Union Electric
Callaway Plant

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October 3, 2002

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C.20555-0001

Gentlemen: ULNRC-04745

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
REVISION TO TECHNICAL SPECIFICATION 5.5.9
"STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM"

AmerenUE herewith transmits an application for amendment to Facility Operating License No. NPF-30 for the Callaway Plant.

Callaway's Technical Specifications (TS) require that steam generator tubes be examined in accordance with the provisions of TS 5.5.9. This section defines tube inspection for the steam generator tubes. Callaway considers the bobbin coil exam to meet the requirements for tube inspection. Rotating pancake coil (RPC) exams at the top of the tubesheet have been performed to supplement the bobbin coil exam to ensure tube structural integrity in accordance with NEI 97-06. RPC at the top of the tubesheet has been used for the past 5 refueling outages and has been recognized as a good industry practice and proactive. Callaway will perform top of the tubesheet inspections using RPC, as described in the Westinghouse Topical Report, WCAP-15932-P, during refuel 12.

This proposed license amendment request (LAR) is being submitted per the guidance provided in Administrative Letter (AL) 98-10 as a correction to a Technical Specification (TS) that was found to be non-conservative in nature. This proposed change would revise TS 5.5.9 to clearly delineate the scope of the steam generator tube inspection required in the tubesheet region. The depth of the tube inspection within the tubesheet is based on the evaluations and justifications presented in the attached WCAP. Since we are submitting this LAR per the guidance of AL 98-10, we do not view this LAR as a request that would require an amendment prior to plant startup following Refuel 12.

A001

Attachments 1 through 5 provide the required Affidavit, Evaluation, Markup of TS pages, Retyped TS pages, and topical report.

It has been determined that this amendment application does not involve a significant hazards consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of this amendment.

Westinghouse has determined that information associated with WCAP-15932-P is proprietary, and is thereby supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets 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 2.790. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790.

If you have any questions on this amendment application, please contact Mr. Dave Shafer at (314) 554-3104.

Very truly yours,

John D. Blosser
Manager, Regulatory Affairs

JMC/
Attachments: 1) Affidavit
2) Evaluation
3) Markup of Technical Specification Page
4) Retyped Technical Specification Page
5) a) Topical Report (Proprietary)
b) Topical Report (Non-proprietary)
c) Proprietary Affidavit

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STATE OF MISSOURI)
) S S
COUNTY OF CALLAWAY)

John D. Blosser, of lawful age, being first duly sworn upon oath says that he is Manager, Regulatory Affairs for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By _____
John D. Blosser
Manager, Regulatory Affairs

SUBSCRIBED and sworn to before me this 3rd day of October, 2002.

CATHY J. CRISP
MY COMM. EXP.
NOTARY SEAL
NOTARY PUBLIC
JAN 29, 2006
STATE OF MISSOURI

Cathy J. Crisp
Notary Public
Notary Seal
State of Missouri
Commissioned for Callaway County
Expiration 1-29-06

ULNRC-04745

ATTACHMENT 2

EVALUATION

EVALUATION

1.0 INTRODUCTION

1.1 This proposed License Amendment Request (LAR) pursuant to 10 CFR 50.90 revises Technical Specification (TS) 5.5.9, Steam Generator (SG) Tube Surveillance Program to clearly delineate the scope of the steam generator tube inspection required in the tubesheet region. The definition for tube inspection is provided in TS 5.5.9.h), and this proposed LAR would provide clarification to this definition.

1.2 Final Safety Analysis Report (FSAR) Section

There are no changes to the Callaway Plant FSAR associated with this amendment application.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change would modify the current definition of tube inspection as defined below:

"Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube repaired by sleeving, the tube inspection shall include the sleeved portion of the tube."

The proposed change to the TS clarifies the scope of the tube inspection required for the region within the SG tubesheet. The proposed change excludes the portion of the tube within the tubesheet below a specified distance determined by WCAP-15932-P. The proposed change reads as follows:

"Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg, including inspection with rotating pancake coil (or equivalent) from the tube expansion transition at the top of the tubesheet to the H* depth as specified in Table 5.5.9-4. For a tube repaired by sleeving, the tube inspection shall include the sleeved portion of the tube."

The H* analysis accounts for the reinforcing effect that the tubesheet has on the external surfaces of the SG tube within the tubesheet region. The H*/P* analysis shows that tube integrity and leakage below the H* distance remain within the existing design limits.

3.0 BACKGROUND

The steam generators at Callaway Plant are Westinghouse Model F, vertical shell and U-tube evaporator, with integral moisture separating equipment. The Model F SG is similar in configuration to the Westinghouse Model 51 SGs. The Model F incorporates several improved features. These features include: Preferential distribution of feedwater to the hot leg portion of the tube bundle, removal of downcomer resistance, blockage of the tube lane, and improvements to the primary and secondary steam separators. Each SG contains 5,626 tubes. The SG tubes are Inconel-600, a nickel-chromium-iron alloy with an outside diameter of 0.688-inches and a wall thickness of 0.040-inches. The tubes are seal welded to the tubesheet cladding. These fusion welds were performed in compliance with Sections III and IX of the ASME Code and were dye penetrant inspected and leakproof tested. The tubesheet is approximately 21 inches thick and after welding, each tube was hydraulically expanded for the full depth of the tubesheet to the secondary surface to eliminate crevices between the tube and tubesheet. The seal weld and hydraulic expansion forms the interface which provides the structural and part of the leaktight boundary between the primary and secondary systems at each end of a SG tube.

Callaway Plant currently inspects SG tubes using the eddy current technique with a bobbin coil examination of the full length of the tube from tube end-hot to tube end-cold, this includes the entire depth of the tubesheet. In addition to this examination a rotating pancake coil (RPC) + Point examination is performed in the hot leg top of tubesheet (+2"/-3"). In addition all bobbin I-code indications that are not dispositioned via a history review are RPC tested.

NEED FOR CHANGE

The need for the proposed change is based on a late emerging issue regarding the Steam generator tube inspections and the identification of indications within the tubesheet region. This change is needed to resolve a regulatory issue to clearly specify the method of examining the SG tubes within the tubesheet region and to determine the exact depth to which this examination needs to extend.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The regulatory requirements associated with steam generator tube inspections include the following:

Criterion 14 – Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15 – Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 19 – Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Criterion 31 – Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual steady state and transient stresses, and (4) size of flaws.

Criterion 32 – Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Regulatory Guide 1.83, Revision 1, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes.

Regulatory Guide 1.121, Revision 0, Bases for Plugging Degraded PWR Steam Generator Tubes.

5.0 TECHNICAL ANALYSIS

Design Basis and Safety Analysis Considerations

The steam generators at Callaway Plant are Westinghouse Model F, vertical shell and U-tube evaporator, with integral moisture separating equipment. The Model F SG is similar in configuration to the Westinghouse Model 51 SG. The Model F incorporates several improved features. Each tube is secured in the

tubesheet of the SG by a seal weld and a hydraulic expansion process. The process expands each tube over its entire length of the tubesheet to eliminate crevices between the tube and tubesheet. The seal weld and hydraulic expansion forms the interface which provides the structural and part of the leaktight boundary between the primary and secondary systems at each end of a SG tube. Located near the top of the tubesheet is a region where the tube transitions from the tubesheet hole diameter to that of the original tube. This region is referred to as the transition region.

Justifications have been developed by Westinghouse Electric Company in WCAP-15932-P to reduce the RPC inspection length of Model F steam generator tubes within the tubesheet from full-length to partial-length for the Callaway Plant. The criteria is referred to as partial-length RPC justifications and show that, based on plant observance of a certain maximum primary-to-secondary side leakage value during normal operation, primary water stress corrosion cracking (PWSCC) below a certain depth into the tubesheet from the tubesheet top will pose neither structural issues such as tube severance and pullout nor excessive leakage during the limiting accident condition, steamline break. If or when the normal operation leakage reaches the maximum level, the plant would plug the leaking tubes.

An evaluation has been performed to develop the specific RPC inspection depth, known as H* ("H-Star"), below which any type of axial or circumferential PWSCC can be accommodated provided normal operation leakage is closely monitored. The determination of H* consists of analyses and testing programs which quantified the tube-to-tubesheet hole surface radial contact pressure of the Model F steam generator tubes for the bounding plant conditions. The tube within the H* length must be non-degraded. H* is reckoned downward from the top of the tubesheet or the bottom of the hydraulic expansion transition, whichever is lower in elevation and it varies with distance of a particular tube from the vertical centerline of the tube bundle. The tubes are grouped in four zones, based on distance from the bundle vertical centerline. The largest H* depth is approximately 8 inches and the smallest is approximately 2.50 inches. Due to tubesheet upward bending during normal operation and during the limiting accident condition, the tube-to-tubesheet hole surface contact pressure varies through the thickness of the tubesheet. The resistance to leakage through a crack and through the tube-to-tubesheet interface to the secondary side is a function of fluid conditions and tube-to-tubesheet contact pressure. Therefore, leakage from a given crack in a given tube, is a function of a crack distance from the tubesheet top and tube distance from the bundle (tubesheet) vertical centerline.

An additional portion of the justification for partial-length RPC inspection of the tube joint is P* ("P-Star"). P* was determined to be 3 inches, reckoned downward from the tubesheet top. The main function of P* is to act as a contingency for potential circumferential cracks slightly below the 3 inch RPC

inspection depth. The intent of P* is to show the acceptability of a tube separation below the P* distance for approximately 95 percent of the operating tubes in the bundle. P* is based on the consideration that an affected tube will be captured by the non-degraded neighboring tube in the same column, thus limiting the vertical motion that could ensue if the tube severed within the tubesheet and provided a small engagement with the tubesheet hole.

The implementation of H* involves performing partial-length RPC (PLRPC) inspection of the tubes within the tubesheet to average depths comparable to the currently specified depth of 3 inches (on average for all of the tubes on the hot leg). The application of H* is invoked if PWSCC is suspected, or confirmed, to be occurring within the tubesheet or when it is concluded there is significant potential for such occurrence. In the case of Callaway, PWSCC within the trigger region of the tube, i.e., below the bottom of the hydraulic expansion transition and the standard 3 inches inspection depth, has been determined by NDE.

The mechanical features of the existing tube-to-tubesheet joint have been analyzed and have been shown to perform the mechanical and hydraulic functions of the tube joint including and below the elevation of H*, including the function of the weld. The H* joint would provide adequate resistance to tube movement and to primary-to-secondary leakage at the limiting condition, the steamline break condition. The development of the H* length conservatively considers the portion of the tube joint below H*, including the weld, as being completely degraded and, therefore, to provide no anchorage or resistance to leakage. A similar approach has been demonstrated to be acceptable for use in cases of tube weld damage due to loose parts. In those cases, the entire length of the hydraulically expanded tube joint, above the weld was demonstrated to be adequate to replace the potentially ineffective weld.

Based on the results of the evaluation provided in WCAP-15932-P, and the results of several inspections performed during past refueling outages wherein PWSCC (axial or circumferential) was indicated in the region below the hydraulic expansion bottom and the 3 inch RPC inspection depth, it has been determined that the RPC inspection depth of the tubes within the tubesheet be the calculated H* depths.

In calculating the H* depth, the tubesheet was divided into four zones, designated A, B, C, and D defined by radial distance from the tubesheet vertical centerline. H* for each zone is determined as the inspection depth based on accident conditions in a specified zone. Zone D, the innermost zone, extending from the vertical centerline of the tubesheet to a radius of 12.0 inches, requires an inspection to a depth of 7.98 inches (including 0.15 inch for the portion of the expansion transition inside the tubesheet). Zone C would extend from a radius of 12.0 inches to a radius of 30.2 inches and would require an inspection to a depth of 7.50 inches. Zone B extends from a radius of 30.2 inches to a radius of 48.6 inches and would require an inspection to a depth of 5.75 inches. Zone A

extends from a radius of 48.6 inches to the periphery of the tubesheet and would require an inspection depth of 2.38 inches. The H* information is summarized in Table 3-1 and 3-2 of WCAP-15932-P.

The justifications developed in WCAP-15932-P to reduce the RPC inspection length of the Model F steam generator tubes within the tubesheet from full-length to partial-length for the Callaway Plant show that, based on plant observance of 75 gpd maximum primary-to-secondary side leakage value during normal operation, PWSCC below the H* depths (in a given zone) into the tubesheet from the tubesheet top will pose neither structural issues such as tube severance and pullout nor excessive leakage during the limiting accident condition. If or when the normal operation leakage reached the 75 gpd level, the plant would plug the leaking tubes, preferably during a refueling outage and leakage during the limiting postulated accident case would be less than the FSAR limit of 1.0 gpm for the affected steam generator.

Summary/Conclusion

The proposed amendment revises TS 5.5.9 to clearly delineate the scope of the steam generator tube inspection required in the tubesheet region. The analyses presented above assess the potential impact of the proposed change on applicable safety analyses. The assessments demonstrate that the change will not adversely affect the design basis, safety analyses, or the safe operation of the plant.

6.0 REGULATORY ANALYSIS

There have been no changes to the plant design such that any of the regulatory requirements in Section 4.0 would come into question. This amendment application revises TS 5.5.9 to clearly delineate the scope of the steam generator tube inspection required in the tubesheet region. The evaluation performed by AmerenUE in Section 5 concludes that Callaway Plant will continue to comply with all applicable regulatory requirements.

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

7.0 NO SIGNIFICANT HAZARDS DETERMINATION

This amendment application revises TS 5.5.9 to clearly delineate the scope of the steam generator tube inspection required in the tubesheet region. The

definition for tube inspection is provided in TS 5.5.9.h), and this LAR would provide clarification to this definition.

The proposed amendment does not involve a significant hazards consideration for Callaway Plant based on the three standards set forth in 10 CFR 50.92(c) as discussed below:

(1) Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change to the Callaway TS is to incorporate a SG tube inspection program based on WCAP-15932-P. The H*/P* analysis takes into account the reinforcing effect the tubesheet has on the external surface of an expanded tube.

Tube-bundle integrity will not be adversely affected by the implementation of the H*/P* tube inspection scope. SG tube burst or collapse cannot occur within the confines of the tubesheet; therefore, the tube burst and collapse criteria of Regulatory Guide (RG) 1.121 are inherently met. Any degradation below the H*/P* distance is shown by the analysis results to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event.

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet, therefore structural criterion is satisfied by the tubesheet constraint. However, severe degradation of the tube within the tubesheet could possibly permit severing of the tube and tube pullout from the tubesheet under the axial forces on the tube. Sections 5.0 and 7.0 of WCAP-15932-P describe the testing and structural analysis that was performed to define the H* length of non-degraded tubing that is sufficient to compensate for the axial forces on the tube to prevent pullout.

Therefore, the proposed change to incorporate the H* inspection scope into the TS maintains existing design limits and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?

Response: No

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This

amendment will not affect the normal method of plant operation or change any operating limits. Tube-bundle integrity will be maintained during all plant conditions upon implementation of the proposed changes to the tube inspection scope. The proposed change does not induce a new mechanism that would result in a different kind of accident from those previously analyzed. No new accident scenarios, transients precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

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

(3) Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change does not impact the margin of safety as defined by RG 1.83 and RG 1.121 and the requirements of General Design Criteria 14, 15, 19, 31, and 32. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion:

Based on the above, AmerenUE concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. AmerenUE has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase

in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

9.0 PRECEDENTS

The TS change requested in this amendment application is similar to those previously approved for San Onofre Nuclear Generating Station Unit 2, and Sequoyah Nuclear Plant Unit 2.

10. REFERENCES

1. Sequoyah Nuclear Plant, Unit 2 License Amendment No. 266, dated May 10, 2002.

2. WCAP-15932-P

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ATTACHMENT 3

MARKUP OF TECHNICAL SPECIFICATION PAGE

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

d) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

e) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective;

f) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving and is equal to 40% of the nominal tube wall thickness. The plugging limit for laser welded sleeves is equal to 39% of the nominal sleeve wall thickness. The plugging limit for Electrosleeves is equal to 20% of the nominal sleeve wall thickness;

g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3, above;

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg, including inspection with rotating pancake coil (or equivalent) from the tube expansion transition at the top of the tubesheet to the H* depth as specified in Table 5.5.9-4. For a tube repaired by sleeving, the tube inspection shall include the sleeved portion of the tube;

i) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections; and

(continued)

TABLE 5.5.9-4

STEAM GENERATOR TUBE INSPECTION H* DEPTHS

H* DEPTHS				
Inspection Depth Zones	D	C	B	A
Radius of the Zone from the Vertical Centerline of the Tubesheet	0"-12.0"	>12.0"-30.2"	>30.2"-48.6"	>48.6"-58.3"
H* Depths for the Zone	7.98"	7.50"	5.75"	2.38"

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ATTACHMENT 4

RETYPED TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

d) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

e) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective;

f) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving and is equal to 40% of the nominal tube wall thickness. The plugging limit for laser welded sleeves is equal to 39% of the nominal sleeve wall thickness. The plugging limit for Electrosleeves is equal to 20% of the nominal sleeve wall thickness;

g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3, above;

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg, including inspection with rotating pancake coil (or equivalent) from the tube expansion transition at the top of the tubesheet to the H* depth as specified in Table 5.5.9-4. For a tube repaired by sleeving, the tube inspection shall include the sleeved portion of the tube;

i) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections; and

(continued)

TABLE 5.5.9-4

STEAM GENERATOR TUBE INSPECTION H* DEPTHS

H* DEPTHS				
Inspection Depth Zones	D	C	B	A
Radius of the Zone from the Vertical Centerline of the Tubesheet	0"-12.0"	>12.0"-30.2"	>30.2"-48.6	>48.6"-58.3"
H* Depths for the Zone	7.98"	7.50"	5.75"	2.38"

(continued)

5.5 Programs and Manuals (continued)

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 a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Rev. 2, and uses the test procedure guidance in Regulatory Guide 1.52, Revision 2, Positions C.5.a, C.5.c and C.5.d.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Filtration	2.0%	70%
Control Room Pressurization	2.0%	70%
Emergency Exhaust System	2.0%	70%

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration	5.4" WG	2000 cfm, ± 200 cfm
Control Room Pressurization	5.4" WG	500 cfm, +500,- 50 cfm
Emergency Exhaust System	5.4" WG	9000 cfm, ± 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI 510-1975 and corrected to design nameplate voltage settings.

ESF Ventilation System	Wattage
Control Room Pressurization	15 ± 2 KW
Emergency Exhaust System	37 ± 3 KW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure, Revision 0". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures, Revision 2".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in the outdoor liquid radwaste tanks listed below that are not

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste System is less than the quantities determined in accordance with the Standard Review Plan, Section 15.7.3:

a. Reactor Makeup Water Storage Tank,
b. Refueling Water Storage Tank,
c. Condensate Storage Tank, and
d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a water and sediment content within limits for ASTM 2D fuel oil.

b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition of new fuel oil to storage tanks; and

c. Total particulate concentration of the stored fuel oil is $\leq$ 10 mg/l when tested every 31 days based on applicable ASTM D-2276 standards.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.1 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

d. Leakage rate acceptance criteria are:

(continued)

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$;

   b) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to ≥ 10 psig.

e. The provisions or Technical Specification SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of Technical Specification SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totalling < 20% percent of the individual total dose need not be accounted for. In the aggregate, at least 80% percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period.

The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year. in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient limits in Specification 3.1.3,

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference Limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor, $F_Q(Z)$ F_Q^{RTP}, K(Z), W(Z) and F_Q Penalty Factors for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2.

7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).

2. WCAP-10216-P-A, REV. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

3. WCAP-10266-P-A, REV. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).

4. NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report' (TAC NO 77268)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."

b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NRC letter, CALLAWAY PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE: PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MA5631 and MA7287), dated March 24, 2000.

2. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January, 1996".

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a report within 12 months following completion of the inspection. This report shall include:

1) Number and extent of tubes and sleeves inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported to the Commission within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment;

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits does rate and cumulative dose rate information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(continued)

5.7 High Area Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the Shift Supervisor/Operating Supervisor or Health Physics Supervision, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

(continued)

5.7 High Area Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably

(continued)

5.7 High Area Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individual qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area, such as PWR containment, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

Westinghouse

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Cust. P.O.
Our ref: SCP-02-66

October 4, 2002

AmerenUE
CALLAWAY PLANT
Improved Justification of Partial-Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren UE Callaway Plant

Dear Mr. Corder:

This letter transmits 10 copies of proprietary (WCAP-15932-P) and non-proprietary (WCAP-15932-NP) versions of "Improved Justification of Partial-Length RPC Inspection of Tube Joints of Model F Steam Generators of AmerenUE Callaway Plant," dated September, 2002, for your submittal to the NRC for review and approval.

In addition to the proprietary and non-proprietary WCAPs, there are four other enclosures for your use:

1. Information which should be included in your NRC transmittal letter.

2. Proprietary Information Notice to be attached to your NRC transmittal letter.

3. Copyright Notice to be attached to your NRC transmittal letter.

4. Westinghouse letter, "Application for Withholding Proprietary Information from Public Disclosure" (CAW-02-1558) with Affidavit CAW-02-1558.

Please transmit the original of Item 4 to the NRC in your transmittal.

If you have any questions, please do not hesitate to contact us.

Sincerely,

WESTINGHOUSE ELECTRIC COMPANY LLC

P. J. McDonough
Customer Project Manager

Attachment

cc:	M. S. Evans	Callaway
	J. H. Center	Callaway
	Bruce Huhmann	Callaway

Our ref. SCP-02-66
October 4, 2002

bcc:	Z. L. Henderson	ECE 510
	P. J. McDonough	ECE 510
	Larry Nelson	WM F2V38

Westinghouse

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Attention: Mr. Samuel J. Collins

Our ref: CAW-02-1558

October 4, 2002

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP- 15932-P, Improved Justification of Partial Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant (Proprietary);

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-02-1558 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 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Ameren-UE.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-02-1558, and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

Cc: G. Shukla/NRR
T. Carter/NRC (5E7)

bcc: H. A. Sepp (ECE 4-7A) 1L, 1A
R. Bastien, (Brussels, Belgium) 1L, 1A
L. Ulloa (Madrid, Spain) 1L, 1A
C. Brinkman, 1L, 1A (Westinghouse Electric Co , 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RLE Administrative Aide (ECE 4-7A) (letter w/affidavit, no attachments)
L. A. Nelson, Waltz Mill
R. J. Sterdis, Waltz Mill
G. W. Whiteman (ECE 410C)

CAW-02-1558

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, 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:

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 4th day
of October, 2002

Lorraine M. Piplica
Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, 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 the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 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 the Westinghouse Electric Company LLC 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 2.790 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 constitutes Westinghouse policy and provides 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 10CFR Section 2.790, 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 "Improved Justification of Partial-Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant", WCAP-15932-P, (Proprietary), dated September 2002. The information is provided in support of a submittal to the Commission, being transmitted by the Ameren-UE letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information is submitted for use by Westinghouse Electric Company LLC for the Callaway Plant. The information is applicable for other licensee submittals in response to certain NRC requirements for justification of a reduction of rotating pancake coil (RPC) inspection length of Model F steam generator tubes within the tubesheet from full-length to partial length.

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

(a) Justify the use of the H* criterion as a basis for limiting the length of eddy current inspection of hydraulically expanded tubes in the tubesheet of region of steam generators.

(b) Justify the use of the P* criterion which acts as a contingency for potential circumferential cracks just below the RPC inspection depth.

(c) Discuss analysis and testing programs used in support of the development of the H* and P* criterion.

Further this information has substantial commercial value as follows:

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

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

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology that was developed by Westinghouse.

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 licensing support documentation 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 2.790 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 2.790(b)(1).

COPYRIGHT NOTICE

Ameren-UE

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC:

Enclosed are:

1. 2 copies of WCAP-15932-P, "Improved Justification of Partial Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant" (Proprietary).

2. 2 copies of WCAP-15932-NP, "Improved Justification of Partial Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant" (Nonproprietary).

Also enclosed are a Westinghouse authorization letter, CAW-02-1558, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit 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 Section 2.790 of the Commission's' regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-02-1558 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.