loss-of-Coolant Methodology for Pressurized Water Reactors," Revision 0, and also adds EMF-2103(P)(A), "Realistic Large Break Loss-of-Coolant Methodology for Pressurized Water Reactors," Revision 2 or higher upon approval of the specific revision by the NRC, to the TS 6.9.1.6.2 listing of analytical methods used to determine the core operating limits. Specifically, TS 6.9.1.6.2.f is revised by replacing "EMF-2087(P)(A), 'SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications,' approved version as specified in the COLR," with "ANP-3011(P), 'Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis,' Revision 1, or EMF-2103(P)(A), 'Realistic Large Break Loss-of-Coolant Methodology for Pressurized Water Reactors,' Revision 2 or higher, approved version as specified in the COLR."

The proposed change also eliminates detail in TS 6.9.1.6 that cross-references each method to the applicable specifications and parameters.

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 TR underlying the proposed HNP methodology has been reviewed and approved by the NRC for use in determining core operating limits and for evaluation of LBLOCA. 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 SE. 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.

The proposed change enables the use of new methodology to re-analyze a LBLOCA. 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 (SSC). 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 SSCs 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 SSCs. 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 TR into the TS or deletion of cross-reference information from the description of the COLR. 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.

5.0 ENVIRONMENTAL CONSIDERATION

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(c)(9), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review."

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

6.0 REFERENCES

1. EMF-2103(NP), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," dated April 2003 (ADAMS Accession No. ML032691424).

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)", dated April 9, 2003 (ADAMS Accession No. ML030760312).

3. ANP-3011(P), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," dated August 2011.

4. ANP-3011(NP), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," dated August 2011.

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

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

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

8. Letter from C. Burton to the NRC, "Shearon Harris, Unit 1, Application for Revision to Technical Specification Core Operating Limits Report References, ANP-2853(NP), Rev. 0, 'Realistic Large Break LOCA Summary Report'," (HNP-10-029), dated March 23, 2010 (ADAMS Accession No. ML100890593).

9. Letter from C. Burton to the NRC, "Shearon Harris Nuclear Power Plant, Unit No. 1, Response to Request for Additional Information, Regarding Amendment to Incorporate A Realistic Large Break Loss of Coolant Accident Methodology into the Core Operating Limits Report (TAC NO. ME3569)," (HNP-10-105), dated December 9, 2010 (ADAMS Accession No. ML103500470).

10. Letter from the NRC to C. Burton (via email), "Shearon Harris, Unit 1, Email Attachment, RLBLOCA Round 2 RAIs Qs," dated February 28, 2011 (ADAMS Accession No. ML110590577).

11. NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 3.0, dated March 31, 2004.

12. Letter from the NRC to G. Gellrich, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)," dated February 18, 2011 (ADAMS Accession No. ML110390224).

13. Letter from the NRC to W. Campbell, "Sequoyah Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Core Operating Limits Report References for Realistic Large Break Loss-of-Coolant Accident Methodology (TAC No. MD8532)," dated September 24, 2008 (ADAMS Accession No. ML073601002).

14. Letter from the NRC to M. Balduzzi, "Palisades Nuclear Plant - Issuance of Amendment Re: Realistic Large Break Loss-of-Coolant Accident (TAC No. MD3492)," dated January 31, 2008 (ADAMS Accession No. ML080110060).

15. Letter from the NRC to T. Walt, "H. B. Robinson Steam Electric Plant, Unit 2 - Issuance of Amendment Regarding Methodology for Large Break Loss-of-Coolant Accident Analyses (TAC No. MC6630)," dated September 20, 2006 (ADAMS Accession No. ML062330018).

16. Letter from the NRC to R. Ridenoure, "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)," dated November 3, 2006 (ADAMS Accession No. ML062900184).

17. Letter from the NRC to D. Christian, "North Anna Power Station, Unit 2 - Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel (TAC No. MB4715)," dated April 1, 2004 (ADAMS Accession No. ML040960040).

18. Letter from C. Burton to the NRC, "Shearon Harris Nuclear Power Plant, Unit 1, Docket No. 50-400/Renewed License No. NPF-63, Request for License Amendment, Measurement Uncertainty Recapture," dated April 28, 2011 (ADAMS Accession No. ML11124A180).

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ATTACHMENT 1
TECHNICAL SPECIFICATION PAGE MARKUPS
(4 pages)

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

f. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), and V(Z) for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 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.

a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 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, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

b. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

(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).

c. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

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

e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

(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, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

f. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," 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).

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

Replace with: ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision1, or EMF-2103 (P)(A), Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 2 or higher, approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

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

i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

(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).

j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

k. XN-NF-82-49(P)(A), "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," 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).

l. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 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, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, 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 CORE OPERATING LIMITS REPORT (Continued)

n. 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).

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,

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,

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ATTACHMENT 2
RETYPED TECHNICAL SPECIFICATION PAGES
(4 pages)

6.9.1.6 CORE OPERATING LIMITS REPORT

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c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5,

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e. Axial Flux Difference Limits for Specification 3/4.2.1,

f. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), and V(Z) for Specification 3/4.2.2,

g. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 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.

a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

b. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

c. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

f. ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, or EMF-2103 (P)(A), Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 2 or higher, approved version as specified in the COLR.

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

k. XN-NF-82-49(P)(A), "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," approved version as specified in the COLR.

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m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

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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.

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.

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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,

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ATTACHMENT 3
CROSS-REFERENCE OF RAIS AND ANP-3011(P)
(3 pages)

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Response to NRC RAIs on Application of RLBLOCA to HNP

The captioned license amendment request incorporates the responses to RAIs that were received on a previous (Reference 8) LAR to add EMF-2103 methodology to the HNP TS. The RAIs were captured in Reference 9 (below numbered as I-1 to I-10) and in Reference 10 (below numbered as II-1 to II-10).

The table that follows summarizes the RAI and action that was taken. As applicable reference sections of ANP-3011 are identified where the response impacts the implementation of RLBLOCA methodology.

No.	Subject	Summary Response
I-1	Legacy LOCA method	Re-submittal incorporates resolution of the previous RAI response. Aside from EMF-2087 no legacy methods need to be removed from TS COLR methods list.
I-2	Deviations from EMF-2103	Revised TS markup in re-submittal responds to RAI. ANP-3011 contains restrictions on the use of EMF-2103.
I-3	Review of Other COLR methodologies.	Response to RAI explains use of other COLR methods currently listed in TS. No other changes needed to be made as discussed in Reference 9.
I-4	Pellet thermal conductivity degradation (RODEX)	Previous response incorporated into ANP-3011 and expanded based on RAI II-1, -6, -7, -8 and -9. ANP-3011 section 6.1 presents expanded discussion and response to outstanding RAI.
I-5	TS 3.4.3 LCO for Pressurizer level versus sampled range.	Response to RAI was unacceptable to NRC; new RAI II-10 was generated. Refer to RAI-II-10 below.
I-6	Safety Injection Accumulator level versus volume.	Confirmation docketed in Reference 9 and was directly responded to NRC request.

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No.	Subject	Summary Response
I-7	Rod to Rod radiation model	Response to I-7 & -8 are understood to be acceptable to NRC per Feb 14, 2011 telecon and absence of additional RAI on this subject. Information is repeated in ANP-3011, Section 5.3.1.
I-8	Rod to Rod radiation model	See entry for I-7
I-9	Single failure as applied to downcomer boiling	Initial response is understood to be acceptable to NRC per Feb 14, 2011 telecon and absence of additional RAI on this subject. ANP-3011 Section 5.5 discusses downcomer boiling but does not repeat the 59 PCT curves presented in Reference 9.
I-10	Decay Heat sampling	RAI I-10 led to RAI II-5. Refer to ANP-3011 Section 6.11 for changes made in response to these RAIs.
II-1 a) and b)	Demonstrate "time-in-life modeling" is bounding	Revised sensitivity studies were not performed. ANP-3011 directly models fresh and once burn rods as illustrated throughout the document. Additional discussion of pellet thermal conductivity issues discussed in IN 2009-23 presented in Section 6.1
II-2	Demonstrate clad balloon and rupture do not occur	Balloon and rupture predicted. See entry for II-3
II-3	Calculate the effect of clad balloon and rupture impact on PCT and local oxidation.	An HNP sensitivity study of clad balloon and rupture with regard to PCT and local oxidation is provided in Section 6.9.
II-4	Application of IN98-29 (pre-accident oxidation)	ANP-3011 includes pre-accident oxidation in calculated total local oxidation (Table 3-5).

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No.	Subject	Summary Response
II-5	Decay Heat model	Refer to ANP-3011 Section 6.11 for changes made in response to these RAIs.
II-6	Provide additional detail on fuel centerline temperature determination	Section 6.1 provides a step by step discussion of how the fuel centerline temperature determination.
II-7	Provide a scatter plot of adjustment to fuel centerline temperature as a function of burn up.	See II-6
II-8	Provide additional plot of PCT node detail	See response to RAI II-6.
II-9	Provide radial temperature profiles for the hot rod	See response to RAI II-6.
II-10	Pressurizer level sampled range	Sample Pressurizer level distribution changed to Gaussian with a 7.4% standard deviation. Sampling up to 92 % permitted. Refer to ANP-3011 Section Table 3-3 and Figure 3-6.

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ENCLOSURE 2
AREVA AFFIDAVIT FOR WITHHOLDING OF PROPRIETARY DATA
(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. (AREVA NP) 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-3011(P), Revision 001, entitled "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," dated August 2011 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 secret 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 5th

day of August 2011.

Kathleen A. Bennett

Kathleen Ann Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/15
Reg. # 110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2015