ATTACHMENT 1

PROPOSED McGUIRE UNIT 1 AND 2 TECHNICAL SPECIFICATION CHANGES

8605230319 69pp.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limits shown in Figure 3.1-0, and
b. Less negative than -4.1×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specifications 3.1.1.3a. - MODES 1 and 2* only.#
Specification 3.1.1.3b. - MODES 1, 2, and 3 only.#

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limits shown in Figure 3.1-0 within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.3.

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SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

REPLACE WITH THE FOLLOWING FIGURE (X)

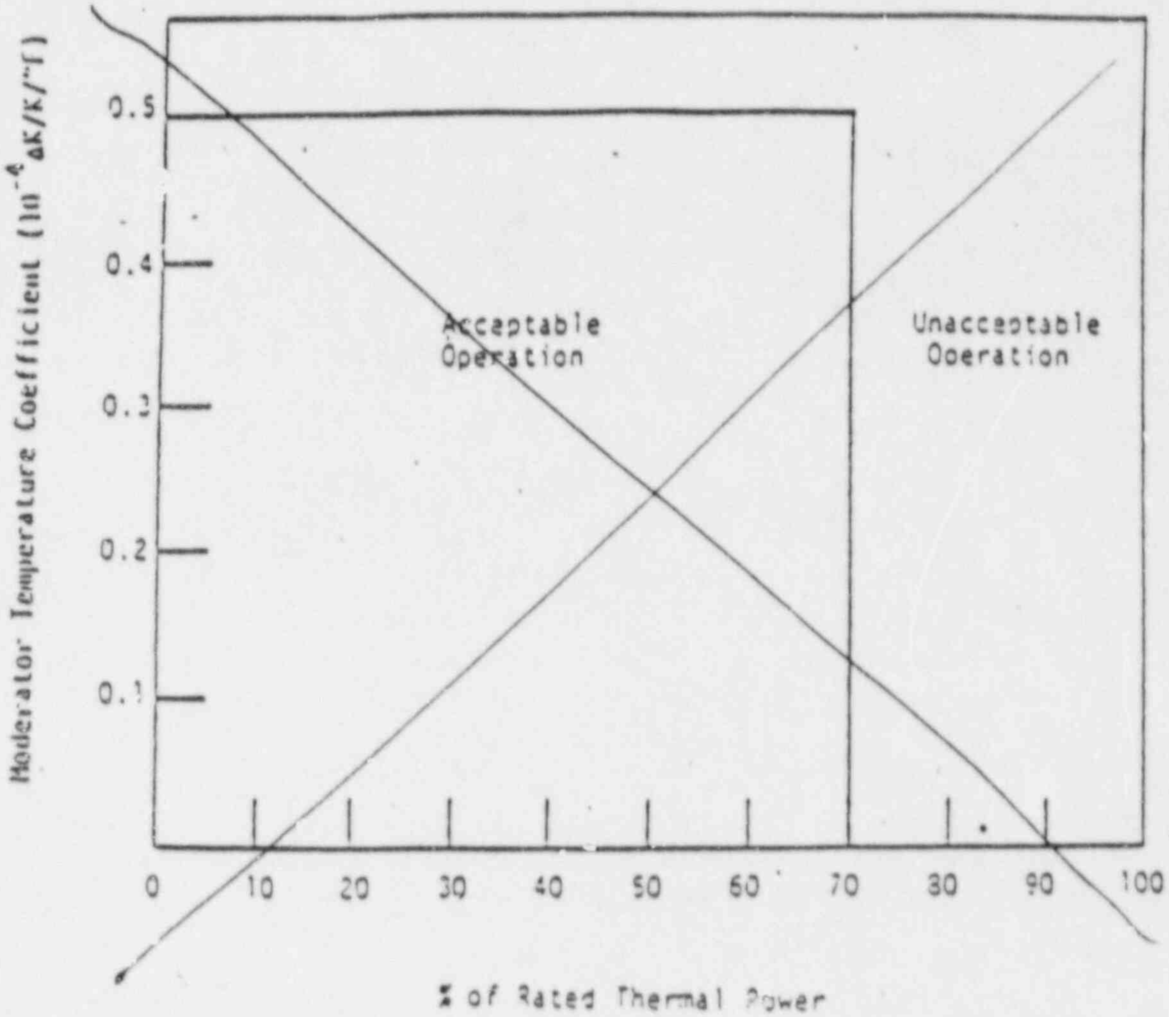
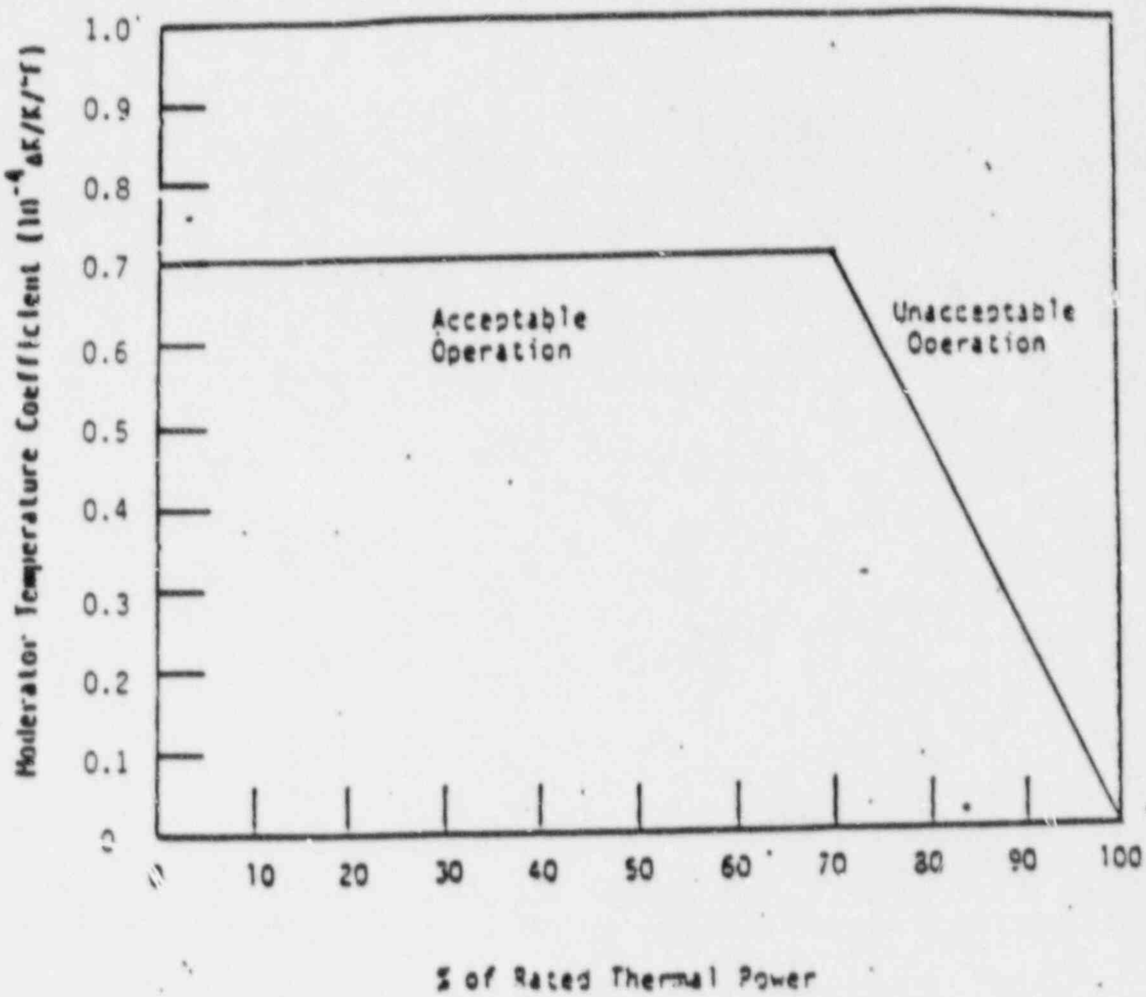


FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

FIGURE (X)



~~FIGURE 3.1-0~~

~~MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL~~

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BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} delta k/k/°F. The MTC value of -3.2×10^{-4} delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} k/k/°F.

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16,321 gallons of 7000-ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000-ppm borated water from the refueling water storage tank (RWST).

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed operational space defined by Figure 3.2-1 for RAOC operation, or

b. within a ±5 percent target band about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. For RAOC operation with the indicated AFD outside of the Figure 3.2-1 limits,

   1. Either restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

   1. Either restore the indicated AFD to within the target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

*See Special Test Exception 3.10.2.

**APL^{ND} is the minimum allowable power level for base load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

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SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitoring Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 3/4.2.2 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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FIGURE Y

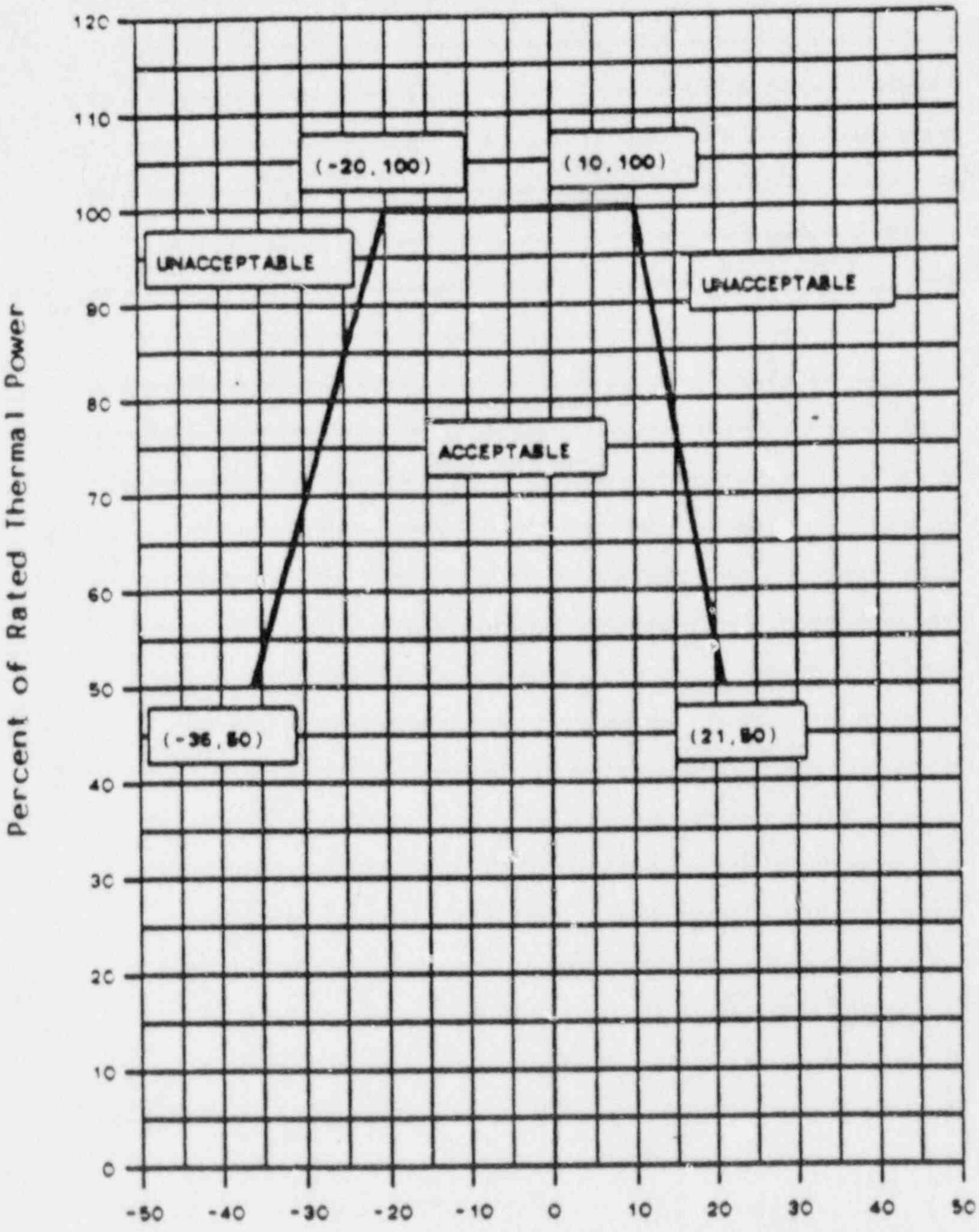

FIGURE 3.2-1a
AFD Limits as a Function of Rated Thermal Power (UNIT 1)
~~McGuire Unit 1 Cycle 4~~

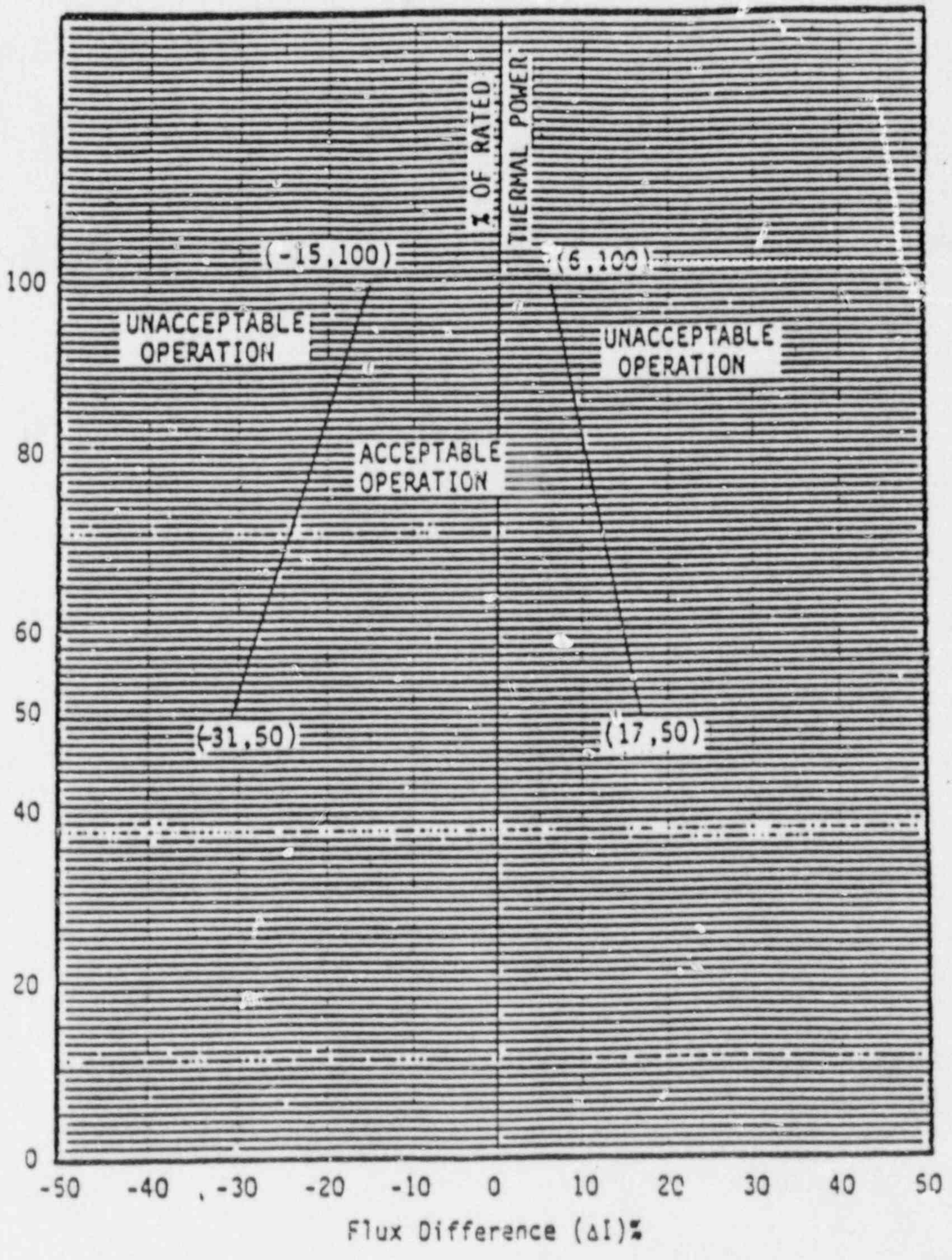

FIGURE 3.2-1b
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER (Unit 2)

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3/4.2 POWER DISTRIBUTION LIMITS

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 2.26 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.

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AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are defined by Figures 3.2-1, i.e. that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limit of which are defined by Figure 3.2-1, and 2) Base Load operation, which is defined as the maintenance of the AFD within a ± 5% band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts the indicated AFD to relatively small target band and power swings (AFD target band of ±5%, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

ATTACHMENT 2
JUSTIFICATION AND SAFETY ANALYSIS

Mr. H.B. Tucker's (DPC) November 14, 1983 letter to Mr. H.R. Denton (NRC/ONRR) described planned changes in the fuel design for McGuire Nuclear Station, Units 1 and 2. McGuire Unit 1 had been operating with a Westinghouse 17x17 low-parasitic (STD) fueled core. It was planned to refuel Unit 1 with Westinghouse 17x17 Reconstitutable Optimized Fuel Assembly (OFA) regions. As a result, future core loadings would range from an approximately 1/3 OFA - 2/3 STD transition core to eventually an all OFA fueled core. Major advantages for utilizing the OFA 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 the water to uranium ratio. This letter provided a Reference Safety Evaluation Report summarizing the evaluation/analysis performed on the region-by-region reload transition from the McGuire Units 1 and 2 STD fueled cores to cores with all optimized fuel. The 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 metho[illegible] described in WCAP-9272 and 9273, "Westinghouse Reload Safety Evaluation Methodology," and the transition effects described for mixed cores in Chapter 18 of WCAP-9500-A, "Reference Core Report - 17x17 Optimized Fuel Assembly." Consistent with the Westinghouse STD reload methodology for analyzing cycle specific reloads, parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate subsequent determination of the applicability of 10 CFR 50.59. Subsequent cycle [illegible]c reload safety evaluations were to verify that applicable safety [illegible] are satisfied based on the reference evaluation/analyses established [illegible] reference report. A summary of the mechanical, nuclear, thermal and hydraulic, and accident evaluations for the McGuire Units 1 and 2 transitions to an all OFA core were given in the reference report.

The results of evaluation/analysis and tests described in the Reference Safety Evaluation Report led to the following conclusions:

a. The Westinghouse OFA reload fuel assemblies for McGuire 1 and 2 are mechanically compatible with the STD design, control rods, and reactor internals interfaces. Both fuel assemblies satisfy the design bases for the McGuire units.

b. Changes in the nuclear characteristics due to the transition from STD to OFA fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

c. The reload OFAs are hydraulically compatible with the STD design.

d. The accident analyses for the OFA transition core were shown to provide acceptable results by meeting the applicable criteria, such as, minimum DNBR, peak pressure, and peak clad temperature, as required. The previously reviewed and licensed safety limits were met. Analyses in support of this safety evaluation establish a reference design on which subsequent reload safety evaluations involving OFA reloads can be based. (Attachment 2A of H.B. Tucker's December 12, 1983 Unit 1/Cycle 2 OFA reload submittal presented those detailed non-LOCA and LOCA accident analyses of the McGuire Units 1 and 2 FSAR impacted by the changes as determined in Section 6.0 of the Reference Safety Evaluation Report).

e. Plant operating limitations given in the Technical Specifications affected by use of the OFA design and positive MTC would be satisfied with the changes noted in Section 7.0 of the report.

McGuire Unit 1 is currently operating in Cycle 3 (including a power coastdown not part of the original cycle design which was evaluated and determined permissable under the provisions of 10CFR 50.59) with Westinghouse 17x17 low parasitic (STD) fuel assemblies and optimized fuel assemblies (OFA) following previous NRC approval of two OFA reload regions (reference Ms. E.G. Adensam's (NRC/ONRR) April 20, 1984 and May 15, 1985 letters to H.B. Tucker), with the third such OFA region scheduled for the upcoming Cycle 4 refueling. Subsequent McGuire Unit 1 cycles are also planned to be refueled with Westinghouse 17x17 OFA's. (McGuire Unit 2 is currently in its cycle 3 refueling outage preparing for operation with its second OFA reload region. This reload (McGuire Unit 2/Cycle 3) is being accomplished under the provisions of 10 CFR 50.59 as indicated in a Tucker to Denton letter of February 21, 1986, with the first OFA reload having received NRC approval via Adensam to Tucker letter dated March 22, 1985).

Attachment 2A is the cycle-specific Reload Safety Evaluation (RSE) for McGuire Unit 1/Cycle 4. The RSE presents an evaluation for McGuire Unit 1, Cycle 4, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology". As indicated above, the NRC has previously approved similar OFA reloads for McGuire Unit 1 (and 2). The November 14, 1983 OFA transition reference safety evaluation licensing submittal (approved by the NRC) justifying the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core contained mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 4 safety evaluation.

All of the accidents comprising the licensing bases which could potentially be affected by the fuel reload have been reviewed for the Cycle 4 design. The results of new analyses and the justification for the applicability of previous results for the remaining analyses are addressed in the cycle specific reload safety evaluation.

As in Cycles 2 and 3, this cycle will contain one region 4 demonstration assembly of an intermediate flow mixer grid fuel assembly design. This demonstration assembly has been previously discussed in Mr. H.B. Tucker's February 20, 1984 letter to Mr. H.R. Denton, in which it was concluded that the demonstration program could be implemented per the requirements of 10CFR 50.59. This assembly will be loaded into the core in a manner which satisfies the requirements of the "Safety Evaluation for the intermediate flow mixer grid (IFM) demonstration fuel assembly in McGuire Unit 1" (Davidson, S.L. (Ed.), February 1984). During the Cycle 2/3 refueling a problem was encountered during routine inspection of removable fuel rods in this demonstration assembly, and one removable rod was not reinserted because of mechanical interference (operation of Unit 1/Cycle 3 with one fuel rod of this assembly removed was evaluated under the provisions of 10 CFR 50.59 and determined permissible). The safety impact for a rod removed with a water hole remaining is addressed in the Cycle 4 reload safety evaluation.

From the evaluation presented in the Cycle 4 Reload Safety Evaluation, it concluded that the Cycle 4 design does not cause the previously acceptable safety limits to be exceeded. This conclusion is based on the following:

1. Cycle 3 burnup is between 11000 and 12127 MWD/MTU.

2. Cycle 4 burnup is limited to 13100 MWD/MTU including a coastdown.

3. There is adherence to all plant operating limitations given in the Technical Specifications as revised by the proposed changes given in Appendix A of the Cycle 4 RSE.

To ensure plant operation consistent with the design and safety evaluation conclusion statements made in the Cycle 4 RSE and to ensure that these conclusions remain valid, Technical Specification changes will be needed for Cycle 4 to incorporate RAOC and a positive moderator temperature coefficient. These changes (presented in Appendix A of the cycle-specific RSE) are discussed in the cycle-specific RSE, along with any necessary justifications. The McGuire Unit 1 Cycle 4 reload design has been performed assuming an increase in the low power beginning of cycle moderator temperature coefficient (MTC) to +7 pcm/degrees F (the MTC technical specification limit is raised to +7 pcm/degrees F up to 70% rated thermal power and ramped to 0 pcm/degrees F at 100% RTP). The unit could startup and operate with the currently approved Technical Specification limit (a "step" of +5 to 0 pcm/degrees F at 70% power) but the probability of entering the action statement requiring interim rod position limits and potential delays in the cycle startup process and a special report to the Commission would be increased. Attachment 2B is the safety evaluation report presenting the evaluations and analyses performed verifying the acceptability of operation of McGuire Unit 1 (and 2) with this increased positive MTC limit of +7 pcm/degrees F, including a description of the proposed change's impact on the FSAR Chapter 15 transients (this report is identified as reference no. 10 in the McGuire Unit 1/Cycle 4 RSE). The McGuire FSAR will be revised accordingly in the appropriate annual FSAR update following approval of this change. The revised Unit 1 RAOC envelope is based upon the F_Q limit of 2.26. The McGuire Unit 1 F_Q limit was increased from 2.15 to 2.26 in submittals associated with the Unit 1/Cycle 3 reload (and for Unit 2 with the Unit 2/Cycle 2 reload). However, the RAOC AFD envelope continued to be based upon the 2.15 F_Q limit until this cycle's RAOC analysis, although the envelope could have been expanded along with the previous limit increase. The revision to the Unit 1 RAOC AFD envelope is simply taking credit for the previously approved higher F_Q limit and brings the nuclear design and LOCA analysis assumption into agreement (Note: Although credit for the 2.26 F_Q limit currently in effect on Unit 2 has not been taken in Unit 2 RAOC analyses performed to date (including the recent Unit 2/Cycle 3 analysis as reflected in the Tucker to Denton letter of February 21, 1986), Duke anticipates similarly expanding the Unit 2 RAOC envelope in the near future to take credit for the higher limit).

Attachment 1 provides copies of the McGuire Units 1 and 2 Technical Specifications with the appropriate Unit 1/Cycle 4 changes indicated. Note that although the changes given in Appendix A of the Unit 1/Cycle 4 RSE are intended to apply only to Unit 1, Appendix A does not always reflect the fact that a change would also apply to Unit 2 unless specifically indicated otherwise since the McGuire Unit 1 and 2 Technical Specifications are combined into one document - this has been accounted for in Attachment 1. Consequently, the McGuire Unit 2 specifications are administratively affected in that a specification currently applying to both McGuire Units 1 and 2 is split into two portions addressing the separate requirements for Units 1 and 2 created by these changes. In addition, although intended to apply only for Unit 1 via being in Appendix A, note that the increased positive MTC limit of +7 pcm/degrees F is also being requested for McGuire Unit 2 because of its desirable effects on fuel cycle flexibility (this is justified by Attachment 2B for Unit 2 as well as Unit 1). Although the increase in the positive MTC limit was assumed in the McGuire 1/Cycle 4 RSE, the change is not a consequence of the reload and therefore is acceptable for immediate implementation on Unit 2. Note that while this increase was not assumed in the McGuire Unit 2/Cycle 3 RSE, it will be needed (assumed) for future reloads and therefore should be approved on Unit 2 sometime prior to McGuire 2/Cycle 4. The McGuire FSAR update will accordingly reflect this change for Unit 2 as well as unit 2. No revisions to the Technical Specification Bases are required by the changes.

The Peaking Factor Limit Report for McGuire Unit 1/Cycle 4 which will be submitted in accordance with Technical Specification 6.9.1.9 provides the elevation dependent W(z) values that are to be used as inputs to define the appropriate fitting coefficients for W(z) interpolations to be performed as a function of cycle burnup and axial elevation for RAOC and Base Load Operation during Cycle 4, and the value for APL^{ND}. The appropriate W(z) function is used to confirm that the Heat Flux Hot Channel Factor, $F_Q(z)$, will be limited to the values specified in the Technical Specifications. The peaking factor report to be submitted for Unit 1 Cycle 4 is based upon the proposed revision to the RAOC AFD envelope. If the revised RAOC limits are not approved in time to support the Unit 1 Cycle 4 startup, confirmation of the validity of the W(z) functions with respect to the existing RAOC AFD envelope or generation of new W(z) functions would be required. An exemption to the 60 days prior to criticality submittal schedule for the peaking factor report may be necessary if revised W(z) functions are required. The use of overly conservative W(z) functions increases the probability of entering the action statement and could lead to very restrictive AFD limits and/or reductions in reactor power.

ATTACHMENT 2A

RELOAD SAFETY EVALUATION

MCGUIRE NUCLEAR STATION

UNIT 1 CYCLE 4

April, 1986

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4578L:6-860415

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for McGuire Unit 1, Cycle 4, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"(1).

McGuire Unit 1 is operating in Cycle 3 with Westinghouse 17x17 low parasitic (STD) and optimized fuel assemblies (OFA). For Cycle 4 and subsequent cycles, it is planned to refuel the McGuire Unit 1 core with Westinghouse 17x17 optimized fuel assemblies. In the OFA transition licensing submittal(2) to the NRC, approval was requested for the transition from the STD fuel design to the OFA design and the associated proposed changes to the McGuire Units 1 and 2 Technical Specifications. The licensing submittal, which has received NRC approval, justifies the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core. The OFA transition licensing submittal(2) contains mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 4 safety evaluation.

All of the accidents comprising the licensing bases(2,3) which could potentially be affected by the fuel reload have been reviewed for the Cycle 4 design described herein. The results of new analyses and the justification for the applicability of previous results for the remaining analyses are addressed in safety evaluations for a Positive Moderator Coefficient(10) and the UHI Elimination(14) licensing submittals.

During the cycle 2/3 refueling a problem was encountered in assembly ZV-1. One removable rod was not reinserted because of mechanical interference. This assembly will remain in the core for Cycle 4. The safety impact for a rod removed with a water hole remaining is presented in Reference 4.

1.2 GENERAL DESCRIPTION

The McGuire Unit 1, Cycle 4 reactor core will be comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 3/4 refueling, 64 STD fuel assemblies will be replaced with 64 Region 6 optimized fuel assemblies. A summary of the Cycle 4 fuel inventory is given in Table 1.

As in Cycles 2 and 3, this cycle will contain one Region 4 demonstration assembly, designated in Figure 1 as 4A, of an intermediate flow mixer grid fuel assembly design. This assembly will be loaded into the core in a manner which satisfies the requirements given in Reference 13.

Nominal core design parameters utilized for Cycle 4 are as follows:

Core Power (MWt)	3411
System Pressure (psia)	2250
Core Inlet Temperature (°F)	558.9
Thermal Design Flow (gpm)	382,000
Average Linear Power Density (kw/ft)	5.43
(based on 144" active fuel length)	

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 4 design does not cause the previously acceptable safety limits to be exceeded. This conclusion is based on the following:

1. Cycle 3 burnup is between 11000 and 12127 MWD/MTU.

2. Cycle 4 burnup is limited to 13100 MWD/MTU including a coastdown.

3. There is adherence to plant operating limitations in the Technical Specifications.

4. The proposed Technical Specification changes discussed in Section 4.0 of this report and provided in Appendix A are approved.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The Region 6 fuel assemblies are Westinghouse OFAs. The mechanical description and justification of their compatibility with the Westinghouse STD design in a transition core is presented in the OFA transition licensing submittal.(2)

Table 1 presents a comparison of pertinent design parameters of the various fuel regions. The Region 6 fuel has been designed according to the fuel performance model(5). The fuel is designed and operated so that clad flattening will not occur, as predicted by the Westinghouse clad flattening model(6). For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in Reference 7, is satisfied.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is described in WCAP-8183, "Operational Experience with Westinghouse Cores."(8) Operating experience for Zircaloy grids has also been obtained from six demonstration 17x17 OFAs(2), four demonstration 14x14 OFAs(2) and two regions of OFA fuel in the McGuire Unit 1 Cycle 2 and 3 designs.

2.2 NUCLEAR DESIGN

The Cycle 4 core loading is designed to meet a $F_Q(z) \times P$ ECCS limit of $\leq 2.26 \times K(z)$. In the event of UHI elimination, the $F_Q(z) \times P$ ECCS limit of $\leq 2.26^* \times K(z)$ will remain applicable to the Cycle 4 design.

Relaxed Axial Offset Control (RAOC) will be employed in Cycle 4 to enhance operational flexibility during non-steady state operation. RAOC makes use of available margin by expanding the allowable ΔI band, particularly at reduced

*Based on the LOCA analyses performed in support of the UHI elimination licensing submittal(14).

power. The RAOC methodology and application is fully described in Reference 9. The analysis for Cycle 4 indicates that no change to the safety parameters is required for RAOC operation. During operation at or near steady state equilibrium conditions, core peaking factors are significantly reduced due to the limited amount of xenon skewing possible under these operating conditions. The Cycle 4 Technical Specifications recognize this reduction in core peaking factors through the use of a Base Load Technical Specification.

Adherence to the F_Q limit is obtained by using the F_Q Surveillance Technical Specification, also described in Reference 9. This provides a more convenient form of assuring plant operation below the F_Q limit while retaining the intent of using a measured parameter to verify operation below Technical Specification limits. F_Q surveillance is only a surveillance requirement and as such has no impact on the results of the Cycle 4 analysis or safety parameters.

Table 2 provides a summary of Cycle 4 kinetics characteristics compared with the current limits based on previously submitted accident analyses.

Table 3 provides the control rod worths and requirements at the most limiting condition during the cycle (end-of-life) for the standard burnable absorber design. The required shutdown margin is based on previously submitted accident analysis. The available shutdown margin exceeds the minimum required.

The loading pattern contains 320 burnable absorber (BA) rods located in 44 BA rod assemblies. Location of the BA rods are shown in Figure 1.

A more Positive Moderator Coefficient as compared to the current value will be utilized during Cycle 4. The safety evaluation is contained in Reference 10 and the associated Technical Specification changes are addressed in Section 4.0 of this report.

2.3 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[2]. The thermal hydraulic safety analyses used for Cycle 4 are based on a reduced design flow rate[15] in comparison to Reference 2. No significant variations in thermal margins will result from the Cycle 4 reload.

The thermal-hydraulic methods used to analyze axial power distributions generated by the RAOC methodology are similar to those used in the Constant Axial Offset Control (CAOC) methodology. Normal operation power distributions are evaluated relative to the assumed limiting normal operation power distribution used in the accident analysis. Limits on allowable operating axial 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 were evaluated relative to the axial power distribution assum[illegible] used to generate DNB core limits and resultant Overtemperature Delta-T setpoints (including the f(ΔI) function). No changes in the DNB core limits are required for RAOC operation.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability has been evaluated considering the consequences of those incidents examined in the FSAR[3] using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at the design power level (Section 1.0) during Cycle 4. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 4 core. The time dependent densification model[11] was used for fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining $F_Q(z)$ at or below 2.26 x K(z).

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[3] were examined. In all cases, it was found that the effects were accommodated within the conservatism of the initial assumptions used in 1) the previous applicable safety analysis, 2) the safety evaluation performed in support of the positive moderator coefficient (+7 pcm/°F) licensing submittal[10] or 3) the safety evaluation performed in support of the UHI Elimination licensing submittal[14].

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 4 parameters in each of these three areas were examined as discussed in the following subsections to ascertain whether new accident analyses were required.

3.2.1 KINETIC PARAMETERS

Table 2 is a summary of the kinetic parameters current limits along with the associated Cycle 4 calculated values. All of the kinetic values fall within the bounds of the current limits except for the minimum moderator temperature coefficient. The safety evaluation for the Positive Moderator Coefficient is contained in Reference 10 and the associated Technical Specification changes are addressed in Section 4.0 of this report.

3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths, and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 4 meets the current limit. Table 3 shows that the Cycle 4 shutdown margin requirements have been satisfied. Table 4 is a summary of the current limit control rod ejection analysis parameters and the corresponding Cycle 4 values.

3.2.3 CORE PEAKING FACTORS

Peaking factors for the dropped RCCA incidents were evaluated based on the NRC approved dropped rod methodology described in Reference 12. Results show that DNB design basis is met for all dropped rod events initiated from full power.

The peaking factors for steamline break and control rod ejection have been evaluated and are within the bounds of the current limits.

4.0 TECHNICAL SPECIFICATION CHANGES

To ensure that plant operation is consistent with the design and safety evaluation conclusion statements made in this report and to ensure that these conclusions remain valid, Technical Specifications changes will be needed for Cycle 4 to incorporate RAOC and the Positive Moderator Temperature Coefficient. These changes are presented in Appendix A.

5.0 REFERENCES

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

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

3. "McGuire Final Safety Analysis Report."

4. "Reload Safety Evaluation McGuire - Unit 1 Cycle 3 - Revision 1," May 1985.

5. Miller, J.V., (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations", WCAP-8785, October 1976.

6. George, R.A., (et. al.), "Revised Clad Flattening Model", WCAP-8381, July 1974.

7. Risher, D. H., (et. al.), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977.

8. Skaritka, J., Iorii, J.A., "Operational Experience with Westinghouse Cores", WCAP-8183, Revision 14, July, 1985.

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

10. Westinghouse Transmittal to Duke Power Company, "Safety Evaluation for Operation of McGuire Units 1 and 2 with a Positive Moderator Coefficient", January 1986.

11. Hellman, J.M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation", WCAP-8219-A, March 1975.

12. Morita, T., Osborne, M. P., et. al., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10297-P-A (Proprietary) and WCAP-10298-A (Non Proprietary), June 1983.

13. Davidson, S. L., (Ed.), "Safety Evaluation for the Intermediate Flow Mixer Grid (IFM) Demonstration Fuel Assembly in McGuire Unit 1", February 1984.

14. Duke Power Company Transmittal to NRC, "McGuire Nuclear Station Safety Analyses for UHI Elimination", March 1986.

15. Duke Power Company Transmittal to NRC," McGuire 2 Cycle 2 OFA Reload", November 1984.

TABLE 1

MCGUIRE UNIT 1 - CYCLE 4

FUEL ASSEMBLY DESIGN PARAMETERS

Region	1	4*	5*	6A*	6B*
Enrichment (w/o U-235)+	2.108	3.205	3.204	3.20	3.40
Density(% Theoretical)+	94.53	95.04	95.05	95.0	95.0
Number of Assemblies	9	60	60	12	52
Approximate Burnup at++ Beginning of Cycle 4 (MWD/MTU)	16942#	20027	14604	0	0
Approximate Burnup at++ End of Cycle 4 (MWD/MTU)	28540#	33731	25240	16745	14617

* Optimized Fuel - Zirc grid

+ All fuel region values are as-built except Region 6 values which are nominal.

++Based on EOC3 = 11560 MWD/MTU, EOC4 = 13100 MWD/MTU (coastdown included)

#The burnups noted are for the Region 1 fuel assemblies being used and are not an average for the whole region.

TABLE 2

MCGUIRE UNIT 1 - CYCLE 4
KINETICS CHARACTERISTICS

	Current Limits	Cycle 4 Design
Minimum Moderator Temperature Coefficient (pcm/°F)*	+5 < 70% of RTP 0 ≥ 70% of RTP	+7 <70% of RTP +7 ramp to 0 from 70% to 100% of RTP
Doppler Temperature Coefficient (pcm/°F)*	-2.9 to -0.91	-2.9 to -0.91
Least Negative Doppler-Only Power Coefficient, Zero to Full Power, (pcm/% power)*	-9.55 to -6.05	-9.55 to -6.05
Most Negative Doppler Only Power Coefficient, Zero to Full Power (pcm/% power)*	-19.4 to -12.6	-19.4 to -12.6
Minimum Delayed Neutron Fraction β_{eff}, (%)	.44	>.44
Minimum Delayed Neutron Fraction β_{eff}, (%) [Ejected Rod at BOL]	.50	>.50
Maximum Differential Rod Worth of Two Banks Moving Together (pcm/in)*	100	<100

*pcm = 10^{-5} $\Delta\rho$

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TABLE 3

END-OF-CYCLE SHUTDOWN REQUIREMENTS AND MARGINS
MCGUIRE UNIT 1 - CYCLE 4

Control Rod Worth (%$\Delta\rho$)	Cycle 3	Cycle 4
All Rods Inserted	6.72	6.95
All Rods Inserted Less Worst Stuck Rod	5.90	5.95
(1) Less 10%	5.32	5.35
Control Rod Requirements (%$\Delta\rho$)		
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	3.18	3.39
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	3.68	3.89
Shutdown Margin [(1) - (2)] (%$\Delta\rho$)	1.64	1.46
Required Shutdown Margin (%$\Delta\rho$)	1.30	1.30

TABLE 4

MCGUIRE UNIT 1 - CYCLE 4
CONTROL ROD EJECTION ACCIDENT PARAMETERS

HZP-BOC	Current Limit*	Cycle 4
Maximum ejected rod worth, %$\Delta\rho$	0.75	<0.75
Maximum F_Q (ejected)	11.0	<11.0
HFP-BOC		
Maximum ejected rod worth, %$\Delta\rho$	0.23	<0.23
Maximum F_Q (ejected)	4.5	<4.5
HZP-EOC		
Maximum ejected rod worth, %$\Delta\rho$	0.90	<0.90
Maximum F_Q (ejected)	20.0	<20.0
HFP-EOC		
Maximum ejected rod worth, %$\Delta\rho$	0.23	<0.23
Maximum F_Q (ejected)	5.9	<5.9

*Based on the safety evaluation performed in support of the Positive Moderator Coefficient licensing submittal[10].

FIGURE 1
CORE LOADING PATTERN
McGUIRE UNIT 1, CYCLE 4

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					5	6B	5	6B	5	6B	5				
2			5	5	6B	4	6B 4	4	6B 4	4	6B	5	5		
3		5	5	6B 4	4	6B 8	4	5 SS	4	6B 8	4	6B 4	5	5	
4		5	6B 4	4	5	1	6A 8	5	6A 8	1	5	4	6B 4	5	
5	5	6B	4	5	4	6B 12	4	4	4	6B 12	4	5	4	6B	5
6	6B	4	6B 8	1	6B 12	4	5	4	5	4	6B 12	1	6B 8	4	6B
7	5	6B 4	4	6A 8	4	5	4	6A 8	4	5	4	6A 8	4	6B 4	5
8	6B	4	5	5	4A *	4	6A 8	1	6A 8	4	4	5	5	4	6B
9	5	6B 4	4	6A 8	4	5	4	6A 8	4	5	4	6A 8	4	6B 4	5
10	6B	4	6B 8	1	6B 12	4	5	4	5	4	6B 12	1	6B 8	4	6B
11	5	6B	4	5	4	6B 12	4	4	4	6B 12	4	5	4	6B	5
12		5	6B 4	4	5	1	6A 8	5	6A 8	1	5	4	6B 4	5	
13		5	5	6B 4	4	6B 8	4	5 SS	4	6B 8	4	6B 4	5	5	
14			5	5	6B	4	6B 4	4	6B 4	4	6B	5	5		
15					5	6B	5	6B	5	6B	5				

90° (left) 270 (right)

0°

X Y	REGION NUMBER BA's
SS	SECONDARY SOURCE

* Demonstration Assembly with Rod Removed

APPENDIX A

TECHNICAL SPECIFICATION

PAGE CHANGES

Modifications to Pages:

3/4 1-5a
3/4 2-4

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limits shown in Figure 3.1-0, and
b. Less negative than -4.1×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specifications 3.1.1.3a. - MODES 1 and 2* only.#
Specification 3.1.1.3b. - MODES 1, 2, and 3 only.#

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limits shown in Figure 3.1-0 within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

   3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0.
#See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

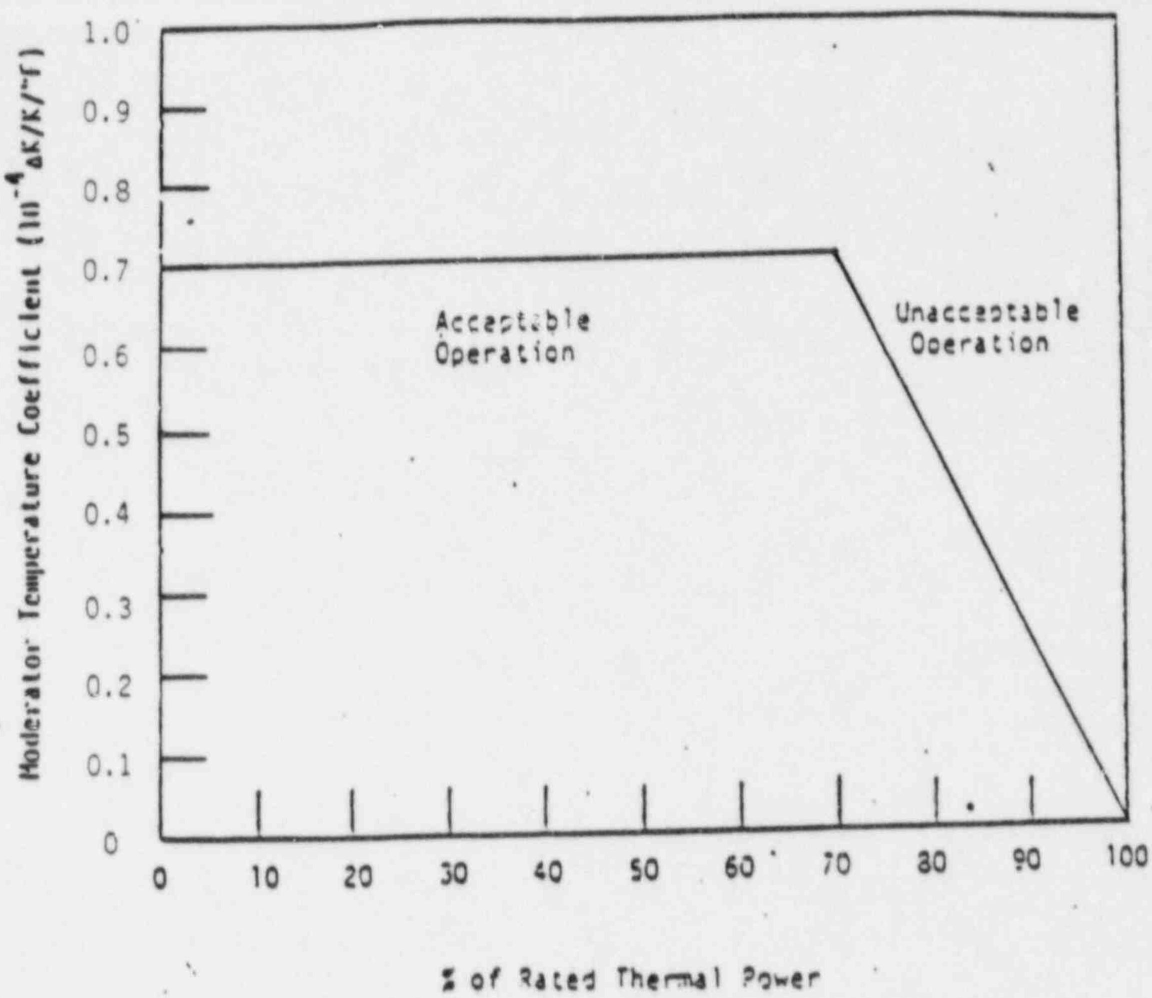

FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

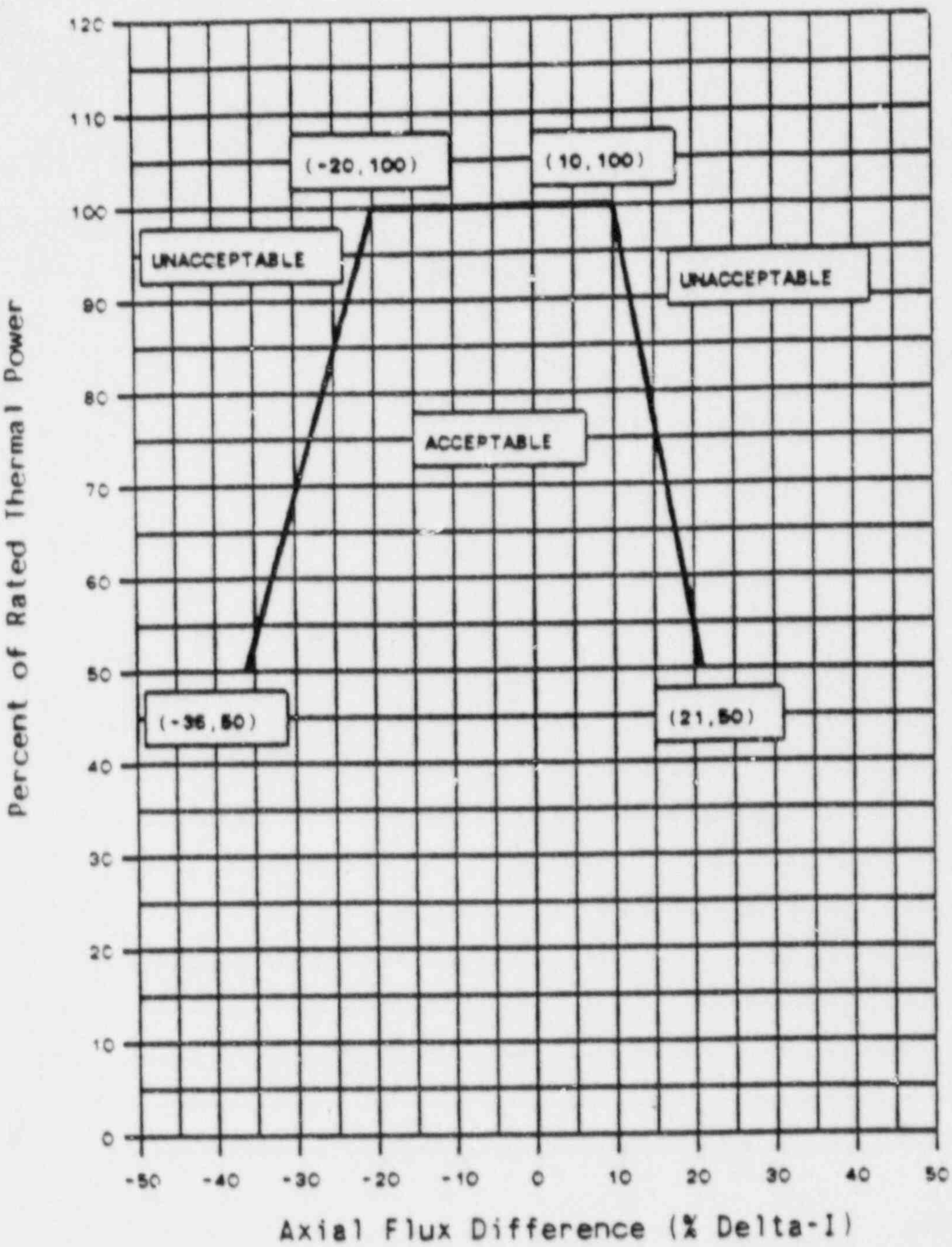

FIGURE 3.2-1
AFD Limits as a Function of Rated Thermal Power
McGuire Unit 1 Cycle 4

ATTACHMENT 2B

SAFETY EVALUATION FOR OPERATION OF

McGUIRE UNITS 1 & 2

WITH A POSITIVE MODERATOR COEFFICIENT

SAFETY EVALUATION FOR OPERATION OF

McGUIRE UNITS 1 & 2

WITH A POSITIVE MODERATOR COEFFICIENT

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CONTENTS

SECTION 1
INTRODUCTION

Safety analyses and evaluations have been performed to support the proposed Technical Specification change for McGuire Units 1 & 2 which would allow a more positive moderator temperature coefficient to exist during power operation. The results of the analysis, which are presented in the following section, show that the proposed change can be accommodated with margin to applicable FSAR safety limits.

The present McGuire Technical Specifications require the moderator temperature coefficient (MTC) to be +5 pcm/°F* or less at all times while the reactor is critical. A positive coefficient at reduced power levels results in a significant increase in fuel cycle flexibility, but has only a minor effect on the safety analysis of the accident events presented in the FSAR.

The proposed Technical Specification change would allow a +7 pcm/°F MTC below 70 percent of rated power, ramping down to 0 pcm/°F at 100 percent power. This MTC is diagrammed in Figure 1. A power-level dependent MTC was chosen to minimize the effect of the specification on postulated accidents at high power levels. Moreover, as the power level is raised, the average core water temperature becomes higher as allowed by the programmed average temperature for the plant, tending to bring the moderator coefficient more negative. Also, the boron concentration can be reduced as xenon builds into the core. Thus, there is less need to allow a positive coefficient as full power is approached. As fuel burnup is achieved, boron is further reduced and the moderator coefficient will become negative over the entire operating power range.

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* 1 pcm = 10^{-5} $\Delta k/k$

SECTION 2

ACCIDENT EVALUATIONS

I. Introduction

The impact of a positive moderator temperature coefficient on the accident analyses presented in Chapter 15 of the McGuire Units 1 & 2 Final Safety Analysis Report (FSAR) has been assessed. Those incidents which were found to be sensitive to positive or minimum moderator coefficients were reanalyzed. In general, these incidents are limited to transients which cause reactor coolant temperature to increase. The analyses presented herein were based on a +7 pcm/°F moderator temperature coefficient, which was assumed to remain constant for variations in temperature.

The control rod ejection and rod withdrawal from subcritical analyses were based on a coefficient which was at least +7 pcm/°F at zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code, on which the analyses are based, is a diffusion-theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature. For all accidents which were reanalyzed, the assumption of a positive moderator temperature coefficient existing at full power is conservative since as diagrammed in Figure 1, the proposed Technical Specification requires that the coefficient be linearly ramped to zero above 70 percent power.

In general, reanalysis was based on the identical analysis methods, computer codes, and assumptions employed in the FSAR; any exceptions are noted in the discussion of each incident. Accidents not reanalyzed included those resulting in excessive heat removal from the reactor coolant system for which a large negative moderator coefficient is conservative. Table 1 gives a list of accidents presented in the McGuire Units 1 & 2 FSAR, and denotes those events reanalyzed for a positive coefficient. The following sections provide discussions for each of the FSAR events.

II. Transients Not Affected By a Positive Moderator Coefficient

The following transients were not reanalyzed since they either result in a reduction in reactor coolant system temperature, and are therefore sensitive to a negative moderator temperature coefficient, or are otherwise not affected by a positive moderator temperature coefficient.

A. RCCA Misoperation

Only the RCCA drop case presented in Section 15.4.3 of the FSAR is potentially affected by a positive moderator temperature coefficient. Use of a positive coefficient in the analysis would result in a larger reduction in core power level following the RCCA drop, thereby increasing the probability of a reactor trip. For the return to power automatic rod control case, a positive coefficient would result in a small increase in the power overshoot. Since the limiting conditions for this accident are at or near 100 percent power where the moderator temperature coefficient must be close to zero or negative, this accident is unaffected by the proposed Technical Specification and thus the analysis was not repeated.

B. Startup of an Inactive Reactor Coolant Loop

An inadvertent startup of an idle reactor coolant loop at an incorrect temperature results in a decrease in core average temperature. As the most negative values of moderator reactivity coefficient produce the greatest reactivity addition, the most limiting case is represented by the analysis reported in the FSAR, Section 15.4.4.

C. Excessive Heat Removal Due to Feedwater System Malfunctions

The addition of excessive feedwater or the reduction of feedwater temperature are excessive heat removal incidents, and are consequently most sensitive to a negative moderator temperature coefficient. Results presented in Section 15.1.1 and 15.1.2 of the FSAR, based on a negative coefficient, represent the limiting case. Therefore, this incident was not reanalyzed.

D. Excessive Load Increase

An excessive load increase event, in which the steam load exceeds the core power, results in a decrease in reactor coolant system temperature. With the reactor in manual control, the analysis presented in Section 15.1.3 of the FSAR shows that the limiting case is with a large negative moderator coefficient. If the reactor is in automatic control, the control rods are withdrawn to increase power and restore the average temperature to the programmed value. The analysis of this case in the FSAR show that the minimum DNBR is not sensitive to moderator temperature coefficient. Therefore, the results presented in the FSAR are still applicable to this incident.

E. Spurious Actuation of Safety Injection

Analysis of a spurious actuation of safety injection at power is presented in Section 15.5.1 of the FSAR. This transient results in a decrease in average coolant temperature and core power and the results are not sensitive to moderator temperature coefficient. Therefore, this incident was not reanalyzed with a positive moderator coefficient.

F. Main Steam Line Depressurization/Rupture of a Main Steam Pipe

Since the steam line der surization or rupture of a main steam pipe is a tempera . reduction transient, minimum core shutdown margin is ar . ated with a strong negative moderator temperature coef ne worst conditions for a steamline break are therefore the analyzed in the FSAR (Section 15.1.4 and 15.1.5).

G. Loss of Coolant Accident (LOCA)

The loss of coolant accident (Section 15.6.5 of the FSAR) is analyzed to determine the core heatup consequences caused by a rupture of the reactor coolant system boundary. The event results in a depressurization of the RCS and a reactor shutdown at the beginning of the transient. This accident was not reanalyzed since the Technical Specification requirement that the temperature coefficient be zero or negative at 100 percent power ensures that the previous analysis basis for this event is not affected.

III. Transients Sensitive to a Positive Moderator Coefficient

A. Boron Dilution

As stated in Section 15.4.6 of the FSAR, a boron dilution incident cannot occur during refueling due to administrative controls which isolate the RCS from potential sources of diluted water. If a boron dilution incident occurs during cold shutdown, hot standby, or startup, the operator must take action to terminate the dilution before the reactor returns critical. Therefore, since a return to criticality is prevented by the operator, the value of the moderator coefficient has no effect during a boron dilution incident in these operating modes. The reactivity addition due to a boron dilution at power will cause an increase in power and reactor coolant system temperature. Due to the temperature increase, a positive moderator coefficient would add additional reactivity and increase the severity of the transient. With the reactor in automatic control, however, the rod insertion alarms provide the operator with adequate time to terminate the dilution before shutdown margin is lost. A boron dilution incident with the reactor in manual control is no more severe than a rod withdrawal at power, which is discussed in Section III.C, and therefore this case was not specifically analyzed. Following reactor trip, the amount of time available before shutdown margin is lost is not affected by the moderator coefficient.

B. Control Rod Bank Withdrawal From a Subcritical Condition

Introduction

A control rod assembly bank withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 15.4.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator and fuel. The reactivity addition due to a positive moderator coefficient results in increases in peak heat flux and peak fuel and clad temperatures.

Method of Analysis

The analysis was performed in the FSAR for a reactivity insertion rate of 75×10^{-5} Δk/sec. This reactivity insertion rate was used in this analysis and is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). The analysis used a moderator temperature coefficient more conservative than a +7 pcm/°F for all appropriate temperature values.

Results and Conclusions

Reanalysis of this event assuming a 75 pcm/sec insertion rate, coupled with a positive moderator temperature coefficient of +7 pcm/°F, yields a peak heat flux which does not exceed the nominal full power value. In the event of a RCCA bank withdrawal accident from a subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limit value and thus, no fuel or clad damage is predicted. Therefore, the conclusions presented in the FSAR remain valid.

C. Uncontrolled Control Rod Bank Withdrawal at Power

Introduction

An uncontrolled control rod bank withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature. A positive moderator coefficient would increase the power mismatch resulting in a faster heatup of the reactor coolant. However, this effect is offset by the fact that the faster heatup and reactivity addition result in an earlier reactor trip from either overtemperature delta-T or high neutron flux. A discussion of this incident is presented in Section 15.4.2 of the FSAR.

Method of Analysis

The transient was reanalyzed employing the same digital computer code and assumptions regarding instrumentation and setpoint errors used for the FSAR. This transient was analyzed at 100, 60 and 10 percent power with a positive moderator coefficient. A constant moderator coefficient of +7 pcm/°F was used in the analysis.

Results and Conclusions

For each initial power level the full range of reactivity insertion rates was reanalyzed. The limiting case for DNB margin remains above the limit DNBR value. These results demonstrate that the conclusions presented in the FSAR are still valid. That is, the core and reactor coolant system are not adversely affected since nuclear flux and overtemperature ΔT trips prevent the core minimum DNB ratio from falling below the limit value for this incident.

D. Loss of Reactor Coolant Flow

Introduction

The loss of flow events presented in FSAR Sections 15.3.1 and 15.3.2 were reanalyzed to determine the effect of a +7 pcm/°F moderator temperature coefficient on the nuclear power transient and the resultant minimum DNBR reached during the incident. The effect on the nuclear power transient would be limited to the initial stages of the incident during which reactor coolant temperature increases; this increase is terminated shortly after reactor trip.

Method of Analysis

Analysis methods and assumptions used in the reevaluation were consistent with those employed in the FSAR. The digital computer codes used to calculate the flow coastdown and resulting system transient were the same as those used to perform the analysis described in the FSAR. The analysis assumed a constant moderator coefficient of +7 pcm/°F.

Results and Conclusions

Results of the analysis show that the minimum DNBR remains above the limit value for these transients.
Therefore, the conclusions of the FSAR analyses remain valid.

E. Locked Rotor

Introduction

The case presented in the FSAR (Section 15.3.3) for this transient was reanalyzed. Following a locked rotor incident, reactor coolant system temperature rises until shortly after reactor trip. A positive moderator coefficient will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the incident. The transient was reanalyzed, however, due to the effect on the nuclear power transient and thus on the peak reactor coolant system pressure and fuel and clad temperatures.

Method of Analysis

The digital computer codes used in the reanalysis to evaluate the pressure transient and thermal transient were the same as those used in the FSAR. The analysis employed a constant moderator coefficient of +7 pcm/°F. Other assumptions used were consistent with those employed in the FSAR.

Results and Conclusions

Analysis of the locked rotor incident with a +7 pcm/°F moderator temperature coefficient shows that the peak reactor coolant system pressure remains below that which would cause stresses to exceed the faulted condition stress limits. The peak clad temperature for the hot spot during the worst transient remains much less than 2700°F and the amount of Zirconium - water reaction is small. Therefore, the conclusions presented in the FSAR remain valid.

F. Turbine Trip

Introduction

Two cases, analyzed for both beginning and end of life conditions, are presented in Sections 15.2.3 of the FSAR:

1. Full credit is taken for the effect of the pressurizer spray and the pressurizer power operated relief valves. Safety valves are also available.

2. No credit is taken for the effect of the pressurizer spray or power operated relief valves. Safety valves are operable.

Although the moderator temperature coefficient will be negative at end of life, all cases were repeated. The result of a loss of load is a core power level which momentarily exceeds the secondary system power extraction causing an increase in core water temperature. The consequences of the reactivity addition due to a positive moderator coefficient are increases in both peak nuclear power and pressurizer pressure.

Method of Analysis

A constant moderator temperature coefficient of +7 pcm/°F was assumed. The method of analysis and assumptions used were otherwise in accordance with those presented in the FSAR.

Results and Conclusions

The beginning of life case system transient response to a total loss of load from 100 percent power assuming pressurizer relief and spray valves was calculated. Peak pressurizer pressure reaches 2531 psia following a reactor trip on high pressurizer pressure. A minimum DNBR well above the limit value is reached shortly after reactor trip.

The transient response to a loss of load assuming no credit for pressure control was also calculated. Peak pressurizer pressure reaches 2572 psia following reactor trip on high pressurizer pressure. The DNBR increases throughout the transient.

The analysis of the beginning of life cases demonstrates that the integrity of the core and the reactor coolant system pressure boundary during a loss of load or turbine trip transient will not be impacted by a +7 pcm/°F moderator reactivity coefficient since the minimum DNB ratio remains well above the limit value, and the peak reactor coolant pressure is less than 110 percent of the design value of 2500 psia. Therefore, the conclusions presented in the FSAR remain valid.

G. Loss of Normal Feedwater/Loss of Offsite Power

Introduction

The loss of normal feedwater and loss of offsite power accidents (Sections 15.2.7 and 15.2.6 of the FSAR) are analyzed to demonstrate the ability of the secondary system auxiliary feedwater to remove decay heat from the reactor coolant system. Following initiation of the event the reactor coolant temperature rises prior to reactor trip due to the reduced heat transfer in the steam generators. Thus, the assumption of a positive moderator temperature coefficient results in a reactivity insertion and resultant increase in core power prior to reactor trip. This is turn increases the amount of heat that must be removed following reactor trip, resulting in a more severe transient.

Method of Analysis

A constant moderator temperature coefficient of +7 pcm/°F was assumed. A conservative core residual heat generation based on the 1979 version of ANS-5.1 was used. The method of analysis and assumptions used were otherwise in accordance with those presented in the FSAR.

Results and Conclusions

The transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS. The pressurizer does not fill with water, assuring that the integrity of the core is not adversely affected. 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

Introduction

The main feedwater pipe rupture accident (Section 15.2.8 of the FSAR) is analyzed to demonstrate the ability of the secondary system auxiliary feedwater to remove decay heat from the reactor coolant system. Following initiation of the event the reactor coolant temperature rises prior to reactor trip due to the reduced heat transfer in the steam generators. Thus, the assumption of a positive moderator temperature coefficient results in a reactivity insertion and resultant increase in core power prior to reactor trip. This is turn increases the amount of heat that must be removed following reactor trip, resulting in a more severe transient.

Method of Analysis

A constant moderator temperature coefficient of +7 pcm/°F was assumed. A conservative core residual heat generation based on the 1979 version of ANS-5.1 was used. The method of analysis and assumptions used were otherwise in accordance with those presented in the FSAR.

Results and Conclusions

The transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS to prevent overpressurization of the RCS or core uncovery. The reactor coolant remains subcooled, assuring that the core remains covered with water. For the case without offsite power, the results verify the natural circulation capacity of the RCS to provide sufficient heat removal capability to prevent RCS overpressurization and fuel or clad damage following reactor coolant pump coastdown.

I. Control Rod Ejection

Introduction

The rod ejection transient is analyzed at full power and hot standby for both beginning and end of life conditions in the FSAR. Since the moderator temperature coefficient is negative at end of life, only the beginning of life cases are affected by a positive MTC. The high nuclear power levels and hot spot fuel temperatures resulting from a rod ejection are increased by a positive moderator coefficient. A discussion of this transient is presented in Section 15.4.8 of the FSAR.

Method of Analysis

The digital computer codes for analysis of the nuclear power transient and hot spot heat transfer are the same as those used in the FSAR. The ejected rod worths and transient peaking factors assumed are conservative with respect to the actual calculated values for current fuel cycles. The analysis used a moderator temperature coefficient more conservative than a +7 pcm/°F for all appropriate temperature values and power levels. This is a conservative assumption since the moderator coefficient actually decreases to zero from 70 percent to 100 percent power.

Results and Conclusions

A peak clad average temperature of 2683°F was reached in the beginning of life hot zero power case. Maximum fuel temperatures were associated with the full power case.

Although the peak hot spot fuel centerline temperature for this transient exceeded the melting point, melting was restricted to less than the innermost 10 percent of the pellet.

As fuel and clad temperatures do not exceed the fuel and clad limits specified in the FSAR, there is no danger of sudden fuel dispersal into the coolant, or consequential damage to the primary coolant loop.

J. Accidental Depressurization of the Reactor Coolant System

Introduction

An accidental depressurization of the reactor coolant system results from an inadvertent opening of a pressurizer relief or safety valve (FSAR Section 15.6.1). Since a safety valve is sized to relieve at a much greater flow rate than a relief valve and will therefore allow a much more rapid depressurization, the case of a safety valve opening is analyzed. This situation initially results in a rapidly decreasing reactor coolant system pressure until the hot leg saturation pressure is reached. With a negative moderator density coefficient (positive MTC), the decrease in pressure results in an increase in core reactivity because the coolant density decreases as the pressure decreases. The most limiting case therefore assumes the reactor is in manual control, such that the increase in core reactivity causes nuclear power and average coolant temperature to increase until the reactor trips. Therefore, the consequence of the reactivity addition due to the +7 pcm/°F moderator coefficient is an increase in peak nuclear power.

Method of Analysis

The method of analysis and assumptions used were the same as those presented in the FSAR except for the following:

1. A constant moderator temperature coefficient of +7 pcm/°F was assumed.

2. The reactor was assumed to operate in the manual mode of operation to prevent rod insertion prior to reactor trip.

3. A least negative Doppler-only power coefficient of reactivity was assumed to augment any power increase due to moderator reactivity.

Results and Conclusions

The system transient response to the inadvertent opening of a pressurizer safety valve with the reactor in manual rod control was calculated. The reactor trips on overtemperature delta-T and the minimum DNBR occurs shortly after control rods begin to drop into the core.

The analysis demonstrates that the integrity of the core during a reactor coolant system depressurization transient is not adversely affected by a positive moderator reactivity coefficient since the minimum DNB ratio remains above the limit value. Therefore, the conclusions presented in the FSAR remain valid.

SECTION 3
CONCLUSIONS

To assess the effect on accident analysis of operation of McGuire Units 1 & 2 with a positive moderator temperature coefficient of +7 pcm/°F safety analyses of transients sensitive to a minimum or positive moderator coefficient were performed. These transients included control rod assembly withdrawal from subcritical, control rod assembly withdrawal at power, loss of reactor coolant flow, locked rotor, turbine trip, loss of normal feedwater, rupture of a main feedwater pipe, control rod ejection, and RCS depressurization. This study indicates that a +7 pcm/°F moderator coefficient does not result in the violation of safety limits for any of the transients analyzed.

Except as noted, the analyses employed a constant moderator coefficient of +7 pcm/°F, independent of power level. The results of this study are conservative for the accidents investigated at full power, since the proposed Technical Specification diagrammed in Figure 1 requires that the coefficient linearly decrease from +7 pcm/°F to 0 pcm/°F from 70 percent to 100 percent of rated power.

Figure 1

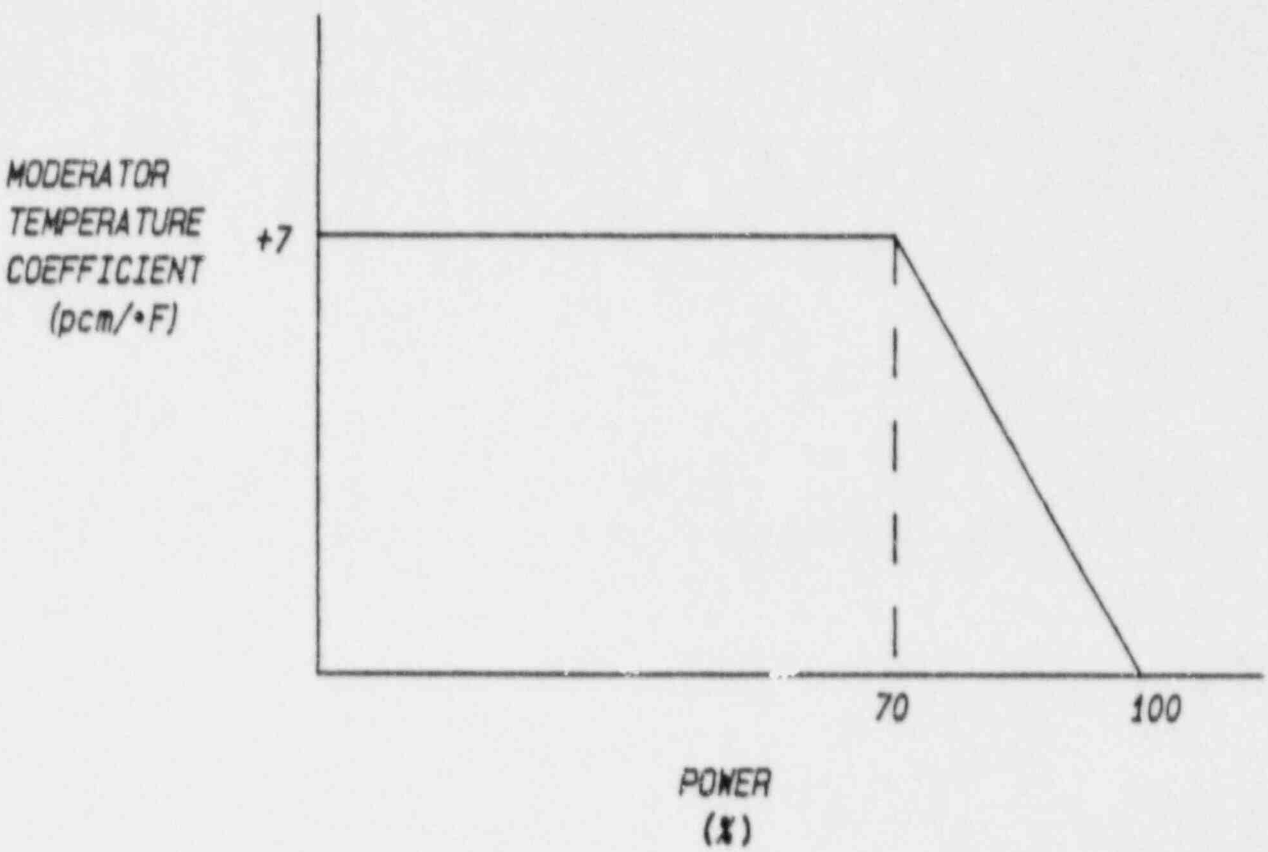

TABLE 1

FSAR ACCIDENTS EVALUATED FOR
POSITIVE MODERATOR COEFFICIENT EFFECTS

	FSAR	Accident	Time in Life
	15.1.1/1.2	Feedwater Malfunction	EOC
	15.1.3	Excessive Load Increase	BOC/EOC
	15.1.4/1.5	Steam Line Depressurization/Break	EOC
*	15.2.2/2.3	Loss of Load [illegible]ine Trip	BOC/EOC
*	15.2.6	Station Blackout	BOC
*	15.2.7	Loss of Feedwater	BOC
*	15.2.8	Feed Line Break	BOC
*	15.3.1/3.2	Loss of Flow	BOC
*	15.3.3	Locked Rotor	BOC
*	15.4.1	RCCA Withdrawal from Subcritical	BOC
*	15.4.2	RCCA Withdrawal at Power	BOC/EOC
	15.4.3	RCCA Misoperation	BOC
	15.4.4	Startup of an Inactive Loop	EOC
*	15.4.6	Boron Dilution	BOC
*	15.4.8	RCCA Ejection	BOC/EOC
	15.5.1	Spurious Actuation of SI	BOC
*	15.6.1	Accidental Depressurization of RCS	BOC
	15.6.5	LOC[illegible]	BOC

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* Accidents Explicitly Addressed

BOC - Beginning of Cycle
EOC - End of Cycle

ATTACHMENT 3

ANALYSIS OF SIGNIFICANT HAZARDS CONSIDERATION

As required by 10 CFR 50.91, this analysis is provided concerning whether the proposed amendments involve significant hazards considerations, as defined by 10 CFR 50.92. Standards for determination that a proposed amendment involves no significant hazards considerations are if operation of the facility in accordance with the proposed amendment would 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.

I. McGuire 1/Cycle 4 Reload Related Technical Specification Changes:

The proposed amendments to incorporate RAOC and a positive moderator temperature coefficient ensure that plant operation is consistent with the design and safety evaluation conclusion statements made in the McGuire Unit 1 Cycle 4 reload safety evaluation and ensure that those conclusions remain valid. The reference safety evaluation report submitted by Mr. H.B. Tucker's November 14, 1983 letter to Mr. H.R. Denton summarized the evaluation performed on the region-by-region reload transition from the McGuire Units 1 and 2 standard (STD) fueled cores to cores with all optimized fuel (OFA). The report examined the differences between the Westinghouse STD design and OFA design and evaluated the effects of these differences for the transition to an all OFA core. The report (approved by the NRC) justifies the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core, and contains summaries of the mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 4 safety evaluation. Subsequent cycle specific reload safety evaluations were to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in the reference report.

The McGuire Unit 1/Cycle 4 reload safety evaluation (Attachment 2A) presents an evaluation which demonstrates that the core reload will not adversely affect the safety of the plant. 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. The results of new analyses and the justification for the applicability of previous results for the remaining analyses is presented in the cycle specific reload safety evaluation. The results of these evaluation/analysis and tests lead to the following conclusions:

a. The Westinghouse OFA reload fuel assemblies for McGuire 1 and 2 are mechanically compatible with the STD design, control rods, and reactor internals interfaces. Both fuel assemblies satisfy the design bases for the McGuire units.

b. Changes in the nuclear characteristics due to the transition from STD to OFA fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

c. The reload OFAs are hydraulically compatible with the STD design.

ses for the OFA transition core were shown to
results by meeting the applicable criteria,
DNBR, peak pressure, and peak clad temperature,
previously reviewed and licensed safety limits

limitations given in the Technical Specifications
reload will be satisfied with the proposed

t is concluded that the Unit 1 Cycle 4 design does
acceptable safety limits to be exceeded. Further,
changes to Unit 1's operating limitations have no
valuated and approved on accident causal mechanisms

ded examples of amendments likely to involve no
siderations (48 FR 14[illegible]). One example of this type
h either may result [illegible]ome increase to the
nces of a previously [illegible]alyzed accident or may reduce
argin, but where results of the change are clearly
criteria with respect to the system or component
ard review plan: for example, a change resulting
f a small refinement of a previously used
design method". Because the evaluations previously
ll of the accidents comprising the licensing bases
ly be affected by the fuel reload were reviewed for
sign and conclude that the reload design does not
acceptable safety limits to be exceeded, the above
ed to this situation. In addition, the NRC has
significant hazards consideration determinations for
1 (and 2) reload amendments. Consequently, example
"For a nuclear power reactor, a change resulting from a
e reloading, if no fuel assemblies significantly
e found previously acceptable to the NRC for a previous
y in question are involved. This assumes that no
s are made to the acceptance criteria for the technical
at the analytical methods used to demonstrate conformance
specifications and regulations are not significantly
NRC has previously found such methods acceptable.", also

f actions not likely to involve a significant hazards
(i), "A purely administrative change to technical
For example, a change to achieve consistently throughout
ecifications, correction of an error, or a change in
Accordingly the changes to Unit 2 specifications which do
content for Unit 2 but which preserve the distinctions
ithin the common document are administrative in nature and
[illegible] hazards considerations.

d. The accident analyses for the OFA transition core were shown to provide acceptable results by meeting the applicable criteria, such as, minimum DNBR, peak pressure, and peak clad temperature, as required. The previously reviewed and licensed safety limits are met.

e. Plant operating limitations given in the Technical Specifications affected by the reload will be satisfied with the proposed changes.

From these evaluations, it is concluded that the Unit 1 Cycle 4 design does not cause the previously acceptable safety limits to be exceeded. Further, the reload and associated changes to Unit 1's operating limitations have no effects not previously evaluated and approved on accident causal mechanisms or probabilities.

The commission has provided examples of amendments likely to involve no significant hazards considerations (48 FR 14870). One example of this type is (vi), "A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where results of the change are clearly within all acceptable criteria with respect to the system or component specified in the standard review plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method". Because the evaluations previously discussed 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 and conclude that the reload design does not cause the previously acceptable safety limits to be exceeded, the above example can be applied to this situation. In addition, the NRC has previously issued no significant hazards consideration determinations for similar McGuire Unit 1 (and 2) reload amendments. Consequently, example (iii) which states "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.", also applies.

Another example of actions not likely to involve a significant hazards consideration is (i), "A purely administrative change to technical specifications: For example, a change to achieve consistently throughout the technical specifications, correction of an error, or a change in nomenclature". Accordingly the changes to Unit 2 specifications which do not change the content for Unit 2 but which preserve the distinctions between units within the common document are administrative in nature and involve no significant hazards considerations.

II. Unit 2 Increased Positive Moderator Temperature Coefficient

The proposed amendments would allow a more positive moderator temperature coefficient (MTC) to exist during power operation on McGuire Unit 2. Safety analyses and evaluations have been performed to support operation of McGuire Units 1 and 2 with a positive MTC of +7 pcm/degrees F below 70% of rated thermal power, ramping down to 0 pcm/degrees F at 100% power. These are described in a safety evaluation report (Attachment 2B) verifying the acceptability of operation of McGuire units 1 and 2 with this increased positive MTC. To assess the effect on accident analysis transients sensitive to a minimum or positive moderator coefficient were analyzed. This study indicates that a +7 pcm/degrees F moderator coefficient does not result in the violation of safety limits for any of these transients.

Since the results of the analysis show that the proposed change can be accomplished with margin to applicable FSAR safety limits, and the change is to an operating limitation involving no changes to hardware or other accident causal mechanisms and can have no effect on accident probabilities, the three standards for determination that no significant hazards considerations are involved are met. In addition, example (vi) cited in Part I above (which covered, among other changes, an identical change on Unit 1) is also applicable to this Unit 2 change.

Based upon the preceding analyses, Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration.

Darl Hood

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 15, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: B.J. Youngblood, Director
PWR Project Directorate #4

Subject: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370
McGuire 1/Cycle 4 OFA Reload

Dear Mr. Denton:

Mr. H.B. Tucker's (DPC) November 14, 1983 letter to Mr. H.R. Denton (NRC/ONRR) described planned changes in the fuel design for McGuire Nuclear Station, Units 1 and 2. Commencing with the first refueling of each of the units, the standard fuel assemblies in use were to be replaced over the next four refuelings with optimized fuel assemblies (OFA). The letter transmitted a reference safety evaluation describing the safety impact of operation with a transition core and an all OFA core. McGuire Unit 1 has begun this process with the NRC having approved the necessary license amendments, and Unit 1/Cycle 3 is currently operating with two OFA reload regions. The third such OFA Reload Region is scheduled for the upcoming cycle 4 refueling. (McGuire Unit 2 is currently in its cycle 3 refueling outage preparing for operation with its second OFA Reload Region.)

Attached are proposed license amendments to facility operating licenses NPF-9 and NPF-17 for McGuire Nuclear Station Units 1 and 2, respectively. The proposed amendments ensure that plant operation is consistent with the design and safety evaluation conclusion statements made in the McGuire Unit 1 Cycle 4 Reload Safety Evaluation (RSE) and ensure that these conclusions remain valid. Note that the McGuire Unit 2 specifications are administratively affected in that a specification currently applying to both McGuire Units 1 and 2 is split into two portions addressing the separate requirements for Units 1 and 2 created by these changes. In addition, the increased positive moderator temperature coefficient limit of +7 pcm/degrees F is being requested for McGuire Unit 2 as well as Unit 1. Although the increase in the positive MTC limit was assumed in the McGuire 1/Cycle 4 RSE, the change is not a consequence of the reload and therefore is acceptable for immediate implementation on Unit 2 (while this increase was not assumed in the McGuire Unit 2/Cycle 3 RSE, it will be needed (assumed) for future reloads and therefore should be approved on Unit 2 sometime prior to McGuire 2/Cycle 4). The McGuire FSAR will be revised to reflect this positive MTC change in the appropriate annual FSAR update following approval.

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Attachment 1 contains the proposed technical specification changes, and Attachment 2 discusses the Justification and Safety Analysis to support the proposed changes. Included in Attachment 2 is: A) the cycle-specific reload safety evaluation for McGuire Unit 1/Cycle 4; and B) the safety evaluation for operation of McGuire Units 1 and 2 with a positive moderator coefficient. The peaking factor limit report for McGuire Unit 1/Cycle 4 which is required in accordance with McGuire Technical Specification 6.9.1.9 will be submitted at least 60 days prior to cycle initial criticality. Pursuant to 10 CFR 50.91, Attachment 3 provides an analysis performed in accordance with the standards contained in 10 CFR 50.92 which concludes that the proposed amendments do not involve a significant hazards consideration. The proposed amendments have been reviewed and determined to have no adverse safety or environmental impact.

It is requested that the proposed amendments receive timely review and approval in view of the current McGuire Unit 1/Cycle 4 startup schedule. Unit 1 end of Cycle 3 refueling shutdown is currently scheduled for May 16, 1986 with Cycle 4 initial criticality scheduled for July 24, 1986. Any changes to this schedule will be provided to the NRC staff. The implications of failure to have these amendments approved by Unit 1/Cycle 4 criticality are described in Attachment 2.

Pursuant to 10 CFR 170.3(y), 170.12(c), and 170.21, Duke Power proposes that this application contains license amendments for McGuire Units 1 and 2 subject to fees based on the full cost of the review (to be calculated using the applicable professional staff rates shown in 10 CFR 170.20) and must be accompanied by an application fee of $150, with the NRC to bill Duke Power at six-month intervals for all accumulated costs for the application or when review is completed, whichever is earlier. Accordingly, please find enclosed a check in the amount of $150.00.

Should there be any questions concerning this matter or if additional information is required, please advise.

Very truly yours,

s/Hal B. Tucker

Hal B. Tucker

PBN/jgm

Attachments

xc: (w/attachments)
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Mr. W.T. Orders
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R.J. Tomonto
G.E. Vaughn
R.P. Wood
T.F. Wyke
Section File: MC-801.01
MC-813.20

HAL B. TUCKER, being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this revision to the McGuire Nuclear Station License Nos. NPF-9 and NPF-17 and that all statements and matters set forth therein are true and correct to the best of his knowledge.

s/Hal B. Tucker
Hal B. Tucker, Vice President

Subscribed and sworn to before me this 15th day of May, 1986.

s/Linda L. Kessler
Notary Public

My Commission Expires:

May 1, 1989