LIST OF FIGURES

Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

p. <u>DOSE EQUIVALENT I-131</u>

DOSE EQUIVALENT I-131 is that concentration of I-131 (μ Ci/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 as listed and calculated based on dose conversion factors derived from ICRP-30.

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0059	I-132
0.1692	I-133
0.0010	I-134
0.0293	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

1. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

2. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the nominal zero power design temperature.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the Integrity of the fuel cladding.

SPECIFICATION

a. The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation.

c. The peak fuel centerline temperature shall be maintained < 4700 °F.

3 Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \; [\; K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f (\Delta I) \;]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, %
T	=	Average temperature, °F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.
$f(\Delta I)$	=	An even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f(\Delta I) = 0$.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average Temperature, °F

T' ≤ [*]°F

K_4 ≤ [*]

K_5 ≥ [*] for increasing T; [*] for decreasing T

K_6 ≥ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop ≥ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint ≥ 55.0 Hz

2. Low voltage setpoint ≥ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level ≥ 5% of narrow range instrument span.

B. A vent pathway shall be provided with an effective flow cross section ≥ 6.4 square inches.

1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, then verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

A. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

B. $\leq \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

A. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding 60 μCi/gram DOSE EQUIVALENT I-131, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within six hours.

B. With the specific activity of the reactor coolant $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within six hours.

C. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 *or* $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Steam and Power Conversion System.

OBJECTIVE

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

SPECIFICATION

a. Main Steam Safety Valves (MSSVs)

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of two MSSVs per steam generator are OPERABLE.

2. The reactor shall not be made critical unless five MSSVs per steam generator are OPERABLE.

3. If the conditions of TS 3.4.a.1 or TS 3.4.a.2 cannot be met within 48 hours, then within 1 hour initiate action to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

b. Auxiliary Feedwater System

1. The Reactor Coolant System shall not be heated > 350°F unless the following conditions are met:

   A. Auxiliary feedwater train "A" and auxiliary feedwater train "B" are OPERABLE and capable of taking suction from the Service Water System and delivering flow to the associated steam generator.

   B. The turbine-driven auxiliary feedwater train is OPERABLE and capable of taking suction from the Service Water System and delivering flow to both steam generators, OR

   The turbine-driven auxiliary feedwater train is declared inoperable.

   C. The auxiliary feedwater pump low discharge pressure trip channels are OPERABLE.

2. When the Reactor Coolant System temperature is > 350°F, any of the following conditions of inoperability may exist during the time interval specified:

   A. One auxiliary feedwater train may be inoperable for 72 hours.

   B. Two auxiliary feedwater trains may be inoperable for 4 hours.

   C. One steam supply to the turbine-driven auxiliary feedwater pump may be inoperable for 7 days.

3. When the Reactor Coolant System temperature is > 350°F, an auxiliary feedwater pump low discharge pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded, the associated auxiliary feedwater train shall be declared inoperable and the OPERABILITY requirements of TS 3.4.b.2 applied.

4. When the Reactor Coolant System temperature is > 350°F, if three auxiliary feedwater trains are discovered to be inoperable, initiate immediate action to restore one auxiliary feedwater train to OPERABLE status and suspend all LIMITING CONDITIONS FOR OPERATION requiring MODE changes until one auxiliary feedwater train is restored to OPERABLE status.

5. If the OPERABILITY requirements of TS 3.4.b.2 above are not met within the times specified, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

6. When reactor power is < 15% of RATED POWER, any of the following conditions may exist without declaring the corresponding auxiliary feedwater train inoperable:

   A. The auxiliary feedwater pump control switches located in the control room may be placed in the "pull out" position.

   B. Valves AFW-2A and AFW-2B may be in a throttled or closed position.

   C. Valves AFW-10A and AFW-10B may be in the closed position.

c. Condensate Storage Tank

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of 39,000 gallons of water is available in the condensate storage tanks.

2. If the Reactor Coolant System temperature is > 350°F and a minimum of 39,000 gallons of water is not available in the condensate storage tanks, reactor operation may continue for up to 48 hours.

3. If the time limit of TS 3.4.c.2 above cannot be met, within 1 hour initiate action to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

d. Secondary Activity Limits

1. The Reactor Coolant System shall not be heated > 350°F unless the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators is ≤ 0.1 μCi/gram.

2. When the Reactor Coolant System temperature is > 350°F, the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators may exceed 0.1 μCi/gram for up to 48 hours.

3. If the requirement of TS 3.4.d.2 cannot be met, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure: 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the SHUTDOWN MARGIN shall be at least that as specified in the COLR

b. Power Distribution Limits

1. At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:

   A. $F_Q^N(Z)$ Limits shall be as specified in the COLR.
   B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If $F_{\Delta H}^N$ not within limits:

   A. Perform the following:

      i. Within 4 hours either, restore $F_{\Delta H}^N$ to within its limit or reduce thermal power to less than 50% of RATED POWER.

      ii. Reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED POWER within 72 hours.

      iii. Verify $F_{\Delta H}^N$ within limits within 24 hours.

   B. If the actions of TS 3.10.b.2.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power increases may proceed provided that $F_{\Delta H}^{N}$ is demonstrated, through incore flux mapping, to be within its limits prior to exceeding the following thermal power levels:

   i. 50% of RATED POWER,
   ii. 75% of RATED POWER, and
   iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If the $F_Q^N(Z)$ equilibrium relationship is not within its limit:

   A. Reduce the thermal power ≥ 1% RATED POWER for each 1% the $F_Q^N(Z)$ equilibrium relationship exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% for each 1% $F_Q^N(Z)$ equilibrium relationship exceeds its limit.

   B. If the actions of TS 3.10.b.3.A are not completed within the specified time, then reduce thermal power to ≤ 5% of RATED POWER within the next 6 hours.

   C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $F_Q^{EQ}(Z)$ transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

4. Power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied. (Note: time requirements may be extended by 25%)

   A. For $F_Q^N(Z)$ equilibrium relationship, once after each refueling prior to thermal power exceeding 75% of RATED POWER; and once within 12 hours after achieving equilibrium conditions, after exceeding, by ≥ 10% of RATED POWER, the thermal power at which the $F_Q^N(Z)$ equilibrium relationship was last verified; and 31 effective full power days thereafter.

   B. For $F_{\Delta H}^{N}$, following each refueling prior to exceeding 75% RATED POWER and 31 effective full power days thereafter.

5. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

6. Power distribution maps using the movable detector system shall be made to confirm the relationship of $F_Q^{EQ}(Z)$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

   A. Once after each refueling prior to exceeding 75% RATED POWER and every 31 effective full power days thereafter.

   B. Once within 12 hours of achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.6.A.

C. If a power distribution map measurement indicates that the F_Q^{EQ} (Z) transient relationship's margin to the limit, as specified in the COLR, has decreased since the previous evaluation, then either of the following actions shall be taken:

i. F_Q^{EQ} (Z) transient relationship shall be increased by the penalty factor specified in the COLR for comparison to the transient limit as specified in the COLR and reverified within the transient limit, or

ii. Repeat the determination of the F_Q^{EQ} (Z) transient relationship once every seven effective full-power days until either i. above is met, or two successive maps indicate that the F_Q^{EQ} (Z) transient relationship's margin to the transient limit has not decreased.

7. If, for a measured F_Q^{EQ}, the transient relationships of F_Q^{EQ} (Z) specified in the COLR are not within limits, then take the following actions:

A. Reduce the axial flux difference limits ≥ 1% for each 1% the F_Q^{EQ} (Z) transient relationship exceeds its limit within 4 hours after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. If the actions of TS 3.10.b.7.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Verify the $F_Q^N(Z)$ equilibrium relationship and the F_Q^{EQ} (Z) transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

i. If the axial flux difference is not within limits, reduce thermal power to less than 50% RATED POWER within 30 minutes.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant total flow rate shall be ≥ 178,000 gallons per minute average and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, at or above 90% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

FIGURE TS 3.1-3
DELETED

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
43. AFW Pump Low Discharge Pressure Trip	Not Applicable	Each refueling cycle	Each refueling cycle	
44. Axial Flux Difference (AFD)	Weekly			Verify AFD within limits for each OPERABLE excore channel

5.2 CONTAINMENT

APPLICABILITY

Applies to those design features of the Containment System relating to operational and public safety.

OBJECTIVE

To define the significant design features of the Containment System.

SPECIFICATION

a. Containment System

1. The Containment System completely encloses the entire reactor and the Reactor Coolant System and ensures that leakage of activity is limited, filtered and delayed such that off-site doses resulting from the design basis accident are within the guidelines of 10 CFR Part 50.67. The Containment System provides biological shielding for both normal OPERATING conditions and accident situations.

2. The Containment System consists of:

   A. A free-standing steel reactor containment vessel designed for the peak pressure of the design basis accident.

   B. A concrete shield building which surrounds the containment vessel, providing a shield building annulus between the two structures.

   C. A Shield Building Ventilation System that causes leakage from the reactor containment vessel to be delayed and filtered before its release to the environment.

   D. An Auxiliary Building Special Ventilation System that serves the special ventilation zone and supplements the Shield Building Ventilation System during an accident condition by causing any leakage from the Residual Heat Removal System (RHRS) and certain small amounts of leakage that might be postulated to bypass the Shield Building Ventilation System to be filtered before their release.

5.3 REACTOR CORE

APPLICABILITY

Applies to the reactor core.

OBJECTIVE

To define those design features which are essential in providing for safe reactor core operations.

SPECIFICATION

a. Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy, ZIRLO™, or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-test-assemblies that have not completed representative testing may be placed in non-limiting core regions. Lead-test-assemblies shall be of designs approved by the NRC for use in pressurized water reactors and their clad materials shall be the materials approved as part of those designs.

b. Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

(3) S.M. Bajorek, et al., WCAP-12945-P-A (Proprietary), Westinghouse Code Qualification Document for Best-Estimate Loss-of –Coolant Accident Analysis, Volume I, Rev. 2, and Volume II-V, Rev.1, and WCAP-14747 (Non-Proprietary) March 1998.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(10) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(11) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(12) S.I. Dederer, et al., WCAP-14449-P-A, Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Rev. 1 (Proprietary and WCAP-14450-NP-A, Rev. 1 (Non-Proprietary), October 1999.

(13) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

(14) WCAP-11397-P-A, "Revised Thermal Design Procedure, "April 1989.

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

   A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

   Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

   A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

      (1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

BASIS - Safety Limits-Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.

BASIS - Safety Limit - Reactor Coolant System Pressure (TS 2.2)

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is ensured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure, 2485 psig) has been established.[2]

The settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to prevent exceeding the SAFETY LIMIT of 2735 psig for all transients except the hypothetical RCCA Ejection accident, for which the faulted condition stress limit acceptance criterion of 3105 psig (3120 psia) is applied. The initial hydrostatic test was conducted at 3107 psig to ensure the integrity of the Reactor Coolant System.

(1) USAR Section 4
(2) USAR Section 4.3

Maximum Coolant Activity (TS 3.1.c)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is limited to ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[15] are analyzed assuming an RCS activity of 1.0 μCi/gram DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike of 60 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. The results obtained from these analyses indicate the control room and off-site dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 50.67 limits.

The action of reducing average reactor coolant temperature to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

[15] USAR Section 14.0

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-sink-event.

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System and supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on analyses performed by WPSC and the Westinghouse Electric Corporation. These analyses conservatively assumed the plant was at 100% initial power and demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident with no steam generator dryout or core damage. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER to provide further margin in the analysis.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses to fulfill the above functions. Since the AFW System is a safety features system, the backup pump is provided. This redundant motor-driven capability is also supplemented by the turbine-driven pump.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.2. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications.

For all other design basis accidents, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

Condensate Storage Tank (TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

[1] USAR Section 8.2.4

Secondary Activity Limits (TS 3.4.d)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The secondary side of the steam generator's activity is limited to ≤ 0.1 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

[2] USAR Section 14.0

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

$F_Q^{EQ}(Z)$ is the measured $F_Q^N(Z)$ obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for $F_Q^N(Z)$ as specified in the COLR has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures, with a high probability, remain less than the 2200 °F limit.

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses. The LOCA analyses are performed for Westinghouse 422 V+ fuel, FRA-ANP heavy fuel and for FRA-ANP standard fuel.

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The $F_{\Delta H}^N$ limit is determined from safety analyses of the limiting DNBR transient events. The safety analyses are performed for FRA-ANP heavy fuel, FRA-ANP standard fuel, and Westinghouse 422 V+ fuel. In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is greater than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^N$ limit.

The use of $F_{\Delta H}^N$ in TS 3.10.b.5.C is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

No penalty for rod bow effects needs to be included in TS 3.10.b.1 for FRA-ANP fuel.[1]

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as specified in the COLR.

3. The control bank insertion limits as specified in the COLR are not violated, except as allowed by TS 3.10.d.2.

4. The axial power distribution, expressed in terms of axial flux difference, is maintained within the limits.

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the FQ(Z) upper bound envelope of FQLIMIT times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor program. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and reactor power is greater than 50 percent or RATED POWER.

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit by an automatic Protection System. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

[1] N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.

Quadrant Power Tilt Limits (TS 3.10.c)

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The 2% tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This symmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level. This monitoring is required by TS 4.1.

The two hour time interval in TS 3.10.c is considered ample to identify a dropped or misaligned rod. If the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core power distribution map using the movable detector system. For a tilt ratio > 1.02 but ≤1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of 2% for each 1% of indicated tilt is required. Power distribution measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment. If a tilt ratio of > 1.02 but ≤1.09 cannot be eliminated after 24 hours, then the reactor power level will be reduced to ≤50%.

If a misaligned rod has caused a tilt ratio > 1.09, then the core power shall be reduced by 2% of rated value for every 1% of indicated power tilt ratio > 1.0. If after eight hours the rod has not been realigned, then the rod shall be declared inoperable in accordance with TS 3.10.e, and action shall be taken in accordance with TS 3.10.g. If the tilt condition cannot be eliminated after 12 hours, then the reactor shall be brought to a minimum load condition; i.e., electric power ≤ 30 MW. If the cause of the tilt condition has been identified and is in the process of being corrected, then the generator may remain connected to the grid.

If the tilt ratio is > 1.09, and it is not due to a misaligned rod, then the reactor shall be brought to a no load condition (i.e., reactor power ≤ 5%) for investigation by flux mapping. Although the reactor may be maintained critical for flux mapping, the generator must be disconnected from the grid since the cause of the tilt condition is not known, or it cannot be readily corrected.

Rod Insertion Limits (TS 3.10.d)

The allowed completion time of two hours for restoring the control banks to within the insertion limits provides an acceptable time for evaluation and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

Operation beyond the rod insertion limits is allowed for a short-time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss-of-flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The time limits of six hours to achieve HOT STANDBY and an additional six hours to achieve HOT SHUTDOWN allow for a safe and orderly shutdown sequence and are consistent with most of the remainder of the Technical Specifications.

Rod Misalignment Limitations (TS 3.10.e)

During normal power operation it is desirable to maintain the rods in alignment with their respective banks to provide consistency with the assumption of the safety analyses, to maintain symmetric neutron flux and power distribution profiles, to provide assurance that peaking factors are within acceptable limits and to assure adequate shutdown margin.

Analyses have been performed which indicate that the above objectives will be met if the rods are aligned within the limits of TS 3.10.e. A relaxation in those limits for power levels < 85% is allowable because of the increased margin in peaking factors and available shutdown margin obtained while OPERATING at lower power levels. This increased flexibility is desirable to account for the nonlinearity inherent in the rod position indication system and for the effects of temperature and power as seen on the rod position indication system.

Rod position measurement is performed through the effects of the rod drive shaft metal on the output voltage of a series of vertically stacked coils located above the head of the reactor pressure vessel. The rod position can be determined by the analog individual rod position indicators (IRPI), the plant process computer which receives a voltage input from the conditioning module, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

The plant process computer converts the output voltage signal from each IRPI conditioning module to an equivalent position (in steps) through a curve fitting process, which may include the latest actual voltage-to-position rod calibration curve.

The rod position as determined by any of these methods can then be compared to the bank demand position which is indicated on the group step counters to determine the existence and magnitude of a rod misalignment. This comparison is performed automatically by the plant process computer. The rod deviation monitor on the annunciator panel is activated (or reactivated) if the two position signals for any rod as detected by the process computer deviate by more than a predetermined value. The value of this setpoint is set to warn the operator when the Technical Specification limits are exceeded.

The rod position indicator system is calibrated once per REFUELING cycle and forms the basis of the correlation of rod position vs. voltage. This calibration is typically performed at HOT SHUTDOWN conditions prior to initial operations for that cycle. Upon reaching full power conditions and verifying that the rods are aligned with their respective banks, the rod position indication may be adjusted to compensate for the effects of the power ascension. After this adjustment is performed, the calibration of the rod position indicator channel is checked at an intermediate and low level to confirm that the calibration is not adversely affected by the adjustment.

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, then the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

Core Average Temperature (TS 3.10.k)

The core average temperature limit is consistent with full power operation within the nominal operational envelope. Either Tavg control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A higher Tavg will cause the reactor core to approach DNB limits.

Reactor Coolant System Pressure (TS 3.10.l)

The RCS pressure limit is consistent with operation within the nominal operational envelope. Either pressurizer pressure control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A lower pressure will cause the reactor core to approach DNB limits.

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant system (RCS) flow limit, as specified in the COLR, is consistent with the minimum RCS flow limit assumed in the safety analysis adjusted by the measurement uncertainty. The safety analysis assumes initial conditions for plant parameters within the normal steady state envelope. The limits placed on the RCS pressure, temperature, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the analyzed transients.

The RCS flow normally remains constant during an operational fuel cycle with all reactor coolant pumps running. At least two plant computer readouts from the loop RCS flow instrument channels are averaged per reactor coolant loop and the sum of the reactor coolant loop flows are compared to the limit. Operating within this limit will result in meeting the DNBR criterion in the event of a DNB-limited event.

DNBR Parameters (TS 3.10.n)

The DNBR related safety analyses make assumptions on reactor temperature, pressure, and flow. In the event one of these parameters does not meet the TS 3.10.k, TS 3.10.l or TS 3.10.m limits, an analysis can be performed to determine a power level at which the MDNBR limit is satisfied.