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

February 9, 1989

Docket No. 50-315
and 50-316

Mr. Milton Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. DPR-58 and Amendment No. 107 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated March 26, 1987 as clarified and supplemented by letters dated August 25, 1987, and June 7 and August 31, 1988. Also, you corrected typographical errors in the original submittal via replacement pages provided to the NRC on January 13, 1988.

These amendments revise the Technical Specifications on boron concentrations; change the moderator temperature coefficient to a ramp function rather than a step function; remove 3-loop operation in Modes 1 and 2; add additional restrictions because of safety analyses; add footnotes such that addition of water from the RWST does not constitute a boron dilution; clarify trip channel positions for different pressure between steam lines in 3-loop operation; require two PORV's be available in Modes 1, 2, and 3; add 4.0.4 and 3.0.4 section exemptions; change values related to boron dilution events, half-loop operation, RCS flow measurement, auxiliary feedwater flow measurements, and flow measurement error; change descriptions of P-12 interlock; simplify power distribution and APDMS requirements; make changes to achieve similarity between units; and make miscellaneous editorial changes.

By letter dated June 7, 1988, the licensee withdrew a portion of their proposed changes related to requirements to submit a peaking factor limit report each cycle. The withdrawal was requested by the NRC since NRC requirements with regard to the report are in the process of being reevaluated. This will be a subject of further licensing action.

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1/1

See Reports

CP

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by

John Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 120 to DPR-58
2. Amendment No. 107 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

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A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

John Stang

John Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

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2. Amendment No. 107to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated March 26, 1987, as clarified and supplemented by letter dated August 25, 1987, January 13, 1988, June 7, 1988, and August 31, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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;

   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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance. The Technical Specifications are to become effective within 120 days of receipt of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 9, 1989

TACHMENT TO LICENSE AMENDMENT

AMENDMENT NO 120 FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO.50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

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MEMBER(S) OF THE PUBLIC

1.35 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 utility, its contractors or its 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.

SITE BOUNDARY

1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access 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 industrial, commercial, institutional and/or recreational purposes.

DESIGN THERMAL POWER

1.38 DESIGN THERMAL POWER shall be a design total reactor core heat transfer rate to the reactor coolant of 3411 MWt. See Table 1.3.

ALLOWABLE POWER LEVEL (APL)

1.39 APL means "allowable power level" which is that power level, less than or equal to 100% RATED THERMAL POWER, at which the plant may be operated to ensure that power distribution limits are satisfied.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = 1.135$

$K_2 = 0.0130$

$K_3 = 0.000659$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +2 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent DESIGN THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of DESIGN THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 2.3 percent of its value at DESIGN THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +2 percent, the ΔT trip setpoint shall be automatically reduced by 1.8 percent of its value at DESIGN THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 \left(T - T''\right) - f_2(\Delta I)\right]$

where:

ΔT_o	=	Extrapolated ΔT at DESIGN THERMAL POWER
T	=	Average temperature, ^{o}F
T''	=	Indicated T_{avg} at DESIGN THERMAL POWER 577.1^{o}F
K_4	=	1.089
K_5	=	0.0177/^{o}F for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.0011 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	f_1 (ΔI) as defined in Note 1 above.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures, because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in the heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore, THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent confidence that the minimum DNBR for the limiting rod is greater than or equal to the applicable design DNBR limit for each fuel type (as defined below). For 4 loop operation, the improved thermal design procedure is used. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit (as defined below), establishes a design DNBR limit value, which must be met in plant safety analyses, using values of input parameters without uncertainties.

The table below indicates the relationship between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design.

4 Loop Operation

	Westinghouse Fuel (15x15 OFA)		Exxon Nuclear Co. Fuel (15x15)	
	(WRB-1 Correlation)		(W-3 Correlation)	
	Typical Cell*	Thimble Cell**	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17	1.30	1.30
Design Limit DNBR	1.32	1.31	1.58	1.50
Safety Analysis Limit DNBR	1.69	1.69	1.58	1.50

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

* represents typical fuel rod
**represents fuel rods near guide tube

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. If axial peaks are more severe than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above the P-8 setpoint, less than or equal to 31% of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip, to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses, but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by less than or equal to 0.71×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. A 0.1 second time delay is incorporated in each of these trips to prevent spurious reactor trips from momentary electrical power transients.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of any one pump breaker above P-8 or the opening of two or more pump breakers below P-8. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.60% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN less than 1.60% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.60% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2##, at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

*See Special Test Exception 3.10.1

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

##With K_{eff} less than 1.0

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

a. In MODE 4:

1. Greater than or equal to 1.6% $\Delta k/k$ when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

b. In MODE 5:

1. Greater than or equal to 1.0% $\Delta k/k$ when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

APPLICABILITY: MODES 4 and 5

ACTION:

With SHUTDOWN MARGIN less than the above limits, immediately initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the above limits:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

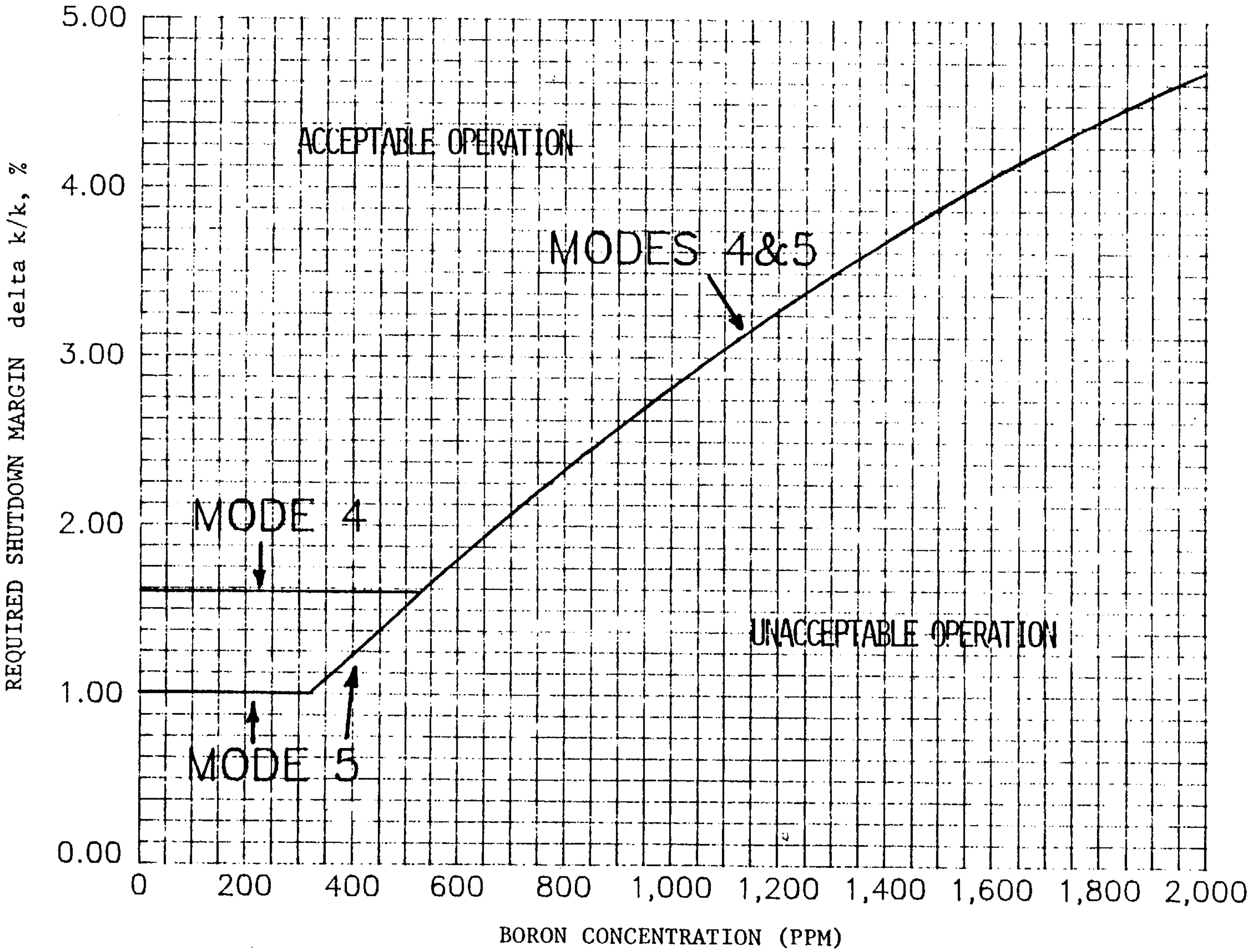


Figure 3.1-3 REQUIRED SHUTDOWN MARGIN

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be greater than or equal to 2000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.*

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system less than 2000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be greater than or equal to 2000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying greater than or equal to 2000 gpm through the reactor coolant system.

*For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

SURVEILLANCE REQUIREMENTS

4.1.1.4 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 3.1.1.4.a, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -3.0×10^{-4} $\Delta k/k/^{o}F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.0×10^{-4} $\Delta k/k/^{o}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.4.b, at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the temperature of the heat traced portion of the flow path is $\geq$ 145°F when a flow path from the boric acid tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.*

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 170°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that, on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2390 psig when tested pursuant to Specification 4.0.5 at least once per 31 days.

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 170°F.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one boric acid transfer pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 At least the above required boric acid transfer pump shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 110 psig,

c. Verifying pump operation for at least 15 minutes, and

d. Verifying that the pump is aligned to receive electrical power from an OPERABLE emergency bus.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORIC ACID TRANSFER PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at $200^{\circ}F$; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 At least the above required boric acid pump shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 110 psig,

c. Verifying pump operation for at least 15 minutes.

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 inoperable or 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 inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The affected rod is restored to OPERABLE status within the above alignment requirements, or

   2. The affected 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, and

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

*See Special Test Exceptions 3.10.2 and 3.10.4

LIMITING CONDITION FOR OPERATION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

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 shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE
FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within $\pm$ 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (228 steps) shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to entering MODE 2:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be fully withdrawn (228 steps).

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be fully withdrawn:

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 3.10.4
#With K_{eff} greater than or equal to 1.0

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 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 3.10.4

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

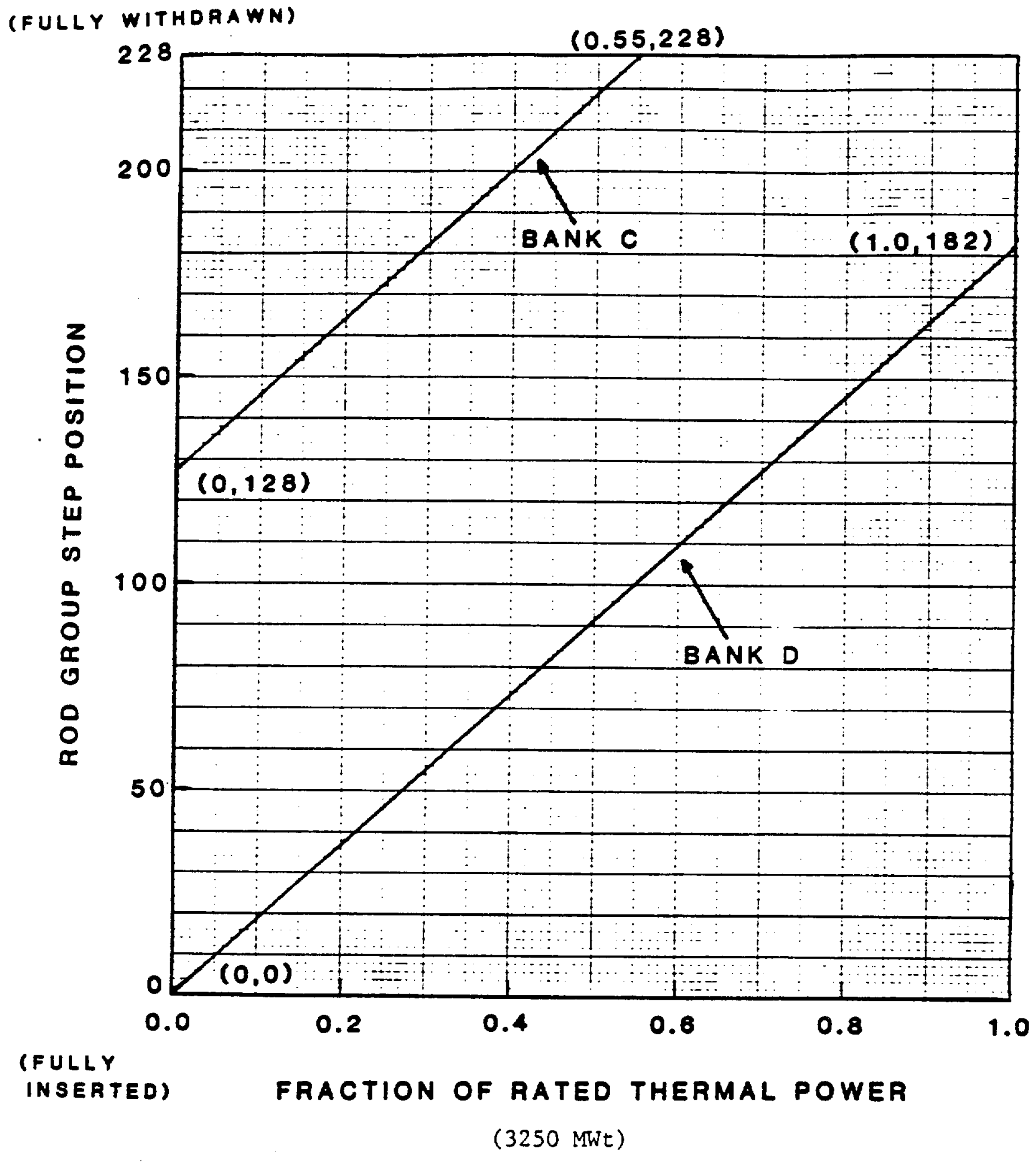


FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER 4 LOOP OPERATION

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (±5% or ±3% flux difference units) about a target flux difference.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, 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) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% 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 Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE 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 AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

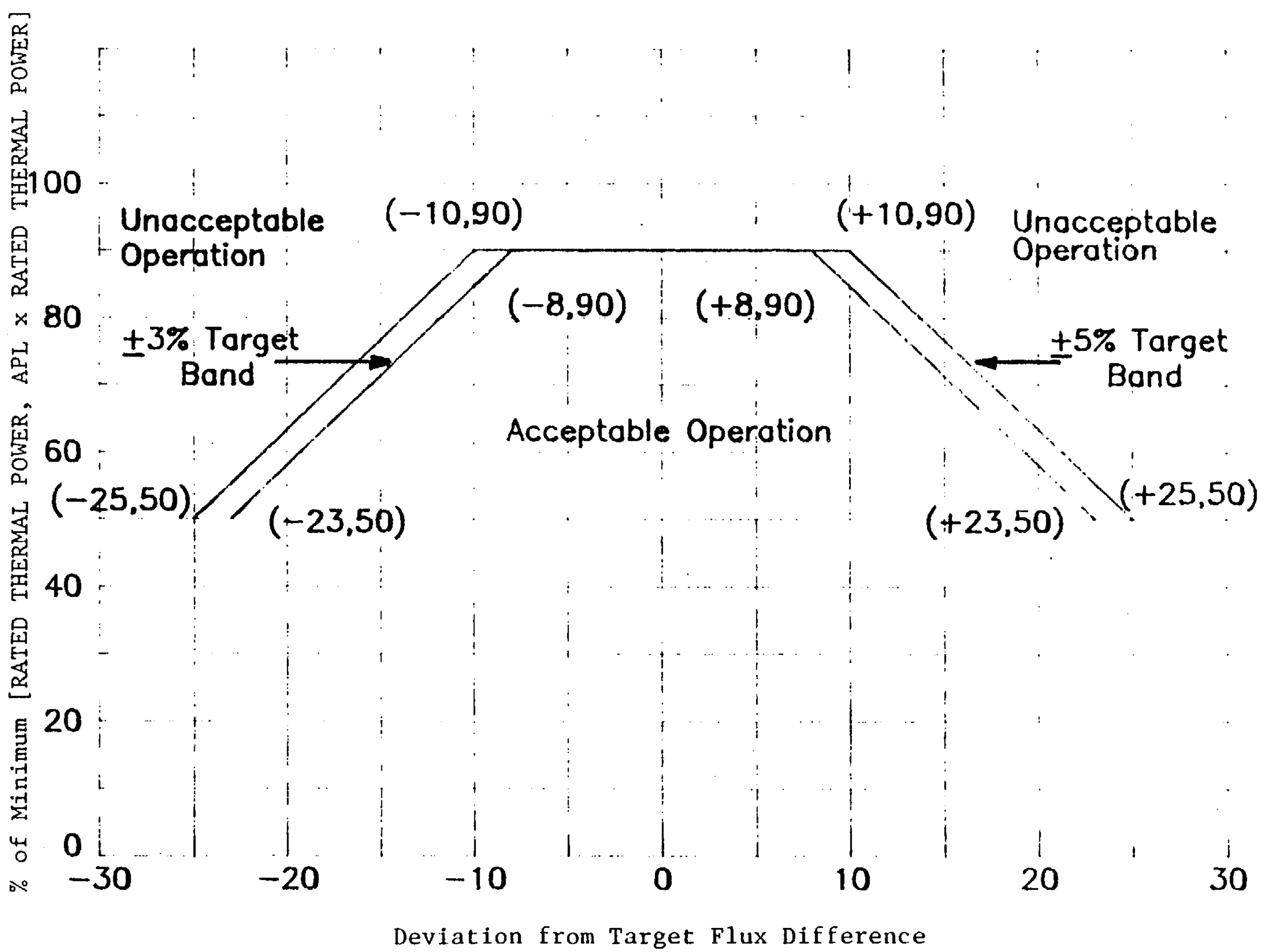
a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. A penalty deviation of one-half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target axial flux difference of each OPERABLE excore channel shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The allowable values of the target band are ± 5% or ± 3%. Redefinition of the target band from ± 3% to ± 5% between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from ± 5% to ± 3% is allowed only in conjunction with the determination of a new target axial flux difference. The provisions of Specification 4.0.4 are not applicable.

FIGURE 3.2-1 ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE



HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

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

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q(Z) \leq \frac{[2.10]}{P}$ [K(Z)]	$F_Q(Z) \leq \frac{[1.82]}{P}$ [K(Z)]	$P > 0.5$
$F_Q(Z) \leq$ [4.20] [K(Z)]	$F_Q(Z) \leq$ [3.64] [K(Z)]	$P \leq 0.5$

- $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$
- $F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- K(Z) is the function obtained from Figure 3.2-3 for Westinghouse fuel and Figure 3.2-2 for Exxon Nuclear Co. fuel.

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 ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit.

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be determined to be within its limit above 5% of RATED THERMAL POWER according to the following schedule:

a. Whenever $F_Q(Z)$ is measured for reasons other than meeting the requirement of 4.2.6.2, or

b. At least once per 31 effective full power days, whichever occurs first.

FIGURE 3.2-2 EXXON FUEL

K(Z) NORMALIZED VS. CORE HEIGHT

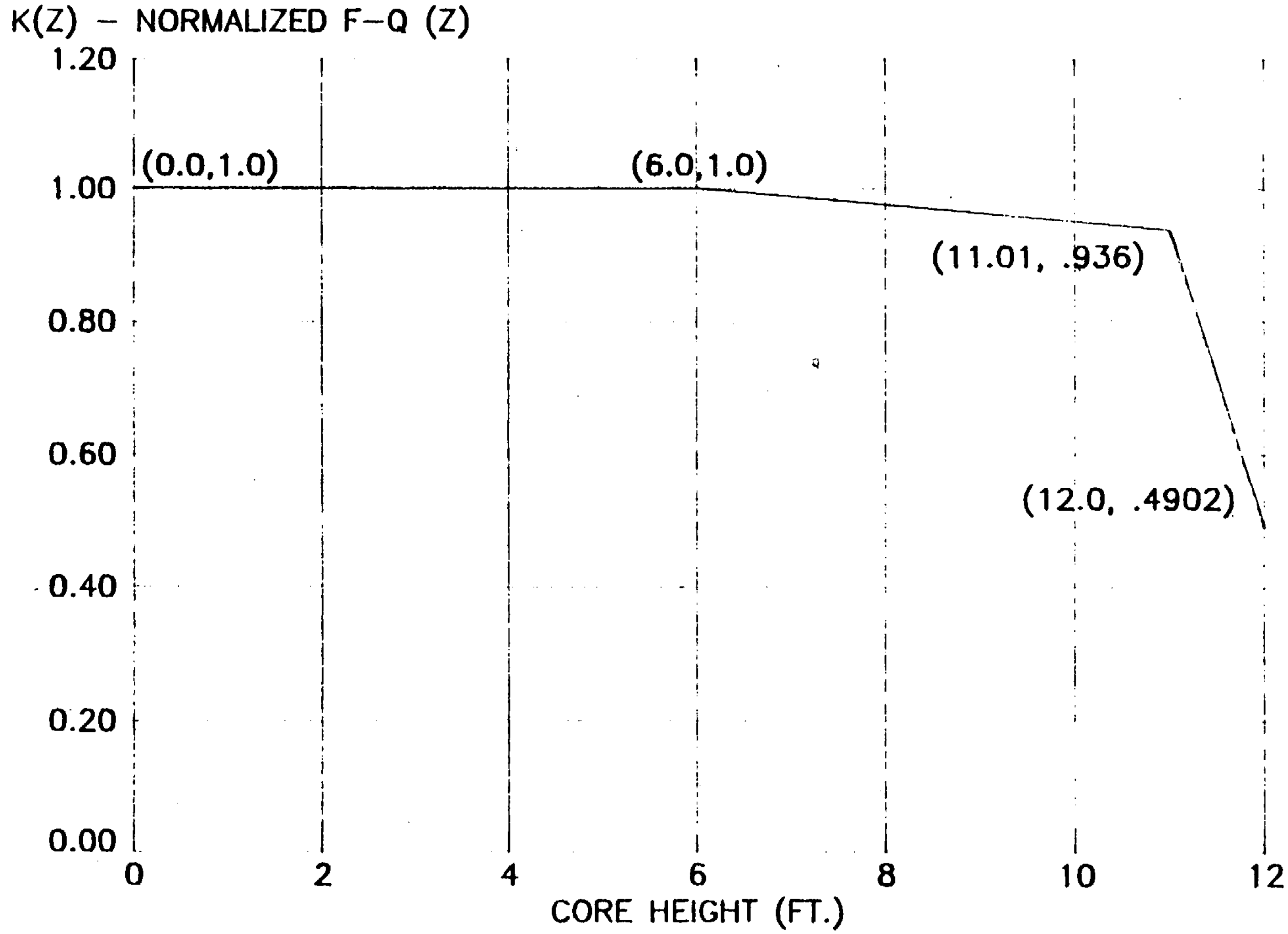
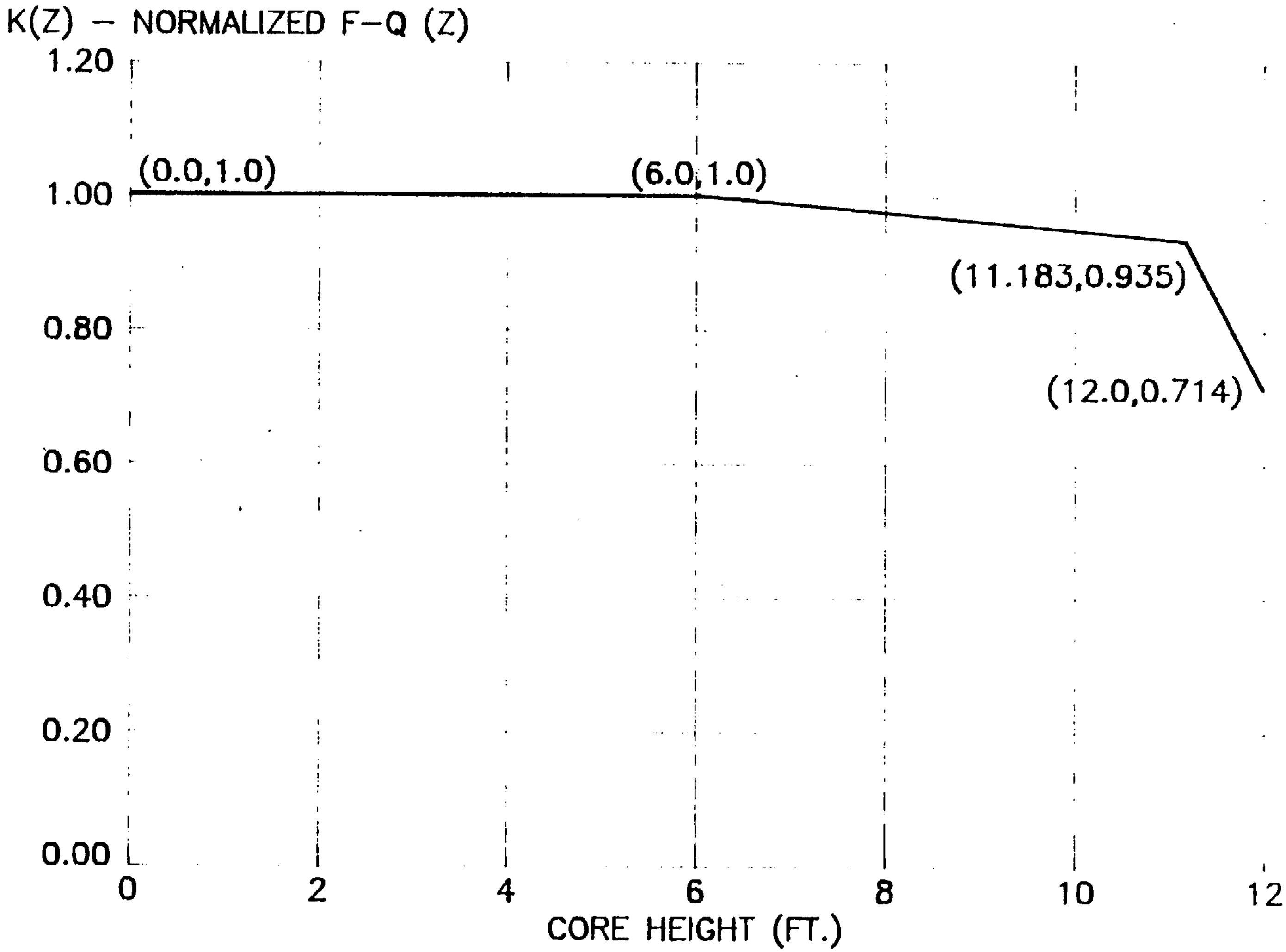


FIGURE 3.2-3 WESTINGHOUSE FUEL
K(Z) NORMALIZED VS. CORE HEIGHT



NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationships:

$F^N_{\Delta H} \leq 1.49\ [1 + 0.3\ (1-P)]$ (for Westinghouse fuel)

and $F^N_{\Delta H} \leq 1.45\ [1 + 0.2\ (1-P)]$ (for Exxon Nuclear Co. fuel)

where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours 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. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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.

c. The provisions of Specification 4.0.4 are not applicable.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Within 2 hours:

a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours 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.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is greater than 75 percent of RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The indicators used to determine RCS total flow rate shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation at RATED THERMAL POWER
Reactor Coolant System T_{avg}	$\leq$ 570.4°F*
Pressurizer Pressure	$\geq$ 2205 psig**
Reactor Coolant System Total Flow Rate	$\geq$ 138.6 x 10^6 lbs/hr***

* Indicated average of at least three OPERABLE instrument loops.

** Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

***Indicated value.

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

3.2.6 THERMAL POWER shall be less than or equal to ALLOWABLE POWER LEVEL (APL), given by the following relationships:

Westinghouse Fuel

$$APL = \min \text{ over } Z \text{ of } \frac{2.10\ K(Z)}{F_Q(Z) x V(Z) x F_p} \text{ x } 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

Exxon Nuclear Co. Fuel

$$APL = \min \text{ over } Z \text{ of } \frac{1.82\ K(Z)}{F_Q(Z) x V(Z) x F_p} \text{ x } 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

- $F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

- $V(Z)$ is the function defined in the Peaking Factor Limit Report.

- F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in max over Z of $\frac{F_Q(Z)}{K(Z)}$ with exposure. Then either of the penalties, F_p, shall be taken:

  F_p = 1.02, or

  F_p = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until two successive maps indicate that the max over Z of

  $\frac{F_Q(Z)}{K(Z)}$ is not increasing.

- The above limit is not applicable in the following core regions.

  1) Lower core region 0% to 10% inclusive.
  2) Upper core region 90% to 100% inclusive.

APPLICABILITY: MODE 1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With THERMAL POWER exceeding APL:

a. Reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes. Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER 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 ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER.

b. THERMAL POWER may be increased to a new APL calculated at the reduced power by either redefining the target axial flux difference or by correcting the cause of the high $F_Q(Z)$ condition.

SURVEILLANCE REQUIREMENTS

4.2.6.1 The provisions of Specification 4.0.4 are not applicable.

4.2.6.2 APL shall be determined by measurement in conjunction with the target flux difference and target band determination* above 15% of RATED THERMAL POWER, according to the following schedule:

a. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which APL was last determined**, or

b. At least once per 31 effective full power days, whichever occurs first.

* APL can be redefined by remeasuring the target axial flux difference in accordance with ACTION statement b of Specification 3.2.6.

**During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

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TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2 and *	$2^{\#}$
3. Power Range, Neutron Flux, High Positive Rate	4	2	3	1, 2	$2^{\#}$
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	$2^{\#}$
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	$2^{\#\#}$ and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	$6^{\#}$
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	$6^{\#}$

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	4	1	7#
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10#
B. Above P-7 and below P-8	1/breaker	2	1/breaker per operating loop	1	11#
21. Reactor Trip Breakers	2	1	2	1, 2 3*,4*,5*	1, 13 14
22. Automatic Trip Logic	2	1	2	1, 2 3*,4*,5*	1 14

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

The provisions of Specification 3.0.4 are not applicable.

High voltage to detector may be de-energized above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied.

a. The inoperable channel is placed in tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of the other channels per Specification 4.3.1.1.1.

c. Either THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.c.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

TABLE 3.3-1 (Continued)

ACTION 8 - (Deleted.)

ACTION 9 - (Deleted.)

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 13 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 1. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels less than 6×10^{-11} amps.	P-6 prevents or defeats the manual block of source range reactor trip.

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels greater than or equal to 11% of RATED THERMAL POWER or 1 of 2 Turbine First Stage Pressure channels greater than or equal to 37 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level. Low flow in a particular loop can be evidenced by either a detected low flow or by the opening of the reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels greater than or equal to 31% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip caused by either a low coolant flow condition in a single loop or a reactor coolant pump breaker trip on a single loop.
P-10	With 3 of 4 Power Range Neutron Flux channels less than 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1,2,3*,4*,5*
2. Power Range, Neutron Flux	S	D(2,8),M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R (6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R (6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)+	M	1, 2
8. Overpower ΔT	S	R(9)+	M	1, 2
9. Pressurizer Pressure--Low	S	R+	M	1, 2
10. Pressurizer Pressure--High	S	R+	M	1, 2
11. Pressurizer Water Level--High	S	R+	M	1, 2
12. Loss of Flow-Single Loop	S	R(8)	M	1

+ The provisions of Specification 4.0.6 are applicable.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow-Two Loops	S	R(8)	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R+	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R+	M	1, 2
16. Undervoltage-Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency-Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker				
A. Shunt Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1,2,3*,4*,5*
23. Reactor Trip Bypass Breaker	N.A.	N.A.	M(12) and S/U(1)(13)	1,2,3*,4*,5*

+The provisions of Specification 4.0.6 are applicable.

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Compare incore to excore axial imbalance above 15% of RATED THERMAL POWER. Recalibrate if absolute difference greater than or equal to 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.

(8) - The provisions of Specification 4.0.4 are not applicable.

(9) - The provisions of Specification 4.0.4 are not applicable for $f_1(\Delta I)$ and $f_2(\Delta I)$ penalties. (See also note 1 of Table 2.2-1)

(10) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip.circuit(s).

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) - Local manual shunt trip prior to placing breaker in service.

(13) - Automatic Undervoltage Trip.

(14) - The provisions of Specification 4.0.4 are not applicable when leaving MODE 1. In such an event, the calibration and/or functional test shall be performed within 24 hours after leaving MODE 1.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14*
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	14*
e. Differential Pressure Between Steam Lines - High					
Four Loops Operating	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3##	14*
Three Loops Operating	3/operating steam line	1####/steam line, any operating steam line	2/operating steam line	3##	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Flow in Two Steam Lines-High					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3##	14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line	3##	15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3##	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
OR, COINCIDENT WITH					
Steam Line Pressure-Low					
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	15
2. CONTAINMENT SPRAY					
a. Manual	2	2	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Steam Flow in Two Steam Lines--High					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3##	14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line	3##	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH EITHER T_{avg} -- Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, $3^{\#\#}$	14^{*}
Three Loops Operating	1 T_{avg}/operating loop	$1^{\#\#\#}$ T_{avg} in any operating loop	1 T_{avg} in any two operating loops	$3^{\#\#}$	15
OR, COINCIDENT WITH					
Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1, 2, $3^{\#\#}$	14^{*}
Three Loops Operating	1 pressure/operating loop	$1^{\#\#\#}$ pressure in any operating loop	1 pressure in any 2 operating loops	$3^{\#\#}$	15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

####Manually trip all bistables which would be automatically tripped in the event pressure in the associated active loop were less than the pressure in the inactive loop. For example, if loop 1 is the inactive loop, then the bistables which indicate low pressure in loops 2, 3, and 4 relative to loop 1 should be tripped.

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operations may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels greater than or equal to 1915 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 2 of 4 T_{avg} channels less than or equal to Setpoint. Setpoint greater than or equal to 541^{o}F.	P-12 allows the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T_{avg}. P-12 in coincidence with high steam flow will result in a steam line isolation. P-12 affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset value, the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T_{avg} is prevented or defeated.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq$ 14.0#/48.0##
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Recirculation Fan	$\leq$ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	$\leq$ 60.0

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
	a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
	b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
	c. Containment Pressure-High	S	R^+	M(3)	1, 2, 3
	d. Pressurizer Pressure--Low	S	R^+	M	1, 2, 3
	e. Differential Pressure Between Steam Lines--High	S	R^+	M	1, 2, 3
	f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low or Steam Line Pressure--Low	S	R^+	M	1, 2, 3
2.	CONTAINMENT SPRAY				
	a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
	b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
	c. Containment Pressure--High-High	S	R^+	M(3)	1, 2, 3

$^+$ The provisions of Specification 4.0.6 are applicable.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R+	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low Pressure--Low	S	R+	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R+	M	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R+	M	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R+	M	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R+	1, 2

+The provisions of Specification 4.0.6 are applicable.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R^{+}	M	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	1, 2, 3
8. LOSS OF POWER				
a. 4 kv Bus Loss of Voltage	S	R^{+}	M	1, 2, 3, 4
b. 4 kv Bus Degraded Voltage	S	R^{+}	M	1, 2, 3, 4

+ The provisions of Specification 4.0.6 are applicable.

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REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be O RABLE and in operation as required by items b, c, and d:

1. Reactor Coolant Loop 1 and its associated st m generator and reactor coolant pump,

2. Reactor Coolant Loop 2 and its associated st m generator and reactor coolant pump,

3. Reactor Coolant Loop 3 and its associated st m generator and reactor coolant pump,

4. Reactor Coolant Loop 4 and its associated st m generator and reactor coolant pump.

b. At least two of the above coolant loops shall be ERABLE and at least one loop in operation if the reactor tri breakers are in the open position, or the control rod drive sy em is not capable of rod withdrawal.*

c. At least three of the above coolant loops shall b OPERABLE and in operation when the reactor trip system breaker are in the closed position and the control rod drive system capable of rod withdrawal.

d. At least three of the above coolant loops shall b OPERABLE and in operation above P-12. (Refer to Technical Specification 3.3.2.1, Table 3.3-3 for instrument ion requirements.)

APPLICABILITY: MODE 3

* All reactor coolant pumps may be de-energized for up to 1 hour rovided (1) no operations are permitted that would cause dilution of the rea or coolant system boron concentration**, and (2) core outlet temperature is aintained at least 10°F below saturation temperature.

** For purposes of this specification, addition of water from th RWST does not constitute a dilution activity provided the boron concentrat n in the RWST is greater than or equal to the minimum required by specifi tion 3.1.2.8.b.2.

HOT STANDBY

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. With less than the number of operating coolant loops required by item d above, restore the required number of coolant loops within 2 hours or lower the reactor coolant system temperature below P-12.

d. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System** and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The coolant loops listed below shall be OPERABLE and in operation as required by items b and c:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal - East, **

6. Residual Heat Removal - West, **

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.***

c. At least three of the above reactor coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

APPLICABILITY: MODES 4 and 5

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 170°F unless 1) the pressurizer water volume is less than 62% of span or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration****, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System**** and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 25% of wide range instrument span at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2485 psig ±1% in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.11 Three power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. PORVs inoperable:*

1. With one PORV inoperable,

within 1 hour either restore the inoperable PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With two PORVs inoperable,

within 1 hour either restore at least one of the inoperable PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; restore at least one of the inoperable PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With three PORVs inoperable,

within 1 hour either restore at least one of the PORVs to OPERABLE status or close their associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Block valves inoperable:*

1. With one block valve inoperable,

within 1 hour either (1) restore the block valve to OPERABLE status, or (2) close the block valve and remove power from the block valve, or (3) close the associated PORV and remove power from the associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

   within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

   within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.**

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested when ACTION 3.4.11.a or 3.4.11.c is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.1.1.2.b and 4.8.2.3.2.d.**

* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.
**The provisions of Specification 4.0.6 are applicable.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the reactor trip breakers are opened; otherwise, be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 4.7-1, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87.2
2	65.4
3	43.6

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TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER		LIFT SETTING (± 1%)*	ORIFICE SIZE
a.	SV-1	1065 psig	16 $in.^2$
b.	SV-1	1065 psig	16 $in.^2$
c.	SV-2	1075 psig	16 $in.^2$
d.	SV-2	1075 psig	16 $in.^2$
e.	SV-3	1085 psig	16 $in.^2$

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops an equivalent discharge pressure of greater than or equal to 1240 psig at 60°F on recirculation flow.

2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of greater than or equal to 1180 psig at 60°F and at a flow of greater than or equal to 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

STEAM GENERATOR STOP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each steam generator stop valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one steam generator stop valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one steam generator stop valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The stop valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each steam generator stop valve that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 5 seconds on any closure actuation signal while in HOT STANDBY with T_{avg} greater than or equal to 541°F during each reactor shutdown except that verification of full closure within 5 seconds need not be determined more often than once per 92 days.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.5.3 The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 when performing PHYSICS TESTS at the beginning of a cycle provided the steam generator stop valves are maintained closed.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. A day tank containing a minimum of 70 gallons of fuel,

   2. A fuel storage system containing a minimum of 42,000 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.6.**

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

**The provisions of Specification 4.0.6 are applicable.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes** and initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

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* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

** For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.* The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per 24 hours.

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the region of acceptable operation shown on Figures 3.4-2 and 3.4-3.

APPLICABILITY: MODE 2

ACTION:

a. With the THERMAL POWER greater than 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figures 3.4-2 and 3.4-3, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figures 3.4-2 and 3.4-3 at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour.

SURVEILLANCE REQUIREMENTS (Continued)

4.10.3.3 Each Intermediate and Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4 and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

NATURAL CIRCULATION TESTS

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.4.1.1 may be suspended during the performance of PHYSICS TESTS and Thermal-Hydraulic Tests, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during PHYSICS TESTS.

4.10.5.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

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 for increased load events 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 1.6% $\Delta k/k$ is initially required to control the reactivity transient and automatic ESF is assumed to be available. 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 for this event.

In shutdown MODES 4 and 5 when heat removal is provided by the residual heat removal system, active reactor coolant system volume may be reduced. Increased SHUTDOWN MARGIN requirements when operating under these conditions is provided for high reactor coolant system boron concentrations to ensure sufficient time for operator response in the event of a boron dilution transient.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 ± 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective 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 pressure vessel is above its minimum RT_{NDT} temperature. Administrative procedures will be established to ensure the P-12 blocked functions are unblocked before taking the reactor critical.

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 separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 170°F, unless the reactor vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from all operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability, usable volume requirement, is 5641 gallons of 20,000 ppm borated water from the boric acid storage tanks or 99,598 gallons of 2400 ppm borated water from the refueling water storage tank. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5.

BORATION SYSTEMS (Continued)

With the RCS average temperature below 200°F, one 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 change in the event the single injection system becomes inoperable.

The boration capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 2890 gallons of 20,000 ppm borated water from the boric acid storage tanks or 76,937 gallons of 2400 ppm borated water from the refueling water storage tank. The boration source volumes of Technical Specification 3.1.2.7 have been conservatively increased to 4300 gallons from the boric acid storage tank and 90,000 gallons from the RWST. These values were chosen to be consistent with Unit 2. The Unit 2 value for the boric acid storage tank volume includes sufficient boric acid to borate to 2000 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 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 (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. 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 ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis for a rod ejection accident.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} greater than or equal to 541°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.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.69 during normal operation and in short term transients, and (b) 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 hot channel 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 (AFD)

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.

Although ~~it~~ is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels above 50% of RATED THERMAL POWER. For THERMAL POWER levels below 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less). During operation at THERMAL POWER levels between 15% and 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less), the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

The upper bound limit (90% or 0.9 x APL of RATED THERMAL POWER (whichever is less)) on AXIAL FLUX DIFFERENCE assures that the $F_Q(Z)$ envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The lower bound limit (50% of RATED THERMAL POWER) is based on the fact that at THERMAL POWER levels below 50% of RATED THERMAL POWER, the average linear heat generation rate is half of its nominal operating value and below that value, perturbations in localized flux distributions cannot affect the results of ECCS or DNBR analyses in a manner which would adversely affect the health and safety of the public.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

The bases and methodology for establishing these limits is presented in topical report WCAP - 8385, "Power Distribution Control and Load Following Procedures."

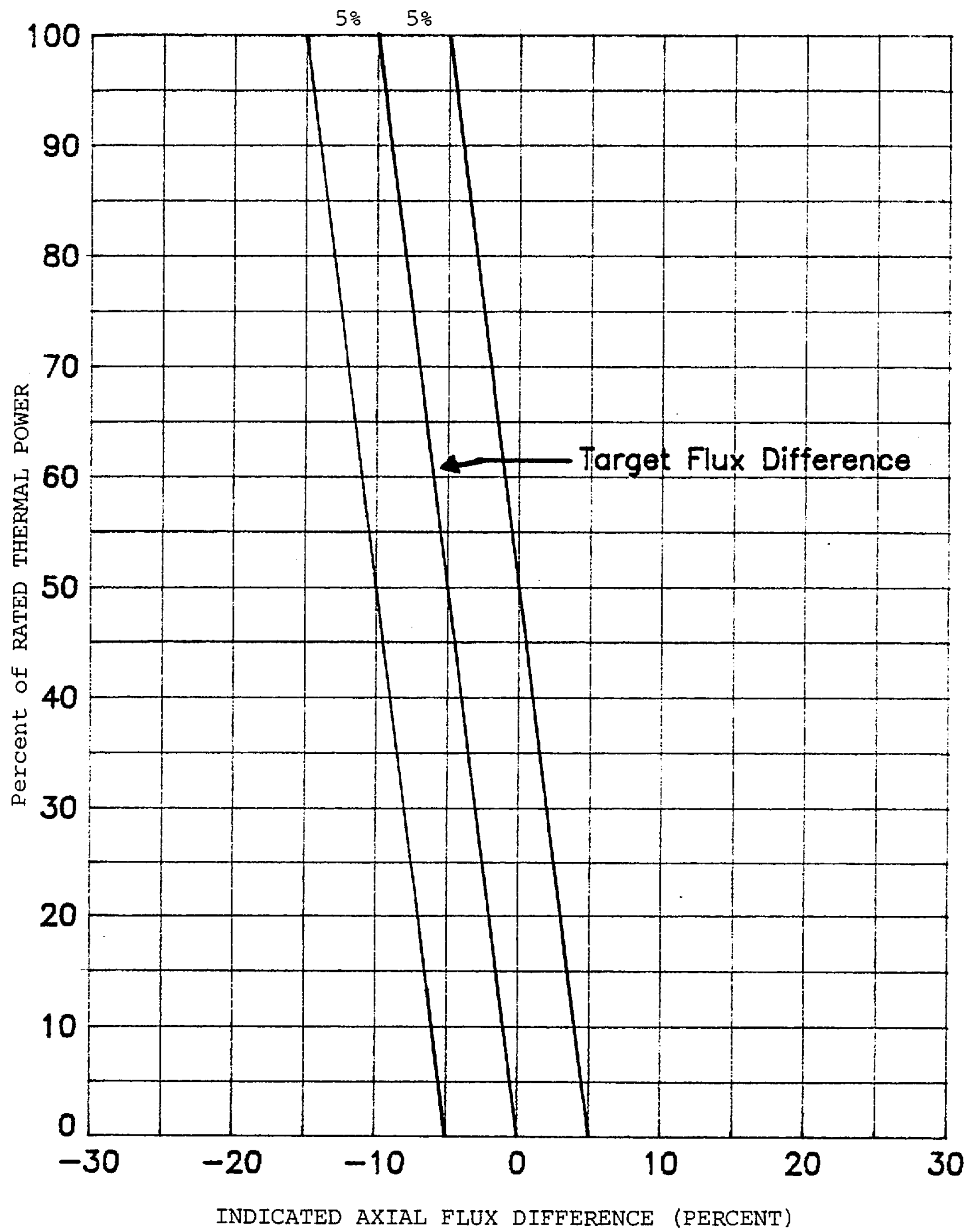

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTORS

The limits on heat flux and nuclear enthalpy rise hot channel factors 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.

Each of these hot channel factors are measurable, but will normally only be determined periodically, as specified in Specifications 4.2.2.1, 4.2.2.2, 4.2.3, 4.2.6.1 and 4.2.6.2. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits, provided conditions (a) through (d) above are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system, and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for, and 4% is the appropriate allowance for a full core map taken with the incore detection system. This 4% measurement uncertainty has been included in the design DNBR limit value. The specified limit for $F^N_{\Delta H}$ also contains an additional 4% allowance for uncertainties. The total allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, affect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup PHYSICS TESTS can be compensated for in F_Q, by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02, but less than 1.09, is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated to be adequate to maintain the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient. The indicated values of T_{avg} and flow include allowances for instrument errors. Measurement uncertainties have been accounted for in determining the DNB parameters' limit values.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the 12-hour surveillance of the RCS flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperature.

3/4.2.6 ALLOWABLE POWER LEVEL - APL

Constant Axial Offset Control (CAOC) operation manages core power distributions such that Technical Specification limits on $F_Q(Z)$ are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients. The V(Z) factor given in the Peaking Factor Limit Report and applied by the Technical Specifications provides the means for predicting the maximum $F_Q(Z)$ distribution anticipated during operation using CAOC taking into account the incore measured equilibrium power distribution. A comparison of the maximum $F_Q(Z)$ with the Technical Specification limit determines the power level (APL) below which the Technical Specification limit can be protected by CAOC. This comparison is done by calculating APL, as defined in specification 3.2.6.

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3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.69 during all normal operations and anticipated transients. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 170°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCP's to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation conditions.

3/4.4.11 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.12 REACTOR COOLANT VENT SYSTEM

The Reactor Coolant Vent System is provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. It has been designed to vent a volume of Hydrogen approximately equal to one-half of the Reactor Coolant System volume in one hour at system design pressure and temperature.

The Reactor Coolant Vent System is comprised of the Reactor Vessel head vent system and the pressurizer steam space vent system. Each of these subsystems consists of a single line containing a common manual isolation valve inside containment, splitting into two parallel flow paths. Each flow path provides the design basis venting capacity and contains two 1E DC powered solenoid isolation valves, which will fail closed. This valve configuration/redundancy serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a remotely-operated vent valve, power supply, or control system does not prevent isolation of the vent path. The pressurizer steam space vent is independent of the PORVs and safety valves and is specifically designed to exhaust gases from the pressurizer in a very high radiation environment. In addition, the OPERABILITY of one Reactor Vessel head vent path and one Pressurizer steam space vent path will ensure that the capability exists to perform this venting function.

The function, capabilities, and testing requirements of the Reactor Coolant Vent System are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirement," November 1980.

The minimum required systems to meet the Specification and not enter into an action statement are one vent path from the Reactor Vessel head and one vent path from the Pressurizer steam space.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. The value of the minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to 80°F to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure from containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is approximately 121 percent of the total secondary steam flow of 14,120,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line = 1, 2 or 3.

X = Total relieving capacity of all safety valves per steam line = 4,288,450 lbs/hour.

Y = Maximum relieving capacity of any one safety valve = 857,690 lbs/hour.

(109) = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1065 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1065 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

The acceptance discharge pressures for the auxiliary feedwater pumps are based on a fluid temperature of 60°F. Water density corrections are permitted to allow comparison of test results which vary depending on ambient conditions.

In addition to its safety design function, the AFW system is used to maintain steam generator level during startup (including low power operation). During this time, the system design allows for automatic initiation of the auxiliary feedwater pumps and their related automatic valves in the flow path.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The boron concentration requirement of specification 3.9.1.b has been conservatively increased to 2400 ppm to agree with the minimum concentration of the RWST.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 107
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated March 26, 1987, as clarified and supplemented by letters dated August 25, 1987, January 13, 1988, June 7, 1988 and August 31, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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;

   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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 107 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance. The Technical Specifications are to become effective within 120 days of receipt of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 9, 1989

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO 107 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO.50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT	REMOVE	INSERT
Index in its entirety	Index pages I - XVIII	3/4 3-11 thru 3-13	3/4 3-11 thru 3-13
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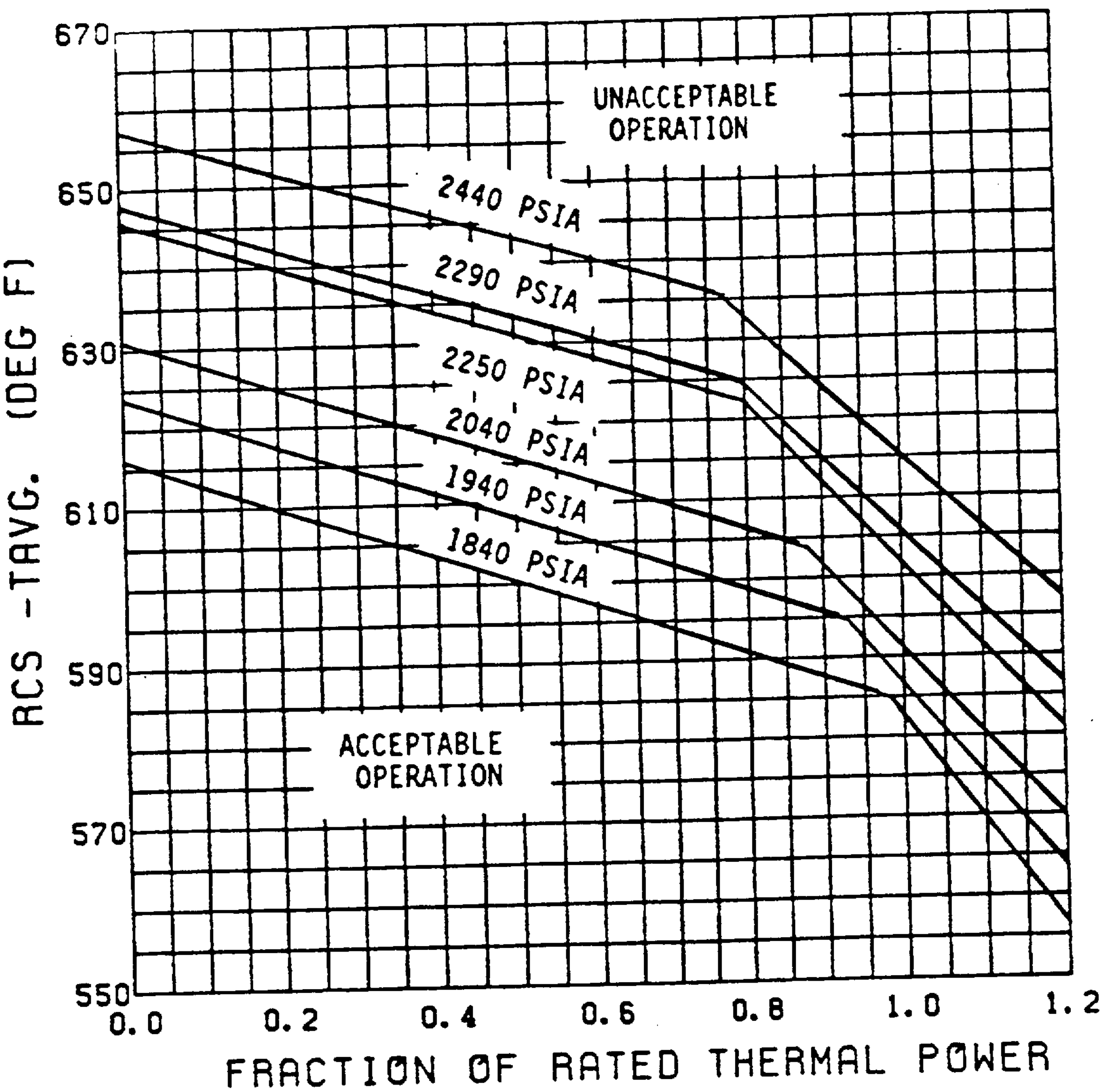


PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T AVG, DEG F)
1840	(0.00,616.2) , (0.98,585.1) , (1.20,556.5)
1940	(0.00,623.8) , (0.93,594.7) , (1.20,563.5)
2040	(0.00,631.0) , (0.88,603.8) , (1.20,569.6)
2250	(0.00,645.9) , (0.80,622.3) , (1.20,580.9)
2290	(0.00,647.9) , (0.80,624.5) , (1.20,586.5)
2440	(0.00,657.4) , (0.77,635.6) , (1.20,597.2)

FIGURE 2.1-1 Reactor Core Safety Limits -
Four Loops in Operation

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

a. In MODE 4:

1. Greater than or equal to 1.6% $\Delta k/k$ when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

b. In MODE 5:

1. Greater than or equal to 1.0% $\Delta k/k$ when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

APPLICABILITY: MODES 4 and 5

ACTION:

With SHUTDOWN MARGIN less than the above limits, immediately initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the above limits:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be greater than or equal to 2000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.*

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system less than 2000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be greater than or equal to 2000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying greater than or equal to 2000 gpm through the reactor coolant system.

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the region of acceptable operation in Figure 3.1-2, and

b. Less negative than -3.9×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a above:

1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limits 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. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report 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.

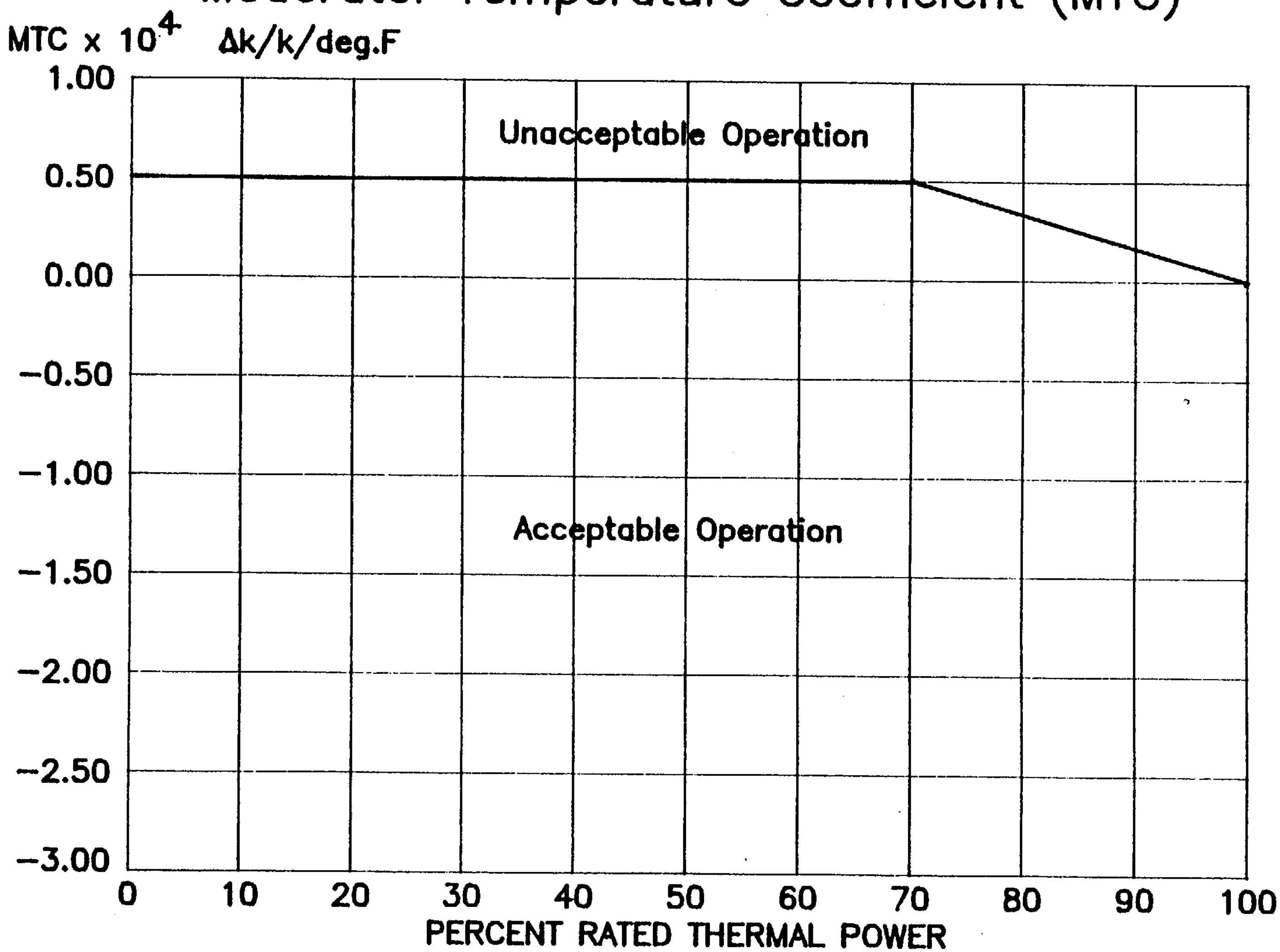
b. With the MTC more negative than the limit of 3.1.1.4.b above, be in HOT SHUTDOWN within 12 hours.

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

See Special Test Exception 3.10.3

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 541°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With $K_{eff} \geq 1.0$.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 145°F when a flow path from the boric acid tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.*

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that, on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2390 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 152°F.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

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 inoperable or 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 inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The affected rod is restored to OPERABLE status within the above alignment requirements, or

   2. The affected 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, and

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

*See Special Test Exceptions 3.10.2 and 3.10.3

LIMITING CONDITION FOR OPERATION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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 shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (228 steps) shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to entering MODE 2:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn (228 steps).

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod, 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 fully withdrawn:

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

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, 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.

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (±5% or ±3% flux difference units) about a target flux difference.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. Above 90% of 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, 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) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

FIGURE 3.2-1 ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE

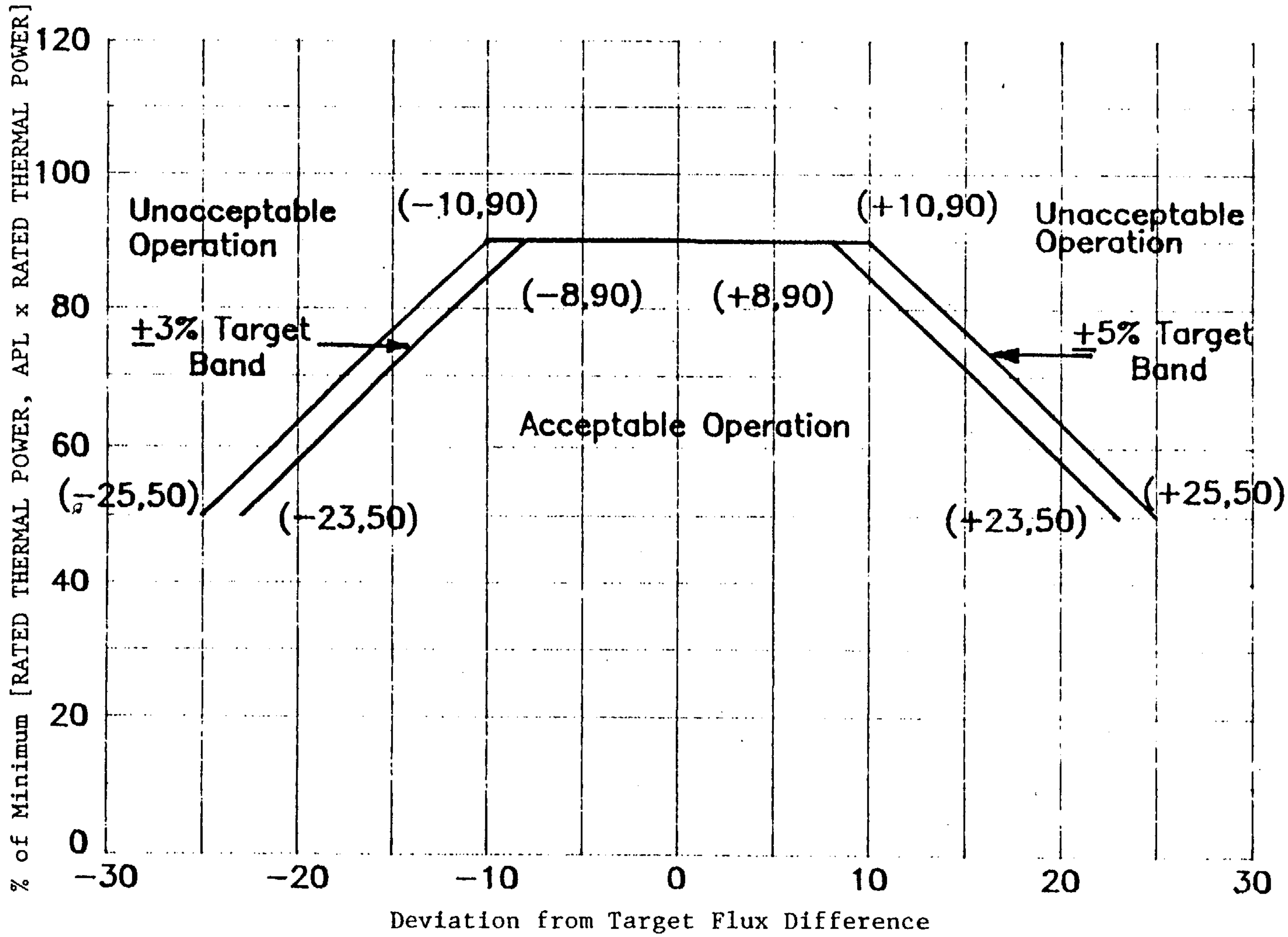

TABLE 3.2-1

DNB PARAMETERS

LIMITS

PARAMETER	4 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 576.3°F. (indicated)**
Pressurizer Pressure	$\geq$ 2205 psig*, **
Reactor Coolant System Total Flow Rate	$\geq 138.6 \times 10^6$ lbs/hr***

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

** Indicated average of at least three OPERABLE instrument loops.

*** 3.5% penalty for measurement uncertainty included in this value.

TABLE 3.2-2

DNB PARAMETERS

PARAMETER	LIMIT
Reactor Coolant System T_{avg}	$\leq$ 549.2°F. (Reactor Subcritical)*
Reactor Coolant System T_{avg}	$\leq$ 576.3°F. (Reactor Critical)*
Pressurizer Pressure	$\geq$ 2176 psig*

Reactor coolant loop operational requirements are contained in Specifications 3.4.1.1, 3.4.1.2.c and 3.4.1.3.c.

* Indicated average of at least three OPERABLE instrument loops.

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

3.2.6 THERMAL POWER shall be less than or equal to ALLOWABLE POWER LEVEL (APL), given by the following relationships:

Westinghouse Fuel

$$APL = \text{min over Z of } \frac{1.97\ K(Z)}{F_Q(Z) x V(Z) x F_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

Exxon Nuclear Co. Fuel

$$APL = \text{min over Z of } \frac{2.10\ K(Z)}{F_Q(Z) x V(Z) x F_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

- $F_Q(Z)$ is the measured hot channel factor, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- V(Z) is the function defined in Figure 3.2-3 which corresponds to the target band.
- F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in peak pin power, $F_{\Delta H}$, with exposure. Then either of the following penalties, F_p, shall be taken:

  F_p = 1.02 or,

  F_p = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until 2 successive maps indicate that the peak pin $F_{\Delta H}$ is not increasing.
- The above limit is not applicable in the following core regions.

  1) Lower core region 0% to 10% inclusive.
  2) Upper core region 90% to 100% inclusive.

APPLICABILITY: MODE 1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With THERMAL POWER exceeding APL:

a. Reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes. Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER 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 ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER.

b. THERMAL POWER may be increased to a new APL calculated at the reduced power by either redefining the target axial flux difference or by correcting the cause of the high $F_Q(Z)$ condition.

SURVEILLANCE REQUIREMENTS

4.2.6.1 The provisions of Specification 4.0.4 are not applicable.

4.2.6.2 APL shall be determined by measurement in conjunction with the target flux difference and target band determination* above 15% of RATED THERMAL POWER, according to the following schedule:

a. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which APL was last determined**, or

b. At least once per 31 effective full power days, whichever occurs first.

*APL can be redefined by remeasuring the target axial flux difference in accordance with ACTION statement b of Specification 3.2.6.

**During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	3	1	6#
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10#
B. Above P-7 and below P-8	1/breaker	2	1/breaker per operating loop	1	11#
21. Reactor Trip Breakers	2	1	2	1, 2 3*,4*,5*	1, 13 14
22. Automatic Trip Logic	2	1	2	1, 2 3*,4*,5*	1 14

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Pressure Before the First Stage channels ≥ 51 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level. Low flow in a particular loop can be evidenced by either a detected low flow or by the opening of the reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels ≥ 31% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip caused by either a low coolant flow condition in a single loop or a reactor coolant pump breaker trip on a single loop.
P-10	With 3 of 4 Power Range Neutron Flux channels < 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2,8),M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)	M	1, 2
8. Overpower ΔT	S	R(9)	M	1, 2
9. Pressurizer Pressure--Low	S	R	M	1, 2
10. Pressurizer Pressure--High	S	R	M	1, 2
11. Pressurizer Water Level--High	S	R	M	1, 2
12. Loss of Flow - Single Loop	S	R(8)	M	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow-Two Loops	S	R(8)	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage-Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency-Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker				
A. Shunt Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1,2,3*,4*,5*
23. Reactor Trip Bypass Breaker	N.A.	N.A.	M(12) and S/U(1)(13)	1,2,3*,4*,5*

TABLE 4.3-1 (Continued)

NOTATION

*	-	With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.
(1)	-	If not performed in previous 7 days.
(2)	-	Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference > 2 percent.
(3)	-	Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.
(4)	-	Manual ESF functional input check every 18 months.
(5)	-	Each train tested every other month.
(6)	-	Neutron detectors may be excluded from CHANNEL CALIBRATION.
(7)	-	Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.
(8)	-	The provisions of Specification 4.0.4 are not applicable.
(9)	-	The provisions of Specification 4.0.4 are not applicable for $f_1(\Delta I)$ and $f_2(\Delta I)$ penalties. (See also note 1 of Table 2.2-1)
(10)	-	The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).
(11)	-	The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.
(12)	-	Local manual shunt trip prior to placing breaker in service.
(13)	-	Automatic Undervoltage Trip.
(14)	-	The provisions of Specification 4.0.4 are not applicable when leaving MODE 1. In such an event, the calibration and/or functional test shall be performed within 24 hours after leaving MODE 1.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	≤ 12.0#/24.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
d. Containment Air Recirculation Fan	≤ 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Not Applicable
b. Feedwater Isolation	Not Applicable
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be OPERABLE and in operation as required by items b, c, and d:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.*

c. At least three of the above coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

d. At least three of the above coolant loops shall be OPERABLE and in operation above P-12. (Refer to Technical Specification 3.3.2.1, Table 3.3-3 for instrumentation requirements.)

APPLICABILITY: MODE 3

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* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration**, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2.

HOT STANDBY

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. With less than the number of operating coolant loops required by item d above, restore the required number of coolant loops within 2 hours or lower the reactor coolant system temperature below P-12.

d. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System** and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The coolant loops listed below shall be OPERABLE and in operation as required by items b and c:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal - East, **

6. Residual Heat Removal - West **

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.***

c. At least three of the above reactor coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

APPLICABILITY: MODES 4 and 5

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 152°F unless 1) the pressurizer water volume is less than 62% of span or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration****, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System**** and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 25% of wide range instrument span at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electrical power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

LIMITING CONDITION FOR OPERATION

3.4.11 Three power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. PORVs inoperable:*

1. With one PORV inoperable,

within 1 hour either restore the inoperable PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With two PORVs inoperable,

within 1 hour either restore at least one of the inoperable PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; restore at least one of the inoperable PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With three PORVs inoperable,

within 1 hour either restore at least one of the PORVs to OPERABLE status or close their associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Block valves inoperable:*

1. With one block valve inoperable,

within 1 hour either (1) restore the block valve to OPERABLE status, or (2) close the block valve and remove power from the block valve, or (3) close the associated PORV and remove power from the associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

ELECTRICAL POWER SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. A day fuel tank containing a minimum volume of 70 gallons of fuel,

2. A fuel storage system containing a minimum volume of 42,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.*

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for Requirement 4.8.1.1.2.a.5.**

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* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

**The provisions of Specifications 4.0.6 and 4.0.7 are applicable.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes** and initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.* The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per 24 hours.

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating start up or PHYSICS TESTS.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F, unless the reactor vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability usable volume requirement is 3700 gallons of 20,000 ppm borated water from the boric acid storage tanks or 118,000 gallons of borated water from the refueling water storage tank. The required RWST volume is based on an assumed boron concentration of 2000 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5. The boration source volume from the boric acid storage tank has conservatively been increased to 5650 gallons. This value was chosen to be consistent with Unit 1.

With the RCS temperature below 200°F, one 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 change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 4300 gallons of 20,000 ppm borated water from the boric acid storage tanks or 90,000 gallons of borated water from the refueling water storage tank. The value for the boric acid storage tank volume includes sufficient boric acid to borate to 2000 ppm. The required RWST volume is based on an assumed boron concentration of 2000 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 mechnical systems and components.

The OPERABILITY of boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

BASES

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE in MODES 4 and 5, an operating RHR loop, connected to the RCS, provides overpressure relief capability. Additionally, if no safety valves are OPERABLE, then all Safety Injection pumps and all but one charging pump will be rendered inoperable to preclude overpressurization due to an inadvertent increase in the RCS inventory.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 80°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The boron concentration requirement of specification 3.9.1.b has been conservatively increased to 2400 ppm to agree with the minimum concentration of the RWST.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference Specifications:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Unit No. 1, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Unit No. 1, Specification 3.3.3.4.

d. Fire Detection Instrumentation, Specification 3.3.3.8.

e. Fire Suppression Systems, Specifications 3.7.9.1, 3.7.9.2, 3.7.9.3 and 3.7.9.4.

f. Seismic Event Analysis, Specification 4.3.3.3.2.

g. Sealed Source leakage in excess of limits, Specification 4.7.8.1.3.

h. Moderator Temperature Coefficient, Specification 3.1.1.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.120 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO.107 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated March 26, 1987, the Indiana Michigan Electric Company (IMEC the licensee) submitted proposed changes to the Technical Specifications (TSs) for D. C. Cook Units 1 and 2. The changes were intended to ensure the TSs reflected the analysis requirements, and to make the TSs more consistent. The licensee has categorized the changes into 12 separate groups (see Attatchment 1 to base document of March 26, 1987). The staff evaluation discussed in Section 2 of this Safety Evaluation (SE) is organized based on the same categorization.

By letter dated August 25, 1987, the licensee further clarified proposed changes addressed in part by Amendment 111 and 94 issued on June 10 1987 for Units 1 and 2, respectively. The first of the clarified changes was related to the moderator temperature coefficients for Unit 2. The licensee proposed to replace the fixed value of the most negative MTC with a curve of permitted values as a function of core power in TS 3/4.1.1.4. Although the licensee proposed the same change for both units in the March 26, 1987 submittal, only the Unit 1 change was granted. The SE for Amendments 111 and 94 erroneously indicated that the change had been previously granted for Unit 2. For further discussion see section 2.8 of this SE. The other clarification addressed in the licensee's August 25, 1987 letter involved an increase in the required Mode 6 boron concentration in TS 3.9.1. The licensee had proposed to increase the required concentration from 2000 ppm to 2400 ppm for both units. The SE for Amendment 111/94 indicated that the changes to TS 3.9.1 were acceptable for both units. However, the Unit 1 page was inadvertently omitted form the NRC's transmittals of the amendment. Therefore, the changes to Unit 1 TS 3.9.1 is being issued at this time.

By letter dated August 31, 1988, the licensee proposed changes to the original submittal in response to NRC telephone requests of April 27, 1987 and August 9, 1988. The changes revised the licensee's submittal with regard to TS 4.0.4 exemptions and with regard to the description of various interlocks described in the TSs. On January 13, 1988, the licensee provided the NRC Project Manager with replacement pages with corrected typographical errors in the original submittal. In both of these cases, the changes were either more restrictive than originally proposed or editorial in nature. The changes did not substantially alter the original submittal and thus did not affect the staff's initial determination of no significant hazards consideration.

By letter dated June 7, 1988, the licensee withdrew a portion of their proposed changes related to requirements to submit a peaking factor limit report each cycle. The withdrawal was requested by the NRC since NRC requirements with regard to the report are in the process of being reevaluated.

2.0 EVALUATION

The following discussion and evaluation is organized in the same form as that used by the licensee in Attachments 1 and 3 of the base document. Due to the large number of proposed changes, specific reference to the relevant portions of these documents in this SE is limited to the 12 broad categories. Where appropriate, reference to the proposed TS page changes (see Attachment 2 to the base document) is made to supplement the evaluation.

2.1 Editorial Changes

The licensee has proposed a number of changes which have been characterized as purely editorial in nature. These changes are listed in the attachments to the licensee's submittal in sufficient detail that listing them again in this evaluation is not deemed necessary. The staff has reviewed those changes identified by the licensee as editorial and has confirmed that the changes are administrative or provide consistency between the Unit 1 and Unit 2 TSs, are not of major safety significance, and are, therefore, acceptable.

2.2 Removal of 3-Loop Technical Specifications

The licensee has proposed to remove the 3-loop TSs associated with Operational Modes 1 and 2 for Unit No. 1 in order to simplify the documentation supplied to the operators. All references to extended 3-loop operation in Modes 1 and 2 have been deleted. Removal of these TSs will not affect operation and are, therefore, acceptable. Similar changes have been previously made for Unit No. 2 and were approved in Amendment 82 to Facility Operating License No. DPR-74 dated May 21, 1986; the present approval for Unit No. 1 thus makes the TSs consistent between the units.

Removal of 3-loop (Modes 1 and 2) TSs is discussed on pages 3-5 of Attachment 1 or the licensee's March 26, 1987 submittal.

In summary, license conditions for the Cook plant prohibited operation with only 3 loops above the P-7 interlock setpoint (approximately 11% of rated power). In practice, the Cook plant chose not to operate in Modes 1 or 2 with only 3 loops operable. Therefore, TSs regarding 3-loop operation in Modes 1 and 2 were unnecessary. The licensee has proposed to delete the subject TSs in an effort to streamline the document. The change places additional restrictions on the plant by prohibiting operation with less than the full compliment of reactor coolant pumps operating. Thus, increased margin to DNB under accident conditions should result and the change is acceptable.

2.3 Additional Restrictions Because of Safety Analyses

Specifcation 3.1.3.1 - Action C.2a -

The number of events requiring reanalysis in order to permit continued operations of Unit 1 with a stuck rod is increased from 1 to 7. This makes the requirement compatible with that of Unit 2, is conservative with respect to the present specification, and is acceptable.

Specifications 3.1.3.1 - Action C.2.c -

This action statement is added to the Specification to make it compatible with the Unit 2 Specifications and the Standard Technical Specifications. Additional power distribution maps are required when operating with a stuck rod. This provides additional assurance that unacceptable power distributions will not occur and is acceptable.

Specification 4.1.3.3 (Unit 1)/4.1.3.4 (Unit 2) -

The revised Specification 4.1.3.3 would require the completion of rod drop time testing prior to entering Mode 2 rather than prior to achieving criticality. This is a conservative change and use of a mode-dependent requirement eases administrative control. Therefore, the staff finds the proposed revisions to be acceptable.

Specifications 4.2.5.2 and 4.2.5.3 -

Requirements for an RCS total flow measurement and channel calibration every 18 months are added. This provides additional assurance of operation within the thermal-hydraulic envelope and is acceptable.

Table 3.3-1, Item 2 and Table 4.3-1, Item 2 -

This item requires that the Power Range Neutron Flux Functional Unit be operational whenever it is possible to withdraw a control rod. This is consistent with the analysis of the zero power rod withdrawal event and is acceptable. This change also provides consistency between the Units 1 and 2 Specifications. An additional Channel Functional Test is also required.

Specifications 3.4.1.2 and 3.4.1.3 -

These Specifications require the reactor coolant pumps to be operable when withdrawal of a control rod at zero power is possible. The safety analysis requirement for Unit 1 is that 2 pumps be running when the rods are capable of withdrawal. However, the Unit 2 safety analysis required 3 pumps to be running under the same conditions. Changes to the Unit 2 TSs, to require 3 pumps running were approved via Amendment 82 to the Unit TSs, dated May 21, 1986. Although the Unit 1 analysis only requires 2 pumps, 3 pumps running was proposed by the licensee to achieve consistency with the Unit 2 TSs. The Unit 1 changes are conservative with respect to the safety analysis and are therefore acceptable.

Specification 3.4.2, Action b -

This action statement was added to the Unit 2 TSs via Amendment 82, dated May 21, 1986, as a result of an analysis of the overpressurization event. Since Unit 1 is similar to Unit 2, its addition to the Unit 1 Specifications is required and is acceptable.

Specification 3.7.1.5 -

The action statements are revised to prohibit operation in Mode 1 with an inoperable steam generator stop valve and to clarify other mode assignments. The revision, which is more restrictive, makes Donald C. Cook Specifications consistent with Standard Technical Specifications and is acceptable.

2.4 Refueling Water Tank Changes

The licensee has added footnotes to appropriate TSs to indicate that additions to the reactor coolant system from the RWST do not constitute a positive reactivity addition (boron dilution event). The RWST boron concentration is set to provide adequate shutdown margin; therefore, these footnotes for clarification are acceptable.

The TS LCO and Bases changes for revising the RWST boron concentration are discussed in Section 2.8 of this evaluation.

2.5 Changes to Differential Pressure Between Steam Lines - High ESF Actuation Signal

The licensee proposed to change a footnote on Table 3.3-3 to clarify the actions to be taken during Mode 3 when only three steam generators are operable. The Differential Pressure Between Steam Line-High actuation of the Engineered Safety Feature (ESF) depends on the comparison of pressures between steam lines. In the above changes to remove 3-loop TSs, the licensee also removed 3 Steam Generator Loop operation from Modes 1 and 2. Three steam generator loop operation is allowed in Mode 3; however, with one of these loops out, the comparison of pressure between steam lines will produce half of the required ESF signal. The current footnote can be misinterpreted as providing that, under the condition of 3 steam loop operation, all the channels on the inoperable loop should be place in the tripped mode. This would create an ESF actuation signal when none was required. The proposed change clarifies the TS requirement and is acceptable.

The licensee proposed to change the footnote designator in the Channels to Trip column of the ESF Actuation Instrumentation Table 3.3-3 and add a corresponding footnote which specifies that the channel on each of the operable loops which indicates high differential pressure with respect to the idle loops be placed in the tripped mode. This action reduces the ESF actuation logic for the active loop differential pressure from 2 out of 3 to 1 out of 2 and thus permits 3-loop operation in Mode 3 since 2 channels per steam line are necessary for a trip.

This change makes the TS for Unit 1 consistent with the previously approved change for Unit 2, and is acceptable.

2.6 Change to Power Operated Relief Valve (PORV) Specification

The licensee proposed to change TS 3/4.4.11 to require that at least two PORV's be available in Operational Modes 1, 2, and 3. This is the same change previously reviewed and approved for Unit No. 2 in Amendment 82 to Facility Operating License No. DPR-74 and is, therefore, acceptable for the present proposal.

2.7 Addition of TS 4.0.4/3.0.4 Exemptions

The licensee proposed to add exemptions to existing TSs to allow operational mode changes under certain specified conditions. With the exception of the exemption related to loss of flow in two loops for Unit No. 2 and the source range neutron flux channel calibration, the changes proposed were reviewed and approved for Unit No. 2 in Amendment 82 to Facility Operating License No. DPR-74 and are acceptable for the present proposal.

The TS 4.0.4 exemption related to loss of flow in two loops had been previously proposed by the licensee for only Unit 2 in their letter No. AEP:NRC:0916I, dated March 14, 1986. The changes had been described in the licensee's submittal, and were reviewed and found acceptable by the NRC in Amendment 82. However, because the licensee inadvertently left the change page out of their submittal, the change was not issued by the NRC in Amendment 82, which transmitted the other changes to the Unit 2 TSs requested by the licensee's March 14, 1986 submittal. The Unit 1 change requested in the licensee's March 20, 1987 submittal is identical to the Unit 2 change which was reviewed previously and found acceptable. On this basis, the change is being approved for Unit 1 and reapproved for Unit 2.

Table 4.3-1, Item 6, requires the Source Range Neutron Flux Channel Calibration at least once per 18 months (R) and the monthly (M) channel functional test. The licensee's proposed TS 4.0.4 exemption is to address the situation where the plant remains in Mode 1 for longer than 18 months. The staff found the change acceptable; however, the wording of the original request would allow entry from shutdown conditions such as Mode 3, hot standby, to startup without the minimum channels operable due to lapsed surveillance. A change was proposed to the licensee, that in such an event the source range calibration and functional test should be performed within 24 hours of leaving Mode 1. The licensee agreed to the NRC's proposed changes with regard to the source range neutron flux calibration TS 4.0.4 exemption. The licensee transmitted reworded revisions of the Table 4.3-1 notation on August 31, 1988. The licensee's suggested wording is acceptable. On the same table functional descriptions, page 3/4 3-9, a description of the P-7 and P-8 interlocks on the reactor coolant pump breakers position trip has been included. The licensee's suggested wording, which was also transmitted on August 31, 1988, is acceptable. Because the changes requested by the NRC and incorporated into the licensee's August 31, 1988, letter are in all cases either more restrictive or administrative in nature, it was concluded that the changes did not require renotice in the Federal Register.

2.8 Changes to Existing TS Values

Shutdown Margin

In Section 3/4.1.1.2, additional material is proposed to define the LCO for shutdown margin in Operational Modes 4 and 5. Changes are proposed as a result of a new analysis of boron dilution event for Modes 4 and 5. In both modes, the requirements on operability of Reactor Coolant Loops and Residual Heat Removal Loops were altered to be consistent with the new analysis. An equivalent change was previously approved for Unit 2 in Amendment 82 to Facility Operating License No. DPR-74, and the present proposal is acceptable on the same basis.

Reactor Coolant Flow

In Section 3/4.1.1.3, a reduction in the minimum reactor coolant system flow rate from 3000 gpm to 2000 gpm is proposed. The change is in recognition of the potential for RHR vortexing and air entrainment at the higher flow rate. The reduction in flow rate requires consideration and reevaluation of the boron dilution event and decay heat removal capability. The licensee has provided these analyses in Attachments 5 and 14 of the base submittal document (AEP:NRC: 0916W). The staff has reviewed the submitted material and has concluded that adequate consideration has been provided to justify the proposed change. Some editorial changes were also made in these sections which are also acceptable. The change to 2000 gpm is further identified in TS Surveillance Requirement 4.9.8.1 which is also acceptable.

In Section 3/4.2.5, the Surveillance Requirements are revised to add channel calibration and reactor coolant flow measurement once per 18 months. In addition monthly RCS flow surveillance is removed since it is redundant to a 12-hour surveillance in existence in the same Specification. Equivalent change were proposed and approved in Amendment 82 to Facility Operating License No. DPR-74 and are equally acceptable for the present proposal.

Delta-T Span

In Table 2.2-1, Notes 3 and 4, the words "delta-T span" are inserted after the uncertainty values in these notes. The revised designation is consistent with the setpoint analysis and is acceptable.

Moderator Temperature Coefficient

In the licensee's June 10, 1987 submittal, it was proposed to modify the Moderator Temperature Coefficient (MTC) curve (TS 3/4.1.1.4). The proposed modification replaced the fixed value of the most negative MTC with a curve of permitted values as a function of core power. The curve would show a constant value of +5 pcm/deg F for core power from 0% to 70% of full power. Above 70%, the allowed value would decrease linearly to 0 pcm/deg F at full power.

The current TS has two values of permitted MTC. Up to 70% of full power, +5 pcm/deg F is permitted, and above that value the MTC must be zero or negative. The licensee provided information form the fuel vendors which indicated that the safety analysis assumptions remain valid. Thus, the staff finds the proposed linear reduction is bounded by previously approved safety analyses and is therefore acceptable.

The licensee's March 26, 1987 letter proposed the change in the MTC function for both units. The change for Unit 1 was granted via Amendment 111 to the Unit 1 TSs. The SE for that amendment erroneously indicated that the change had been previously approved for Unit 2. In reality, the change had not been previously approved and in fact was proposed for the first time by the licensee in the March 26, 1987 submittal. The proposed change has been found acceptable for Unit 2, and is therefore being issued at this time.

RWST and BAST Boron

For the proposed modification to Specifications 3/4.1.2.7, 3/4.1.2.8, 3/4.5.1 and 3/4.5.5, the changes to the minimum volumes and boron concentrations for the RWST and BAST were previously found acceptable in the SE supporting the June 10, 1987 Amendments. However, only Unit 2 TSs were changed for boron concentrations. The evaluation was performed for both units, therefore, the changes to Unit 1 TSs are acceptable.

Auxiliary Feedwater

In Section 4.7.1.2, the surveillance requirements on auxiliary feedwater pumps are changed to provide assurance that these pumps can deliver sufficient flow to fulfill the requirements of the safety analyses for the plant. A similar change was previously made and accepted for Unit 2 and is likewise acceptable for Unit 1.

DNB Parameters Error Allowances

Footnotes are added to Table 3.2-1 for the Unit 2 TSs to document the allowance for flow measurement error in the development of the Reactor Coolant System analysis which was submitted by the licensee as part of the information for Amendment 82 to the Unit 2 TSs. Also, readability allowances are included as a footnote to Table 3.2-2 for reactor coolant system average temperature and pressurizer pressure. This change also documents analyses submittal as part of the information for Amendment 82. These changes correct a previous licensee omission, clarify the TSs, and are acceptable.

2.9 Changes to the P-12 Interlock Description

The P-12 ESF Interlock permits the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T-average. The licensee proposed to rewrite the TSs in terms of the actual setpoint value of 541°F which is the low-low bistable trip point. In addition, the functional description is rewritten for clarity. These are basically editorial changes to eliminate potential confusion in the previous specification and are acceptable.

2.10 Simplifications to Power Distribution and APDMS TSs

The current power distribution TSs for Unit 1 are those for a plant using Constant Axial Offset Control and having an Axial Power Distribution Monitoring System (APDMS). In addition, the F_Q limits for EXXON fuel is burnup dependent. Unit 2 has a simplified F_Q (z) specification with Allowable Power Level (APL) surveillance rather than F_{xy} surveillance. The proposed changes would make the Power Distribution Specifications consistent between the two units.

The Specifications affected by this group of changes include 3/4.2.1, 3/4.2.2, 3/4.2.3, and 3/4.2.6. The changes in those Specifications due to deletion of the APDMS and the adoption of APL surveillance were approved for Unit 2 in Amendment 82 to Facility Operating License No. DPR-74 dated May 21, 1986. These changes are also acceptable for D.C. Cook Unit 1.

A further change in Unit 1 Specification 3.2.2 is the replacement of the exposure dependent F_Q limit for EXXON fuel with a constant limit. The licensee has stated that current approved EXXON LOCA models do not require the use of a burnup designation. The staff finds this change to be acceptable. The licensee proposes to remove the requirement in Specification 3.2.2, Action a, to perform the reduction of the Overpower delta-T setpoint in hot standby. An evaluation by the licensee has shown that the reduction may be performed in Mode 1. This is consistent with Standard Technical Specifications and is acceptable.

In a related change, the licensee proposed to add Specification 6.9.1.11 to allow a Peaking Factor Limit Report to be submitted 15 days prior to initial criticality. By letter dated May 1, 1987, the NRC established revised guidance for Unit 1. The licensee agreed to consider any further changes regarding the Peaking Factor Limit Report as a separate action. By letter dated June 7, 1988, the licensee withdrew the proposed TS changes regarding the peaking factor limit report.

2.11 Changes to Increase the Similarity Between Unit 1 TSs and Unit 2 TSs

(a) The P-8 setpoint in Table 3.3-1, Reactor Trip System Instrumentation, is changed from 51 percent to 31 percent coolant flow. The P-8 interlock prevents or defeats the manual block of reactor trip on low coolant flow in a single loop. The lowering of the setpoint is a conservative step and is acceptable.

(b) A footnote related to the conditions to which the pressurizer code safety valve lift settings correspond is added to applicable specifications. This change to Unit 1 TSs makes the TSs more consistent between the two units and is basically administrative in nature. The staff finds the change acceptable.

(c) The Limiting Condition for Operation and Surveillance Requirements for the Refueling Water Storage Tank are changed to require a minimum solution temperature of 80°F and a surveillance test every 24 hours. These changes were previously approved in Amendment 111 to Facility Operating License No. DPR-58 and Amendment 94 to Facility Operating License No. DPR-74 dated June 10, 1987.

(d) A proposed change to the bounding volumes for boration sources was previously approved in the June 10, 1987 Amendment identified in Part 2.11(c) above.

(e) A requirement to have at least three coolant loops operating above the P-12 setpoint has been added to Specification 3.4.1.2. This requirement has been added to assure that the requirements of Table 3.3-3, Engineered Safety Features Actuation System, may be met and is acceptable.

(f) The required boron concentration for refueling has been increased in Specification 3.9.1b to be consistent with the requirement for the RWST. This is acceptable and is applicable to both units.

(g) Specifications 3/4.1.2.b and 3.4.1.3.b have been altered for Unit 2 to require a minimum number of coolant pumps to be operable when it is possible to withdraw rods. This is consistent with the safety analysis of the rod withdrawal event and is acceptable. (See discussion under 2.3 above.)

2.12 Clarification and Removal of Temporary Requirements

The licensee has proposed three changes in this category, which are discussed as follows:

Specifications in Table 3.3-3 footnotes and LCO Section 3.4.1.2 are changed to reflect an analysis previously submitted. During the review of the revised setpoints for Resistance Temperature Detectors (RTS's) and related TS requirements, certain commitments were made for temporary application to the previous Cycle 6 for Unit No. 1. These commitments were necessary to permit certain startup testing pending the correction of RTD calibration data and were approved in Amendment 91 to Facility Operating License DPR-58 dated September 3, 1985. In the present submittal, the necessary corrections were made and the temporary conditions are no longer necessary. The affected TSs are Table 3.3-3, Items 1.f and 4.d (footnote and references are removed) and an editorial change to LCO 3.4.1.2. Based on the staff's review of the additional material, these changes are found to be acceptable.

Secondly, Specifications 3.10.3.b, 3.10.4.b, and 3.10.5.b are reworded to assure clarity. Because the intent of the Specifications is not changed, these changes are acceptable.

The third proposed change is the removal of the reactor trip portion of Table 3.3-5, Item 8a (Steam Generator Water Level - High High), which identifies the Engineered Safety Features Response Times. The basis for this deletion is provided by the licensee in Attachment 6 of the base submittal document. Based on the staff's review of the material which concludes that the results of the feedwater malfunction transient would be essentially unchanged by the modification, the deletion is found acceptable. A similar change was proposed for the Unit 2 TS for consistency, and is acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

The TS changes which the staff found acceptable are summarized in the attached (Attachment 1) table. The table also indicates which section of this SE describes the changes.

Bases changes were also proposed by the licensee which complement the proposed TS changes proposed by the licensee. These changes were reviewed by the NRC staff and found acceptable.

Typographical errors in the licensee's original submittal were corrected by the licensee via replacement pages provided to the NRC on January 13, 1988. These changes were editorial in nature and thus did not require renotice in the Federal Register.

4.0 EVALUATION RESULTS

Based on the review discussed above, the staff concludes that the proposed TSs changes in the categories identified are acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 CONCLUSIONS

The staff has 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 health and safety of the public.

Principal Contributors:

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Date: February 9, 1989

Attachments:
1. Table of TS changes

ATTACHMENT 1

SUMMARY DESCRIPTIONS FOR D. C. COOK UNIT 1 PROPOSED TECHNICAL SPECIFICATIONS

PAGE	SECTION	SER SECTION	DESCRIPTION
1-7	Definition 1.39	2.1	APL made a defined term.
2-1	2.1.1	2.2	Removed reference to Figure 2.1-2 and three loop operation.
2-3	Figure 2.1-2	2.2	Figure is removed.
2-8	Table 2.2-1	2.2	Parameters for three loop operation are removed.
2-9	Table 2.2-1 Notes 3 & 4	2.8	Words "ΔT span" are added.
3/4 1-1	3.1.1.1	2.1	APPLICABILITY changed to MODES 1, 2, and 3.
		2.1	Mathematical symbols are written out in words.
		2.1	Specification title is changed.
3/4 1-2	4.1.1.1.1.e	2.1	Surveillance changed to MODE 3 only.
3/4 1-3	3.1.1.2 4.1.1.2	2.8	Revised to include MODE 4 and MODE 5 in the same specification. Revised Technical Specification requirements based on dilution accident analysis in MODES 4 and 5.
		2.1	Mathematical symbols are written out in words.
		2.1	Specification title is changed.
3/4 1-3a	4.1.1.2.b	2.1	Specification 4.1.1.2.b is moved to new page 3/4 1-3a.
3/4 1-3b	Figure 3.1-3	2.1	New figure is added.
3/4 1-3a; 3/4 1-3b		2.1	Pages added due to length of new specification.
3/4 1-4	3.1.1.3 4.1.1.3	2.1	"reactor pressure vessel" is changed to "reactor coolant system".

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
		2.1	Mathematical symbols are written out in words.
		2.8	Flow rate requirement reduced to 2000 gpm.
		2.4	Footnote added.
3/4 1-7	3.1.2.1	2.4	Footnote added.
3/4 1-11	3.1.2.3	2.4	Footnote added.
	ACTION c	2.1	"ar" is changed to "are".
	4.1.2.3.1	2.1	Mathematical symbols are written out in words.
	4.1.2.3.2.b	2.1	Period is added.
3/4 1-13	3.1.2.5	2.4	Footnote added.
	4.1.2.5.b	2.1	Mathematical symbols are written out in words.
3/4 1-14	3.1.2.6	2.1	"STATUS" is changed to "status".
	4.1.2.6.b	2.1	Mathematical symbols are written out in words.
3/4 1-18; 3/4 1-19; 3/4 1-19a	3.1.3.1	2.1	The words "which are inserted in the core" are removed.
	3.1.3.1 ACTION b	2.1	The word "bank" is replaced by the words "group step counter".
	ACTION c	2.1	The words "due to causes other than addressed by ACTION a, above," are added.
	ACTION c.1 ACTION c.2	2.1	"The rod" is changed to "The affected rod".
	ACTION c.2.a	2.3	This ACTION statement is replaced.
	ACTION c.2.c	2.3	This ACTION statement is added.

PAGE	SECTION	SER SECTION	DESCRIPTION
	ACTION c.2.d ACTION c.2.e	2.1	Current ACTION statements c.2.c and c.2.d are renumbered.
		2.1	Words added to emphasize that when ACTION c.2 is chosen that items a, b and c must be performed plus the choice of either d or e.
	ACTION c.2.d	2.1	Mathematical symbols are written out in words.
	ACTION c.2.e	2.2	Reference to Figure 3.1-2 is removed.
	Table 3.1-1	2.1	Table referred to in Action 2.C is added.
3/4 1-21	3.1.3.3	2.1	Mathematical symbols are written out in words.
		2.1	"(228 steps)" is added.
		2.1	APPLICABILITY changed to MODES 1 and 2.
		2.2	ACTION statement b removed.
	4.1.3.3	2.3	Words "prior to entering MODE 2" replace "prior to reactor criticality".
3/4 1-22	3.1.3.4	2.1	"(228 steps)" is added.
		2.1	Mathematical symbols are written out in words.
3/4 1-23	3.1.3.5	2.2	Reference to Figure 3.1-2 is removed.
	ACTION b	2.1	"figures" becomes "figure".
		2.1	Mathematical symbols are written out in words.
3/4 1-24; 3/4 1-25; 3/4 1-26	Figure 3.1-1 Figure 3.1-2	2.1	Rod Group Insertion Limits figure for 4 Loop Operation is redrawn with labeled endpoints.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
		2.2	Rod Group Insertion Limit figure for 3 Loop Operation is removed.
		2.1	Rod Group Insertion Limit figure for 4 Loop Operation is renamed Figure 3.1-1.
		2.1	Pages 3/4 1-25 and 3/4 1-26 are removed.
3/4 2-1	3.2.1	2.1	"3.4.2" becomes "3/4.2" in title.
		2.1	APL footnote is removed.
		2.1	Mathematical symbols are written out in words.
3/4 2-2	3.2.1.a.2.c	2.10	Exemption from AFD requirements for APDMS calibration is removed.
	3.2.1.d	2.1	Action d is removed.
3/4 2-3	4.2.1.3 4.2.1.4	2.10	$F_Q^M(Z)$ is changed to APL. Referenced specification number has changed.
3/4 2-4	Figure 3.2-1	2.1	Figure is redrawn.
3/4 2-5	3.2.2	2.1	Description of $F_Q(Z)$ penalties moved from surveillance to LCO.
3/4 2-5	3.2.2	2.10	The F_Q limit for Exxon fuel is changed to fixed value of 2.04.
		2.1	Definitions for P, $F_Q(Z)$, and K(Z) reworded with no change of meaning.
	3.2.2.a	2.10	Modified existing ACTION statement a.1 to remove the requirement to lower the Overpower ΔT (OPΔT) in hot standby.
		2.1	"F_Q" is changed to $F_Q(Z)$".
		2.10	ACTION 3.2.2.a.2 is removed.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 2-5; 3/4 2-6	3.2.2	2.1	ACTION statement b is moved from page 3/4 2-6 to page 3/4 2-5.
3/4 2-6; 3/4 2-7; 3/4 2-8; 3/4 2-9	4.2.2.2	2.10	Much of this surveillance requirement has been moved to APL Specification 3.2.6.
3/4 2-8(a)	Figure 3.2-3	2.1	The V(Z) function provided by Exxon Nuclear Co. is removed from Technical Specifications. This page is to be removed from T/S.
3/4 2-10	Figure 3.2-2	2.1	The figure is redrawn.
		2.1	The page number is changed to 3/4 2-7.
3/4 2-11	Figure 3.2-3	2.1	The figure is redrawn.
		2.1	The page number is changed to 3/4 2-8.
3/4 2-12	3.2.3	2.1	Mathematical symbols are written out in words.
		2.1	"power" is changed to "POWER".
		2.1	The page number is changed to 3/4 2-9.
3/4 2-13	4.2.3	2.1	4.2.3.1 is changed to 4.2.3.
		2.1	The page number is changed to 3/4 2-10.
3/4 2-14; 3/4 2-15	3/4.2.4	2.1	The page numbers are changed to 3/4 2-11 and 12, respectively.
		2.1	"LIMITS" is added to title.
		2.1	Mathematical symbols are written out in words.
	ACTION b.2	2.1	"trip" is changed to "Trip".

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 2-16	3/4.2.5	2.1	The page number is changed to 3/4 2-13.
	4.2.5.2 4.2.5.3	2.3	Surveillance Requirement 4.2.5.2 is expanded and clarified.
	4.2.5.4	2.7	Exemption from Specification 4.0.4 is added for primary flow surveillances.
3/4 2-17	Table 3.2-1	2.2	The parameters for three loop operation are removed.
		2.1	The parameters for Design Thermal Power are removed.
		2.1	Units used for pressure changed from psia to psig.
		2.1	1.386×10^8 changed to 138.6×10^6.
		2.8	Footnotes are added for RCS T_{avg} and RCS Total Flow Rate.
		2.1	This page number is changed to 3/4 2-14.
3/4 2-18; 3/4 2-19; 3/4 2-20; 3/4 2-21; 3/4 2-22; 3/4 2-23; 3/4 2-24	3.2.6	2.10	This entire Technical Specification is changed to an Allowable Power Level (APL) Technical Specification.
		2.1	The new APL Technical Specification is on pages 3/4 2-15 and 3/4 2-16.
3/4 2-17 through 3/4 2-24		2.1	Pages 3/4 2-17 through 3/4 2-24 are deleted.
3/4 3-2	Table 3.3-1	2.1	This page is intentionally left blank.

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PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 3-3	Table 3.3-1 Item 2	2.3	Power Range, Neutron Flux Functional Unit has an added applicable mode:*.
	Item 3	2.1	Comma is added.
	Item 5	2.1	"Intermediate Range, Neutron Flux" is typed onto two lines.
	Item 7	2.2	References to three loop operation are removed.
	Item 8	2.2	References to three loop operation are removed.
3/4 3-4	Item 13	2.1	"in" added.
	Item 14	2.1	"loops" is changed to "loop".
3/4 3-5	Item 16	2.1	Slash replaced by hyphen.
	Item 20B	2.7	Reactor Coolant Pump Breaker Position Trip Above P-7 has an added exemption from 3.0.4 applicability.
	Item 22	2.1	Clarifications made to properly identify which ACTION statements apply to applicable mode.
3/4 3-6	Table 3.3-1 Notation	2.2	Footnote ** is removed.
	ACTION 2.b	2.1	Words "of the other channels" are added.
	ACTION 2.c	2.1	Mathematical symbols are written out in words.
3/4 3-8	Table 3.3-1 ACTION 14	2.1	"OPEARABLE" is changed to "OPERABLE".
	ACTION 9	2.2	Action 9 is removed.
		2.1	"Deleted" replaces Action 9, and Action 8 which were previously removed.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
		2.1	Mathematical symbols are written out in words.
3/4 3-9	Table 3.3-1	2.1	Mathematical symbols are written out in words.
		2.11	The value of P-8 is changed to 31% RTP.
		2.1	Description of P-7 and P-8 interlocks clarified to provide detail on loss of flow instrumentation.
3/4 3-12	Table 4.3-1 Item 2	2.3	Power Range, Neutron Flux Functional Unit has an additional Channel Functional Test (S/U(1)).
3/4 3-12; 3/4 3-13	Item 2, 5, 6, 7, 8, 12 & 13	2.7	Power, Intermediate, and Source Range Neutron Flux, Loss of Flow Single Loop and Two Loop Functional Units have added exemptions from Specification 4.0.4. Overpower ΔT and Overtemperature ΔT Functional Units have added exemptions from Specification 4.0.4 for $f_1(\Delta I)$ and $f_2(\Delta I)$ penalties.
3/4 3-14	Table 4.3-1 Notation	2.1	Mathematical symbols are written out in words.
		2.7	Footnotes 8, 9, and 14 are added.
3/4 3-16; 3/4 3-17; 3/4 3-18; 3/4 3-20; 3/4 3-21	Table 3.3-3 Item 1.e 1.f 4.d	2.2	References to three loop operation in Modes 1 and 2 are removed.
		2.5	Reference to ### footnote for Differential Pressure Between Steam Lines-High Functional Unit changed to #### footnote.
	Items 1.f & 4.d	2.12	References to Footnote ** are removed.
3/4 3-22	Table 3.3-3	2.5	Footnote #### is added.
		2.12	Footnote ** is removed.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 3-23	Table 3.3-3	2.9	Reworded Condition and Setpoint, Function description for P-12 interlock.
		2.1	Mathematical symbols are written out in words.
3/4 3-29	Table 3.3-5 Item 8a	2.12	Reactor trip is removed from description.
3/4 3-31	Table 4.3-2 Item 1c	2.1	Period is changed to comma.
3/4 3-33	Table 4.3-2 Item 6d	2.1	Loss of Main Feedwater Pumps Mode 3 Surveillance Requirement deleted.
3/4 3-33a	Table 4.3-2 Item 8.b	2.1	"Loss of Voltage" is changed to "Degraded Voltage".
3/4 3-49; 3/43-50	3.3.3.6 4.3.3.6	2.10	This entire Technical Specification is removed.
3/4 4-2	3.4.1.2	2.3	Criterion for the operability of reactor coolant loops are established based on the status of the reactor trip system breakers and/or the control rod system.
		2.11	Criterion for the operability of reactor coolant loops based on P-12 is added.
		2.1	Existing text reorganized for convenience. ACTION b becomes ACTION d.
	3.4.1.2 ACTION d Footnote *	2.4	** is added to ACTION d and footnote *; footnote ** is added.
3/4 4-2a	4.4.1.2.1 4.4.1.2.2	2.1	Surveillances, footnotes and ACTION d moved from previous page.
3/4 4-3;	3.4.1.3	2.3	Criteria for the operability of

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 4-3a			reactor coolant loops are established based on the status of the reactor trip system breakers and/or the control rod system.
		2.1	Existing text reorganized for convenience. ACTION b becomes ACTION c.
	3.4.1.3 ACTION c Footnote ***	2.4	**** is added to ACTION c and footnote ***; footnote **** is added.
		2.1	Changed 62.00% to 62%. Removed underlining.
3/4 4-3b; 3/4 4-3c; 3/4 4-3d	3.4.1.4 4.4.1.4	2.2	The entire Technical Specification is removed.
		2.1	Pages 3/4 3-c and 3/4 3-d are to be removed.
3/4 4-4	3.4.2	2.4	Footnote ** added.
		2.11	Footnote * is added.
	ACTION b	2.3	ACTION statement added.
3/4 4-5	3.4.3 4.4.3	2.11	Footnote * is added.
3/4 4-35	3.4.11	2.6	ACTION changed to only allow one PORV or block valve inoperable. Making more than one PORV inoperable without shutting down the reactor is not allowed.
		2.1	Reference to Section 6.9.1.9 is deleted.
3/4 4-36	4.4.11.1	2.1	Portions of expanded ACTION statement and surveillance requirements moved to p. 3/4 4-36.
	4.4.11.2	2.1	Reference to Section 6.9.1.9 is deleted.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
		2.1	Footnote * is changed to **.
	4.4.11.3	2.1	Reference to Surveillance 4.8.2.3.2.c is changed to 4.8.2.3.2.d.
3/4 7-1	3.7.1.1	2.2	ACTION b is modified to remove three loop operation in Modes 1 and 2.
3/4 7-3	Table 3.7-2	2.2	Table is removed.
3/4 7-4	Table 4.7-1	2.11	Footnote * is added.
3/4 7-5	4.7.1.2	2.8	Discharge pressures for auxiliary feedwater pump flow testing changed.
		2.1	Mathematical symbols are written out in words.
3/4 7-10	3.7.1.5	2.3	ACTION statements are revised.
	4.7.1.5.1	2.1	Specification 4.7.1.5 is renumbered 4.7.1.5.1.
	4.7.1.5.2	2.7	Exemption from Specification 4.0.4 is added for entry into Mode 3.
		2.7	Exemption from Specification 4.0.4 is provided for entry into Mode 2 with stop valves closed for PHYSICS TESTS.
3/4 8-5	3.8.1.2	2.4	Footnote added.
		2.1	Footnote * is changed to **.
3/4 9-1	3.9.1	2.1	Mathematical symbols are written out in words.
		2.4	Footnote added.
	3.9.1.b ACTION	2.11	The required boron concentration for refueling is increased to 2400 ppm.
3/4 9-2	3.9.2	2.4	Footnote added.
3/4 9-9	3.9.8.1 4.9.8.1	2.1	Mathematical symbols are written out in words.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
		2.8	Flow rate requirement reduced to 2000 gpm.
		2.4	Footnote added.
3/4 10-2	4.10.2.2	2.1	Referenced specifications are renumbered.
		2.1	Mathematical symbols are written out in words.
		2.1	Reference to the Augmented Startup Test Program is removed.
3/4 10-3	3.10.3.b	2.12	Specification is reworded.
		2.1	"reactor trip setpoints" is rewritten as "Reactor Trip Setpoints".
	3.10.3.b ACTION 4.10.3.2	2.1	Mathematical symbols are written out in words.
3/4 10-5	3.10.4.b	2.12	Specification is reworded.
		2.1	"reactor trip setpoints" is rewritten as "Reactor Trip Setpoints".
	ACTION	2.1	"THERMA1" is changed to "THERMAL".
	3.10.4.b ACTION 4.10.4.1	2.1	Mathematical symbols are written out in words.
3/4 10-6	3.10.5.b	2.12	Specification is reworded.
		2.1	Mathematical symbols are written out in words.
	4.10.5.1	2.1	"the" is added.
B 2-1; B 2-1a	2.1.1 (Bases)	B	References to three loop operation and Figure 2.1-2 are removed.
		B	Headings are clarified; footnotes are added.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
B 2-5	Overtemperature WT (Bases)	B	Paragraph referring to three loop operation is removed.
	Overpower WT (Bases)	B	Added reference to f(WI) penalty for OPWT.
	Pressurizer Pressure (Bases)	B	Added reference to the use of the pressurizer pressure high trip in the loss of load event.
B 2-5; B 2-6	2.2.1 (Bases)	B	Moved text from page B 2-6 to B 2-5.
B 2-6	Loss of Flow (Bases)	B	The value of the P-8 setpoint is changed to 31%. This sentence is reworded.
		B	References to three loop operation are removed.
		B	Mathematical symbols are written out in words.
B 2-6; B 2-7; B 2-8	2.2.1 (Bases)	B	Moved text from page B 2-7 to B 2-6, and from page B2-8 to B 2-7. Page B 2-8 may now be deleted.
B 3/4 1-1	3/4.1.1.1 3/4.1.1.2 (Bases)	B	Revision to Shutdown Margin Basis.
		B	350°F is changed to 200°F.
		B	Mathematical symbols are written out in words.
	3/4.1.1.3 (Bases)	B	Flow rate requirement reduced to 2000 gpm.
		B	Circulation time is increased to 45 minutes.
B 3/4 1-2	Minimum Temp. for Criticality (Bases)	B	Revised discussion of interaction between minimum temperature for criticality requirement and P-12 reset point; paragraph reworded for consistency with Unit 2. Technical Specifications.

(Continued)

PAGE	SECTION	SER SECTION	DESCRIPTION
B 3/4 1-2; B 3/4 1-3	3/4.1.2 (Bases)	B	"above" is changed to "below".
		B	Revisions were made to description of the RWST and BAST as boration sources.
		B	Mathematical symbols are written out in words.
		B	pH value limits are added.
B 3/4 2-1	3/4.2	B	"F_Q(ZP)" is changed to "F_Q(Z)".
		B	Updated minimum DNBR limit.
B 3/4 2-2	3/4.2.1	B	The word "of" is added.
		B	"signifigance" is changed to "significance".
		B	"F_Q(ZP)" is changed to "F_Q(Z)".
		B	Description of burnup dependent F_Q envelope is removed.
		B	Period replaced by comma.
B 3/4 2-3	Figure B 3/4 2-1	B	Figure is redrawn.
B 3/4 2-4	3/4.2.2 3/4.2.3	B	The words "nuclear enthalpy hot channel factor" changed to "nuclear enthalpy rise hot channel factor".
		B	References to F_Q and F_{WH} expanded to include proposed APL Technical Specification.
B 3/4 2-5	3/4.2.3	B	"physics tests" is changed to "PHYSICS TESTS".
		B	Section on burnup dependent F_Q for Exxon fuel removed.

PAGE	SECTION	SER SECTION	DESCRIPTION
B 3/4 2-6	3/4.2.5 (Bases)	B	Discussion of flow rate surveillances are included.
	3/4.2.6 (Bases)	B	This section is changed to an Allowable Power Level (APL) Technical Specification.
B 3/4 3-3	3/4.3.3.6 (Bases)	B	This section is removed.
B 3/4 4-1	3/4.4.1 (Bases)	B	References to three loop operation are removed.
		B	Updated minimum DNBR limit.
		B	P-8 is changed to 31% of RTP.
		B	Additional operable loops are required with control rods capable of withdrawal.
B 3/4 4-1; B 3/4 4-2	3/4.4.2 3/4.4.3 (Bases)	B	Text is moved from page B 3/4 4-1 to page B 3/4 4-2.
B 3/4 4-13	3/4.4.11 3/4.4.12 (Bases)	B	Periods are converted to slashes.
B 3/4 5-3	3/4.5.5 (Bases)	B	pH value limits are changed.
		B	Discussion of the difference between the analysis value and Technical Specification value of the RWST temperature is added.
B 3/4 7-1	3/4.7.1.1 (Bases)	B	References to three loop operation are removed.
		B	Reference to Table 3.7-2 is changed to Table 3.7-1. This basis is condensed to one page.
B 3/4 7-2	3/4.7.1 (Bases)	B	Variable definitions are moved to previous page.

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PAGE	SECTION	SER SECTION	DESCRIPTION
B 3/4 9-1	3/4.9.1	B	The basis section from STS is substituted for existing basis and is augmented with a discussion of the increase in boron concentration requirement to 2400 ppm.
	3/4.9.5 (Bases)	B	"CORE ALTERNATIONS" is changed to "CORE ALTERATIONS".

PAGE	SECTION	SER SECTION	DESCRIPTION
2-2	Figure 2.1-1	2.1	Curve for 2250 psia is added.
3/4 1-3	4.1.1.2	2.1	Change "greater than" to "greater than or equal to".
		2.1	Mathematical symbols are written out in words.
	3.1.1.2.b	2.1	Period added.
3/4 1-4	3.1.1.3 4.1.1.3	2.8	Flow rate requirement reduced to 2000 gpm.
		2.1	Mathematical symbols are written out in words.
3/4 1-5; 3/4 1-6; 3/4 1-6a	3.1.1.4 Figure 3.1-2	2.8	The upper limit on MTC for operation above 70% RTP is changed. The upper limit is now graphically displayed.
	4.1.1.4.b	2.1	Specified 300 ppm surveillance at "RATED THERMAL POWER equilibrium boron concentration".
		2.1	Mathematical symbols are written out in words.
		2.1	The new MTC limits are now graphically displayed in Figure 3.1-2 on new page 3/4 1-6a.
3/4 1-8	3.1.2.1	2.4	Footnote added.
	4.1.2.1.a	2.1	"/ 145" is changed to "greater than or equal to 145°F".
3/4 1-11	3.1.2.3	2.1	"the" is removed from footnote.
		2.1	"ar" is changed to "are".
		2.1	Mathematical symbols are written out in words.
3/4 1-18	3.1.3.1 ACTION c.1 ACTION c.2	2.1	"The rod" is changed to "The affected rod".

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PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 1-18; 3/4 1-19	ACTION c.2.b	2.1	ACTION c.2.b is moved from page 3/4 1-19 to page 3/4 1-18.
	ACTION c.2	2.1	Words added to emphasize that when ACTION c.2 is chosen that items a, b and c plus the choice between items d and e must be performed.
3/4 1-19	ACTION c.2.d	2.1	Mathematical symbols are written out in words.
	ACTION c.2.e	2.1	Reference to Figure 3.1-2 is removed.
	4.1.3.1.1	2.1	References to part length rods are removed.
	4.1.3.1.2	2.1	The words "in the core" are removed.
3/4 1-23	3.1.3.4	2.1	"(228 steps)" is added.
		2.1	Mathematical symbols are written out in words.
	4.1.3.4	2.3	Words "prior to entering Mode 2" replace "prior to reactor criticality".
3/4 1-24	3.1.3.5	2.1	"(228 steps)" is added.
		2.1	Mathematical symbols are written out in words.
3/4 1-25	3.1.3.6	2.1	"figures" is changed to "figure".
		2.1	Mathematical symbols are written out in words.
3/4 1-27		2.1	Page is removed.
3/4 2-1		2.1	APL footnote is removed.
3/4 2-4	Figure 3.2-1	2.1	Figure is redrawn.
3/4 2-16	Table 3.2-1	2.8	Footnote added to document flow allowance for measurement error. Analysis value reduced by the value of the allowance.

PAGE	SECTION	SER SECTION	DESCRIPTION
	Table 3.2-1	2.1	Footnote *** is added.
		2.1	Asterisks moved to right hand column.
	Footnote **	2.8	The words "at least three" are added.
3/4 2-18	Table 3.2-2	2.8	Allowance for readability included for RCS Tavg and Pressurizer Pressure. The allowance was calculated consistently with footnote *.
3/4 2-19	3.2.6	2.1	ALLOWABLE POWER LEVEL is capitalized.
		2.1	Expression for APL is revised to more accurately reflect the meaning of APL.
		2.1	Second "$F_Q(Z)$" is replaced by "measured hot channel factor".
3/4 2-19; 3/4 2-20		2.1	ACTION statements are moved from page 3/4 2-19 to page 3/4 2-20.
3/4 3-3	Table 3.3-1 Items 13 & 14	2.1	"in" is added.
3/4 3-4	Table 3.3-1 Items 21 & 22	2.1	Clarifications made to properly identify which ACTION statements apply to each applicable mode.
3/4 3-8	P-7 and P-8 Descriptions	2.1	Description of P-7 and P-8 interlocks clarified to provide detail on loss of flow instrumentation.
3/4 3-11	Functional Unit 6	2.7	TS 4.0.4 exemption added for condition where unit is leaving Mode 1.
3/4 3-12	Table 4.3-1 Item 13	2.7	Loss of Flow-Two Loops Functional Unit has an added exemption from Specification 4.0.4.
3/4 3-13	Note 14	2.7	Note 14 added.

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PAGE	SECTION	SER SECTION	DESCRIPTION
3/4 3-28	Table 3.3-5 Item 8a	2.1	Reactor trip is removed from description.
3/4 4-2	3.4.1.2.d	2.11	Criterion for the operability of reactor coolant loops based on P-12 is added.
	ACTION b ACTION c	2.11	ACTION statements added to address too few reactor coolant loops when control rods are capable of withdrawal. Old ACTION b becomes ACTION d.
3/4 4-2a	ACTION d	2.1	ACTION d and footnote moved from previous page.
3/4 4-3 3/4 4-3a	3.4.1.3 ACTION b	2.11	ACTION statement added to address too few reactor coolant loops when control rods are capable of withdrawal. Old ACTION b becomes ACTION c.
3/4 4-4	3.4.2	2.4	Footnote added.
3/4 4-32	3.4.11 ACTIONS a.1, a.2	2.1	"inoperable" added to clarify which PORV must be restored.
3/4 8-5	3.8.1.2	2.4	Footnote added.
		2.1	Existing footnote * is changed to footnote **.
3/4 9-2	3.9.2	2.4	Footnote added.
		2.1	Footnote removed.
3/4 9-8	3.9.8.1	2.8	Flow rate requirement reduced to 2000 gpm.
		2.1	Mathematical symbols are written out in words.
3/4 10-3	3.10.3.b	2.12	Specification is reworded.

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PAGE	SECTION	SER SECTION	DESCRIPTION
		2.1	"reactor trip setpcints" is changed to "Reactor Trip Setpoints".
	3.10.3.b ACTION 4.10.3.1	2.1	Mathematical symbols are written out in words.
3/4 10-4	3.10.4.b	2.12	Specification is reworded.
	4.10.4.1	2.1	"the" is added.
		2.1	Mathematical symbols are written out in words.
6-19	6.9.2.h	2.1	Moderator Temperature Coefficient is added to the Special Reports list.
	6.9.2.e	2.1	Comma is removed.
B 3/4 1-3	3/4.1.2	B	Revisions made to the description of the RWST as a boration source.
B 3/4 4-1a 4-2	3/4.4.2 3/4.4.3 3/4.4.4 (Bases)	B	Text is combined to one page; B 3/4 4-19 is removed .
B 3/4 5-3	3/4.5.5 (Bases)	B	pH value limits are changed.
B 3/4 9-1	3/4.9.1 (Bases)	B	Bases section enhanced to incorporate standard TS text and a discussion of the increase in boron concentration to 2400 ppm.