Progress Energy

10 CFR 50.90

Serial: RNP-RA/09-0103

DEC 16 2009

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
REGARDING STEAM GENERATOR ALTERNATE REPAIR CRITERIA

Ladies and Gentlemen:

In accordance with the provisions of the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power and Light Company, also known as Progress Energy Carolinas (PEC), Inc., is submitting a request for an amendment to the Technical Specifications (TS) contained in Appendix A of the Operating License for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed amendment would modify TS 5.5.9 to revise the steam generator (SG) alternate repair criteria and TS 5.6.8 to revise the SG tube inspection reporting requirements.

Specifically, the proposed change would revise requirements in TS Section 5.5.9, called alternate repair criteria, which would allow inspection of the tube to start within the tubesheet region (a minimum of 17.28 inches below the top of the tubesheet) and add requirements to report indications in this region and primary to secondary leakage that could be attributed to the uninspected portion of the tube within the tubesheet. The change is being proposed as a temporary requirement until the next scheduled inspection.

Attachment I provides an Affirmation pursuant to 10 CFR 50.30(b).

Attachment II provides a description of the proposed change, a technical justification for the proposed change, a summary of commitments, a No Significant Hazards Consideration Determination, and an Environmental Impact Consideration.

Attachment III provides a markup of the current TS pages and Attachment IV provides retyped pages for the proposed TS.

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A001
NRR

A plant-specific analysis for HBRSEP, Unit No. 2, is provided in Attachments V and VI. Attachment V provides a non-proprietary version of the report WCAP-17091-NP, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)," that can be released for public disclosure. Attachment VI provides a proprietary version of the report, designated WCAP-17091-P, that should be withheld from public disclosure. Attachment VII provides an Affidavit from Westinghouse regarding the proprietary nature of WCAP-17091-P, as required by 10 CFR 2.390.

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

PEC requests approval of the proposed license amendment by April 16, 2010, to allow for the use of the alternate repair criteria for inspections planned during the next refueling outage, which is currently scheduled to begin on April 17, 2010.

If you have any questions concerning this matter, please contact Mr. Curt Castell at (843) 857-1626.

Sincerely,

B. C. White
Manager - Support Services - Nuclear

BCW/cac

Attachments:
I. Affirmation
II. Request for Technical Specifications Change Regarding Steam Generator Alternate Repair Criteria
III. Markup of Technical Specifications Pages
IV. Retyped Technical Specifications Pages
V. WCAP-17091-NP, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)" – Non-Proprietary
VI. WCAP-17091-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)" – Proprietary
VII. Westinghouse Affidavit Regarding Proprietary WCAP-17091-P

c: Ms. S. E. Jenkins, Manager, Infectious and Radioactive Waste Management Section (SC)
Mr. A. Gantt, Chief, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. T. Orf, NRC Project Manager, NRR
NRC Resident Inspectors, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/09-0103 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 12/16/09

B. C. White
Manager - Support Services – Nuclear
HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING STEAM GENERATOR ALTERNATE REPAIR CRITERIA

Description of the Current Condition and the Proposed Change

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., is proposing a change to the Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

The H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Technical Specifications Section 5.5.9 specifies the repair criteria requirements for the Steam Generator (SG) Program, as follows:

"c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding the following criteria shall be plugged: 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the repair criteria for each 12 month period until the next inspection of the tube.

The following alternate tube repair criteria shall be applied as an alternative to the preceding criteria, until the end of Operating Cycle 25:

Flaws found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging. Tubes with flaws identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet shall be plugged upon detection."

TS 5.5.9.d states the provisions for the tube inspections, as follows:

"d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet (until the end of Operating Cycle 25 the required inspection length extends 17 inches below the top of the tubesheet on the tube hot leg side to 17 inches below the top of the tubesheet on the tube cold leg side), and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations."

Subsection 3. of TS 5.5.9.d states:

"3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack."

Additionally, TS 5.6.8 provides the requirements for the steam generator tube inspection report, as follows:

"h. The number of indications including location, size, and orientation, and whether the indications initiated in the primary or secondary side of the tube for each indication detected in the upper 17 inches of the tubesheet region of the tube.

i. The operational cycle primary to secondary leakage rate observed in each SG during the cycle preceding the tube inspection that is the subject of this report and the corresponding calculated accident leakage from the lower 4.81 inches of the tube for the most-limiting accident in the most-limiting SG. If the calculated accident leakage rate for any steam generator is less than two times the total observed operational primary to secondary leakage rate, the report should describe how it was determined."

Alternate repair criteria are proposed to be revised in TS 5.5.9.c, as follows:

"c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding the following criteria shall be plugged: 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the repair criteria for each 12 month period until the next inspection of the tube.

The following alternate tube repair criteria shall be applied as an alternative to the preceding criteria, until the end of Operating Cycle ~~25~~**27**:

~~Flaws found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging. Tubes with flaws identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet shall be plugged upon detection~~ **Tubes with service-induced flaws located greater than 17.28 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 17.28 inches below the top of the tubesheet shall be plugged upon detection**."

The requirements of TS 5.5.9.d are proposed to be revised, as follows:

"d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet (until the end of Operating Cycle ~~25~~**27** the required inspection length extends 17**.28** inches below the top of the tubesheet on the tube hot leg side to 17**.28** inches below the top of the tubesheet on the tube cold leg side), and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations."

Subsection 3 of TS 5.5.9.d is proposed to be revised, as follows:

"3. If crack indications are found in any **portion of a** SG tube **not excluded above**, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack."

The reporting requirements of TS 5.6.8.h and i are proposed to be revised by replacement with three requirements h, i, and j, as follows:

"h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection that is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 17.28 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided."

Technical Justification for the Proposed Change

HBRSEP, Unit No. 2, is currently in Operating Cycle 26. The unit has three Westinghouse Model 44F steam generators, which were installed as replacement steam generators in 1984. There are 3214 thermally treated Alloy 600 tubes in each of the steam generators. The tubes have an outside diameter of 0.875 inch, an average wall thickness of 0.050 inch, and there are six stainless steel tube support plates and a flow distribution baffle. The tube support plate holes are quatrefoil shaped and the flow distribution baffle holes are round. A total of 32 tubes have been plugged.

HBRSEP, Unit No. 2, has typically used bobbin probes for inspecting the length of tubing within the tubesheet; however, the bobbin probe is not capable of reliably detecting stress corrosion cracks (SCCs) in the tubesheet region, should such cracks be present. For this reason, the bobbin probe inspections have been supplemented with rotating coil probes in a region extending from 4 inches above the top of the tubesheet to 2 inches below the top of the tubesheet. This zone includes the tube expansion transition zone located at the top of the tubesheet. The expansion transition zone contains residual stresses, which are considered likely locations for SCC, should it develop.

Prior to the fall of 2004, no instances of stress corrosion cracking affecting the tubesheet region of thermally treated Alloy 600 tubing had been reported at any nuclear power plants in the United States. However, in the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station Unit 2 (Catawba), which utilized a different SG design (Model D5 SGs). Similar to the SGs at HBRSEP, Unit No. 2, the Catawba SGs use thermally treated Alloy 600 tubing that were hydraulically expanded against the tubesheet. The crack-like indications at Catawba were found in a tube overexpansion (OXP), in the tack expansion region, and near the tube-to-tubesheet weld. An OXP is created when the tube is expanded into a tubesheet bore hole that is not perfectly round. These out-of-round conditions were created during the tubesheet manufacturing as a result of drill bit wandering or chip gouging. The tack expansion region is an approximately 1-inch long expansion at each tube end. The purpose of the tack expansion is to facilitate performing the tube-to-tubesheet weld, which is made prior to the hydraulic expansion of the tube over the full tubesheet depth.

Since the initial findings at Catawba in the fall of 2004, other nuclear plants have found crack-like indications in tubes within the tubesheet. These plants include Braidwood Unit 2, Byron Unit 2, Comanche Peak Unit 2, Surry Unit 2, Vogtle Unit 1, and Wolf Creek Unit 1. Most of the crack-like indications were found in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary water stress corrosion cracking.

The NRC previously granted License Amendment No. 214 for HBRSEP, Unit No. 2, to exclude the portion of the tubes below 17 inches from the top of the hot leg tubesheet on a one-time basis. This amendment expired at the end of Operating Cycle 25. The unit is currently operating in Cycle 26 with a planned SG inspection during Refueling Outage 26, which is scheduled to begin on April 17, 2010.

The proposed changes will renew and slightly change the exclusion distance for the tube examination and plugging requirements within the tubesheet region of the steam generator. These changes are based on the technical justifications contained in the report WCAP-17091-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)," June 2009. Non-proprietary and proprietary versions of that report are provided in Attachments V and VI of this letter.

The evaluation in WCAP-17091-P is based on the use of finite element model structural analysis and a bounding leak rate evaluation based on contact pressure between the tube and the tubesheet during normal and postulated accident conditions. The limited tubesheet inspection criteria were developed for the tubesheet region of the HBRSEP, Unit No. 2, Model 44F SG considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited tubesheet inspection criteria were selected to prevent pull out of a severed tube from the tubesheet due to axial end cap loads acting on the tube and to ensure that the accident induced leakage limits are not exceeded. WCAP-17091-P provides the technical justification for limiting the inspection in the tubesheet expansion region to less than the full depth of the tubesheet.

The determination of the portion of the tube that requires eddy current inspection within the tubesheet is based upon evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in WCAP-17091-P. The tube-to-tubesheet radial contact pressure provides resistance to tube pull out and resistance to leakage during plant operation, including transients and postulated accident conditions. The constraint that is provided by the tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are satisfied by the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope as described below, the existing operational leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur.

Primary to secondary leakage from tube degradation is assumed to occur in several design basis accidents: Steam line break (SLB), locked rotor, and control rod ejection events. The radiological dose consequences associated with this assumed leakage are evaluated to ensure that they remain within regulatory limits (e.g. 10 CFR Part 100, 10 CFR 50.67, GDC 19). The accident induced leakage performance criteria are intended to ensure the primary to secondary leak rate during any accident does not exceed the primary to secondary leak rate assumed in the accident analysis. Radiological dose consequences define the limiting accident condition for the H* value. The current UFSAR assumed primary to secondary accident induced leak rate is 150 gallons per day (gpd) through any one SG for SLB, and 0.3 gpm total for each of the rod ejection and locked rotor events. Thus, the limiting accident is SLB. It should be noted that both of the above leak rate values, are at accident conditions.

The leak rate ratio (accident induced leak rate to operational leak rate) is directly proportional to the change in differential pressure and inversely proportional to the dynamic viscosity. An increase in temperature results in an increase in leak rate because dynamic viscosity decreases with an increase in temperature. However, for the postulated SLB event, a plant cool down event would occur and the subsequent temperature in the reactor coolant system (RCS) would not be expected to exceed the temperatures at plant no load conditions. Thus, an increase in leakage would not be expected to occur as a result of the temperature change. The increase in leakage would only be a function of the increase in primary to secondary pressure differential. The resulting leak rate ratio for the SLB event is 1.82 for HBRSEP, Unit No. 2 (Table 9-7 of WCAP-17091-P).

The SLB leak rate factor for HBRSEP, Unit No. 2, is 1.82 (Table 9-7 in WCAP-17091-P). Multiplying this factor by the TS operational leak rate limit of 75 gpd through any one SG indicates that an assumed primary to secondary accident induced leak rate of 136.5 gpd or greater through any one SG is required to ensure that the limiting design basis accident assumption is not exceeded. This condition is satisfied by the current UFSAR assumed primary to secondary accident induced leak rate of 150 gpd through any one SG for SLB.

The other design basis accidents, such as the postulated locked rotor event and the control rod ejection event, are conservatively modeled using the design specification transients that result in increased temperatures in the SG hot and cold legs for a period of time. Leakage would be expected to increase due to decreasing viscosity and increasing differential pressure for the duration of time that there is a rise in RCS temperature. For transients other than a SLB, the length of time that a plant with Model 44F SGs will exceed the normal operating differential pressure across the tubesheet is less than 30 seconds for the locked rotor event, and less than 10 seconds for the control rod ejection event. As the accident induced leakage performance criteria is defined in gallons per minute, the leak rate for a locked rotor event can be integrated over a minute for comparison to the limit. Time integration permits an increase in acceptable leakage during the time of peak pressure differential by approximately a factor of two for the locked rotor event because of the short duration (less than 30 seconds) of the elevated pressure differential, and by a factor of six for the control rod ejection event (less than 10 seconds). This translates into an effective reduction in the leakage factor by the same factor for each event. Therefore, the locked rotor event leak rate factor of 1.56 for HBRSEP, Unit No. 2, is adjusted downward to a factor of 0.78 (Table 9-7). Similarly, the control rod ejection event leak rate factor is reduced by a factor of six, from 2.21 to 0.37 (Table 9-7). Due to the short duration of the transients above normal operating pressure differential, no leakage factor is required for the locked rotor and control rod ejection events (i.e., the leakage factor is under 1.0 for both transients). Thus, SLB remains the limiting accident and 1.82 remains the limiting leak rate factor for HBRSEP, Unit No. 2 (Table 9-7 in WCAP-17091-P).

Plant-specific operating conditions are used to generate the overall leakage factor ratios that are used in the condition monitoring and operational assessments. The plant-specific data provide the initial conditions for application of the transient input data. The results of the analysis of the plant-specific inputs to determine the bounding plant for each model of SG and to assure that the design basis accident contact pressures are greater than the normal operating contact pressure are contained in Section 6 of WCAP-17091-P.

The leak rate factor of 1.82 for a postulated SLB has been calculated as shown in Table 9-7. A factor of 1.82 will be applicable to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). The leak rate factor of 1.82 in Table 9-7 applies to both hot and cold legs. Specifically, for the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.82 and added to the total leakage from any other source and compared to the assumed accident induced leak rate. For the OA, the difference between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.82 and compared to the observed operational leakage.

WCAP-17091-P redefines the primary pressure boundary. The tube to tubesheet weld no longer functions as a portion of this boundary. The hydraulic expansion of the tube from the top of the tubesheet down to 17.28 inches below the top of the tubesheet now functions as the primary pressure boundary in the area of the tube and tubesheet, maintaining the structural and leakage integrity over the full range of SG operating conditions, including the most limiting accident conditions. The evaluation in WCAP-17091-P determined that degradation in tubing below 13.31 inches from the top of the tubesheet does not require inspection or repair (plugging). The inspection of the portion of the tubes above 13.31 inches below the top of hot and cold tubesheet for tubes that have been hydraulically expanded in the tubesheet provides a high level of confidence that the structural and leakage performance criteria are maintained during normal operating and accident conditions. WCAP-17091-P, Section 8.0, recommended a final H* value of 13.31 inches from the top of the tubesheet for the entire bundle of tubes. However, for conservatism an H* value of 17.28 inches has been chosen based on applying statistical evaluation to the 13.31 inch result.

The additional reporting requirements being added to TS 5.6.8 will establish the appropriate reporting criteria for the tubes that required repair under the proposed alternate repair criteria and a quantification of the operational and accident-induced leakage that could potentially be attributable to the uninspected region of the SG tubes.

The proposed change will establish appropriate inspection requirements to ensure SG integrity under normal operation and accident conditions for HBRSEP, Unit No. 2. These changes will allow the inspections of the HBRSEP, Unit No. 2, SGs to be conducted appropriately and prevent unnecessary plugging of SG tubes that should not be removed from service.

Additionally, the following commitments will be implemented with this license amendment:

- Monitoring for tube slippage as part of the steam generator tube inspection program will be conducted.

- A one-time verification of the expansion locations will be conducted to determine if any significant deviations exist from the top of the tubesheet to the bottom of the expansion transion (BET). If any significant deviations are found, the condition will be entered into the plant corrective action program and be dispositioned.

- Areas of inservice SG tubes within the tubesheet that are not fully expanded will be included in the scheduled inspection and the 17.28 inch tubesheet inspection limitation will not be applied to the areas of these tubes that are not fully expanded.

No Significant Hazards Consideration Determination

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., is proposing a change to the Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change will revise TS 5.5.9, Steam Generator (SG) Program, and TS 5.6.8, Steam Generator Tube Inspection Report, to incorporate requirements that will allow the use of alternate repair criteria that establishes an appropriate scope of the tube inspection and repair for the portion of the tube within the tubesheet region.

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   The proposed change does not involve physical changes to any plant structure, system, or component. The inspection of the portion of the steam generator tubes within the tubesheet region is being changed to identify the appropriate scope of inspection and the criteria for plugging tubes that are found with degradation. The proposed requirements will continue to ensure that the probability of a steam generator tube rupture accident is not increased. 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 inspection and repair requirements 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. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed accident.

   Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

   The proposed change does not involve any physical alteration of plant systems, structures, or components. 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. The proposed inspection and repair criteria will establish appropriate requirements to ensure that the steam generator tubes are properly maintained. As a result, no new failure modes are being introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

   The proposed change defines the safety significant portion of the tube that must be inspected and repaired. WCAP-17091-P identifies the specific inspection depth below which any type of tube degradation is shown to have no impact on the performance criteria in NEI 97-06 Rev. 2, "Steam Generator Program Guidelines" and TS 5.5.9, "Steam Generator (SG) Program."

   The proposed change that alters the SG inspection and reporting criteria maintains the required structural margins of the SG tubes for normal, transient, and accident conditions. Nuclear Energy Institute 97-06, "Steam Generator Program Guidelines," and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and GDC 32, "Inspection of Reactor Coolant Pressure Boundary," by reducing the probability and consequences of a SGTR. The probability and consequences of a SGTR are reduced by establishing the limiting safe conditions for tube wall degradation. RG 1.121 uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

   For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse report WCAP-17091-P defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary to secondary leakage during applicable plant conditions. The steam line break accident leak rate factor for HBRSEP, Unit No. 2, is 1.82 (Table 9-7 in WCAP-17091-P). Multiplying this factor by the room temperature TS operational leak rate limit of 75 gpd through any one SG indicates that an assumed primary to secondary accident induced leak rate of 136.5 gpd or greater through any one SG is required to ensure that the limiting design basis accident assumption is not exceeded (at room temperature). This condition is satisfied by the current UFSAR assumed primary to secondary accident induced leak rate of 150 gpd through any one SG for SLB.

   Therefore, the proposed change does not involve a significant reduction in any margin of safety.

Based on the above discussion, Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., has determined that the requested change does not involve a significant hazards consideration.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., has reviewed this request and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

Proposed Change

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., is proposing a change to the Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change will revise TS 5.5.9, Steam Generator (SG) Program, and TS 5.6.8, Steam Generator Tube Inspection Report, to incorporate requirements that will allow the use of alternate repair criteria that establishes the appropriate scope of the tube inspection and repair for the portion of the tube within the tubesheet region.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. The use of the proposed alternate repair criteria has no negative impact on effluent releases. The limits on primary to secondary leakage are not being increased. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The proposed change does not involve physical plant changes, or introduce any new mode of plant operation. The inspection requirements will not change the access requirements for these inspections such that radiation exposure would be increased. Therefore, the proposed change does not result in a significant increase in individual or cumulative occupational radiation exposures.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/09-0103
5 pages including cover page

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING <u>STEAM GENERATOR ALTERNATE REPAIR CRITERIA</u>

MARKUP OF TECHNICAL SPECIFICATIONS PAGES

Tubes with service-induced flaws located greater than 17.28 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 17.28 inches below the top of the tubesheet shall be plugged upon detection.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 75 gallons per day per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding the following criteria shall be plugged: 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the repair criteria for each 12 month period until the next inspection of the tube.

The following alternate tube repair criteria shall be applied as an alternative to the preceding criteria, until the end of Operating Cycle ~~25~~ 27:

~~Flaws found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging. Tubes with flaws identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet shall be plugged upon detection.~~

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet (until the end of Operating Cycle ~~25~~ 27 the required inspection length extends 17.28 inches below the top of the tubesheet on the tube hot leg side to 17.28 inches below the top of the tubesheet on the tube cold leg side), and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program
(continued)

integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

not excluded above

portion of a

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of critical parameters, their sampling frequency, sampling points, and control band limits;

(continued)

5.6 Reporting Requirements (continued)

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. 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.

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

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

Replace with h., i., and j. on the next page

~~h. The number of indications including location, size, and orientation, and whether the indications initiated in the primary or secondary side of the tube for each indication detected in the upper 17 inches of the tubesheet region of the tube.~~

~~i. The operational cycle primary to secondary leakage rate observed in each SG during the cycle preceding the tube inspection that is the subject of this report and the corresponding calculated accident leakage from the lower 4.81 inches of the tube for the most-limiting accident in the most-limiting SG. If the calculated accident leakage rate for any steam generator is less than two times the total observed operational primary to secondary leakage rate, the report should describe how it was determined.~~

<Insert new requirements h., i., and j.>

h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection that is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 17.28 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/09-0103
5 pages including cover page

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING <u>STEAM GENERATOR ALTERNATE REPAIR CRITERIA</u>

RETYPED TECHNICAL SPECIFICATIONS PAGES

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program
(continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 75 gallons per day per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding the following criteria shall be plugged: 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the repair criteria for each 12 month period until the next inspection of the tube.

The following alternate tube repair criteria shall be applied as an alternative to the preceding criteria, until the end of Operating Cycle 27:

Tubes with service-induced flaws located greater than 17.28 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 17.28 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet (until the end of Operating Cycle 27 the required inspection length extends 17.28 inches below the top of the tubesheet on the tube hot leg side to 17.28 inches below the top of the tubesheet on the tube cold leg side), and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program
(continued)

integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of critical parameters, their sampling frequency, sampling points, and control band limits;

(continued)

5.6 Reporting Requirements (continued)

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

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5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. 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.

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

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

h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection that is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 17.28 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

5.6 Reporting Requirements (continued)

5.6.8 Steam Generator Tube Inspection Report (continued)

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

United States Nuclear Regulatory Commission
Attachment VII to Serial: RNP-RA/09-0103
8 pages including cover page

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING <u>STEAM GENERATOR ALTERNATE REPAIR CRITERIA</u>

Westinghouse Affidavit Regarding Proprietary WCAP-17091-P

Westinghouse

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Our ref: CAW-09-2599

June 12, 2009

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-17091-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)," dated June 2009 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-09-2599 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Progress Energy.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-09-2599, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

W. J. Lovely / FOR
J.A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: G. Bacuta (NRC OWFN 12E-1)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared R. M. Span, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

R. M. Span

R. M. Span, Principal Engineer
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 12th day of June, 2009

Sharon L. Markle

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Principal Engineer, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-17091-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)," dated June 2009 (Proprietary), for submittal to the Commission, being transmitted by Progress Energy Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for H.B. Robinson Unit 2 is expected to be applicable to other licensee submittals in support of implementing an alternate repair criterion, called H*, that does not require an eddy current inspection and plugging of the tubes below a distance of 13.31 inches from the top of the tubesheet.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the analyses, methods, and testing which support the implementation of an alternate repair criterion, designated as H*, for a portion of the tubes within the tubesheet of the H.B. Robinson Unit 2 steam generators.

(b) Assist the customer in obtaining NRC approval of the Technical Specification changes associated with the alternate repair criterion.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for the purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculation, evaluation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.