APPENDIX A TECHNICAL SPECIFICATION CHANGES

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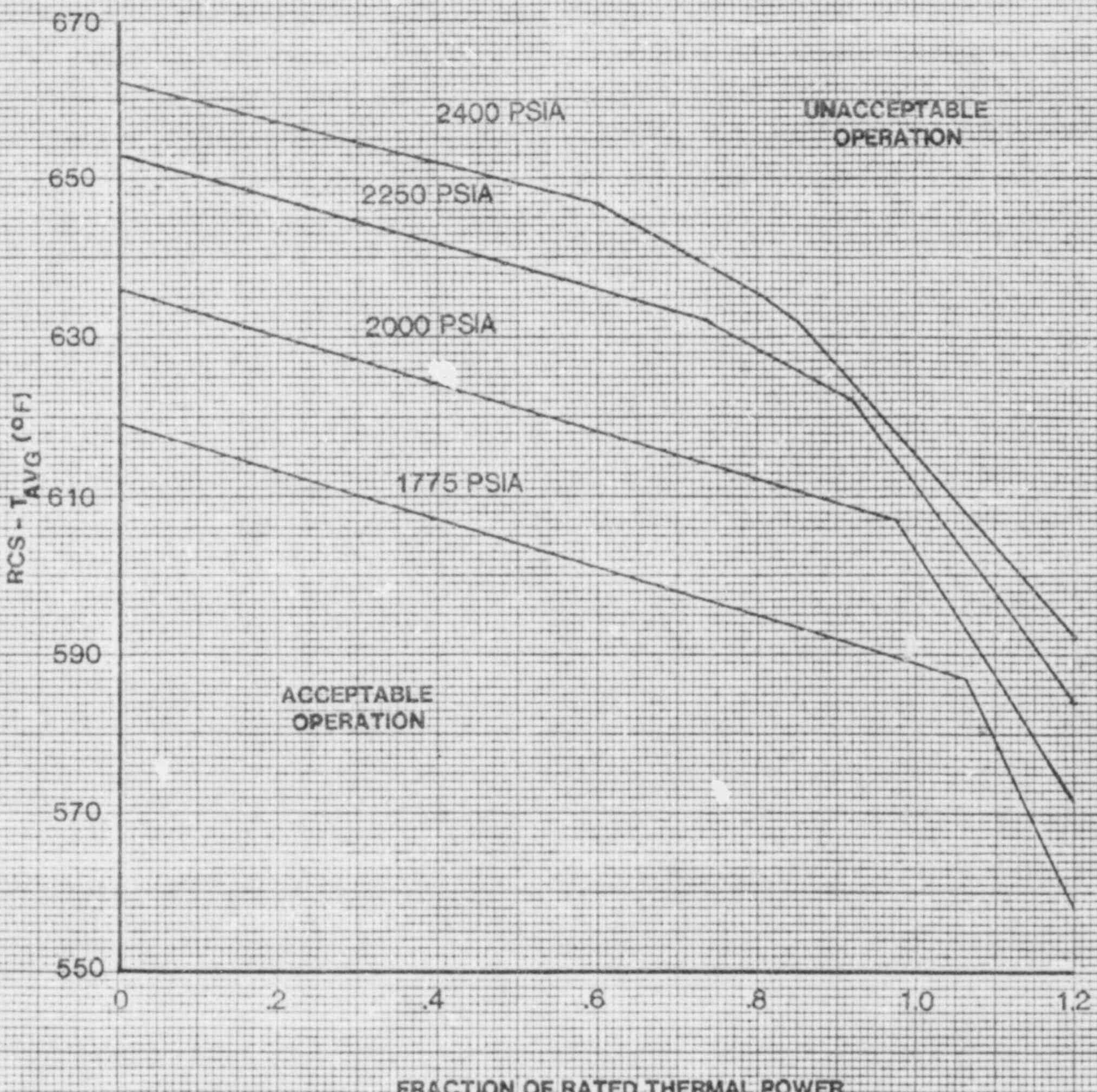


Figure 2.1-1. Reactor Core Safety Limit – Four Loops in Operation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	$\leq$ 10% Turbine Impulse Pressure Equivalent	$\leq$ 11% Turbine Impulse Pressure Equivalent
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Trip	$\leq$ 35% of RATED THERMAL POWER	$\leq$ 36% of RATED THERMAL POWER
23. Power Range Neutron Flux - (P-10) - Enable Block of Source, Intermediate, and Power Range (low setpoint) Reactor Trips	$\geq$ 10% of RATED THERMAL POWER	$\geq$ 9% of RATED THERMAL POWER
24. Reactor Trip P-4	Not Applicable	Not Applicable
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	$\leq$ 50% of RATED THERMAL POWER	$\leq$ 51% of RATED THERMAL POWER

NOTATION

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

where: $\frac{1}{1 + \tau_1}$ = Lag compensator on measured ΔT

τ_1 = Time constants utilized in the lag compensator for $\Delta T_3 \tau_1$ = 2 secs.

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.15

K_2 = 0.011

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

$\frac{1 + \tau_2 S}{1 + \tau_3 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_2, & τ_3 = Time constants utilized in the lead-lag controller for T_{avg}, τ_2 = 33 secs., τ_3 = 4 secs.

T = Average temperature °F

$\frac{1}{1 + \tau_4 S}$ = Lag compensator on measured T_{avg}

τ_4 = Time constant utilized in the measured T_{avg} lag compensator, τ_4= 2 secs.

T' $\leq$ 578.2°F (Nominal T_{avg} at RATED THERMAL POWER)

K_3 = 0.00055

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

S = Laplace transform operator (sec^{-1})

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

(i) for $q_t - q_b$ between - 29 percent and + 5 percent $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -29 percent, the ΔT trip setpoint shall be automatically reduced by 1.50 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by 0.86 percent of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_5 S}{1 + \tau_5 S}\right) \left(\frac{1}{1 + \tau_4 S}\right) T - K_6[T\left(\frac{1}{1 + \tau_4 S}\right) - T''] - f_2(\Delta I)\}$

Where: $\frac{1}{1 + \tau_1 S}$ = as defined in Note 1

τ_1 = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ 1.087

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_5 S}{1 + \tau_5 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: (Continued)

τ_5	=	Time constant utilized in the rate-lag controller for T_{avg}, τ_5 = 10 secs.
$\frac{1}{1 + \tau_4 S}$	=	as defined in Note 1
τ_4	=	as defined in Note 1
K_6	=	0.0011 for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 578.2°F)
S	=	as defined in Note 1
$f_2(\Delta I)$	=	0 for all ΔI

NOTE 3: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2 percent.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 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, 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 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOWRATE AND R

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R_1, R_2 shall be maintained within the regions of allowable operation shown on Figure 3.2-3 for 4 loop operation:

Where:

a. $R_1 = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$,

b. $R_2 = \dfrac{R_1}{[1 - RBP\ (Bu)]}$,

c. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

d. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$, and

e. RBP (Bu) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

APPLICABILITY: MODE 1

ACTION:

With the combination of RCS total flow rate and R_1, R_2 outside the regions of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R_1, R_2 to within the above limits, or

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

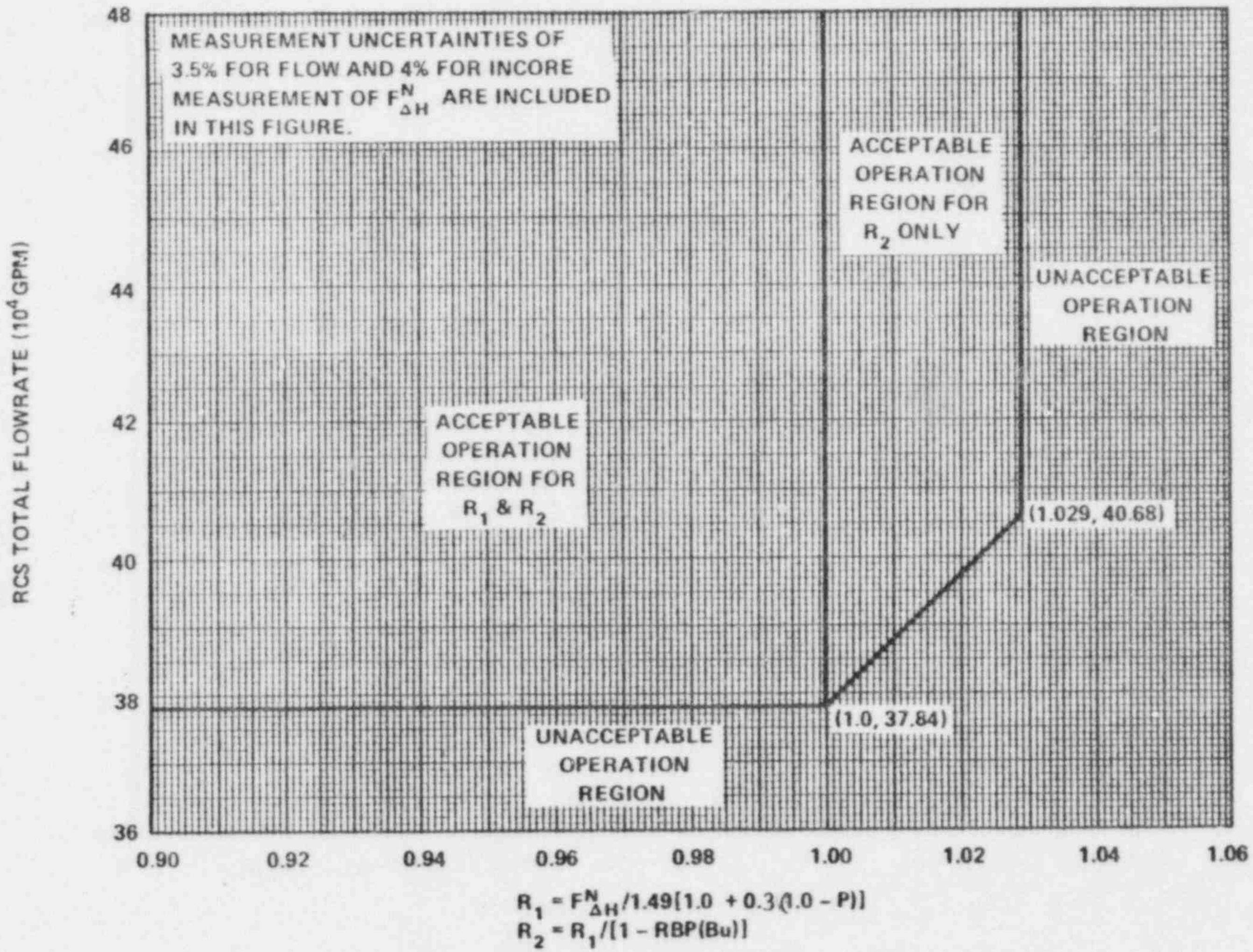


FIGURE 3.2-3 RCS Total Flowrate Versus R_1 and R_2 – Four Loops in Operation

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the allowed operational space defined by Figure 3.2-1.

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the Figure 3.2-1 limits,

   1.) Either restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

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

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50 percent of RATED THERMAL POWER by:

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

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

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

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

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside the limits.

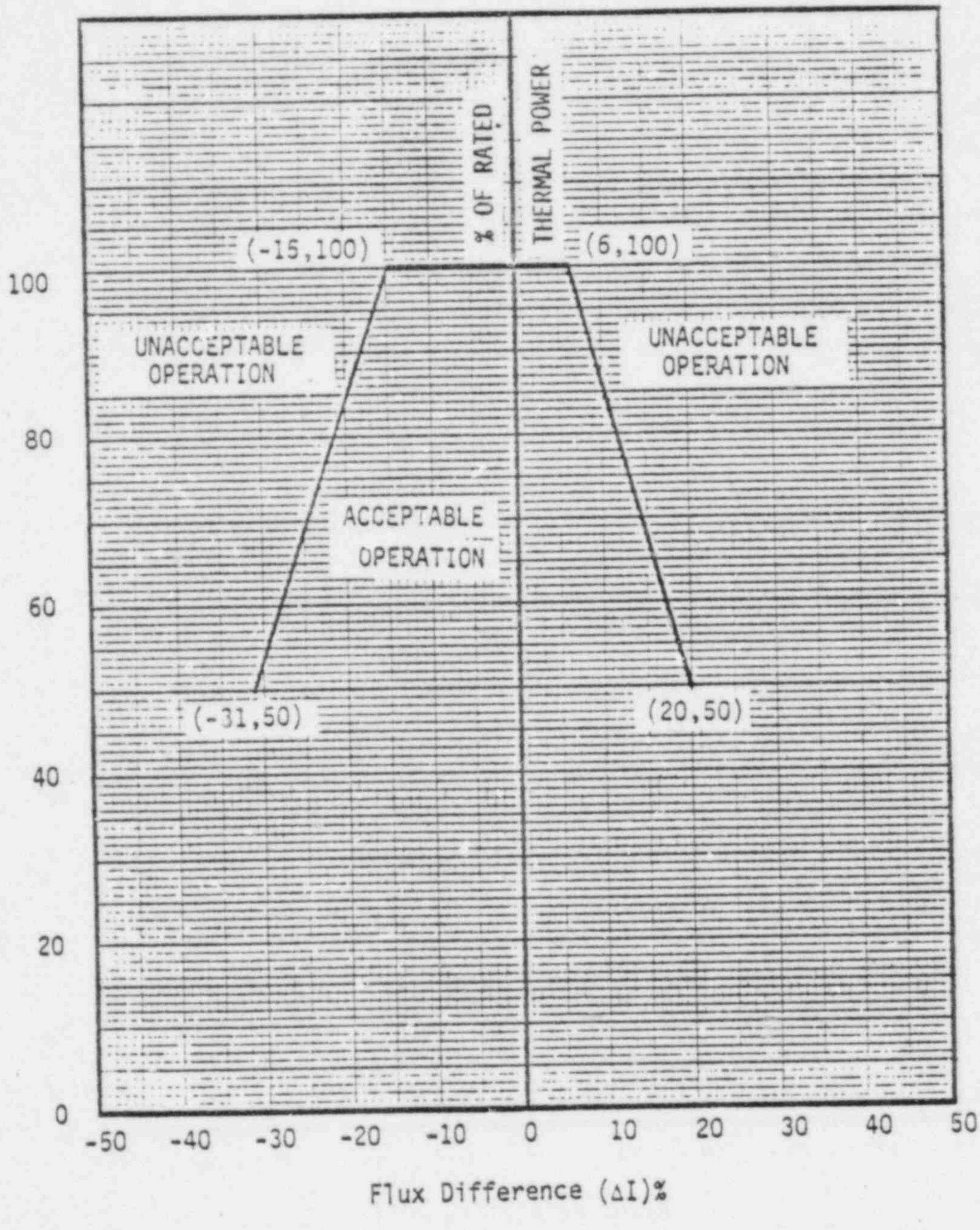


FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER
(TYPICAL EXAMPLE)

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-FQ(Z)

LIMITING CONDITION FOR OPERATION

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

$$F_Q(z) \leq \left[\frac{2.237}{P}\right] [K(\dot{Z})] \text{ for } P > 0.5$$

$$F_Q(z) \leq \left[\frac{2.237}{0.5}\right] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(z)$ exceeding its limit:

1. Reduce THERMAL POWER at least 1 percent for each 1 percent $F_Q(z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1 percent (in ΔT span) for each 1 percent $F_Q(z)$ exceeds the limit.

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5 percent to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.237 \quad x\ K(z)}{P\ x\ W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.237 \quad x\ K(z)}{W(z)\ x\ \ 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{limit} is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.14.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

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

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

SURVEILLANCE REQUIREMENTS (Cont)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

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

1. $F_Q{}^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.2.c, or

2. $F_Q{}^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

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

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

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

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q{}^M(z) \times W(z)}{\frac{2.237}{P} \times K(z)} \right] \right) -1 \right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q{}^M(z) \times W(z)}{\frac{2.237}{0.5} \times K(z)} \right] \right) -1 \right\} \times 100 \quad \text{for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above

SURVEILLANCE REQUIREMENTS (continued)

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

1. Lower core region 0 to 15 percent inclusive.

2. Upper core region 85 to 100 percent inclusive.

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained:

a. between 85°F* and 105°F in the containment upper compartment, and

b. between 100°F* and 125°F in the containment lower compartment.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature not conforming to the above limits, restore the air temperature to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment upper compartment average air temperature shall be the weighted average** of all ambient air temperature monitoring stations located in the upper compartment. As a minimum, temperature readings will be obtained at least once per 24 hours from the following locations:

Location
a. Elev. 743 ft.
b. Elev. 786 ft.
c. Elev. 786 or 845 ft.

4.6.1.5.2 The primary containment lower compartment average air temperature shall be the weighted average** of all ambient air temperature monitoring stations located in the lower compartment. As a minimum, temperature readings will be obtained at least once per 24 hours from the following locations:

Location
a. Elev. 722 ft.
b. Elev. 700 ft.
c. Elev. 685 or 703 ft.

* Lower limit may be reduced to 60°F in MODES 2, 3 and 4.

** The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperable temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Three pairs (three purge supply lines and three purge exhaust lines) of containment purge system lines may be open; the containment purge supply and exhaust isolation valves in all other containment purge lines shall be closed. Operation with purge supply or exhaust isolation valves open for either purging or venting shall be limited to less than or equal to 1000 hours per 365 days.

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

ACTION:

With a purge supply or exhaust isolation valve open in excess of the above cumulative limit, or with more than one pair of containment purge system lines open, close the isolation valve(s) in the purge line(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 The position of the containment purge supply and exhaust isolation valves shall be determined at least once per 31 days.

4.6.1.9.2 The cumulative time that the purge supply and exhaust isolation valves are open during the past 365 days shall be determined at least once per 7 days.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(z)$ Heat flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(z)$ upper bound envelope of 2.237 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

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

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

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

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 13 steps from the group demand position.

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

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

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

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flow and $F^N_{\Delta H}$ may be "traded off" against one another to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5 percent for RCS total flow rate and 4 percent for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

R_1, as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is the value used in the various safety analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g. peak clad temperature, and thus is the maximum "as measured" value allowed. R_2, as defined, allows for the inclusion of a penalty for for Rod Bow on DNBR only. Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allow for "trade off" in excess of R equal to 1.0 for the purpose of offsetting the Rod Bow DNBR penalty.

The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 and W 8691 Rev. 1 (partial rod bow test data).

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

The hot channel factor $F_Q{}^M(z)$ is measured periodically and increased by a cycle and height dependent power factor, W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.14.

THIS FIGURE DELETED

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

3/4.2.4 QUADRANT POWER TILT RATIO

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

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

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the intial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.3 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.
2. Cause(s) for the unplanned release.
3. Actions taken to prevent recurrence.
4. Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.14 The W(z) function for normal operation shall be provided to the Director, Nuclear Reactor Regulations, Attention: Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 60 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.

Any information needed to support W(z) will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

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

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

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. All REPORTABLE OCCURRENCES submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1 and 6.8.4.
f. Records of radioactive shipments.
g. Records of sealed source and fission detector leak tests and results.
h. Records of annual physical inventory of all sealed source material of record.

APPENDIX B PEAKING FACTOR LIMIT REPORT

Sequoyah Nuclear Plant Unit 1 - Peaking Factor Limit Report

This peaking factor limit report is provided in accordance with paragraph 6.9.1.14 of the Sequoyah unit 1 technical specifications.

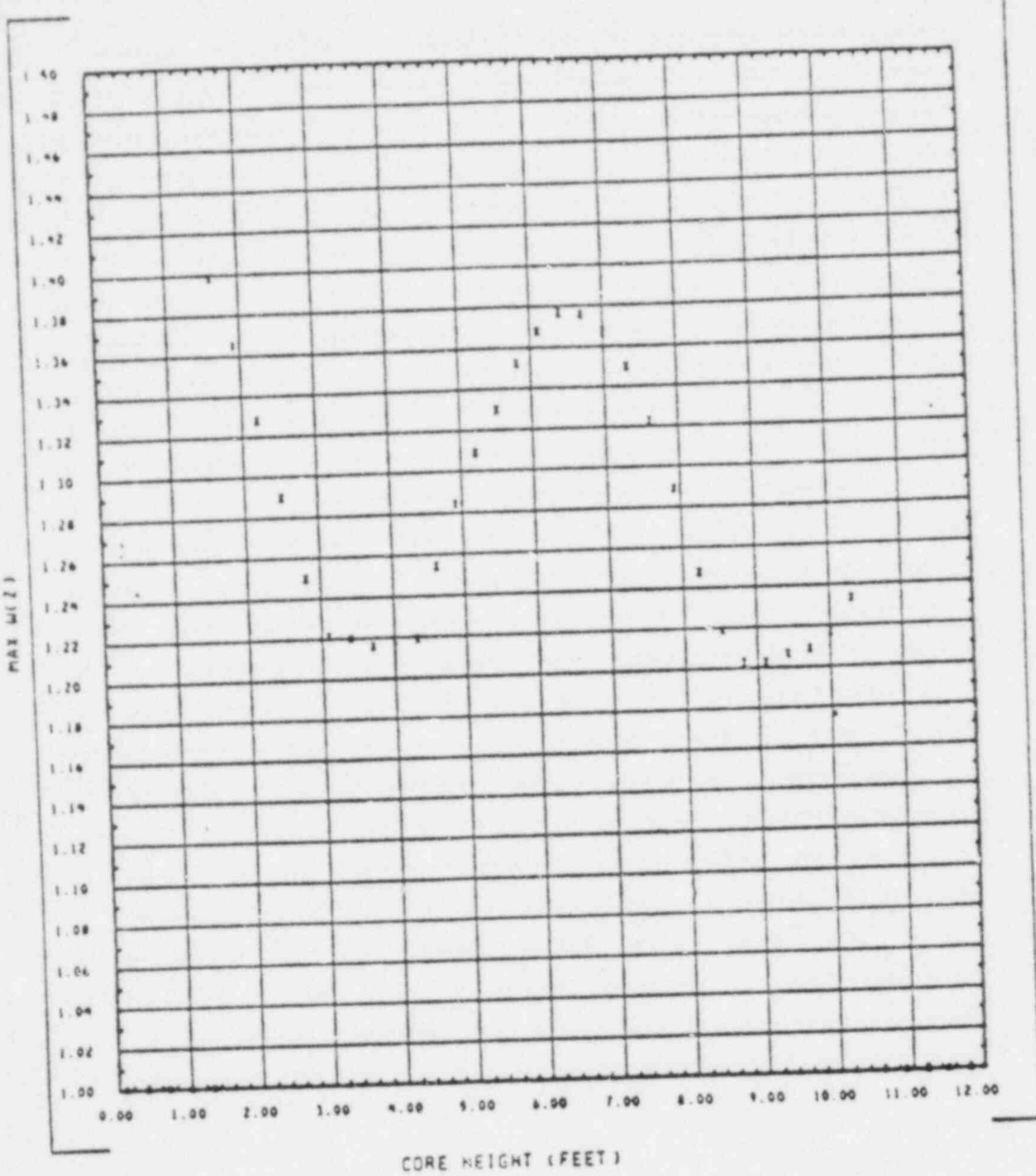
The cycle 2 W(z) function for RAOC operation is shown in figure 1. W(z) was calculated using the method described in NS-EPR-269, letter from E. P. Rahe (Westinghouse) to C. H. Burlinger (NRC) August 31, 1982.

This W(z) function is used to confirm that the heat flux hot channel factor, $F_Q(z)$, will be limited to the technical specification values of:

$$F_Q(z) \leq \frac{2.237}{P}[K(z)] \quad \text{for} \quad P > 0.5 \text{, and}$$

$$F_Q(z) \leq 4.474\,[K(z)] \quad \text{for} \quad P \leq 0.5$$

This W(z) function, when applied to a power distribution measured under equilibrium conditions, demonstrates that the initial conditions assumed in the LOCA are met along with the ECCS acceptance criteria of 10 CFR 50.46.



+(a,c)

HEIGHT (FEET)	MAX W(Z)	
.15	1.000	*
.45	1.000	*
.75	1.000	*
1.05	1.000	*
1.35	1.000	*
1.65	1.399	
1.95	1.365	
2.25	1.328	
2.55	1.289	
2.85	1.250	
3.15	1.221	
3.45	1.220	
3.75	1.216	
4.05	1.219	
4.35	1.219	
4.65	1.254	
4.95	1.284	
5.25	1.309	
5.55	1.329	
5.85	1.352	
6.15	1.368	
6.45	1.376	
6.75	1.376	
7.05	1.367	
7.35	1.350	
7.65	1.323	
7.95	1.289	
8.25	1.248	
8.55	1.219	
8.85	1.203	
9.15	1.203	
9.45	1.206	
9.75	1.209	
10.05	1.217	
10.35	1.233	
10.65	1.000	*
10.95	1.000	*
11.25	1.000	*
11.55	1.000	*
11.85	1.000	*

* Top and bottom 15% excluded as per Technical Specification 4.2.2.2.g

FIGURE 1

TYPICAL WESTINGHOUSE RELOAD CORE

RAOC W(Z)

APPENDIX C RELOAD TEST PROGRAM

The reload core design will be verified by performance of the following tests:

1. Control rod drop times,
2. Critical boron concentration measurements,
3. Control rod bank worth measurements using rod swap method,
4. Moderator temperature coefficient measurements, and
5. Flux distribution measurements using the incore flux mapping system.

APPENDIX D REMOVAL OF ROD CONTROL RESTRICTIONS

400 Chestnut Street Tower II

July 22, 1982

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

In 1979 Westinghouse Electric Corporation identified to the NRC, Core Performance Branch, by letters dated November 15 and November 28, 1979 (Reference letter numbers NS-TMA-2162 and NS-TMA-2167), a concern with regard to certain assumptions utilized in the dropped rod accident safety analysis applicable to some Westinghouse NSSS designs. This concern was derived primarily from the potential for an unanalyzed power overshoot while in automatic control following selected dropped rod events which did not result in a reactor trip. The concern was applicable to all Westinghouse plants which rely upon the power range neutron flux high-negative rate reactor trip to mitigate the consequences of the dropped rod accident. Operating plants were notified of an unreviewed safety question under 10 CFR 50.59 and nonoperating plants notified of a significant deficiency under 10 CFR 50.55(e). Westinghouse recommended, and NRC subsequently required, certain operational restrictions above 90-percent power (either manual rod control or restricted rod insertion limits when in automatic rod control) to address this concern on an interim basis and to provide further evaluation.

It is our understanding that a meeting was held between members of the Core Performance Branch staff and Westinghouse to discuss the Westinghouse dropped rod evaluation process. This process demonstrated that the DNB design basis can be met for this FSAR Chapter 15 condition II event. We have been notified by Westinghouse that this evaluation process results in conclusions that will allow removal of the interim operating requirements on rod control and insertion.

It is also our understanding that an agreement has been reached between Westinghouse and members of the Core Performance Branch staff that the removal of operating requirements would take place after the NRC review of the information subsequently submitted by Westinghouse letter dated January 20, 1982 (Reference letter number NS-EPR-2545). This letter serves as notification that the dropped rod evaluation process documented by Westinghouse letter NS-EPR-2545, dated January 20, 1982 applies to Sequoyah unit 1, cycle 2 and Sequoyah unit 2, cycle 2.

The results confirm that the DNB design basis is met for the dropped rod accident. Based upon this method, it can be concluded that the interim restrictions on rod control and insertion will no longer be necessary.

We formally request the NRC to review the material submitted by Westinghouse (NS-EPR-2545, January 20, 1982), and subsequently remove the interim operational restrictions effective with the startup of cycle 2 for both units. Approval is needed before startup of Sequoyah unit 1, cycle 2, presently scheduled for refueling outage in September 1982.

If you have any questions concerning this matter, please get in touch with J. E. Wills at FTS 858-2683.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L.M. Mills

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 22nd day of July 1982
Paulette G. White
Notary Public
My Commission Expires 9-5-84

cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303