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

August 22, 1986

Docket Nos.: 50-369
and 50-370

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.60 to Facility Operating License NPF-9 and Amendment No.41 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No.60 to Facility Operating License NPF-9 and Amendment No. 41to 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 May 15, 1986, and supplemented May 23, June 6 and 30, and August 4 and 12, 1986.

The amendments change the Technical Specifications to reflect the third of several refueling stages involved in the continuing transition to the use of optimized fuel assemblies in Unit 1. This includes a change in axial flux difference limits for Unit 1. For both units, the amendments include changes associated with a more positive moderator temperature coefficient and changes in peripheral fuel assemblies subject to baffle jetting. The latter changes are revised for consistency with your supporting evaluations and are in accordance with the agreement with Mr. Bob Gill during my telephone discussion of August 13, 1986. The amendments are effective as of their date of issuance.

A copy of the related safety evaluation supporting Amendment No.60 to Facility Operating License NPF-9 and Amendment No. 41to Facility Operating License NPF-17 is enclosed.

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Notice of issuance of 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.60 to NPF-9
2. Amendment No.41 to NPF-17
3. Safety Evaluation

cc w/enclosures: See next page

DISTRIBUTION:
See attached page

	DSH	PWDL
PWR#4/DPWR-A	PWR#4/DPWR-A	PWR#4/DPWR-A
MDuncan/mac	DHood	for BJYoungblood
08/15/86	08/14/86	08/20/86

Mr. H. B. Tucker
Duke Power Company

McGuire Nuclear Station

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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. 60
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 May 15, 1986, and supplemented May 23, June 6 and 30, and August 4 and 12, 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.

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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.60, 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 22, 1986

PWR#4/DPWR-A	DSH PWR#4/DPWR-A	OGC/BETH	PWOL PWR#4/DPWR-A
MDuncan:mac	DHood	J. Goldberg	for BJYoungblood
07/30/86	08/14/86	8/18/86	08/20/86

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. 41
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 May 15, 1986, and supplemented May 23, June 6 and 30, and August 4 and 12, 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. 41, 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 22, 1986

	DSH		BWY
PWR#4/DPWR-A	PWR#4/DPWR-A	OGC/BETH	PWR#4/DPWR-A
MDuncan:mac	DHood	J. Goldberg	for BJYoungblood
07/30/86	08/14/86	08/18/86	08/20/86

ATTACHMENT TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 41

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. The corresponding overleaf page is also provided to maintain document completeness.

Amended Page	Overleaf Page
IV	
V	
3/4 1-5a	
3/4 2-3	
3/4 2-4	
5-6	5-5

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

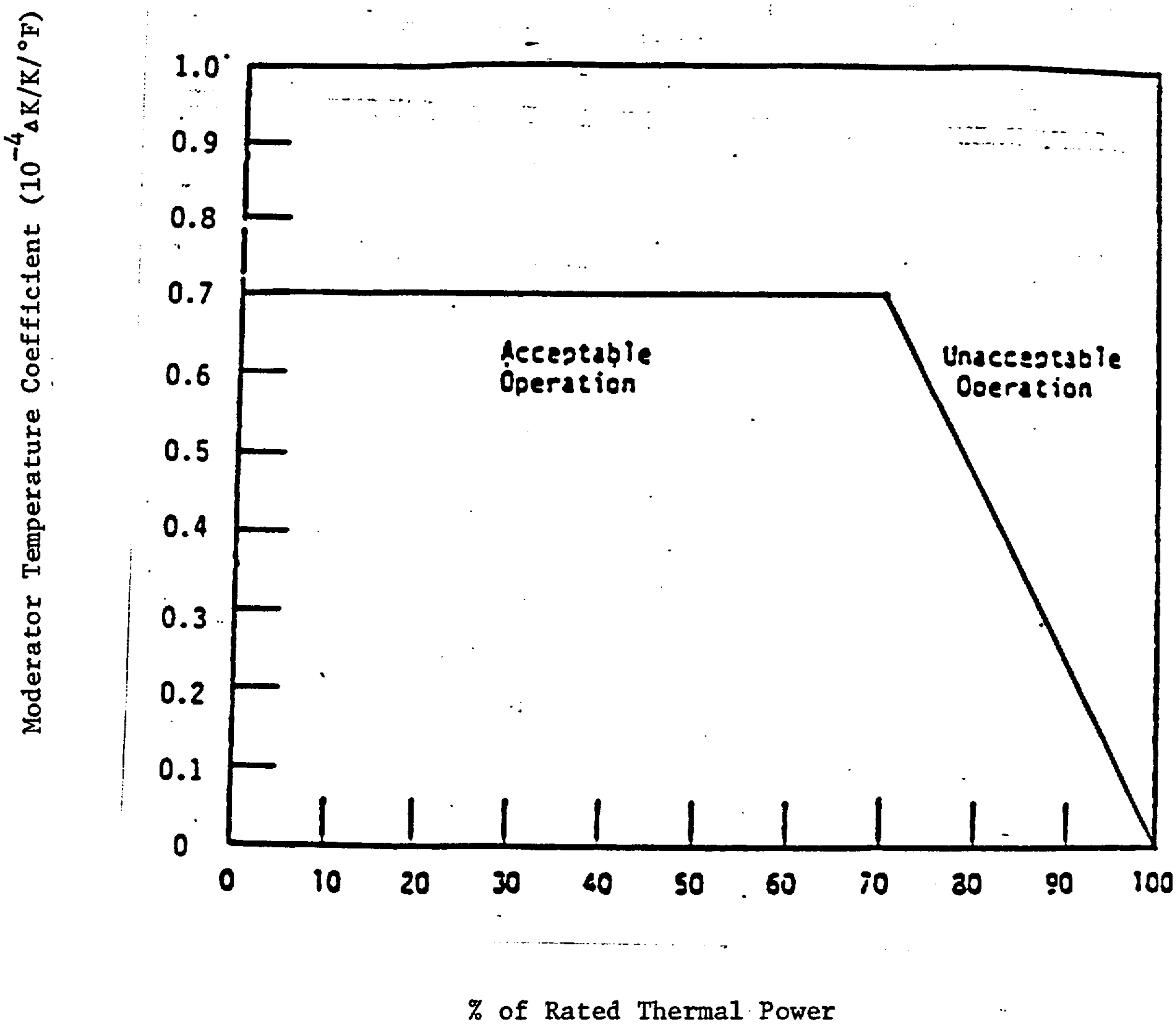


FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

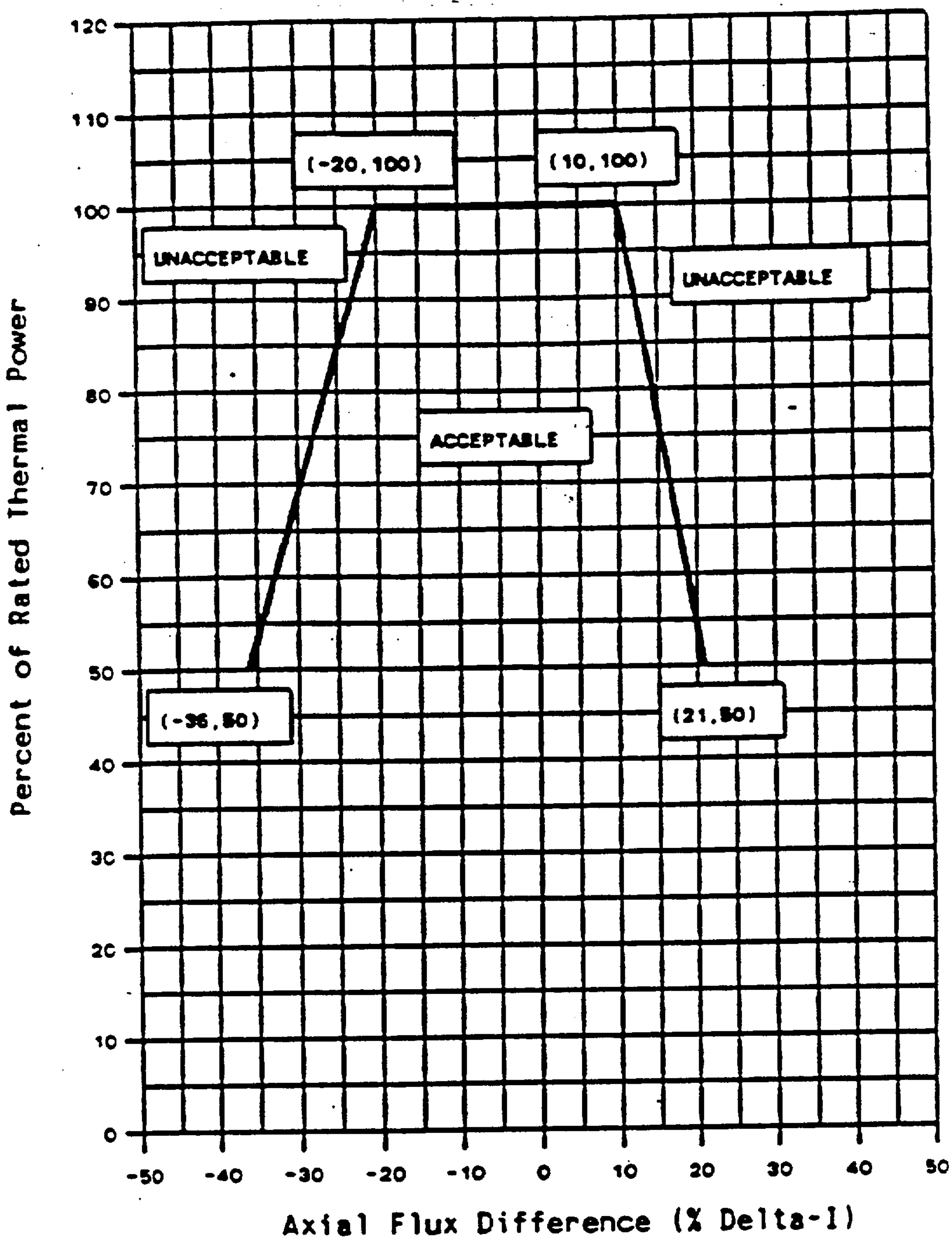


Figure 3.2-1a

AFD Limits as a Function of Rated Thermal Power (Unit 1)

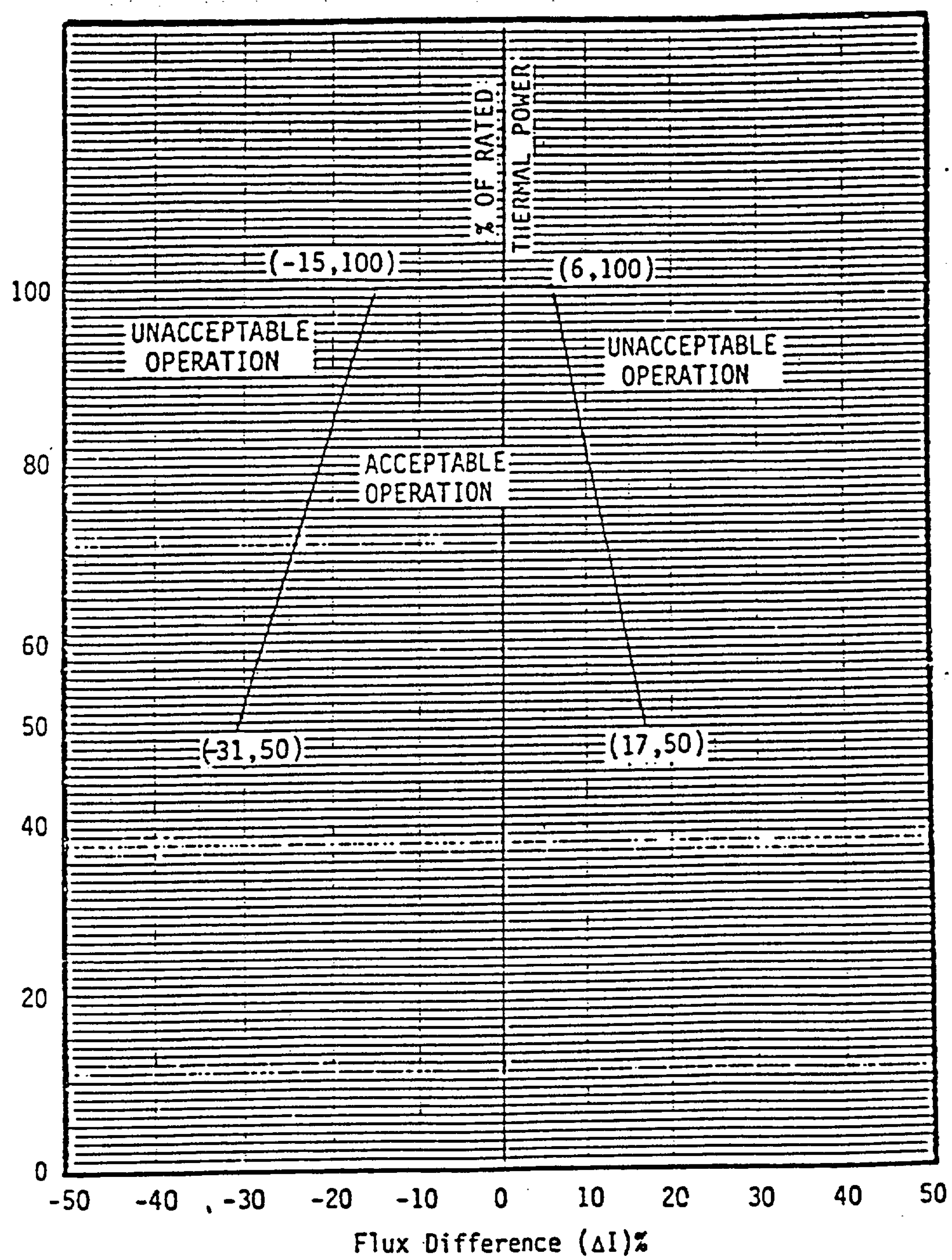
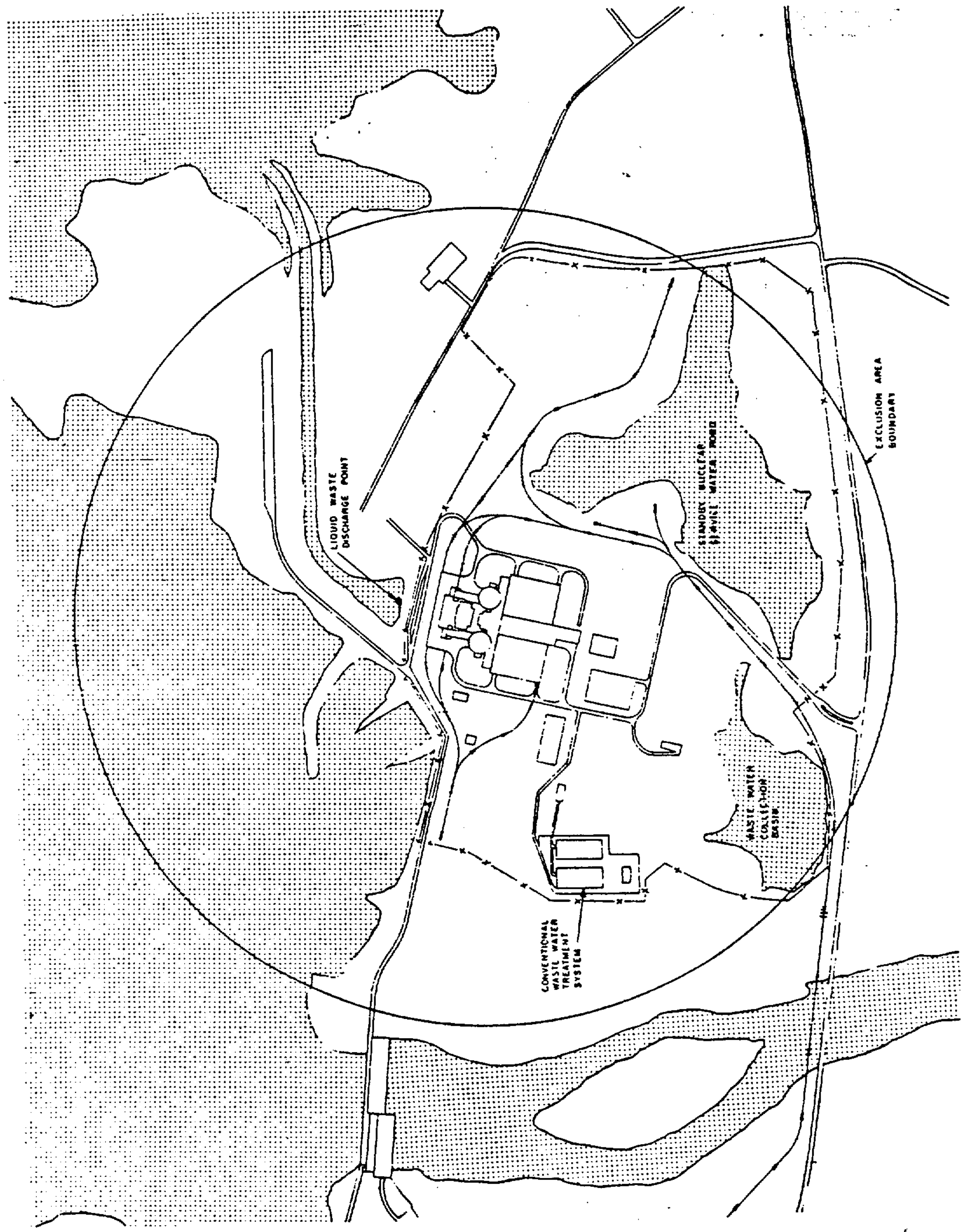


FIGURE 3.2-1
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER (Unit 2)



SITE BOUNDARY FOR
LIQUID EFFLUENTS
McGUIRE NUCLEAR STATION

FIGURE 5.1-4

5.2.1.2 REACTOR BUILDING

a. Nominal annular space = 5 feet.

b. Annulus nominal volume = 427,000 cubic feet.

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 177 feet.

d. Nominal inside diameter = 125 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.25 feet.

g. Dome inside radius = 87 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 15.0 psig and a temperature of 250°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4, except that limited substitutions of fuel rods by filler rods consisting of Zircaloy-4 or stainless steel, or by vacancies, may be made in peripheral fuel assemblies if justified by cycle-specific reload analyses. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material for Unit 1 control rods shall be 80% silver, 15% indium, and 5% cadmium. The nominal values of absorber material for Unit 2 control rods shall be 100% boron carbide (B_4C) for 102 inches and 80% silver, 15% indium, and 5% cadmium for the 40-inch tip. All control rods shall be clad with stainless steel tubing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 60TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

McGUIRE NUCLEAR STATION, UNITS 1 AND 2

I. INTRODUCTION

By letter dated May 15, 1986 (Ref. 1), Duke Power Company (the licensee) made application to amend Facility Operating Licenses NPF-9 and NPF-17 for McGuire Nuclear Station Units 1 and 2, respectively, to reflect (1) the Unit 1-Cycle 4 refueling, (2) a related Technical Specification (TS) change for Unit 1 and (3) a TS change for both units related to application of a positive moderator temperature coefficient. Supplemental information in support of the proposed amendments was provided by licensee's letters dated May 23, June 6 and 30, and August 4 and 12, 1986.

The reload for Cycle 4 reflects the third in a series of four refueling stages for Unit 1 associated with the transition to the use of Optimized Fuel Assemblies (OFA) at McGuire Nuclear Station. The transition began with the Unit 1 refueling for Cycle 2 which was authorized April 20, 1984, by License Amendment 32 (Ref. 2). The TS change for Unit 1 only involves revision of permissible Axial Flux Difference (AFD) limits for Relaxed Axial Offset Control (RAOC) for Cycle 4. The previous TS figure for AFD limits is retained, but for Unit 2 only. The change applicable to both units consists of increasing the allowable positive Moderator Temperature Coefficient (MTC). Prior to these amendments, the TS allowed a +5 pcm/°F MTC at power levels up to 70%, and 0 at power levels above 70%. The revised TS allows an MTC of +7 pcm/°F up to 70%, decreasing linearly above 70% power to 0 pcm/°F at 100% power.

By letters dated August 4 and 12, 1986, the licensee provided a description and revised Reload Safety Evaluation (RSE) for modifications to the McGuire Unit 1-Cycle 4 core reload. During the Cycle 4 refueling outage at McGuire Unit 1, a damaged fuel assembly was found. The damage is attributed to "baffle jetting" (i.e., flow through baffle joints impinging on fuel rods, resulting in induced vibration and subsequent damage). As a result, the Cycle 4 loading pattern was modified to remove the damaged fuel assembly and insert two modified, partially irradiated fuel assemblies, one in the location where the damage occurred during Cycle 3 and the other, as a precaution, in another location where a baffle gap was observed. The licensee's letters included a request for changes to TS 5.3.1 associated with the fuel assembly modifications and deletion of an obsolete sentence.

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II. EVALUATION

1. General Design

McGuire Units 1 and 2 were initially loaded with Westinghouse 17x17 low parasitic fuel assemblies. This fuel is designated by Westinghouse as STD. Commencing with Unit 1 Cycle 2, the McGuire units have been refueling with 17x17 reconstitutable Westinghouse Optimized Fuel Assemblies (OFA). The OFA fuel has similar design features to the STD fuel. The major differences are the use of six intermediate (mixing vane) Zircaloy grids for the OFA fuel versus six intermediate (mixing vane) Inconel grids for STD fuel; a reduction in fuel rod, guide thimble, and instrumentation tube diameters; and the replacement of a standard bottom nozzle with a reconstitutable bottom nozzle.

The OFA fuel has been designed to be mechanically compatible with the STD design, reactor internal interfaces, fuel handling and refueling equipment, and spent fuel storage racks. The top and bottom Inconel (non-mixing vane) grids of the OFA are nearly identical to the Inconel grids of the STD design. The only difference is that the spring and dimple heights have been modified to accommodate the reduced diameter of the OFA fuel rod. The six intermediate (mixing vane) grids are made of Zircaloy rather than Inconel which is currently used in the STD design. The Zircaloy grids have thicker straps than the Inconel grids and the Zircaloy grid height is slightly larger than the Inconel grid height. These dimensional changes were made to compensate for differences in material strength. The Westinghouse 17x17 OFA grid design, which was described in WCAP-9500-A, has been reviewed and approved by the NRC staff (Ref. 3).

The Cycle 4 loading for Unit 1 consists of 64 new OFA assemblies. Thus, the core is constituted almost entirely of OFA fuel, except for 13 remaining STD assemblies. As in Cycles 2 and 3, Cycle 4 also contains one demonstration assembly of an intermediate flow mixer grid design. During the Cycle 2/3 refueling, one removable rod was not reinserted in the demonstration assembly because of mechanical interference. The safety impact for a rod removed (i.e., the effect of an open water channel or "vacancy" within the fuel assembly) was presented in the Cycle 3 RSE (Ref. 6) and accepted by the NRC staff in its SER approving Cycle 3 operation (Ref. 7). The demonstration assembly with its open water channel is retained for Cycle 4 and is included in the licensee's RSE.

The staff review and approval for the transition to OFA (Ref. 2) was based upon a submittal from the licensee dated November 14, 1983 (Ref. 4). This report examined the differences between the Westinghouse OFA and STD designs and evaluated the effects of these differences for the transition to an all OFA core. The evaluation considered the standard reload design methods described in the approved report WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology" (Ref. 5) and the transition effects described for mixed cores in Reference 3. These documents (References 2-4) justified the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core.

In addition to the above, the licensee provided a Reload Safety Evaluation (RSE) for McGuire 1 Cycle 4 as an attachment to Reference 1. The RSE presents a cycle-specific evaluation for Cycle 4 which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the approved reload design methods of Reference 5, WCAP-9272.

The two assemblies selected for modification and use to prevent damage from baffle jetting are from the Region 4 batch already in use in the core. For these fuel assemblies 2 rows of four rods from the corner nearest the baffle joints were to be removed and replaced with stainless steel rods to eliminate the potential for failure. This was accomplished except difficulties were encountered in loading the stainless steel rods in two end cells in one assembly. It was decided that these locations could be left as open water channels. The assembly with the water channels was loaded in the core location having a potential problem, but not the one in which the fuel damage occurred in Cycle 3.

2. Nuclear Design

The Cycle 4 core loading is designed to meet $[F_Q(z) \times P]$ ECCS limit of $\leq 2.26 \times K(Z)$. The cycle-specific RAOC analysis based upon the above $F_Q(Z)$ limits results in a new curve of AFD limits as a function of power. This was submitted as a change to TS Figure 3.2-1a for Unit 1 only. Since this figure was calculated with approved RAOC methods (Ref. 8) and maintains the above $F_Q(Z)$ limits approved for Cycle 3 (Ref. 7), the change to TS Figure 3.2-1a is acceptable. The previous Figure 3.2-1 continues to apply to Unit 2, and is redesignated "Figure 3.2-1b (Unit 2)". This change is purely administrative and is, therefore, also acceptable.

The RSE provides a table of Cycle 4 kinetics characteristics which are compared with the current limits based on previously approved accident analyses.

The RSE also provides a table showing the results of the calculated Cycle 4 control rod worths and requirements at the most limiting condition during the cycle (end-of-life). These results include a standard 10% allowance for calculational uncertainty. From these results, the staff concludes that sufficient control rod worth will be available to provide the required shutdown margin for Cycle 4 operation.

A more positive MTC than the current value is specified for Cycle 4. This is evaluated elsewhere in this SER.

3. Thermal and Hydraulic Design

The thermal hydraulic methodology, DNBR correlation and core DNB limits used for Cycle 4 are consistent with the current licensing basis described in Reference 4 and approved in Reference 2.

The power distributions produced by the cycle-specific RAOC analysis were analyzed for normal operation and Condition II events. Limits on the allowable operating flux difference as a function of power level from these considerations were found to be less restrictive than those resulting from LOCA F_Q considerations. The Condition II analyses generate DNB core limits and resultant Over-Temperature Delta-T setpoints. No changes resulted from the Cycle 4 analysis.

The licensee's evaluation also indicates that the modified peripheral fuel assemblies do not significantly alter the flow velocity profile and grid support conditions, and therefore do not adversely affect the performance of the fuel. This is also confirmed by the experience of several domestic nuclear power plants which have previously used solid SS rods to prevent damage from baffle jetting.

We therefore conclude that the Cycle 4 thermal-hydraulic analysis is acceptable.

4. Accident Analysis

The Cycle 4 kinetics parameters all fall within the bounds upon which the previous applicable safety analysis is based, except for the proposed change to the positive MTC. Thus no accident revaluation would be required, except for the effect of the MTC change. The licensee provided a report on the effect of the MTC change on accident analysis as an attachment to Reference 1. The analysis applies to both McGuire Units 1 and 2, and is evaluated as follows:

The licensee has assessed the impact of a positive MTC of 7 pcm/°F on the accident analyses presented in Chapter 15 of the FSAR. Those incidents which were found to be sensitive to positive or near-zero moderator coefficients were reanalyzed. These incidents are limited to transients which cause the reactor coolant temperature to increase. Accidents not reanalyzed included those resulting in excessive heat removal from the reactor coolant system, for which a large negative moderator coefficient is more limiting, and those for which heatup effects following reactor trip are not sensitive to the moderator coefficient. We agree with the licensee's conclusions about which transients did and did not require reanalysis.

The transients not reanalyzed are:

(1) RCCA misalignment/drop.
(2) Startup of an inactive reactor coolant loop.
(3) Excessive heat removal due to feedwater system malfunctions.
(4) Excessive load increase.
(5) Spurious actuation of safety injection.
(6) Rupture of a main steam pipe.
(7) Loss-of-coolant accident (LOCA)

The incidents reanalyzed, with two exceptions, used a +7 pcm/°F moderator temperature coefficient, assumed to remain constant for variations in temperature. This is conservative, since the proposed change will require the coefficient to ramp to zero at full power. The two exceptions are the rod ejection and the rod withdrawal from subcritical accidents, for which the computer model cannot accept a constant coefficient. The coefficient decrease which occurred during the transients was less than the proposed change, which is acceptable.

The transients reanalyzed and their results are:

A. Boron Dilution

Boron dilution accidents during refueling or startup are terminated by operator action. The proposed MTC does not reduce the time available for operator action in these modes below the acceptable value of 30 minutes from the time the operator is alerted to reactor criticality. This is acceptable. The dilution analysis for power conditions with the reactor in automatic control assumes operator action based on the rod insertion alarm. Analysis of the transient shows the time for operator action remains above the acceptable value of 15 minutes. The dilution from power with the reactor in manual control is bounded by the rod withdrawal transient. Boron dilution accident results will therefore remain acceptable with the proposed MTC.

B. Control Rod Bank Withdrawal from a Subcritical Condition

This transient results in an uncontrolled addition of reactivity leading to a power excursion causing a heatup of the moderator and fuel. The time the core is critical before a reactor trip is very short so that the RCS temperature does not increase significantly; hence the effect of a positive MTC is small. The analysis results show a transient average heat flux which does not exceed the steady state full power value and an increased core water temperature that remains below the full power value. The results show that the DNBR remains above the applicable limit of 1.17 during the transient, which is acceptable.

C. Uncontrolled Control Rod Bank Assembly Withdrawal at Power

This transient produces a mismatch in steam flow and core power, resulting in an increase in RCS temperature. However, the results show that the nuclear flux and overtemperature T trips prevent the core minimum DNBR from falling below the applicable safety analysis limits of 1.47 or 1.49 for thimble and typical cells, respectively, for this transient, which is acceptable.

D. Loss of Coolant Flow

The most severe loss of flow transient is caused by the simultaneous loss of power to all four reactor coolant pumps (RCPs). This case was reanalyzed to determine the effect of the positive MTC on the nuclear power transient and the resultant effect on the minimum DNBR reached during the transient. The minimum DNBR remains above the applicable safety analysis limits of 1.47 or 1.49 for the thimble and typical cells, respectively, during the transient, which is acceptable.

E. Locked Rotor

The locked rotor event is reanalyzed because of the potential effect of the positive MTC on the nuclear power transient and thus on the RCS pressure and fuel temperature. A positive MTC will not affect the time to DNB because DNB is conservatively assumed to occur at the beginning of the transient. The results show peak RCS pressure and peak pellet average and peak cladding temperatures less than the limits used in the previously approved FSAR analyses, which is acceptable.

F. Loss of External Electric Load

The loss of external electric load transient is reanalyzed for both the beginning-of-life (BOL) and end-of-life cases. Since the MTC will be negative at end-of-life, the end-of-life results are essentially the same as in the FSAR. Two beginning-of-life cases are analyzed: (1) reactor in the automatic rod control mode with operation of the pressurizer spray and pressurizer power operated relief valves (PORV); and (2) reactor in the manual control mode with no credit for pressurizer spray or PORVs. The result of a loss of load is a core power that momentarily exceeds the secondary system power removal, causing an increase in RCS coolant temperature. The reactivity addition due to a positive MTC causes an increase in both nuclear power and RCS pressure. The result for the control rods in the automatic control and assuming pressurizer spray and relief at BOL is an RCS pressure of 2531 psia following a reactor trip on high pressurizer pressure. A minimum DNBR well above the applicable limits is reached shortly after reactor trip. The result for the case of rods in manual control with no credit for pressure control is a peak RCS pressure of 2572 psia following a reactor trip on high pressure. The minimum DNBR increases throughout the transient. Since the DNBR remains above the applicable limits and the peak RCS pressure is less than 110% of design, the conclusions presented in the previously approved FSAR analysis are still applicable.

G. Loss of Normal Feedwater/Loss of Offsite Power

These accidents are analyzed to show the ability of the secondary system auxiliary feedwater to remove decay heat from the reactor coolant system. The results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS.

For the case without offsite power, the results verify the natural circulation capacity of the RCS to provide sufficient heat removal capability to prevent fuel or clad damage following reactor coolant pump coastdown.

H. Rupture of a Main Feedwater Pipe

This accident is analyzed to demonstrate the ability of the secondary system auxiliary feedwater to remove heat from the RCS. The results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS to prevent overpressurization or core uncovery. For the case without offsite power, the results verify the natural circulation capacity of the RCS to prevent overpressurization and fuel or clad damage following reactor coolant pump coastdown.

I. Control Rod Ejection

The rod ejection transient is reanalyzed only for BOC since the MTC will be negative at EOC and the existing FSAR analysis remains applicable for EOC. The higher nuclear power levels and hotspot fuel temperatures resulting from a rod ejection are increased by a positive MTC. The results from the BOC reanalysis show that the fuel and clad temperatures are within the limiting values specified in the existing FSAR analysis. The peak hotspot fuel centerline temperature exceeded the melting temperature for the full power case; however, melting was restricted to less than the innermost ten percent of the pellet. The fuel and clad temperatures do not exceed the limits specified in the previously approved FSAR analysis. Therefore, the results of the control rod ejection reanalysis are acceptable.

J. Accidental Depressurization of the Reactor Coolant System

The acceptance criteria for the accidental depressurization of the RCS are shown to be satisfied by predicting a minimum DNBR above the allowable safety analysis limits for this accident of 1.47 and 1.49 for thimble and typical cells, respectively.

Since the reanalysis of the affected plant transients does not result in exceeding any of the fuel limits or safety limits specified in the previously approved reference or FSAR analyses, we conclude that operation with the specified positive moderator temperature coefficient (i.e., a coefficient of +7 pcm/°F up to 70% power, and decreasing linearly from this to 0 pcm/°F at full power) will not pose an undue risk to the health and safety of the public, and is therefore acceptable. The analysis is applicable to both McGuire Units 1 and 2, and therefore the proposed revision of the TS to incorporate the MTC for both units is acceptable.

The licensee also performed an evaluation of the effect of the modified assemblies on the core nuclear design. The results demonstrate that there is no adverse effect on the parameters used in the accident analysis for Cycle 4. Because of this, the licensee concluded that the results of the accident evaluation (discussed above) remain valid. Because the assembly modifications are made at the edge of the core in low power locations, we agree that such changes have little or no effect on the performance or accident evaluation of the core. We, therefore, conclude that our above findings concerning the acceptability of the Cycle 4 reload remain valid.

The licensee proposed a change to Section 5.3.1 of the Design Features section of the Technical Specifications to allow other than 264 fuel rods to be used in fuel assemblies. The specific wording change as initially proposed in the licensee's letter of August 4, 1986, was broadly stated; the specific change would have provided for substitution of SS or Zircaloy-4 fillers and vacancies without limit as to quantity or location within the core, whereas the supporting RSE had assumed limited substitutions of filler rods or vacancies, located only in peripheral fuel assemblies. To eliminate this inconsistency the licensee requested during telephone discussions with the staff that this portion of the request be revised by adding the following words to the end of the first sentence in TS 5.3.1: ", except that limited substitutions of fuel rods by filler rods consisting of Zircaloy-4 or stainless steel, or by vacancies, may be made in peripheral fuel assemblies if justified by cycle-specific reload analyses." We find this to be consistent with the RSE. On the basis of our above conclusion that such assembly modifications have little or no effect on core performance or accident analyses, the favorable experience of other nuclear plants with such substitutions, and the similarities of Unit 1 and Unit 2, we find the change, as revised, is acceptable for Unit 1 and Unit 2.

The licensee also proposed to delete from TS 5.3.1 the sentence, "The initial core loading shall have a maximum enrichment of 3.15 weight percent of U-235." Because this sentence applied only to the initial cores, it is now obsolete and its deletion has no effect upon safety. Therefore, this change is administrative and acceptable.

III. FINDINGS OF EXIGENT CIRCUMSTANCES

The discovery of fuel damage due to baffle jetting during the current Unit 1-Cycle 4 refueling outage has created the need for additional changes to the McGuire Technical Specifications which were not included in the Commission's initial "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Proposed No Significant Hazards Consideration Determination" (51 FR 23484) dated June 27, 1986. These additional changes were limited to TS 5.3.1 and were noticed by 51 FR 28463 on August 7, 1986. In 51 FR 28463, the Commission stated that it had determined that failure to act on the licensee's additional request in a timely way would result in extending the current refueling shutdown for McGuire Unit 1. The licensee's scheduled date for completing the current refueling outage and achieving criticality was noted to be August 26, 1986 (and is now projected to be August 24, 1986). Therefore, the Commission noted that it had insufficient time to issue its usual 30-day notice of the proposed action for public comment.

The licensee contacted the Commission promptly upon discovery of the damaged fuel. The licensee also interacted promptly with Westinghouse to establish corrective measures (involving a redesigned core with two fuel assemblies modified by limited use of solid SS rods) and to obtain revisions to the Reload Safety Evaluation (RSE) needed to complete the licensee's corrective action selection process and to accompany the licensee's supplemental application for amendments. Completion of the revised RSE and the associated application was further impacted by difficulties encountered in loading replacement SS rods in two locations within one of the two modified fuel assemblies, which resulted in a decision to leave these two locations as open water channels and, hence, to also incorporate this change in the RSE and supplemental application for amendments. The licensee's supplemental application for amendments was filed with the Commission promptly upon completion of the supporting RSE by Westinghouse.

The Commission finds that the licensee was unable to avoid the circumstances which created the need for prompt Commission review and approval, and that the licensee has used its best efforts to provide the Commission an opportunity for noticing the proposed action subject to a public comment period less than the usual 30 days. Accordingly, we conclude that the licensee has not delayed its application to take advantage of the Exigency License Amendment provisions of 10 CFR 50.91 and has used its best efforts to provide a reasonable opportunity (at least 15 days) for public noticing and comment.

IV. FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has determined that the amendments involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the amendments does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The evaluations discussed above show that all of the accidents comprising the licensing bases which could potentially be affected by the fuel reload were reviewed for the Unit 1/Cycle 4 design. These evaluations conclude that the reload design does not cause the previously acceptable safety limits, as specified in the Standard Review Plan, to be exceeded; therefore, the amendments do not involve a significant reduction in a margin of safety or a significant increase in the consequences of an accident previously evaluated. From these evaluations, the Commission also finds that the core reload with its associated changes in operating limitations (i.e., MTC and AFD limits) and modified peripheral fuel assemblies has no effect on any accident causal mechanisms. Therefore, the amendments do not involve a significant increase in the probability of an accident previously evaluated. The modifications to the peripheral fuel assemblies, as noted above in our evaluation, have little or no effect on the performance or accident evaluation of the core and they do not create the possibility of a new or different kind of accident from any accident previously evaluated. Similarly, the changes other than those for peripheral fuel assemblies do not involve changes in hardware nor the introduction of new or novel changes in procedures. Therefore, the amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes of an administrative nature which delete an obsolete sentence from TS 5.3.1 and redesignate an existing TS Figure for Unit 2 only, have no safety implication and do not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Therefore, based on the evaluations given above, the Commission has determined that the amendments involve no significant hazards consideration.

V. ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve 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 exposure. The NRC staff has made a determination that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet 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 these amendments.

VI. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (51 FR 23484) on June 27, 1986 and (51 FR 28463) on August 7, 1986. We have also determined that this action involves no significant hazards consideration and that exigency circumstances exist which justify taking this action on an expedited basis. We have consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments.

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.

References

(1) Letter to H.R. Denton (NRC) from H.B. Tucker (Duke Power), "McGuire Nuclear Station Docket Nos. 50-369 and 50-370 McGuire 1/Cycle 4 OFA Reload," May 15, 1986.

(2) Letter to H.B. Tucker (Duke Power) from E.G. Adensam (NRC), "Issuance of Amendment 32 to Facility Operating License NPF-9 and Amendment No. 13 to Facility Operating License NPF-17, MCGuire Nuclear Station, Units 1 and 2," April 20, 1984.

(3) "Reference Core Report - 17 X 17 Optimized Fuel Assembly," WCAP-9500-A, 1981.

(4) Letter to H.R. Denton (NRC) from H.B. Tucker (Duke Power), "McGuire Nuclear Station, Docket Nos. 50-369 and 50-370," November 14, 1983.

(5) Davidson, S.L. et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

(6) "Reload Safety Evaluation - McGuire Unit Cycle 3 Revision 1," May 1985.

(7) Letter to H.B. Tucker (Duke Power) from E.G. Adensam (NRC), "Issuance of Amendment No. 43 to Facility Operating License NPF-9 and Amendment No. 24 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2," May 15, 1985.

(8) Miller, R.W., et al., "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," WCAP-10217-A, June 1983.

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Dated: August 22, 1986

DATED: August 22, 1986

AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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