Progress Energy

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JUL 21 2009

Serial: HNP-09-055
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

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 TS 6.9.1.6 to add NRC-approved topical report EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," to the Core Operating Limits Report methodologies list. This change will allow the use of thermal-hydraulic analysis code S-RELAP5 for Chapter 15 non loss-of-coolant accident (LOCA) transients in the HNP safety analyses.

Topical Report (TR) EMF-2310(P)(A), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," was approved by the NRC on May 11, 2001, for the application of the S-RELAP5 thermal-hydraulic analysis computer code to Chapter 15 non-LOCA transients. EMF-2310(P)(A), Revision 1, approved by the NRC on May 19, 2004, updated Section 5.6 of the TR.

The NRC Safety Evaluation on the use of the S-RELAP5 code contained a restriction that specific plant submittals include justification of the nodalization used, input parameters, options selected, and all of the parameters that influence the progressions of the event and its mitigation. Accordingly, this submittal contains HNP's plant-specific analysis ANP-2693, "Loss of Forced Reactor Coolant Flow Analysis for Harris Nuclear Plant."

HNP requests approval of the proposed License Amendment by July 2010, with implementation to occur within 60 days of approval. This requested approval date has been administratively selected to accommodate a normal NRC review time.

A001
NRR

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

This document contains no new Regulatory Commitment.

Please refer any question regarding this submittal to Mr. Dave Corlett at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on [JUL 21 2009].

Sincerely,

Christopher L. Burton

Christopher L. Burton
Vice President
Harris Nuclear Plant

CLB/kms

Enclosures:
1. Evaluation of the Proposed Change
2. Affidavit for Withholding of Proprietary Information
3. AREVA Report No. ANP-2693(P), Revision 0 (Proprietary)
4. AREVA Report No. ANP-2693(NP), Revision 0 (Non-Proprietary Version)

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
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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.n; relocate current Technical Specification 6.9.1.6.2.n to 6.9.1.6.2.o.*

1. SUMMARY DESCRIPTION

2. DETAILED DESCRIPTION

2.1 EMF-2310(P)(A), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

2.2 EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

3. TECHNICAL EVALUATION

3.1 Identification of Event

3.2 Justification of Nodalization

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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 (HNP), Unit No. 1.

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 that 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 a 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

HNP TS 6.9.1.6.2.b currently authorizes the use of ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, " to determine the core operating limits. HNP is submitting this License Amendment Request (LAR) for approval of the NRC-accepted Topical Report (TR) EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," as a Core Operating Limits Report (COLR) reference to HNP TS.

EMF-2310(P)(A) contains the application of the S-RELAP5 thermal-hydraulic analysis computer code to Chapter 15 non loss-of-coolant (LOCA) transients. S-RELAP5, which is an updated version of NRC approved ANF-RELAP, has been approved by the NRC as a replacement for the ANF-RELAP code.

The addition of EMF-2310(P)(A), Revision 1 as an authorized COLR reference for HNP will allow the use of the S-RELAP5 thermal-hydraulic analysis code methodology for Chapter 15 non-LOCA accidents in the HNP safety analyses. TR EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," will then be added to HNP TS as 6.9.1.6.2.n. The core operating limits to be developed for HNP using the methodologies approved in EMF-2310(P)(A), Revision 1, will be established in accordance with the applicable limitations as documented in the NRC Safety Evaluation (SE) Reports for EMF-2310(P)(A).

The current TS 6.9.1.6.2.n, "Mechanical Design Methodologies," will be re-designated as 6.9.1.6.2.o, in order to retain the current grouping of the reference reports.

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2.1 EMF-2310(P)(A), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

NRC's review and approval of EMF-2310(P)(A), Revision 0 for use in determining core operating limits authorizes the S-RELAP5 code in the analysis of certain non-LOCA events in pressurized water reactors. Specifically, the NRC concluded in its letter and Safety Evaluation (SE) dated May 11, 2001 (Reference 1), for TR EMF-2310(P)(A), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" that:

> "The staff finds that the assessment performed in support of the S-RELAP5 application to Chapter 15 non-LOCA events is adequate in that it compares code results with ANF-RELAP results for the selected LOFT transients and for plant calculations. . . .
>
> The staff concludes that the S-RELAP5 code is capable of addressing the thermal-hydraulic response of the target non-LOCA events in a conservative manner and is, therefore, an acceptable replacement for the ANF-RELAP code."

However, the NRC's acceptance 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 is expected to include, as a minimum, the nodalization, defense of the chosen parameters, any needed sensitivity studies, justification of the conservative nature of the input parameters and calculated results (Reference 1). This requirement for plant specific demonstration of the applicability of the TR was subsequently clarified by the NRC in a letter issued November 13, 2003 (Reference 5).

In accordance with the above requirement, this submittal includes both proprietary and non-proprietary, designated as P (proprietary) and NP (non-proprietary), versions of AREVA NP Inc. report ANP-2693, "Loss of Forced Reactor Coolant Flow Analysis for Harris Nuclear Plant, Unit 1."

2.2 EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

EMF-2310(P)(A), Revision 1 updated Section 5.6, "CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)," in the following three areas:

- The Dilution Front Model will be used when the Residual Heat Removal (RHR) system is in operation

- All control rods will be assumed to be inserted in Modes 4 and 5
- Complete mixing of the fluid is assumed prior to entry of the diluted fluid into the core

The associated SE, approved by the NRC on May 18, 2004 (Reference 2), concluded that the S-RELAP5 systems analysis code, as previously reviewed and approved by the NRC, remained acceptable for use in the analysis of Chapter 15 non-LOCA events.

NRC review of the provided EMF-2310(P)(A), Revision 1 material did not duplicate the analysis of the material previously accepted in the initial TR review, concluding only that the methodology in Revision 1 was capable of addressing the thermal-hydraulic response of the boron dilution event in a conservative manner. Since the original requirement for a plant specific demonstration of the applicability of the S-RELAP5 code was not modified in EMF-2310(P) Revision 1, such an analysis is still needed in a Licensee's request for approval to use S-RELAP5.

3. TECHNICAL EVALUATION

A Loss of Forced Reactor Coolant Flow (LOCF) event analysis, which is HNP FSAR Event 15.3.2, has been chosen to demonstrate the specific application of the S-RELAP5 model to HNP. Since LOCF is included in the Table 1 ("Applicable SRP Chapter 15 Events") of the SE (Reference 1), it is an acceptable non-LOCA event that may be analyzed using the S-RELAP5 methodology.

Enclosures 4 and 5 of this submittal are AREVA NP Inc. report ANP-2693, Revision 0, "Loss of Forced Reactor Coolant Flow Analysis for Harris Nuclear Plant, Unit 1." This report provides the plant specific analysis of the LOCF event for HNP performed using the S-RELAP5 computer code, demonstrating the application of the approved TR (Reference 1) methodology to HNP.

3.1 Identification of Event

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the event, the immediate effect of the loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature, which could result in Departure from Nuclear Boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

A reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. A reactor trip on reactor coolant pump undervoltage is provided to protect against

conditions which can cause a loss of voltage to all reactor coolant pumps, such as loss of offsite power. In accordance with FSAR Section 15.3.2.2, the AREVA report on the LOCF event for HNP considers the underfrequency event on the power supply to the reactor coolant pumps as more limiting than the pump power supply undervoltage trip. Therefore, due to a more rapid flow coastdown, the underfrequency initiating event bounds an undervoltage initiating event.

3.2 Justification of Nodalization

Figures 3.1 to 3.3 of the AREVA report contain the reactor vessel, Reactor Coolant System (RCS) piping and steam generator nodalization diagrams based on a HNP plant specific model. The 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 and a Reactor Coolant Pump (RCP). 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.

3.3 Chosen Parameters

The parameters and equipment states used in the AREVA analysis were chosen to provide a conservative estimate of the challenge to DNB, with biasing and assumptions for key input parameters consistent with the NRC approved methodology. Since the margin to the DNB limit is minimized at the beginning of the event, the event is analyzed from full power initial conditions. The analysis is performed with assumptions and models that are designed to minimize the departure from nucleate boiling ratio (DNBR). Table 3.1 of the AREVA report contains the key assumptions and Table 3.2 the biasing of key parameters, both consistent with the approved methodology of the sample problem.

3.4 Sensitivity Studies

This event is controlled primarily by the primary system flow coast down. The S-RELAP5 code assessments in Reference 2 validate the model relative to this controlling parameter. The biasing of input parameters is chosen to produce a conservative estimate of the challenge to DNB for this application. Thus, no additional input parameter sensitivity studies are needed for this application.

3.5 Results

The sequence of events and the results for the underfrequency event are provided in Table 4.1 of the AREVA report, with responses to key system variables in Figures 4.1 to 4.7. The results of the LOCF for the underfrequency case using the S-RELAP5 model compare to the values obtained using the ANF-RELAP model, as provided in FSAR Section 15.3.2. Modeling of the pressurizer power-operated relief valves (PORV) operation, which tends to keep the core pressure low, reduces the core thermal margin as represented by the DNBR. The results indicate that the PORVs open and the transient DNBR does not fall below the safety limit DNBR. Based on these results, the acceptance criteria for this event are satisfied by the use of the S-RELAP5 model.

AREVA NP Inc. report ANP-2693 concludes that the event acceptance criteria are met based on the results of this analysis. Since the core power does not increase appreciably during this event, the challenge to the fuel centerline melt Specified Acceptable Fuel Design Limit (SAFDL) is not limiting and margin exists to the DNB SAFDL. Additionally, because system temperatures and pressures increase less significantly for a loss of flow event compared to complete loss of load type events, the pressurization transient does not present a severe challenge to the maximum pressure criterion. Therefore, the use of S-RELAP5 is successful in meeting the event acceptance criteria.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

TR EMF-2310(P)(A) pertains to non-LOCA accident and transient analyses that are part of the HNP licensing basis. The regulatory bases for these analyses are found in the General Design Criteria (GDC) (Reference 3). The GDCs that pertain to each of the analyses are listed in the Standard Review Plan (SRP) (Reference 4).

The definition of evaluation models of LOCA events per 10 CFR 50.46, which can also be applied to non-LOCA analysis, 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, written for LOCA analyses but also considered applicable to non-LOCA analyses, 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-2310(P)(A) as provided in the NRC SE on the topical report. HNP has not taken any deviations from the NRC approved SEs for TR EMF-2310(P)(A).

The AREVA analysis submitted (Enclosures 4 and 5) satisfy the NRC limitations contained in the SE regarding site-specific application for approval of EMF-2310(P)(A). Therefore, the addition of EMF-2310(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. 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 Reports. In the May 11, 2001, NRC SE, the NRC concluded that the S-RELAP5 code is capable of addressing the thermal-hydraulic response of the target non-LOCA events in a conservative manner and is therefore an acceptable replacement for the ANF-RELAP code. The May 19, 2004, SE for Revision 1 to EMF-2310(P)(A), concluded that the code remained acceptable for use for the non-LOCA events.

   The proposed change, by itself, does not impact the current design bases. The proposed change enables the use of new methodology to re-analyze certain

events. Revised analyses 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, other than allowing for fuel design in accordance with NRC approved methodologies. No new or different equipment is being installed. 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).

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.

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6. REFERENCES

1. Letter from NRC to Framatome ANP, Acceptance for Referencing of Licensing Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" (TAC NO. MA7192), May 11, 2001 (ML011310533)

2. Letter from NRC to Framatome ANP, Final Safety Evaluation for Topical Report EMF-2310(P), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" (TAC NO. MC0329), May 19, 2004 (ML041400499)

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

4. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, April 1996 (ML033580677)

5. Letter from NRC to Framatome ANP, Clarification of Safety Evaluation for EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" (TAC NO. MB7338), November 13, 2003

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ATTACHMENT 1
TECHNICAL SPECIFICATION PAGE MARKUPS
(4 Pages)

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).

n. INSERT A FROM NEXT PAGE

INSERT A - TS 6.9.1.6.n as follows:

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

O.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,

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

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

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(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

ATTACHMENT 2
RETYPED TECHNICAL SPECIFICATION PAGES
(3 Pages)

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).

n. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", approved version as specified in the COLR.

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).

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:

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c. Nondestructive examination techniques utilized for each degradation mechanism,

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

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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SPECIAL REPORTS

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

AREVA
AFFIDAVIT PURSUANT TO 10 CFR 2.390
For AREVA Report No. ANP-2693(P), Revision 0 (Proprietary)
(3 Pages – Single Sided)

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-2693(P), Revision 0, "Loss of Forced Reactor Coolant Flow Analysis for Harris Nuclear Plant, Unit 1," dated May 2009 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 7th

day of May 2009.

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