Docket Nos.: 50-369
and 50-370

June 22, 1987

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. 73 to Facility Operating License NPF-9 and Amendment No. 54 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 73 to Facility Operating License NPF-9 and Amendment No. 54 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 April 9, 1987, and supplemented May 18 and June 15, 1987.

The amendments change the Technical Specifications to reflect the third of several refueling stages involved in the transition to the use of optimized fuel assemblies in Unit 2. This includes for both units an increase (from 2.26 to 2.32) in the limit for the heat flux hot channel factor and accompanying axial flux difference limits. The amendments also include changes regarding the submittal of peaking factor limit reports and appropriate changes to the Technical Specification index pages. The amendments are effective as of their date of issuance.

A copy of the related safety evaluation supporting Amendment No. 73 to Facility Operating License NPF-9 and Amendment No. 54 to Facility Operating License NPF-17 is enclosed.

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. 73 to NPF-9
2. Amendment No. 54 to NPF-17
3. Safety Evaluation

cc w/enclosures: See next page

* See previous concurrence

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
DHood/mac	*MDuncan	BJYoungblood
06/17/87	06/ /87	06/ /87

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PDR ADOCK 05000369
P PDR

Mr. H. B. Tucker
Duke Power Company

McGuire Nuclear Station

cc:
Mr. A.V. Carr, Esq.
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. Robert Gill
Duke Power Company
Nuclear Production Department
P. O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
Route 4, Box 529
Hunterville, North Carolina 28078

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

L. L. Williams
Area Manager, Mid-South Area
ESSD Projects
Westinghouse Electric Corporation
MNC West Tower - Bay 239
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dr. John M. Barry
Department of Environmental Health
Mecklenburg County
1200 Blythe Boulevard
Charlotte, North Carolina 28203

Chairman, North Carolina Utilities
Commission
Dobbs Building
430 North Salisbury Street
Raleigh, North Carolina 27602

Mr. Dayne H. Brown, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

DATED: June 22, 1987

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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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. 73
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 April 9, 1987, and supplemented May 18 and June 15, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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. 73 , 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/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Technical Specification
  Changes

Date of Issuance: June 22, 1987

* See previous concurrence

PD#II-3/DRP-II/I	PD#II-3/DRP-I/II	OGC/BETH	PD#II-3/DRP-I/II
DHood:mac	*MDuncan		BJYoungblood
06/17/87	06/ /87	06/16/87	06/22/87

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. 54
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 April 9, 1987, and supplemented May 18 and June 15, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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. 54 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/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Technical Specification
Changes

Date of Issuance: June 22, 1987

* See previous concurrence

PD#II-3/DRP-II/I	PD#II-3/DRP-I/II	OGC/BETH	PD#II-3/DRP-I/II
DHood:mac	*MDuncan		BJYoungblood
06/17/87	06/ /87	06/18/87	06/22/87

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 54

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

V
XXII
3/4 2-3
3/4 2-4
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6-21

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

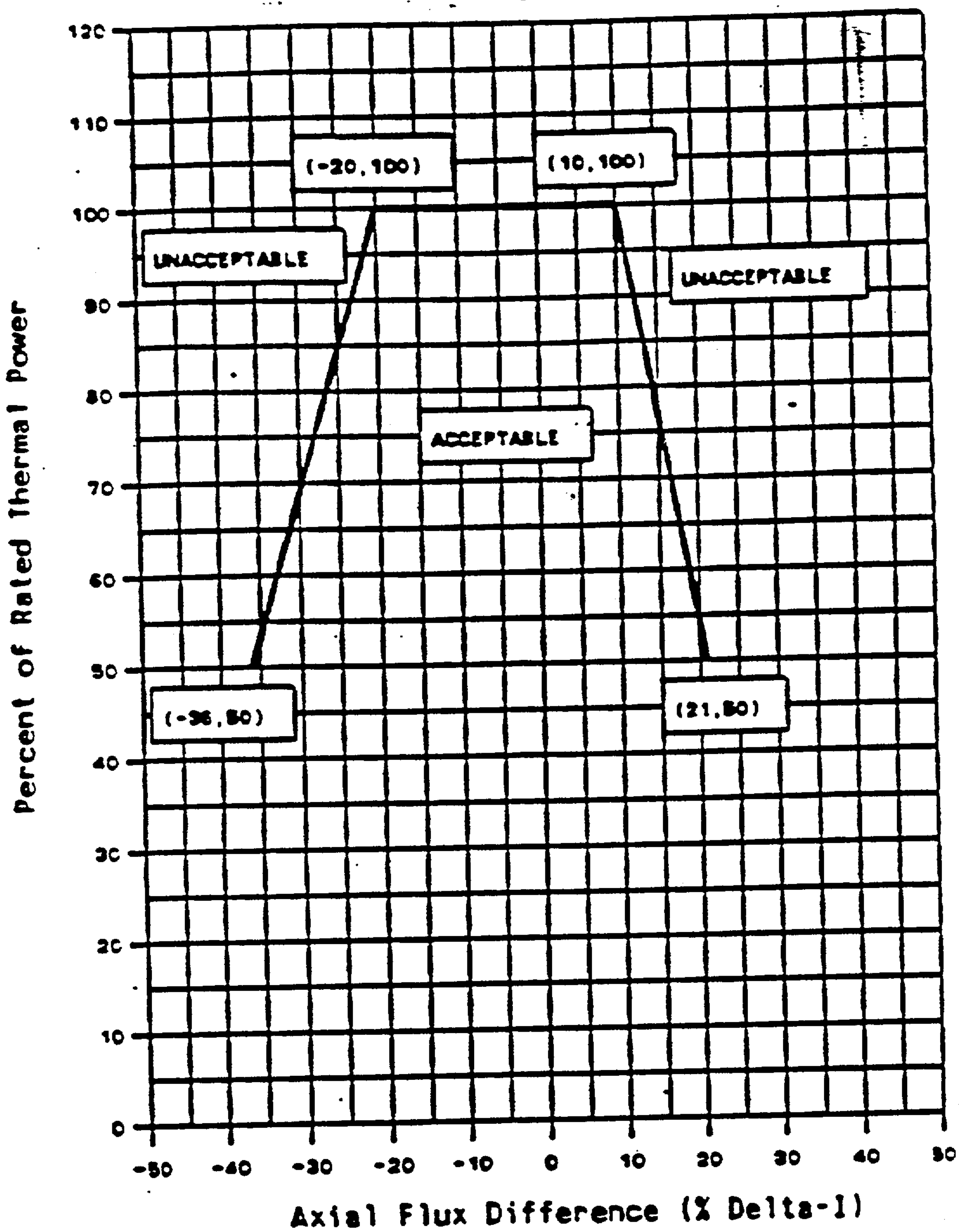


Figure 3.2-1

AFD Limits as a Function of Rated Thermal Power

This page deleted.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationship:

$$F_Q(Z) \leq \frac{[2.32]}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{[2.32]}{0.5} [K(Z)] \text{ for } P \leq 0.5$$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.32 \quad x\ K(z)}{P\ x\ W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.32 \quad x\ K(z)}{W(z)\ x\ 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.32 is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

---

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.32}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.32}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of 3.2-1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e., and 4.2.2.2f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within ±5% of target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{2.32 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.32. K(z) is given in Figure 3.2-2. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{2.32 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.32.

K(Z) is given in Figure 3.2-2. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and
2. At least once per 31 effective full power days.

e. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or
2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilbrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with the following expression:

$$[(\text{max. over z of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.32}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plan regions:

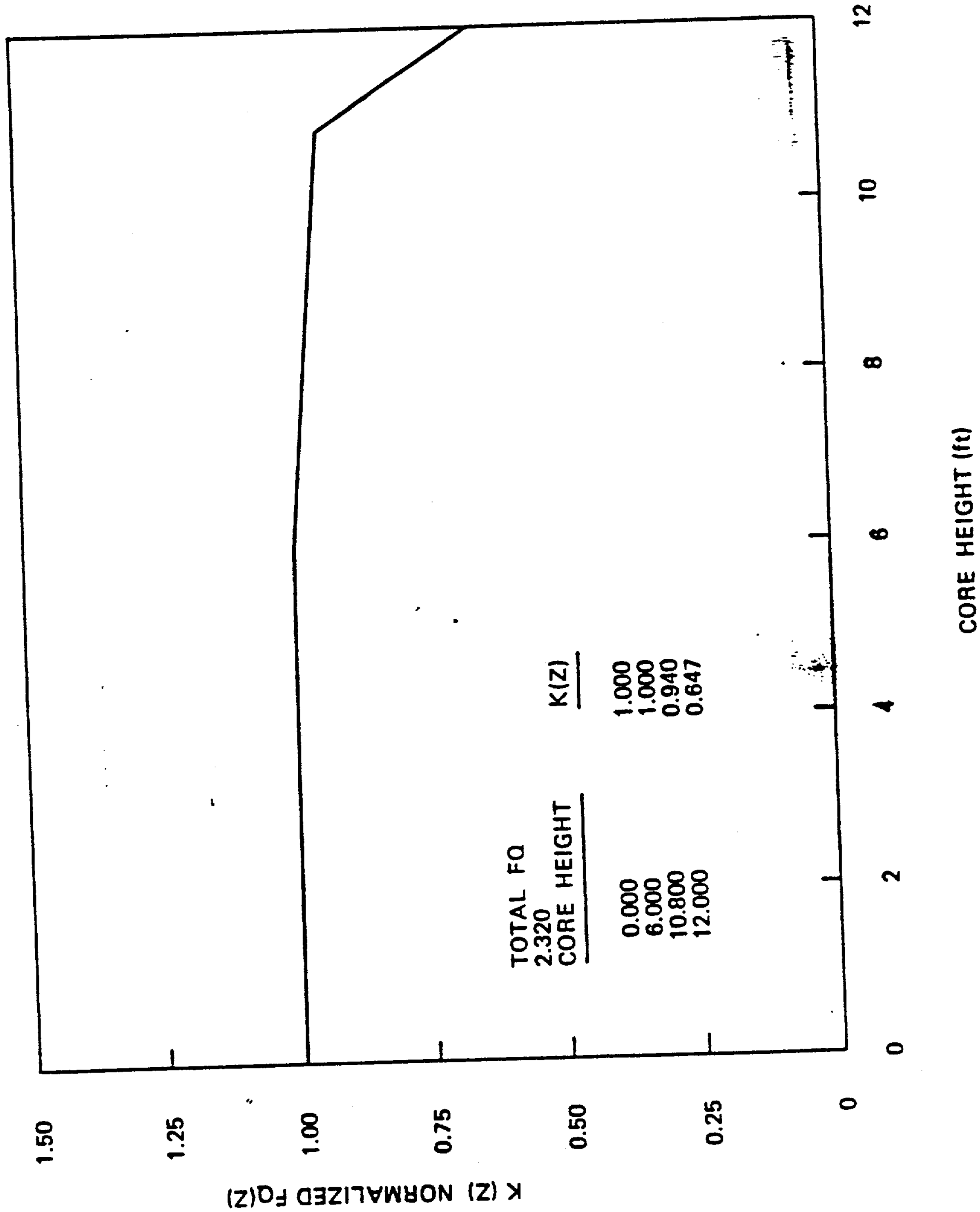
1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

FIGURE 3.2-2

K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT



BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

6.9.1.9 PEAKING FACTOR LIMIT REPORT

The W(z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(z) functions for RAOC and Base Load Operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC*. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(z) functions for ROAC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC document control desk with copies to the regional administrator and the resident inspector within 30 days of their implementation.

Any information needed to support W(z), $W(z)_{BL}$ and APL^{ND} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

*WCAP-10216 "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification".

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 54 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 letter dated April 9, 1987 and supplements dated May 18 and June 15, 1987, (Ref. 1), Duke Power Company (the licensee) proposed amendments to change the Technical Specifications (TSs) to reflect the third refueling of McGuire Unit 2 and its fourth fuel cycle. (The refueling for Unit 2 Cycle 4 would continue the transition to use of optimized fuel assemblies (OFAs) initiated during the first refueling and would replace an additional 64 standard fuel assemblies with OFAs; thus, 181 of the total 193 fuel assemblies in Cycle 4 would be OFAs.) The existing TS figures for axial flux difference limits as a function of rated thermal power would be relabeled such that the existing figure for Unit 1 only (Figure 3.2-1a) would apply to both Units 1 and 2, and the existing figure for Unit 2 only (Figure 3.2-1b) would be deleted. The TS Index would be updated consistent with these changes.

The proposed amendments would also increase the limit specified for heat flux hot channel factor (F_Q) for both Unit 1 and Unit 2 from the present value of 2.26 to 2.32. This change would be reflected in each of several TSs presently specifying 2.26, including TSs 3.2.2, 4.2.2.2c, 4.2.2.2f, 4.2.2.3, 4.2.2.4c, 4.2.2.4f.2, and Bases 3/4.2.1. A corresponding change would be made to TS Figure 3.2-2 which shows normalized F_Q as a function of core height (i.e., the revised normalized figure would be based upon a total F_Q of 2.32 rather than 2.26.)

The title of TS 6.9.1.9, "Radial Peaking Factor Limit Report" would be changed to "Peaking Factor Limit Report." This change would also be reflected in the

TS Index. The schedule in TS 6.9.1.9 for providing the peaking factor limit report to the NRC would be changed from 60 days before cycle initial criticality (or 60 days before W(Z) functions and the value for APL^{ND} would become effective) to 30 days after implementation. The change would also update the NRC addressee specified in TS 6.9.1.9 for receipt of the peaking factor limit report (i.e., the NRC's Core Performance Branch would be changed to the NRC Document Control Desk, with copies also specified to be provided to the Regional Administrator and the Resident Inspector) based upon changes in the Commission's regulations (51 FR 40303). TS 6.9.1.9 would also be supplemented to specify that the methodology used to generate the W(Z) functions for Relaxed Axial Offset Control (RAOC) and base load operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC (i.e., from WCAP-10216 "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specifications".) If changes to these methods are deemed necessary, the revised TS 6.9.1.9 would specify that such changes are to be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendments of previously submitted documentation.

The licensee's letter of April 9, 1987 provided a description and Reload Safety Evaluation (RSE) for the McGuire Unit 2-Cycle 4 core reload and the associated Peaking Factor Limit Report (PFLR). By letter dated May 18, 1987 the licensee corrected certain references cited in the April 9, 1987 submittal to clarify that appropriate methodology had been used for the RSE. By Letter dated June 15, 1987, the licensee noted that two fuel assemblies had received damage and would not be reused during Unit 2 Cycle 4 operation. Consequently, the core loading pattern was revised to exclude these two assemblies and a revised RSE and PFLR was provided to the NRC. The revised pattern did not change the results of the licensee's safety evaluation, the conclusions of the April 9 RSE, or the proposed revisions to the TSs.

EVALUATION

1. Large Break LOCA Analysis and Increased F_Q

The licensee's RSE included LOCA analyses to justify the increase in F_Q from

2.26 to 2.32 because the limiting event which determines the allowable value of F_Q is the large-break LOCA.

During a large break LOCA, depressurization of the Reactor Coolant System (RCS) would result in a reactor trip and safety injection signals. Although the injection of borated water from the ECCS would complement void formation to shutdown the fission process, the presence of boron is not accounted for in this aspect of the LOCA analysis. Similarly, no credit is taken for control rod insertion, leaving void formation as the credible mechanism to terminate the fission process in the early phase of the transient. Injection of the borated water provides for heat transfer from the core and prevents excessive clad temperature. Once the RCS depressurizes to about 600 psia, the accumulators would begin injecting borated water. The analysis assumes loss of offsite power; hence, reactor coolant pumps are assumed to trip and to coast down. After the depressurization (blowdown) phase of the transient, refill of the reactor vessel begins with emergency core cooling water which was not assumed to be operational until this time. The refill phase is completed when the water level reaches the bottom of the fuel rods. The next phase, reflood, occurs as ECCS water covers the core and terminates the core temperature rise. Continued operation of the ECCS pumps supplies water for long term cooling. The boric acid concentration in the primary water is sufficient to prevent criticality.

The analysis was performed with an NRC approved code, the 1981 and 1983 versions of the Westinghouse LOCA-ECCS evaluation model, BART (Refs. 3 and 4). This code included the evaluation model revisions approved by the NRC (Refs. 2, 5 and 6) regarding (a) a modelling change in WREFLOOD which was found to increase the peak cladding temperature by about 20°F and (b) a systematic input error in the BART code which caused low values of hot assembly bundle power to be used, and which was found to increase peak cladding temperature by about 100°F.

The blowdown, refill and reflood stages of the transient were analyzed with the methodologies described in Reference 7. Reference 7 also describes the interfaces among the various computer codes and the features of the codes that ensure compliance with 10 CFR 50, Appendix K. The various computer codes involved in the analysis are:

- SATAN-VI: - Analyzes the thermal hydraulic transient in the reactor coolant system during blowdown. It includes RCS pressure, enthalpy, density and mass and energy flow (Ref. 8).

- LOTIC: - Calculates the containment pressure transient during the three phases of the LOCA analysis (Ref. 9).

- WREFLOOD: - Determines the core flooding rate, the coolant pressure and temperature and the quench front height during the refill and reflood phases of the LOCA (Ref. 10). (See also BASH.)

- LOCTA-IV: - Computes the thermal transient of the hottest fuel rod during the three phases (Ref. 11).

- BASH: - Provides an analysis of the steam/water flow phenomena during core reflood (Ref. 13).

The computational model and codes used for this analysis have been approved by the NRC and comply with the requirements of Appendix K to 10 CFR 50.

The initial conditions and the numerical values of the input parameters for the analysis were conservatively determined by the licensee. Because of the substantial design similarity of the Unit 1 and Unit 2 Cycle 4 cores and RCS, the analysis applies to both Unit 1 and Unit 2. The double ended, cold leg, guillotine break was shown to be the limiting case (Ref. 12). Some of the main parameters and initial conditions for this limiting case include:

Core power - 102% of 3411 MWt
Peak linear power - 102% of 12.88 KW/ft
Heat flux hot-channel factor (F_Q) - 2.32
Accumulator water volume (nominal) - 950 ft^3, each
Moody discharge coefficients (C_D) - 0.4, 0.6, 0.8
Steam generator tube plugging - 5.0% each

The criteria to be satisfied in a large break LOCA analysis are described in 10 CFR 50.46. The criteria are: (1) peak cladding temperature shall not exceed 2,200°F; (2) localized maximum cladding oxidation must not exceed 17% during or after quenching; (3) cladding chemical interaction with water and steam (maximum hydrogen generation) must not exceed 1.0% of all the metal in the cladding cylinders surrounding the fuel; (4) calculated changes in core geometry shall be such that the core remains amenable to cooling; and (5) after the successful initial operation of the ECCS the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

The results of the licensee's analysis showed that all the applicable criteria are satisfied. Specifically,

1. Peak cladding temperature was 1,841°F for the worst case ($C_D = 0.6$). This is less than 2200°F and is, therefore, acceptable.

2. Local maximum cladding oxidation during or after quenching was 2.71%. This is less than 17% and is, therefore, acceptable.

3. Cladding chemical interaction of all the metal in the cladding cylinders surrounding the fuel was less than 0.30%. This does not exceed 1.0% and is, therefore, acceptable.

4. The cladding temperature transient was terminated at a time when the core geometry was still amenable to cooling.

5. After the successful initial operation of the ECCS, the calculated core temperature remains at an acceptably low level and the decay heat is removed for an extended period of time.

On the basis of the above results for the large break LOCA performed with an F_Q of 2.32, we find the proposed increase in F_Q for Units 1 and 2 to be acceptable because the revised analyses, performed with suitable input parameters and based upon methodology which satisfies the criteria of Appendix K to 10 CFR 50, provide results which meet the requirements of 10 CFR 50.46.

2. Unit 2/Cycle 4 Reload Safety Evaluation

On April 20, 1984, the Commission issued Amendment No. 32 to Facility Operating License NPF-9 to change the Technical Specifications to permit changes in operating limits related to the transition to the use of optimized fuel assemblies (OFA) in McGuire Unit 1. Commission approval of the transition from the standard fuel assembly to the OFA loading was based upon the licensee's safety analyses (Ref. 14 and 15) which examined the mechanical, nuclear, thermal hydraulic and accident evaluation aspects and justified the compatibility of the OFA design with the standard design in the transition loadings as well as the full OFA core. A similar amendment for Unit 2 (Amendment 23) was issued March 22, 1985.

Accordingly, beginning with their first refuelings for Cycle 2, Unit 1 and Unit 2 operated with the first stage of a transition core consisting of approximately 1/3 Westinghouse 17x17 OFAs and 2/3 Westinghouse 17x17 low-parasitic fuel assemblies (STDs). During Cycle 3, each unit contained about 2/3 OFAs and 1/3 STDs. Unit 1 is currently operating in Cycle 4 and Unit 2 will achieve Cycle 4 by its present refueling. In Cycle 4, 181 of the total 193 fuel assemblies of the Unit 2 core will be OFAs.

The major differences between STDs and OFAs are the use of Zircaloy grids for the OFAs versus Inconel grids for STDs and reduction in fuel rod diameter. The OFA fuel has similar design features compared to the STD fuel, which has had substantial operating experience in a number of nuclear plants. Major advantages for utilizing the OFAs are: (1) Increased efficiency of the core by reducing the amount of parasitic material and (2) Reduced fuel cycle costs due to an optimization of water to uranium ratio.

The proposed amendments provide for plant operation consistent with the design and safety evaluation conclusions in the licensee's McGuire Unit 2 Cycle 4 Reload Safety Evaluation (RSE). The changes to the Technical Specifications reflect adjustments in the limiting conditions and surveillance requirements for (1) axial flux difference and (2) heat flux hot channel factor, consistent with the parameters used in the RSE. A summary of the cycle specific aspects of the nuclear, thermal hydraulic, and mechanical analysis, and the NRR findings, follow:

The Cycle 4 reload has been designed to meet ECCS limits based upon an F_Q of 2.32 and Relaxed Axial Offset Control (RAOC) has been employed to the extent permitted by the new F_Q value (Ref. 1 and 16). The Cycle 4 nuclear characteristics are within the range of, and are bounded by, the Cycle 3 nuclear characteristics (Ref. 1, Attachment 2A). The significant core parameter (e.g., enrichment, fuel density, fuel burnup, moderator temperature coefficient, Doppler temperature coefficient, minimum delayed neutron fraction, maximum bank differential worth, control rod (and worst stuck rod) worths and shutdown margin) are within the design limits which we have approved for Cycle 3 and are, therefore, acceptable.

The thermal-hydraulic methodology, the DNBR correlation, and the Cycle 4 DNB limits are consistent with the current and accepted licensing basis. Operating power distributions were evaluated relative to the assumed limiting operating power distribution used in the accident analyses. Limits on allowable operating axial flux difference as a function of power level, were found to be less restrictive than those resulting from the LOCA F_Q considerations previously discussed. No changes in the DNB core limits are required. No variation in the thermal margin will result from the Cycle 4 reload. Hence, the Unit 2 Cycle 4 thermal-hydraulic design is acceptable.

The mechanical design of Unit 2 Cycle 4 is within the limits of Cycle 3 design and the compatibility of the OFA core has been justified in the OFA loading submittal (Ref. 14). The fuel has been designed and will be operated so that clad flattening will not occur (Ref. 1 and 17). In that portion of the core pattern designated as "region 6," a rod plenum spring smaller than that of previous fuel regions is used. This new spring design (Ref. 18) satisfies a change in the non-operational 6g loading design criterion to 4g axial and 6g lateral loading with dimensional stability. This reduced spring force reduces the potential for pellet chipping in the fuel rod. We find that the mechanical fuel design for Unit 2 Cycle 4 is within the limits of previously accepted designs and is, therefore, acceptable.

Accordingly, we conclude that the Unit 2 Cycle 4 design does not cause the previously acceptable safety limits to be exceeded and is, therefore, acceptable.

3. Administrative Changes

By previous Amendments 32 (Unit 1)/13(Unit 2) and Amendments 42 (Unit 1)/23 (Unit 2), McGuire was changed to a type of F_Q function for which the title "Radial Peaking Factor Limit" was no longer appropriate. The previous amendments failed to correct the title of TS 6.9.1.9. The present amendments correct the title by deleting "Radial." Also, during its licensing review of another nuclear plant (Vogtle Electric Generating Station), the Commission determined that the safety of a plant would not be affected if the peaking factor limit report required by TS 6.9.1.9 were submitted 30 days after implementation rather than 60 days before criticality, provided the methodology used was previously reviewed and approved by the NRC and changes to this methodology are subject to the requirements of 10 CFR 50.59. The change in the McGuire schedule implemented by these amendments includes these conditions in the revised TS 6.9.1.9. The amendments also update the NRC addressee for receipt of the report consistent with 51 FR 40303.

The TS Index is updated consistent with appropriate changes implemented by these amendments.

The above changes are purely administrative and have no adverse impact upon safety. They are, therefore, acceptable.

REFERENCES

1. Letter from H. B. Tucker, Duke Power Company to NRC, dated April 9, 1987, with supplements dated May 18, 1987 and June 15, 1987.

2. Letter from H. R. Denton to PWR Applicants and Licensees, "Westinghouse ECCS Evaluation Models" (Generic Letter 86-16) dated October 22, 1986.

3. Letter from J. R. Miller, NRC, to E. P. Rahe, Westinghouse, "Acceptance for Referencing of the 1981 Version of the Westinghouse Large Break ECCS Evaluation Model," December 1, 1981.

4. Letter from C. O. Thomas, NRC, to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9561, BART A-1: A Computer Code for Best Estimate Analyses of Reflood Transients," dated December 21, 1983.

5. Letter from Charles E. Rossi, NRC to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9561, Addendum 3, Revision 1," dated August 25, 1986.

6. Young, M. Y., "Addendum to BART-A1: A Computer Code for the Best Estimate Analysis of Reload Transients" (Special Report: Thimble modeling in Westinghouse ECCS Evaluation Model): WCAP-9561-P, Addendum 3, Revision 1, dated July 1986.

7. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, dated July 1974.

8. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, dated June 1974.

9. Hsieh, T., et al., "Long Term Ice Condenser Containment LOTIC Code Supplement 1" WCAP-8355 Supplement 1 May 1975, WCAP-8354P, July 1974.

10. Kelly, R. D., et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD)," WCAP-8171, (WCAP-8170P) dated June 1974.

11. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, dated June 1974.

12. Salvatori, R., "Westinghouse ECCS Sensitivity Studies," (WCAP-8340P) WCAP-8356, dated July 1974.

13. Letter form W. J. Johnson, Westinghouse to J. Lyons, NRC, "BASH Methodology Enhancements," WCAP-10266, Addendum 2, dated March 1987, No. NS-NRC-87-3212.

14. Duke Power Company submittal to NRC, "Safety Evaluation for McGuire Units 1 and 2, Transition to Westinghouse 17x17 Optimized Fuel Assemblies" dated December 1983.

15. Letter from H. B. Tucker, Duke Power Company to H. R. Denton, NRR, "RTD Bypass Manifold Removal," dated October 29, 1985.

16. Miller, R. W., et al., "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specifications" WCAP-10217-A, dated June 1983.

17. Georgie, R. A., et al., "Revised Clad Flattening Model" WCAP-8381, dated July 1974.

18. Letter from E. P. Rahe, Jr., Westinghouse to L. E. Phillips, NRC, "Fuel Handling Load Criteria," dated April 12, 1984 (NS-EPR-2893).

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 and changes in surveillance requirements. 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.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 18977) on May 20, 1987 and consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments. Licensee submittals since publication of 52 FR 18977, dated May 18 and June 15, 1987, correct certain references in the initial submittal, reconfirm that the initial safety limits are met for minor changes in the fuel loading pattern, and do not alter the proposed changes as identified in 52 FR 18977 or alter the staff's proposed no significant hazards consideration determination.

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: L. Lois, RSXB
D. Hood, PD#II-3

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