Docket

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

March 30, 1993

Docket No. 50-482

Mr. Bart D. Withers
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Withers:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M84946)

The Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications in response to your application dated October 28, 1992, as supplemented by your letters dated January 28, 1993 and March 8, 1993.

The amendment revises various Technical Specifications to support the use of VANTAGE 5H fuel with intermediate flow mixers, include results of revised transient, thermal-hydraulic, and nuclear design analyses, allow main steam safety valve setpoint tolerance increases, and relocate cycle-specific parameters to the Core Operating Limits Report. The Core Operating Limits Report should be submitted to the NRC upon its issuance and will satisfy the reporting requirements associated with Cycle 7 operation.

As discussed with your staff and included in your March 8, 1993 letter, the staff has inserted the dates for the NRC Safety Evaluations referenced in Technical Specification 6.9.1.9. A minor change was made in your requested revision of Technical Specification 6.9.1.9 to reflect the fact that the Safety Evaluation for the Wolf Creek Nuclear Operating Corporation's Transient Analysis Methodology topical report has not been issued. An editorial change was also made to the format of Technical Specification 6.9.1.9. These changes were discussed with your staff and were found to be acceptable. Completion of the NRC review of the transient analysis topical report is expected in the near future. The review performed to date provided the necessary assurance of the adequacy of your Technical Specification amendment request. Upon issuance of the Safety Evaluation for the transient analysis topical report, a revision to Technical Specification 6.9.1.9 to reflect the approval will be coordinated with your staff.

DF01
1/1

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 61 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. NPF-42

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

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated October 28, 1992 and supplemented by letters dated January 28, 1993 and March 8, 1993, 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, as amended, the provisions of the Act, and the rules and regulations of the Commission;

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

   D. The issuance of this license 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. NPF-42 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 61 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and is to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C Black

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 30, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Revised 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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
I	I
II	II
IV	IV
V	V
XIX	XIX
XXI	XXI
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
2-2	2-2
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
3/4 1-8	3/4 1-8
3/4 1-12	3/4 1-12
3/4 1-14	3/4 1-14
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 1-22	--
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9

REMOVE	INSERT
3/4 2-10	3/4 2-10
3/4 2-14	3/4 2-14
3/4 2-15	3/4 2-15
--	3/4 2-16
3/4 3-43	3/4 3-43
3/4 7-3	3/4 7-3
3/4 9-1	3/4 9-1
3/4 9-16	3/4 9-16
3/4 10-2	3/4 10-2
3/4 10-4	3/4 10-4
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	--
B 3/4 3-4	B 3/4 3-4
5-6	5-6
5-7	5-7
5-8	5-8
6-21	6-21
--	6-21a
--	6-21b

SECTION PAGE

DEFINITIONS (Continued)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION PAGE

BASES

SECTION PAGE

SECTION PAGE

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

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

DOSE EQUIVALENT I-131

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

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

REACTOR TRIP SYSTEM RESPONSE TIME

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

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

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

SLAVE RELAY TEST

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

WASTE GAS HOLDUP SYSTEM

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

TABLE 1.1

FREQUENCY NOTATION

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

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 four 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, and comply with the requirements of Specification 6.7.1.

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, and comply with the requirements of Specification 6.7.1.

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, and comply with the requirements of Specification 6.7.1.

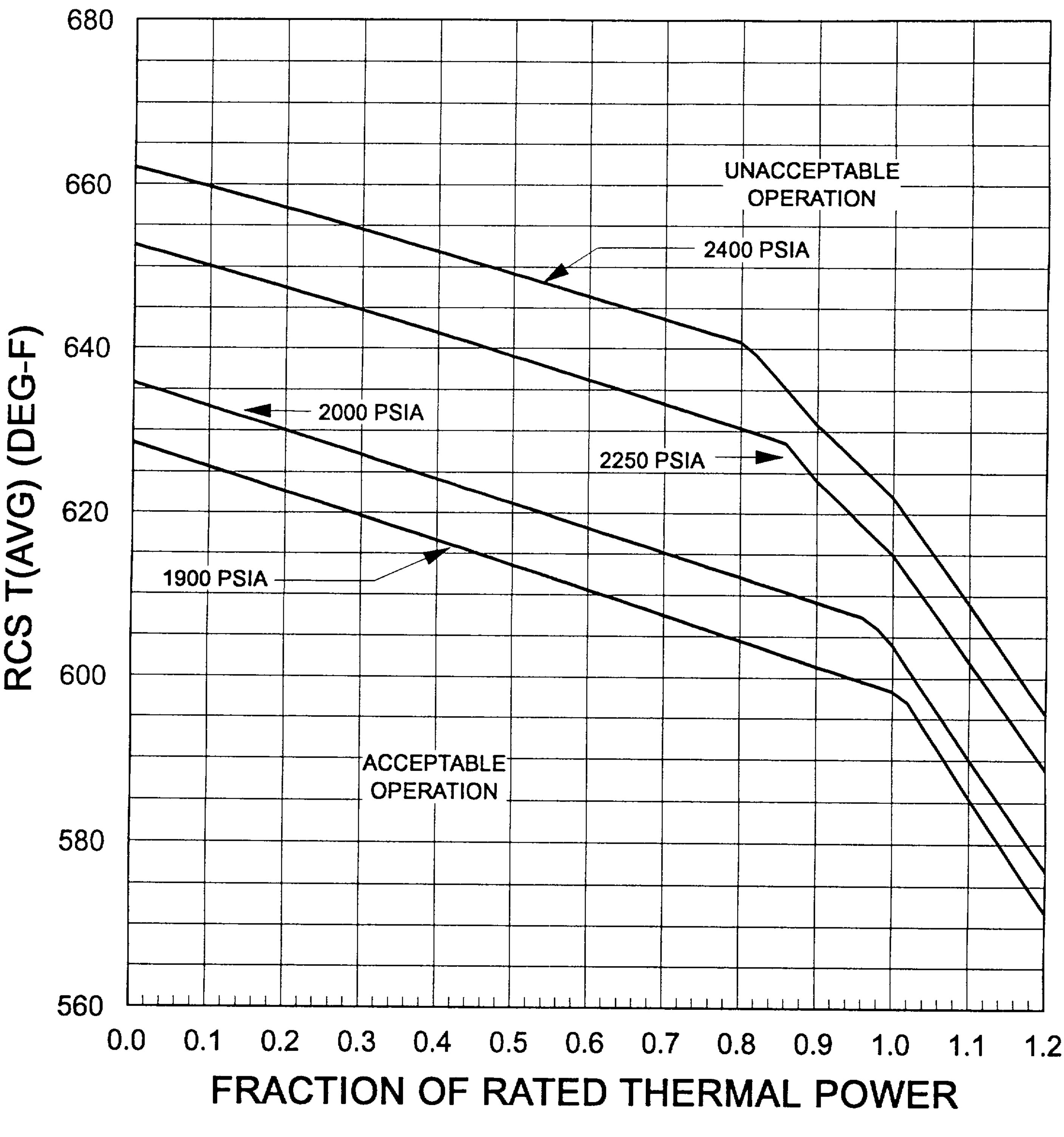

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	7.3	5.11	2.42	See Note 1	See Note 2
8. Overpower ΔT	4.60	2.93	0.14	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.7	0.71	2.49	≥1915 psig	≥1906 psig
10. Pressurizer Pressure-High	7.5	0.71	2.49	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	1.96	≤92% of instrument span	≤93.9% of instrument span

*RTP = RATED THERMAL POWER
**Loop design flow = 93,600 gpm

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 6 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 2 s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.10;
K_2	=	0.0137/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 16 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000671;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron 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 -25% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -25% the ΔT Trip Setpoint shall be automatically reduced by 1.8% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 1.384% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.6% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 6 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 2 s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_4	=	1.10;
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature;
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constant utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00128/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;

T = Average temperature, °F;

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F);

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.67% ΔT span.

2.1 SAFETY LIMITS

BASES

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 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 through the DNBR correlations. DNBR correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation as specified in the CORE OPERATING LIMITS REPORT (COLR). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit. For plant conditions which fall outside the range of applicability of the DNB correlation above, the W-3 correlation is used.

In addition, DNB margin is maintained by performing safety analyses to a higher value than the correlation limit, called the safety analysis limit DNBR. The margin between the safety analysis limit DNBR and the correlation limit DNBR is used to cover known DNBR penalties and provide margin for design flexibility. The safety analysis limit DNBR is specified in the COLR.

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 safety analysis limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on the design $F_{\Delta H}$ specified in the COLR and a reference cosine with a peak of 1.55 for axial power shape.

2.1.1 REACTOR CORE (Continued)

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping and valves are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at greater than or equal to 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% $\Delta k/k$ for four loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron 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 1.3% $\Delta k/k$:

a. Within 1 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 verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

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*See Special Test Exception Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3, 4, or 5, 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 ± 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 4.1.1.1.1e. above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limits specified in Figure 3.1-1 for the all rods withdrawn, beginning of cycle life (BOL), THERMAL POWER condition,

b. Less negative than the EOL limit specified in the COLR for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

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

ACTION:

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

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

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

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

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

#See Special Test Exception Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

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

a. The MTC shall be measured and compared to the BOL hot zero power limit specified in Figure 3.1-1 prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit, at least once per 14 EFPD during the remainder of the fuel cycle.

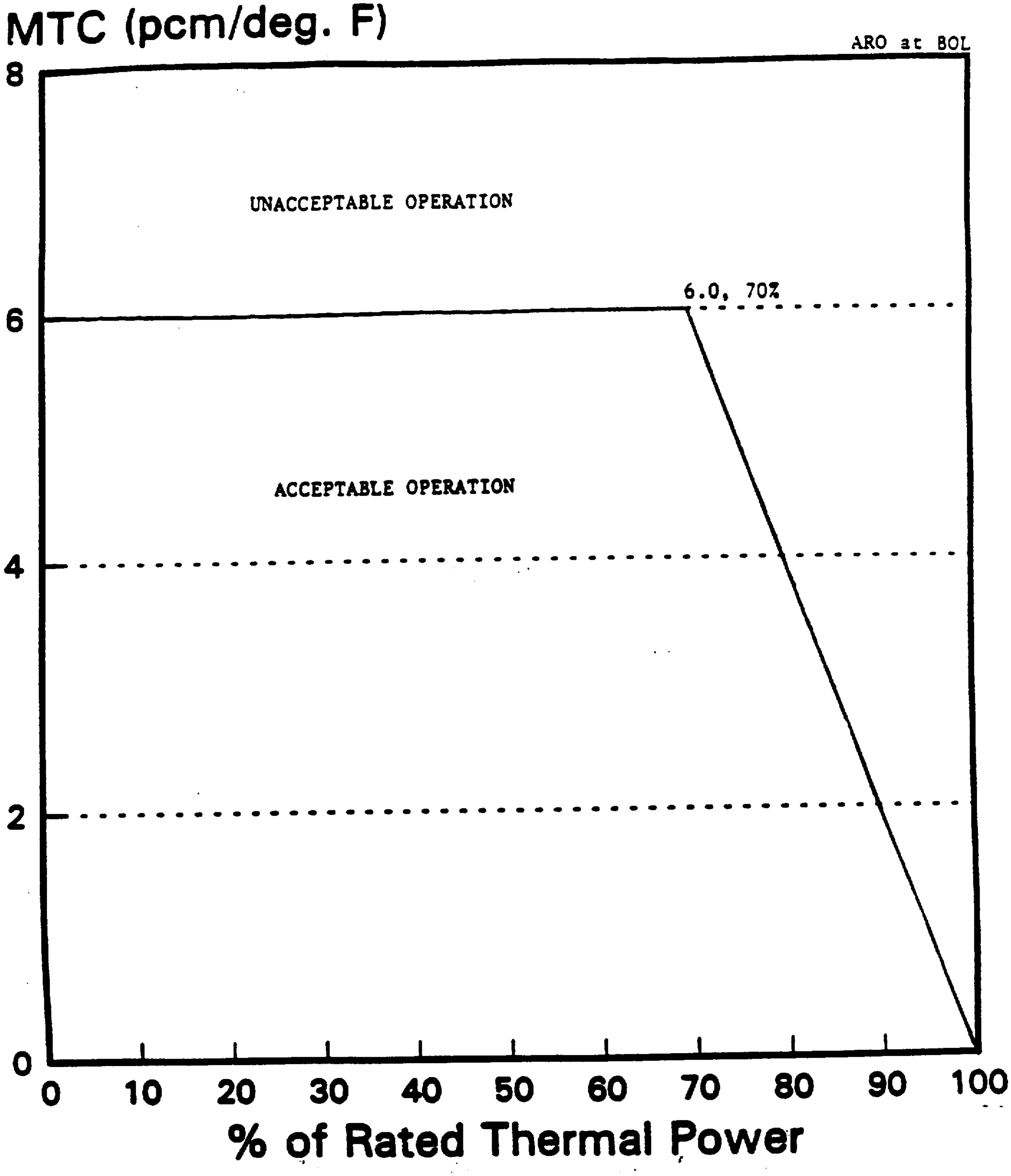

Figure 3.1-1
BOL MODERATOR TEMPERATURE COEFFICIENT VS. POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°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 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.

*See Special Test Exception Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System if the Boric Acid Storage System is OPERABLE as given in Specification 3.1.2.5a. for MODES 5 and 6 or as given in Specification 3.1.2.6a. for MODE 4; or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank is OPERABLE as given in Specification 3.1.2.5b. for MODES 5 and 6 or as given in Specification 3.1.2.6b. for MODE 4.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. 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;

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the Reactor Coolant System.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 2968 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 55,416 gallons,

2) A minimum boron concentration of 2400 ppm, and

3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,
2) Verifying the contained borated water volume, and
3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water sources shall be OPERABLE as required by Specification 3.1.2.2 for MODES 1, 2, and 3 and one of the following borated water sources shall be OPERABLE as required by Specification 3.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

   1) A minimum contained borated water volume of 17,658 gallons,

   2) Between 7000 and 7700 ppm of boron, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

   1) A minimum contained borated water volume of 394,000 gallons,

   2) Between 2400 and 2500 ppm of boron,

   3) A minimum solution temperature of 37°F, and

   4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources in MODE 1, 2 or 3, restore the storage system 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 at least 1.3% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable in MODE 1, 2, or 3, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the boron concentration in the water,

   2) Verifying the contained borated water volume of the water source, and

   3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

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: The ACTION to be taken is based on the cause of inoperability of control rods as follows:

CAUSE OF INOPERABILITY	ACTION	
	One Rod	More Than One Rod
a) Immovable as a result of excessive friction or mechanical interference or known to be untrippable.	(1)	(1)
b) Misaligned from its group step counter demand height or from any other rod in its group by more than ± 12 steps (indicated position).	(3)	(2)
c) Inoperable due to a rod control urgent failure alarm or other electrical problem in the rod control system, but trippable.	(4)	(4)

ACTION 1 - 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.

ACTION 2 - Be in HOT STANDBY within 6 hours.

ACTION 3 - POWER OPERATION may continue provided that within 1 hour:

1. The rod is restored to OPERABLE status within the above alignment requirements, or
2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod 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.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

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.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Restore the rod to within the insertion limit specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

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

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the allowed operational space specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 above 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the limits specified in the COLR within 15 minutes, or

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

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q^{MA}(X,Y,Z) \le \frac{[F_Q^{RTP}]}{P} [K(Z)] \text{ for } P > 0.5, \text{ and}$$

$$F_Q^{MA}(X,Y,Z) \le \frac{[F_Q^{RTP}]}{0.5} [K(Z)] \text{ for } P \le 0.5.$$

Where:

$F_Q^{MA}(X,Y,Z)$ = the measured heat flux hot channel factor, $F_Q^{M}(X,Y,Z)$, increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty,

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

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

K(Z) = the normalized $F_Q(X,Y,Z)$ limit as a function of core height, as specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable THERMAL POWER at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 2 hours and declare the AFD monitor alarm inoperable until the AFD alarm setpoints are reset to the modified limits; and

c. 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% $F_Q^{MA}(X,Y,Z)$ exceeds the limit; and

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

LIMITING CONDITION FOR OPERATION (Continued)

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_q(X,Y,Z)$ 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 $F_Q^M(X,Y,Z)$ shall be evaluated to determine if $F_Q(X,Y,Z)$ is within its limit by:

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

b. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or

2. After exceeding by 20% or more of RATED THERMAL POWER the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined*;

c. Satisfying the relationship presented in Specification 3.2.2;

d. Satisfying the following relationship:

$F_Q^M(X,Y,Z) \leq [F_Q(X,Y,Z)]^{NOM}$

where $[F_Q(X,Y,Z)]^{NOM}$ represents the nominal design power distribution increased by an allowance for the expected deviation between the nominal design power distribution and the measurement and is specified in the COLR.

If the above relationship is not satisfied, then for that location perform the following:

1. Calculate the % margin to the maximum allowable design as follows:

$$\text{\% Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{OP}}\right) 100$$

$$\text{\% Reactor Protection Setpoint (RPS) Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{RPS}}\right) 100$$

where, $[F_Q^L(X,Y,Z)]^{OP}$ and $[F_Q^L(X,Y,Z)]^{RPS}$ are the Operational and RPS design peaking limits and are specified in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.d.1, above. If the minimum margin is less than 0, EITHER of the following actions shall be taken:

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

SURVEILLANCE REQUIREMENTS (Continued)

a. Within 2 hours, control the AFD to within new AFD limits that are determined by:

Reduced negative AFD Limit =

The negative AFD Limit in Specification 3.2.1
plus
the absolute value of the quantity
[Op Mar NSLOPE * Minimum Operational Margin],

Reduced positive AFD Limit =

The positive AFD Limit in Specification 3.2.1
minus
the absolute value of the quantity
[Op Mar PSLOPE * Minimum Operational Margin],

where, the Op Mar NSLOPE and Op Mar PSLOPE are specified in the COLR, and

declare the AFD monitor alarm inoperable until the AFD alarm setpoints are modified to the limits of 4.2.2.2.d.2.a, or

b. Comply with the ACTION requirements of Specification 3.2.2, treating the margin violation in 4.2.2.2.d.1, above, as the amount by which $F_Q^{MA}(X,Y,Z)$ is exceeding its limit.

3. Find the minimum RPS margin of all locations examined in 4.2.2.2.d.1, above. If the minimum margin is less than 0, the following action shall be taken:

Within 72 hours, reduce the negative $f_1(\Delta I)$ limit and the positive $f_1(\Delta I)$ limit of the OTΔT as follows:

Reduced negative $f_1(\Delta I)$ Limit =

$f_1(\Delta I)$ of Table 2.2-1
plus
the absolute value of the quantity
[the RPS Mar NSLOPE * Minimum RPS Margin],

Reduced positive $f_1(\Delta I)$ Limit =

$f_1(\Delta I)$ of Table 2.2-1
minus
the absolute value of the quantity
[the RPS Mar PSLOPE * Minimum RPS Margin],

where, RPS Mar NSLOPE and RPS Mar PSLOPE are specified in the COLR.

e. The limits in Specification 4.2.2.2.d are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive,

2. Upper core region from 85 to 100%, inclusive,

3. Grid Plane Regions, and

4. Core plane regions within +/- 2% of core height (+/- 2.88 inches) about the bank demand position of the Bank "D" control rods.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by the following relationship:

$$F\Delta HR^{M}(X,Y) \leq F\Delta HR^{L}(X,Y)$$

where,

$F\Delta HR^{M}(X,Y)$ = the maximum measured radial peak ratio defined in the Core Operating Limits Report (COLR).

$F\Delta HR^{L}(X,Y)$ = the maximum allowable radial peak ratio defined and specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%* for each 1% that $F\Delta HR^{M}(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

   1. Restore $F\Delta HR^{M}(X,Y)$ to within the limit for RATED THERMAL POWER, or

   2. Reduce the Power Range Neutron Flux - High Trip Setpoint at least RRH% for each 1% that $F\Delta HR^{M}(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit, either:

   1. Restore $F\Delta HR^{M}(X,Y)$ to within the limit for RATED THERMAL POWER, or

   2. Perform the following actions:

      a. Reduce the OTΔT K_1 term by at least TRH** for each 1% that $F\Delta HR^{M}(X,Y)$ exceeds the limit, and

      b. Verify through incore mapping that $F\Delta HR^{M}(X,Y)$ is restored to within the limit for the THERMAL POWER allowed by ACTION a, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

* RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F\Delta HR^{M}(X,Y)$ exceeds $F\Delta HR^{L}(X,Y)$ and is specified in the COLR.

**TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F\Delta HR^{M}(X,Y)$ exceeds the limit and is specified in the COLR.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a and/or c.2.b, above; subsequent POWER OPERATION may proceed provided that $F\Delta HR^M(X,Y)$ is demonstrated, through incore flux mapping, to be within the limit specified in the COLR prior to exceeding the following THERMAL POWER levels:

  1. A nominal 50% of RATED THERMAL POWER,
  2. A nominal 75% of RATED THERMAL POWER, and
  3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F\Delta HR^M(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ is within its limit by:

a. Measuring $F\Delta HR^M(X,Y)$ according to the following schedule:

  1. Prior to operation above 75% of RATED THERMAL POWER at the beginning of each cycle, and
  2. At least once per 31 Effective Full Power Days.

b. Satisfying the following relationship:

$$F\Delta HR^M(X,Y) \le F\Delta HR^{NOM}(X,Y)$$

where, $F\Delta HR^{NOM}(X,Y)$ represents the nominal design power distribution increased by an allowance for the expected deviation between the nominal design power distribution and the measurement and is specified in the COLR.

If the above relationship is not satisfied, then for that location perform the following:

  1. Calculate the % margin to the maximum allowable design as follows:

$$\% \ F_{\Delta H} \text{ Margin} = \left(1 - \frac{F\Delta HR^M(X,Y)}{F\Delta HR^L(X,Y)}\right) 100$$

  2. Find the minimum margin for all locations examined in 4.2.3.2.b.1, above. If the minimum margin is less than 0, comply with the ACTION requirements of Specification 3.2.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. 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; and

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 at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 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, and

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 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, and

c. Reactor Coolant System (RCS) Flow Rate

APPLICABILITY: MODE 1.*

ACTION:

a. With parameter 1 or 2 of Table 3.2-1 exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

b. With the RCS total flow rate outside the region of acceptable operation shown on Table 3.2-1:

1. Within 2 hours either:

   a. Restore the total flow rate to within the above limit, or

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER.

2. Within 6 hours either:

   a. Restore the total flow rate to within the above limit, or

   b. Reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER.

3. Within 72 hours of initially being outside the above limit, verify that the RCS total flow rate is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

*See Special Test Exception Specification 3.10.4 for 3.2.5.c.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION 1.b and/or 3, above; subsequent POWER OPERATION may proceed provided that the indicated RCS total flow rate is demonstrated to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

   a. A nominal 50% of RATED THERMAL POWER,

   b. A nominal 75% of RATED THERMAL POWER, and

   c. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.5.1 The provisions of Specification 4.0.4 are not applicable to Specification 3.2.5.c.

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

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

4.2.5.4 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.5.5 The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation
1. Indicated Reactor Coolant System T_{avg}	$\leq$592.5°F
2. Indicated Pressurizer Pressure	$\geq$2220 psig*
3. Reactor Coolant System Flow Rate	$\geq 38.4 \times 10^4$ GPM

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

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

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

a. Recalibration of the Excore Neutron Flux Detection System,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$.

ACTION:

a. With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING* (±3%)**	ORIFICE SIZE
Loop 1	Loop 2	Loop 3	Loop 4		
V055	V065	V075	V045	1185 psig	16.0 sq. in.
V056	V066	V076	V046	1197 psig	16.0 sq. in.
V057	V067	V077	V047	1210 psig	16.0 sq. in.
V058	V068	V078	V048	1222 psig	16.0 sq. in.
V059	V069	V079	V049	1234 psig	16.0 sq. in.

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

**After testing, the valves will be left at ±1%.

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 motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary 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

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1535 psig on recirculation flow when tested pursuant to Specification 4.0.5;

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1625 psig at a flow of greater than or equal to 120 gpm when the secondary steam supply pressure is greater than 900 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 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. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

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 30 gpm of a solution containing greater than or equal to 7000 ppm boron 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 the limit specified in the COLR, whichever is the more restrictive.

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 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.

4.9.1.3 Valves BG-V178 and BG-V601 shall be verified locked closed and secured in position at least once per 31 days.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

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

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 Spent fuel assemblies stored in Region 2 shall be subject to the following conditions:

a. The combination of initial enrichment and cumulative exposure shall be within the acceptable domain of Figure 3.9-1, and

b. No spent fuel assemblies shall be placed in Region 2, nor shall any storage location be changed in designation from being in Region 1 to being in Region 2, while refueling operations are in progress.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The burnup of each spent fuel assembly stored in Region 2 shall be ascertained by analysis of its burnup history and independently verified, prior to storage in Region 2. A complete record of such analysis shall be kept for the time period that the spent fuel assembly remains in Region 2 of the spent fuel pool.

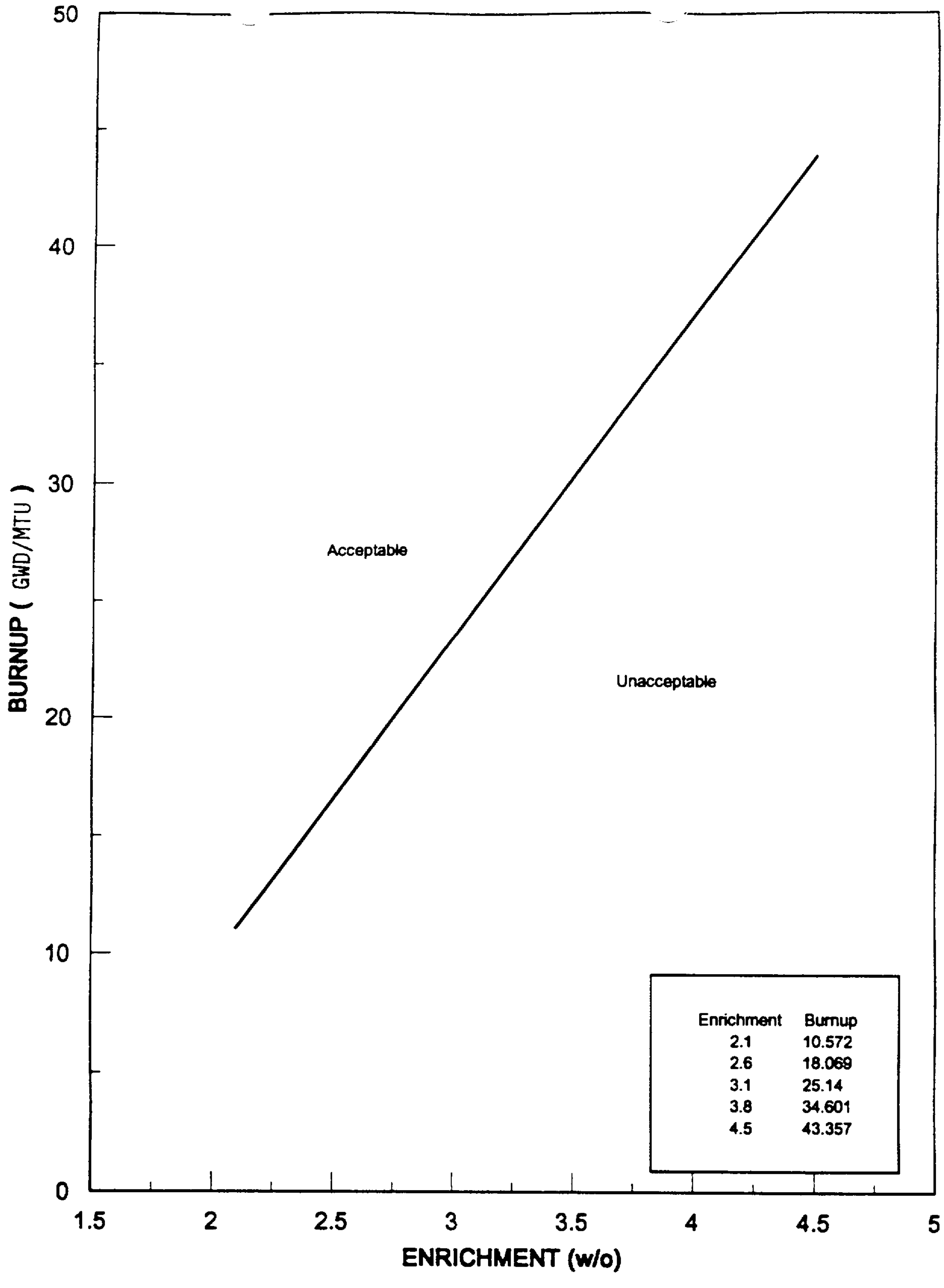

FIGURE 3.9-1 WOLF CREEK MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT FOR STORAGE IN REGION 2

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 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, 3.1.3.5, 3.1.3.6, 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, 3.2.3, and 3.2.5.c 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, 3.2.3, or 3.2.5.c being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 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, 3.2.3, and 3.2.5.c, 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 requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specification 4.2.2.2,

b. Specification 4.2.3.2, and

c. Specification 4.2.5.2.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 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,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

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

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

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 an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.2.3, 3.2.5.c and 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

   1) The THERMAL POWER does not exceed the P-10 Interlock Setpoint, and

   2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-10 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-10 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-10 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-10 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

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

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

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

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

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC End of Life (EOL) value specified in the CORE OPERATING LIMITS REPORT (COLR). The 300 ppm surveillance limit MTC value specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value specified in the COLR.

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

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

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

3/4.1.2 BORATION SYSTEMS

The Boration Systems 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) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature equal to or greater than 350°F a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,754 gallons of 2400 ppm borated water from the RWST. With the RCS average temperature less than 350°, only one boron injection flow path is required.

BORATION SYSTEMS (Continued)

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

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.3% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2968 gallons of 7000 ppm borated water from the boric acid storage tanks or 14,071 gallons of 2400 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. In the case of the boric acid tanks, all of the contained volume is considered usable. The required usable volume may be contained in either or both of the boric acid tanks.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

MOVABLE CONTROL ASSEMBLIES (Continued)

For purposes of determining compliance with Specification 3.1.3.1, any immovability of a control rod invokes ACTION Statement 3.1.3.1.a. Before utilizing ACTION Statement 3.1.3.1.c, the rod control urgent failure alarm must be illuminated or an electrical problem must be detected in the rod control system. The rod is considered trippable if the rod was demonstrated OPERABLE during the last performance of Surveillance Requirement 4.1.3.1.2 and met the rod drop time criteria of Specification 3.1.3.4 during the last performance of Surveillance Requirement 4.1.3.4.

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

The power reduction and shutdown time limits given in ACTION statements 3.1.3.2.a.2, 3.1.3.2.b.2, and 3.1.3.2.c.2, respectively, are initiated at the time of discovery that the compensatory actions required for POWER OPERATION can no longer be met.

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

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

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to the DNBR design limit specified in the CORE OPERATING LIMITS REPORT (COLR) 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 certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(X,Y,Z)$ Heat Flux Hot Channel Factor, is defined as the 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, at assembly (X,Y);

$F_{\Delta H}(X,Y)$ 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 at assembly (X,Y).

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ limits are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD limits have been adjusted for measurement uncertainty.

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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the AFD limits and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

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

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

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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

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

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

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

$F_{\Delta H}(X,Y)$ will be maintained within its limits provided Conditions a. through d. above are maintained. The limits on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, are specified in the COLR as Maximum Allowable Radial Peak Ratio limits, obtained by dividing the Maximum Allowable Peak (MAP) limit by the axial peak for assembly location (X,Y). By definition, the Maximum Allowable Radial Peak Ratio limits will result in a DNBR for the limiting transient that is equivalent to the DNBR calculated with the design $F_{\Delta H}(X,Y)$ value specified in the COLR and a limiting reference axial power shape.

$F_Q^M(X,Y,Z)$ and $F\Delta HR^M(X,Y)$ are measured periodically to provide assurance that they remain within their limits. A peaking margin calculation is performed, when necessary, to provide the basis for reducing THERMAL POWER, for reducing the width of the AFD limits, and for reducing the $f_1(\Delta I)$ limits of the OTΔT trip setpoints.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

QUADRANT POWER TILT RATIO (Continued)

The 2-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 control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the Reactor Coolant System T_{avg} and the pressurizer pressure 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 USAR assumptions and have been analytically demonstrated adequate to maintain a DNBR above the safety analysis limit DNBR specified in the CORE OPERATING LIMITS REPORT (COLR) throughout each analyzed transient. The indicated T_{avg} value of 592.5°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595°F and 2205 psig respectively, with allowance for measurement uncertainty.

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.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins completely offset any rod bow penalties. This is the margin between the correlation DNBR limit and the safety analysis limit DNBR. These limits are specified in the COLR.

The applicable values of rod bow penalties are referenced in the USAR.

When RCS flow rate and $F_{\Delta H}(X,Y)$, per Specification 3.2.3, are measured, no additional allowances are necessary prior to comparison with the limits in the COLR. Measurement uncertainties of 2.5% for RCS total flow rate and 4% for $F_{\Delta H}(X,Y)$ have been allowed for in determination of the design DNBR value.

DNB PARAMETERS (Continued)

The measurement uncertainty for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, an inspection is performed of the feedwater venturi each refueling outage.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified in Table 3.2-1. This surveillance also provides adequate monitoring to detect any core crud buildup.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) Feedwater System isolates, (4) the emergency diesel generators start, (5) containment spray pumps start and automatic valves position, (6) containment isolates, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates safety injection actuation on low pressurizer pressure and low steamline pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steamline pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated ACTION will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Control Room Emergency Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q^M(X,Y,Z)$ or $F_{\Delta H}^M(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Neutron Flux channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4 or stainless steel or by vacancies may be made if justified by a cycle-specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum nominal enrichment of 3.10 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rod assemblies shall be hafnium, silver-indium-cadmium, or a mixture of both types. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total volume of the Reactor Coolant System, including pressurizer and surge line, is 12,135 ± 100 cubic feet at a nominal T_{avg} of 557°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the USAR. This is based on new fuel with an enrichment of 4.45 weight percent U-235 in Region 1 and on spent fuel with combination of initial enrichment and discharge exposures, shown in Figure 3.9-1, in Region 2, and

b. A nominal 9.236 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 2040 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1344 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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CORE OPERATING LIMITS REPORT (COLR)

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

1. Specification 3.1.1.3: Moderator Temperature Coefficient (MTC) EOL limits

2. Specification 3.1.3.5: Shutdown Rod Insertion Limit

3. Specification 3.1.3.6: Control Rod Insertion Limits

4. Specification 3.2.1: Axial Flux Difference (AFD)

5. Specification 3.2.2: Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$

6. Specification 3.2.3: Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$

7. Specification 3.9.1.b: Refueling Boron Concentration

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

a. NRC Safety Evaluation Report dated October 29, 1992, for the "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station" (ET-90-0140, ET 92-0103)

   (Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$

b. NRC Safety Evaluation Report (upon issuance) for the "Transient Analysis Methodology for the Wolf Creek Generating Station" (ET-91-0026, ET 92-0142, WM 93-0010, WM 93-0028)

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient [MTC])

c. NRC Safety Evaluation Report dated March 26, 1993, for the "Qualification of the Steady State Core Physics Methodology for the Wolf Creek Generating Station" (ET 92-0011, WM 93-0038)

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient (MTC); Specification 3.1.3.5 - Shutdown Rod Insertion Limit; Specification 3.1.3.6 - Control Rod Insertion Limits; Specification 3.2.1 - Axial Flux Difference; Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$; Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$; Specification 3.9.1.b - Refueling Boron Concentration)

CORE OPERATING LIMITS REPORT (COLR) (Continued)

d. NRC Safety Evaluation Report dated March 10, 1993, for the "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station" (ET 92-0032, ET 93-0017)

(Methodology for Specification 3.1.3.6 - Control Rod Insertion Limits; Specification 3.2.1 - Axial Flux Difference)

e. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7" (NA 92-0073, NA 93-0013, NA 93-0054)

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$, [Use of WRB-2 Correlation with VIPRE-01 Code])

f. NRC Safety Evaluation Report dated November 13, 1986, for "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code" (WCAP-10266-P-A, Rev. 2)

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$

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

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

SPECIAL REPORTS

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

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

RECORD RETENTION (Continued)

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

RECORD RETENTION (Continued)

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those Unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the Unit Staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Manual;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the PSRC and the NSRC;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

DOCKET NO. 50-482

WOLF CREEK GENERATING STATION

1.0 INTRODUCTION

By application dated October 28, 1992, and supplemented by letters dated January 28, 1993 and March 8, 1993, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes reflect the use of VANTAGE 5H fuel with intermediate flow mixers, the licensee's performance of nuclear, thermal-hydraulic and safety analyses using a revised methodology and incorporating some changes to key assumptions, and the relocation of cycle specific parameters to a Core Operating Limits Report (COLR).

The most significant changes include increases in allowable core peaking based on revised thermal-hydraulic and accident analyses; introduction of an allowable positive moderator temperature coefficient; a decrease in the reactor coolant thermal design flow; an increase in the main steam safety valve setpoint tolerance; an increase in required shutdown margin in Mode 5, cold shutdown, to address boron dilution concerns; and the adoption of the Core Operating Limits Report.

The supporting analyses were performed assuming an increase in rated thermal power and a range of reactor coolant system temperatures. The technical specifications and related plant changes related to these parameters are not included in the proposed changes addressed by this Safety Evaluation.

The January 28 and March 8, 1993 submittals provided clarifying information and did not change the initial no significant hazards considerations determination published in the Federal Register on January 6, 1993.

2.0 EVALUATION

The fuel assembly design being introduced for Cycle 7 operation is a standard Westinghouse design which has been utilized at other facilities and has been reviewed and approved by the staff (Ref. 1). The analyses supporting the proposed changes to the technical specifications were performed using the NRC approved methodologies which are listed in the proposed revision to Technical Specification 6.9.1.9, which would add the requirement to submit a Core Operating Limits Report.

An exception to the use of approved methodologies is the Wolf Creek Nuclear Operating Corporation's transient analysis methodology topical report. The staff's review of the transient analysis topical report is nearing completion and no significant issues have been identified. The staff's review of the transient analysis topical report is sufficiently complete to conclude that the analysis performed in support of Cycle 7 operation was performed using an acceptable methodology. Technical Specification 6.9.1.9 has been modified from the licensee's amendment request to reflect the pending approval of the licensee's "Transient Analysis Methodology for the Wolf Creek Generating Station." This change was discussed with the licensee and was determined to be acceptable.

The only significant change from the specifics of an NRC reviewed methodology is the core thermal-hydraulics performed for the Cycle 7 reload design. The evaluation of the Cycle 7 core thermal-hydraulics model is provided below and this Safety Evaluation has been added to the list of approved methodologies provided by Technical Specification 6.9.1.9.

The introduction of the intermediate flow mixing grids (IFMs) associated with the new fuel assemblies being inserted for Cycle 7 required the licensee to repeat the process described in Reference 2 including the modelling of the IFMs and use of the associated critical heat flux correlation (WRB-2). The methodology outlined in the approved topical report was followed but the specific DNBR limits changed as a result of the revised fuel design and use of the WRB-2 correlation. Upon the determination of the DNBR limit, performance of the statistical core design and determination of operating limits was performed in accordance with the approved methodology.

The licensee performed a qualification of the NRC approved WRB-2 correlation (Ref. 3) using the VIPRE-01 thermal-hydraulic analysis code. This qualification was performed in the same manner as was used for the WRB-1 correlation in Reference 2. The licensee's analyses resulted in the determination of a DNBR correlation limit (to provide 95% probability at 95% confidence) of 1.14 for the WRB-2 correlation using the VIPRE-01 computer code. In accordance with the staff's Safety Evaluation for the VIPRE-01 code (Ref. 4), the vendor's correlation limit of 1.17 [WRB-2 and the THINC code, (Ref. 3)] will be utilized since the licensee's limit was determined to be less than the original correlation limit using the vendor's computer code. The staff has reviewed the licensee's analyses and has determined that a correlation limit of 1.17 is appropriate for use by WCNOC with the WRB-2 correlation and VIPRE-01 computer code.

Following the determination of the DNBR correlation limit, the licensee reperformed the statistical core design analyses which had been presented in Reference 2. This included the incorporation of uncertainties as discussed in Reference 2 with the exception of the axial peaking factors. The presence of the IFMs and their effect on predicted DNBR was stated to result in difficulties in obtaining an acceptable fit for the response surface model. The licensee has therefore treated the uncertainties associated with axial peaking in the traditional manner of a conservative DNBR penalty. The incorporation of the remaining parameters was consistent with the approved methodology of Reference 2. The resultant DNBR statistical design limit for the WRB-2/VIPRE-01 analysis is 1.31. The accounting for the axial peaking uncertainties, transition core penalty, lower plenum flow anomaly, rod bow penalty, and design margin results in a DNBR thermal design limit (TDL) of 1.80. The TDL value of 1.80 is used in subsequent thermal-hydraulic analyses to ensure that 95% protection at 95% confidence is provided by DNBR related operating limits and protection systems. The staff's review of the licensee's analyses has determined that the analyses are consistent with the methodology presented in Reference 2 and can be used for Cycle 7 and subsequent reload analyses.

The specific technical specification changes proposed by the licensee can be divided into four major categories. These are:

1) Changes resulting from the licensee's reload design methodologies;

2) Changes in key analysis assumptions incorporated into the licensee's analyses and loss of coolant accident analysis;

3) Adoption of the Core Operating Limits Report; and

4) Editorial changes resulting from the above major changes.

Each of the major changes is discussed below. The editorial changes required due to the referencing of technical specifications which have been revised or otherwise necessary for clarification have also been reviewed and determined to be acceptable.

TS 1-2: Core Operating Limits Definition

The proposed change introduces a definition for the Core Operating Limits Report. The staff's evaluation determined that the proposed change is consistent with the guidance in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." The staff finds the proposed change acceptable.

TS 2-1: Figure 2.1-1

The proposed change in the figure results from changes in the assumptions related to parameters such as reactor coolant system flow rate and maximum allowable peaking factors. The changes also result from physical differences in the fuel design and the associated critical heat flux correlation used to predict the occurrence of departure from nuclear boiling (DNB). In addition, the change in the limits reflect the use of the licensee's methodology, including statistical core design, instead of the analytical methodology associated with the existing technical specifications. The limits were determined using the licensee's approved methodology (Ref. 2) as discussed above for the specific application to Cycle 7. The assumptions were adequately justified and, where appropriate, were incorporated in other proposed technical specification changes. The staff finds the proposed change acceptable.

TS 2.2: Table 2.2-1

Several terms associated with the Overtemperature Delta-T (OTDT) and Overpower Delta-T (OPDT) setpoints were changed to reflect changes incorporated into the transient analyses, the revised safety limits provided in TS Figure 2.1-1, and revised setpoint calculations performed by the licensee. The methodology used to determine the setpoints is in accordance with the licensee's approved methodology (Ref. 2) as well as established procedures and practices used in the determination of protection system setpoints. The revised OTDT and OPDT time constants were incorporated into the reanalysis of Updated Safety Analysis Report (USAR) Chapter 15 transients and were determined to provide adequate protection against violation of core safety limits or other applicable acceptance criteria. The f(delta-I) penalties were determined in accordance with the approved methodology addressed by Reference 5. The staff finds the proposed changes acceptable.

TS 3/4 1.1.1 and TS 3/4 1.1.2: Boration Control

The proposed change increases the required shutdown margin in Mode 5, cold shutdown, to address concerns regarding the boron dilution transient and the boron dilution mitigation system. The reanalysis of the boron dilution transients is presented in the supporting documentation. The staff determined that the increase in required shutdown margin for Mode 5 operation was appropriate to address identified problems with the previous boron dilution analysis. The inclusion of Mode 5 in TS 3/4 1.1.1 and deletion of TS 3/4 1.1.2 is considered appropriate given that the shutdown margin requirements have become the same for the various modes of operation.

TS 3/4 1.1.3: Moderator Temperature Coefficient (MTC)

The MTC technical specification increases the beginning of cycle (most positive) limit from a constant 0 pcm/deg F to a limit which is a function of power level with the most positive limit being 6 pcm/deg F from 0% rated thermal power (RTP) to 70% RTP. The limit decreases linearly from 6 pcm/deg F at 70% RTP to 0 pcm/deg F at 100% RTP. This positive limit is introduced to the technical specifications as Figure 3.1-1. The value of the end of cycle (most negative) MTC limit (-41 pcm/deg F) has been transferred to the Core Operating Limits Report in accordance with guidance of Generic Letter 88-16. The supporting analyses have shown that the limits associated with both the most positive and most negative values of MTC are acceptable in that all FSAR Chapter 15 transient analyses continue to meet the applicable acceptance criteria. The staff finds the proposed increase in the positive MTC limit and the relocation of the most negative MTC limit to the COLR to be acceptable.

TS 3/4 1.3.5/1.3.6: Shutdown and Control Rod Insertion Limits

The only change to TS 3/4 1.3.5 and TS 3/4 1.3.6 is that the actual insertion limits are relocated to the COLR. No other changes to the limiting conditions for operation, required actions, or surveillance requirements are introduced. This relocation is consistent with the guidance of Generic Letter 88-16 and is acceptable to the staff.

TS 3/4 2.1: Axial Flux Difference (AFD)

The only change to TS 3/4 2.1 is that Figure 3.2-1, "Axial Flux Difference Limits as a Function of Rated Thermal Power," is relocated to the COLR. No other changes to the limiting conditions for operation, required actions, or surveillance requirements are introduced. This relocation is consistent with the guidance of Generic Letter 88-16 and is acceptable to the staff.

TS 3/4 2.2: Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$)

TS 3/4 2.2 is changed to reflect the licensee's performance of the nuclear analysis and related surveillances in accordance with the approved methodology described in Reference 5. The change is similar to the Babcock and Wilcox methodology described in Reference 6. In addition to the methodology related changes, the relocation of the F_Q limits and related parameters to the COLR is also proposed. The staff's review of the proposed changes concluded that, with the exception of several minor terminology and plant specific implementation differences, the proposed changes are consistent with the model technical specification included in Reference 6. The staff finds the proposed changes acceptable.

It is noted that the licensee has utilized a more recent loss of coolant accident evaluation model (Ref. 7) in order to increase the F_Q limit from 2.32 to 2.50. The evaluation model has been approved by the staff and is included in the list of approved methodologies provided in TS 6.9.1.9.

TS 3/4 2.3: Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X,Y)$

TS 3/4 2.3 is changed to reflect the licensee's performance of the nuclear analysis and related surveillances in accordance with the approved methodology described in Reference 5. The change is similar to the Babcock and Wilcox methodology described in Reference 6. In addition to the methodology related changes, the relocation of the $F_{\Delta H}$ limits and related parameters to the COLR is also proposed. The staff's review of the proposed changes concluded that, with the exception of several minor terminology and plant specific implementation differences, the proposed changes are consistent with the model technical specification included in Reference 6. As part of the changes, requirements associated with reactor coolant system flow rate are moved to TS 3/4 2.5, DNB parameters. The staff finds the proposed changes acceptable.

It is noted that the licensee has included increased nuclear enthalpy rise assumptions in the analysis for Cycle 7. The $F_{\Delta H}$ limit has been increased from 1.55 to 1.65. The increase has been justified by the supporting analysis and related NRC approved methodologies. The related topical reports as well as this Safety Evaluation have been included in the list of approved methodologies provided by TS 6.9.1.9.

TS 3/4 2.5: DNB Parameters

The basic change to TS 3/4 2.5 is the addition of reactor coolant flow rate requirements to the specification. The required flow rate has been reduced from the current value of 384,400 gpm to 384,000 gpm. The reduced flow was assumed in the supporting analyses, including the core thermal-hydraulic analyses which determined the core safety limits and protection system setpoints. In addition to adding the reactor coolant flow requirement to the limiting condition for operation, the existing action and surveillance requirements associated with reactor coolant system flow (existing TS 3/4 2.4) have been added to TS 3/4 2.5. The staff determined that the existing action and surveillance requirements remain applicable and therefore finds the proposed change acceptable.

TS 3/4 7.1: Table 3.7.2, Steam Line Safety Valves Per Loop

The proposed change involves a revision to the required as found lift setting of the main steam safety valves from +/- 1% of the setpoint to +/- 3% of the setpoint. The increased tolerance was included in the supporting analyses and it was determined that adequate protection from secondary system over-pressurization was maintained. In order to prevent exceeding the setpoint by an excessive amount over the course of several cycles, the proposed change includes a requirement to leave the main steam safety valves within +/- 1% of the setpoint following inservice testing. The testing requirements, including the proposed as found and as left tolerances, will continue to meet the requirements of the licensee's ASME Section XI inservice testing program. The staff finds the proposed changes acceptable.

TS 3/4 9.1: Boron Concentrations (Refueling Operations)

The only change to TS 3/4 9.1 is that required boron concentration for the reactor coolant system and refueling canal during refueling operations is relocated to the COLR. No other changes to the limiting conditions for operation, required actions, or surveillance requirements are introduced. This relocation is consistent with the guidance of Generic Letter 88-16 and is acceptable to the staff.

TS 3/4 9.12: Figure 3.9-1, Burnup Vs. Enrichment

TS Figure 3.9-1 provides the required fuel assembly burnup as a function of initial enrichment for storage of a spent fuel assembly in Region 2 of the spent fuel storage pool. The figure needed to be reanalyzed due to the replacement of inconel spacer grids with zircaloy spacer grids when going from the Westinghouse Standard to VANTAGE 5H fuel assembly design. The previous figure had included credit for the neutron absorption characteristics for the inconel spacer grids. The maximum initial enrichment assumption was reduced from 4.50 weight percent to 4.45 weight percent when the spent fuel pool criticality analyses were performed for the transition to VANTAGE 5H fuel assemblies. The net effect of the change in spacer grid material and the reduction in the maximum initial enrichment is a minor change to the burnup versus enrichment limits provided in Figure 3.9-1. The staff finds the proposed change acceptable.

TS 5.3: Design Features - Fuel Assemblies

The proposed change deletes the maximum enrichment value from TS 5.3. However, the maximum enrichment value is retained in TS 5.6, Fuel Storage. The proposed change is a deletion of a redundant specification and is acceptable to the staff.

TS 5.6: Design Features - Fuel Storage

The proposed change deletes the specific value for the uncertainty allowance in the calculation of spent fuel pool subcriticality but retains the reference to the applicable FSAR section. In addition, Figure 5.5-1, which is a duplication of TS Figure 3.9-1, is deleted and the TS references Figure 3.9-1. The maximum enrichment limit is reduced from 4.50 weight percent to 4.45 weight percent to account for insertion of VANTAGE 5H fuel assemblies into the spent fuel pool. The reduction in the enrichment limit as well as the editorial changes to this specification are acceptable to the staff.

TS 6.9.1.9: Core Operating Limits Report

The proposed change deletes the requirement for the submittal of a radial peaking factor limit report and introduces the requirement for the submittal of a Core Operating Limits Report (COLR), which documents specific values of cycle-specific parameters using NRC-approved methodologies. The licensee's proposed change is in accordance with the guidance of Generic Letter 88-16 and lists the associated technical specifications and methodologies approved by the staff for the determination of cycle-specific parameters. The requirement that only the listed NRC-approved methodologies may be used to determine core operating limits in the COLR ensures safe operation of the facility within approved acceptance criteria. Therefore, removal of cycle-specific parameters, as proposed, is acceptable.

Based on the above, the staff finds the proposed TS changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment. The state official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located with the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 601 dated January 6, 1993). In addition, the amendment changes recordkeeping or reporting requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. WCAP-10444-P-A, Addendum 2A, "VANTAGE 5H Fuel Assembly," Westinghouse Electric Corporation, February 1989.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to Topical Report TR-90-0025 W01, Core Thermal-Hydraulic Analysis Methodology for the Wolf Creek Generating Station, October 29, 1992.

3. WCAP-10444-P-A, "VANTAGE 5 Fuel Assembly Reference Core Report," Westinghouse Electric Corporation, September 1985.

4. Letter from Rossi, C.E. (NRC) to Blaisdell, J.A. (URGA), "Acceptance for Referencing Licensing Topical Report, EPRI NP-2511-CCM, "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," May 1, 1986.

5. NSAG-007, "Wolf Creek Nuclear Operating Corporation Reload Safety Evaluation Methodology for Wolf Creek Generating Station, Wolf Creek Nuclear Operating Corporation," March 11, 1992, as supplemented by Forrest T. Rhodes' letter dated February 3, 1993.

6. BAW-10163P-A, "Core Operating Limits Methodology for Westinghouse Designed PWRs," Babcock & Wilcox, June 1989.

7. WCAP-10266-P-A, Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code," March 1987.

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