

March 15, 1990

Dockets Nos. 50-369
and 50-370

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: ISSUANCE OF AMENDMENT NO.105 TO FACILITY OPERATING LICENSE NPF-9 AND
AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NPF-17 - MCGUIRE
NUCLEAR STATION, UNITS 1 AND 2 (TACS 75826/75827)

The Nuclear Regulatory Commission has issued the enclosed Amendment No.105 to Facility Operating License NPF-9 and Amendment No. 87 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 17, as supplemented January 29, 1990.

The amendments change the TSs for Units 1 and 2 by replacing the values of cycle-specific parameter limits in core-related specifications with a reference to a Core Operating Limits Report (COLR) which will contain the values of these limits. These amendments also include the addition of the COLR to the Definitions section of the TSs and to the reporting requirements in the TS Administrative Controls. Additionally, the amendments change the TS Table of Contents and delete several obsolete footnotes.

A copy of the related Safety Evaluation supporting the amendments is also enclosed. Notice of issuance of the amendments will be included in the Commission's

Sincerely,

/s/

Darl Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No.105 to NPF-9
2. Amendment No. 87 to NPF-17
3. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

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DATED: March 15, 1990

AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-9

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-9 filed by the Duke Power Company (the licensee) dated January 17, as supplemented January 29, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.105 , are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 15, 1990



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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. NPF-17

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-17 filed by the Duke Power Company (the licensee) dated January 17, as supplemented January 29, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 15, 1990

ATTACHMENT TO LICENSE AMENDMENT NO.105

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

<u>Remove Pages</u>	<u>Insert Pages</u>
I & II	I & II
V	V
XXII	XXII
1-2 thru 1-7	1-2 thru 1-7
1-8	1-8*
3/4 1-4 & 1-5	3/4 1-4 & 1-5
3/4 1-14 & 1-15	3/4 1-14 & 1-15
3/4 1-20 thru 1-22	3/4 1-20 thru 1-22
3/4 2-1	3/4 2-1
3/4 2-3	3/4 2-3
3/4 2-6 thru 2-9	3/4 2-6 thru 2-9
3/4 2-9a & 2-9b	3/4 2-9a & 2-9b
3/4 2-12	3/4 2-12
3/4 2-14 & 2-15	3/4 2-14 & 2-15
3/4 2-17	3/4 2-17
3/4 3-45	3/4 3-45
B 3/4 1-1	B 3/4 1-1
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1 & 2-2	B 3/4 2-1 & 2-2
B 3/4 2-2a	B 3/4 2-2a
B 3/4 2-4	B 3/4 2-4
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*Overleaf page provided to maintain document completeness.

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DEFINITIONS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
 - 1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or operator action during periods when containment isolation valves may be opened under administrative controls pursuant to Specification 4.6.1.1.a; or
 - 2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.
- b. All equipment hatches are closed and sealed,
- c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of Specification 3.6.1.2, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

\bar{E} - AVERAGE DISINTEGRATION ENERGY

1.12 \bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

DEFINITIONS

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

- a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or
- c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H₂O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.

DEFINITIONS

PURGE - PURGING

1.24 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR BUILDING INTEGRITY

1.27 REACTOR BUILDING INTEGRITY shall exist when:

- a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,
- b. The Annulus Ventilation System is in compliance with the requirements of Specification 3.6.1.8, and
- c. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

DEFINITIONS

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.32 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.33 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a SOLIDIFICATION agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP).

SOURCE CHECK

1.34 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:

- a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and
- b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.36 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

DEFINITIONS

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.37 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.38 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.39 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.40 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.41 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM

1.42 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

TABLE 1.1
FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

APPLICABILITY: Figure 3.1-0 and COLR Figure 1 Limits - MODES 1 and 2* only.#
End of Cycle Life (EOL) Limit - MODES 1, 2, and 3 only.#

ACTION:

- a. With the MTC more positive than the limit specified in Figure 1 of the COLR, 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 limit specified in Figure 1 of the COLR 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 EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

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

#See Special Test Exception 3.10.3.

REACTIVITY CONTROL SYSTEMS

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 specified in the COLR, 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 the 300 ppm surveillance limit specified in the COLR (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 the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

- a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.
- b. With more than one full-length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.
- c. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:
 1. The rod is restored to OPERABLE status within the above alignment requirements, or
 2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or
 3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:
 - a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;
 - b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

ACTION (Continued)

- c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours; and
 - d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.
- d. With more than one full-length rod trippable but inoperable due to causes other than addressed by ACTION a above, POWER OPERATION may continue provided that:
- 1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and
 - 2. The inoperable rods are restored to OPERABLE status within 72 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

- a. Restore the rod to within the insertion limit specified in the COLR, or
- b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

- a. Within 15 minutes prior to withdrawal of any rods in Control Banks A, B, C or D during an approach to reactor criticality, and
- b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

- a. Restore the control banks to within the limits within 2 hours, or
- b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or
- c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.

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

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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 as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or
- b. the target band specified in the COLR 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 limits specified in the COLR,
 1. Either restore the indicated AFD to within the COLR 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 COLR specified 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 limits specified in the COLR.

*See Special Test Exception 3.10.2.

** APL^{ND} is the minimum allowable (nuclear design) power level for base load operation and is specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

Where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

$K(Z)$ = the normalized $F_Q(Z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

- a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and
- b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

- a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.
- b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.
- c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP}}{P \times W(z)} \times K(z) \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP}}{W(z) \times 0.5} \times K(z) \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(z)$ is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and $W(z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP} , $K(z)$, and $W(z)$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

- d. Measuring $F_Q^M(z)$ according to the following schedule:
 1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or
 2. At least once per 31 Effective Full Power Days, whichever occurs first.

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

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

- e. With measurements indicating

$$\begin{array}{c} \text{maximum} \\ \text{over } z \end{array} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

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

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

- 2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that
- $$\begin{array}{c} \text{maximum} \\ \text{over } z \end{array} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

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

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

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

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

- 2) One of the following actions shall be taken:

- a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or
- b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or
- c) Verify that the requirements of Specification 4.2.2.3 for base load operation are satisfied and enter base load operation.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

- a. Prior to entering base load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain base load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) during this time period. Base load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

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

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the F_Q limit. $K(z)$ is the normalized $F_Q(z)$ as a function of core height. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP} , $K(z)$, and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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

4.2.2.4 During base load operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

- a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} .
- b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

* APL^{ND} is the minimum allowable (nuclear design) power level for base load operation in Specification 3.2.1.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

- c. Satisfying the following relationship:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. F_Q^{RTP} is the F_Q limit.

$K(Z)$ is the normalized $F_Q(Z)$ as a function of core height. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP} , $K(Z)$, and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

- d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:
1. Prior to entering base load operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and
 2. At least once per 31 effective full power days.
- e. With measurements indicating

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

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

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

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

- f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:
1. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation specified in the CORE OPERATING LIMITS REPORT (COLR) for four loop operation:

Where:

$$a. \quad R = \frac{F_{\Delta H}^N}{F_{\Delta H}^{RTP} [1.0 + MF_{\Delta H} (1.0 - P)]},$$

$$b. \quad P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}},$$

c. $F_{\Delta H}^N$ = Measured values of $F_{\Delta H}^N$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F_{\Delta H}^N$ shall be used to calculate R since the figure specified in the COLR includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 1.7% for flow and 4% for incore measurement of $F_{\Delta H}^N$,

d. $F_{\Delta H}^{RTP}$ = The $F_{\Delta H}^N$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

e. $MF_{\Delta H}$ = The power factor multiplier specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

- b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.
- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:
 1. A nominal 50% of RATED THERMAL POWER,
 2. A nominal 75% of RATED THERMAL POWER, and
 3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

- 4.2.3.1 The provisions of Specification 4.0.4 are not applicable.
- 4.2.3.2 The combination of indicated RCS total flow rate determined by process computer readings or digital voltmeter measurement and R shall be within the region of acceptable operation specified in the COLR:
 - a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and
 - b. At least once per 31 Effective Full Power Days.
- 4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.
- 4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.
- 4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

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INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

- a. At least 75% of the detector thimbles,
- b. A minimum of two detector thimbles per core quadrant, and
- c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

- a. Recalibration of the Excore Neutron Flux Detection System,
- b. Monitoring the QUADRANT POWER TILT RATIO, or
- c. Measurement of $F_{\Delta H}^N$ and $F_Q(Z)$

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

- a. Recalibration of the Excore Neutron Flux Detection System, or
- b. Monitoring the QUADRANT POWER TILT RATIO, or
- c. Measurement of $F_{\Delta H}^N$ and $F_Q(Z)$

3/4.1 REACTIVITY CONTROL SYSTEMS

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 End Of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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 EOL MTC value.

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2000 gallons of 7000-ppm borated water from the boric acid storage tanks or 10,000 gallons of 2000-ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The control rod insertion limit and shutdown rod insertion limits are specified in the CORE OPERATING LIMITS REPORT per specification 6.9.1.9.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

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;
- $F_{\Delta H}^N$ 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 the F_Q^{RTP} limit specified in the CORE OPERATING LIMITS REPORT (COLR) 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.

POWER DISTRIBUTION LIMITS

BASES

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND} , the limits on AFD are defined in the COLR, 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 limits of which are defined in the COLR, and 2) base load operation, which is defined as the maintenance of the AFD within a COLR specified 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 as specified in the COLR, $APL^{ND} \leq \text{power} \leq APL^{BL}$ 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.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. These limits are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

- a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps from the group demand position;
- b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;
- c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and
- d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}^N$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on the figure specified in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the power level is decreased) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}^N$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in the figure specified in the COLR, accounts for $F_{\Delta H}^N$ less than or equal to the $F_{\Delta H}^{RTP}$ limit specified in the COLR. This value is used in the various accident analyses where $F_{\Delta H}^N$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Margin between the safety analysis limit DNBRs and the design limit DNBRs is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty (2%) and the appropriate fuel rod bow DNBR penalty (WCAP - 8691, Rev. 1).

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate

POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of the figure specified in the COLR. Measurement errors of 1.7% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in the figure specified in the COLR. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the figure specified in the COLR.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or base load operation, $W(z)$ or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. $W(z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by base load operation which result in less severe transient values. The $W(z)$ function for normal operation and the $W(z)_{BL}$ function for base load operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference limits, target band, and APL^{ND} for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_Q^{RTP} , $K(Z)$, $W(Z)$, APL^{ND} and $W(Z)_{BL}$ for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $MF_{\Delta H}$, limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).
(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)
2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor ($W(Z)$ surveillance requirements for F_Q Methodology.)
3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

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



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO.105 TO FACILITY OPERATING LICENSE NPF-9
AND AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NPF-17
DUKE POWER COMPANY
DOCKETS NOS. 50-369 AND 50-370
MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated January 17, 1990, as supplemented January 29, 1990, Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) for McGuire Nuclear Station, Units 1 and 2. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a "Core Operating Limits Report" (COLR) for the values of those limits. The proposed changes also include the addition of the "Core Operating Limits Report" to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket that was endorsed by the Babcock and Wilcox Owners Group. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16 dated October 4, 1988.

These amendments also delete several obsolete footnotes which permitted, subject to specified requirements, operation of McGuire Unit 1 during a portion of its fuel cycle 6 when the number of available movable detector thimbles in the core was less than 75%. This deletion is in accordance with the licensee's request of January 29, 1990.

While the January 29, 1990, submittal was not referenced in the February 7, 1990, Federal Register notice, the changes proposed in the submittal were described in the notice. The licensee also provided a sample COLR by letter dated January 29, 1990. This supplemental submittal did not alter the action noticed or affect the initial determination of no significant hazards consideration.

2.0 EVALUATION

2.1 Cycle-Specific Parameter Limits

The licensee's proposed changes to the TSs are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

- (1) The Definition section of the TSs was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

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- (2) The following specifications and bases were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

3/4.1.3.5	Reactivity Control Systems - Shutdown Rod Insertion Limit
3.1.3.6	Reactivity Control Systems - Control Rod Insertion Limits
Figure 3.1-1	Rod Bank Insertion Limits Versus Relative Power
3.2.1	Axial Flux Difference (AFD)
Figure 3.2-1	AFD Limits as a Function of Rated Thermal Power
3/4.2.2	Heat Flux Hot Channel Factor ($F_0(Z)$)
Figure 3.2-2	$K(Z)$ - Normalized $F_0(Z)$ as a Function of Core Height
3/4.2.3	Power Distribution Limits - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor
Figure 3.2-3	RCS Flow Rate Versus R - Four Loops in Operation
Bases 3/4.1.1.3	Moderator Temperature Coefficient
Bases 3/4.1.3	Movable Control Assemblies
Bases 3/4.2.1	Axial Flux Difference
Bases 3/4.2.2 and 3/4.2.3	Heat Flux Hot Channel Factor, and RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor

- (3) Specification 6.9.1.9, "Peaking Factor Limit Report," was revised and retitled "Core Operating Limits Report" so as to address the reporting requirements of the Administrative Controls section of the TSS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using the NRC-approved methodology in WCAP-9272-P-A, WCAP-10216-P-A and WCAP-10266-P-A, Revision 2, and consistent with all applicable limits of the safety analysis. Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in the TSS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable.

As part of the implementation of Generic Letter 88-16, the NRC staff also reviewed a sample COLR provided by the licensee's submittal of January 29, 1990. The staff concludes that the format and content of the sample COLR are acceptable.

2.2 Movable Incore Detector Thimbles

By letter dated January 29, 1990, the licensee requested deletion of several footnotes, and associated references to them. These footnotes and references had been previously added to the TSs by License Amendments 101 (Unit 1) and 83 (Unit 2). The footnotes applied only during fuel Cycle 6 of McGuire Unit 1 and addressed operation with less than 75% (but with at least 50%) of the total movable detector thimbles in the core available. These footnotes and their references appear on TS pages 3/4 2-7, 3/4 2-9, 3/4 2-9b, 3/4 2-14, and 3/4 3-45.

Because McGuire Unit 1 has now completed operation with fuel Cycle 6, these footnotes, and references to them, are obsolete. Accordingly, their removal from the TSs is purely administrative and has no adverse safety impact. This change is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The amendments also involve changes in recordkeeping, reporting, or administrative procedures or requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (55 FR 4268) on February 7, 1990. The Commission consulted with the State of North Carolina. No public comments were received, and the State of North Carolina did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: March 15, 1990