Docket Nos.: 50-369
and 50-370

August 19, 1986

Mr. H.B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: ISSUANCE OF AMENDMENT NO.59 TO FACILITY OPERATING LICENSE NPF-9 AND AMENDMENT NO. 40TO FACILITY OPERATING LICENSE NPF-17 - MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No.59 to Facility Operating License NPF-9 and Amendment No.40 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated June 24, 1986, as revised August 5, 13, and 18, 1986, and supplemented July 1 and 23, 1986.

The amendments change the Technical Specifications to modify steam generator tube plugging requirements for tube defects located in the tubesheet region. The amendments are effective as of their date of issuance and until the end of the fifth refueling outage of the applicable unit.

Minor changes to Bases 3/4.4.5 have been made in accordance with telephone discussions with Mr. J. Day of your company on August 14, 1986. These changes to the Bases more accurately reflect tube degradation detection capabilities and the requirement that F* tubes not be degraded within the F* distance.

A copy of the related safety evaluation supporting Amendment No.59 to Facility Operating License NPF-9 and Amendment No. 40to Facility Operating License NPF-17 is enclosed.

Notice of issuance of the amendments will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/S/

Darl Hood, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 59to NPF-9
2. Amendment No. 40to NPF-17
3. Safety Evaluation

cc w/enclosures: See next page

DSH		PWOl
PWR#4/DPWR-A	PWR#4/DPWR-A	for PWR#4/DPWR-A
DHood/rad	MDuncan	BJYoungblood
08/18/86	08/18/86	08/18/86

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 59
License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-9 filed by the Duke Power Company (the licensee) dated June 24, 1986, as revised August 5, 13, and 18, 1986 and supplemented July 1 and 23, 1986 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.59, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Darl Hood, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment:
Technical Specification
Changes

Date of Issuance: August 19, 1986

DSH
PWR#4/DPWR-A
DHood/rad
08/18/86

PWR#4/DPWR-A
MDuncan
08/18/86

OGC-Beth
Johnson
08/19/86
w/noted changes

PWR#4/DPWR-A
for BJYoungblood
08/18/86

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-17 filed by the Duke Power Company (the licensee) dated June 24, 1986, as revised August 5, 13, and 18, 1986 and supplemented July 1 and 23, 1986 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 4Q are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Darl Hood, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment:
Technical Specification
Changes

Date of Issuance: August 19, 1986

DSH
PWR#4/DPWR-A
DHood/rad
08/18/86

PWR#4/DPWR-A
MDuncan
08/18/86

OGC-Beth
JOHNSON
08/19/86
w/noted changes

Pwool
PWR#4/DPWR-A
for BJYoungblood
08/18/86

ATTACHMENT TO LICENSE AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Amended Page	Overleaf Page
3/4 4-12	3/4 4-11
3/4 4-14	3/4 4-13
3/4 4-15	
B 3/4 4-3	

3/4.4.5 STEAM GENERATORS

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 Table 4.4-2. 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:

SURVEILLANCE REQUIREMENTS (Continued)

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

3) A tube inspection (pursuant to Specification 4.4.5.4.a.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. In addition to the 3% sample, all F* tubes will be inspected.

d. The tubes selected as the second and third samples (if required by Table 4.4-2) 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.

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)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

    1) Reactor-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

    2) A seismic occurrence greater than the Operating Basis Earthquake,

    3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, and

    4) A main steam line or feedwater line break.

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 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

NOTE 1: The application of F* expires at the end of the fifth fuel cycle for each respective unit.

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 after the field hydrostatic test and 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 top 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 2 inches.

11) F* TUBE is a tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) in the F* distance.

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 Table 4.4-2.

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. The results of inspections of F* tubes shall be reported to the Commission in a report to the Director, ONRR, 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.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

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. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness. For tubes with degradation below the F* distance, and not degraded within the F* distance, plugging is not required.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

McGUIRE NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letters dated June 24, 1986, July 1 and 23, 1986, August 5, 13, and 18, 1986, Duke Power Company (the licensee for McGuire Nuclear Station, Units 1 and 2) requested a revision to the Technical Specifications (TS), Section 3/4.4.5., "Steam Generators." This revision seeks to change the plugging limit definition in TS 4.4.5.4.a. and would exclude from plugging those tubes with indications approximately 2 inches or greater below the top of the tubesheet provided that the top 2 inches of the tube within the tubesheet is not degraded. Westinghouse Reports WCAP 11224 and WCAP 11225 "Tubesheet Region Plugging Criterion," which are part of the TS amendment request, address the issue of plugging full depth hardroll expanded steam generator tubes which may have experienced degradation within the tubesheet area and provide the technical justification for the licensee's TS change request. WCAP 11225 is a nonproprietary version of WCAP 11224.

Existing plant TS tube plugging criteria apply throughout the tube length and do not take into account the reinforcing effect of the tubesheet on the external surface of the tube. The presence of the tubesheet will constrain the tube and will complement its integrity in that region by precluding tube deformation beyond its expanded outside diameter. The resistance to both tube rupture and tube collapse is significantly strengthened by the tubesheet. In addition, the proximity of the tubesheet significantly affects the leak behavior of throughwall tube cracks in this region, i.e., no significant leakage relative to plant technical specification allowables is to be expected. For these reasons, consideration of the use of an alternate criterion for plugging is justified.

The purpose for the development of the proposed criterion is to obviate the need to remove a tube from service (by plugging) due to 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* and referred to as the F* criterion, below the top of the tubesheet below which tube degradation of any extent does not necessitate plugging. The F* criterion, according to the licensee's evaluation, provides the same level of protection for tube degradation in the tubesheet region as that afforded by Regulatory Guide (RG) 1.121 for degradation located outside the tubesheet region. Limitations on the use of the F* criterion have also been discussed by the licensee.

EVALUATION

1. Westinghouse Reports WCAP 11224 and WCAP 11225

a. Engagement Distance Determination

The purpose of this development was the identification of a distance, designated F* (and identified as the F* criterion), below the top of the tubesheet below which tube degradation of any extent does not necessitate plugging. This criterion would be used in determining whether or not plugging of full depth hardroll expanded steam generator tubes is necessary for degradation which has been detected in that portion of the tube which is within the tubesheet.

Tube rupture in the conventional sense, i.e., characterized by an axially oriented "fishmouth" opening in the side of the tube, is not possible within the tubesheet. The reason for this is that the tubesheet material prevents the wall of the tube from expanding outward in response to the internally acting pressure forces. The forces which would normally act to cause crack extension are transmitted into the walls of the tubesheet, the same as for a nondegraded tube, instead of acting on the tube material. Thus, axially oriented linear indications, e.g., cracks, cannot lead to tube failure within the tubesheet and may be considered on the basis of leakage effects only.

Likewise, a circumferentially oriented tube rupture is resisted because the tube is not free to deform in bending within the tubesheet. When degradation has occurred such that the remaining tube cross sectional area does not present a uniform resistance to axial loading, bending stresses are developed which may significantly accelerate failure. When bending forces are resisted by lateral support loads, provided by the tubesheet, the acceleration mechanism is mitigated and a tube separation mode similar to that which would occur in a simple tensile test results. Such a separation mode, however, requires the application of significantly higher loads than for the unsupported case.

The proposed criterion forms the basis for obviating the need to remove a tube from service (by plugging) due to detection of indications, e.g., by eddy current testing (ECT), in a region extending over most of the length of tubing within the tubesheet. This evaluation applies to the McGuire Units 1 and 2 Westinghouse Model D steam generators and assesses the integrity of the tube bundle for tube ECT indications occurring in 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 cooldown, 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) or feedwater line break (FLB).

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 even in the event of a circumferential break in the tube below the F* distance. The necessary engagement length applicable to the McGuire Units 1 and 2 steam generators was found by Westinghouse to be less than the two inch F* distance proposed by the licensee for McGuire based on preload analysis performed by Westinghouse.

Verification that the Westinghouse value is conservative was demonstrated by both pullout and hydraulic proof testing of tubes in tubesheet simulating collars. Application of the F* criterion provides a level of protection for tube degradation in the tubesheet region commensurate with that afforded by RG 1.121 for degradation located outside the tubesheet region.

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 along 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.

The presence of the elastic preload also 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.

Tubes are installed in the steam generator tubesheet by a hardrolling process which 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 has been analyzed to determine the length of hardroll required to preclude axial tube forces from being transmitted further along 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 which would exhibit installed preload characteristics like the tubes in the tubesheet.

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.

The test program was designed to simulate the interface of a tube-to-tubesheet full depth hardroll for a model D steam generator. 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.

Following the taking of all post roll measurements, the collars were saw cut to within a small distance from the tube wall. The collars were then split for removal from the tube, and tube ID and OD measurements repeated. In addition, the axial length of the tube within the collar was measured both before and after collar removal.

The data recorded was that necessary to determine the interfacial conditions of the tubes and collars. These consisted of the ID and OD of the tubes prior to and after rolling and removal from the collars as well as the inside and outside dimensions of each collar before and after tube rolling. Two orthogonal measurements were taken at each of six axial locations within the collars and tubes. In addition, gage marks were put on the tubes so that any axial deformation that occurred during collar removal might be monitored. The remainder of the data of particular interest was calculated from these specific dimensions. The calculated dimensions included wall thickness, change in wall thickness for both rolling and removal of the tubes from the collars, and percent of springback. Using the measured and calculated physical dimensions, an analysis of the tube deflections was performed to determine the amount of preload radial stress present following the hardrolling.

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.

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 conservative.

The plant operating pressure influences the preload directly based on the application of the pressure load to the ID 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 dilatation of the tubesheet holes, thus, reducing the amount of tube-to-tubesheet preload. Each of these effects was quantitatively treated.

The maximum amount of tubesheet bow loss of preload will occur at the top of the tubesheet. Since F* is measured from the bottom of the hardroll transition (BRT) or the top of the tubesheet, and leakage is to be restricted by the portion of the tube above F*, the potential for the tube section above F* to experience a net loosening during operation was considered for evaluation. The effects of the three identified mechanisms affecting the preload are considered as follows:

1) Thermal Expansion Tightening - The mean coefficient of thermal expansion for the Inconel tubing between ambient conditions and 600°F is 7.80×10^{-6} in/in/°F. That for the steam generator tubesheet is 7.28×10^{-6} in/in/°F. Thus, there is a net difference of 0.52×10^{-6} in/in/°F in the expansion property of the two materials. Considering a temperature difference of 550°F between ambient and operating conditions the increase in preload between the tube and the tubesheet was calculated.

   The results indicate that the increase in preload radial stress due to thermal expansion is positive during both normal operating and faulted conditions.

2) Internal Pressuring Tightening - The maximum normal operating differential pressure from the primary-to-secondary side of the steam generator is during a loss of load transient. The internal pressure acting on the wall of the tube will result in an increase of the radial preload in proportion to the increase in primary side pressure during the transient.

   In actuality, the increase in preload will be dependent on the internal pressure of the tube since water at secondary side pressure would not be expected between the tube and the tubesheet.

   For both normal and faulted conditions the results indicate that the preload radial stress is increased.

3) Tubesheet Bow Loosening - An analysis of the Model D3 steam generator tubesheet was performed to evaluate the loss of preload stress that would occur as a result of tubesheet bow. Basically, the deflection of the tubesheet was used to find the stresses active on the top surface and then the presence of the holes was accounted for. For the location where the loss of preload is a maximum, the radial preload stress would be reduced during normal operation and faulted SLB operating conditions. During LOCA the differential operating pressure is from secondary to primary. Thus, the radial preload will increase as the tubesheet bows downward.

Combining the room temperature hardroll preload with the thermal and pressure effects results in a net positive operating preload during normal and faulted operation. In addition to restraining the tube in the tubesheet, this preload should effectively retard leakage from indications in the tubesheet region of the tubes.

The applied loads to the tubes which 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 which 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 which is conservatively 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. 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 length 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 for sliding of nickel on mild steel under "greasy" conditions.

For the maximum normal pressure applied load with a safety factor of 3, the length of hardroll required is exceeded by the McGuire conservative value for F* of two inches.

Similarly, the required engagement length for faulted conditions using a safety factor corresponding to an ASME Code safety factor of 1.0/0.7 for allowable stress for faulted conditions is similarly exceeded by the McGuire F* value.

The F* value thus determined for the required length of hardroll engagement below the 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 position of the ECT indication must be added to the criterion for the final calculation of F*. A conservative allowance for uncertainty in ECT position indication is available in the conservative F* distance of 2 inches for the McGuire Units 1 and 2 TS 3/4.4.5.

b. Rolled Tube Pullout Tests

The engagement distance determination discussed above 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 of the testing verify the calculation as being conservative.

Pullout tests were conducted on several actual rolled 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 D3 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. After rolling, an inside circumferential cut was machined through the wall of the 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.

Based on the observed pullout forces, the coefficient of friction assumed in the engagement distance determination was verified to be conservative.

c. Rolled Tube Hydraulic Proof Tests

Similar to the rolled tube pullout tests, pressure tests were conducted on rolled joints and 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 D3 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 water.

These proof tests showed that even for rolled joints considerably less than the F* distance in length at less than nominal wall thinning, pressure induced axial forces of several thousands of pounds or greater are necessary to cause the tube to release from the tubesheet. Thus, the preload based calculation of required engagement distance is indicated to be conservative.

d. Primary-to-Secondary Leakage Considerations

As described above, to apply the F* criterion the applicable tube must have a certain minimum length of hardroll engagement below the top of the tubesheet or the BRT, whichever is greater relative to the top of the tubesheet. For McGuire Units 1 and 2 the licensee has conservatively established an F* distance of two inches. Because of the interfence fit created by the hardrolling operation, no leakage is expected to occur between the tube and the tubesheet regardless of the condition of the tube below the F* distance. This was confirmed by the hydraulic proof test specimens which were pressurized up to and in excess of the faulted operating conditions.

Because of the difficulties 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.

e. 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 significant margin between tube collapse strength and the limiting secondary-to-primary differential pressure condition, even in the presence of circumferential or axial indications.

2. Technical Specification Changes

The licensee first proposed TS changes to implement the F* criterion in à letter dated August 5, 1986. Based upon discussions with the Commission the licensee revised these proposed TS changes by letters dated August 13 and 18, 1986. The following addresses the changes in the TS proposed in the August 18, 1986 letter implementing the F* criterion.

a. The TS is changed to contain a definition of the F* distance (i.e., the distance into the tubesheet from the top 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 two inches) and a definition of a F* tube, (i.e., a tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) within the F* distance).

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

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

d. 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 provided the tube is not degraded (i.e., no indications of cracking) within the F* distance. The restriction on no degradation within the F* distance has been incorporated (1) because of limitations in accurately sizing stress corrosion cracking (i.e., stress corrosion cracking that appears by ECT to be shallow may in fact be considerably deeper), and (2) because the engagement distance analysis and the testing program were based upon tubes that did not contain imperfections.

e. The application of the F* criterion is being approved for about two cycles of operation (i.e., until the end of the fifth fuel cycle) for each McGuire unit. This time provision is included in the proposed TS at the request of the Commission to give the Commission the opportunity to review and evaluate the results of subsequent inspections before extending or revising Commission approval for use of the F* criterion.

f. The Bases are supplemented to reflect the addition of F* criterion to the TS.

The Commission has reviewed the TS changes as revised through the August 18, 1986, letter and finds them acceptable. These TS changes provide acceptable implementation of the F* criterion as analyzed in the Westinghouse Reports and evaluated in this Safety Evaluation Report.

Accordingly, the Commission concludes that tubes can safely be left in-service with eddy current indications of pluggable magnitude that are located below the F* distance provided the tube is not degraded within the F* distance. The F* distance is defined as two 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, the Commission concludes 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 sufficient 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. Nevertheless, until behavior of F* tubes has been confirmed by actual operation, the Commission concludes that its approval of these amendments should be limited to about two cycles of operation for each McGuire unit.

ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, the Commission has determined that issuance of the amendments will have no significant impact on the environment (51 FR 29173).

CONCLUSION

Notice of opportunity for a prior hearing was published in the Federal Register on July 16, 1986 (51 FR 25777). No requests for a hearing were received.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Darl S. Hood, PWR #4 Licensing-A
C. Sellers, PAEB, PWR Licensing-A

Dated: August 19, 1986

U.S. NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. to Facility Operating License No. NPF-9, and Amendment No. to Facility Operating License No. NPF-17, issued to Duke Power Company (the licensee), which revised the Technical Specifications for operation of the McGuire Nuclear Station, Units 1 and 2 (the facility) located in Mecklenburg County, North Carolina. The amendments are effective as of the dates of their issuance.

The amendments change the Technical Specifications to modify steam generator tube plugging requirements for tube defects located in the tubesheet region.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on July 16, 1986 (51 FR 25777). No request for a hearing or petition for leave to intervene was filed following this notices.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact (51 FR 2973) related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated April 1976 and its addendum dated January 1981.

For further details with respect to the action see (1) the application for amendment dated June 24, 1986, as revised August 5, 13, and 18, 1986, and as supplemented July 1 and 23, 1986, (2) Amendment No.59 to License No. NPF-9 and Amendment No. 40to License No. NPF-17, and (3) the Commission's related Safety

Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28242. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington,D.C. 20555, Attention: Director, Division of PWR Licensing-A.

Dated at Bethesda, Maryland this 19th day of Aug. 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Paul O'Connor, Acting Director
PWR Project Directorate #4
Division of PWR Licensing-A

PWR#4/DPWR-A	PWR#4/DPWR-A	OGC-Beth	PWR#4/DPWR-A
MDuncan/rad	DHood		for BJYoungblood
08/19/86	08/19/86	08/19/86	08/20/86

Dated: August 19, 1986

AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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