Oa-1 Hood

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

Tac 61512/3

June 30, 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 License Amendments - Supplement

Dear Mr. Denton:

My letter of May 15, 1986 (and related letters dated May 23, 1986 and June 6, 1986) submitted 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. Included among these amendments was a proposed Technical Specification change which allows a more positive moderator temperature coefficient (MTC) to exist during power operation (+7 pcm/degrees F below 70% of rated thermal power, ramping down to 0 pcm/degrees F at 100% RTP). This increased positive MTC limit was also requested for Unit 2 because of its desirable effects on fuel cycle flexibility.

In telecons on June 11 and 20, 1986 between Mr. D.S. Hood et. al. of your staff and Mr. P.B. Nardoci et. al. (DPC) during which various NRC concerns related to the subject amendments were discussed, certain additional information regarding the increased positive MTC change was requested to support NRC review of the proposed amendments. This additional information concerning the specific criteria and predicted values of safety analyses and evaluations performed for transients sensitive to a positive moderator coefficient and other details related to the change is provided in Attachments 1 and 2. Note that the McGuire FSAR page markups and new figures and tables reflecting the increased positive MTC change contained in Attachment 2 will be used to revise the McGuire FSAR in the appropriate annual FSAR update following approval of this change.

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Duke would like to reiterate its request that the proposed amendments receive timely review and approval with respect to the McGuire Unit 1/Cycle 4 startup schedule. Unit 1 is currently in its end of Cycle 3 refueling outage (which began May 15, 1986). However, Cycle 4 initial criticality (previously scheduled for July 24, 1986) is currently indeterminate as a result of recently discovered damage to a fuel assembly. Duke will keep the NRC apprised of any schedule changes as well as resolution of the damaged fuel assembly problem.

Since this letter contains information supplementing that provided in my May 15, 1986 submittal which is currently under review and is bounded by the contents of that submittal, the previous amendments, justification and safety analyses, and significant hazards considerations remain valid, and no additional amendment fees are necessary. Should there be any questions concerning this matter or if additional information is required, please advise.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PBN/10/jgm
Attachments

xc: (w/attachments)
Dr. J. Nelson Grace, Regional Administrator
U.S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. Dayne Brown, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
P.O. Box 12200
Raleigh, North Carolina 27605

Mr. Darl Hood
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. W.T. Orders
Senior Resident Inspector
McGuire Nuclear Station

bxc: (w/attachment 1 only)
N.A. Rutherford
J.G. Torre
S.A. Gewehr
J.B. Day
M.L. Bellville
D.R. Bradshaw
A.V. Carr
L.H. Flores
R.C. Futrell
E.M. Geddie
R.M. Gribble
W.A. Haller
G.P. Horne
M.S. Kitlan MNS
E.O. McCraw
D.S. Marquis
D.W. Perone (W)
R.B. Priory
W.D. Reckley
H.T. Snead
G.B. Swindlehurst
R.J. Tomonto
G.E. Vaughn
R.P. Wood
T.F. Wyke

(w/attachments)
R.L. Gill
P.M. Abraham
K.S. Canady
R.H. Clark
T.L. McConnell
C.D. Markham (W)
Section File: MC-801.01
MC-813.20

ATTACHMENT 1

INCREASED POSITIVE MTC TECHNICAL SPECIFICATION CHANGE
IMPACT ON FSAR CHAPTER 15 TRANSIENTS
SPECIFIC CRITERIA AND PREDICTED VALUES

The Duke submittal of May 15, 1986 included a proposed Technical Specification change which allows the moderator temperature coefficient (MTC) to be +7 pcm/degrees F at power levels below 70% RTP, ramping down to 0 pcm/degrees F at 100% rated thermal power. The supporting documentation for the proposed change identified those transient analyses in Chapter 15 of the FSAR which required evaluation due to the increase in allowable MTC. Those analyses which included an assumption regarding beginning-of-cycle MTC were reanalyzed incorporating the increased allowable MTC assumption. The submittal did not include detailed information regarding the reanalysis for each affected transient but, instead, simply stated each satisfied the applicable safety and regulatory criteria. A future FSAR update (following approval of the amendment) will include the details of the reanalysis for each affected transient with the level of information provided being the same as that currently provided in the FSAR (reference Attachment 2).

Each of the transients sensitive to a positive moderator coefficient originally discussed in Section III of the safety evaluation for operation of McGuire Units 1 and 2 with a positive moderator coefficient (reference Attachment 2B of the May 15, 1986 submittal) are discussed in additional detail below. While specific criteria for each transient are provided, specific values for parameters as predicted by the analyses may not be given. For those cases in which exact calculational results are not provided, Duke believes that the information provided is adequate to enable NRC review of the proposed revision to the allowable MTC and is consistent with the level of detail provided by Duke and other licensees in previously reviewed and approved similar type licensing requests.

A. Boron Dilution

The acceptance criteria for the analytical results of the boron dilution events are shown to be satisfied by demonstrating that the operator has adequate time to address the dilution event and/or demonstrating that the dilution transient is less severe than other analyzed reactivity transients. The boron dilution from subcritical during refueling requires that 30 minutes be available after the operator is alerted of the dilution event before the reactor becomes critical. The alarm taken credit for alerting the operator is the high flux at shutdown alarm. The boron dilution during refueling analysis is not impacted by the proposed +7 pcm/degrees F MTC. The analysis of dilution during startup is also not impacted since the operator is made aware of the dilution by a reactor trip due to high neutron flux and the loss of shutdown margin after the reactor trip is not impacted by the MTC and thus continues to satisfy the 15 minute criterion for time available for operator action. The dilution analysis for power conditions with the reactor in automatic control assumes operator notification via the rod insertion alarm.

Evaluation of the transient demonstrates that the time from alarm to the loss of shutdown margin remains greater than the 15 minute criterion. The dilution event from power with the reactor in manual control is bounded by the rod withdrawal transient and demonstrates the adequacy of the reactor trip system to prevent DNB. The time available after the reactor trip to prevent loss of shutdown margin is not affected by the MTC.

B. Control Rod Bank Withdrawal from a Subcritical Condition

The acceptance criteria for the rod bank withdrawal from subcritical/startup conditions analysis are related to the fuel design limits DNB and CLFM. The FACTRAN code is used to predict fuel temperatures and the results of the +7 pcm/degrees F MTC analysis show the fuel temperature increases are relatively small and do not introduce CLFM concerns. The DNB criterion was shown to be satisfied using the THINC computer code and WRB-1 CHF correlation which has a minimum DNBR limit of 1.17. Uncertainties are accounted for by using conservative initial condition assumptions and penalties and so the calculated DNBR is compared directly to the 1.17 correlation limit.

C. Uncontrolled Control Rod Bank Withdrawal at Power

The acceptance criteria for the rod bank withdrawal at power analyses are related to the fuel design limits DNB and CLFM. The analyses include a range of initial power levels and reactivity insertion rates. The CLFM criterion is shown to be satisfied by the calculated peak heat flux being limited to less than 118% RTP by the high flux trip function. The DNB concern is evaluated using the THINC computer code, WRB-1 correlation, and Improved Thermal Design Procedure (ITDP). The combination of uncertainties via ITDP plus the introduction of margin for miscellaneous issues result in an analysis DNBR limit of 1.47 or 1.49 for thimble and typical cells, respectively. The calculated DNBRs are compared to the appropriate analysis DNBR limit (1.47 or 1.49).

D. Loss of Reactor Coolant Flow

The acceptance criteria associated with the partial and complete loss of flow transients are that the fuel design limits (DNB and CLFM) are not exceeded and the reactor coolant system pressure remains below 110% design pressure. The LOFTRAN and FACTRAN computer code results show that the pressure does not exceed 2750 psia and that the heat flux does not exceed CLFM limits. The DNB analysis is performed using the THINC computer codes, the WRB-1 CHF correlation, and the ITDP methodology. The calculated DNBRs are compared to the appropriate analysis DNBR limit of 1.47 or 1.49 for thimble and typical cells, respectively.

E. Locked Rotor

The acceptance criteria for the locked rotor analysis are that the reactor coolant pressure remains less than 110% design pressure and based on an acceptable fuel damage model that no fuel failure results. The LOFTRAN computer code calculates the system response and predicts the reactor coolant system does not exceed 2750 psia. The analysis of the fuel using FACTRAN predicts peak clad temperatures less than the 2700 degrees F limit assuming DNB occurs at the initiation of the transient. The predicted maximum system pressure is 2613 psia and the maximum clad temperature at the core hot spot is 2009 degrees F in the results of the +7 pcm/degrees F MTC analysis.

F. Turbine Trip

The acceptance criteria for the turbine trip transient analysis are that the reactor coolant pressure remains less than 2750 psia and fuel design limits are not exceeded. The LOFTRAN computer code is used to demonstrate that the pressurizer safety valves are able to maintain the system pressure under 2750 psia and secondary side integrity is maintained. The DNB analysis employs the THINC computer code, WRB-1 CHF correlation, and ITDP methodology. The calculated DNBRs are compared to the appropriate analysis DNBR limit of 1.47 or 1.49 for thimble and typical cells, respectively.

G. Loss of Normal Feedwater/Loss of Offsite Power

The acceptance criteria for the loss of feedwater and loss of offsite power transient analyses are that the reactor coolant and secondary systems not exceed 110% design values, the minimum DNBR remains above appropriate limits, and the performance of heat removal systems are adequate to maintain core cooling. The LOFTRAN computer code is used to model primary and secondary systems and demonstrate plant components/systems are adequate to prevent exceeding pressure limits or fuel integrity limits and provide the necessary decay heat removal capability.

H. Rupture of a Main Feedwater Pipe

The acceptance criteria for the rupture of a main feedwater pipe analysis are that reactor coolant and secondary systems remain below pressure limits and the performance of heat removal systems are adequate to maintain core cooling. The LOFTRAN computer code is used to demonstrate that pressure limits are not exceeded, the primary coolant remains subcooled, and the core remains covered.

I. Control Rod Ejection

The control rod ejection analysis is demonstrated to satisfy applicable acceptance criteria by limiting the allowable calculated fuel pellet enthalpy at the core hot spot (225 cal/gm and 200 cal/gm for unirradiated and irradiated fuel, respectively), limiting the allowable clad temperatures predicted at the hot spot to 2700 degrees F, limiting the peak calculated reactor coolant pressure to the appropriate design limit, and limiting the calculated volume of fuel melting at the core hot spot to less than 10%. The pressure response analysis is described in WCAP-7588, Rev.1A (January 1975) and remains applicable for the +7 pcm/degrees F MTC analysis. The peak clad temperature predicted was 2683 degrees F associated with the BOL HZP case. Maximum fuel temperatures are associated with the full power cases but fuel melting was calculated to be less than the innermost 10% of the pellet.

J. Accidental Depressurization of the Reactor Coolant System

The acceptance criteria for the accidental depressurization of the reactor coolant system are shown to be satisfied by predicting a minimum DNBR greater than the minimum allowable DNBR. The transient was analyzed using the THINC computer code, WRB-1 CHF correlation, and ITDP methodology. The predicted DNBR is compared to the analysis limits of 1.47 and 1.49 for thimble and typical cells, respectively.

ATTACHMENT 2

INCREASED POSITIVE MTC TECHNICAL SPECIFICATION CHANGE

FSAR CHAPTER 15 REVISIONS

NOTE: FSAR section page markups are attached for each of the Chapter 15 accidents that were reanalyzed as a result of the proposed Technical Specification change to allow a +7 pcm/degrees F moderator temperature coefficient. In addition however, miscellaneous revisions not related to this change are also included in these markups. This material will be used in subsequent updating of the McGuire Units 1 and 2 Final Safety Analysis Report following approval of the proposed change.