September 29, 1983

Docket No. 50-328

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Subject: Issuance of Amendment No. 21 to Facility Operating License No. DPR-79 - Sequoyah Nuclear Plant, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. DPR-79.

The amendment changes the Technical Specifications to accommodate the Unit 2 cycle 2 reload operations and to change the requirement for testing of containment protective fuses from a destructive type of testing to visual inspection. The amendment is in response to your letters dated July 1 and July 27, 1983.

A copy of the related safety evaluation supporting Amendment No. 21 to Facility Operating License DPR-79 is enclosed.

Sincerely,

S/

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 21 to DPR-79
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:DL:LB #4	DL:LB #4	DL:LB #4				
SURNAME	MDuncan/hmc	CStahle	EAdensam				
DATE	9/26/83	9/ /83	9/ /83				

SEQUOYAH

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Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

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U.S. Environmental Protection
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ATTN: EIS Coordinator
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Hamilton County Courthouse
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Nashville, Tennessee 37203

OFFICE							
SURNAME							
DATE							

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. DPR-79

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

   A. The applications for amendment to the Sequoyah Nuclear Plant, Unit 2 (the facility) Facility Operating License No. DPR-79 filed by the Tennessee Valley Authority (licensee), dated July 1 and July 27, 1983, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, the provisions of the Act, and the rules and regulations of the Commission;

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 21, are hereby incorporated into the license.

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DATE							

The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Appendix A Technical
Specification Changes

Date of Issuance: September 29, 1983

Check for Comments
on Petition immediately
before issuing;
If any come back to [illegible]

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4	AD:L:DL		
SURNAME	MDuncan/hmc	OStahle	[illegible]	EAdensam	TNovak		
DATE	9/26/83	9/[illegible]/83	9/27/83	9/28/83	9/29/83		

ATTACHMENT TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Amended Page
2-2
2-7
2-8
2-9
2-10
B2-1
3/4 2-1
3/4 2-2
3/4 2-3
3/4 2-4
3/4 2-5
3/4 2-6
3/4 2-6a
3/4 2-8
3/4 2-11
3/4 3-44
3/4 6-10
3/4 6-26
B3/4 2-1
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B3/4 2-4
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DATE ▶							

September 29, 1983

AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE DPR-79 - SEQUOYAH UNIT 2

DISTRIBUTION w/enclosures:

Docket No. 50-327/328
LB #4 r/f
C. Stahle
M. Duncan
OELD
E. Adensam
R. Hartfield, MPA
R. Diggs, ADM
D. Eisenhut/R. Purple
J. Souder
T. Barnhart (4)
E. L. Jordan, DEQA: I&E
J. M. Taylor, DRP: I&E
L. J. Harmon, IE File
D. Brinkman, SSPB
H. Denton

bcc w/enclosures:

NRC PDR
Local PDR
NSIC
PRC System
A. Rosenthal, ASLAB
ASLBP
ACRS (16)
W. Jones (10)

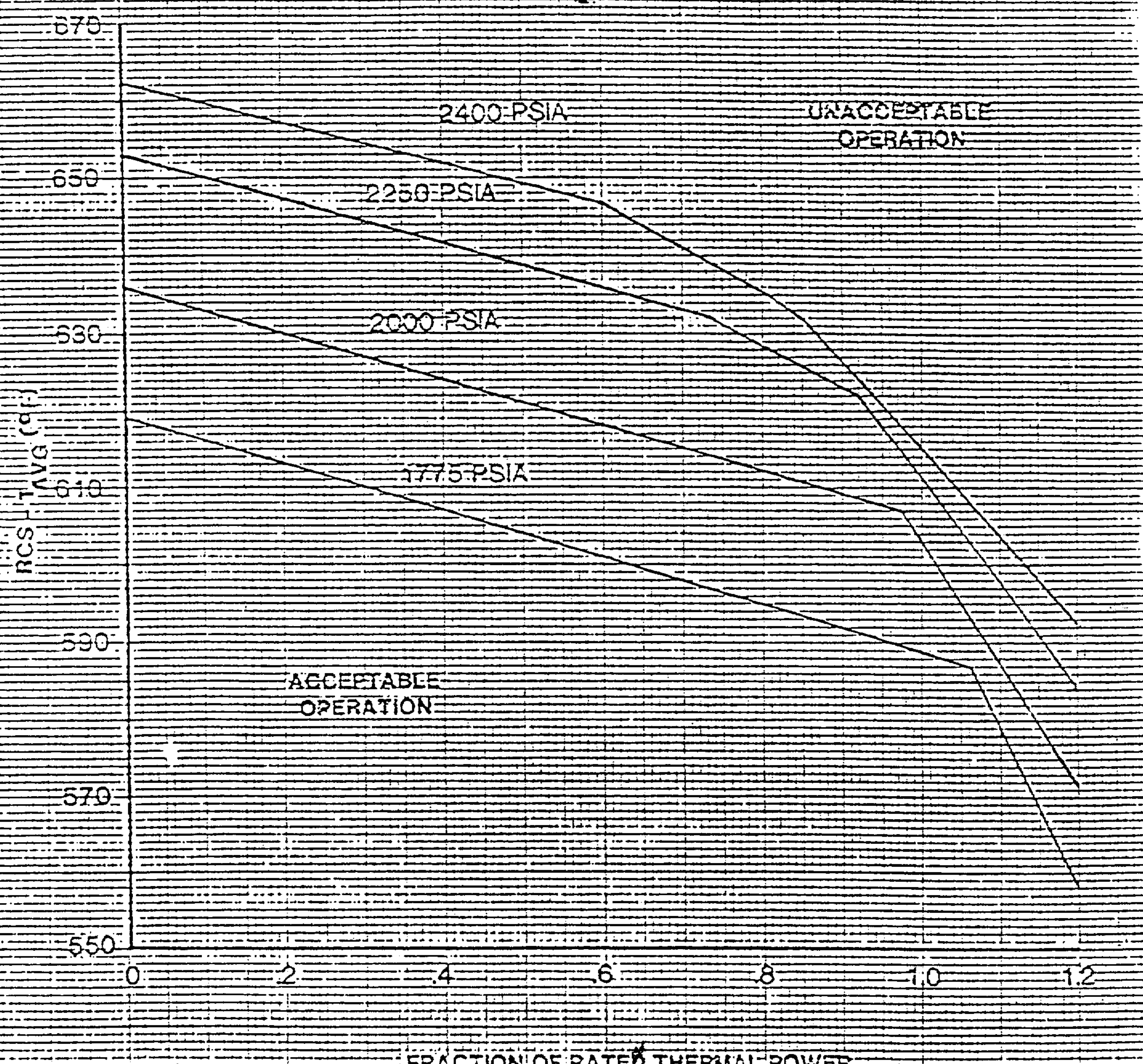


Figure 2.1-1. Reactor Core Safety Limit - Four Loops in Operation

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

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.

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.

These 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

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

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% of 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% 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.

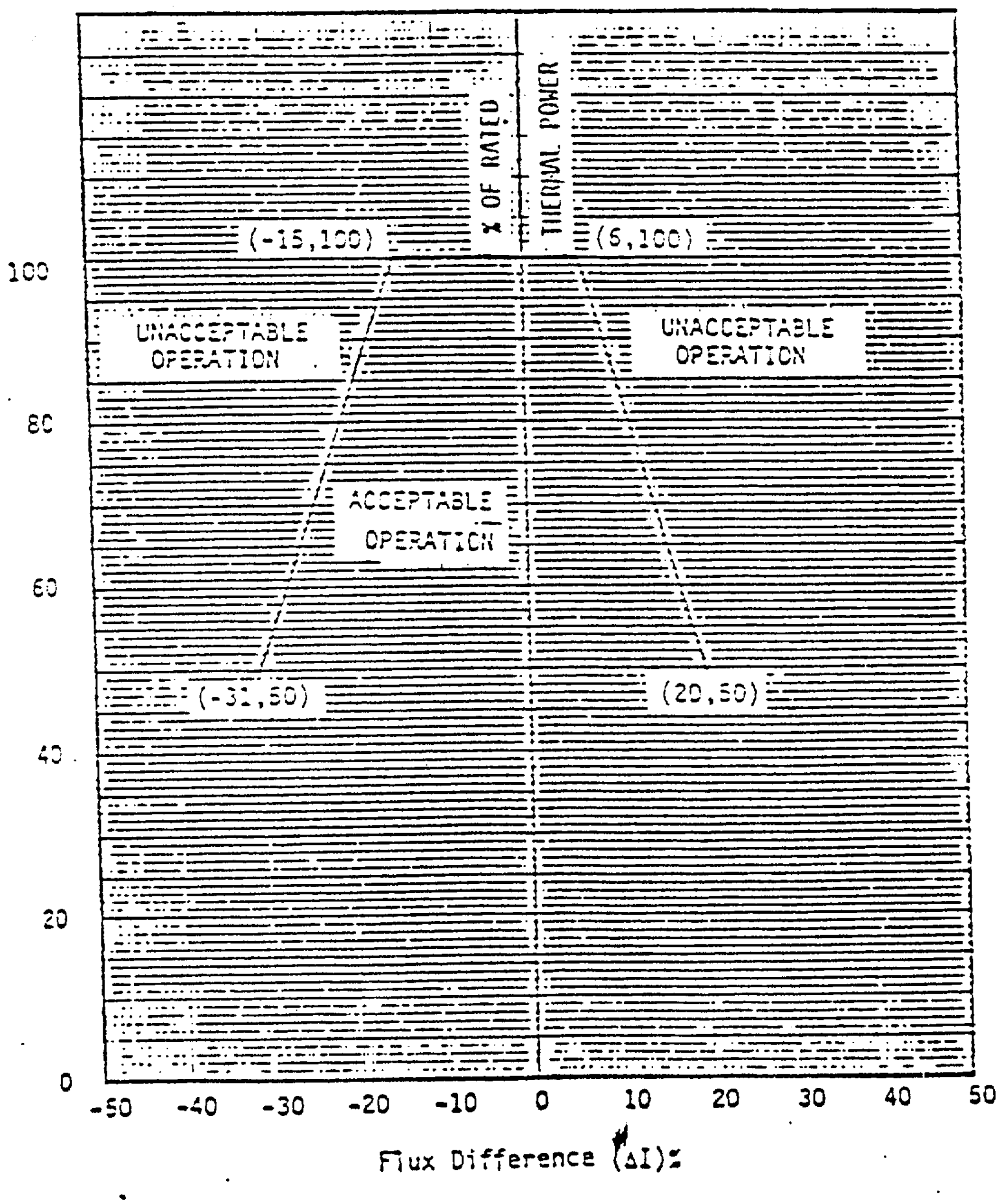


FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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

LIMITING CONDITION FOR OPERATION

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

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

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

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

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:

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

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

b. 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% to account for measurement uncertainties.

c. Satisfying the following relationship:

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

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

where $F_Q^M(z)$ is 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 (Continued)

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 present $F_Q(z)$ 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 for further increased by 5 percent to account for measurement uncertainty.

3/4.2.3 RCS FLOW RATE 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 region 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. $R_2 = \dfrac{R_1}{[1\text{-RBP(BU)}]}$,

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).

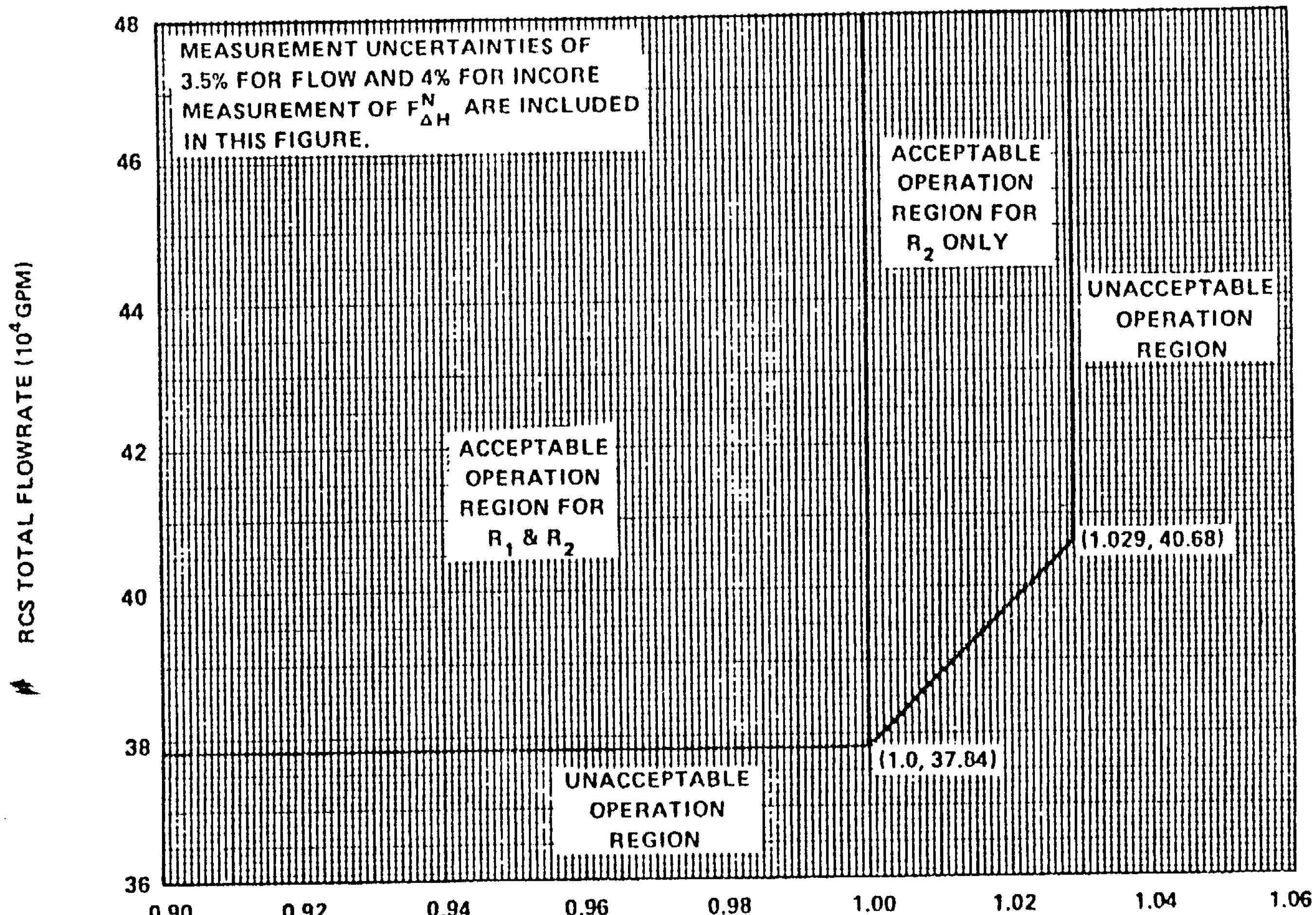
APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours, either:

1. 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.



$R_1 = F^N_{\Delta H}/1.49[1.0 + 0.3(1.0 - P)]$
$R_2 = R_1/[1 - RBP(Bu)]$

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

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

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

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, and $F_Q(Z)$

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$.

HYDROGEN MITIGATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 The primary containment hydrogen mitigation system shall be operable.

APPLICABILITY: MODES 1 and 2.

ACTION

With one train of hydrogen mitigation system inoperable, restore the inoperable train to OPERABLE status within 7 days or increase the surveillance interval of S.R. 4.6.4.3 from 92 days to 7 days on the operable train until the inoperable train is returned to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen mitigation system shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and verifying that at least 66 of 68 igniters are energized.*

b. At least once per 18 months by verifying the temperature of each igniter is a minimum of 1700°F

*Inoperable igniters must not be on corresponding redundant circuits which provide coverage for the same region.

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 (AFD) 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 compuer 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 ± 13 steps from the group demand position.

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

c. The control rod insertion limits of 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 of $F^N_{\Delta H}$ 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 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.

THIS FIGURE DELETED

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

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 tolerance 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.

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 on 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 initial 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.

COMBUSTIBLE GAS CONTROL (Continued)

is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The operability of at least 66 of 68 igniters in the hydrogen control distributed ignition system will maintain an effective coverage throughout the containment. This system of ignitors will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1200 pounds of ice per basket contains a 10% conservative allowance for ice loss through sublimation which is a factor of 10 higher than assumed for the ice condenser design. The minimum weight figure of 2,333,100 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the event that observed sublimation rates are equal to or lower than design predictions after three years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighed each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

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 Regulation, 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 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.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 1 of the curcuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. For the lower voltage circuit breakers the nominal trip setpoint and short-circuit response time are listed in Table 3.8-1. Testing of these circuit breakers will consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.*

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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*Surveillance requirement 4.8.3.1.a.3 may be suspended until the completion of Cycle 2 provided that the following surveillance requirement is implemented:

A fuse inspection and maintenance program will be maintained to ensure that:

1. the proper size and type of fuse is installed,
2. the fuse shows no sign of deterioration, and
3. the fuse connections are tight and clean.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.21 TO FACILITY OPERATING LICENSE DPR-79

TENNESSEE VALLEY AUTHORITY

INTRODUCTION

TVA requested in their letters of July 1 and July 27, 1983, several changes to the Technical Specifications for Sequoyah Units 1 & 2. Additional information on the requested changes was provided in letters of August 3, September 7, and September 19, 1983. Also the reference section in the fuel reload descussion provides a comprehensive listing of material that was utilized in the analysis.

One change for Unit 2 is to accommodate cycle 2 fuel reload operations. For this reload, sixty-eight new fuel assemblies will replace spent fuel from the first cycle. The new assemblies are the same as the assemblies in place, except for minor grid modifications to minimize interactions of grid spacing during fuel handling. Also some new burnable absorber rods will be utilized in cycle 2 that have been previously accepted for use in other nuclear plants. Also, the TVA request of July 1, 1983, requested a number of technical specification changes to improve plant operations which are applicable to both Units. These are removing operating restrictions on control rod operations, and adding requirements on the hydrogen control system.

Technical Specification changes regarding the testing of containment protective fuses from a destructive type of testing to visual inspection was requested. Every 18 months, 10% of the protective fuses are to be tested to ensure their integrity. At Sequoyah there are three types of protective fuses: 6900 and 480 volt fuses crimped inline and 480 volt fuses located in clip type holders. Removal of the fuses for testing may compromise cable and holder integrity.

FUEL RELOAD

By letter dated July 1, 1983, the Tennessee Valley Authority (TVA), licensee for Sequoyah Nuclear Plant Unit 2, submitted a request (Ref. 1) for a change in the plant Technical Specifications to accommodate the Unit 2, Cycle 2 reload. The submittal included a reload safety evaluation (RSE) that contained a description of the changes, a justification for the changes, and the proposed Technical Specifications. As stated in the RSE, the reload analysis was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology" (Ref. 2).

FUEL SYSTEM DESIGN

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when required, (c) the number of fuel rod failures is not under-estimated for postulated accidents, and (d) coolability is always maintained. Our evaluation of the information provided in support of Cycle 2 operation of Sequoyah Unit 2 is described below with regard to these review objectives.

Description

The Sequoyah 2 Cycle 2 core will be comprised of 193 fuel assemblies manufactured by Westinghouse Electric Corporation (W). Sequoyah 2 is one of the first W plants to utilize a W 17X17 fuel assembly design. This design is intended to extend fuel capability beyond that of the earlier 15X15 design in other W reactors of this approximate size. The primary intent of the design is to reduce stored energy in fuel rods for LOCA conditions. For Cycle 2 operation, 68 Region 1 fuel assemblies will be replaced with 68 Region 4 (Reload 1) assemblies.

The mechanical design of the Region 4 assemblies is the same as the Region 1 assemblies with the exception of a reconstitutable bottom nozzle design and grid modifications intended to minimize potential grid-to-grid interaction during fuel handling. Table I provides a comparison of pertinent design parameters.

In the RSE (Ref. 1), it is stated that a new Wet Annular Burnable Absorber (WABA) rod design will be utilized for Cycle 2 operation. The WABA design is described in WCAP-10021, Revision 1 (Ref. 3).

Design Evaluation

WABA - The WABA rod design consists of annular pellets of aluminum oxide and boron carbide ($Al_2O_3B_4C$) burnable absorber material encapsulated within two concentric Zircaloy tubings. The reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod. The topical report describing the WABA design (Ref. 3) was approved, and the utilization of WABA rods in Sequoyah 2 is thus approved subject to certain conditions. These conditions concern surveillance and the analysis of core bypass flow.

With regard to surveillance of the WABA rods, Westinghouse proposed a program that will consist of a visual (binocular) examination of approximately 10 percent of the WABA assemblies for evidence of anomalies or loss of structural integrity of the rodlets. In addition, the surveillance program would include review of routine in-core instrumentation measurements taken during the cycle. Such measurements would be used to help monitor the reactivity worth of the WABAs for comparison with predictions and with the results of the visual examination. Westinghouse offered to perform this surveillance for the first two plants to utilize WABAs. In response to a staff question (Ref. 4) concerning whether Sequoyah Unit 2 is subject to and will follow the provisions of the proposed W surveillance program, TVA stated (Ref. 5) that the program will indeed be carried out at Sequoyah Unit 2, although it was not originally intended to be one of the two WABA surveillance plants.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
SEQUOYAH UNIT 2 - CYCLE 2

REGION	1	2	3	4
Enrichment (W/O U-235)*	2.124	2.617	3.101	3.50
Geometric Density (Percent Theoretical)*	94.6	94.7	94.6	94.5
Number of Assemblies	5	72	48	68
Approximate Burnup at Beginning of Cycle 2 (MWD/MTU)	14,400	16,400	10,500	0
Approximate Burnup Predicted for EOC 2	24,100	27,400	23,100	11,500

*All fuel regions except region four are as-built values: Region four values are nominal. An average density of 94.5 percent theoretical was used for Region 4 evaluations.

Miscellaneous Design Considerations - In addition to surveillance of the WABA rods, additional information was requested (Ref. 4) concerning a few miscellaneous items that were not fully addressed in the RSE (Ref. 1). The questions and responses (Ref. 5) are addressed below.

Because the RSE presented only beginning-of-cycle (BOC) 2 burnup values for each fuel region, but not predicted (approximate) end-of-cycle (EOC) 2 burnups, our first question requested the EOC target burnups. The previously missing burnup values are shown in Table 1. They are consistent with second cycle burnups in other W plants and require no further comment.

We also requested further information (Ref. 4) about the reconstitutable bottom nozzles and spacer grid modifications, which are new features in the Region 4 fuel. In response (Ref. 5) TVA indicated that the reconstitutable bottom nozzle feature is the same as that introduced on other W plants such as Trojan, Farley Units 1 and 2, Salem Unit 1, North Anna Units 1 and 2, and (very recently) Beaver Valley Unit 1, and that this feature and the spacer grid modifications (which have also been implemented in fuel assemblies in the fore-mentioned plants) were evaluated by the licensee and determined not to involve an unreviewed safety question or Technical Specification change. Therefore, these design changes were performed under the provisions of 10 CFR 50.59. The Region 4 modified design features thus appear to be within the state-of-the-art and to be relatively minor. They are acceptable, therefore.

Another staff question concerned the fuel performance model (Ref. 6) used for the Cycle 2 analysis. That model (PAD 3.3) had been approved subject to certain restrictions. In response, TVA indicated that the approved W fuel performance model, with restrictions as modified in the NRC's safety evaluation for Addendum 1 to reference 6, was used in the design of Region 4 fuel. Inasmuch as the fuel performance analyses were performed with an approved model, it is acceptable without further review.

The final staff question (Ref. 4) concerned W's fuel rod revised internal pressure design basis. Although the RSE indicated that the internal pressure design basis, as described in WCAP 8964 (Ref. 7), was satisfied for Cycle 2 operation, we pointed out to TVA that an amended criterion had to be used to assure acceptable consequences for transients and accidents. In response (Ref. 5) TVA stated that the amended rod internal pressure criterion, was specified for Sequoyah Unit 2 Cycle 2 operation. We accept that statement without further review.

Non-Fuel Bearing Component (NFBC) Holddown Springs - During a recent refueling of McGuire Unit 1, several broken NFBC holddown springs were discovered (Ref. 8). McGuire 1, like Sequoyah 2, is an upper-head-injection (UHI) W plant. Such plants have holddown assemblies for non-fuel-bearing components such as thimble plugs and secondary sources. The primary function of the holddown assemblies is to provide an axial force on the NFBCs sufficient to oppose flow-induced lift forces during reactor operation. For the UHI plants, the holddown assemblies function also as part of the injection system, and so the springs in these assemblies are of special design with central turns of larger diameter than the end turns to allow radial flow area for emergency coolant. Because broken springs could have an impact on UHI flow, with resultant effects on the LOCA peak cladding temperatures, and because broken springs could also result in fuel damage from loose parts, TVA was asked (Ref. 4) to provide reasonable assurance for Cycle 2 operation that broken springs would not occur or that the potential effects of loose parts, UHI flow restrictions, and increases in LOCA peak cladding temperature would not be significant.

In response (Ref. 5) TVA stated that, although 17 broken springs were identified via examinations performed during refueling, there were no double-ended breaks. This observation was important because double-ended breaks would have greater potential for UHI blockage or loose parts. With regard to Cycle 2 operation, TVA stated (Ref. 5) that all 94 NFBC holddown springs will be replaced with springs of a new design in which mean stress levels are reduced and material grain size is reduced for better fatigue properties. Inasmuch as all the original NFBC springs will be replaced with the new design, which has been reviewed and approved for McGuire, we conclude that there is reasonable assurance that NFBC holddown springs will not be a problem during Cycle 2 operation of Sequoyah 2.

NUCLEAR DESIGN

The Cycle 2 loading is designed to meet an $F_Q(z)$XP ECCS analysis limit of $\leq$2.237XK(z). The kinetics characteristics for Cycle 2 are identical with those used for the previously submitted accident analysis except for the most negative Doppler temperature coefficient. The effect of this difference was considered in the analysis. The PALADON code which has been reviewed and approved by the staff was used to perform the nuclear design analyses. Control requirements analyses show that adequate shutdown margin exists for Cycle 2. The control rod insertion limits remain unchanged from Cycle 1.

The Cycle 2 analysis was performed with the following changes to the power distribution control procedure:

(1) The partial power multiplier for $F^N_{\Delta H}$ was changed from 0.2 to 0.3.

(2) The constant axial offset control procedure was replaced with the relaxed axial offset control (RAOC) procedure.

(3) The presently used surveillance procedure for $F_Q(z)$ was replaced with procedure.

These changes were requested in order to increase operational flexibility and permit optimization of the loading pattern.

The change in the partial power multiplier for $F^N_{\Delta H}$ allows increased $F^N_{\Delta H}$ at reduced power while maintaining the same $F^N_{\Delta H}$ limit at full power. The allowable $F^N_{\Delta H}$ is described by $F^N_{\Delta H} \leq 1.55\ [1+0.3(1-P)]$. The increase in allowable

$F^N_{\Delta H}$ at low power eliminates the need to change the rod insertion limits to satisfy the peaking factor criteria at low power with control rod bank at the insertion limit. The effect of this change was considered in the Cycle 2 analysis.

The RAOC procedure is described in a W topical report (Ref. 9). This report has been reviewed by the staff which concluded that the RAOC procedure was an acceptable method for power distribution control in W reactors.

The revised $F_Q(z)$ surveillance is described in a W topical report (Ref. 9) This report has been reviewed by the staff which concluded that the revised technique accomplished the same ends as the present one and is acceptable.

Control Rod Operations

The licensee requested removal of the interim operating restrictions imposed as a result of deficiencies in the analysis of the control rod drop event. Westinghouse submitted a topical report supporting the removal of these restrictions when certain analyses are performed. The staff has approved this report and the analyses have been performed for Sequoyah 2 Cycle 2. Thus the interim operating restrictions may be removed.

THERMAL HYDRAULIC

Cycle 2 reload fuel has no significant variation from Cycle 1 that would affect the thermal margins. However, the licensee has proposed that the coefficient of the power dependent term in the $F^N_{\Delta H}$ equation of the Technical Specification will be increased to 0.3 from its present value of 0.2. This has the effect of increasing the allowed radial peaking factor at low powers. This change was also accepted and incorporated in Unit 1 Cycle 2 reload.

Sequoyah Unit 2 Cycle 2 reload will contain 288 new WABA rods. This number is well within the maximum numbers of WABA rods allowed in reloads as specified in Table 7.2 of reference 3.

Based on the above and since the proposed Technical Specification change does not result in violation of SAFDL, and the number of WABA rods is well within the acceptable configuration, we conclude that the proposed Cycle 2 operation is acceptable.

ACCIDENTS AND TRANSIENTS

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR for four loop operation have been examined. In most cases, it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis. For the incidents which were reanalyzed it was determined that the applicable design basis limits are not exceeded and thus the conclusions presented in the FSAR are still valid.

The only kinetic parameter not within the limiting range of values used in the previous safety analysis was the Doppler temperature coefficient (DTC). The change was small and since the DTC represents only a small portion of the total negative reactivity feedback, the effect is negligible and no accidents were reanalyzed.

Cycle 2 has a trip reactivity insertion rate which is different from that used for Cycle 1. Investigation of the transients showed that only the locked rotor and loss of flow analyses may be affected. These transients were reanalyzed and there were no changes to the NRC safety conclusions.

Peaking factor evaluation for the rod out of position and hypothetical steamline break accidents resulted in a minimum DNBR greater than the design limit DNBR. Thus, these accidents were not investigated further.

The hot-zero power beginning of life rod ejection accident was reanalyzed because the Cycle 2 maximum F_Q exceeded the Cycle 1 values. The results show that all acceptance criteria specified in reference 10 were satisfied. Thus, the safety conclusions remain valid.

The change in the allowable $F^N_{\Delta H}$ as function of power resulted in a change to the k constants in the overtemperature Delta-T and overpower Delta-T setpoint equations and a change to the overtemperature Delta-T $f(\Delta I)$ function. Since the overtemperature Delta-T trip is used in the bank withdrawal at power accident, this accident was reanalyzed with the new overtemperature Delta-T setpoints. The results show that the minimum DNBR remains above the limit, and thus, are acceptable.

TECHNICAL SPECIFICATION

We have reviewed the proposed changes to Technical Specifications 2.1, 2.2, 3/4.2.1, 3/4.2.2, 3/4.2.3, and 6.9.1.14 and the bases and find them acceptable.

In accordance with proposed Technical Specification 6.9.1.14 TVA submitted a peaking factor report for Cycle 2 as an appendix to their RSE. The report was revised in a letter dated August 3, 1983. We find this report acceptable.

RELOAD CONCLUSIONS

We have reviewed the information submitted on the Cycle 2 operation of Sequoyah Unit 2. We find the proposed Cycle 2, Region 4 refueling to be acceptable from a fuel system mechanical design standpoint.

We have reviewed the nuclear design and find the proposed reload to be acceptable. The use of the RAOC procedure and the use of the proposed F_Q surveillance procedures are acceptable. The change of the partial power multiplier for $F^N_{\Delta H}$ from 0.2 to 0.3 is acceptable. The interim operating procedure for the rod drop protection may be discontinued for Cycle 2.

We have reviewed the proposed Technical Specification changes 2.1.1, 2.2.1, 3/4.2.1, 3/4.2.2, 3/4.2.3, and 6.9.1.14 and find them acceptable.

Hydrogen Control Requirements

In their letter of July 1, 1983, TVA requested changes to the Technical Specifications of Unit 2 to make them consistent with the approved Unit 1 specifications. The permanent hydrogen igniter system has been previously demonstrated to be acceptable for mitigating the consequences of hydrogen protection during degraded core accidents. The Unit 2 hydrogen mitigation system has been changed during the outage period to duplicate the Unit 1 system (Amendment No. 24).

Containment Protective Fuses

Fuses are utilized in nuclear power plants as overcurrent protective devices for high power electrical curcuits penetrating the reactor containment. Every 18 months, the technical specifications call for testing 10% of the fuses to ensure their integrity. Unit 1 technical specifications were (Amendment No. 20) modified to permit visual inspection rather than destructive testing of a certain number of fuses. The Unit 2 change is identical to that for Unit 1. Therefore, the revisions will make the requirements for both Units consistent in this area.

REFERENCES

1. Letter from L. M. Mills (TVA) to E. Adensam (NRC) with "Reload Safety Evaluation -- Sequoyah Unit 2, Cycle 2," July 1, 1983.

2. F. M. Bordelon, et al, "Westinghouse Reload Safety Evaluation Methodology," Westinghouse Report WCAP-9273, March 1978.

3. Letter from E. P. Rahe, Jr. (W) to C. O. Thomas (NRC), "Westinghouse WABA Evaluation Report," W Report WCAP-10021, Revision 1, (Proprietary), October 18, 1982.

4. Letter from E. G. Adensam (NRC) to Mr. H. G. Parris (TVA), "Sequoyah Unit 2 Cycle 2 Fuel Reload Questions," September 1, 1983.

5. (a) M. Tokar and C. Stahle (NRC) Telecummunication with J. E. Wills, et al, (TVA), September 2, 1983.

   (b) Letter from L. M. Mills (TVA) to E. Adensam (NRC) with response to NRC Questions on Sequoyah 2 Cycle 2 Reload, September 7, 1983.

6. "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," Westinghouse Report WCAP-8785 (Non-Proprietary) and WCAP-8720 (Proprietary), October 1976.

7. "Safety Analysis for Revised Fuel Rod Internal Pressure Design Basis," Westinghouse Report WCAP-8964 (Non-Proprietary), March 31, 1977.

8. Letter from H. B. Tucker (Duke Power Company) to J. P. O'Reilly (NRC) with Reportable Occurrence Report No. 369/83-11, March 24, 1983.

9. E. P. Rahe, Jr. (W) to C. Berlinger (NRC) "Relaxation of Constant Axial Offset Control" NS-EPR-2649 Parts A and B, August 31, 1982.

10. D. H. Risher, "An Evaluation of the Rod Ejection Accident in Westinghouse PWR's Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (48 FR 36130) on August 15, 1983, and consulted with the State of Tennessee. No public comments were received and the State of Tennessee did not have any comments.

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 29, 1983

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