September 4, 1992

Docket No. 50-400

Mr. R. A. Watson
Senior Vice President
Nuclear Generation
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Watson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING STEAM GENERATOR F* TUBE PLUGGING CRITERIA - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M82815)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP). This amendment changes the Technical Specifications in response to your request dated February 7, 1992.

The amendment revises Operating License (NPF-63) to allow the use of an alternate steam generator tube plugging criterion for the portion of the tubes within the tubesheet. The proposed revision would amend Technical Specification (TS) 4.4.5 specifying an F* (F-star) distance within the tubesheet below which indications would not require repair or plugging.

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

Sincerely,

Orignal signed by:

Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 31 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

CP-1

LA:PD21:DRPE	PM:PD21:DRPE	OGC	D:PD21:DRPE	*DET:EMCB	
PAnderson	NLe		EAdensam	JStrosnider	
8/28/92	8/27/92	8/2/92	9/4/92	8/28/92	

Mr. R. A. Watson
Carolina Power & Light Company

Shearon Harris Nuclear Power Plant,
Unit 1

cc:

Mr. H. Ray Starling
Manager - Legal Department
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Resident Inspector/Harris NPS
c/o U. S. Nuclear Regulatory Commission
Route 1, Box 315B
New Hill, North Carolina 27562

Mr. Gerald E. Vaughn, Vice President
Harris Nuclear Project
Harris Nuclear Plant
P. O. Box 165
New Hill, North Carolina 27562

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

Public Service Commission
State of South Carolina
P.O. Drawer 11649
Columbia, South Carolina 29211

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street
Suite 2900
Atlanta, Georgia 30323

Mr. C. S. Hinnant
Plant General Manager
Harris Nuclear Plant
P. O. Box 165
New Hill, North Carolina 27562

Mr. Dayne H. Brown, Director
Division of Radiation Protection
N. C. Department of Environmental,
Commerce & Natural Resources
P. O. Box 27687
Raleigh, North Carolina 27611-7687

Mr. R. B. Starkey
Vice President
Nuclear Services Department
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

[redacted]
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
E. Adensam
P. Anderson
N. Le
OGC
D. Hagan (MNBB 3302)
G. Hill (4) (P1-37)
Wanda Jones (P-130A)
C. Grimes (11E22)
J. Strosnider
K. Karwoski
ACRS (10)
OPA
OC/LFMB
E. Merschoff, RII

cc: Harris Service List

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
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 7, 1992, 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. 31 , 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

Elinor G. Adensam

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 4, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 31

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 4-14	3/4 4-14
3/4 4-16	3/4 4-16
	3/4 4-16a
3/4 4-17	3/4 4-17
3/4 4-19	3/4 4-19
B3/4 4-3	B3/4 4-3

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 and B) 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.

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 must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

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 from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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;

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

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. This definition 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.

7. 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 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; and

9. 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 prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

SURVEILLANCE REQUIREMENTS (Continued)

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

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

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 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 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 inspected,

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

3. Identification of tubes plugged.

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 defective tubes* and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes* and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug 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 defective tubes* and inspect 2S tubes in each other S.G.	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
		Notification to NRC pursuant to Specification 4.4.5.5.c.	Additional S.G. is C-3	Inspect all tubes in each S.G. and plug 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.

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 = 500 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 500 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 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. 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. 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated February 7, 1992, Carolina Power & Light Company (CP&L or the licensee) submitted a request for changes to the Shearon Harris Nuclear Power Plant, Unit 1, (SHNPP) Technical Specifications (TS). The requested changes would revise the SHNPP Operating License to allow the use of an alternate steam generator tube plugging criterion for the portion of the tubes within the tubesheet.

The proposed revision would amend Technical Specification 4.4.5 specifying an F* distance within the tubesheet below which indications of degradation would have no impact on the determination of integrity of a steam generator tube. As a result, steam generator tubes with degradation below the F* distance in the tubesheet region would not require repair or plugging. The licensee believes the proposed change is safe because the presence of the tubesheet in conjunction with the hardroll tube installation process significantly reduces the potential for tube failure and/or leakage within the tubesheet area when compared to the free span portion of the tube. The presence of the tubesheet provides for constraint of the tube, and the tubesheet complements the integrity of the tube by minimizing the amount of deformation a tube can undergo beyond its expanded outside diameter. The proximity of the tube and tubesheet, due to the hardroll expansion, leads to limiting the amount of primary-to-secondary leakage.

The purpose for the development of the proposed criterion is to obviate the need to repair a tube (by sleeving) or remove a tube from service (by plugging) due to the detection of indications, generally by eddy current testing (ECT), in a region extending over most of the length of tubing within the tubesheet. This Safety Evaluation assesses the integrity of the tube bundle with ECT indications on tubes within the tubesheet under normal operating and postulated accident conditions.

The proposed criterion identifies a distance designated F* (referred to as the F* criterion), below the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation, below which tube degradation of any extent does not necessitate either sleeving or plugging. The F*

criterion, according to the licensee's evaluation, provides a similar level of protection for tube degradation in the tubesheet region as that afforded by Regulatory Guide 1.121 for degradation located outside the tubesheet region. Limitations on the use of the F* criterion have also been discussed by the licensee.

2.0 EVALUATION

Engagement Distance Determination

The licensee determined a distance, designated F*, below the bottom of the roll transition or top of the tubesheet, whichever is lower in elevation, for which tube degradation of any extent does not necessitate remedial action, e.g., sleeving or plugging. This criterion would be used in determining whether or not repairing or plugging of full depth hardroll expanded steam generator tubes is necessary for any potential degradation which has been detected in that portion of the tube within the tubesheet.

The proposed criterion forms the basis for obviating the need to repair a tube (by sleeving) or to remove a tube from service (by plugging) due to the detection of indications, i.e., by eddy current testing (ECT), in a region extending over most of the length of tubing within the tubesheet. This evaluation applies to the Westinghouse Model D steam generators at SHNPP and assesses the integrity of the tube bundle, for tube ECT indications occurring on the length of tubing within the tubesheet, relative to

1) maintenance of tube integrity for all loadings associated with normal plant conditions, including startup, operation in power range, hot standby, and cool down, as well as all anticipated transients;

2) maintenance of tube integrity under postulated limiting conditions of primary-to-secondary and secondary-to-primary differential pressure, e.g., steamline break (SLB); and,

3) limitation of primary-to-secondary leakage consistent with accident analysis assumptions.

The F* criterion provides for sufficient engagement of the tube-to-tubesheet hardroll such that pullout forces that could be developed during normal or accident operating conditions would be successfully resisted by the elastic preload between the tube and tubesheet.

In order to evaluate the applicability of any developed criterion for indications within the tubesheet, some postulated type of degradation must necessarily be considered. For this evaluation, it was postulated that a circumferential severance of a tube could occur, contrary to existing plant operating experience. However, implicit in assuming a circumferential severance to occur is the consideration that degradation of any extent could be demonstrated to be tolerable below the location determined acceptable for the postulated condition.

When the tubes have been hardrolled into the tubesheet, any axial loads developed by pressure and/or mechanical forces acting on the tubes are resisted by frictional forces developed by the elastic preload that exists between the tube and the tubesheet. For some specific length of engagement of the hardroll, no significant axial forces will be transmitted further down the tube; and that length of tubing, i.e., F*, will be sufficient to anchor the tube in the tubesheet. In order to determine the value of F* for application in Model D steam generators, a testing program was conducted to measure the elastic preload of the tubes in the tubesheet.

Tubes are installed in the SG tubesheet by a hardrolling process that expands the tube to bring the outside surface into intimate contact with the tubesheet hole. The roll process and roll torque are specified to result in a metal-to-metal interference fit between the tube and the tubesheet.

A test program was conducted by Westinghouse to quantify the degree of interference fit between the tube and the tubesheet provided by the full depth mechanical hardrolling operation. The data generated in these tests have been analyzed to determine the length of hardroll required to preclude axial tube forces from being transmitted further down the tube, i.e., to establish the F* criterion. The amount of interference was determined by installing tube specimens in collars specifically designed to simulate the tubesheet radial stiffness. A hardroll process representative of that used during steam generator manufacture was used in order to obtain specimens that would exhibit installed preload characteristics like the tubes in the tubesheet. The test configuration consisted of six cylindrical collars. A mill annealed, Inconel 600 (ASME SB-163) tubing specimen was hard rolled into each collar using a process which simulated actual tube installation conditions.

Once the hardrolling was completed, the test collars were removed from the tube specimens; and the springback of the tube was measured. The amount of springback was used in an analysis to determine the magnitude of the interference fit, which is, therefore, representative of the residual tube-to-tubesheet radial load in Westinghouse Model D steam generators.

During plant operation, the amount of preload will change depending on the pressure and temperature conditions experienced by the tube. The room temperature preload stresses, i.e., radial, circumferential and axial, are such that the material is nearly in the yield state if a comparison is made to ASME Code minimum material properties. Since the coefficient of thermal expansion of the tube is greater than that of the tubesheet, heatup of the plant will result in an increase in the preload and could result in some yielding of the tube. In addition, the yield strength of the tube material decreases with temperature. Both of these effects may result in the preload being reduced upon return to ambient temperature conditions, i.e., in the cold condition. However, based on the results obtained from the pullout tests, this is not expected to be the case as, even with a very high thermal relaxation soak, the results show the analysis to be bounding.

The plant operating pressure influences the preload directly based on the application of the pressure load to the inner diameter of the tube, thus, increasing the amount of interface loading. The pressure also acts indirectly to decrease the amount of interface loading by causing the tubesheet to bow upward. This bow results in a dilation of the tubesheet holes, thus reducing the amount of tube-to-tubesheet preload. Each of these effects may be quantitatively treated.

Analytically combining the room temperature hardroll preload with the thermal, pressure, and tubesheet bow effects resulted in a net positive operating preload during normal and faulted operating conditions. In addition to restraining the tube in the tubesheet, it is expected that this preload would effectively retard leakage from indications in the tubesheet region of the tubes.

The applied loads to the tubes that could result in pullout from the tubesheet during all normal and postulated accident conditions are predominantly axial and due to the internal to external pressure differences. For a tube that has not been degraded, the axial pressure load is given by the product of the pressure with the internal cross-sectional area. However, for a tube with internal degradation, e.g., cracks oriented at an angle to the axis of the tube, the internal pressure may also act on the flanks of the degradation. Thus, for a tube that is postulated to be severed at some location within the tubesheet, the total force acting to remove the tube from the tubesheet is given by the product of the pressure and the cross-sectional area of the tubesheet hole. The force resulting from the pressure and internal area acts to pull the tube from the tubesheet, and the force acting on the end of the tube tends to push the tube from the tubesheet. For this analysis, the tubesheet hole diameter has been used to determine the magnitude of the pressure forces acting on the tube. Any other forces such as fluid drag forces in the U-bends and vertical seismic forces, are negligible by comparison.

The calculation of the required engagement distance is based on determining the distance for preload frictional forces to equilibrate the applied operating loads. The axial friction force was found as the product of the radial preload force and the coefficient of friction between the tube and the tubesheet. The value assumed for the coefficient of friction was based on laboratory experiments for a rolled joint under hydraulic loading conditions. For the maximum normal pressure applied load, including a safety factor of three, the length of hardroll required is exceeded by the value for F* that CP&L has proposed. The proposed value for F* is 1.6 inches. Similarly, the required engagement length for faulted conditions, using a safety factor corresponding to an American Society of Mechanical Engineers Code (ASME Code) safety factor of 1.0/0.7 for allowable stress for faulted conditions, is exceeded by the CP&L value for F*.

The F* value determined for the required length of hardroll engagement below the bottom of the hardroll transition (BRT) or the top of the tubesheet, whichever is greater relative to the top of the tubesheet, is sufficient to resist tube pullout during both normal and postulated accident condition loadings. Furthermore, the uncertainty in determining the position of the

eddy current flaw indication and the edge of the roll transition zone has been accounted for in the CP&L proposed F* value of 1.6 inches.

Rolled Tube Pullout Tests

The engagement distance determination was calculated from a derived preload force and an assumed static coefficient of friction for tube to tubesheet contact. A direct measurement of this static coefficient of friction is difficult. However, a simple pull test on a rolled tube joint provided both support for the derived preload force (less the effects of thermal expansion and internal pressure tightening) and an indirect measurement of the static coefficient of friction. The results verify the previously calculated results.

Pullout tests were conducted on several actual rolled tubes that have joints with various amounts of wall thinning. As with the preload tests, the test configuration consisted of mill annealed, Inconel 600 (ASME SB-163) Model D tubing, hard rolled into carbon steel collars with an outside diameter to simulate tubesheet rigidity. Inside surface roughness values of the collars were measured and recorded. The specification of surface roughness for the fabrication of the collars was the same as that used for the fabrication of the Model D tubesheets. Prior to rolling, the tubing was tack rolled and welded to the collar similar to the installation of tubes in the steam generators. The hard rolling was done in a direction away from the weld and in all aspects simulated actual tube installation conditions. After rolling, an inside circumferential cut was machined through the wall of a tube at a controlled distance from the bottom of the hardroll transition (opposite the tube weld). The machined cut simulated a severed tube condition. To simulate any possible effect of reduced preload force due to tube yielding during manufacturing heat treatment and during reactor operation, the samples were subjected to a heat soak. For the tests, there was no increase in the preload due to thermal expansion of the tube relative to the collar. Based on the observed pullout, the coefficient of friction assumed in the engagement distance determination was comparable relative to a dry interface between the tube and collar.

Rolled Tube Hydraulic Proof Tests

Similar to the rolled tube pullout tests, pressure tests were conducted on rolled joints with nominal degrees of wall thinning. As with the preload and pullout tests, the test configuration consisted of mill annealed, Inconel 600 (ASME SB-163) Model D tubing, hard rolled into carbon steel collars with an outside diameter to simulate tubesheet rigidity. As with the pullout test samples, a machined cut was used to simulate a severed tube condition. To simulate any possible effects of reduced preload force due to tube yielding during manufacturing heat treatment, these samples were also subjected to a heat soak. The pressure tests were performed at room temperature using deionized water.

The tests showed that even for rolled joints with lengths less than F* distance and less-than-nominal wall thinning, pressure induced axial forces of several thousands of pounds or greater were necessary to cause the tube to

release from the tubesheet; thus, the preload based calculations of required engagement distance was verified to be adequate.

Limitation of Primary-To-Secondary Leakage

As described above, the F^* criterion requires a minimum length of hardroll engagement below the top of the tubesheet or the bottom of the roll transition, whichever is greater relative to the top of the tubesheet. For SHNPP, an F^* distance of 1.6 inches has been proposed. The presence of the elastic preload presents a significant resistance to flow of primary-to-secondary or secondary-to-primary water for degradation which has progressed fully through the thickness of the tube wall. In effect, no leakage would be expected if a sufficient length of hardroll is present. This has been demonstrated in high pressure fossil boilers where hardrolling of tube to the tubesheet joints is the only mechanism resisting flow, and in steam generator sleeve-to-tube joints made by the Westinghouse hybrid expansion joint process. This was also confirmed by the hydraulic proof test specimens which showed no leakage at pressures exceeding normal operating conditions. Because of the difficulty in accurately sizing stress corrosion crack indications, the technical specifications require that no indications of cracking can be present within the F* distance in tubes to which the F* criterion is applied. This requirement has the effect of preventing the start of a leak path.

Tube Integrity Under Postulated Limiting Conditions

The final aspect of the evaluation is to demonstrate tube integrity under the postulated loss of coolant accident (LOCA) condition of secondary-to-primary differential pressure. A review of tube collapse strength characteristics indicates that the constraint provided to the tube by the tubesheet gives a margin between tube collapse strength and the limiting secondary-to-primary differential pressure condition, even in the presence of circumferential or axial indications.

3.0 EVALUATION OF PROPOSED TECHNICAL SPECIFICATIONS

The licensee proposed revisions to its Technical Specifications to implement the F* criterion in a letter dated February 7, 1992 (Serial: NLS-92-017). The following addresses the changes proposed in the Technical Specifications (TS) for implementation of the F* criterion.

1. The TS contains a definition of the F* distance, which is 1.6 inches, and a definition of an F* tube, which is a tube left in service by application of the F* criterion.

2. The TS contain a specific provision for reinspection of F* tubes in addition to the normal TS required sampling.

3. Special reports containing the results of inspection or reinspection of F* tubes are to be submitted to the Commission prior to restart.

4. The F* criterion or plugging limit is defined such that tubes need not be plugged because of ECT indications, equal to or greater than 40% through-wall, that are below the F* distance from the top of the tubesheet (or from the top of the last hardroll whichever is lower) provided the tube is not degraded within the F* distance. The restriction on no degradation within the F* distance means that there are no indications of cracking. This restriction has been incorporated because of the difficulty in accurately sizing stress corrosion cracking. It is recognized that stress corrosion cracking that appears by ECT to be shallow may in fact be considerably deeper. In addition, the engagement distance analysis and the testing program were based upon tubes that do not contain imperfections.

This TS provides acceptable implementation of the F* criterion as analyzed in the Westinghouse Report (WCAP 12816) and evaluated in Section 2.0 of this Safety Evaluation Report.

4.0 SUMMARY

Based on a review of the licensee's submittal, the staff concludes that tubes can be left in service with eddy current indications of pluggable magnitude that are below the F* distance provided the tube is not degraded within the F* distance. The F* distance is defined as 1.6 inches from the top of the tubesheet or from the top of the last hardroll, whichever is lower.

From the results of the testing and analysis, it is concluded that following the installation of a tube by the standard hardrolling process, a residual radial preload stress exists due to the plastic deformation of the tube and tubesheet interface. This residual stress is expected to restrain the tube in the tubesheet while providing a leak limiting seal condition even if the tube is completely severed circumferentially at the F* distance below the top of the tubesheet.

The staff concludes that the proposed TS changes on Steam Generators Surveillance Requirements, Section 3/4.4.5, as detailed in the letter dated February 7, 1992, may be incorporated in an amendment to the operating license for the SHNPP.

5.0 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.

6.0 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 (57 FR 11104). 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.

7.0 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.

Principal Contributor: K. Karwoski

Date: September 4, 1992