November 23, 1998

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: ISSUANCE OF AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING CHANGES TO THE TECHNICAL SPECIFICATION FOR STEAM GENERATOR TUBE SLEEVING - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M98097)

Dear Mr. Scarola:

The Nuclear Regulatory Commission has issued Amendment No. 85 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No. 1, in response to your request dated February 27, 1997, as supplemented by letter dated August 24, 1998. This amendment changes Technical Specification (TS) 3/4.4.5, "Steam Generators," by adding sleeve installation as an alternative to tube plugging for repairing degraded steam generators. The amendment incorporates into the TS (1) reference to a Combustion Engineering, Inc. topical report to describe steam generator tube sleeving techniques, (2) sleeve/tube inspection scope and expansion criteria, (3) plugging limits for sleeved tubes, and (4) a requirement to perform a post-weld heat treatment of free span welds.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's regular bi-weekly Federal Register notice.

Sincerely,

Original signed by:

Scott C. Flanders, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 85 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See next page

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OFFICE	PM:PDII-3	LA:PDII-2	OGC	D:PDII-3	
NAME	SFlanders	EDunnington		FHebdon	
DATE	11/17/98	11/20/98	11/ /98	11/20/98	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

FFICIAL RECORD

AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File
PUBLIC
PDII-1 Reading
J. Zwolinski
OGC
G. Hill (2)
E. Sullivan
E. Murphy
A. Keim
ACRS
L. Plisco, RII

cc: Harris Service List

AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

[redacted]
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NAME	SFlanders	EDunnington	[illegible]	FHebdon	
DATE	11/17/98	11/2/98	11/ /98	11/20/98	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

OFFICIAL RECORD

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 23, 1998

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: ISSUANCE OF AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING CHANGES TO THE TECHNICAL SPECIFICATION FOR STEAM GENERATOR TUBE SLEEVING - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M98097)

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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-400

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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. James Scarola
Carolina Power & Light Company

Shearon Harris Nuclear Power Plant
Unit 1

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. NPF-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated February 27,1997, as supplemented by letter dated August 24, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 85 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 23, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 85

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-16	3/4 4-16
3/4 4-16a	3/4 4-16a
3/4 4-17	3/4 4-17
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
---	3/4 4-20a
3/4 4-23	3/4 4-23
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 4.4-2 A, B and C. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

   2. Tubes in those areas where experience has indicated potential problems, and

* When referring to a steam generator tube, the sleeve shall be considered part of the tube if the tube has been repaired per Specification 4.4.5.4.a.12.

4.4.5.2 (Continued)

3. A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Tables 4.4-2 A, B and C) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

d. Each inspection shall include a sample of those tubes expanded in the preheater section of the steam generator. The first sample size, second sample size and subsequent inspection shall follow Table 4.4-2B.

e. In addition to the 3% sample, all tubes for which the alternate plugging criteria (F*) has been previously applied shall be inspected in the tubesheet region.

f. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.e, previous defects or imperfections in the area covered by the sleeve pressure boundary are not considered an area requiring reinspection.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube or sleeve;

3. Degraded Tube means a tube, including the sleeve if the tube has been repaired, containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be repaired, (i.e. sleeved) or removed from service and is equal to 40% of the nominal tube wall thickness. Plugging will be required for all tubes that have been repaired by sleeving where degradation in the sleeve pressure boundary is detected. The portion of the tube and sleeve for which indications of wall degradation must be evaluated is that portion constituting the pressure boundary. The pressure boundary is defined as follows:

   i. For a free span sleeve (both ends welded), the pressure boundary consists of that portion between the upper and lower weld centerlines.

   ii. For a tubesheet area sleeve (lower end rolled), the pressure boundary consists of that portion between the upper weld centerline and the lower rolled joint centerline.

   iii. For the tube, the pressure boundary consists of those portions of the tube above (and below for transition zone sleeves) the sleeve weld centerline.

   The plugging limit does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

7. Unserviceable describes the condition of a tube or sleeve if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube;

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator, including the sleeve if the tube has been repaired, performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10. F* Distance is the distance into the tubesheet from the face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet), that has been conservatively chosen to be 1.6 inches.

11. Alternate Tube Plugging Criteria does not require the tube to be removed from service or repaired when the tube degradation exceeds the plugging limit so long as the degradation is in that portion of the tube from F* to the bottom of the tubesheet. This definition does not apply to tubes with degradation (i.e., indications of cracking) in the F* distance.

12. Tube Repair refers to sleeving, as described in Combustion Engineering Report CEN-630-P, Rev. 01 which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventative measure. A post-weld heat treatment during installation will be performed. 100% of the weld zones will be visual test (VT) inspected upon installation.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 4.4-2A, B and C.

4.4.5.5 Reports

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 in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube (including sleeves) inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special 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 in a Special Report pursuant to Specification 6.9.2 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.

d. The results of the inspection of F* tubes shall be reported to the Commission in a report, prior to the restart of the unit following the inspection. This report shall include:

   1. Identification of F* tubes, and

   2. Location and size of the degradation.

   NRC approval of this report is not required prior to restart.

TABLE 4.4-2A

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes* and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes* and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes*
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes* and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Specification 4.4.5.5.c.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes.* Notification to NRC pursuant to Specification 4.4.5.5.c.	N/A	N/A

$S = \frac{9}{n}\%$ where n is the number of steam generators inspected during an inspection.

* Defective tubes which fall under the Alternate Tube Plugging Criteria do not have to be plugged or repaired.

TABLE 4.4-2B

STEAM GENERATOR TUBE INSPECTION - TUBE EXPANDED IN PREHEATER REGION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of S of the tubes expanded in the preheater section	C-1	None	N/A	N/A
	C-2	Plug or repair defective tubes and inspect all other expanded tubes in this Steam Generator	C-1	N/A
			C-2	Plug or repair defective tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all expanded tubes in this Steam Generator, plug or repair defective tubes and inspect all expanded tubes in each other Steam Generator. Notification to NRC pursuant to Specification 4.4.5.5.c.	All other S.G.s are C-1	None
			One or more S.G.s C-2 but no additional S.G.s are C-3	Plug or repair defective tubes
			Additional S.G. is C-3	Plug or repair defective tubes. Notification to NRC pursuant to Specification 4.4.5.5.c.

$S = \frac{9}{n}\%$ where n is the number of steam generators inspected during an inspection.

TABLE 4.4-2C

STEAM GENERATOR TUBE SLEEVE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of each type of installed sleeve per S.G. (1)	C-1	None	N/A	N/A
	C-2	Plug tubes containing defective sleeves and inspect all remaining installed sleeves in this S.G.	C-1	None
			C-2	Plug tubes containing defective sleeves
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all installed sleeves in this S.G., plug tubes containing defective sleeves and inspect 100% of the installed sleeves in each other S.G.	Each other S.G. is C-1	None
		Special Report to NRC per Specification 6.9.2	Each other S.G. is C-2	Plug tubes containing defective sleeves
			Each other S.G. is C-3	Inspect all installed sleeves in each S.G. and plug tubes containing defective sleeves Special Report to NRC per Spec. 6.9.2

(1) Each sleeve type is considered a separate popluation for determination of sample expansion.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 31 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted by multiplying the observed leakage by the square root of the quotient of 2235 divided by the test pressure.

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired or plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair or plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. This plugging limit does not apply to imperfections located below the F* region of any given tube. The F* criterion can be applied only if the tube geometry in the region selected as the F* distance falls within the analytical limits of WCAP-12816. A sleeved tube must be plugged if degradation is detected in the sleeve pressure boundary. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

OPERATIONAL LEAKAGE (Continued)

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The maximum allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letters dated February 27, 1997, and August 24, 1998, Carolina Power & Light Company (CP&L or the licensee), submitted a request to change the technical specifications at the Harris Nuclear Plant (HNP), Unit 1. The August 24, 1998, supplemental letter provided clarifying information only, and did not change the initial no significant hazards consideration determination. The proposed amendment would permit sleeving repairs of defective steam generator (SG) tubing with Combustion Engineering (CE) leaktight sleeves. The proposed changes incorporate references to a CE topical report describing SG tube sleeves, incorporate sleeve/tube inspection scope and expansion criteria, define sleeve plugging limits as plug upon detection of degradation of sleeve, incorporate a post-weld heat treatment of free span welds, and reduce the allowable primary-to-secondary leakage through any one SG to 150 gallons per day (gpd).

The CE topical report, CEN-630-P, Rev. 01, "Repair of 3/4-inch O.D. Steam Generator Tubes Using Leak Tight Sleeves," dated November 1996, addresses issues identified in 1996 at Prairie Island Unit 1. At issue were indications detected in weld joints of CE sleeves resulting from inadequate cleaning. Because the bulk of the technical and regulatory issues for the present request are identical to those reviewed in the previous safety evaluations (SEs), this SE discusses only those issues warranting revision, amplification, or inclusion based on current experience.

Details of prior staff evaluations of CE sleeves may also be found in the SEs for Waterford Steam Electric Station, Unit 3, Docket Number 50-382, dated December 14, 1995; Byron Nuclear Power Station, Units 1 and 2 and Braidwood Nuclear Power Station, Units 1 and 2, Docket Numbers 50-454, 50-455, 50-456 and 50-457, dated April 12, 1996; Zion Nuclear Power Station, Units 1 and 2, Docket Numbers 50-295 and 50-304, dated October 29, 1996; Prairie Island, Units 1 and 2, Docket Numbers 50-282 and 50-306, dated November 4, 1997; and Beaver Valley, Unit 1, Docket Number 50-334, dated November 25, 1997. These evaluations relate to the proposed Shearon Harris license amendment.

2.0 BACKGROUND

Previous staff evaluation of CE sleeves addressed the technical adequacy of the sleeves in the four principal areas of pressure-retaining component design: structural requirements, material of construction, welding, and non-destructive examination. The staff found the analyses and tests that were submitted to address these areas of component design to be acceptable.

The function of sleeves is to restore the structural and leakage integrity of the tube pressure boundary. Consequently, structural analyses were performed for a variety of loadings including design pressure, operating transients, and other parameters selected to envelope loads imposed during normal operating, upset, and accident conditions. Stress analyses of sleeved tube assemblies were performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. Detailed in the generic topical, the structural integrity of the sleeve design has been investigated analytically and verified by laboratory tests of sleeve mockups. These analyses, along with the results of qualification testing and previous plant operating experience, were cited to demonstrate that the sleeved tube assembly is capable of restoring SG tube integrity.

The sleeve material is a nickel-iron-chromium alloy, Alloy 690, a Code-approved material (ASME SB-163) incorporated in ASME Code Case N-20. The staff has determined that the use of Alloy 690 thermally treated (TT) sleeves is an improvement over the Alloy 600 material used in the original SG tubing. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirm test results regarding the improved corrosion resistance of Alloy 690 TT over that of Alloy 600. The NRC staff has concluded, as a result of these laboratory corrosion tests, that Alloy 690 is acceptable in meeting the guidelines in Regulatory Guide (RG) 1.85, "Materials Code Case Acceptability ASME Section III, Division 1," Revision 24, dated July 1986. NRC staff has approved use of Alloy 690 TT tubing in replacement SGs as well as sleeving applications.

Two CE sleeve types are proposed. They are a tube sheet sleeve and a tube support (TS) sleeve. A tube sheet sleeve is designed to restore the portion of a tube in the vicinity of the top of the SG tubesheet. A TS sleeve can be used to span a support plate elevation or be used on a freespan section of tube.

The CE sleeves are installed using gas tungsten arc welding to join the sleeve to the parent tube at the upper (free span) end of the tube sheet sleeve and at both ends of a TS sleeve. The weld joint is the subject of the modifications to the installation processes described in topical report CEN-630-P, Rev. 01. The lower joint of the tube sheet sleeve is hard-rolled into the tubesheet below the expansion zone. There are no changes from the previous topical reports with respect to the rolled joint.

Leak resistance of the rolled joint has been demonstrated through laboratory tests. The rolled joint is controlled to provide a leaktight structural joint. Bounding calculations and laboratory tests have verified that, should leakage develop in the welded or rolled joints, it would not exceed 1 gpm and, thus, the 10 CFR Part 100 guidelines for radiological release would not be impacted, even under the most severe postulated conditions.

3.0 EVALUATION

Experience with all types of SG tube sleeves has revealed certain issues outside the scope of basic sleeve design and qualification discussed in previous SEs. These issues involve weld preparation, weld acceptance inspections, inservice inspection expansion criterion, sleeve plugging limits, post-weld heat treatment, and primary-to-secondary leakage limits.

During the spring 1996 refueling outage at Prairie Island Unit 1, roughly 60 previously installed weld joints in CE-sleeved tubes were discovered to have eddy current test (ET) indications. This discovery was the result of the licensee employing a new, more sensitive ET probe for its periodic inspection of SG tubes. Tube/sleeve assemblies were removed from the SGs for metallurgical examination and root cause determination. It was found that the ET indications were due to entrapped oxides and/or weld shrinkage within the sleeve-to-tube weld. The cause of these weld defects was traced to a previously revised tube cleaning procedure. Although the discovered weld defects did not significantly impair the structural integrity (strength) of the welds, they did pose a small leakage potential which is contrary to ASME Code requirements for a welded joint.

As a result of the metallurgical examination, the tube cleaning procedure was revised and post-cleaning visual inspections (VT) were adopted. The initial weld acceptance inspection, an ultrasonic test (UT), was revised to give greater sensitivity. As added measures, an optional VT of the completed welds was added to the installation procedure and the initial baseline ET, normally used only as reference for later periodic reinspection, was modified to supplement the UT as part of the initial weld acceptance inspection. All of these refinements to the sleeving procedure were confirmed using a large number of laboratory samples and field mockups. These modifications were incorporated into a new generic topical report, CEN-630-P, Rev. 01, referenced above, and are discussed in more detail in the following sections.

3.1 Weld Preparation

Prior to performing any weld, the surface of the metal(s) to be welded must be cleaned. For sleeve installation, the inner diameter of the parent tube at the desired weld location must be cleaned of service-induced oxides. For the CE sleeving process, this is accomplished using motorized wire brushes.

Based upon the metallurgical examination of the Prairie Island samples, CE revised the cleaning method to ensure optimum removal of service-induced oxides. The revised cleaning procedure entailed some equipment changes. More significantly, from a quality assurance standpoint, a 100% VT of the cleaning process was instituted. After the wire brush cleaning step, every tube is given a VT using a remote fiber optic camera system to confirm that adequate surface cleaning has been accomplished. CE advises that the 100% VT is an interim step until enough field experience is gained to consider adoption of a statistical sampling plan in the future. The licensee states that it will perform the visual inspection prior to sleeve installation as documented in the CE topical report CEN-630-P, Rev. 01.

3.2 Weld Acceptance Inspections

To verify the acceptability of sleeve welds, the sleeves are examined using a combination of VT, UT, and ET at different stages of the installation process. The analysis of inspection data from three diverse methods of inspection improves the ability to detect fabrication-induced defects. In addition, ET data are also used as a baseline for comparison with data obtained in future required periodic inspections.

In accordance with the ASME Code, Section XI, initial examinations of sleeve welds are performed. Historically, sleeve welds were accepted based upon VT and UT examinations, and ET was used for an initial baseline inspection for comparison with later required periodic inspections. The reason for now using the different types of nondestructive examinations (NDE) for initial acceptance as opposed to the previous practice is due to the differences between potential flaws arising from initial installation versus service-induced degradation. The different NDE techniques have normally been better suited for the respective types of anticipated flaws.

The Prairie Island experience suggested that using only the VT and UT, in place at that time, as initial acceptance examinations may not be sufficient in every circumstance. As a result, the weld acceptance NDE was modified to include:

* 100 percent UT with an enhanced digitized amplitude system
* 100 percent ET using the Plus Point probe

The original UT procedure was based upon the absence of a mid-wall reflection. When fusion existed, the mid-wall reflection (mid-wall of the fused sleeve and tube combination) would not appear since no interface would exist. The Prairie Island experience led CE to discover that lack of fusion caused by axially oriented oxide inclusions from an inadequately prepared surface would not be detected by the UT procedure in use. The oxides are sound conductors and do not cause a large sound reflection.

In the enhanced UT procedure, the back wall signal from the outside of the parent tube is also monitored for presence in the fused area. Additionally, the back wall signal strength is examined for excessive attenuation. Attenuation beyond the normal amount can be interpreted, along with other signal artifacts, as either a weld that is too narrow or one with inclusions or patches of unfused material. The modified UT procedure was extensively tested on laboratory-produced welds containing a variety of inclusion/lack of fusion defects. Samples were then destructively examined and the metallurgical sections compared with the UT results. Comparison of results demonstrated that the revised UT procedure was highly reliable. No significant defects were undetected by the enhanced UT procedure.

ET with the Plus Point probe is now part of the sleeve weld acceptance criteria. CE also discovered that weld shrinkage and circumferentially oriented oxide inclusions from a poorly cleaned weld would not be detected by UT. CE has shown the Plus Point probe reliably detects these process-induced weld defects and blowholes. CE has also shown the ET can reliably locate the position of the defect with respect to the weld centerline which CE defines as the pressure boundary. ET indications located above the weld centerline that meet UT requirements can be left in service. Any ET indication found below the weld centerline requires the tube to be plugged. For the lower welds on tube support sleeves, this criterion is appropriately modified so that indications below the weld center line may be left in service.

In performing the initial inspection, the licensee will use EPRI "PWR Steam Generator Tube Examination Guidelines" Appendix G qualified personnel and Appendix H qualified ECT techniques. For future sleeve/tube inspections, the licensee stated it will follow the most current revision of the EPRI guidelines in terms of inspection scope and expansion criteria as well as personnel and technique qualifications.

During a recent installation of welded sleeves at Kewaunee Nuclear Power Plant, the licensee visually identified weld zone indications that were not identified with either eddy current or ultrasonic inspection techniques. Therefore, this finding indicates that all three inspection methods are needed to ensure acceptable sleeve welds. CP&L proposes to modified the HNP Technical Specification 4.4.5.4a.12 to include 100% visual inspection of sleeves upon installation.

3.3 Inservice Inspection Requirements

Included in the licensee's proposed amendment request are changes that would require the licensee to perform an inspection of 20% of sleeves at each refueling outage. The minimum sample requirements for tube inspections which are specified in "Steam Generator Tube Sample Selection and Inspection" within Technical Specification 4.4.5 are established to assess the overall condition of the SG. Sleeved tubes are of a slightly different configuration and may be more susceptible to stress corrosion cracking than unrepaired tubing; therefore, the inservice inspection requirements currently specified in the technical specifications may not be sufficient to address the condition of these tubes.

The licensee has proposed to include additional inservice inspection requirements in the technical specifications to address sleeves. These additional requirements are reflected in TS 4.4.5.2 and Tables 4.4-2 A,B, and C. The changes would require the initial inspection of at least 20% of all installed sleeves. The proposal is consistent with current industry guidance for SG sleeve examinations. EPRI recommends a 20% sample inspection for sleeves. In addition to this licensee proposal, the results from inspections would be classified, and depending on the classification, may require additional sleeve inspections.

3.4 Sleeve Plugging Limits

The sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and ASME Code Section III allowable stress values and pressure stress equations. According to RG 1.121 criteria, an allowance for NDE uncertainty and postulated operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to evaluate sleeve degradation. The staff assumes a tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% through-wall per cycle for the purpose of determining the sleeve plugging limit. The sleeve plugging limit, which was calculated based on the most limiting of normal, upset, or faulted conditions for 3/4-inch outside diameter steam generator tubes in Westinghouse-designed generators, was determined to be 30% of the sleeve nominal wall thickness based on ASME Code minimum material properties in accordance with staff positions. However, the licensee has proposed to remove sleeves from service upon detection of service-induced degradation of the sleeve material or any portion of the sleeve-to-tube weld. The licensee has modified HNP TS 4.4.5.4a.6 and TS Bases 3/4.4.5 to reflect this commitment. Removal from service of tubes with sleeves with detectable service-induced degradation provides assurance that the minimum acceptable wall thickness will not be violated during the next subsequent cycle of operation.

3.5 Post-Weld Heat Treatment

Accelerated corrosion tests confirm that a post weld heat treatment (PWHT) significantly improves the intergranular stress corrosion cracking resistance of the Alloy 600 parent tube material in the weld zone. In its February 27, 1997, submittal, the licensee committed to performing PWHT of the welded joints in accordance with the CE generic sleeving report and the NRC staff position. This commitment is reflected by appropriate text inclusion to the HNP Technical Specification 4.4.5.4a.12.

3.6 Reactor-to-Secondary Leakage Limit

The licensee proposes to modify Limiting Condition for Operation 3.4.6.2.c. The modification changes the reactor-to-secondary leakage limit of 500 gpd through any one SG to 150 gpd through any one SG. TS Bases 3/4.4.6.2 was also changed to reflect this modification. This modification will ensure that SG tube integrity is maintained in the event of a main steam line rupture or under loss of coolant accident conditions.

The staff concludes the proposed sleeving repairs, as described in the CE sleeve topical report, can be accomplished to produce sleeved tubes of acceptable structural integrity, leak tightness and corrosion resistance. The staff also finds the proposed preservice inspection methods for examining the conditions of the welds acceptable.

The NRC staff concluded the repair of SG tubes using welded sleeves designed by CE is acceptable, supplemented by additional licensee commitments, as discussed above and reflected by technical specifications revisions, to: 1) include visual inspection of sleeves upon installation, and performing PWHT of the free span weld joints; 2) incorporate TS table 4.4-2C for sleeve inservice inspection sample size and expansion criteria; 3) modify TS tables 4.4-2 A and B to include sleeve repair as an acceptable action for tubes identified as defective during a steam generator inspection; 4) define the sleeve plugging limit of plugging on detection of degradation of the sleeve; and 5) reduce the reactor-to-secondary leakage through any one SG to 150 gpd. Therefore, the staff finds the proposed TS changes acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The

Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 17225). Accordingly, the amendment 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 need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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