ATTACHMENT 8

HBR2
CYCLE 10 RELOAD

TECHNICAL SPECIFICATION
CHANGES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists.

b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 590°F.

c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 620°F.

d. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in Figure 2.1-1 or if the thermal power level, coolant pressure, or Reactor Coolant System average temperature violates the limits specified above.

Basis

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 maintaining 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, thermal power, reactor coolant temperature and pressure, have been related to DNB through the XNB DNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, 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 DNB ratio, DNBR, during normal operational transients and anticipated transients is limited to 1.17. A DNB ratio of 1.17 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.(1) The DNB ratio limit of 1.17 is a conservative design limit which is used as a basis for setting core safety limits. Based on rod bundle DNB tests, no fuel rod damage is expected at this DNB ratio or greater.

The curves of Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is not less than 1.17. The area where clad integrity is assured is below these lines. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperature are shown for each pressure at powers lower than approximately 75%. The temperature limits at low power are

considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.17 but are such that the plant conditions required to violate the limits are precluded by the self actuated safety valves on the steam generators. An arbitrary upper safety limit of 118% thermal power is shown. This limit is based on the high flux trip including all uncertainties.

Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculations of the curves shown in Figure 2.1-1. The figure also includes the effects of uprating to 2300 MWt.[3]

The limits specified for one loop operation and natural circulation result in DNB ratios greater than 1.17.

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 covered by Specification 3.10. Somewhat worse hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.10-1 ensure that the DNB ratio is always greater at part power than at full power.

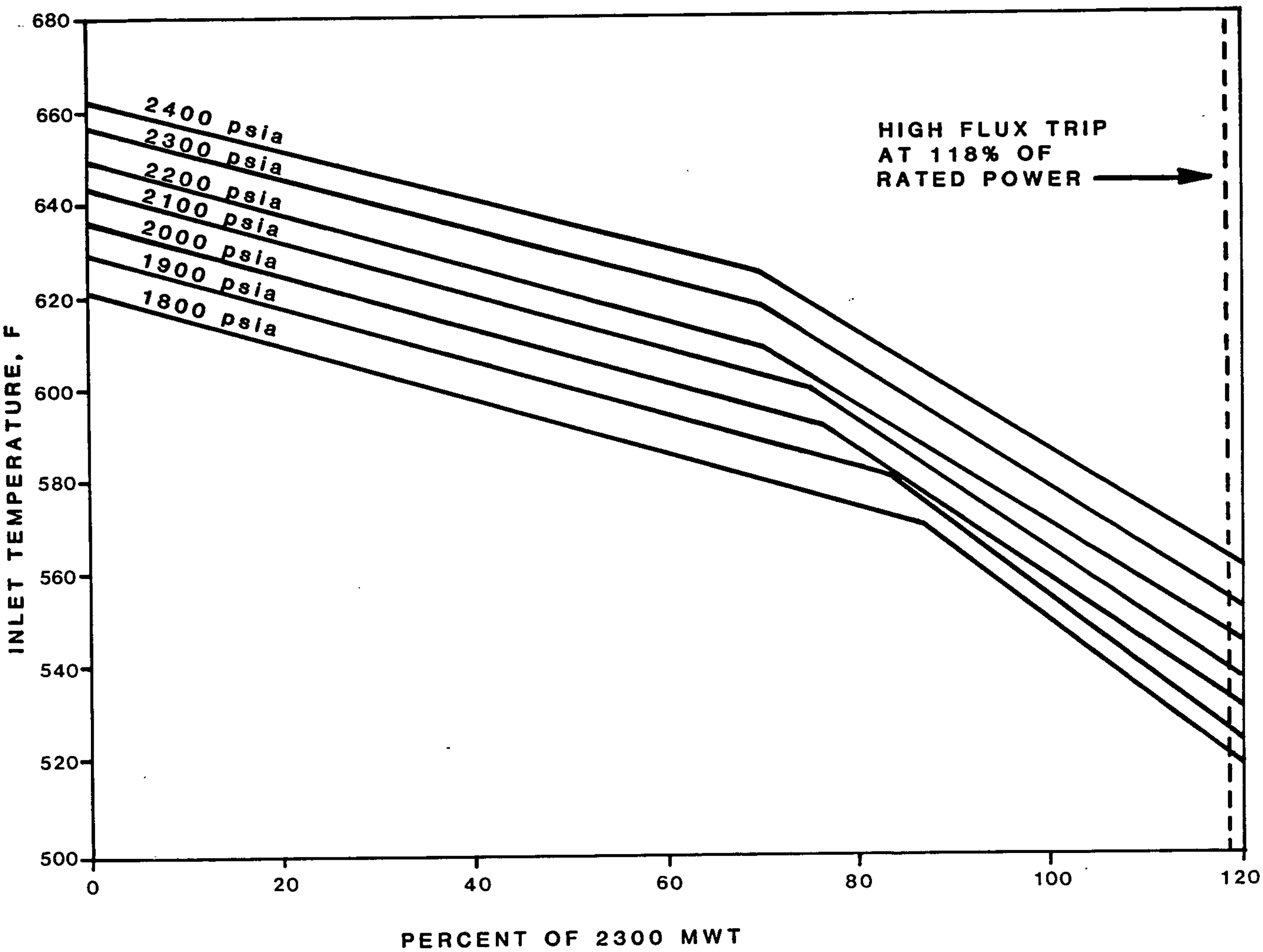
The safety limit curves given in Figure 2.1-1 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the FSAR.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.17[2] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of

+2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

References

(1) XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."
(2) FSAR Section 15.
(3) WCAP-8243, "H. B. Robinson Unit 2 - Justification for Operation at 2300 Mwt; December 1973."



CORE PROTECTION BOUNDARIES FOR 3-LOOP OPERATION
Figure 2.1-1

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow and pressurizer level.

Objective

To provide for automatic protection action in the event that the principal process variables approach a safety limit.

Specification

2.3.1 Protective instrumentation settings for reactor trip shall be as follows:

2.3.1.1 Start-up protection

a. High flux, power range (low setpoint) $\leq$ 25% of rated power.

2.3.1.2 Core protection

a. High flux, power range (high setpoint) $\leq$ 109% of rated power

b. High pressurizer pressure $\leq$ 2385 psig.

c. Low pressurizer pressure $\geq$ 1835 psig.

d. Overtemperature Δ T

$$\leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} (T - T') + K_3 (P - P') - f(\Delta I) \right\}$$

where:

ΔT_o	= Indicated ΔT at rated thermal power;
T	= Average temperature, °F;
P	= Pressurizer pressure, psig;
K_1	$<$ 1.1565;
K_2	= 0.01228;
K_3	= 0.00089;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	= The function generated by the lead-lag controller for T_{avg} dynamic compensation;
τ_1 & τ_2	= Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 20 seconds, τ_2 = 3 seconds;
T'	= 575.4°F Reference T_{avg} at rated thermal power;
P'	= 2235 psig (Nominal RCS Operating Pressure);
S	= Laplace transform operator, sec^{-1};

and f(ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start-up tests such that:

(1) For (q_t - q_b) within +12% and -17%, where q_t and q_b are 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 (2300 Mwt), f(ΔI) = 0. For every 2.4% below rated power (2300 Mwt) level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power (2300 Mwt) level, the permissible negative flux difference range is extended by -1 percent.

(2) For each percent that the magnitude of (q_t - q_b) exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17%, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

e. Overpower Δ T

$$\leq \Delta T_o \{ K_4 - K_5 (\frac{\tau_3 S}{1 + \tau_3 S}) T - K_6 (T - T') - f(\Delta I) \}$$

where:

ΔT_o = Indicated ΔT at rated thermal power, °F;
T = Average temperature, °F;
T' = 575.4°F Reference T_{avg} rated thermal power;
K_4 < 1.07;
K_5 = 0.0 for decreasing average temperature, 0.02 sec/°F for increasing average temperature;
K_6 = 0.00277 for T > T' and 0 for $T \leq T'$;
S = Laplace transform operator, sec^{-1};
$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation;
τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, τ_3 = 10 seconds;
$f(\Delta I)$ = As defined in d. above

f. Low reactor coolant loop flow $\geq$ 90% of normal indicated flow.

g. Low reactor coolant pump frequency $\geq$ 57.5 Hz.

h. Undervoltage $\geq$ 70% of normal voltage.

2.3.1.3 Other Reactor Trips

a. High pressurizer water level $\leq$ 92% of span.

b. Low-low steam generator water level $\geq$ 14% of narrow range instrument span.

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power.

2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

Basis

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from lower power. This trip value was used in the safety analysis.(1)

In the power range of operation, the overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.(2)

The source and intermediate range reactor trips do not appear in the specification, as these settings are not used in the transient and accident analysis (FSAR Section 15). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss-of-coolant accident.(3)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[4], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to Specification 2.3.1.2.d.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed in Section 7.2.2 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The setpoints in the Technical Specifications ensure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figure 2.1-1.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5] The undervoltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error[2] and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[6]

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 45% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.17 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification, as these settings are not used in the transient and accident analysis. (FSAR Section 15)

<u>References</u>

(1) FSAR Section 15.4
(2) FSAR Section 15.0
(3) FSAR Section 15.6
(4) FSAR Section 15.4.2
(5) FSAR Section 15.3
(6) FSAR Section 15.2

3.0 LIMITING CONDITIONS FOR OPERATION

Except as otherwise provided for in each specification, if a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in hot shutdown within eight hours and in COLD SHUTDOWN within the next 30 hours unless corrective measures are taken that permit operation under the permissible Limiting Condition for Operation statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. With reactor power less than 2% of rated thermal power and less than two reactor coolant pumps in operation, one of the following actions shall be taken:

   1. maintain a shutdown margin of at least 4% $\Delta k/k$
   2. open the lift disconnect switches for all control rods not fully withdrawn, or
   3. open reactor trip breakers.

b. Power operation with less than three loops in service is prohibited.

c. At least one reactor coolant pump or residual heat removal pump shall be in operation when $T_{avg} > 200°F$ and reactor power is less than 2% of rated thermal power. In the event this condition cannot be satisfied, the following action shall be taken.

   1. Proceed to establish a boron concentration in the reactor coolant equal to or greater than that concentration needed to maintain a shutdown margin of 1% $\Delta k/k$ at 200°F.

d. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the steam generator temperature is no higher than 50°F higher than the temperature of the reactor coolant system.

Basis

Specification 3.1.1.1.a contains requirements designed to limit the consequences of the uncontrolled bank withdrawal at low or subcritical power conditions as analyzed in the safety analysis. The requirement of two reactor coolant pumps in operation below 2% power is consistent with the assumptions utilized in the bounding transient that was analyzed. The specification makes allowance for less than two pumps in operation by specifying either of three actions that must be taken. Either maintaining the specified shutdown margin, opening the lift disconnect switches on the control rods or opening the reactor trip breakers will prevent the occurrence of the postulated uncontrolled bank withdrawal transient, therefore allowing the two pump requirement to be lifted.

Maintaining a shutdown margin of 4% $\Delta k/k$ is sufficient to prevent a return to criticality if the worth of the two most reactive control rod banks are simultaneously withdrawn as is the assumption of the postulated transient.

Specification 3.1.1.1.b requires that all three reactor coolant pumps be operating during power operation to provide core cooling in the event that a loss of flow occurs. The flow provided will keep DNB well above 1.17. Therefore, cladding damage and release of fission products to the reactor coolant will not occur.

Specification 3.1.1.1.c is designed to allow for adequate mixing of the reactor coolant to maintain a uniform boron concentration during dilution, and to provide a means of boron injection. Should no residual heat removal pump or reactor coolant pump be available, boration via natural circulation shall be initiated. A boron concentration corresponding to 1% $\Delta k/k$ at 200°F (which assumes most reactive rod stuck out) would prevent a return to criticality during the cooldown phase of the postulated steam line break event. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

The purpose of Specification 3.1.1.1.d is to limit pressure surges exhibited in the RCS during a RCP startup. These pressure surges can be controlled in one of two ways. One method would be to require a steam bubble in the pressurizer and thus control pressure using pressurizer controls. The other method would be to limit the temperature difference (< 50°F) between the RCS average temperature and the idle pump's cold leg water temperature.

3.1.1.2 Steam Generator

At least two steam generators shall be operable whenever the average primary coolant temperature is above 350°F.

Basis

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The reactor cannot be made critical without water in all three steam generators, since the low-low steam generator water level trip prevents this mode of operation. Two operable steam generators are therefore adequate.

3.1.1.3 Pressurizer (Pzr)

a. At least one Pzr code safety valve shall be operable whenever the Reactor Head is on the vessel and the RCS is not open for maintenance.

b. The Pzr, including necessary spray and heater control systems, shall be operable before the reactor is made critical.

c. Whenever the RCS temperature is above 350°F or the reactor is critical:

1. All three pressurizer code safety valves shall be operable. Their lift settings shall be maintained between 2485 psig and 2560 psig.

2. At least 125 kw of pressurizer heaters capable of being powered from an emergency power source shall be operable.

d. If the requirements of 3.1.1.3.c.2 are not met and at least 125 kw or Pzr heaters capable of being powered from an emergency source cannot be provided within 72 hrs., commence a normal plant shutdown and cooldown to an RCS average temperature of less than or equal to 350°F.

Basis

The pressurizer is necessary to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve setpoint.[1] Below 350°F and 450 psig in the Reactor Coolant System (RCS), the Residual Heat Removal System can remove decay heat and thereby control system temperature. The pressurizer

safety valves are sized to protect the RCS against overpressure without taking credit for the steam bypass system.[2] If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than the capacity of a single valve. One valve therefore provides adequate defense against overpressurization of the RCS for primary coolant temperatures less than 350°F and two valves provide protection for any temperature.

ASME Section III of the Code allows a maximum variation in the setpoint of 3 percent above the design set pressure.

The requirement that 125 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency power source provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

References

(1) FSAR Table 5.4.6-1
(2) FSAR Section 15.2

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +5.0 pcm/°F less than 50% of rated power, or
b) +5.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specified in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant

pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[1] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as calculated by ASME Code Appendix G methods, and an additional safety margin of 40°F must be maintained above the applicable heatup curve at all times.

If the specified shutdown margin is maintained (Section 3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of one percent subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

(1) FSAR Section 4.3

TABLE 3.5-1

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1.	High Containment Pressure (HI Level)	Safety Injection*	$\leq$ 5 psig
2.	High Containment Pressure (HI-HI Level)	a. Containment Spray** b. Steam Line Isolation	$\leq$ 25 psig
3.	Pressurizer Low Pressure	Safety Injection*	$\geq$ 1700 psig
4.	High Differential Pressure Between any Steam Line and the Steam Line Header	Safety Injection*	$\leq$ 150 psi
5.	High Steam Flow in 2/3 Steam Lines***	a. Safety Injection* b. Steam Line Isolation	$\leq$ 40% (at zero load) of full steam flow $\leq$ 40% (at 20% load) of full steam flow $\leq$ 110% (at full load) of full steam flow
	Coincident with Low T_{avg} or Low Steam Line Pressure		$\geq$ 541°F T_{avg} $\geq$ 600 psig steam line pressure
6.	Loss of Power		
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage) Time Delay	Trip Normal Supply Breaker	328 Volts $\pm$ 1 Volt .75 $\pm$ .25 sec.

TABLE 3.5-1 (Continued)

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6. (Cont'd)	b. 480V Emerg. Bus Undervoltage (Degraded Voltage) Time Delay	Trip Normal Supply Breaker	412 Volts $\pm$ 1 Volt 10.0 Second Delay $\pm$ 0.5 sec.
7.	Containment Radioactivity High	Ventilation Isolation	$\leq$ 2 X Reading at the Time the Alarm is Set with Known Plant Conditions

* Initiates also containment isolation (Phase A), feedwater line isolation and starting of all containment fans.

** Initiates also containment isolation (Phase B).

*** Derived from equivalent ΔP measurements.

TABLE 3.5-3 (Continued)

INSTRUMENTATION OPERATING CONDITIONS FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
2.	CONTAINMENT SPRAY			
	a. Manual*	2	0**	Cold Shutdown
	b. High Containment Pressure* (Hi-Hi Level)	2/set	1/set	Cold Shutdown
3.	LOSS OF POWER			
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage)	2/bus[a]	1/bus[b]	Main Hot Shutdown
	b. 480V Emerg. Bus Undervoltage (Degraded Voltage)	2/bus	1/bus	Maintain Hot Shutdown[c]

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* Also initiates a Phase B containment isolation.
** Must actuate two switches simultaneously.
*** When primary pressure is less than 2000 psig, channels may be blocked.
**** When primary temperature is less than 547°F, channels may be blocked.
***** In this case the 2/3 high steam flow is already in the trip mode.
(a) During testing and maintenance of one channel, may be reduced to 1/bus.
(b) During testing and maintenance of one channel, may be reduced to 0/bus.
(c) The reactor may remain critical below the power operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specification

3.6.1 Containment Integrity

a. The containment integrity (as defined in 1.7) shall not be violated unless the reactor is in the cold shutdown condition.

b. The containment integrity shall not be violated when the reactor vessel head is removed unless a shutdown margin greater than 10% $\Delta k/k$ is constantly maintained.

c. Positive reactivity changes shall not be made by rod drive motion when the containment integrity is not intact except for rod drop tests in which case the shutdown margin is maintained $\geq$ 1% $\Delta k/k$.

d. Positive reactivity changes shall not be made by boron dilution when the containment integrity is not intact unless the shutdown margin is maintained $\geq$ 1% $\Delta k/k$.

3.6.2 Internal Pressure

If the internal pressure exceeds 1 psi or the internal vacuum exceeds 1.0 psi, the condition shall be corrected within eight (8) hours or the

Regarding internal pressure limitations, the containment design pressure of 42 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 2 psig.[(1)] The containment is designed to withstand an internal vacuum of 2.0 psi.[(2)]

References

(1) FSAR Section 6.2.1
(2) FSAR Section 3.8.1.3

The relative humidity (R.H.) of the air processed by the refueling filter systems should be less than the R.H. used during the testing of the charcoal adsorbers in order to assure that the adsorbers will perform under accident conditions as predicted by the test results. Heaters have been installed upstream of the Spent Fuel Building filters to assure of an R.H. of less than 70 percent for the air processed by the Spent Fuel Building filter system. If an R.H. in the containment atmosphere exceeds 70 percent, operation of the containment purge system will be terminated until this specification can be met. If the Spent Fuel Building filter system is found to be inoperable, all fuel handling and fuel movement operations in the Spent Fuel Building will be terminated until the system is made operable.

The temperature limit specified for the fuel cask handling crane is based on the recorded ambient temperature at the time of the 125% load test. The limit is imposed to assure adequate toughness properties of the crane structural materials.

References

(1) FSAR Section 9.4.1
(2) FSAR Section 4.3
(3) FSAR Section 9.4.1
(4) H. B. Robinson Unit 2 Radiological Assessment of Postulated Accidents, XN-NF-84-68(P), July 1984.

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions $\geq$ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

3.10.2 <u>Power Distribution Limits</u>

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < 4.64 \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < 1.65\ (1 + 0.2(1-P)\ )$

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q(Z)$ including the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}$ including a 1.04 measurement uncertainty factor. K(Z) is based on the function given in Figure 3.10-3, and Z is the axial location of F_Q.

3.10.2.1.1 Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.2 $F_Q(Z)$ shall be determined to be within the limit given in 3.10.2.1 by satisfying the following relationship for the middle axial 80% of the core at the time of the target flux determination:

$$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5$$

$$F_Q(Z) < 4.64 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5$$

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

where V(Z) is defined in Figure 3.10-4 which corresponds to the target band and $P > 0.5$.

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference

b) Reduce the reactor power by the maximum percent calculated with the following expression for the middle axial 80% of the core:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.32}{P} \times K(Z)}\right] -1 \right] \times 100\%$$

c) Comply with the requirements of Specification 3.10.2.2.2.

3.10.2.2.2 The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:

$$\text{APL} = \text{minimum over Z of } \frac{2.32 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

where $F_Q(Z)$ is the measured $F_Q(Z)$, including the engineering factor $F_Q^E = 1.03$ and the measurement uncertainty factor $F_u^N = 1.05$ at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the variation function defined in Figure 3.10-4 which corresponds to the target band. K(Z) is the function defined in Figure 3.10-3.

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

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

At power levels in excess of APL of rated power, the APDMS will be employed to monitor $F_Q(Z)$. The limiting value is expressed as:

$$\left[F_j(Z)\ S(Z)\right]_{max} \leq \frac{2.103/P}{\overline{R}_j\ (1 + \sigma_j)}$$

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$).

b. $\overline{R}_j$, for thimble j, is determined from core power maps and is by definition:

$$\overline{R}_j = \frac{1}{6} \sum_{i=1}^{6} \frac{F_{Qj}}{\left[F(Z)_{ij}\ S(Z)\right]_{max}}$$

F_{Qj} is the value obtained from a full core map including S(Z), but without the measurement uncertainty factor F_u^N or the engineering uncertainty factor, F_Q^E. The quantity $F(Z)_{ij}$ S(Z) is the measured value without inclusion of the instrument uncertainty factors F_Q^a. Those uncertainty factors, $F_u^N = 1.05$, $F_Q^a = 1.02$, as well as the engineering factor $F_Q^E = 1.03$, have been included in the limiting value of 2.103/P.

c. σ_j is the standard deviation associated with the determination of $\overline{R}_j$.

d. S(Z) is the inverse of the K(Z) function given in Figure 3.10-3.

This limit is not applicable during physics tests and excore detector calibrations.

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F_{\Delta H}^N$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

3.10.2.2.2 or $F_Q(Z)$ shall be measured and a target axial flux difference re-established at least once every seven (7) effective full power days until two successive measurements indicate enthalpy rise hot channel factor, $F^N_{\Delta H}$, is not increasing.

3.10.2.3 The reference equilibrium-indicated axial flux difference as a function of power level (called the target flux difference) shall be determined in conjunction with the measurement of $F_Q(Z)$ as defined in Specification 3.10.2.1.1.*

3.10.2.4 The indicated axial flux difference shall be considered outside of the limits of Sections 3.10.2.5 through 3.10.2.9 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.

3.10.2.5 Except during physics tests, and except as modified by 3.10.2.6 through 3.10.2.9 below, the indicated axial flux difference shall be maintained within the applicable target band about the target flux difference (defines the target band on axial flux difference).

3.10.2.6 At a power level greater than 90 percent or 0.9 x APL** (whichever is less) of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent or 0.9 x APL (whichever is less) of rated power.

3.10.2.7 At a power level between 50 percent and 90 percent or 0.9 x APL (whichever is less) of rated power,

---

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

** APL is the Allowable Power Level defined in Specification 3.10.2.2.2.

a. The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed the limits shown in Figure 3.10-5. If the cumulative time exceeds one hour, then the reactor power shall be reduced immediately to no greater than 50 percent of rated power and the high neutron flux setpoint reduced to no greater than 55 percent of rated power.

b. A power increase to a level greater than 90 percent or 0.9 x APL (whichever is less) of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.8 At a power level no greater than 50 percent of rated power

a. The indicated axial flux difference may deviate from its target band.

b. A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributing to the one-hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent or 0.9 x APL (whichever is less) of rated power.

3.10.2.9 Calibration of the excore detectors will be performed at a power level no greater than 90% or 0.9 x APL (whichever is less) of rated power. The indicated axial flux difference may deviate from its target band during the calibration provided the flux difference does not exceed the limits shown in Figure 3.10-5.

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are shown in Figure 3.10-4. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference.

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50 percent of rated power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint to be less two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

b. If the tilt condition is not eliminated after 24 hours, the power range high flux setpoint shall be reset to 55 percent of rated power. Subsequent reactor operation would be permitted up to 50 percent of rated power for the purpose of measurement and testing to identify the cause of the tilt condition.

3.10.3.2 Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:

a. The core power level shall be reduced by 2 percent of rated values for every 1 percent of indicated power tilt exceeding 1.0, and

b. If the tilt condition is not eliminated within two hours, the reactor shall be brought to a hot shutdown condition.

c. After correction of the misaligned rod, reactor operation will be permitted to 50 percent of rated power until the indicated quadrant tilt falls below 1.09.

3.10.3.3 If the indicated quadrant tilt exceeds 1.09 and there is not a simultaneous indication of rod misalignment, except as stated in Specification 3.10.3.2.c, the reactor shall immediately be brought to a hot shutdown condition.

equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kW/ft. Second, the minimum DNBR in the core must not be less than 1.17 in normal operation or in short term transients.

In addition to the above, the initial steady state conditions for the peak linear power for a Loss-of-Coolant Accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the ECCS Acceptance Criteria. To aid in specifying the limits on power distribution the following hot channel factors are defined.

a. F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

b. F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

c. F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface

e. Axial power distribution control procedures, which are given in terms of axial flux difference within the target band about the target flux difference, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of axial offset which is defined as the difference in power between the top and bottom halves of the core.

For operation at a fraction P of full power, the design limits are met, provided the limits of Specification 3.10.2.1 are not exceeded.

The permitted relaxation in $F_{\Delta H}^{N}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions a through e are observed, these hot channel factors limits are met.

The procedures for axial power distribution control referred to above include operator control of flux difference to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a value which corresponds to the full power target flux difference established in conjunction with incore power distribution measurements. The target flux difference varies with power level.

The target value of the flux difference is determined at equilibrium xenon conditions. The control rods must be positioned in accordance with their insertion limits. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and the specified deviation of ΔI is permitted from the indicated reference value. The periodic updating of the target flux difference is necessary to reflect the impacts of core burnup on power distribution.

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the allowable range shown in Figure 3.10-5 for 90 percent or 0.9 x APL (whichever is less). Therefore, while the deviation exists, the power level is limited to 90 percent or 0.9 x APL (whichever is less) of rated power or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled with the target band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent of rated power is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

An upper bound envelope of peaking factors has been determined from extensive analysis considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10.2 The specifications on power distribution control ensure that xenon distributions are not developed, which at a later time, could cause greater local power peaking even though the flux difference is then within limits. The results of a Loss-of-Coolant Accident analysis based on this upper bound envelope indicate that a peak clad temperature would not exceed the 2200°F limit. The nuclear analyses of credible power shapes consistent with the power distribution control procedures have shown that the F_Q^T limit is not exceeded.

For transient events, the core is protected from exceeding 21.1 kw/ft locally, and from going below a minimum of DNBR of 1.17 by automatic protection on power, flux difference, pressure and temperature.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_Q^N there is a 5 percent allowance for uncertainties(5) which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured F_Q^N 5 percent less than the limit, for example, at rated power even on a worst case basis. When a measurement is taken, experimental error must be allowed for, and 5 percent is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

In the specified limit of $F_{\Delta H}^N$, there is an 8 percent allowance for design prediction uncertainties, which means that normal operation of the core is expected to result in $F_{\Delta H}^N$ at least 8 percent less than the limit at rated power. The uncertainty to be associated with a measurement of $F_{\Delta H}^N$ by the movable incore system, on the other hand, is 4 percent, which means that the normal operation of the core shall result in a measured $F_{\Delta H}^N$ at least 4 percent less than the value at rated power. The logic behind the larger design uncertainty in the case is that (a) abnormal perturbation in the radial power shape (e.g., rod misalignment) affects $F_{\Delta H}^N$ in most cases without necessarily

affecting F_Q^N, and can limit it to the desired value; (b) while the operator has some control over F_Q^N through F_z^N by motion of control rods, he has no direct control over $F_{\Delta H}^N$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_Q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

Quadrant power tilts are based upon the following considerations. The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions, measured as part of the startup physics testing, 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 is consistent with the assumptions used in power capability analyses. It is not intended that extended reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

During normal plant startup, quadrant power tilt ratio may exceed 1.02 due to instrumentation instabilities as a result of rodded configurations and low excore detector signal levels below 50 percent of full power. Sustained power operation below 50 percent of full power would require a renormalization of the calculational methods for determining power tilt to compensate for change in signal levels once equilibrium conditions are met.

The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt conditions 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 physics map utilizing the movable detector system. For a tilt condition $\leq$ 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 two percent

for each one percent of indicated tilt is required. Physics 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.

In the event the tilt condition of 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for low power physics testing. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55 percent of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition. If a tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor power shall be brought to a hot shutdown condition for investigation.

However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2 percent for each one percent the tilt ratio exceeds 1.0) for the two-hour period necessary to correct the rod misalignment.

The specified rod drop time is consistent with safety analyses that have been performed.(1)

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

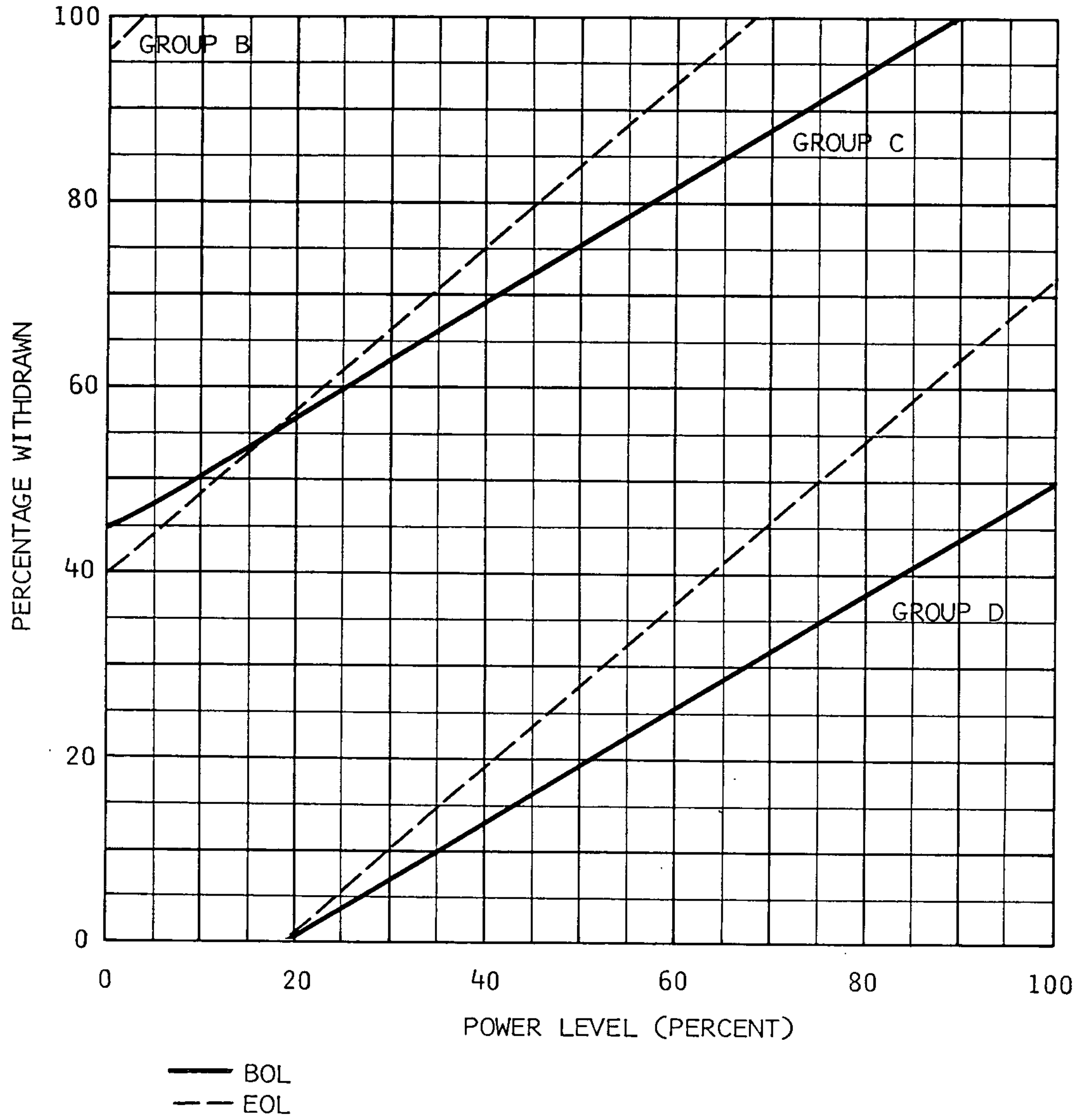
Normal reactor operation causes significant pellet cracking and fragmentation. Consequently, handling of irradiated fuel assemblies can result in relocation of these fragments against the cladding. Calculations show that high cladding stresses can occur if the reactor power increase is rapid during the subsequent startup.

The 72-hour period allows for stress relaxation of the clad before the ramp rate requirement is removed, thereby reducing the potential harmful effects of possible pellet or fragment relocation.

The 3 percent limit is imposed to minimize the effects of adverse cladding stresses resulting from part power operation for extended periods of time. The time period of 30 days is based upon the successful power ramp demonstrations performed on Zircaloy clad fuel in operating reactors, resulting in no cladding failures.

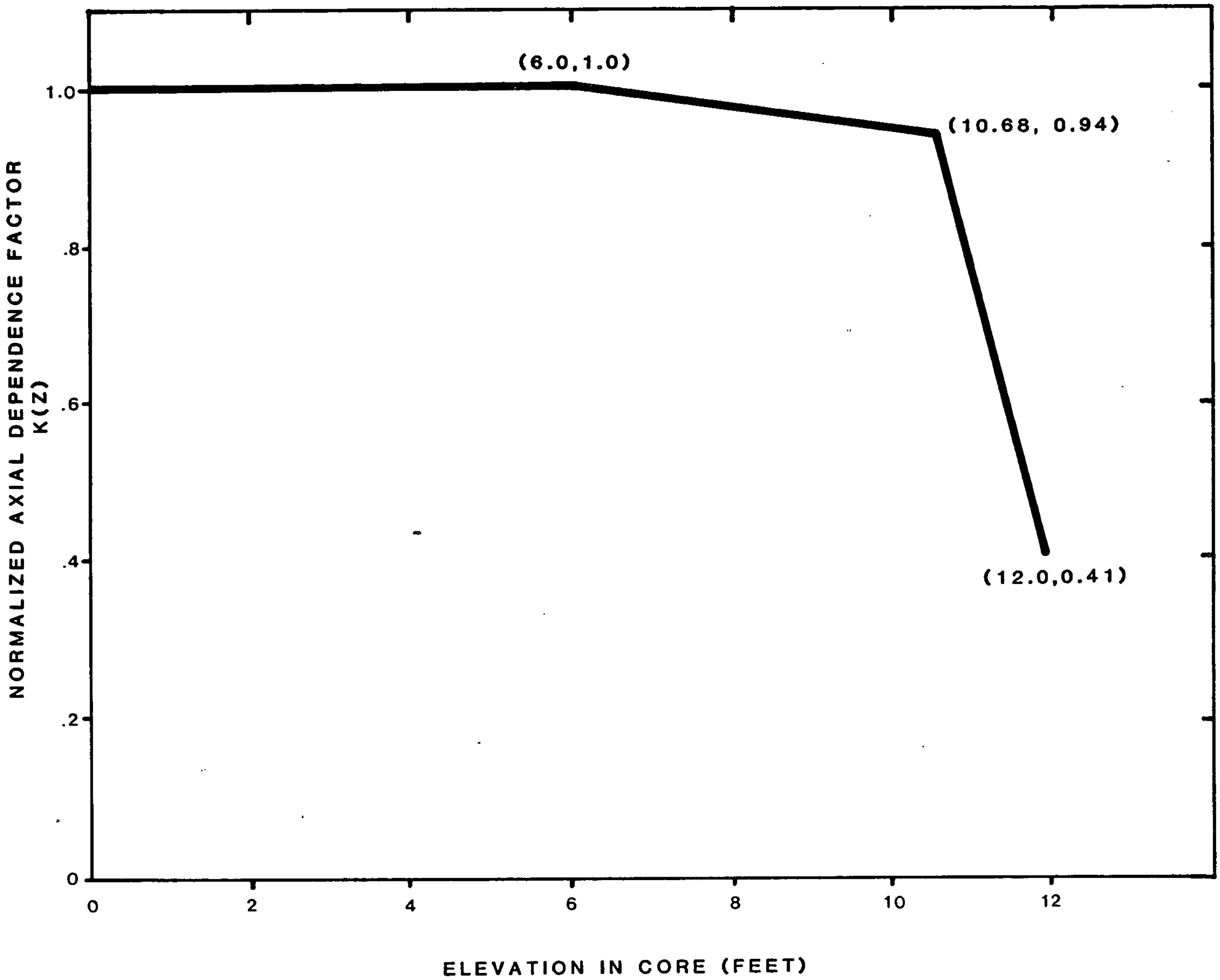
References

(1) FSAR Section 15.0
(2) FSAR Section 7.7
(3) FSAR Section 15.4
(4) FSAR Section 15.4
(5) FSAR Section 15



Amendment No.

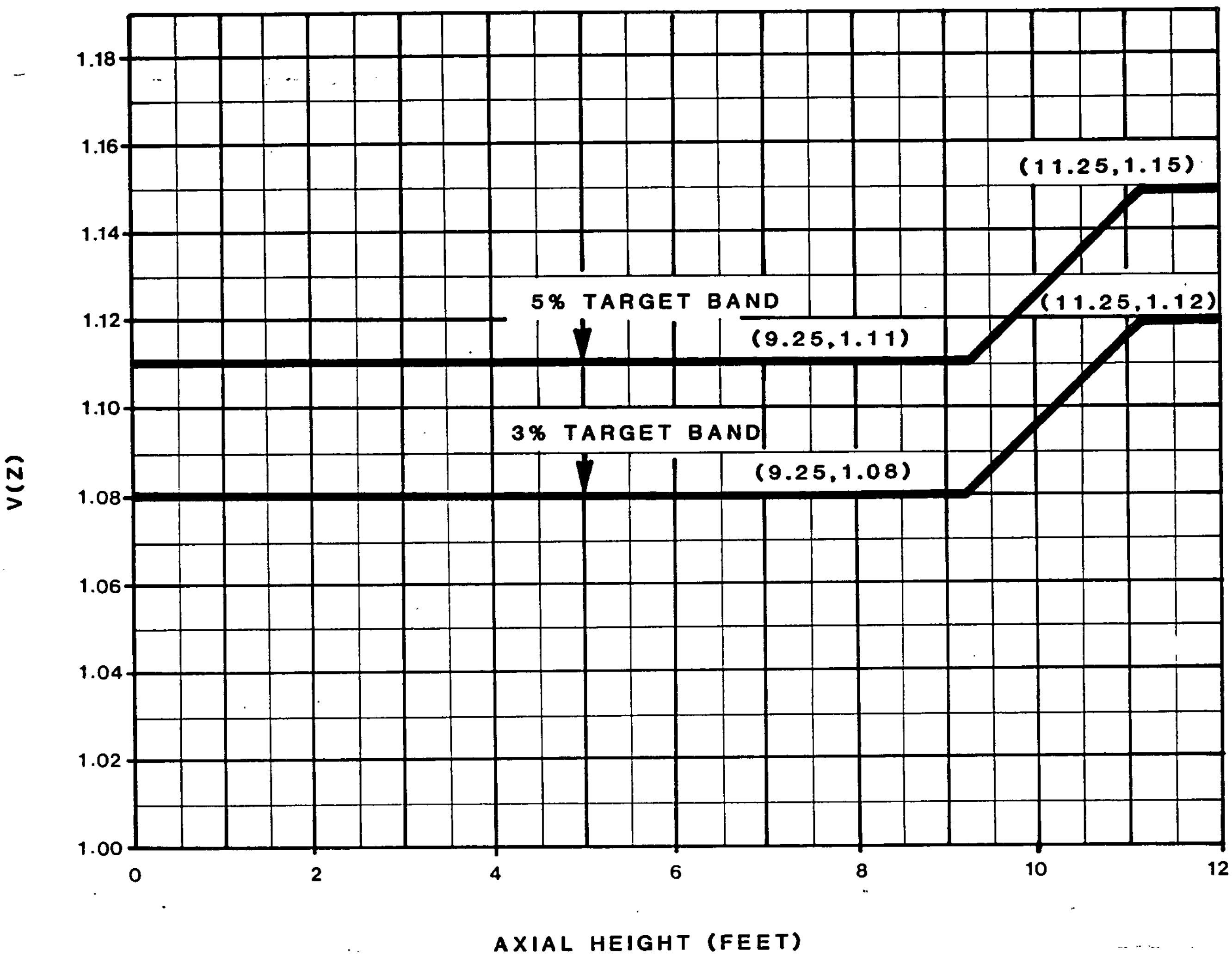
H.B. ROBINSON Unit #2 Carolina Power & Light Company **Technical Specifications**	**Control Group Insertion Limits For Three Loop Operation**	FIGURE **3.10-1**



NORMALIZED AXIAL DEPENDENCE FACTOR FOR
F_q = 2.32 VERSUS ELEVATION

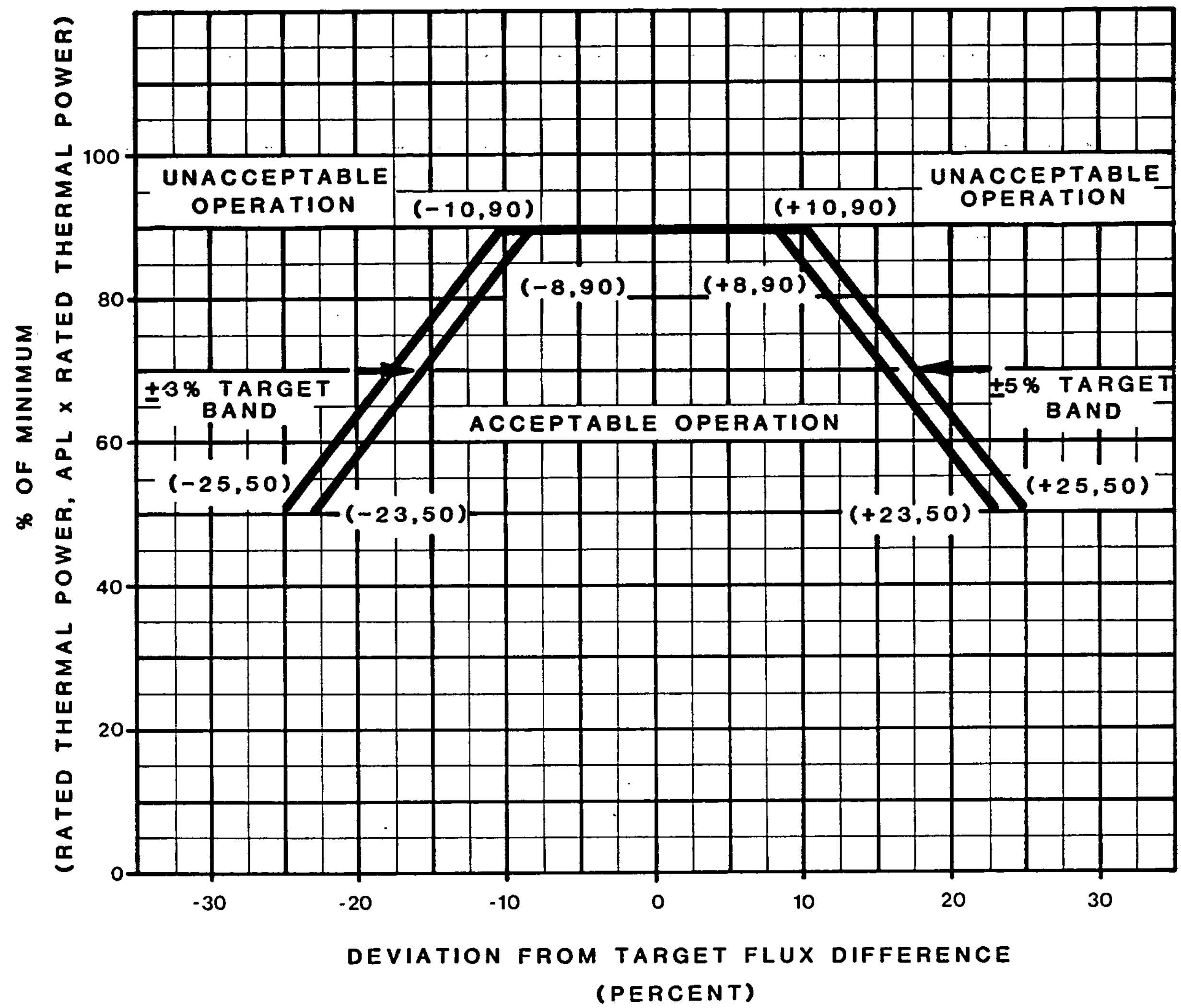
$F_{\Delta H}^{N} = 1.65$

Figure 3.10-3



V(Z) AS A FUNCTION OF CORE HEIGHT

Figure 3.10-4



ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE

Figure 3.10-5

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the in-core instrumentation systems, for the calibration of the excore symmetrical offset detection system.

Specification

3.11.1 A minimum of 16 total accessible thimbles and at least 2 per quadrant and sufficient movable in-core detectors shall be operable during recalibration of the excore symmetrical offset detection system.

3.11.2 Power shall be limited to 90% of rated power if recalibration requirements for the excore symmetrical offset detection system identified in Table 4.1-1 are not met.

Basis

The Movable In-Core Instrumentation System(1) has five drives, five detectors, and 48 thimbles in the core. Each detector can be routed to twenty or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

To calibrate the excore detector system, it is only necessary that the Movable In-Core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due, for example, to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it sill compensate for an error of 10% in the excore protection system. Experience at the Beznau No. 1 and R. E. Ginna plants has shown that drift due to the core on instrument channels is very slight. Thus, limiting the operating levels to 90% of the rated power is very conservative.

Reference

(1) FSAR Section 7.7.1.5

4.11 REACTOR CORE

Applicability

Applies to surveillance of the reactor core.

Objective

To ensure the integrity of the fuel cladding.

Specification

4.11.1 APDMS Operation

4.11.1.1 Prior to establishing normal operation with APDMS, at least six maps will be taken to determine applicable values of $\overline{R}$ and σ for surveillance thimbles.

4.11.1.2 Plant operation up to rated power shall be permitted for the purposes of obtaining the initial maps of Specification 4.11.1.1, provided the APDMS is operational and hot channel factors are shown to be below the limiting values set forth in Specification 3.10.2. Suitably conservative values of $\overline{R}$ and σ shall be derived from maps previously run during the current fuel cycle for use in the APDMS system during this initial period.

4.11.1.3 Subsequent updates of $\overline{R}$ and σ shall employ the last six maps in accordance with Specification 4.11.1.1.

4.11.1.4 Each power distribution map will be based on flux traverses obtained from 36 or more of the 48 monitoring channels.

4.11.2 Except during physics tests and EXCORE calibrations, axial surveillance of F(Z)S(Z) shall consist of traverses with the movable incore detectors in appropriate pairs of detector paths, taken every eight hours, or a frequency of approximately 0, 10,

30, 60, 120, 180, 240, 360, and 480 minutes following accumulated control rod motion in any one direction of five steps or more, exclusive of control rod movement within 15 steps from the top of the core. From the traverses, determination of F(Z)S(Z) shall be made and shown to result in a value less than the limiting value specified in 3.10.2. If the APDMS is out of service, reactor operation above APL of rated power can be continued for fourteen equivalent full power days provided that traverses are taken manually at equivalent frequencies, and a log of accumulated rod motion and time of manual traverses is kept.

4.11.3 The following criteria will be used for selecting the channels for measuring F(Z)S(Z):

a. The channel is not acceptable if it contains a control rod allowed by the insertion limits at power levels requiring APDMS.

b. For the latest full core power map, i, channels, j, are acceptable if:

$$\frac{R_{ij} - \overline{R}_j}{\overline{R}_j} \leq 2\sigma_j$$

Basis

The $\overline{R}$ technique provides a means for using many of the monitoring thimbles to determine $F_Q(Z)$ without fully mapping the core. Frequent core maps assure that appropriate values of $\overline{R}$ are being used for each thimble.

Upon return to power following a refueling outage or other situation where establishment of normal APDMS operation is required, power operation above APL of rated power is desirable to establish hot channel factors at full power.

By using maps that have been previously obtained during the power ascension and deriving conservative values of $\overline{R}$ and σ from these maps for use in the APDMS, operation of the plant within the peaking factor limitations can be ensured.

If the APDMS is out of service, adequate monitoring of the core power distribution can be maintained for a limited period of time by manual actuation of the flux mapping system and calculation of the values of F(Z)S(Z).

5.3 REACTOR

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 68 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods which are all pre-pressurized. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rod locations occupied by rods consisting of natural or slightly enriched uranium pellets, solid inert materials, or a combination of the aforementioned.(1)

5.3.1.2 Deleted.

5.3.1.3 Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.9 weight percent of U-235.

5.3.1.4 Deleted.

5.3.1.5 There are 45 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain 144 inch segments of silver-indium-cadmium alloy clad with the stainless steel.(2)

5.3.1.6 Up to 10 grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the Reactor Coolant System complies with the Code requirements.(3)

5.3.2.2 All piping, components and supporting structures of the Reactor Coolant System are designed to Class I requirements.

5.3.2.3 The nominal liquid volume of the Reactor Coolant System, at rated operating conditions, is 9343 cubic feet.[4]

References

(1) FSAR Section 4.2.1
(2) FSAR Section 4.2.2
(3) FSAR Table 3.2.2-1
(4) FSAR Table 5.1.0-1

SINGLE FAILURE ANALYSIS

Attachment 11

TABLE 1: SUMMARY OF THE RESULTS

Transients	Worst Single Failure	Transients Not Analyzed	Comment
		15.1.1 Feedwater Malfunctions that result in a decrease in feedwater temperature	This increase in heat removal by the secondary is not severe enough to drop Reactor Coolant System pressure to the SI setpoint. Reactor trip and turbine trip prevent drastic cooldown of reactor coolant system.
		15.1.2 Feedwater System Malfunctions that result in an increase in feedwater flow	This increase is heat removal by the secondary is not severe enough to drop Reactor Coolant System pressure to the SI setpoint. Reactor trip and turbine trip prevent drastic cooldown of the reactor coolant system.
15.1.3 Excessive Increase in Secondary Steam Flow	Steam Driven Auxiliary Feedwater Pump fails to deliver flow		
		15.1.4 Inadvertent Opening of SG Relief or PORV	Bounded by Excessive Increase in Secondary Steam Flow (15.1.3) and hand calculations.
15.1.5 Main Steamline Break with Loss of Offsite Power	For fuel thermal limits: failure of one of two diesel generators to start For containment integrity for MSLB inside containment: failure of one of two diesel generators to start		
Main Steamline Break with Offiste Power available	For fuel thermal limits: failure of one of three SI pumps For offsite dose for MSLB outside containment: continued normal feedwater injection at reduced flow		

(3610NH/pgp)

TABLE 1: SUMMARY OF THE RESULTS (Continued)

	Transients	Worst Single Failure		Transients Not Analyzed	Comment
15.2.1	Loss of External Electric Load	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.2.2	Turbine Trip	Bounded by Loss of Load (15.2.1)
			15.2.3	Loss of Condenser Vacuum and other events resulting in Turbine Trip	Bounded by Loss of Load (15.2.1)
			15.2.4	Inadvertent Closure of MSIV's	Bounded by Loss of Load (15.2.1)
			15.2.6	Loss of Non-Emergency AC Power to Station Auxiliaries	Bounded by Complete Loss of Flow (15.3.1) and Loss of Normal Feedwater Flow (15.2.7)
15.2.7	Loss of Normal Feedwater Flow	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.2.8	Feedwater System Pipe Break	Bounded by Steamline Break (15.1.5)
15.3.1	Loss of Forced Primary Coolant Flow	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
15.3.3	Reactor Coolant Pump Shaft Seizure	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.3.4	Reactor Coolant Pump Broken Shaft	Bounded by Shaft Seizure (15.3.3)
15.4.1	Uncontrolled RCCA Withdrawal from Subcritical or Low Power Startup Condition	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			
15.4.2	Uncontrolled RCCA Withdrawal from Power	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			

TABLE 1: SUMMARY OF THE RESULTS (Continued)

	Transients	Worst Single Failure		Transients Not Analyzed	Comment
15.4.3	RCCA Misoperation	Steam Driven Auxiliary Feedwater Pump fails to deliver flow			
			15.4.4	Startup of an Inactive Coolant Loop at Incorrect Temperature	Power operation with less than three loops is not allowed
15.4.6	CVCS Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant				The operational modes of refueling and startup are analyzed to show that adequate time exists to secure inadvertent boron dilution before criticality occurs. No ESF systems are involved.
			15.4.7	Inadvertent Loading of a Fuel Assembly into an Improper Location	Administrative procedures preclude occurrence of this event.
15.4.8	Spectrum of RCCA Ejection Accidents	Steam driven Auxiliary Feedwater Pump fails to deliver flow			
			15.5.1	Inadvertent Operation of ECCS	Shutoff head of high pressure SI pumps is 1500 psia < 1750 psia trip setpoint pressure.
			15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	Effect on Reactor Coolant System Pressure is completely mitigated by the Reactor Protection System and relief valves.
15.6.1	Inadvertent Opening of Pressurizer Safety or PORV	Failure of one of three SI pumps to deliver flow			
			15.6.2	Loss of Reactor Coolant from Rupture of Small Pipes or from Cracks in Large Pipes which actuate the ECCS	Bounded by large break LOCA (15.6.5)

TABLE 1: SUMMARY OF THE RESULTS (Continued)

	Transients	Worst Single Failure	Transients Not Analyzed	Comment
15.6.3	Steam Generator Tube Rupture	Steam driven Auxiliary Feedwater Pump fails to deliver flow		
15.6.5	LOCA	Failure of one diesel generator to start		

(3610NH/pgp)

XN-NF-84-68 (P)

H. B. ROBINSON UNIT 2
RADIOLOGICAL ASSESSMENT
OF POSTULATED ACCIDENTS

A F F I D A V I T

STATE OF Washington)
ss.
COUNTY OF Benton)

I, Richard B. Stout, being duly sworn, hereby say and depose:

1. I am Manager, Licensing and Safety Engineering, for Exxon Nuclear Company, Inc. ("ENC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with ENC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the document XN-NF-84-68(P) entitled "H.B. Robinson Unit 2 Radiological Assessment of Postulated Accidents" referred to as "Document". Information contained in this Document has been classified by ENC as proprietary in accordance with the control system and policies established by ENC for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by ENC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to Carolina Power and Light Company and the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of ENC and would be helpful to competitors of ENC when competing with ENC.

7. The information contained in the Document is considered to be proprietary by ENC because it reveals certain distinguishing aspects of radiological assessment procedures which secure competitive advantage to ENC for fuel design optimization and improved marketability, and includes information utilized by ENC in its business which affords ENC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it extremely valuable insights into radiological assessent procedures and would result in substantial harm to the competitive position of ENC.

9. The Document contains proprietary information which is held in confidence by ENC and is not available in public sources.

10. In accordance with ENC's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside ENC only as required and under suitable agreement providing for non-disclosure and limited use of the information.

11. ENC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. This Document provides information which reveals the radiological assessment procedures developed by ENC over the past several years. ENC has invested thousands of dollars and many man-years of effort in developing the BWR thermal hydraulic analysis methods revealed in the Document. Assuming a competitor had available the same background data and incentives as ENC, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as ENC.

13. Based on my experience in the industry, I do not believe that the background data and incentives of ENC's competitors are sufficiently similar to the corresponding background data and incentives of ENC to reasonably expect such competitors would be in a position to duplicate ENC's proprietary information contined in the Documents.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

Richard B Stout

SWORN TO AND SUBSCRIBED

before me this 10 day of

July, 1984.

Susan E. Backus

NOTARY PUBLIC