ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-89-33)

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SEP 17 1980

WRB-1 correlation and W-3 Correlation for conditions outside the range of WRB-1

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.

Correlations have

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.

Insert 1

~~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 Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30 or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. R118

the safety analysis DNBR limit

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: R118

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$ R118

where P is the fraction of RATED THERMAL POWER

INSERT 1

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above ~~1.30~~ the safety analysis DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate

SAFETY LIMITS

BASES

analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 89% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 89% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below ~~1.30~~ during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature Delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature Delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below ~~1.30~~ during normal operational transients and anticipated transients with 3 loops in operation.

the safety analysis DNBR limit

ROD DROP TIME

2.7

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position# shall be less than or equal to $\boxed{2.2}$ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: R112

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

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

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months

#Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn, inclusive. R112

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F_{\Delta H}^{N}$

~~3/4.2.3 RCS FLOWRATE AND R~~

3.2.3 $F_{\Delta H}^{N}$ shall be limited by the following relationship:

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:~~

$$F_{\Delta H}^{N} \leq 1.55 [1.0 + 0.3 (1.0 - P)]$$

R23

Where:

~~a. $R_1 = \frac{F_{\Delta H}^{N}}{1.49 [1.0 + 0.3 (1.0 - P)]}$,~~

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

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

~~d. $F_{\Delta H}^{N}$ = Measured values of $F_{\Delta H}^{N}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F_{\Delta H}^{N}$ 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 $F_{\Delta H}^{N}$, 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.~~

Insert 2

INSERT 2

With F^N_H exceeding its limit:

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

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% of greater RATED THERMAL POWER.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2 and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the regions of acceptable operation shown in Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

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

Insert 3

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the regions of acceptable operation of Figure 3.2-3:

Insert 3

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The measured $F^N_{\Delta H}$ shall be increased by 4% for measurement uncertainty.

SURVEILLANCE REQUIREMENTS (Continued)

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

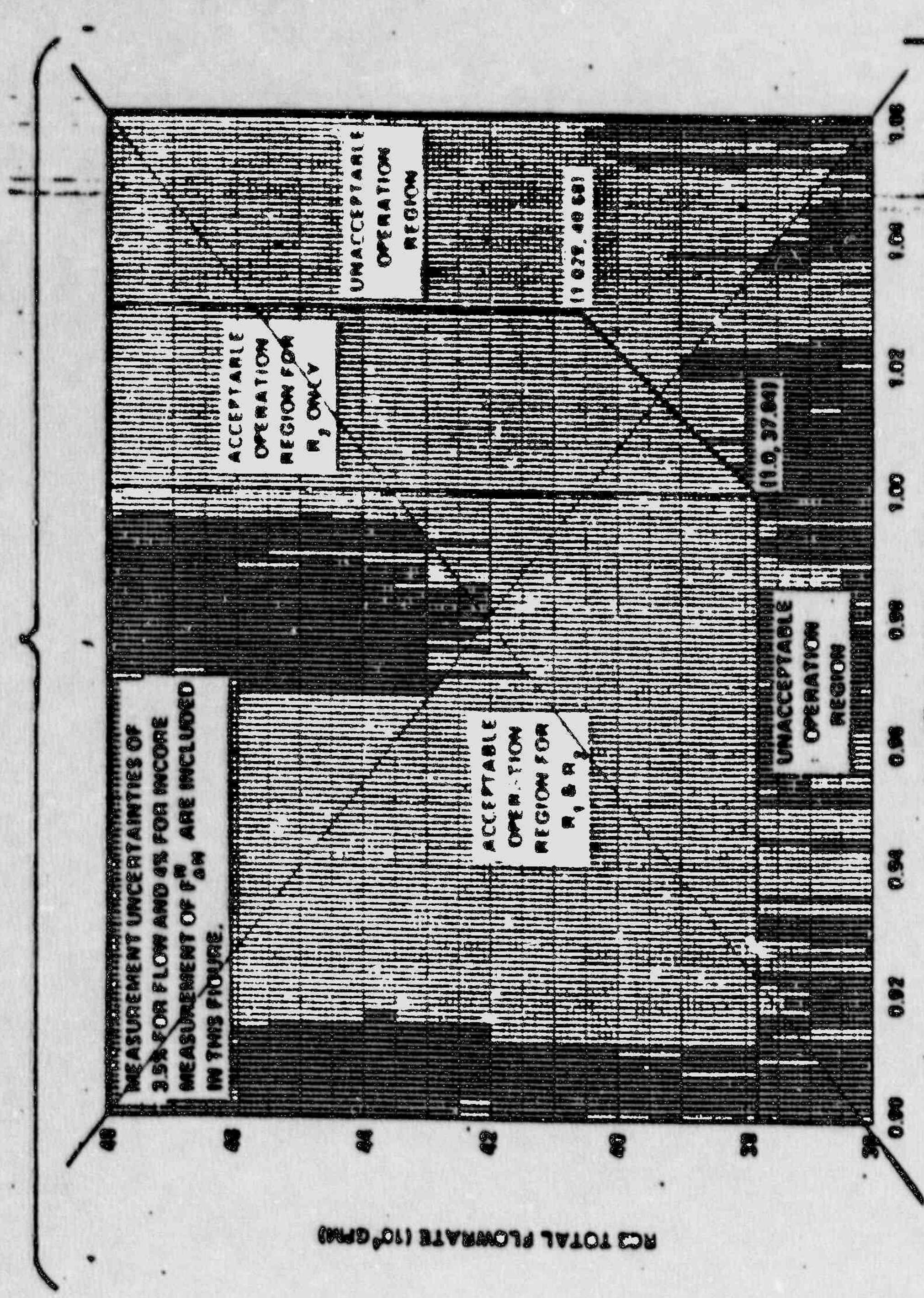
b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the values of R_1 and R_2, obtained per Specification 4.2.3.2, are assumed to exist.

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

4.2.3.5 The RCS flow rate shall be determined by measurement at least once per 18 months.

Delete



R23

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

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

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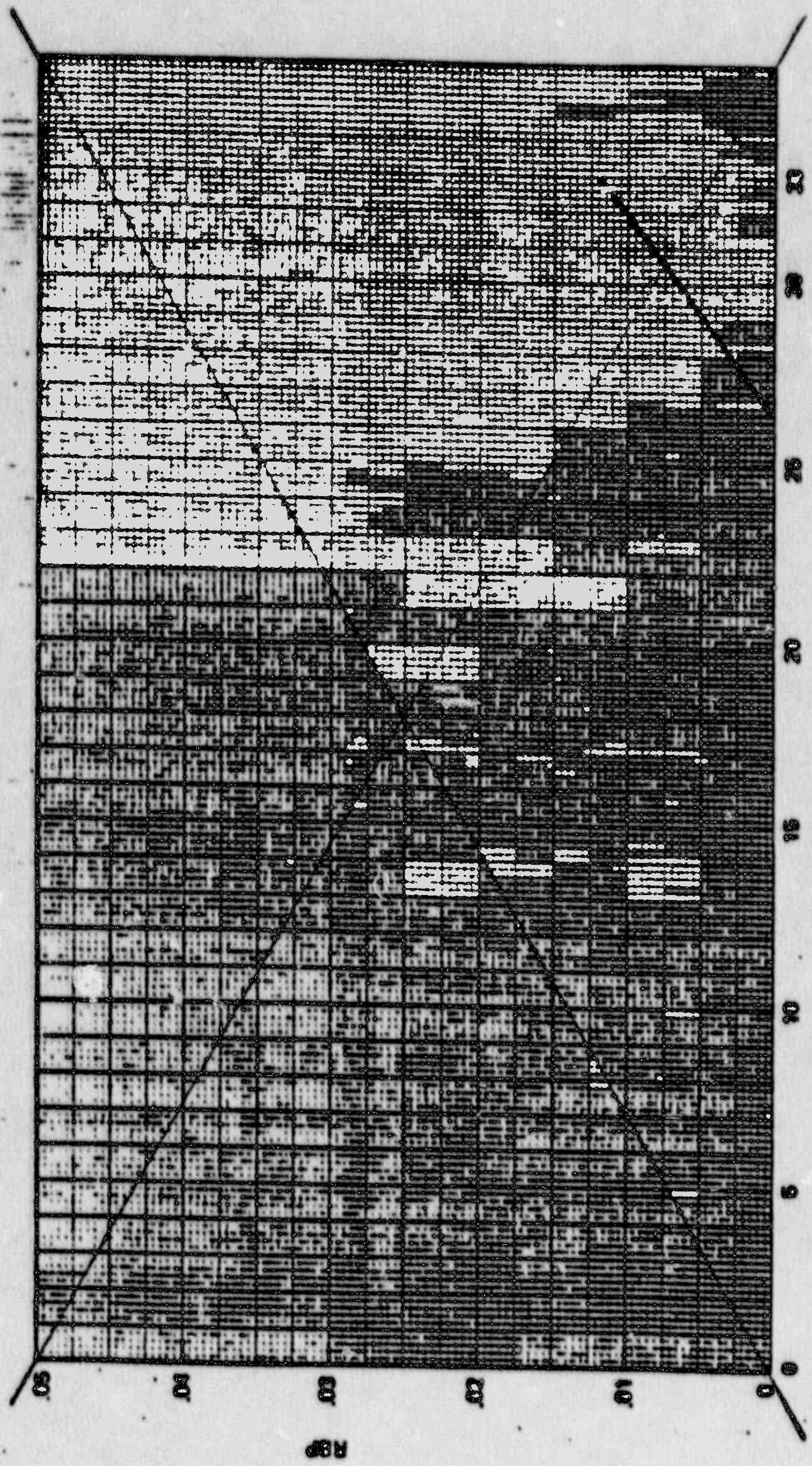

FIGURE 3.2-4 ROD BOW PENALTY VERSUS REGION AVERAGE BURNUP

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. REACTOR COOLANT SYSTEM TOTAL FLOW RATE

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

~~4.2.5~~ 4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 THE RCS FLOW RATE SHALL BE DETERMINED BY MEASUREMENT AT LEAST ONCE PER 18 MONTHS.

4.2.5.3 THE RCS TOTAL FLOW RATE INDICATORS SHALL BE SUBJECTED TO A CHANNEL CALIBRATION AT LEAST ONCE PER 18 MONTHS.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Total Flow	$\geq$ 378400 gpm #

R45

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

Includes a 3.5 % flow measurement uncertainty

R45

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL - FQ(Z) and $F^N_{\Delta H}$

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. R23

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.

hot channel factors

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.

R23

DELETE

INSERT 4

$F^N_{\Delta H}$ 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 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.

INSERT 4

LIMIT

The ~~relaxation in~~ $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

~~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.~~

Insert 5

The hot channel factor $F^{M}_{Q}(z)$ 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. R23

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.

INSERT 5

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty.

The applicable value of rod bow penalty is referenced in the FSAR.

Margin in excess of the rod bow penalty is available for plant design flexibility.

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 ≥ 1.3 throughout each analyzed transient.

greater than or equal to the safety analysis DNBR limit

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. R23

THE SAFETY ANALYSIS DNBR LIMIT

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above ~~1.30~~ during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times. R88

In MODE 4, a single reactor coolant loop or residual heat removal (RHR) loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE. R16

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. R16

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no

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2.1 SAFETY LIMITS

BASES

WRB-1 correlation and W-3 correlation for conditions outside the range of WRB-1

2.1.1 REACTOR CORE

Correlations have

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.

Insert 1

~~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 Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30 or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. R104

the safety analysis DNBR limit

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.5[illegible] and a reference cosine with a peak of 1.55 for axial power shape. An allow[illegible] is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression: R21

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

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 Delta T 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. BR

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

the safety analysis DNBR limit

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate

INSERT 1

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 89% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 89% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below ~~1.30~~ during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below ~~1.30~~ during normal operational transients and anticipated transients with 3 loops in operation.

the safety analysis DNBR limit

ROD DROP TIME

2.7

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position# shall be less than or equal to $\boxed{2.2}$ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: R98

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: Modes 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.* R20

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*For cycle 1, this surveillance is to be completed before the next cooldown or by August 5, 1983, whichever is earlier. R20

#Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥222 and ≤231 steps withdrawn, inclusive. R98

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

POWER DISTRIBUTION LIMITS

~~3/4.2.3 RCS FLOWRATE AND R~~

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

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:~~

R 21

Where:

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

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

$F^N_{\Delta H} \leq 1.55 [1.0 + 0.3(1.0 - P)]$

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

~~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.~~

Insert 2

INSERT 2

With F^N_H exceeding its limit:

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

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% of greater RATED THERMAL POWER.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

Insert 3

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

Insert 3

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The measured $F^N_{\Delta H}$ shall be increased by 4% for measurement uncertainty.

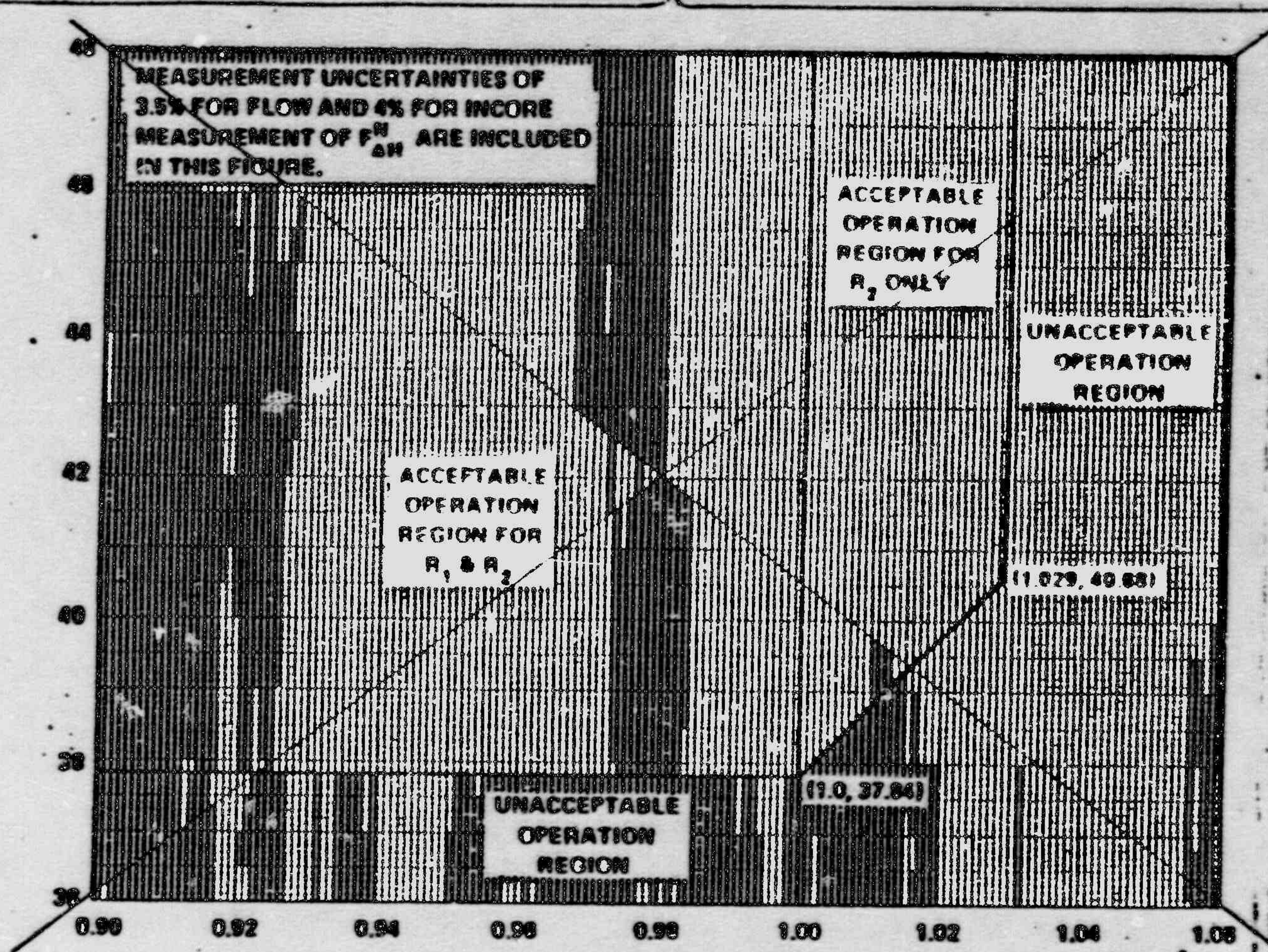
SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained values of R_1 and R_2, obtained per Specification 4.2.3.2, are assumed to exist.

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

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

Delete



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

R21

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

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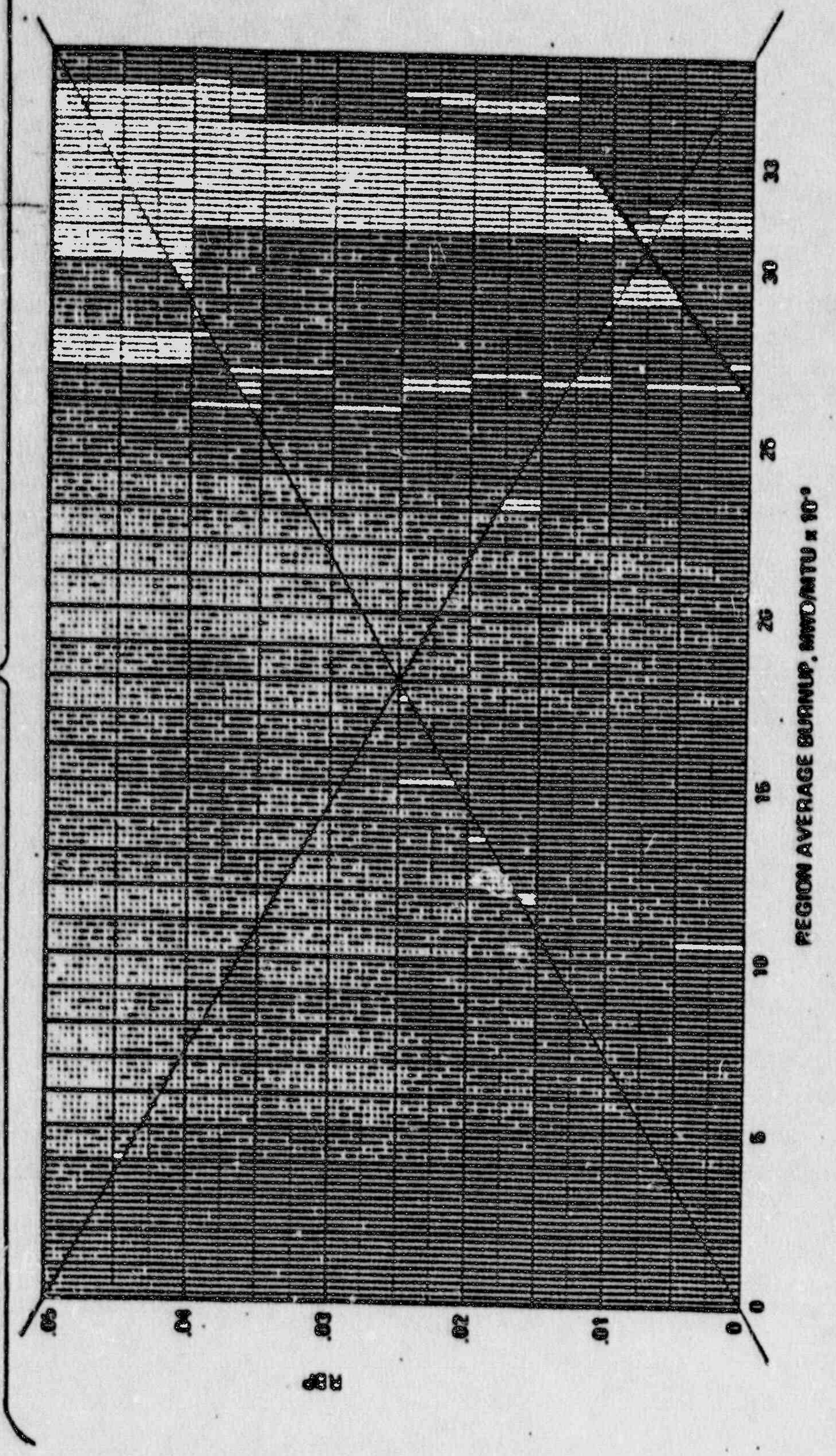


FIGURE 3.2-4 ROD BOW PENALTY VERSUS REGION AVERAGE BURNUP

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

C. REACTOR COOLANT SYSTEM TOTAL FLOW RATE

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 ~~4.2.5~~ Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 THE RCS FLOW RATE SHALL BE DETERMINED BY MEASUREMENT AT LEAST ONCE PER 18 MONTHS.

4.2.5.3 THE RCS TOTAL FLOW RATE INDICATORS SHALL BE SUBJECTED TO A CHANNEL CALIBRATION AT LEAST ONCE PER 18 MONTHS.

1

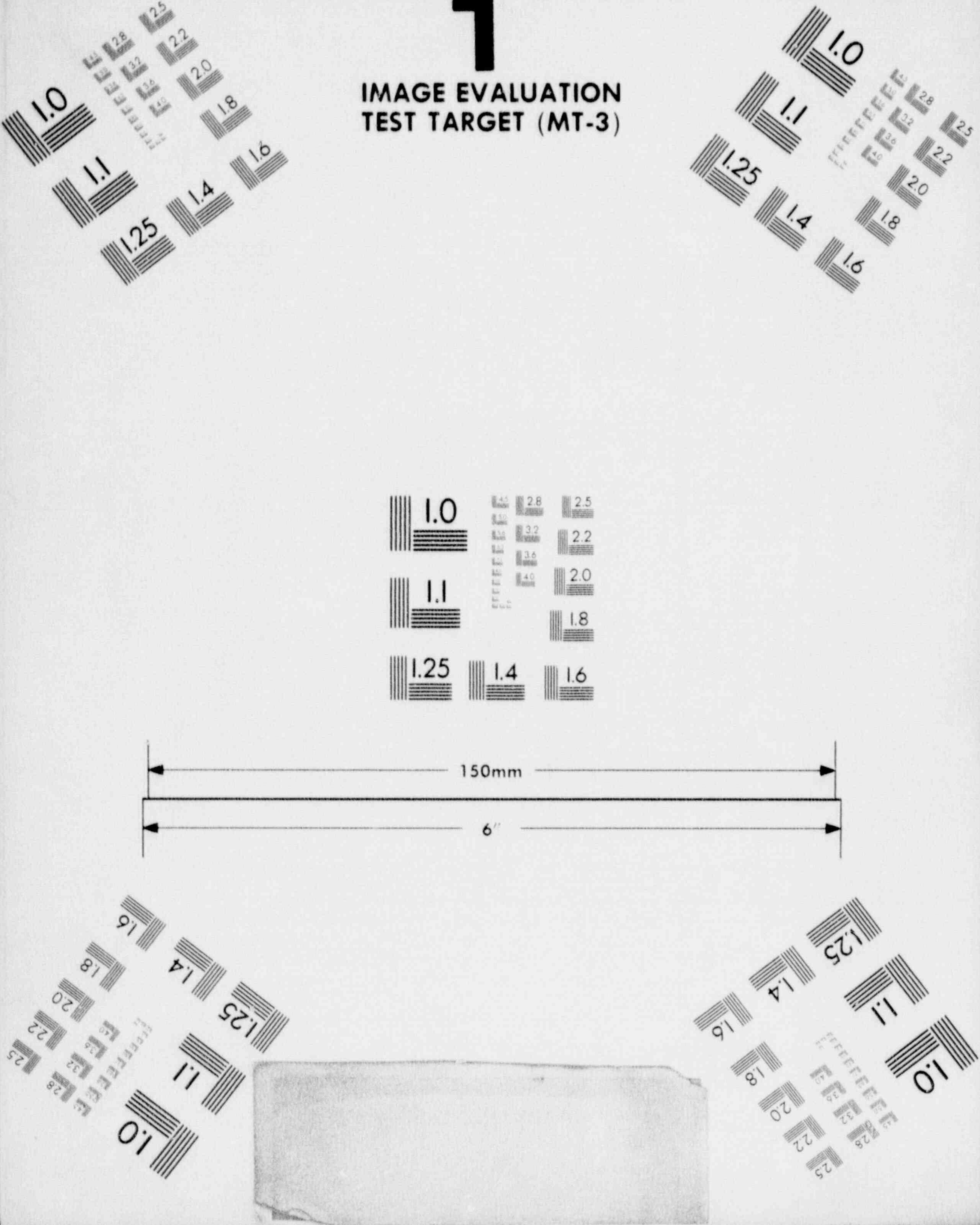
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Flow Rate	$\geq$ 378400 gpm #

R33

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

Includes a 3.5% flow measurement uncertainty

R33

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL - FQ(Z) and $F^N_{\Delta H}$

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.

R

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.

hot channel factors

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 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.

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INSERT 4

~~$F^N_{\Delta H}$ 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 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.~~

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The ~~relaxation in~~ $F^N_{\Delta H}$ LIMIT as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

~~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.~~

Insert 5

The hot channel factor $F^M_Q(z)$ 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. R

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.

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INSERT 5

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty.

The applicable value of rod bow penalty is referenced in the FSAR.

Margin in excess of the rod bow penalty is available for plant design flexibility.

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. R

greater than or equal to the
safety analysis DNBR limit

THE SAFETY ANALYSIS DNBR LIMIT

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times. R75

In MODE 4, a single reactor coolant loop or residual heat removal (RHR) loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

In MODE 5 single failure considerations require that two RHR loops be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-89-33)

DESCRIPTION AND JUSTIFICATION FOR
VANTAGE 5 HYBRID FUEL UPGRADE

ENCLOSURE 2

Description of Change

The Tennessee Valley Authority (TVA) plans to refuel and operate Sequoyah Nuclear Plant (SQN) Units 1 and 2 with Vantage 5 Hybrid (V5H) fuel that incorporates low-pressure-drop zircaloy grids and removable top nozzles, integral fuel burnable absorbers, and extended burnup capabilities. This upgraded fuel will also contain debris filter bottom nozzles, snag resistant grids, and standardized pellets. The evaluations performed for this fuel upgrade accommodate the effects from the following modifications planned for the Cycle 4 outages for each unit:

1. Resistance temperature detector bypass elimination
2. Eagle 21 digital protection system
3. Upper head injection removal
4. Boron injection tank removal
5. New steamline break protection
6. Reactor trip on steam flow/feed flow mismatch

As a result of this fuel upgrade, TVA proposes to modify the SQN Units 1 and 2 technical specifications (TSs) to revise the bases for safety limits to change the W-3 correlation to the WRB-1 correlation and to revise the associated departure from nucleate boiling ratio (DNBR) limits; to revise TS 3.1.3.4 to incorporate a new rod drop time of less than or equal to 2.7 seconds; to revise TS 3.2.3 to delete the rod bow penalty as a function of burnup in the $F^N_{\Delta H}$ (Nuclear Enthalpy Hot Channel Factor) equation and delete Figure 3.2-3; and revise Table 3.2-1 of TS 3.2.5 to define departure from nucleate boiling (DNB) related reactor coolant system (RCS) total flow rate limit, including uncertainties, to be 378,400 gallons per minute (gpm).

Reason for Change

The change of the W-3 correlation to the WRB-1 correlation and the revision to the design DNBR limits in the bases of the safety limits and the increase in the minimum rod drop time are required to allow implementation of the improved fuel design for V5H fuel. The deletion of rod bow penalty as a function of burnup in the $F^N_{\Delta H}$ equation and deletion of Figure 3.2-3 reflect new evaluation methodologies for the effect of fuel rod bow on DNB. The new methodologies provide a basis to eliminate unnecessary power distribution penalties and to simplify the specification. Relocation of the RCS flow rate requirement from TS 3.2.3 to TS 3.2.5 is also the result of new evaluation methodologies for the effect of rod bow on DNB. This change clearly defines the DNB flow parameter limit.

In summary the proposed changes are primarily the result of the following three items:

1. Use of a new DNB correlation
2. Increased rod drop time because of the reduced guide tube diameter for V5H zircaloy grids
3. Incorporation of current methodology to assess the rod bow penalty

Justification for Change

As discussed in the safety evaluation for the fuel upgrade (Enclosure 4), the previously reviewed and licensed safety limits for SQN are met with the upgraded fuel. The new fuel design has provided satisfactory operational performance in fuel assembly demonstration programs since the early 1980s. The V5H fuel is both mechanically and hydraulically compatible with the current SQN fuel assemblies, control rods, and reactor internals interfaces.

The V5H fuel satisfies the current design bases for SQN, and it meets design requirements for hydraulic stability and structural integrity under seismic/loss of coolant accident (LOCA) loads, with margins comparable to 17 x 17 STD (standard) fuel assemblies. Nuclear characteristics are comparable within the range normally seen from cycle to cycle because of fuel management effects.

No change in fuel rod design criteria, methods, or model is necessary with transition to V5H, with the exception of a new DNB correlation. Based upon the information provided in the evaluation, the SQN plant operational limits will be satisfied with the proposed changes.

The evaluation considered the effects of the proposed TS changes on the following areas:

1. Mechanical, nuclear, and thermal-hydraulic fuel assembly design
2. Non-LOCA accidents
3. LOCA accidents
4. Environmental consequences of accidents

These areas have been evaluated for the impact of all proposed changes, including the transition core effects (with a mixed core fuel loading with both V5H and 17 x 17 STD fuel). The required analyses, as described in the fuel upgrade evaluation, were performed by Westinghouse Electric Corporation using methods and procedures previously approved by NRC.

DNB Correlation Change

The calculational methods currently used for 17 x 17 STD fuel assemblies and described in the SQN Final Safety Analysis Report (FSAR) are applicable to V5H fuel assemblies, except for the DNB correlation. The new correlation basis for DNB performance is the WRB-1 correlation.

The WRB-1 correlation established a DNB limit that provides for the margin of safety required by the current FSAR (i.e., DNB will not occur on at least 95 percent of the limiting fuel rods during normal and operational transients and any transient condition arising from faults of moderate frequency at a 95 percent confidence level). The WRB-1 correlation takes credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations.

Increased Rod Drop Time

The V5H fuel design incorporates a snag resistant, low-pressure-drop zircaloy grid. The zircaloy grid will provide for an enchanced performance relative to the current 17 x 17 STD fuel used at SQN.

Utilization of zircaloy as a grid material instead of Inconel reduces the source of cobalt in the core. Consequently, radiation fields caused by the transport of activated cobalt should be lower. The snag-resistant feature results from outer grid straps that are modified to reduce the potential for grid damage and assembly hang-up from assembly interactions during fuel assembly removal. The zircaloy grid also contains features that minimize hydraulic resistance.

In order to maintain mechanical compatibility between the V5H grid and guide tube, a reduction in the V5H guide tube diameter was required. The allowable rod drop time of TS 3.1.3.4 must be increased because of the increased dashpot effect resulting from the guide tube diameter reduction.

Elimination of Rod Bow Penalty

Fuel rod bow has been observed in Westinghouse cores. The phenomena of fuel rod bowing must be accounted for in the DNB safety analyses of Condition I and II events. The current licensing basis offsets the DNB effects of rod bow by partially accommodating it with margin in the W-3 correlation. The remainder of the rod bow penalty is applied as a penalty on the $F^N_{\Delta H}$ evaluation.

New statistical methods have been developed by Westinghouse that verify that the past treatment of rod bow penalty provided an overestimation of the effects on DNB. Application of the new methods to SQN for the standard and the V5H fuel products has verified the reduction in rod bow penalty. The reduction allows for accommodation of the entire penalty in the establishment of the safety limit DNBR.

The requested change to SQN TS 3.2.3 will remove an unnecessary power peaking penalty and simplify the format of the specification. The current specification format includes RCS flow not only to maintain the minimum required RCS flow but also to provide for an additional offset against rod bow penalty. The use of RCS flow to compensate for rod bow penalty will no longer be required.

The flow limit (and its associated uncertainty factor) as it relates to DNB will be moved to TS 3.2.5, DNB parameters, as discussed below. Similar TS changes related to rod bow penalty have been performed for Farley Units 1 and 2, North Anna Units 1 and 2, Beaver Valley Unit 1, and Salem Unit 2.

Definition of DNB Parameter RCS Flow Limit

The RCS flow limit and its associated uncertainty factor have been moved to TS 3.2.5, DNB parameters, which now establishes a minimum allowable RCS flow to prevent violation of the safety limit DNB during normal operation and accident conditions.

The minimum flow rate is based on a thermal design flow rate of 365,600 gpm plus the application of a correction for measurement uncertainties (minimum flow rate = 365,600 gpm x uncertainty factor). The uncertainty factor of 3.5 percent is based on flow measurement uncertainties and feedwater venturi fouling. Therefore, the RCS flow limit for SQN's units is 365,600 gpm x 1.035, or 378,400 gpm.

Environmental Impact Evaluation

The proposed change request does not involve an unreviewed environmental question because operation of SQN Units 1 and 2 in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the Staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or decisions of the Atomic Safety and Licensing Board.

2. Result in a significant change in effluents or power levels.

3. Result in matters not previously reviewed in the licensing basis for SQN that may have a significant environmental impact.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-89-33)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
FOR VANTAGE 5 HYBRID FUEL UPGRADE

ENCLOSURE 3

Significant Hazards Evaluation

TVA has evaluated the proposed TS change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of SQN in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The evaluations of the mechanical, nuclear, and thermal-hydraulic design effects support the conclusion that the requested changes are within the current design criteria established in the FSAR. Consequently, no new mechanisms have been introduced to increase the probability of a previously analyzed accident occurring. The accident evaluations (both LOCA and non-LOCA) exhibit results that maintain the confidence level in the physical integrity of the fission product boundaries as defined in the FSAR. Therefore, the consequences of the accidents do not increase.

(2) Create the possibility of a new or different kind of accident from any previously analyzed.

The evaluations performed established that the FSAR design criteria and system responses during normal and accident conditions are bounding with respect to the proposed changes. The changes will not affect the function of any protection system, and they will not introduce hardware that is different in design criteria requirements. Therefore, no new mechanisms have been introduced that would create the possibility of a new or different kind of accident from those previously analyzed.

(3) Involve a significant reduction in a margin of safety.

The evaluations performed addressed all design criteria and accident analyses. In performing the evaluations, the safety limits established by the FSAR and TS were not modified such as to reduce the difference between the safety limit and the limit defined as the failure point of a fission product boundary. Therefore, the margins that were assumed in the accident analyses remain bounding for the proposed changes.

ENCLOSURE 4

PLANT SAFETY EVALUATION FOR SEQUOYAH NUCLEAR PLANT

UNITS 1 AND 2

VANTAGE 5 HYBRID FUEL UPGRADE

PLANT SAFETY EVALUATION

FOR

SEQUOYAH NUCLEAR PLANT

UNITS 1 AND 2

VANTAGE 5H FUEL UPGRADE

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Editors:

B. W. Gergos
L. V. Tomasic

Contributors: F. Baskerville, R. Brashier, C. Brockhoff, J. Grover, D. Kelly, R. Kemper, P. Kersting, J. Killimayer, N. Pogorzelski, L. Schaub, W. Schivley

Approved: ______________________
E. H. Novenstern, Manager
T/H Design & Fuel Licensing

WESTINGHOUSE ELECTRIC CORPORATION
Commercial Nuclear Fuel Division
P. O. Box 3912
Pittsburgh, Pennsylvania 15230

Attachment to THFL-89-699

PLANT SAFETY EVALUATION
FOR THE
SEQUOYAH NUCLEAR PLANTS UNITS 1 AND 2 FUEL UPGRADE

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1.0 INTRODUCTION AND SUMMARY

It is planned to refuel and operate the Sequoyah Nuclear Plant Units 1 and 2 with Westinghouse fuel employing advanced fuel product features. As a result, future core loadings will have fuel assemblies that incorporate a low pressure drop Zircaloy grid, Removable Top Nozzles, Integral Fuel Burnable Absorber and extended burnup capability. This upgraded fuel feature is known as VANTAGE 5 Hybrid (VANTAGE 5H) and has been submitted as Addendum 2-A (Reference 1) to the "VANTAGE 5 Reference Core Report," WCAP-10444-P-A (Reference 2). In addition to the above features, future Sequoyah reloads will also contain Debris Filter Bottom Nozzles, snag resistant grids and standardized pellets; these additional features have been implemented in other Westinghouse reload cores and do not require prior NRC approval since safety evaluations have shown that a 10CFR50.59 determination can be made for these design features. A brief summary of the upgraded fuel features are given below.

VANTAGE 5H Assembly - The VANTAGE 5H fuel assembly design evolved from the current VANTAGE 5, Optimized (OFA) and Standard (STD) fuel assembly designs. It is based on substantial design and operating experience. Design features from each of these previous designs are incorporated into the VANTAGE 5H fuel assembly design. The most significant VANTAGE 5H design change is the use of Zircaloy grids with 0.374 inch OD standard fuel rods. To accommodate the Zirc grids, the VANTAGE 5H thimble tube diameter was modified to be the same as the 17x17 OFA or VANTAGE 5 fuel.

Removable Top Nozzle (RTN) - The RTN differs from the current design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate attachment and removal; and the nozzle plate thickness was reduced to provide additional space for fuel rod growth. In conjunction with the RTN, a long tapered fuel rod bottom end plug is used to facilitate removal and reinsertion of the fuel rods.

Integral Fuel Burnable Absorber (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 fuel stack. In a typical reload core, approximately one third of the fuel rods in the feed region are expected to include IFBAs. IFBAs provide power peaking and moderator temperature coefficient control.

Extended Burnup - The VANTAGE 5H fuel design will be capable of achieving extended burnups. The basis for designing to extended burnup is contained in Reference 3.

Debris Filter Bottom Nozzle (DFBN) - This bottom nozzle is designed to inhibit debris from entering the active fuel region of the core and thereby improves fuel performance by precluding debris related fuel failures. The DFBN is a low profile bottom nozzle design made of stainless steel, with reduced plate thickness and leg height to provide additional space for fuel rod growth. The DFBN is structurally and hydraulically equivalent to the existing standard bottom nozzle.

Snag-Resistant Grids - The snag-resistant grids contain outer grid straps which are modified to prevent assembly hangup from grid strap interference during fuel assembly removal. This was accomplished by changing the grid strap corner geometry and the addition of guide tabs on the outer grid strap.

Standardized Fuel Pellet - The standardized pellet is a refinement to the current pellet designs with the objective of improving manufacturability while maintaining or improving performance. This design incorporates a reduced pellet length and modification to the previous dish size. The chamfer feature which was included in the pellets for Sequoyah Unit 2 will be introduced for Sequoyah Unit 1.

The Sequoyah Units 1 and 2 Plant Safety Evaluation (PSE) is to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present Sequoyah Unit 1 and Unit 2 cores to a core containing the above upgraded features.

The PSE utilizes the standard reload design methods described in Reference 4 and will be used as a basic reference document in support of future Sequoyah Units 1 and 2 Reload Safety Evaluations (RSEs) for upgraded fuel reloads. Sections 2.0 through 5.0 of the PSE provides the

results of the Mechanical, Nuclear, Thermal and Hydraulic and the Accident Evaluations. Appendix A gives a summary of the changes to the Technical Specifications required and the corresponding change pages. The Significant Hazards Evaluation and the Nuclear Safety Evaluation Checklist are provided in Appendices B and C respectively.

Consistent with the Westinghouse standard reload methodology, Reference 4, parameters are chosen to maximize the applicability of the PSE evaluations for future cycles. The objective of subsequent cycle specific RSEs will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this safety evaluation.

The results of evaluation/analysis described herein lead to the following conclusions:

1. The Westinghouse fuel assemblies containing VANTAGE 5H and the additional upgraded fuel features for the Sequoyah Units are mechanically compatible with the current fuel assemblies, control rod and reactor internals interfaces. The current design bases for Sequoyah Units 1 and 2 have been changed as described in this report to accommodate the VANTAGE 5H design.

2. Changes in the nuclear characteristics due to the transition to upgraded fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

3. The upgraded reload fuel assemblies are hydraulically compatible with the fuel assemblies from previous reload cores.

4. The core design and safety analyses results documented in this report show the core's capability for operating safely for the rated Sequoyah design thermal power.

5. Previously reviewed and licensed safety limits are met when the Sequoyah Units are reloaded with upgraded fuel as described in this report. Plant operating

limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Appendix A of this report. A reference has been established upon which to base Westinghouse reload safety evaluations for future reloads with the upgraded fuel features.

2.0 DESIGN FEATURES

2.1 Introduction

The mechanical design of the upgraded fuel assemblies for Sequoyah Units 1 and 2 is the same as previous reload fuel assemblies except the upgraded fuel assemblies will incorporate several fuel design improvements. These improvements include the VANTAGE 5H design features described in Section 1 with the addition of Debris Filter Bottom Nozzles, snag-resistant grids and standardized fuel pellets.

2.2 VANTAGE 5H Fuel Assembly

The primary mechanical difference between the 17x17 VANTAGE 5H design relative to the 17x17 STD fuel currently in the Sequoyah Units is the use of the VANTAGE 5H Zircaloy grid. A comparison of the STD and VANTAGE 5H fuel assembly design parameters is given in Table 2-1. Figure 2-1 demonstrates the similarity of the two designs and shows a comparison of overall dimensions.

Comparative fuel assembly flow testing pressure drop results indicate that the VANTAGE 5H and the STD 17x17 fuel assembly are hydraulically equivalent. Full assembly testing has confirmed that the VANTAGE 5H fuel assembly has hydraulic stability and that the fuel rod contact wear with the spacer grids is within the allowable design limits.

The major components that determine the structural integrity of the fuel assembly are the grids. Mechanical testing and analysis of the VANTAGE 5H Zircaloy grid and fuel assembly have demonstrated that the VANTAGE 5H structural integrity under seismic/LOCA loads will provide margins comparable to the STD 17x17 fuel assembly design and will meet all design bases.

2.3 VANTAGE 5H Zircaloy Grid

The VANTAGE 5H Zircaloy grid is based on the OFA Zircaloy grid design and operating experience. The grid strip thickness, type of strap welding, basic mixing vane design and pattern, method of thimble tube attachment, type of fuel rod support (6 point), material and envelope are identical to the OFA Zircaloy grid.

In order to demonstrate early performance of the Zircaloy grid design, fuel assembly demonstration programs were conducted inserting OFA fuel assemblies containing Zircaloy grids into 14x14, 15x15 and 17x17 cores. Subsequent to the satisfactory performances observed in these programs, the OFA with Zircaloy grids were loaded and have operated successfully since the early 1980's in many Westinghouse cores. This experience is documented in Reference 5. Relative to the OFA grid, the VANTAGE 5H grid includes features that minimize hydraulic resistance while maintaining required structural capability. This evaluation of the VANTAGE 5H grid performance is based on the extensive design and irradiation experience with previous grid designs and full grid testing completed with the VANTAGE 5H grid design.

2.4 Fuel Rod Performance

The 0.374 inch OD fuel rod used in the VANTAGE 5H fuel assembly is the same as that used in the Sequoyah 17x17 STD fuel assemblies. The design basis, methodology, and models are the same as those previously described in References 2 and 3 with the exception that the NRC approved Westinghouse fuel performance models in Reference 6 are being used. No changes in fuel rod design criteria, methods or models are necessary because of the transition to VANTAGE 5H fuel.

FIGURE 2-1

Comparison of the 17X17 VANTAGE 5H Fuel Assembly
and the 17X17 STD Fuel Assembly

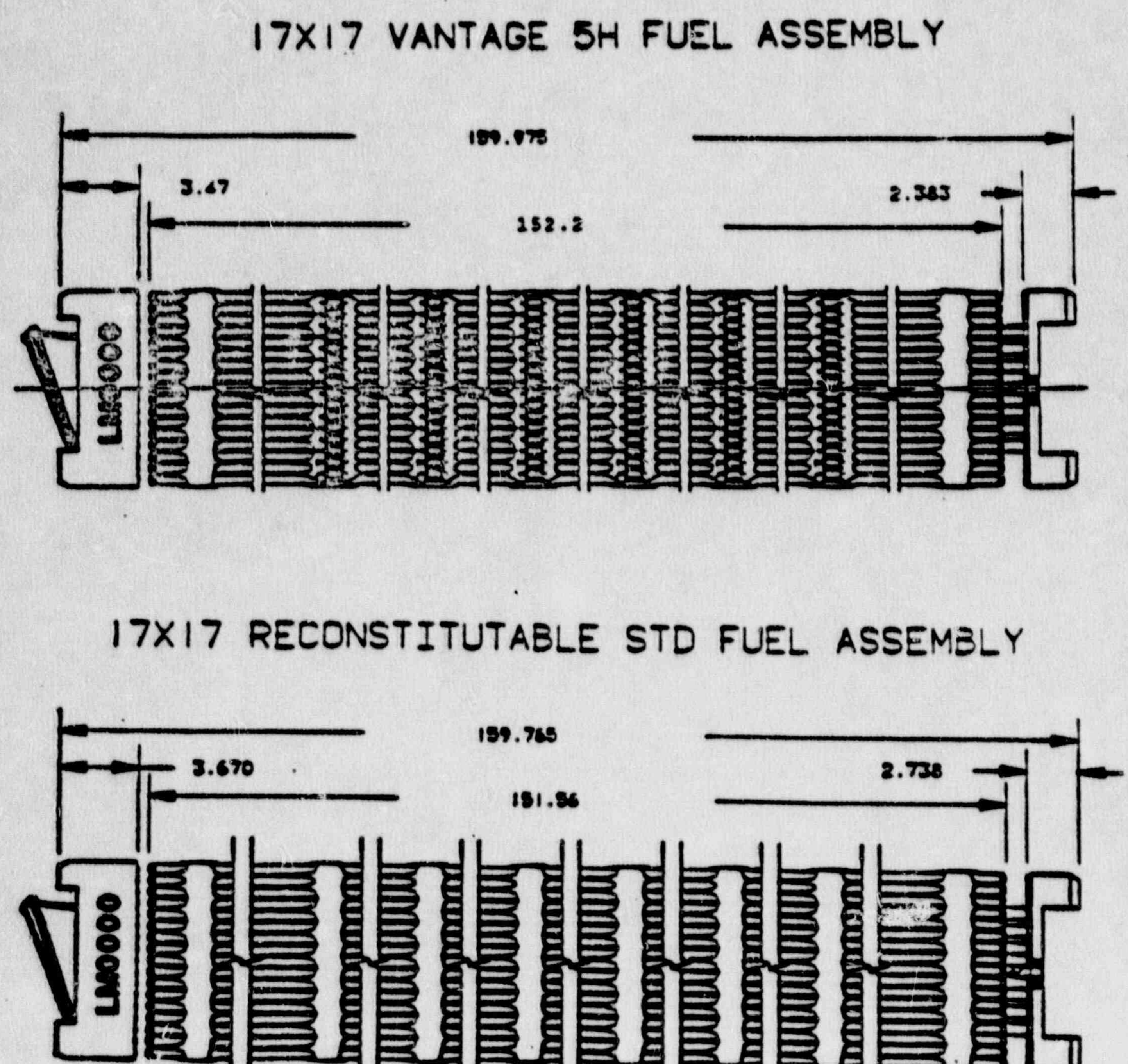


TABLE 2-1

Comparison of 17X17 Standard
and VANTAGE 5 Hybrid Fuel Assembly
Mechanical Design Parameters

	Standard	VANTAGE 5H
Fuel Assembly Overall Length, inch	159.8	160.0
Fuel Rod Overall Length, inch	151.6	152.2
Assembly Envelope, inches	8.426	8.426
Fuel Rod Pitch, inch	0.496	0.496
Number of Fuel Rods/Assembly	264	264
Number of Guide Thimbles/ Assembly	24	24
Number of Instrumentation Tube/Assembly	1	1
Fuel Tube Material	Zirc-4	Zirc-4
Fuel Tube Clad OD, inch	0.374	0.374
Fuel Rod Clad Thickness, inch	0.0225	0.0225
Fuel Clad Gap, mil	6.5	6.5 (uncoated pellets)
Fuel Pellet Diameter, inch	0.3225	0.3225 (uncoated pellets)
Fuel Rod End Plugs	Standard	Tapered
Fuel Pellet Length		
Enriched fuel, inch	0.387	0.387
Unenriched fuel, inch	NA	0.545

TABLE 2-1 (Cont)

Comparison of 17X17 Standard
and VANTAGE 5 Hybrid Fuel Assembly
Mechanical Design Parameters

	Standard	VANTAGE 5H
Relative Clad Thickness/ Diameter Ratio	1.0	1.0
Relative Moderator/Fuel Ratio for Assembly	1.0	1.0
Relative UO_2/Rod	1.0	1.0
Guide Thimble Material	Zirc-4	Zirc-4
Guide Thimble OD, inch	0.482	0.474
Guide Thimble Wall Thickness, inch	0.016	0.016
Grid Material Inner Mid Grid (6)	Inconel	Zirc-4
Edges Modified	No	Yes
Grid Material, End Grids (2)	Inconel	Inconel
Grid Types Utilized		
Inconel Mid Grids	Yes	No
Zircaloy Mid Grids	No	Yes
Inconel Top & Bottom Grids	Yes	Yes
Inner Spring (Mid Grids)	Vertical	Non-Vertical

TABLE 2-1 (Cont)

Comparison of 17X17 Standard
and VANTAGE 5 Hybrid Fuel Assembly
Mechanical Design Parameters

	Standard	VANTAGE 5H
Grid Fabrication Inconel Grids	Brazed joining of interlocking stamped straps	Brazed joining of interlocking stamped straps
Zircaloy Mid Grid	None	Laser weld joining of interlocking stamped straps
Grid/Guide Thimble Attach. Inconel Grids	Thimbles bulged together with sleeves prebrazed	Thimbles bulged together with sleeves prebrazed
Grid/Guide Thimble Attach. Zircaloy Mid Grids	None	Thimbles bulged together with sleeves laser prewelded to straps
Top Nozzle	Welded stainless steel Standard	Removable stainless steel reduced height removable design
Compatible with Fuel Handling Equipment	Yes	Yes

3.0 NUCLEAR DESIGN

3.1 Introduction and Summary

This section provides an assessment of the nuclear design impact from the upgrade in fuel product from 17x17 STD to 17x17 VANTAGE 5H for the Sequoyah units. The fuel design changes impacting nuclear design result from the incorporation of Zircaloy grids, IFBA and increased discharge burnup. The effect of these changes on the core physics characteristics are small and explicitly modeled in the neutronics codes. The specific values of core safety parameters, e.g. power distributions, peaking factors, rod worths, are primarily loading pattern dependent. The variations in the loading pattern dependent safety parameters are also expected to be typical of the normal cycle to cycle variations for the standard fuel reloads. No change in Technical Specifications related to core neutronics behavior result as a consequence of the upgrade in fuel product.

In summary, the change from the current all standard fuel core to a core containing the upgraded fuel product will not cause changes to the current Sequoyah FSAR nuclear design bases.

3.2 Methodology

No changes to the nuclear design philosophy or methods are necessary because of the upgraded fuel product. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the FSAR safety evaluation for each reload cycle. This philosophy is described in Reference 4. These key safety parameters will be evaluated for each Sequoyah reload cycle. If one or more of the parameters fall outside the bounds assumed in the safety analysis, the affected transient will be re-evaluated and the results documented in the RSE for that cycle.

3.3 Design Evaluation

The 0.374 inch diameter fuel rod has had extensive nuclear design and operating experience with the current Sequoyah 17x17 STD fuel assembly design. The Zircaloy grid material has also had extensive nuclear design and operating experience with the current 17x17 VANTAGE 5 and 17x17

OFA fuel assembly designs. IFBAs are used for power distribution control similar to fixed burnable absorbers. The actual peaking factor characteristics are loading pattern dependent. IFBAs may be used with fixed burnable absorbers in the same assembly to increase design flexibility.

3.4 Conclusion

The VANTAGE 5H and STD assembly are neutronically similar. Any variation in design is expected to be within the cycle to cycle variations for the STD core. The nuclear design philosophy and process is unchanged due to the VANTAGE 5H fuel.

4.0 THERMAL AND HYDRAULIC DESIGN

4.1 Introduction and Summary

This section describes the calculational methods used for the thermal-hydraulic analysis, the DNB performance and the hydraulic compatibility during the transition from mixed-fuel cores to an all VANTAGE 5H core. Based on minimal hardware design differences and prototype hydraulic testing of the fuel assemblies, it is concluded in Reference 1 that the STD and VANTAGE 5H fuel assembly designs are hydraulically compatible. Table 4-1 summarizes the thermal-hydraulic design parameters for Sequoyah Units 1 and 2 that were used in this analysis. In addition, the current rod bow methodology is implemented to reduce the rod bow penalty described in Sequoyah Technical Specifications. The thermal-hydraulic design criteria and methods remain the same as those presented in the Sequoyah FSAR with the exception noted in the following sections. All of the current FSAR thermal-hydraulic design criteria are satisfied.

4.2 Methodology

Calculational methods currently used on the 17x17 STD fuel assembly and described in the Sequoyah FSAR are applicable to the evaluation of the core containing both 17x17 STD and VANTAGE 5H fuel assemblies, except for the DNB correlation. The analyses for STD and VANTAGE 5H fuel will utilize the WRB-1 DNB correlation (Reference 7). The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations. As documented in Reference 7, a 95/95 limit DNBR of 1.17 is appropriate for the STD fuel assemblies.

The WRB-1 DNB correlation is applicable to VANTAGE 5H fuel since, from a DNB perspective, the VANTAGE 5H assembly is virtually identical to the 17x17 STD design. As documented in Reference 1, the use of the WRB-1 DNB correlation with a 95/95 limit DNBR of 1.17 is applicable to the VANTAGE 5H fuel assembly.

DNBR margin is maintained for the STD and VANTAGE 5H fuel by performing the DNB safety analysis to a DNBR limit of 1.38. Comparing this limit of 1.38 to the WRB-1 correlation limit of

1.17 results in 15.2% DNBR margin. This DNBR margin is more than sufficient to offset the DNBR penalty due to rod bow (see Section 4.4).

Table 4-2 summarizes the DNBR margin and the allocation of margin for the Sequoyah plants. DNBR margin in excess of the allocation to cover DNBR penalties is available for plant design flexibility.

4.3 Hydraulic Compatibility

The STD fuel assembly and VANTAGE 5H design have been shown in Reference 1 to be hydraulically compatible.

4.4 Effects of Fuel Rod Bow on DNBR

The phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Currently the rod bow penalty is partially offset by the generic DNBR margin. The remaining rod bow penalty is applied as a penalty on $F_{\Delta H}^{N}$ as described in the Sequoyah Technical Specifications, which defines the $F_{\Delta H}^{N}$ as a function of region average burnup to 33,000 MWD/MTU. For this application the rod bow penalty was reassessed using improved methodology.

Using the current methodology described in References 8, 9 and 10, a rod bow penalty of <1.5% is applicable to 17x17 STD fuel assemblies. Based on the similarities between 17x17 STD and VANTAGE 5H fuel assemblies, (i.e. fuel rod diameter, fuel rod pitch and grid spacing), this penalty is also applicable to VANTAGE 5H fuel assembly.

This penalty is the maximum rod bow penalty at an assembly average burnup of 24,000 MWD/MTU. For burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F_{\Delta H}^{N}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory. Therefore, no additional rod bow penalty is required at burnups greater than 24,000 MWD/MTU.

4.5 Fuel Temperature Analysis

The fuel temperatures for use in safety calculations for the VANTAGE 5H fuel are identical to those for the STD fuel.

4.6 Transition Core Effect

The VANTAGE 5H hydraulic test program showed identical results for the VANTAGE 5H Zircaloy mixing vane grid and the STD fuel Inconel mixing vane grid. Therefore, no transition core DNBR penalty is necessary.

4.7 Conclusion

The thermal-hydraulic analysis has shown that 17x17 STD and VANTAGE 5H fuel assemblies are hydraulically compatible and sufficient DNBR margin in the safety limit DNBR exists to cover the rod bow penalty. All current thermal-hydraulic design criteria are satisfied.

TABLE 4-1

Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters		Design Parameters
Reactor Core Heat Output, MWt		3411
Reactor Core Heat Output, 10^6 BTU/hr		11,642
Heat Generated in Fuel, %		97.4
Core Pressure, Nominal, psia		2250
Radial Power Distribution		1.55[1+0.3(1-P)]
Limit DNBR for Design Transients		
	(STD)	1.38
	(V5H)	1.38
DNB Correlation		
	(STD)	WRB-1
	(V5H)	WRB-1
HFP Nominal Coolant Conditions		
Vessel Thermal Design Flow Rate (including Bypass),		
10^6 lbm/hr		138.0
GPM		365,600
Core Flow Rate* (excluding Bypass, based on TDF)		
10^6 lbm/hr		127.7
GPM		338,180
Fuel Assembly Flow Area for Heat Transfer, ft^2	(STD)	51.1
	(V5H)	51.3
Core Inlet Mass Velocity, 10^6 lbm/hr-ft^2 (Based on TDF)	(STD)	2.50
	(V5H)	2.49

* Based on design bypass flow of 7.5%.

TABLE 4-1 (Continued)

Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters		Design Parameters
Nominal Vessel/Core Inlet Temperature, °F		546.7*
Vessel Average Temperature, °F		578.2
Core Average Temperature, °F		582.2
Vessel Outlet Temperature, °F		609.7
Average Temperature Rise in Vessel, °F		63.0
Average Temperature Rise in Core, °F		67.6
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	(STD)	59,700
	(V5H)	59,700
Average Heat Flux, BTU/hr-ft^2	(STD)	189,800
	(V5H)	189,800
Average Linear Power, kw/ft		5.44
Peak Linear Power for Normal Operation,+ kw/ft		12.6
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F		4700

---

* Safety analysis inlet temperature is 548.2°F

+ Based on 2.32 F_Q peaking factor

TABLE 4-2

DNB Margin Summary

DNB Correlation	WRB-1
Correlation Limit	1.17
Safety Limit	1.38
DNBR Margin+	15.2%
DNBR Penalties	
Rod Bow	<1.5%
Transition Core	0.0%

+ DNBR margin between the safety limit DNBR and the correlation limit DNBR.

5.0 ACCIDENT ANALYSIS

5.1 Non-LOCA Accidents

This section summarizes the reanalysis and evaluations performed for the Sequoyah Units upgrade to VANTAGE 5H fuel.

The major effect of changing from STD 17x17 fuel to VANTAGE 5H fuel on the non-LOCA transients is the increased Rod Control Cluster Assembly (RCCA) drop time. The VANTAGE 5H fuel assembly has a thimble tube I.D. of 0.442 inches. STD fuel has a thimble tube I.D. of 0.450 inches. The smaller VANTAGE 5H thimble tube will increase the RCCA drop time from a current limit of 2.2 seconds to 2.7 seconds. This slower drop time will affect the results of the fast non-LOCA limiting transients such as Partial and Complete Loss of Forced Reactor Coolant Flow, RCCA Bank Withdrawal from Subcritical, Single Reactor Coolant Pump Locked Rotor and Rod Ejection. These transients were reanalyzed and a summary of the results of the analyses are discussed on the following pages. In addition, Startup of an Inactive Loop was analyzed because of the change in the DNBR correlation.

Non-LOCA events not mentioned above were not reanalyzed for the increased rod insertion time for one or more of the following reasons:

1) Transient results are insensitive to the rod insertion rate.
2) Reactor trip was not assumed or explicitly modeled in the analysis.
3) Reactor trip has no effect on the minimum or maximum value of the critical parameter of interest.
4) Event may be impacted, but the magnitude of the impact is small when compared to the margin to the design basis limit.

The non-LOCA analyses and evaluations for VANTAGE 5H were performed taking into account the effects from the following programs:

- Eagle 21 Digital Protection System
- Resistance Temperature Detector (RTD) Bypass Elimination
- New Steamline Break Protection

- Low Feedwater Flow Reactor Trip Elimination
- Boron Injection Tank (BIT) Removal
- Upper Head Injection (UHI) Removal

Uncontrolled RCCA Withdrawal from a Subcritical Condition (FSAR Section 15.2.1)

The RCCA withdrawal from subcritical is characterized by a rapid power increase. The power excursion is retarded by Doppler feedback and the transient is terminated by a reactor trip. Due to the high rate at which the power increases, this transient can be sensitive to RCCA drop time.

The RCCA withdrawal from subcritical was reanalyzed with a RCCA drop time to the dashpot of 2.7 seconds. Transient results are shown in Figures 5.1-1 through 5.1-3. Figure 5.1-1 shows the neutron flux transient. The neutron flux overshoots the full power nominal value, but this occurs for only a very short time period. The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 5.1-2. The beneficial effect on the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 5.1-3 shows the response of the average fuel and cladding temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limits.

The core and the Reactor Coolant System (RCS) are not adversely affected, since the combination of thermal power and the coolant temperature result in a minimum DNBR well above the limiting value. Thus no fuel or clad damage will occur.

Uncontrolled RCCA Bank Withdrawal at Power (FSAR Section 15.2.2)

This event is analyzed to show that the DNB design basis is met. The Overtemperature ΔT setpoints are not changed, so the time of reactor trip remains the same. The transient minimum DNBR occurs immediately following the reactor trip. A conservative evaluation of the effects of the 0.5 second increase in control rod drop time was done by extrapolating the transient DNBR results assuming that the reactor trip was delayed by 0.5 second. The extrapolations showed that ample margin to the DNB limit still exists with a 0.5 second delay. The conclusions of the FSAR remain valid.

Rod Cluster Control Assembly Misalignment (FSAR Section 15.2.3)

These events are analyzed to show that the DNB design basis is met. The dropped rod analysis was updated for VANTAGE 5H fuel and the increased RCCA drop time using the current Westinghouse methodology. For a 2.7 second drop time, the maximum rod worth which is considered in the analysis is 500 pcm. For rod worths above this limit, a reactor trip is assumed to occur and no further analysis is needed. The negative flux rate trip protection system might not, however, detect rod worths less than 500 pcm. Therefore analyses are performed each cycle to ensure that the DNB design limit is met for dropped rod worths less than 500 pcm. The Statically Misaligned RCCA events are not impacted by the increase in rod drop time. Thus, the analysis results are unaffected and the conclusions of the FSAR remain valid.

Uncontrolled Boron Dilution (FSAR Section 15.2.4)

Acceptable results for this event are demonstrated by showing that there is sufficient time for operator action to terminate a dilution prior to a return to criticality, which could lead to core damage. For the case analyzed at power with manual rod control, the time of trip is determined by comparison to the RCCA bank withdrawal at power analysis. As previously noted, the time of reactor trip on Overtemperature ΔT is unchanged. The mechanics of the reactor trip are not explicitly modeled in the analysis. Thus, the analysis results are unaffected by the increased rod insertion time and the conclusions of the FSAR remain valid.

Partial and Complete Loss of Forced Reactor Coolant Flow (FSAR Section 15.2.5 & 15.3.4)

The loss of flow transients are characterized by a rapid decrease in core flow. If the reactor is not tripped promptly DNB may occur with subsequent fuel damage. Thus these transients can be sensitive to the RCCA drop time. The partial and complete loss of flow transients were reanalyzed with a RCCA drop time (time to dashpot) of 2.7 seconds. Figures 5.1-4 through 5.1-6 show the results of the partial loss of flow. The results of the complete loss of flow transient are shown in Figures 5.1-7 through 5.1-9. In both cases the minimum DNBR remains above the limit. The underfrequency Loss of Flow transient was also analyzed and the result was that the minimum DNBR remains above the limit.

Startup of an Inactive Reactor Coolant Loop (FSAR Section 15.2.6)

This event is analyzed to show that the DNB design basis is met. Minimum DNBR occurs immediately following a reactor trip on high neutron flux. This transient was reanalyzed to be

consistent with the analysis for VANTAGE 5H fuel. Figures 5.1-10 and 5.1-11 show the results. The minimum DNBR remains above the limit. The conclusions of the FSAR remain valid.

Loss of External Electrical Load and/or Turbine Trip (FSAR Section 15.2.7)

This event is analyzed to show that the DNB design basis is met and that primary and secondary side system pressures do not exceed 110% of design values. Four cases are analyzed which are:

BOL with pressurizer pressure control

BOL without pressurizer pressure control

EOL with pressurizer pressure control

EOL without pressurizer pressure control

The increased rod insertion time to the dashpot will not result in system pressures exceeding 110% of design values. Pressure transients from the current analysis of record were evaluated by extrapolation assuming the reactor trip was delayed 0.5 seconds. In all cases there was ample margin to account for the slight expected pressure rise due to the slower drop times. The increased rod drop time to the dashpot will not result in DNBR below the design limit. DNBR for the BOL without pressure control and both EOL cases rises continuously throughout the transients. Therefore, the increased insertion time will have no effect on the minimum DNBR for these cases. DNBR during the BOL with pressure control case initially rises and then decreases to about the initial value at the time of reactor trip for this case and the increased rod drop time will not result in a DNBR below the design basis.

Loss of Normal Feedwater (FSAR Section 15.2.8)

This event is analyzed to show that adequate heat removal capability exists via the Auxiliary Feedwater System to remove core decay heat, stored energy and RCS pump heat following reactor trip. This is demonstrated by ensuring that the RCS heatup is turned around prior to the time when coolant expansion causes the pressurizer to become filled with water. The loss of feedwater transient is a slow long-term heatup event and is not sensitive to the rate at which control rods are inserted following a reactor trip. The results of the current analysis of record and conclusions of the FSAR remain valid.

Loss of Offsite Power to the Station Auxiliaries (FSAR Section 15.2.9)

This event is analyzed to show that adequate heat removal capability exists via natural circulation flow as aided by the Auxiliary Feedwater System to remove core decay heat and stored energy following reactor trip. This is demonstrated by ensuring that the RCS heatup is turned around prior to the time when coolant expansion causes the pressurizer to become filled with water. The RCS volumetric expansion is not affected since the total RCS flow and vessel outlet temperature remain the same. This transient is a slow long-term heatup event and this aspect of this transient is not sensitive to the rate at which the rods are inserted during a reactor trip. With respect to DNB criterion, this event is bounded by the Complete Loss of Forced Reactor Coolant Flow analysis which was reanalyzed and shown to be acceptable. The conclusions of the current analysis of record and the FSAR remain valid.

Excessive Heat Removal due to Feedwater System Malfunctions (FSAR Section 15.2.10)

This event is analyzed to show that the DNB design basis is met. The reactor is tripped on a turbine trip signal and minimum DNBR occurs shortly afterward. A conservative evaluation of the effects of the 0.5 second increase in control rod insertion time was done by extrapolating the transient DNBR results assuming that reactor trip was delayed by 0.5 seconds. The extrapolation showed that ample margin to the DNB limit still exists with a 0.5 second delay. The conclusions of the FSAR remain valid.

Excessive Load Increase Incident (FSAR Section 15.2.11)

This event is analyzed to show that the DNB design basis is met following a step load increase from rated power. Cases are analyzed at BOL and EOL conditions with and without rod control. In all cases analyzed, the reactor stabilized without a reactor trip. Therefore, the increased control rod insertion time will have no effect on this event. The conclusions of the FSAR remain valid.

Accidental Depressurization of the Reactor Coolant System (FSAR Section 15.2.12)

The limiting criteria for this event is the DNB design basis. This transient is terminated by a reactor trip on Overtemperature ΔT. Minimum DNBR occurs immediately following reactor trip. A conservative evaluation of the effect of the 0.5 second increase in control rod drop time was done by extrapolating the transient DNBR results assuming reactor trip was delayed by 0.5 seconds. The extrapolations showed that abundant margin to the DNB limit still exists with a 0.5 second increase in rod insertion time. The conclusions of the FSAR remain valid.

Accidental Depressurization of the Main Steam System (FSAR Section 15.2.13) and Major Rupture of a Main Steamline (FSAR Section 15.4.2.1)

The inadvertent opening of a steam generator relief or safety valve case is analyzed to show that the DNB design basis is met. The steam system piping failure cases are analyzed to show that the core remains intact and in place and that the radiation doses do not exceed the guidelines of 10CFR100. This is demonstrated by showing that the DNB design basis is met, even though DNB and possible clad perforation are not necessarily unacceptable for these cases.

The limiting steamline break cases are analyzed from hot shutdown initial conditions. The transient is started assuming the reactor is tripped and the core is at the minimum design shutdown margin. Therefore the 0.5 second increase in rod insertion time will have no effect on the results of this analysis. The conclusions of the FSAR remain valid.

Spurious Operation of the Safety Injection System at Power (FSAR Section 15.2.14)

The spurious operation of the safety injection system produces a negative reactivity transient causing a reduction in core power. The power reduction causes a decrease in reactor coolant average temperature and consequent coolant shrinkage. Pressurizer pressure and level decrease until the reactor is tripped. During the transient the DNB ratio never decreases below the initial value, therefore the 0.5 second increase in control rod insertion time will have no effect on the minimum DNBR. The conclusions of the FSAR remain valid.

Single Rod Cluster Control Assembly Withdrawal at Full Power (FSAR Section 15.3.6)

With the reactor in automatic or manual control, an upper bound of the number of fuel rods experiencing DNBR below the safety analysis limit is 5% of the total fuel rods in the core. Because the analyses takes no credit for the control rods entering the core at the time of trip, a 0.5 second increase in the RCCA drop time would have no effect on the results. The conclusions of the FSAR remain valid.

Major Rupture of a Main Feedwater Pipe (FSAR Section 15.4.2.2)

This event is analyzed to show that adequate heat removal capability exists using the auxiliary feedwater system to remove core decay heat, stored energy and RCS pump heat following reactor trip. This is demonstrated by ensuring that the RCS heatup is turned around prior to the time

at which the hotlegs would become saturated and the peak primary and secondary pressures would exceed 110% of design values.

The feedline break is a long term heatup event and is not sensitive to the rate at which the control rods are inserted following a reactor trip. The heatup transient continues for many minutes following the reactor trip. The 0.5 second increases in control rod insertion time will result in an insignificant increase in the integrated heat produced by the core during the transient. No significant increase in hotleg temperature or system pressures would occur due to the increase in control rod insertion time. The conclusion of the current record of analysis and the FSAR remain valid.

Single Reactor Coolant Pump Locked Rotor (FSAR Section 15.4.4)

The accident is postulated as an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected coolant pump is rapidly reduced leading to an initiation of a reactor trip on a low flow signal. If the reactor is not tripped promptly clad temperature may exceed the limit value of 2700°F and RCS pressure may increase above that which would cause stresses to exceed the faulted condition stress limits. This transient can be very sensitive to the RCCA drop time.

The locked rotor transient was reanalyzed with a RCCA drop time of 2.7 seconds. The results of the analysis are shown in Figures 5.1-12 through 5.1-14 and Table 5.1-2.

Since the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

Since the peak clad surface temperature calculated for the hot spot remains less than 2700°F and the amount of Zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability.

Rupture of a Control Rod Drive Mechanism Housing (FSAR Section 15.4.6)

The RCCA ejection transients are characterized by a rapid power burst. Due to the speed at which the power increases this transient can be sensitive to the RCCA drop time. The RCCA transients were reanalyzed with a RCCA drop time (time to dashpot) of 2.7 seconds. The limiting

transients were reanalyzed with a RCCA drop time (time to dashpot) of 2.7 seconds. The limiting criteria for this event (References 12 and 13) are:

1) Average fuel pellet enthalpy at the hot spot limited to below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

2) Fuel melting limited to less than the innermost 10 percent of the pellet at the hot spot. (Melting is assumed to occur at 4900°F for BOL conditions and 4800°F for EOL conditions)

The results of the analysis and a summary of parameters used in the analysis are shown in Table 5.1-1. All cases analyzed meet the acceptance criteria.

Mass/Energy Release (Steamline Breaks) (FSAR Section 6.2)

The mass and energy releases inside containment following a steamline break are used in the containment integrity analysis and are insensitive to the rate at which the control rods are inserted. The 0.5 second increase in control rod insertion time would increase the integrated energy produced by the core by an insignificant amount. The total RCS flow rate will be the same and no significant change in the overall system response will occur. Therefore, the conclusions of the FSAR with respect to containment integrity are not affected.

The mass and energy releases outside containment following a steamline break are used to assure that environmental conditions used for instrument qualifications are maintained. As with the mass and energy released inside containment, the 0.5 second increase in control rod insertion time would increase the integrated energy produced by the core by an insignificant amount. The total RCS flow rate will be the same and no significant change in the overall system response will occur.

Steamline Break with Coincident Rod Withdrawal at Power

This event is analyzed to show that the DNB design basis is met. The transient minimum DNBR occurs immediately following the reactor trip. The limiting case was examined with the new RCCA drop time of 2.7 seconds, and results were compared with those obtained from the previous RCCA drop time of 2.2 seconds. The decrease in minimum DNBR was negligible and ample margin to the DNB limits exists for the increased RCCA drop time of 2.7 seconds.

TABLE 5.1-1
Rod Cluster Control Assembly Ejection
Accident Results

Time in Life	BOL	BOL	EOL	EOL
Power Level	102	0	102	0
Ejected Rod Worth (%ΔK)	0.20	0.75	0.21	0.97
Delayed Neutron Fraction	0.55	0.55	0.44	0.45
Feedback Reactivity Weighting	1.3	2.4	1.6	3.63
Trip Reactivity (%ΔK)	4.0	2.0	4.0	2.0
F_Q before Rod Ejection	2.52	--	2.52	--
F_Q after Rod Ejection	7.11	14.05	7.88	26.0
Number of Operational Pumps	4	2	4	4
Max. Fuel Average Temperature (°F)	4097	3156	4001	3744
Max. Fuel Center Temperature (°F)	4971	3610	4871	4391
Max. Fuel Stored Energy (cal/gm)	180	132	174	161

TABLE 5.1-2

Summary of Results for Locked Rotor Transients

Maximum Reactor Coolant System Pressure (psia)	2603
Maximum Clad Temperature (^{o}F) Core Hot Spot	2026
Amount of $Zr \cdot H_2O$ at Core Hot Spot (% by Weight)	0.70

FIGURE 5.1-1

Startup from Subcritical
Nuclear Power versus Time

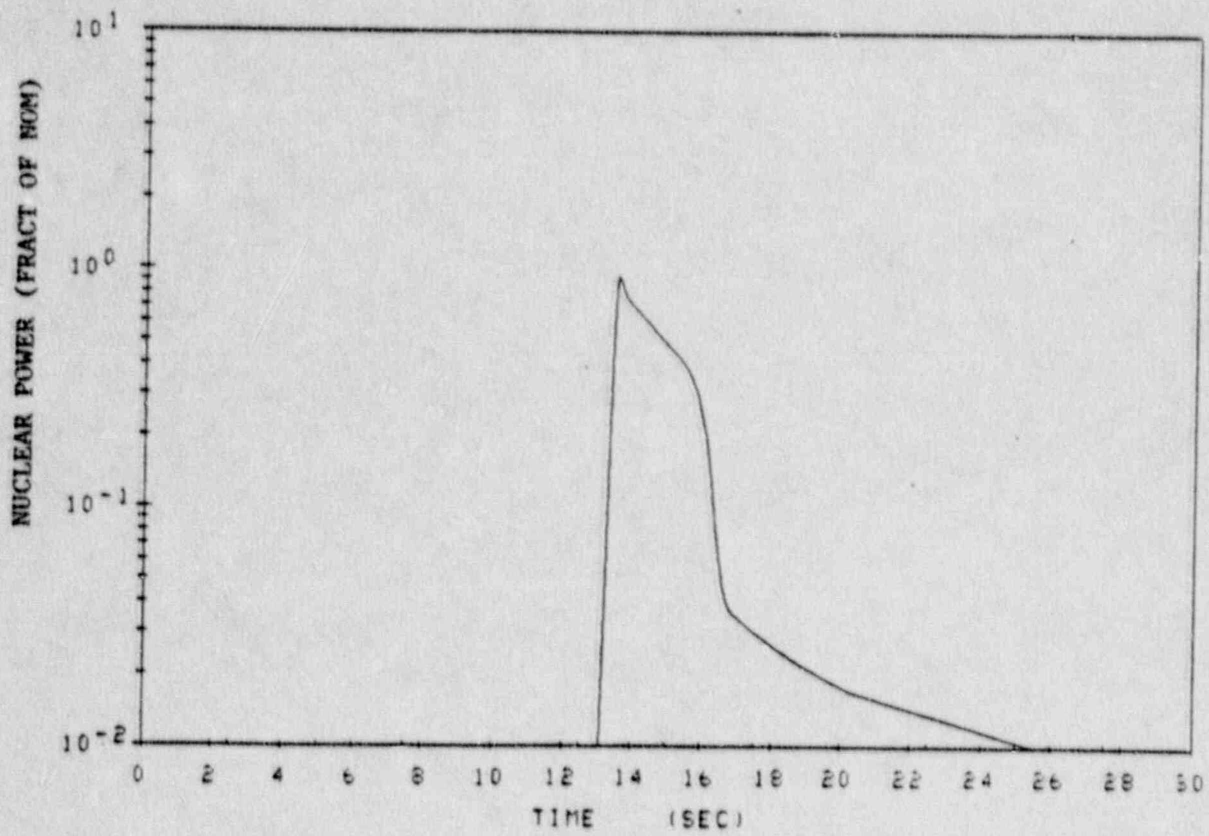


FIGURE 5.1-2

Startup from Subcritical
Heat Flux versus Time

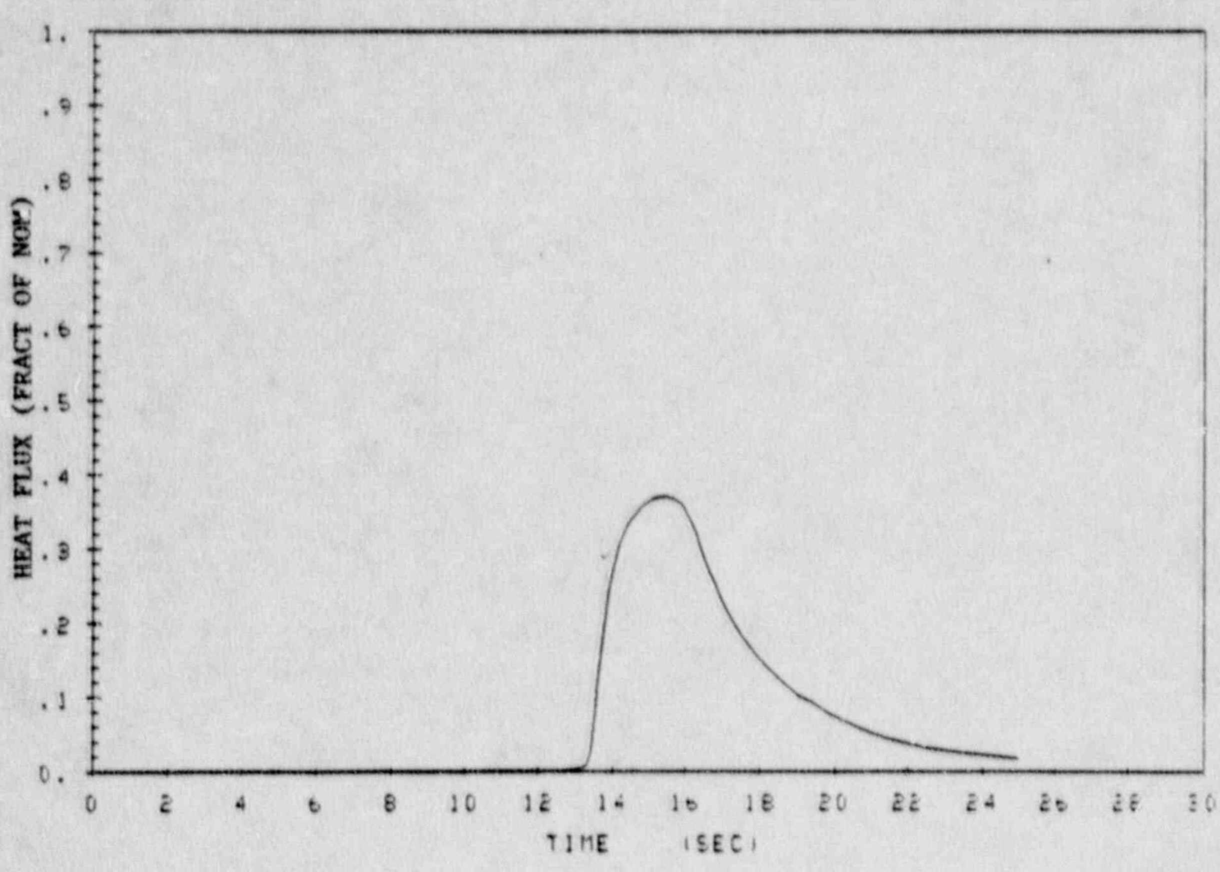


FIGURE 5.1-3

Uncontrolled Rod Withdrawal from a Subcritical Condition
Fuel and Clad Temperature versus Time

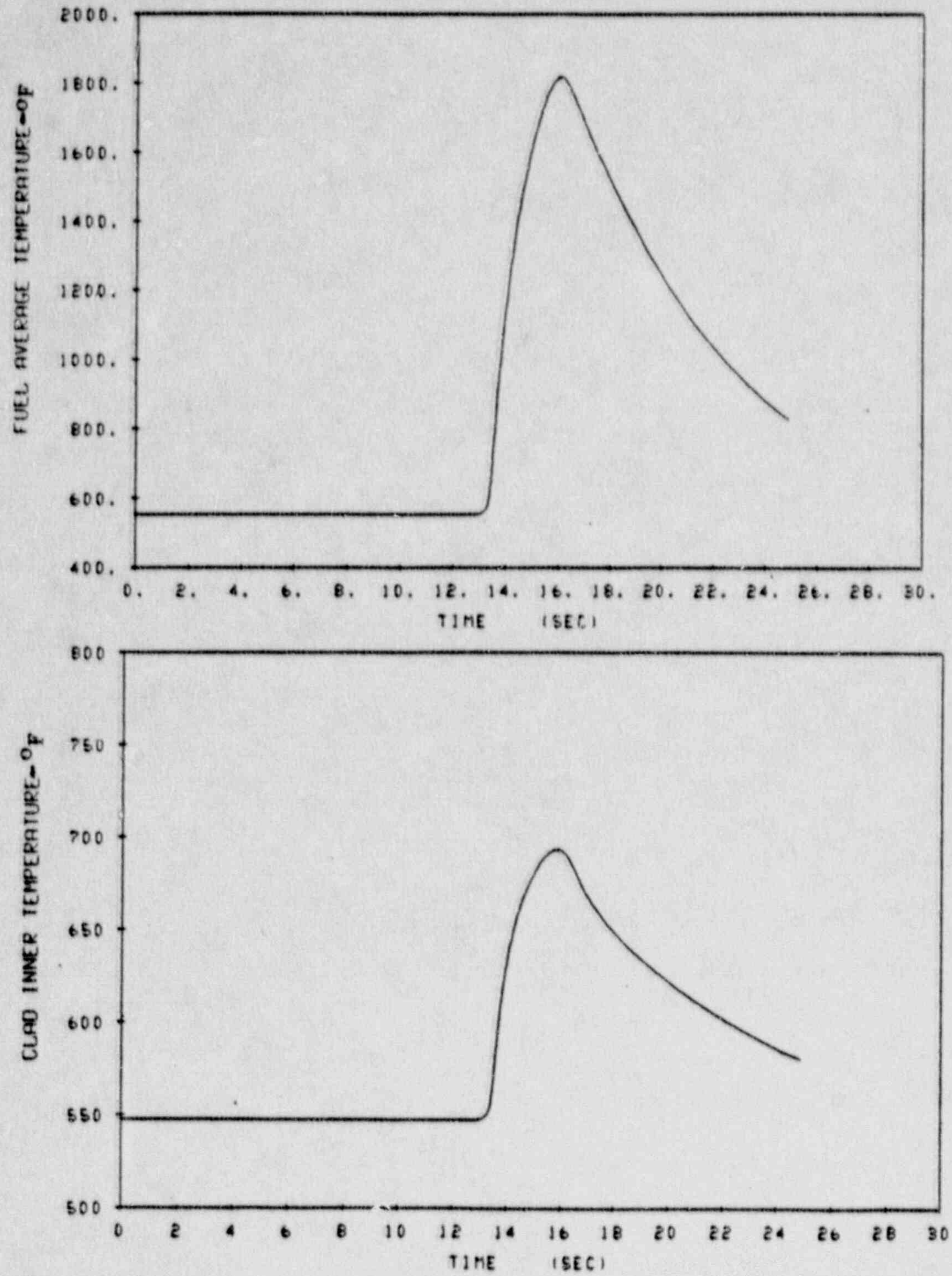


FIGURE 5.1-4

Partial Loss of Forced Reactor Coolant Flow
Core and Loop Flow versus Time

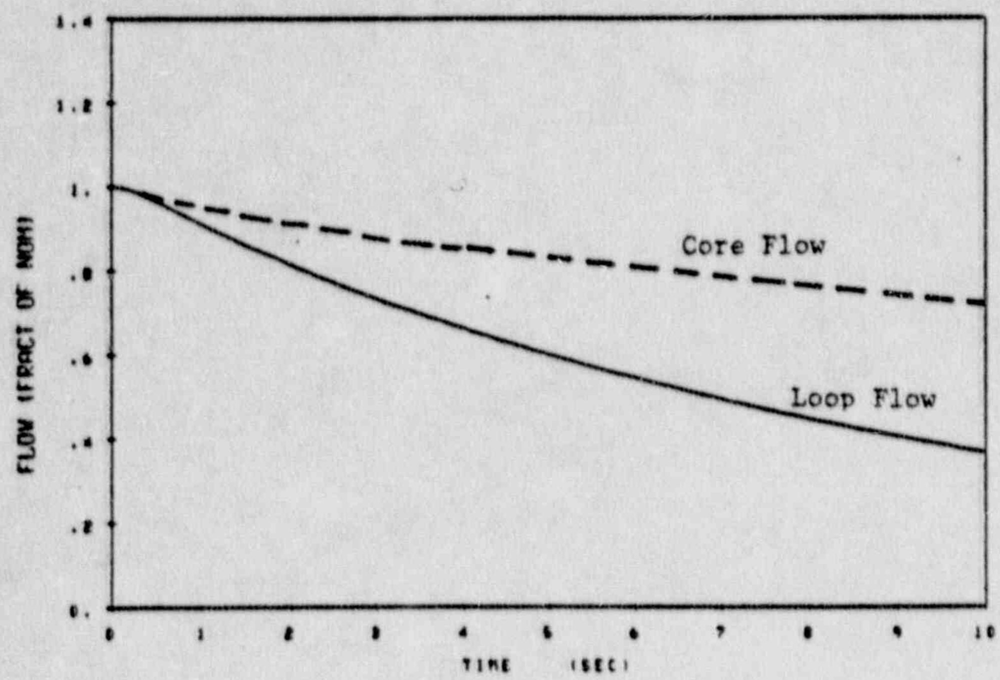


FIGURE 5.1-5

Partial Loss of Forced Reactor Coolant Flow
Heat Flux versus Time

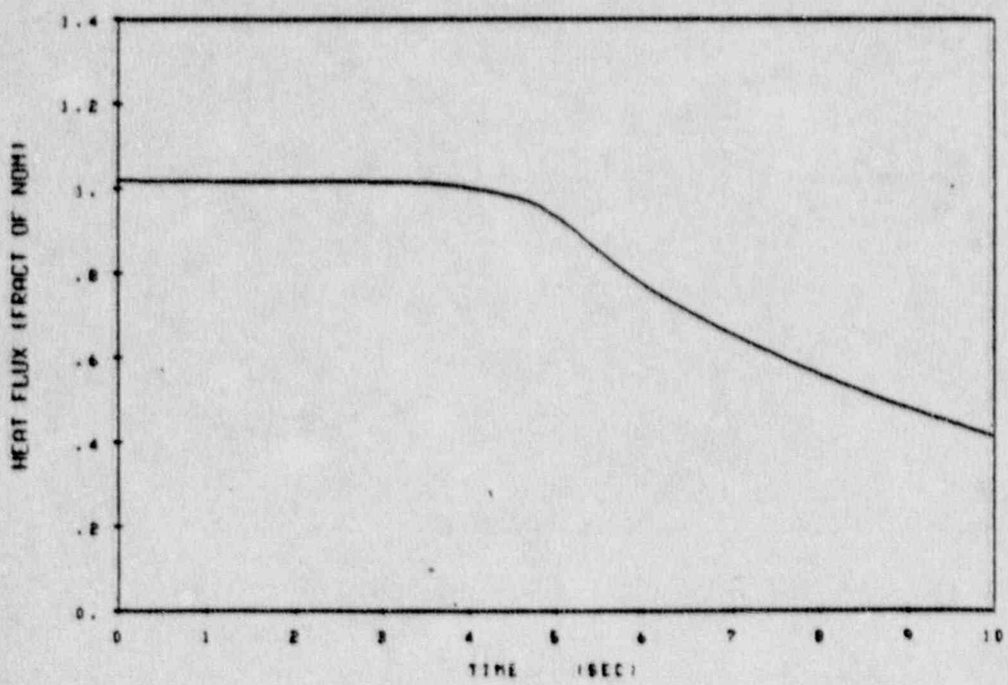


FIGURE 5.1-6

Partial Loss of Forced Reactor Coolant Flow
DNBR versus Time

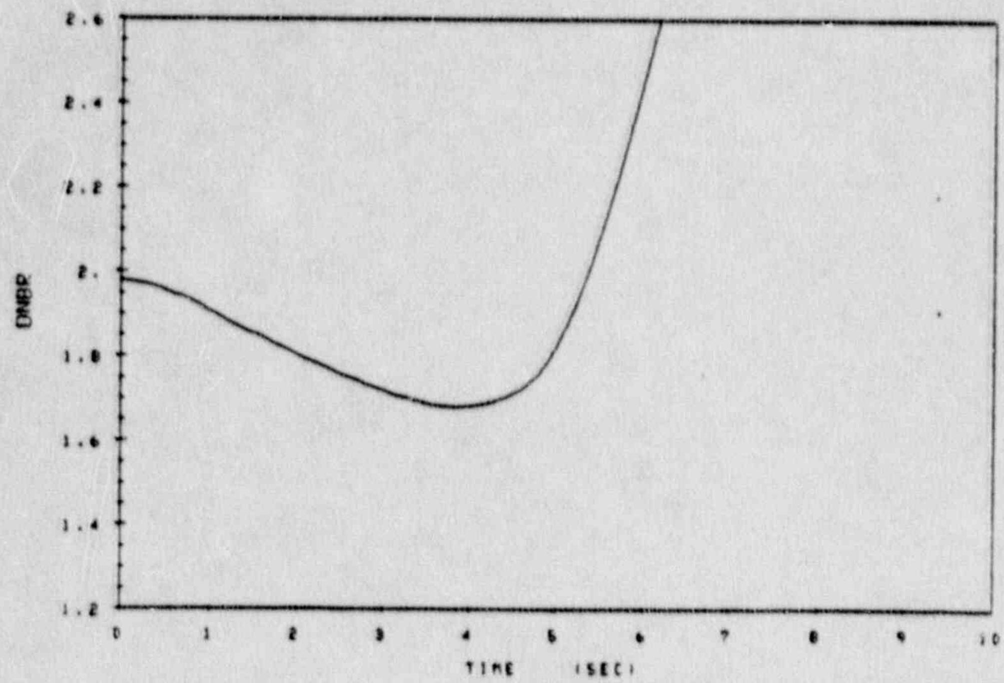


FIGURE 5.1-7

Complete Loss of Forced Reactor Coolant Flow
Core and Loop Flow versus Time

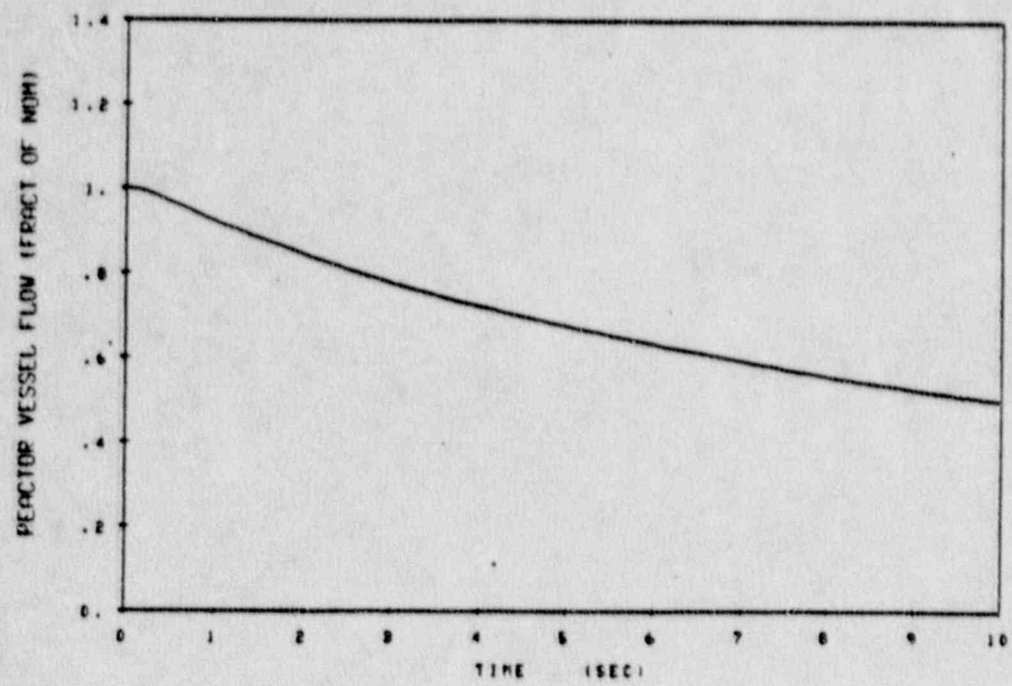


FIGURE 5.1-8

Complete Loss of Forced Reactor Coolant Flow
Heat Flux versus Time

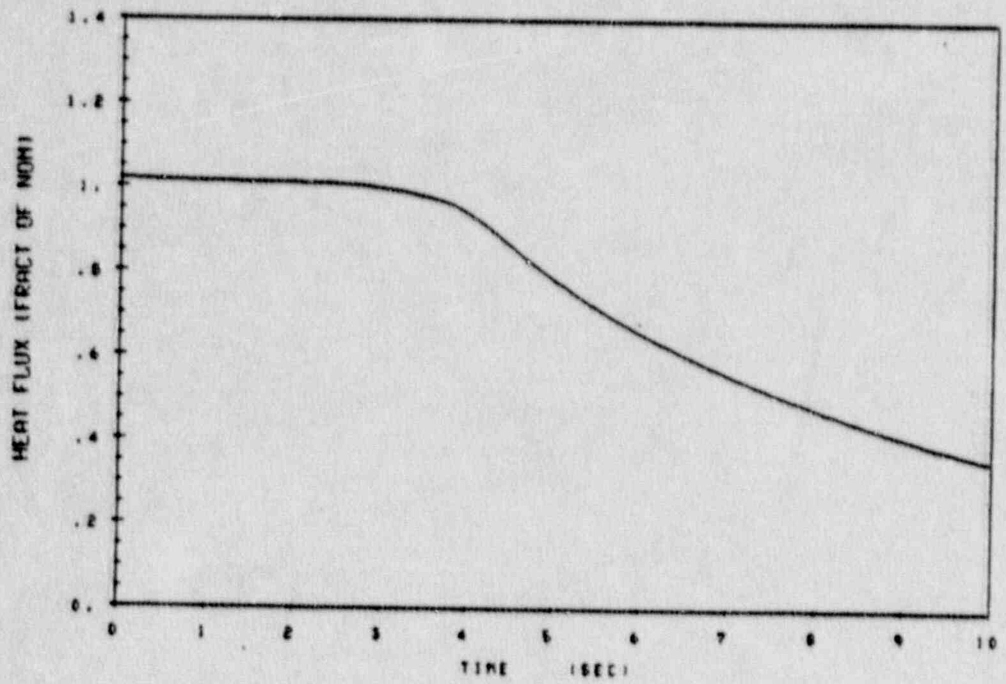


FIGURE 5.1-9

Complete Loss of Forced Reactor Coolant Flow
DNBR versus Time

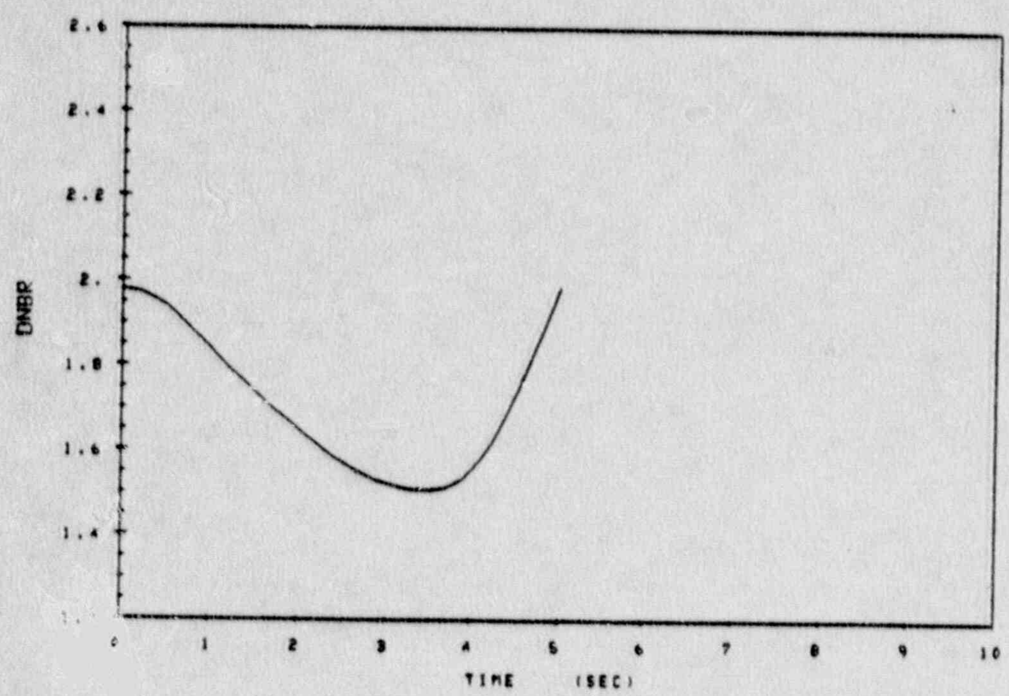


FIGURE 5.1-10

Startup of an Inactive Loop
Heat Flux versus Time

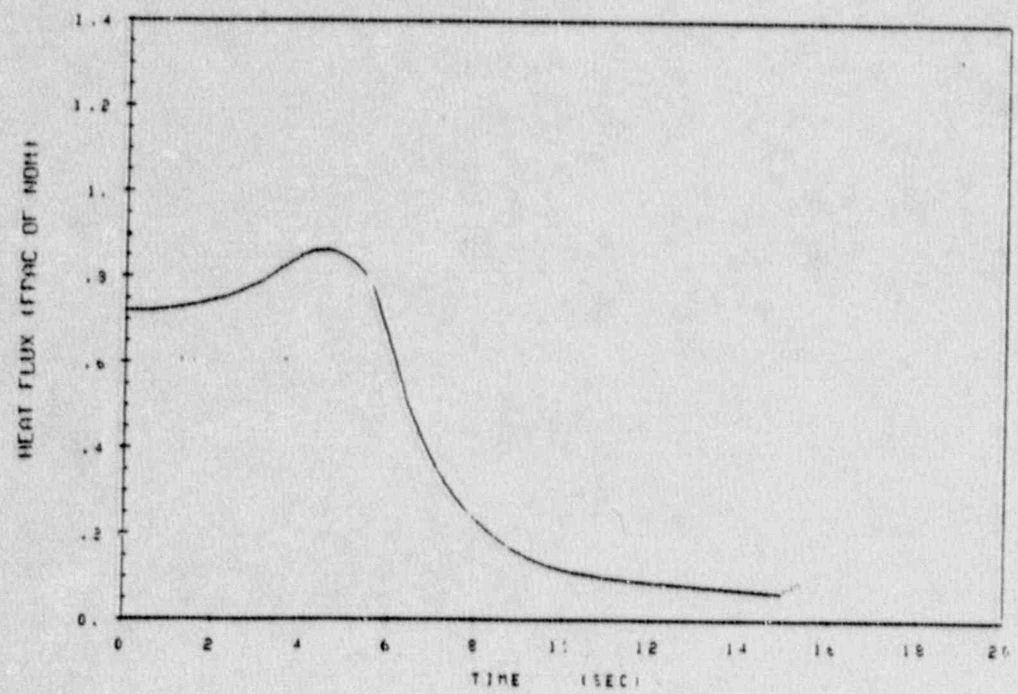


FIGURE 5.1-11

Startup of an Inactive Loop
DNBR versus Time

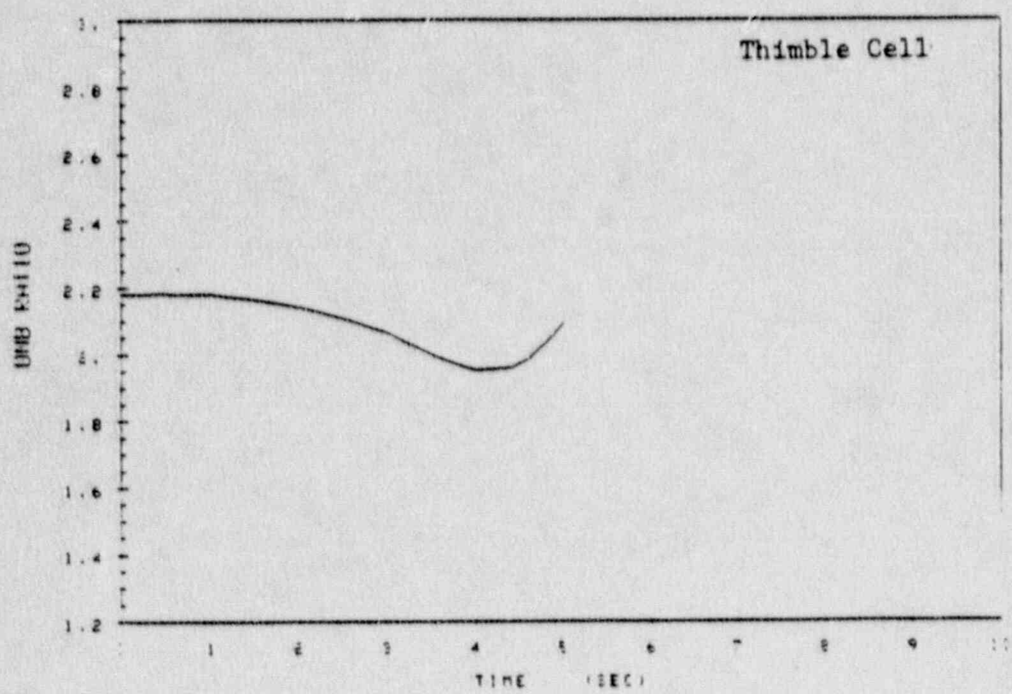


FIGURE 5.1-12

Single Reactor Coolant Pump Locked Rotor
Core Flow versus Time

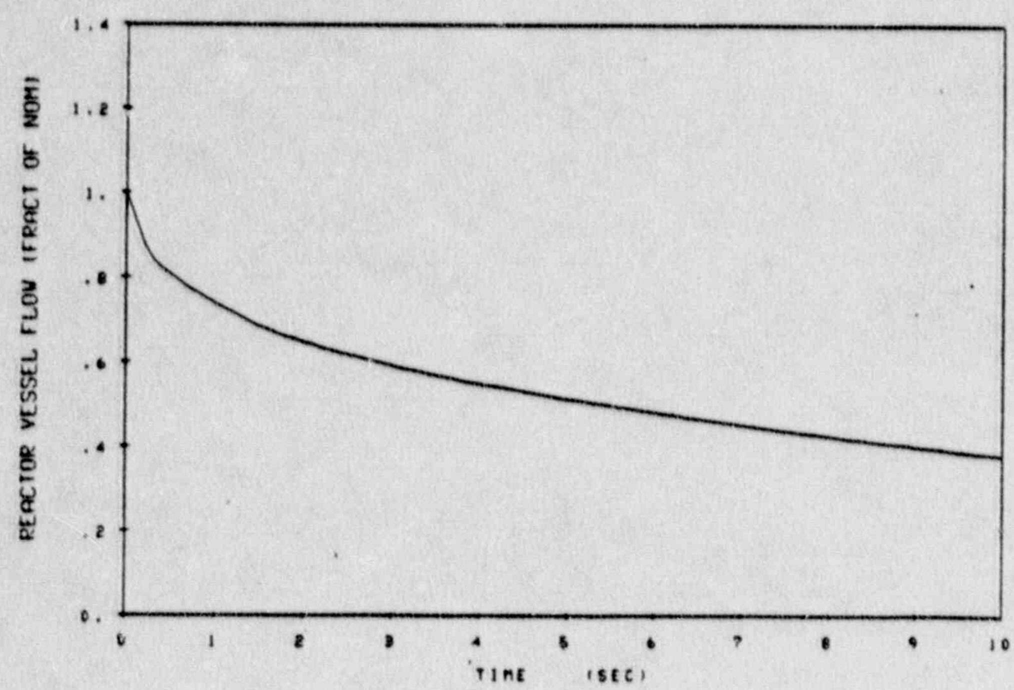


FIGURE 5.1-13

Single Reactor Coolant Pump Locked Rotor
Nuclear Power and Heat Flux versus Time

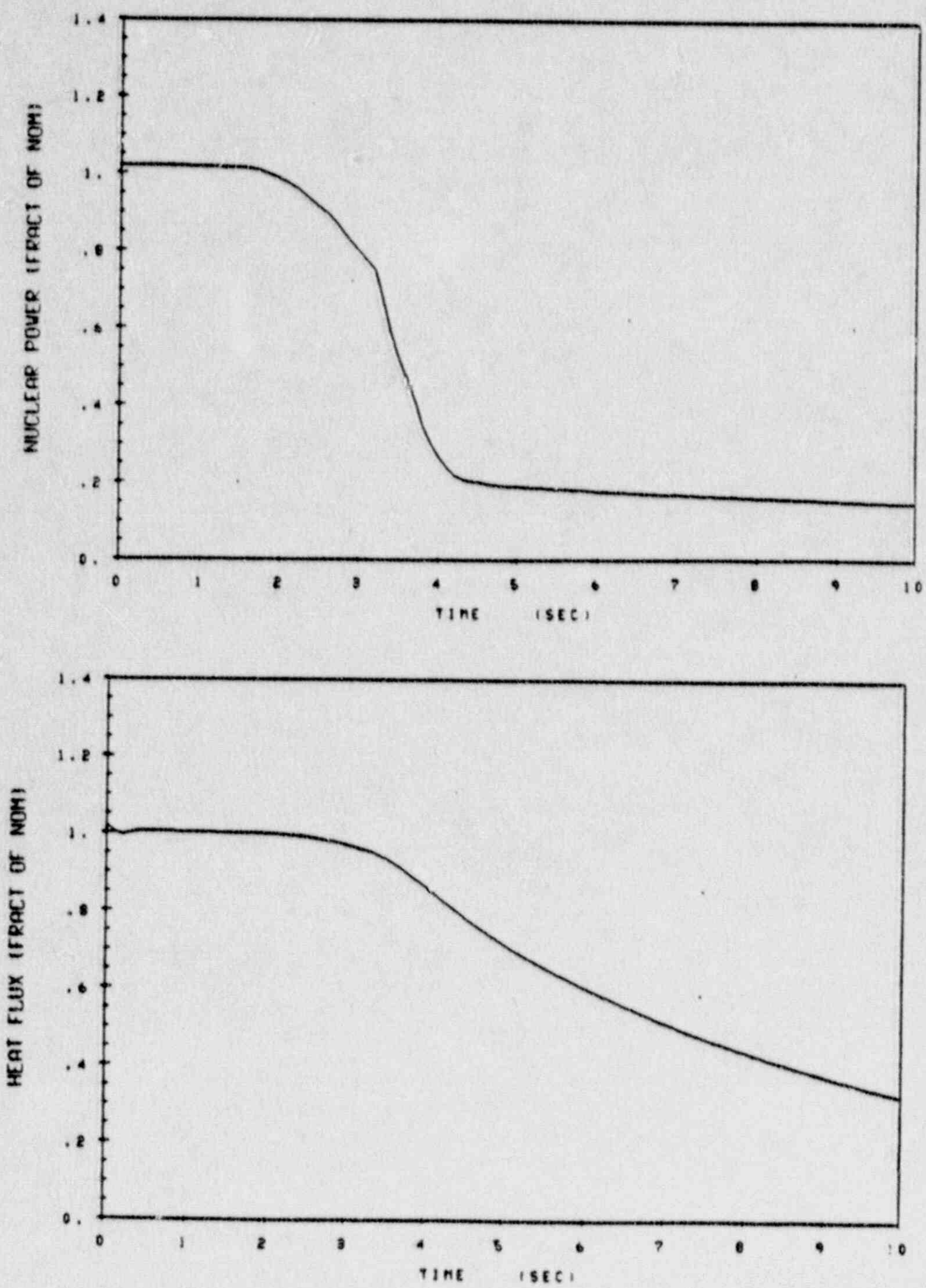
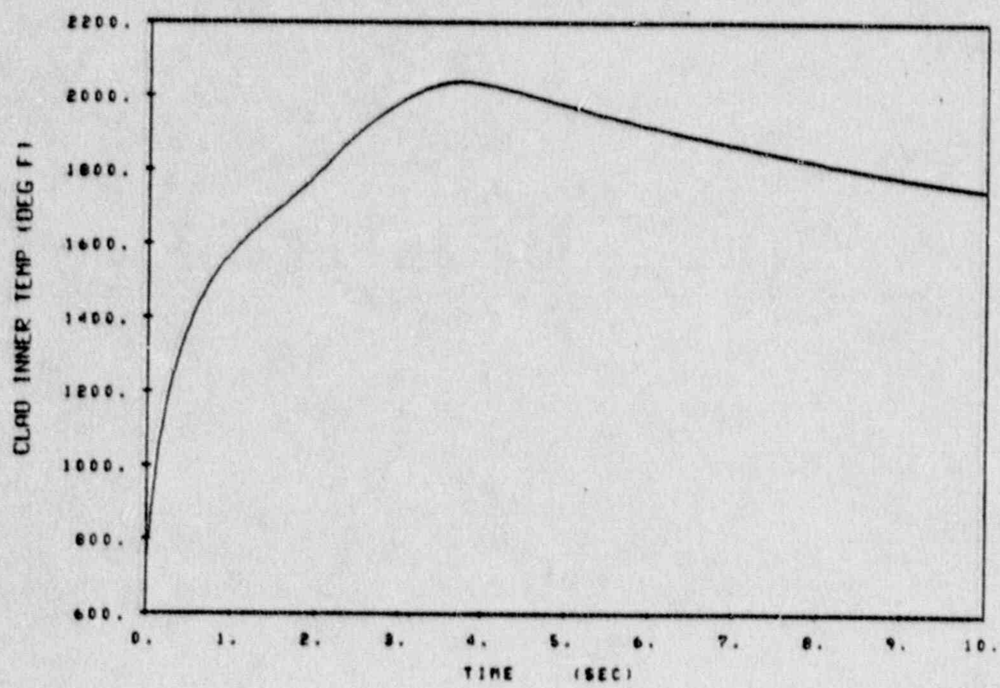


FIGURE 5.1-14

Single Reactor Coolant Pump Locked Rotor
Clad Temperature versus Time



5.2 LOCA Accidents

This section summarizes the evaluations performed to assess the effect of VANTAGE 5H fuel on the Sequoyah Units licensing bases LOCA analyses. As noted in Reference 1, the majority of the VANTAGE 5H fuel features have no adverse effect on the licensing bases LOCA analyses due to the mechanical and hydraulic similarity to 17x17 STD fuel. Reference 1 documents that transitioning from 17x17 STD fuel to 17x17 VANTAGE 5H fuel without intermediate flow mixers (IFMs) results in no transition core Peak Cladding Temperature (PCT) penalty. The only item which can potentially affect the LOCA analyses is the increase in RCCA rod drop time.

Large Break LOCA (FSAR Section 15.4.1)

The Large Break LOCA analysis for the Sequoyah Units has been executed using the Westinghouse 1981 Evaluation Model with BASH for 17x17 standard fuel. It resulted in a calculated PCT of 2001°F for the limiting Cd=0.6 DECLG break. This analysis considered the upper head injection (UHI) system to be removed from service.

An evaluation has been performed to consider the effects on the analysis of the VANTAGE 5H fuel. The large break LOCA evaluation model does not take credit for the negative reactivity introduced by the control rods. Instead, the reactor is brought to a subcritical condition by the presence of voids in the core caused by the rapid depressurization of the RCS. Since credit is not taken for the negative reactivity introduced by the control rods, the increase in rod drop time will have no effect on the current FSAR large break analysis. Furthermore, sensitivity studies have demonstrated that VANTAGE 5H fuel is less limiting than 17x17 STD fuel in analyses performed using the 1981 Evaluation Model with BASH.

Based on the discussion given above, the use of VANTAGE 5H will not result in an increase in the peak clad temperature for the Sequoyah Units. Therefore, these changes are acceptable and the resulting peak clad temperature remains within the regulatory limits.

Small Break LOCA (FSAR Section 15.3.1)

The small break LOCA licensing analysis for the Sequoyah Units for UHI removal has predicted a peak clad temperature of 2105.5°F using the NOTRUMP Westinghouse Small Break Evaluation Model. The only VANTAGE 5H feature which affects the Small Break LOCA analysis

is the increase in rod drop time. The Westinghouse small break model assumes that reactor core is brought to a subcritical condition by the negative reactivity of the control rods. The increase in the rod drop time to a maximum value of 2.7 seconds has been modeled in the UHI removal small break LOCA analysis. There is no increase in clad temperature due to the introduction of VANTAGE 5H fuel. The revised UHI removal licensing basis for the Sequoyah Units would therefore be applicable independent of the use of VANTAGE 5H fuel.

Steam Generator Tube Rupture (FSAR Section 15.4)

The steam generator tube rupture (SGTR) accident is analyzed to ensure that offsite doses remain below the limits defined in 10CFR100. The primary thermal-hydraulic factors affecting this conclusion are: the extent of fuel failure that occurs during the event (or whether DNB occurs), the primary to secondary flow through the ruptured tube, and the mass and energy releases to the atmosphere from the steam generator with the ruptured tube. The amount of fuel failure assumed for the SGTR analysis in the Sequoyah FSAR is 1%, which is assumed to be independent of the fuel type or the transient conditions. The primary to secondary break flow and the mass released to the atmosphere from the ruptured steam generator are dependent upon the RCS and secondary system operating parameters. The change from 17x17 STD fuel to VANTAGE 5H fuel does not affect these parameters and therefore has no effect on the SGTR analysis.

Blowdown Reactor Vessel and Loop Forces (FSAR Section 3.9.3.5)

The major factors in determining the resulting forces from a postulated LOCA on the vessel and the internals are the reactor coolant system primary fluid temperature and pressure. Since VANTAGE 5H does not change the primary side design temperatures and pressures which are modeled in the forces analysis, there will be no effect on the LOCA hydraulic forces.

Post LOCA Long-Term Core Cooling, Boron Evaluation (FSAR Section 15.4.1)

The Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long-Term Cooling' is defined in WCAP-8339. The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water residing in the RCS and sump after a LOCA. Since credit for the control rods is not taken for a large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor

core remaining subcritical assuming all control rods out. This will be demonstrated for each reload design with VANTAGE 5H fuel on a cycle specific basis.

Since the use of VANTAGE 5H fuel will not affect the sources of borated and non-borated water assumed in the long term cooling calculation, it is concluded that there would be no change to the long term cooling capability of the ECCS system. As noted above, this licensing commitment is checked by Westinghouse on a cycle by cycle basis ensuring compliance with this requirement independent of this safety evaluation.

Hot Leg Switchover to Prevent Potential Boron Precipitation (FSAR Sections 15.4.1 and 6.3)

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This recirculation time is dependent on power level, and the RCS, RWST, and accumulator water volumes and boron concentrations. The VANTAGE 5H fuel will have no effect on the assumptions for the RCS, RWST, and the accumulators in the hot leg switchover calculation. Thus, there is no effect on the post-LOCA hot leg switchover time.

Rod Ejection Mass and Energy Release for Dose Calculations (FSAR Section 15.5.7)

A review of the Sequoyah Updated FSAR Chapter 15.5.7 shows no LOCA analysis of the primary side mass and energy release due to a hypothesized rupture of a control Rod Drive Mechanism (CRDM). Instead, simplifying assumptions were made to determine the release of radioactive material from the primary side as a result of a CRDM rupture. Therefore, the use of VANTAGE 5H fuel at the Sequoyah Units will have an insignificant impact on the calculated consequences of a rod ejection accident and is acceptable.

5.3 Environmental Consequences of Accidents

This section summarizes the impact of the upgrade to VANTAGE 5H fuel for the Sequoyah Units on the radiological consequences of accidents. The change to VANTAGE 5H fuel affects the accident doses only insofar as the extent of fuel damage is increased and/or the coolant mass releases to the environment are increased.

Included in this evaluation is consideration of extended fuel burnup of 48,000 ± 500 MWD/MTU for the reload batch average discharge. This extended burnup has a peak fuel pin burnup of < 60,000 MWD/MTU. The impact of extended fuel burnup on core source terms used in accident analyses is addressed in References 3 and 11. Based on Reference 11, the extended fuel burnup would have no impact on the radiological consequences of any of the design basis accidents, except for the fuel handling accident, because the doses for these accidents are due to the release of short-lived isotopes of krypton, xenon, and iodine which, because of their short half-lives, do not increase with burnup. There is an increase in Kr-85 inventory associated with extended burnup but Kr-85 does not significantly impact radiological consequences.

The reactor coolant system source terms provided in FSAR Section 11.1 are not significantly affected by the implementation of extended fuel burnup. While a few long-lived isotopes would increase appreciably, most isotopes would remain virtually unchanged and others would be reduced. The operation of the reactor would remain limited by Technical Specification 3.4.8 which requires that the specific activity of the primary coolant be limited to less than or equal to $100/\bar{E}$ microcuries per gram.

Environmental Consequences of a Postulated Loss of A.C. Power to the Plant Auxiliaries (FSAR Section 15.5.1)

The VANTAGE 5H fuel features do not result in any fuel damage associated with this accident nor is there an increase in the mass releases from the secondary coolant system. As discussed above, extended fuel burnup does not increase the source terms associated with this accident. The conclusions of the FSAR remain valid.

Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture (FSAR Section 15.5.2)

The VANTAGE 5H fuel features do not affect this accident since it involves an auxiliary system that is separate from the operation of the reactor. As discussed above, extended fuel burnup does not increase the source terms associated with this accident. The conclusions of the FSAR remain valid.

Environmental Consequences of a Postulated Loss of Coolant Accident (FSAR Section 15.5.3)

The VANTAGE 5H fuel features do not result in any increase in the fuel damage associated with determining the radiological consequences of this accident since the level of core damage assumed is based on the guidance of Regulatory Guide 1.4. If a mechanistic determination of the fuel damage were to be used, with or without the implementation of the VANTAGE 5H fuel features, there would be a reduction in the level of fuel damage. Mass releases from the secondary coolant system are not considered for this accident. As discussed above, extended fuel burnup does not increase the source terms associated with this accident. The conclusions of the FSAR remain valid.

Environmental Consequences of a Postulated Steam Line Break (FSAR Section 15.5.4)

The VANTAGE 5H fuel features do not result in any fuel damage associated with this accident nor is there an increase in the mass releases from the secondary coolant system. As discussed above, extended fuel burnup does not increase the source terms associated with this accident. The conclusions of the FSAR remain valid.

Environmental Consequences of a Postulated Steam Generator Tube Rupture (FSAR Section 15.5.5)

The VANTAGE 5H fuel features do not result in any fuel damage associated with this accident nor is there an increase in the mass releases from the secondary coolant system. As discussed above, extended fuel burnup does not increase the source terms associated with this accident. The conclusions of the FSAR remain valid.

Environmental Consequences of a Postulated Fuel Handling Accident (FSAR Section 15.5.6)

The VANTAGE 5H fuel features do not affect this accident. However, based on Reference 11, the implementation of extended fuel burnup would result in an increase in the gap fraction assumed

for I-131 from the ten percent value specified in Regulatory Guide 1.25 to twelve percent. Since I-131 is the primary contributor to the thyroid dose, this would be expected to increase the thyroid dose by twenty percent.

While it is expected that analysis of the I-131 gap fraction for the fuel management scheme specific to the Sequoyah fuel would show that the peak rod gap fraction does not exceed ten percent, the fuel handling accident doses have been reanalyzed utilizing the increased gap fraction value. The analysis bases have also been revised to counteract the impact of the increased I-131 gap fraction. There are two changes. One involves the use of accident meteorology that is based on a larger data base than was used previously (and thus more accurately reflects the Sequoyah site). The other involves the use of Regulatory Guide 1.52 guidance for charcoal filter efficiency credit in place of the guidance provided in Regulatory Guide 1.25. The results of the reanalysis show that for the fuel handling accident occurring in the Auxiliary Building the thyroid doses reported in the FSAR of 71 rem at the Site Boundary (SB) and 8.3 rem for the Low Population Zone (LPZ) remain bounding. For the fuel handling accident located inside the containment the reanalysis shows that the SB thyroid dose of 118 rem reported in the FSAR remains bounding but that the LPZ thyroid dose of 13.8 rem is exceeded. The revised LPZ thyroid dose is less than 20 rem.

The conclusions of the FSAR remain valid (i.e., that the doses for the fuel handling accident remain below the limits of 10 CFR 100). Although the LPZ dose has increased somewhat for the case in which the fuel handling accident occurs inside the containment, that dose is less than the acceptance criterion of the present Standard Review Plan and is less than the reported doses for the SB.

Environmental Consequences of a Postulated Rod Ejection Accident (FSAR Section 15.5.7)

The VANTAGE 5H fuel features do not result in any increase in the fuel damage associated with this accident nor is there an increase in the mass releases from the secondary coolant system. As discussed above, extended fuel burnup does not increase the source terms associated with this accident. The conclusions of the FSAR remain valid.

6.0 REFERENCES

1. Davidson, S. L. ed. et al., "VANTAGE 5H Fuel Assembly," WCAP-10444-P-A, Addendum 2A, April 1988.

2. Davidson, S. L. ed. et al., "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A, September 1985.

3. Davidson, S. L. ed. et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

4. Davidson, S. L. ed. et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-A, July 1985.

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8. Skaritka, J., ed., "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (P), July 1979.

9. "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), NS-EPR-2572, March 16, 1982.

10. Letter from C. Berlinger (NRC) to E. P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.

11. D. A. Baker, W. J. Bailey, C. E. Beyer, F. C. Bold, and J. J. Tawil, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR-5009 (PNL-6258), February 1988.

12. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident of Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1A.

13. NS-NRC-89-3466, "Use of 2700°F PCT Acceptance Limit in non-LOCA Accidents," W. J. Johnson (Westinghouse) to Mr. Robert C. Jones (NRC), October 23, 1989.

APPENDIX A
Summary of Technical Specification Changes

SUMMARY OF TECHNICAL SPECIFICATION CHANGES
FOR SEQUOYAH UNIT 1

Page	Section	Description	
B 2-1 B 2-3 B 2-5 B 3/4 2-5	2.1.1 Basis 3/4.2.5 Basis	Change W-3 correlation to WRB-1 correlation and added safety analysis DNBR limit.	This change reflects the DNB correlation used in analyses.
3/4 1-19	3.1.3.4	Revised rod drop time to less than or equal to 2.7 seconds.	This change is a result of changes in the fuel due to the VANTAGE 5H fuel design. The effect of this increase on the safety analysis has been considered.
3/4 2-10 3/4 2-11 3/4 2-12 3/4 2-13 3/4 2-14 B 3/4 2-1 3/4 2-2 B 3/4 2-4	3/4.2.3 3/4.2.2 and 3/4.2.3 Basis	FΔH and Rod Bow Penalty (delete Rod Bow Penalty (RBP) as a function of burnup in FΔH equation and delete figure 3.2-3).	Use of the new rod bow penalty methodology reduces the rod bow penalty. The reduced penalty is accounted for in the analysis by using available DNBR margin. The new methodolgy is defined in the references below*.
3/4 2-18 3/4 2-19 B 3/4 2-5	3/4.2.5 3/4.2.5 Basis	DNB parameter	This change is a result of the revision of the FΔH Tech Specs with respect to flow.

* Skaritka, J., (Ed.) "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Prop), July 1979.

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Letter C. Berlinger (NRC) to E.P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1985.

Recommended Modifications to

the Technical Specifications for

Sequoyah Unit 1

2.1 SAFETY LIMITS

BASES

WRB-1 correlation and W-3 correlation for conditions outside the range of WRB-1

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.

Correlations have

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.

Insert 1

~~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 Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30 or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. the safety analysis DNBR limit R118

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: R118

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$ R118

where P is the fraction of RATED THERMAL POWER

INSERT 1

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).

SAFETY LIMITS

BASES

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

the safety analysis DNBR limit

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above ~~1.30~~ for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate

SAFETY LIMITS

BASES

analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 89% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 89% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below $\boxed{1.30}$ during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature Delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature Delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below $\boxed{1.30}$ during normal operational transients and anticipated transients with 3 loops in operation.

the safety analysis DNBR limit

ROD DROP TIME

2.7

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position# shall be less than or equal to $\boxed{2.2}$ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: R112

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

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

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

#Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of $\geq$ 222 and $\leq$ 231 steps withdrawn, inclusive. R112

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

~~3/4.2.3 RCS FLOWRATE AND R~~

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

~~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:~~

R23

$F^N_{\Delta H} \leq 1.55 [1.0 + 0.3(1.0-P)]$

Where:

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

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

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

~~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.~~

Insert 2

INSERT 2

With F^N_H exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% of greater RATED THERMAL POWER.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2 and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the regions of acceptable operation shown in Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

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

Insert 3

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the regions of acceptable operation of Figure 3.2-3:

Insert 3

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The measured $F^N_{\Delta H}$ shall be increased by 4% for measurement uncertainty.

SURVEILLANCE REQUIREMENTS (Continued)

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

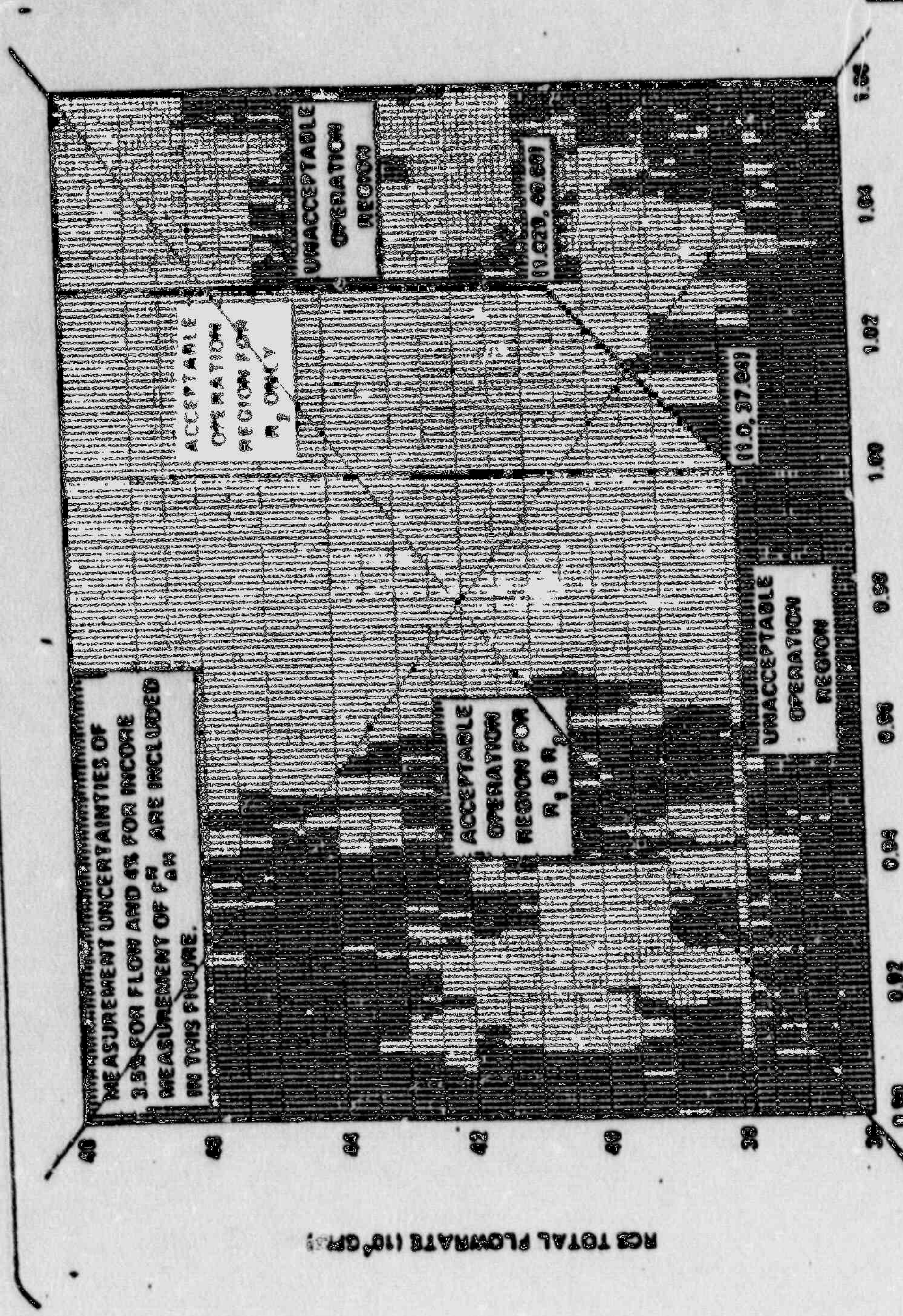
4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the values of R_1 and R_2, obtained per Specification 4.2.3.2, are assumed to exist.

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

4.2.3.5 The RCS flow rate shall be determined by measurement at least once per 18 months.

Delete

R23



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

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

Delete

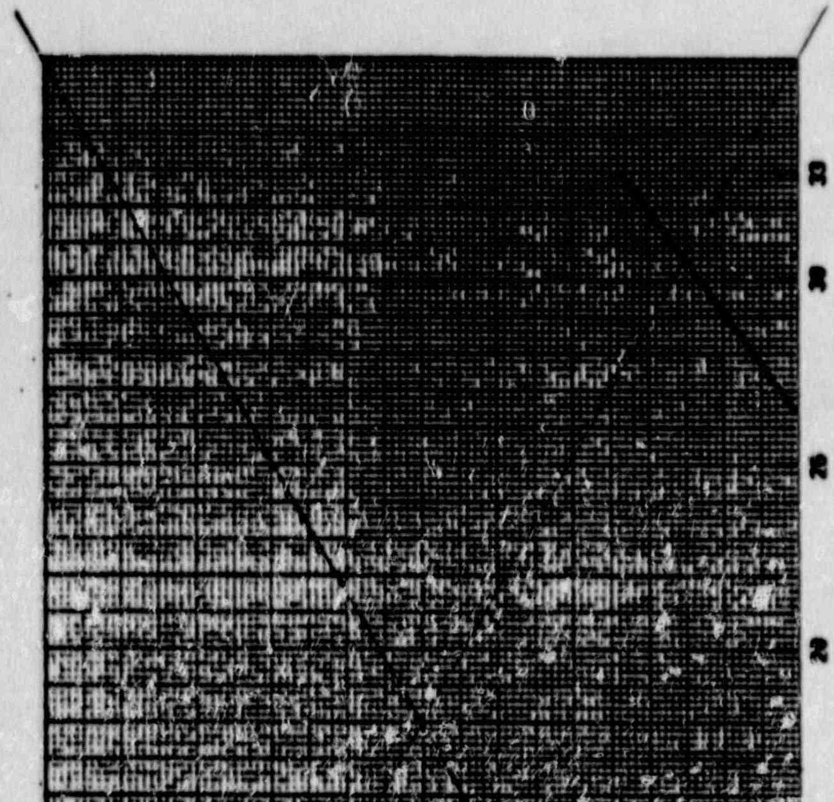


FIGURE 3.2-4 ROD BOW PENALTY VERSUS REGION AVERAGE BURNUP

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1

~~4.2.5~~ Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System	$\geq$ 378400 gpm #

R45

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

Includes a 3.5 % flow measurement uncertainty

R45

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL - FQ(Z) and $F_{\Delta H}^{N}$

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.

$F_{\Delta H}^{N}$ 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. R23

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.

hot channel factors are

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. R23

Insert 4

$F^N_{\Delta H}$ 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 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.

INSERT 4

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

POWER DISTRIBUTION LIMITS

BASES

~~The penalties applied to $F_{\Delta H}^{N}$ 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.~~ R23

Insert 5

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.

INSERT 5

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty.

The applicable value of rod bow penalty is referenced in the FSAR.

Margin in excess of the rod bow penalty is available for plant design flexibility.

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 [illegible] 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. R23

greater than or equal to the
safety analysis DNBR limit

SUMMARY OF TECHNICAL SPECIFICATION CHANGES

FOR SEQUOYAH UNIT 2

Page	Section	Description	
B 2-1 B 2-3 B 2-5 B 3/4 2-5	2.1.1 Basis 3/4.2.5 Basis	Change W-3 correlation to WRB-1 correlation and added safety analysis DNBR limit.	This change reflects the DNB correlation used in analyses.
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* Skaritka, J., (Ed.) "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Prop), July 1979.

Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E.P. Rahe, Jr. (Westinghouse) to J.R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E.P. Rahe, Jr. (Westinghouse) to J.R. Miller (NRC), NS-EPR-2572, dated March 16, 1982.

Letter C. Berlinger (NRC) to E.P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1985.

Recommended Modifications to

the Technical Specifications for

Sequoyah Unit 2

WRB-1 correlation and W-3 Correlation for conditions outside the range of WRB-1

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.

Correlations have

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.

Insert 1

~~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 Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than ~~1.30~~ or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. R104

the safety analysis DNBR limit

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: R21

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

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 Delta T 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. BR

INSERT 1

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

the safety analysis DNBR limit

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above $\boxed{1.30}$ for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate

LIMITING SAFETY SYSTEM SETTINGS

BASES

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 89% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 89% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below ~~1.30~~ during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below ~~1.30~~ during normal operational transients and anticipated transients with 3 loops in operation.

the safety analysis DNBR limit

ROD DROP TIME

2.7

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position# shall be less than or equal to $\boxed{2.2}$ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: R98

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: Modes 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.* R20

---

*For cycle 1, this surveillance is to be completed before the next cooldown or by August 5, 1983, whichever is earlier. R20

#Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of ≥222 and ≤231 steps withdrawn, inclusive. R98

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F_{\Delta H}^{N}$

POWER DISTRIBUTION LIMITS

~~3/4.2.3 RCS FLOWRATE AND R~~

3.2.3 $F_{\Delta H}^{N}$ shall be limited by the following relationship:

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:~~

R 21

$F_{\Delta H}^{N} \leq 1.55 [1.0 + 0.3(1.0-F$

Where:

~~a. $R_1 = \frac{F_{\Delta H}^{N}}{1.49 [1.0 + 0.3 (1.0 - P)]}$,~~

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

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

~~d. $F_{\Delta H}^{N}$ = Measured values of $F_{\Delta H}^{N}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F_{\Delta H}^{N}$ 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 $F_{\Delta H}^{N}$, 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.~~

Insert 2

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INSERT 2

With F^N_H exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% of greater RATED THERMAL POWER.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

Insert 3

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

Insert 3

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The measured $F^N_{\Delta H}$ shall be increased by 4% for measurement uncertainty.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained values of R_1 and R_2, obtained per Specification 4.2.3.2, are assumed to exist.

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

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

Delete

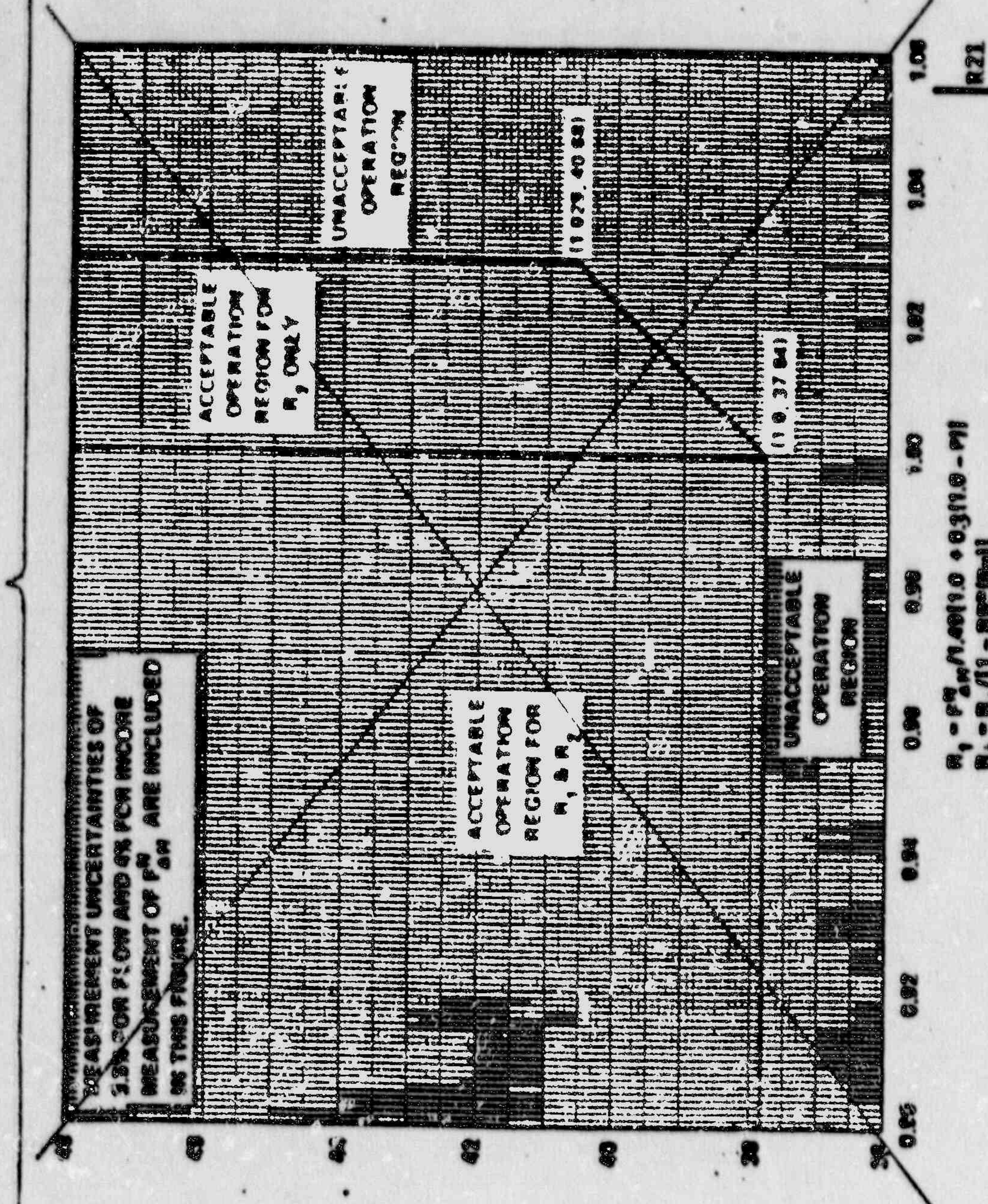

FIGURE 3.2-3 RCS Total Flowrate Versus R_1 and R_2 – Four Loops in Operation

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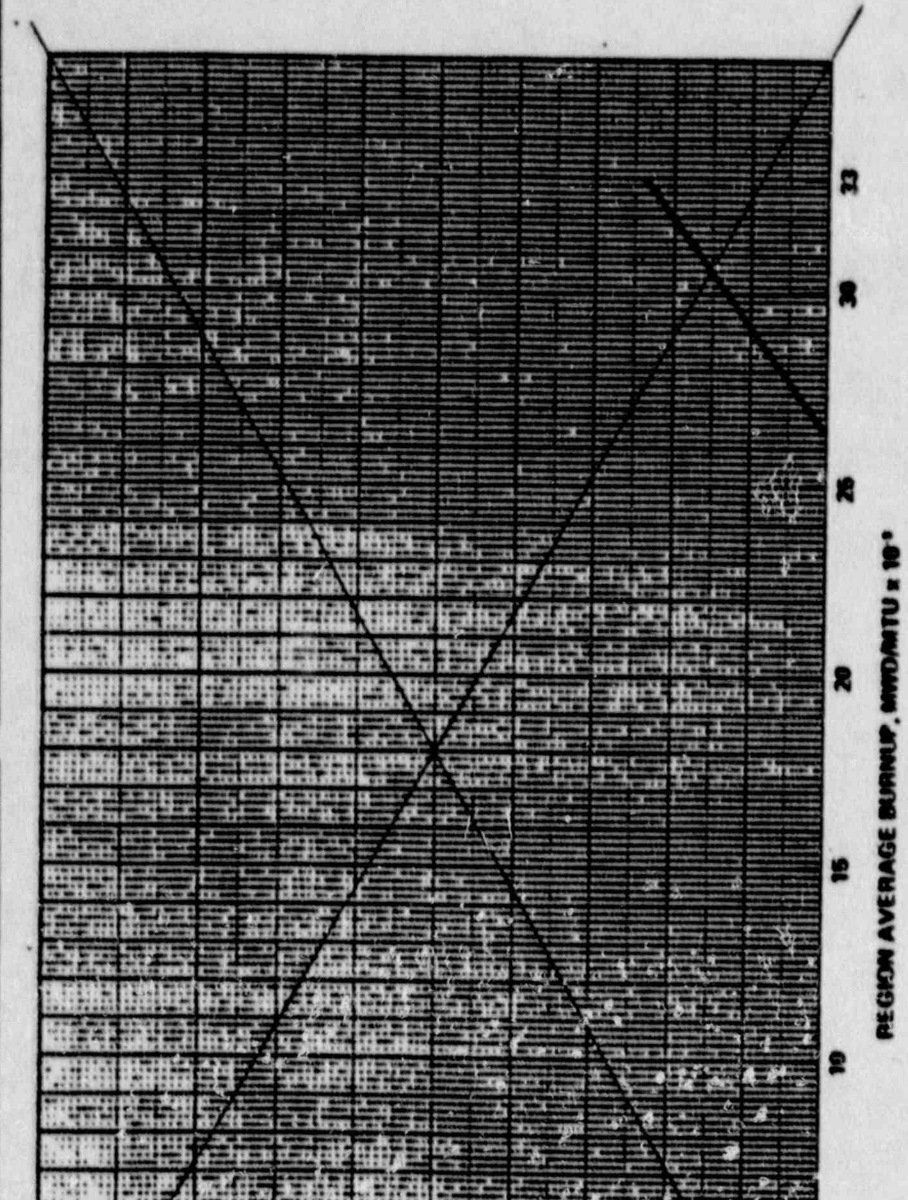

FIGURE 3.2-4 ROD BOW PENALTY VERSUS REGION AVERAGE BURNUP

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1

$\boxed{4.2.5}$ Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	≤ 583°F
Pressurizer Pressure	≥ 2220 psia*
Reactor Coolant System	≥ 378400 gpm #

R33

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

Includes a 3.5 % flow measurement uncertainty

R33

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL - FQ(Z) and $F^N_{\Delta H}$

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. R 2

$F^N_{\Delta H}$ 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.

hot channel factors are

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.

Insert 4

~~$F^N_{\Delta H}$ 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 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.~~

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INSERT 4

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

~~The penalties applied to $F_{\Delta H}^{N}$ 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.~~

Insert 5

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. R

3/4.2.4 QUADRANT POWER TI[illegible]IO

The quadrant powe[illegible] 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.

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INSERT 5

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty.

The applicable value of rod bow penalty is referenced in the FSAR.

Margin in excess of the rod bow penalty is available for plant design flexibility.

POWER DISTRIBUTION LIMITS

BASES

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. R

greater than or equal to the
safety analysis DNBR limit

APPENDIX B

Significant Hazards Evaluation

SIGNIFICANT HAZARDS EVALUATION

A. Background

The Tennessee Valley Authority (TVA) plans to refuel and operate the Sequoyah Nuclear Plants with Westinghouse VANTAGE 5 Hybrid (V5H) advanced fuel product features that incorporate low pressure drop zircaloy grids and Removable Top Nozzles, Integral Fuel Burnable Absorbers and extended burnup capability. This upgraded fuel will also contain Debris Filter Bottom Nozzles, snag resistant grids, and standardized pellets. These features have been implemented in other Westinghouse reload cores.

In addition, the Plant Safety Evaluation for the Sequoyah Fuel Upgrade implements the current rod bow methodology to reduce the rod bow penalty described in the Sequoyah Technical Specifications.

Finally, the evaluations performed for this fuel upgrade also accommodate effects from the following programs:

1. RTD Bypass Elimination
2. Eagle 21 Digital Protection System
3. Upper Head Injection (UHI) Removal
4. Boron Injection Tan (BIT) Removal
5. New Steamline Break Protection
6. Low Feedwater Flow Reactor Trip Elimination

B. Proposed Changes

As a result of the fuel upgrade, the following changes are proposed for the Sequoyah Unit 1 and Unit 2 Technical Specifications:

1. Modify the Bases for Safety Limits to change the W-3 correlation to the WRB-1 correlation and to revise the associated design Departure from Nucleate Boiling Ratio (DNBR) limits.

2. Modify Specification 3.1.3.4 to incorporate a new rod drop time of less than or equal to 2.7 seconds.

3. Modify Specification 3.2.3 to delete the Rod Bow Penalty as a function of burnup in the $F_{\Delta H}$ equation and delete Figure 3.2-3.

4. Modify Table 3.2-1 of Specification 3.2.5 to define the DNB-related Reactor Coolant System (RCS) Total Flow Rate limit, including uncertainties, to be 378,400 gpm.

C. Reasons for Proposed Changes

Changes 1 and 2 are required to allow implementation of the improved fuel design for Westinghouse V5H fuel.

Change 3 is required to incorporate new evaluation methodologies for the effects of fuel rod bow on DNB. The new methodologies provide a basis to eliminate unnecessary power distribution penalties and to simplify the specification.

Change 4 is required to relocate the RCS Total Flow Rate requirement from Specification 3.2.3 to 3.2.5, as a result of Change 3, and to clearly define the DNB flow parameter limit. This limit includes flow measurement uncertainties.

Overall, the proposed changes for this License Amendment Request (LAR) are the result of three primary differences:

1. Increased rod drop time due to the reduced guide tube diameter for the V5H zircaloy grids

2. The use of a new DNB correlation

3. Incorporation of the current methodology to assess the rod bow penalty

D. Justification for Proposed Changes

As discussed in the safety evaluation for the fuel upgrade, the previously reviewed and licensed Safety Limits for Sequoyah are met with the upgraded fuel. The new fuel design has provided satisfactory operational performance in fuel assembly demonstration programs since the early 1980's. The V5H fuel is both mechanically and hydraulically compatible with the current Sequoyah fuel assemblies, control rods, and reactor internals interfaces.

The V5H fuel satisfies the current design bases for Sequoyah and it meets design requirements for hydraulic stability and structural integrity under seismic/LOCA loads, with margins comparable to 17x17 STD fuel assemblies. Nuclear characteristics are comparable within the range normally seen from cycle to cycle due to fuel management effects.

No change in fuel rod design criteria, methods, or model are necessary with transition to V5H, with the exception of a new DNB correlation. Based upon the information provided in the evaluation, the Sequoyah plant operational limits will be satisfied with the proposed changes.

The evaluation considered the effects of the proposed Technical Specification changes on the following areas:

a. Mechanical, Nuclear, and Thermal-Hydraulic Fuel Assembly Design

b. Non-LOCA Accidents

c. LOCA Accidents

d. Environmental Consequences of Accidents

These areas have been evaluated for the impact of all proposed changes in this LAR, including the transition core effects (with a mixed core fuel loading with both V5H and 17x17 STD fuel). The required analyses as described in the fuel upgrade evaluation were performed by Westinghouse using methods and procedures previously approved by the NRC.

1. DNB Correlation Change (Change 1)

The calculational methods currently used for 17x17 STD fuel assemblies and described in the Sequoyah FSAR are applicable to V5H fuel assemblies, except for the DNB correlation. The new correlation basis for DNB performance is the WRB-1 correlation.

The WRB-1 correlation establishes a DNB limit which provides for the margin of safety required by the current FSAR (i.e., DNB will not occur on at least 95 percent of the limiting fuel rods during normal and operational transients and any transient condition arising from faults of moderate frequency at a 95 percent confidence level). The WRB-1 correlation takes credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations.

2. Increased Rod Drop Time (Change 2)

The V5H fuel design incorporates a snag resistant, low pressure drop, zircaloy grid, The Zircaloy grid will provide for an enhanced performance relative to the current Westinghouse 17x17 STD fuel product.

Utilization of zircaloy as a grid material instead of inconel reduces the source of cobalt in the core. Consequently, radiation fields due to the transport of activated cobalt should be lower. The snag resistant feature results from outer grid straps which are modified to reduce the potential for grid damage and assembly hang-up from assembly interactions during fuel assembly removal. The zircaloy grid also contains features that minimize hydraulic resistance.

In order to maintain mechanical compatibility between the V5H grid and guide tube, a reduction in the V5H guide tube diameter was required. The allowable rod drop time of Specification 3.1.3.4 must be increased due to the increased dashpot effect resulting from the guide tube diameter reduction.

3. Elimination of Rod Bow Penalty (Change 3)

Fuel rod bow has been observed in Westinghouse cores. The phenomena of fuel rod bowing must be accounted for in the DNB safety analyses of Condition I and II events. The current licensing basis offsets the DNB effects of rod bow by partially accommodating it with margin in the W-3 correlation. The remainder of the rod bow penalty is applied as a penalty on the $F\Delta H$ Technical Specification.

New statistical methods have been developed by Westinghouse which verify that the past treatment of rod bow penalty provided an overestimation of the effects on DNB. Application of the new methods to Sequoyah for the Standard and the V5H fuel products has verified the reduction in rod bow penalty. The reduction allows for accommodation of the entire penalty in the establishment of the Safety Limit DNBR.

The requested change to Specification 3.2.3 will remove an unnecessary power peaking penalty and simplify the format of the Specification. The current specification format includes reactor coolant system flow not only to maintain the minimum required RCS flow but to provide for an additional offset against rod bow penalty. The use of reactor coolant system flow to compensate for rod bow penalty will no longer be required.

The flow limit (and its associated uncertainty factor) as it relates to DNB will be moved to Specification 3.2.5, DNB parameters (see the discussion for Change 4). Similar Technical Specifications changes related to rod bow penalty have been performed for Farley 1/2, North Anna 1/2, Beaver Valley 1, and Salem 2.

4. Definition of DNB Parameter RCS Flow Limit (Change 4)

The RCS flow limit and its associated uncertainty factor have been moved to Specification 3.2.5, DNB Parameters, which now established a minimum allowable reactor coolant system (RCS) flow to prevent violation of the Safety Limit DNB during normal operation and accident conditions.

The minimum flow rate is based on a thermal design flow rate of 365,600 gpm plus the application of a correction for measurement uncertainties (minimum flow rate = 365,600 gpm x uncertainty factor). The uncertainty factor of 3.5% is based on flow measurement uncertainties and feedwater venturi fouling. Therefore, the RCS flow limit for the Sequoyah units is 365,600 gpm x 1.035, or 378,400 gpm.

E. Determination of Significant Hazards

Pursuant to 10CFR50.91, TVA has determined that operation of the facility in accordance with the proposed LAR does not involve any significant hazards considerations as defined by NRC regulations in 10CFR50.92. The following discussion describes how the proposed amendment satisfies each of the three standards of 10CFR50.92(c).

Operation of the Sequoyah Units in accordance with the proposed Technical Specification changes:

a. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The evaluations of the mechanical, nuclear, and thermal-hydraulic design effects support the conclusion that the requested changes are within the current design criteria established in the FSAR. Consequently, no new mechanisms have been introduced to increase the probability of a previously analyzed accident occurring. The accident evaluations (both LOCA and non-LOCA) exhibit results which maintain the confidence level in the physical integrity of the fission product boundaries as defined in the FSAR. Therefore, the consequences of the accidents do not increase.

b. Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The evaluations performed establish that the FSAR design criteria and system responses during normal and accident conditions are bounding with respect to the proposed changes. The changes will not affect the function of any protection system and they will not introduce hardware which is different in design criteria requirements. Therefore, no new mechanisms have been introduced that would create the possibility of a new or different kind of accident from those previously analyzed.

c. Does not involve a significant reduction in the margin of safety.

   The evaluations performed by Westinghouse addressed all design criteria and accident analyses. In performing the evaluations, the Safety Limits established by the FSAR and Technical Specifications were not modified such as to reduce the difference between the Safety Limit and the limit defined as the failure point of a fission product boundary. Therefore, the margins which were assumed in the accident analyses remain bounding for the proposed changes.

F. Conclusion

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists. This guidance (51 FR 7750) includes examples of the types of amendments that are considered not likely to involve significant hazards considerations.

Based on the evaluation summarized above, TVA has concluded that the proposed Technical Specification changes correspond to the examples in 51 FR 7750 for Amendments Considered Not Likely to Involve Significant Hazards Consideration. Additionally, the proposed changes are consistent with the requirements of 10CFR50.36 and 10CFR50.59.

APPENDIX C

Nuclear Safety Evaluation Checklist

Westinghouse Reference No(s). ______________________

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) SEQUOYAH UNITS 1 & 2

2) CHECK LIST APPLICABLE TO: V5H FUEL UPGRADE

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 (b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10CFR50.59 (a) (1)

(3.1) Yes_X_ No___ A change to the plant as described in the FSAR?
(3.2) Yes___ No_X_ A change to procedures as described in the FSAR?
(3.3) Yes___ No_X_ A test or experiment not described in the FSAR?
(3.4) Yes_X_ No___ A change to the plant technical specifications? (See Note on Page 2)

4) CHECK LIST - PART B - 10CFR50.59 (a) (2) (Justification for Part B answers must be included on page 2.)

(4.1) Yes___ No_X_ Will the probability of an accident previously evaluated in the FSAR be increased?
(4.2) Yes___ No_X_ Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes___ No_X_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes___ No_X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.5) Yes___ No_X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.6) Yes___ No_X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(4.7) Yes___ No_X_ Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59 (c) and submitted to the NRC pursuant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification based upon the written safety evaluation (1) for answers given in Part A (3.4) and Part B of this Safety Evaluation Check List:

SEE OTHER SECTIONS OF THIS PLANT SAFETY EVALUATION REPORT

(1) Reference to document(s) containing written safety evaluation: ______

FOR FSAR UPDATE

Section: ______ Pages: ______ Tables: ______ Figures: ______

FSAR MARK-UPS: CHAPERS 4 & 15 (TO BE FORWARDED UNDER SEPARATE COVER)

TECHNICAL SPECIFICATION MARK-UPS: SEE APPENDIX A OF THIS REPORT

...ription of Change:

... OF THIS PLANT SAFETY EVALUATION REPORT

...Tomasic ______ Date: 11.20.89

...masic ______ Date: ______

...on File ______ Date: ______

______ Date: 11/20/...

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59 (c) and submitted to the NRC pursant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification based upon the written safety evaluation (1) for answers given in Part A (3.4) and Part B of this Safety Evaluation Check List:

SEE OTHER SECTIONS OF THIS PLANT SAFETY EVALUATION REPORT

(1) Reference to document(s) containing written safety evaluation:

FOR FSAR UPDATE

Section:________ Pages:_________ Tables:___________ Figures:___________

FSAR MARK-UPS: CHAPERS 4 & 15 (TO BE FORWARDED UNDER SEPARATE COVER)

TECHNICAL SPECIFICATION MARK-UPS: SEE APPENDIX A OF THIS REPORT

Reason for / Description of Change:

SEE OTHER SECTIONS OF THIS PLANT SAFETY EVALUATION REPORT

Prepared by (Nuclear Safety): L.V.Tomasic Date: 11.20.89

Coordinated With Engineer(s): Signatures On File Date:_________

Coordinating Group Manager(s): Signatures On File Date:_________

Nuclear Safety Group Manager: S.D.Rupprecht Date: 11/24/89