May 5, 1983

Docket No. 50-369

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. 21 to Facility Operating License NPF-9 - McGuire Nuclear Station, Unit 1

The Nuclear Regulatory Commission has issued Amendment No. 21 to Facility Operating License NPF-9 for the McGuire Nuclear Station, Unit 1, located in Mecklenburg County, North Carolina.

This amendment is in response to your letters dated February 3 and April 28, 1983. The amendment adds to the operating license a condition for verifying acceptability of Model D2 steam generator design modifications. The amendment is effective as of its date of issuance.

A copy of the related safety evaluation report supporting Amendment No. 21 to Facility Operating License NPF-9 is enclosed. Also enclosed is a copy of a related notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 21
2. Safety Evaluation
3. Federal Register Notice

cc w/encl:
See next page

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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DUKE POWER COMPANY

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
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 (licensee) dated February 3 and April 28, 1983, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   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, Facility Operating License NPF-9 is amended as follows:

   A. Add paragraph 2.C.(12) to read:

      (12) Steam Generator Design Modification

      The licensee shall conduct the inspection, testing and monitoring program as described in the attachment to

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Hal B. Tucker's letters of February 3 and April 28 (revised), 1983. The licensee shall not make any major modifications to this program unless prior NRC approval is received.

Major modifications are defined as:

a. Elimination of any identified testing, inspection or monitoring,

b. Changes in the frequency of performing the identified testing, inspection or monitoring, and

c. Reduction in the scope of any of the identified testing, inspection or monitoring.

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

FOR THE NUCLEAR REGULATORY COMMISSION

S/

Darrell G. Eisenhut, Director
Division of Licensing

Date of Issuance: May 5, 1983

*NOTE: SEE PREVIOUS WHITE FOR CONCURRENCE

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SURNAME	*MDuncan/hmc	*RBirkel	*Cutchin	EAdensam	TNovak	DEisenhut	
DATE	4/13/83	4/13/83	4/18/83	5/5/83	5/5/83	5/5/83	

DUKE POWER COMPANY

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. NPF-9

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   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 (licensee) dated February 3, 1983, 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;

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b. Changes in the frequency of performing the identified testing, inspection or monitoring, and

c. Reduction in the scope of any of the identified testing, inspection or monitoring.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

Date of Issuance:

No legal objection to form of amendment

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SURNAME	MDuncan/hmc	RBirkel	Cutchin	EAdensam	TNovak	DEisenhut	
DATE	4/13/83	4/13/83	4/18/83	4/ /83	4/ /83	4/ /83	

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NPF-9

DUKE POWER COMPANY

I. INTRODUCTION

By letter dated February 3, 1983, Duke Power Company submitted a report entitled "McGuire Nuclear Station - Unit 1 Steam Generator Monitoring Programs," which outlined the specific actions and surveillance programs relative to the McGuire - Unit 1 Model D2 steam generator modification. The proposed surveillance and monitoring program is based on the recommendations made by the Design Review Panel (DRP) in their January 1983 report "D2/D3 Steam Generator Design Modification." The staff evaluation (NUREG-0966, March 1983) of the DRP report concluded that the modification of the D2/D3 steam generators is acceptable and that the modified steam generators can be operated at 100% of their design capacity. The licensee also proposed a programmatic license condition to verify acceptability of the modification.

The DRP identified three specific items to be addressed by each of the utility owners installing the proposed preheater modifications. These items are as follows:

1. Provisions should be made for initial monitoring of inlet pressure oscillations;

2. Plant-specific provisions for assuring feedwater flow and/or feedwater temperature restrictions are met should be described, where applicable;

3. Inservice inspection, eddy current testing and tube vibration monitoring programs and schedules should be described, where applicable.

The means by which each of the above items will be implemented on McGuire Unit 1 and the schedule for programs in item 3 are described in the licensee's submittal of February 3, 1983, with additional information provided in submittals dated March 1, 1983, March 14, 1983, and April 28, 1983 (revised).

II. DISCUSSION AND EVALUATION OF PLANT SPECIFIC ITEMS

A. Inlet Pressure Monitoring

In Section 5.2.13 of its report, the DRP recommended that the pressure oscillations in the feedline be initially monitored throughout the design operating flow range. To accomplish this, the licensee has proposed a

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pressure monitoring program to record data during the power escalation period following installation of the preheater modification.

The intent of this monitoring is to monitor all pressure variations which could affect the fatigue usage factors of the bolts and welds. This is to by accomplished be using a pressure transducer installed in the feedline elbows. Measurements will be made over the design operating flow range, i.e., from 17% power, where main feed flow is initiated, up to 100% power. Power escalation will be made in 3% increments. Measurements will be made during the period that power is increasing as well as at each 3% increment.

The analysis by Westinghouse for steady state pressure fluctuation resulted in the development of curves of allowable peak-to-peak pressure oscillations versus frequency. These were developed for critical modification components most subject to this loading and are based on limiting the oscillating pressure stresses at any frequency to the endurance limit for the material.

Acceptance criteria for this test will be established to verify that the plant measurements fall within the bounding values used by Westinghouse in the analysis of the manifold.

Based on our review of the proposed program for inlet pressure monitoring we find that the licensee has met the requirements of DRP item 1. We therefore find the program acceptable.

B. Feedwater System Changes

The McGuire steam generators are provided with separate inlet connections for main feedwater and auxiliary feedwater piping. The auxiliary feedwater system (AFWS) is used to provide makeup to the steam generators during plant startup until steam generator flow requirements approach the AFWS design capability. At this stage of plant warmup, the main feedwater system is actuated. When main feedwater is introduced into the lower main feedwater inlet nozzles, the colder water that has stagnated downstream of the main feedwater isolation valve is injected into the steam generator. Westinghouse has calculated that the ensuing thermal transient will result in an overstressed condition on several of the proposed inlet distribution manifold bolts. The problem relates to a combination of low feedwater line purge flow and cold feedwater in the feedline between the isolation valve and the steam generator. The DRP recommended that each utility provide some plant specific method to alleviate this situation.

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A bypass line and locked-open manual valve will be provided to bypass the main feedwater check valve at the steam generator to allow backflow into the main feedwater line. Backflow flushing of the main feedwater lines will be performed during plant warmup when the AFWS is being used for steam generator makeup prior to actuation of the main feedwater system. Hot effluent from the steam generator will bypass the check valve and preheat the main feedwater inlet piping up to the main feedwater isolation valve outside containment. Backflow will continue until the main feedwater piping is adequately preheated as determined by three thermocouples provided inboard of the containment isolation valve. Two new sections of feedwater piping will be added to permit the bypass backflow to be directed to the condenser hotwell during plant warmup. These new sections of piping will be interconnected with the existing main feedwater flow tempering line. The existing flow tempering line is used to provide a small main feedwater flow through the AFWS inlet to the steam generators. One of the new recirculation piping sections will be located in the doghouse (steam and feedwater valve compartment) adjacent to the containment and the other new recirculation piping section and valves will be located in the turbine building. The new bypass line and manual bypass valve will be located in the containment. An orifice in the new pipeline located in the doghouse will limit backflow through the bypass line to 40 gpm.

The piping and valves in the turbine building are nonseismic Category I and have no safety-related function and, therefore, are not protected from natural phenomena, including tornado missiles. The bypass valve in the reactor building and the piping between the feedwater piping and flow tempering piping (FW-FT piping) in the doghouse are seismic Category I, Quality Group B. The safety-related portion of the system is located in seismic Category I, flood, and tornado protected structures. Thus, the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Classification," Positions C.1 and C.2 are satisfied. The essential equipment is separated from the effects of internally generated missiles. The applicant indicated that the new components could not credibly produce missiles, similar to the original components and previously reviewed in Sections 3.5.1.1 and 3.5.1.2 of the McGuire FSAR. The utility has provided the results of a high energy pipe break analysis using the guidelines of the Standard Review Plan Sections 3.6.1 and 3.6.2 for the new piping. This analysis included additional pipe break locations and the effects of pipe whip, jet impingement, flooding environmental effects, and the potential loss of any safety-related equipment in the area. Any safety equipment which is required to operate after the high energy pipe break is protected by shielding from jet impingement and other effects of discharging fluids such as splashing or dripping. There is no moderate energy piping being added by this modification. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are satisfied.

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This modification is for each unit and there is, therefore, no sharing between units. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The feedwater system is not required to transfer heat under accident conditions and, therefore, General Design Criteria 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water Systems," are not applicable. The open bypass line could result in a 40 gpm leakage path around the steam generator check valve. However, the licensee has verified that redundant isolation valves are provided downstream of the bypass line in the return line to the condenser. The redundant isolation valves receive signals to close from corresponding control trains to isolate the return line on automatic start of the AFWS. Therefore, the addition of the bypass line has no adverse effect on minimum AFWS flow requirements for any accident previously analyzed. Thus, we conclude that the requirements of General Design Criterion 44, "Cooling Water," are satisfied with respect to this feedwater modification not affecting the performance of the AFW system.

Based on the above, we conclude that the modification to the feedwater system meets the requirements of General Design Criteria 2, 4, and 44 with respect to its protection against natural phenomena, missile and environmental effects, and in not affecting the performance of the AFW system in mitigating the consequences of an accident, and meets the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to its seismic classification and is, therefore, acceptable. The modification to the feedwater system meets the applicable acceptance criteria of SRP Section 10.4.7.

The licensee, in proposing reverse flushing of the feedline which would eliminate the thermal transient causing a high usage factor on certain bolts and welds in the modification manifold assembly, has met the requirements of DRP item 2. The feedwater system piping changes being made by the licensee are therefore acceptable.

C. Monitoring Program

In the DRP evaluation report the DRP recommended that each utility develop inspection, testing and monitoring programs specific to their plant(s). These programs are designed to verify the hydraulic performance of the modification and give early indication of any structural problems with the manifold. The DRP's recommended surveillance program included visual inspection of the manifold assembly and baseline ECT of the affected first five rows of tubes in the preheater sections after manifold installation and visual and ECT after a 6 month full power operational period. Tube vibration monitoring of installed accelerometers during power escalation was also recommended. The licensee has supplemented the recommended DRP surveillance as described in the following paragraphs.

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1) Visual Inspection

The visual inspections proposed by the licensee follow the DRP's recommendations. The visual inspections are intended to provide an early indication of any unexpected loss of structural integrity. Therefore, a visual inspection of the accessible areas of the modified components will be performed. Inspection access will be through the radiography port in the feedwater piping upstream of the steam generator nozzle. The inspection will be performed using a fiber optics borescope and will be recorded by videotape or still photographs for future reference. Specific items to be inspected include bolts and welds for erosion, fretting wear, corrosion and cracking. The results of the subsequent inspection will be compared with the as-built condition of the manifold. Any questionable or unusual visual indications will be evaluated to determine the need for corrective action. If corrective action is required, a report detailing the problem and the corrective action will be submitted to the NRC staff prior to subsequent power operation.

The visual inspection described above will be performed following reassembly of the feedwater piping after modification installation and again at the first refueling shutdown. The subsequent schedule and the extent of inspection are described in the licensee's April 28, 1983 (revised) submittal.

The proposed manifold visual examination should be performed in accordance with Boiler and Pressure Vessel Code Section XI IWA-2211 Visual Examinations VT-1.

2) Tube Vibration Monitoring

The DRP endorsed the Westinghouse recommendation for tube vibration monitoring for the first plants modified. Accordingly, four tubes in the 'A' steam generator in McGuire - Unit 1 will be instrumented with two accelerometers each to provide an early indication of manifold performance prior to eddy current testing. Due to the uncertainty in the relationship between tube vibration and wear, no short term acceptance criteria have been established. However, the results of these measurements are expected to be useful in assessing the long term potential for the manifold to reduce the wear rate to an acceptably small value. Axial location of each accelerometer is given in Table 1. Acceleration readings will be recorded for off-line analysis by Westinghouse and the licensee. Limited on-line analysis will be performed to verify the validity of the recorded data.

TABLE 1

Axial Location of Tube Mounted Accelerometers

Row	Column*	Evaluation
49	31 N	Halfway between plates 5 and 6
49	31 N	Halfway between plates 6 and 7
49	40 W	Halfway between plates 5 and 6
49	40 W	At Plate 3
49	60 W	Halfway between plates 5 and 6
49	60 W	At Plate 7
49	71 N	At Plate 3
49	71 N	Halfway between plates 5 and 6

*W - window Tube

N - Non-window Tube

Column numbers are those used by Westinghouse. Licensee numbers columns as a mirror image during ECT.

Data will be recorded during power escalation following installation of the modification. As a minimum, data will be recorded at the following power levels during steady state conditions; 40%, 50%, 60%, 70%, 75%, 80%, 85%, 90%, 95%, and 100%. Appropriate plant data will be recorded concurrent with the accelerometer data for correlation purposes.

For McGuire Unit 1, the licensee will, in addition to measurements taken after startup, take and record data from each of the accelerometers at 100% power during the subsequent operating period approximately halfway between startup and the end of the operational period. These data will be compared with the initial data to verify no significant change in tube behavior.

The design modification is intended to reduce the tube vibration response to acceptable levels, i.e., levels corresponding to 40% power with the original design. The proposed vibration monitoring program should be able to verify that the modification achieves this objective in the tubes 49-31, 49-40, 49-60 and 49-71 to be instrumented with accelerometers.

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The NRC staff and its consultants have reviewed this proposal and concur with it. In particular, the selected tubes include two window tubes (49-40 and 49-60), two non-window tubes (49-31 and 49-71), a tube on the periphery of the bundle (49-31) which is exposed to "skimming" flow, and a central tube (49-60). It should be noted that tubes 49-40 and 49-71 were previously instrumented and data were obtained under operating conditions with the original design.

Because relatively large tube-to-tube support plate (TSP) hole diametral clearances still remain, the potential for a tube to float within a TSP exists. Further, calculations based on the assumption of uniform flow show that at high power levels fluidelastic instability is possible if a tube can vibrate in a TSP-inactive mode. Therefore, a primary purpose of the accelerometer measurements will be to first determine if any of the tubes are vibrating in a TSP-inactive mode, and secondly, to determine if a threshold power level exists above which large amplitudes indicative of an instability occur.

The accelerometer data will be recorded on magnetic tape for subsequent data analysis. The data analysis should include frequency spectra, in the form of power spectral density (PSD) plots, and root mean square (RMS) values which are readily obtained by integration under the PSD curve. Accelerometer signals should be double integrated to obtain displacement data, and the PSDs and RMS values should be obtained for both acceleration and displacement.

Dominant frequency peaks should be identified from the PSD curves and compared with results from vibration analyses of the tubes for different assumed support arrangements to determine if the tube is vibrating in a support-inactive mode. The RMS displacement should be plotted as a function of power level. An abrupt increase in displacement response at a given power level, coupled with a simultaneous sharpening of the frequency response spectra, is indicative of a fluidelastic instability. Additionally, the variation of displacement response with power level (flow velocity) can typically be approximated with a power function relationship. If the exponent on the power level is on the order of 2-3, one can reasonably assume that the response is caused by turbulent buffeting. On the other hand an exponent of four or more may indicate a fluidelastic instability.

An assessment of long term potential of the manifold to reduce tube wear rate will be made after the first refueling outage and acceptance criteria for tube vibration will be established and submitted for staff review. Based on our review of the proposed vibration monitoring program, we find that it has met the requirements relative to vibration monitoring in DRP item 3. We therefore find the program acceptable.

3) Eddy Current Testing

The primary method for assessing the effectiveness of the steam generator modification in reducing the rate of tube wear will be eddy current testing (ECT). The same ECT methods will be used for testing after modification installation that were used in previous ECT examinations to allow comparison of results. The first five rows (45 to 49) will be examined. Although not specifically required, the peripheral tubes (tubes adjacent to the wrapper) will also be examined using the same techniques.

The above inspection will be performed after completion of the modification on each steam generator. This inspection will serve as the baseline inspection for the modified steam generator. A second ECT examination will be performed after the proposed period of operation. This second examination will include the same tubes examined during the initial examination. Subsequent ECT examinations will be performed as required by the McGuire Technical Specifications (which used Regulatory Guide 1.83 for determining inspection frequency) and as outlined in the licensee's April 28, 1983 (revised) submittal.

4) Loose Parts Monitoring

The DRP's recommended surveillance program did not include the use of loose parts monitoring as one means of assuring the continued structural integrity of the installed manifold. McGuire - Unit 1 has an installed loose parts monitoring system (LPMS). This system includes a sensor on the lower head of each steam generator. This system, although intended for detecting loose parts in the primary system, has high enough sensitivity to detect a loose manifold. Although extremely unlikely, if a signal is detected on the LPMS which indicates that one of the manifolds is loose, the unit will be shutdown, NRC will be notified and appropriate corrective action taken.

McGuire Technical Specifications require that daily channel checks, monthly operational tests and 18 month calibrations be performed on the LPMS. Further, the technical specifications require that the LPMS be operable. A report must be submitted to NRC if any channel is inoperable for more than thirty days.

5) Plugged Tubes

The DRP's evaluation report did not address the presence of plugged tubes in a modified steam generator. Operation of McGuire Unit 1 steam generator in the unmodified condition resulted in the plugging of six tubes in November 1982 (one tube which did not have significant wear was plugged in July 1982 due to a misinterpretation of the eddy current signal). These tubes cannot be monitored by eddy current technique directly. Integrity of these six tubes will be inferred from eddy current information on active tubes. If ECT measurements show that some wear has occurred over several inspection intervals, a wear rate for the plugged tubes will be estimated. If a previously plugged tube is thus evaluated to have reached a defect of 80% through wall, a detailed structural evaluation will be performed to demonstrate its integrity prior to returning to service.

The Duke Power Co.'s proposed program of inspection, testing and monitoring will provide sufficient performance verification of the modified steam generators. The licensee has supplemented the DRP's recommended surveillance program with loose parts monitoring for on line detection of sounds, due to foreign objects, loose parts, or a loose manifold, emanating from the area of the installed modification. The McGuire Unit 1 ECT program will also include examination of peripheral tubes in addition to the recommended first five rows of tubes in the preheater section. Tube vibration monitoring of the McGuire Unit 1 will include taking and recording of accelerometer measurements at half way through the proposed period of power operation in addition to the measurements recommended to be taken during power ascension to 100% power.

As a general recommendation, any sensors (such as accelerometers and loose parts transducer) that were utilized in monitoring tests of the original design steam generator should be left in the same locations for the monitoring/tests of the modified design. Results from the original design can then be used as a baseline against which results with the modified design can be compared.

III. DISCUSSION AND EVALUATION OF RADIOLOGICAL CONSIDERATIONS

A. ALARA Guidelines

The staff has evaluated the DRP's radiological assessment of the radiation protection measures established by Westinghouse for the Westinghouse Preheat Steam Generator D2/D3 Design Modification, including those measures intended to ensure that doses will be maintained as low as is reasonably achievable (ALARA). Our assessment is based on the DRP utilization of the criteria outlined in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" in the DRP's radiological assessment of the design modification, and its assessments, primarily those provided in Section 4.4, "Radiological Considerations," Section 5.5, "Radiological Consideration/ALARA," and Section 6.0, "Summary" of the January 1983 DRP Evaluation Report. Information provided in other sections was also considered where it contributed to our assessment of the ALARA features of design, planning, installation, maintenance, and inspection. We have additionally evaluated information specific to the McGuire radiation protection/ALARA program which has previously been submitted in the McGuire Final Safety Analysis Report (FSAR). This has been evaluated and found acceptable by the staff in our McGuire Safety Evaluation Report (SER).

Westinghouse's proposed D2/D3 Design Modification Program provides for radiation protection/ALARA measures throughout the design and preparation stage, the performance of the modification, and during post-modification recovery and operations. The McGuire Unit 1 radiation protection/ALARA program has those features essential for compatibility with the Westinghouse design modification program, and contains radiation protection/ALARA elements designed to ensure adequate radiological protection for workers and promote ALARA doses on tasks associated with the modification. These proposed measures are consistent with 10 CFR 20.1(c) and Regulatory Guide 8.8, and are, therefore, acceptable to the NRC staff for the planned modification of McGuire Unit 1.

B. Evaluation

Based upon our evaluation we find the proposed measures consistent with 10 CFR 20.1(c) and Regulatory Guide 8.8 and, therefore, acceptable. The licensee performed a radiological assessment of the proposed modification for the McGuire Unit 1 prior to task initiation to determine the applicability of proposed worker radiological protective measures and ALARA considerations, and to determine how best to integrate this program with their own facility radiation protection program. A similar assessment will be performed for Unit 2. Where significant differences in any of the radiological parameters considered by Westinghouse exist (e.g., equipment, dose rates, radiation sources, doses, training), these will be evaluated and compensating radiation protection/ALARA actions described. During and upon completion of the modification, the licensee will perform a summary radiological assessment of the task, as is recommended in C.3.c of Regulatory Guide 8.8, to enable the staff to evaluate the radiological results of the modification and determine if additional or different radiological controls need to be considered. This will include the following:

(1) The collective occupational dose estimate shall be updated weekly. If the updated estimate exceeds the person-rem estimate by more than 10%, the licensee shall provide a revised estimate, including the reasons for such changes, to the NRC within 15 days of determination.

(2) A final report shall be provided to the NRC within 60 days after completion of the repair. This report will include:

   (a) a summary of the occupational dose received by major task, and

   (b) a comparison of estimated doses with the doses actually received.

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IV. CONCLUSIONS

We find that McGuire Unit 1 with modifications to the preheater sections of the steam generators may be operated at full power for a period of approximately 200 days without undue risk to the public health and safety. Upon shutdown, in addition to the proposed visual inspections and eddy current tests of preheater tubes, technical specification eddy current tests of 3% of the steam generator tubes should be conducted.

The proposed manifold visual examinations should be performed in accordance with Boiler and Pressure Vessel Code Section XI IWA-2211 Visual Examinations VT-1. In addition, all of the visual examinations of the manifold and ECT of the preheater tubes after installation shall be performed in accordance with the extent and schedule of examination as specified in the licensee's April 28, 1983 (revised) submittal.

In addition we conclude the following:

1. Inlet Pressure Monitoring

   Duke Power plans to monitor the pressure at the feedline inlet nozzle during the power escalation period following the installation of the inlet modification. The licensee will be monitored pressure throughout the design operating flow range. In addition, the licensee will verify that the acceptance criteria established by Westinghouse from test data are applicable to the McGuire steam generators and represent bounding and conservative values.

2. Tube Vibration Monitoring

   With respect to the placement of accelerometers and data gathering techniques, due to uncertainty in the relationship between tube vibration and wear rate, no short term acceptance criteria have been established. An assessment of long term potential of the manifold to reduce tube wear rate will be made after the first refueling outage and acceptance criteria for tube vibration will be established and submitted for staff review.

3. ALARA

   a. Perform dose and ALARA pre-modification assessments for McGuire specific actions, and

   b. Provide a post-task summary radiological assessment as outlined herein.

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We also conclude that the license condition proposed by the licensee be included in the McGuire Unit 1 Facility Operating License:

"The licensee shall conduct the inspection, testing and monitoring program as described in the attachment to Hal B. Tucker's letters of February 3, 1983, and April 28, 1983 (revised). The licensee shall not make any major modifications to this program unless prior NRC approval is received.

"Major modifications are defined as:

a. Elimination of any identified testing, inspection or monitoring,

b. Changes in the frequency of performing the identified testing, inspection or monitoring, and

c. Reduction in the scope of any of the identified testing, inspection or monitoring."

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

V. ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Principal Contributors: L. Frank, Materials Engineering Branch, DE
J. Rajan, Mechanical Engineering Branch, DE
J. Ridgely, Auxiliary Systems Branch, DSI
R. Serbu, Radiological Assessment Branch, DSI
R. Birkel, Licensing Branch No. 4, DL

Date: May 5, 1983

AD:L:DL
TNovak
5/ /83

*NOTE: SEE PREVIOUS WHITE FOR CONCURRENCE

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DATE	4/13/83	4/13/83	4/14/83	4/14/83	4/14/83	4/14/83	5/5/83

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J. Ridgley, Auxiliary Systems Branch, DSI
R. Serbu, Radiological Assessment Branch, DSI
R. Birkel, Licensing Branch No. 4, DL

Date:

Ralph/Helen – Provision which should include [illegible] – E

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DATE	4/13/83	4/13/83	4/14/83	4/14/83	4/14/83	4/15/83	4/ /83	4/ /83

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-369

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

FACILITY OPERATING LICENSE NO. NPF-9

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Facility Operating License No. NPF-9, issued to Duke Power Company (licensee) for the McGuire Nuclear Station, Unit 1 (the facility) located in Mecklenburg County, North Carolina.

This amendment adds to the operating license a condition for verifying acceptability of Model D2 steam generator design modifications. The amendment is effective as of its date of issuance.

Issuance of this amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) Duke Power Company letter dated February 3 and April 28, 1983, (2) Amendment No. 21 to Facility Operating License No. NPF-9 and (3) the Commission's related Safety Evaluation.

OFFICE							
SURNAME							
DATE							

These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., and the Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223. 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 Licensing.

Dated at Bethesda, Maryland, this 5th day of May 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

S/

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

No legal objection to issuance of FR notice

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

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May 11, 1983

Docket No. 50-369

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: McGuire Nuclear Station, Unit 1 (DUKE POWER COMPANY)

One ~~XXX~~ signed originals of the Federal Register Notice identified below is ~~are~~ enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License~~(s) or~~ Amendment(s).
- ☐ Other: ______________________

Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	DL:LB#4					
SURNAME →	MDuncan					
DATE →	5/11/83					

NRC FORM 102 (1-76)

May 5, 1983

AMENDMENT NO. 21 TO
FACILITY OPERATING LICENSE NPF-9 - McGUIRE NUCLEAR STATION, UNIT NO. 1

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Docket No. 50-369
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