E R R A T A
Issued 05/05/89

Houston Lighting & Power correspondence ST-HL-AE-3021, dated March 30, 1989 was inadvertently distributed with typographical errors. Pages affected by this change have been annotated "Errata, 05/05/89". Please replace your copy with the attached revised ST-HL-AE-3021.

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ATTACHMENT

Annotated Revisions
to
FSAR and Technical Specification bases

TABLE 4.1-1

REACTOR DESIGN COMPARISON TABLE

THERMAL AND HYDRAULIC DESIGN PARAMETERS	W. B. McGuire UNITS 1 & 2	South Texas Project UNITS 1 & 2	
1. Reactor Core Heat Output, MW_t	3,411	3,800	
2. Reactor Core Heat Output, 10^6 BTU/hr	11,641	12,969	
3. Heated Generated in Fuel, %	97.4	97.4	
4. System Pressure, Nominal, psia	2,250	2,250	
5. System Pressure, Minimum Steady State, psia	2,220	2,220	
6. Minimum Departure from Nucleate Boiling Ratio for Design Transients	>1.30	~~>1.30~~ 1.17	18
7. DNB Correlation	"R" (W-3 with Modified Spacer Factor)	~~"R" (W-3 with Modified Spacer Factor)~~ WRB-1	
Peaking Factors to Prevent DNB			
7.1 Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$	1.55	1.52	54
7.2 Axial Power Shape (Chopped Cosine-Peak to Average)			
a) At normal full power operation	1.55	1.55	
b) At overpower conditions	1.55	1.61	
COOLANT FLOW			
8. Total Thermal Flow Rate, 10^6 lb_m/hr	140.3	141.3	54
9. Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	134.0	135.0	18
10. Effective Flow Area for Heat Transfer, ft^2	51.1	51.1	
11. Average Velocity Along Fuel Rods, ft/sec	16.7	~~16.7~~ 16.9	
12. Average Mass Velocity, 10^6 lb_m/hr-ft^2	2.62	2.64	54 18

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reactivity compensation. The core is also designed to have an overall negative moderator temperature coefficient of reactivity so that average coolant temperature or void content provides another, slower compensatory effect. Nominal power operation is permitted only in a range of overall negative moderator temperature coefficient. The negative moderator temperature coefficient can be achieved through use of fixed burnable poison and/or control rods by limiting the reactivity held down by soluble boron.

Burnable poison content (quantity and distribution) is not stated as a design basis other than as it relates to accomplishment of a non-positive moderator temperature coefficient at power operating conditions discussed above.

4.3.1.3 Control of Power Distribution.

Basis

The nuclear design basis is that, with at least a 95 percent confidence level:

1. The fuel will not be operated at greater than 13.3 kW/ft under normal operating conditions including an allowance of 2 percent for calorimetric error and not including power spike factor due to densification.

2. Under abnormal conditions including the maximum overpower condition, the fuel peak power will not cause melting as defined in Section 4.4.1.2.

3. The fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis (i.e., the DNBR shall not be less than ~~1.30,~~ as discussed in Section 4.4.1) under Condition I and II events including the maximum overpower condition. 53
the design limit DNBR,

4. Fuel management will be such as to produce rod powers and burnups consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

The above basis meets GDC 10.

Discussion

Calculation of extreme power shapes which affect fuel design limits is performed with proven methods and verified frequently with measurements from operating reactors. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state.

Even though there is good agreement between measured peak power calculations and measurements, a nuclear uncertainty margin (Section 4.3.2.2.7) is applied to calculated peak local power. Such a margin is provided both for the analysis for normal operating states and for anticipated transients.

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Bases

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the Reactor Coolant System (RCS) or the Emergency Core Cooling System (ECCS) (when applicable) assures that the following performances and safety criteria requirements are met:

1. Fuel damage (defined as penetration of the fission product barrier, i.e. the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant cleanup system and are consistent with the plant design bases.

2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (see above definition) although sufficient fuel damage might occur to preclude immediate resumption of operation.

3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy the above criteria, the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling Design Basis.

Basis

There will be at least a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events), at a 95 percent confidence level. ~~Historically this has been conservatively met by limiting the minimum departure from nucleate boiling ratio (DNBR) to 1.30, and for this application a minimum DNBR of 1.30 will continue to be used.~~ Insert A

Discussion

By preventing DNB, adequate heat transfer is assured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis as it

Insert B

Insert A

This criterion has been conservatively met by adhering to the following thermal design basis: there must be at least a 95% probability that the minimum departure from nucleate boiling ratio (DNBR) of the limiting power rod during Conditions I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95% probability with 95% confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.

Insert B

Historically, the DNBR limit has been 1.30 for Westinghouse applications. In this application, the WRB-1 correlation (Reference A) is used. With the significant improvement in the accuracy of the critical heat flux prediction by using this correlation instead of the previous correlation, a DNBR ~~correlation~~ design limit of 1.17 is applicable during steady state, normal operational transients and anticipated transients.

A plant-specific margin has been considered in the analyses. A safety analysis DNBR limit of 1.27 was used in the safety analyses. The plant allowance available between the DNBR limit used in the safety analyses and the design limit DNBR (7.8%) will be used to offset the effects of the RCS flow anomaly and fuel rod bowing on DNBR and to provide for flexibility in the design, operation and analyses for the South Texas plants.

For conditions outside the range of parameters for the WRB-1 correlation (refer to Section 4.4.2.2.1), the W-3 correlation is used and a DNBR design limit of 1.30 applies for pressures equal to or greater than 1000 psia. For low pressure (500-1000 psia) applications of the W-3 correlation, a design limit DNBR of 1.45 applies (Ref. C ~~below~~).

(Note to HL&P - "Reference C" appears on this page)

4.4.2.2 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology. The minimum DNBR's for the rated power, design overpower and anticipated transient conditions are given in Table 4.4-1. The minimum DNBR in the limiting flow channel will be downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

DNBR's are calculated by using the correlation and definitions described in the following Subsections 4.4.2.2.1 and 4.4.2.2.2. The THINC-IV computer code [4.3-18, 4.3-49] is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Subsection 4.4.4.3.1 (nuclear hot channel factors) and in Subsection 4.4.2.2.4 (engineering hot channel factors).

4.4.2.2.1 DNB Technology: Early experimental studies of DNB were conducted with fluid flowing inside single heated tubes or channels and with single annulus configurations with one or both walls heated. The results of the experiments were analyzed using many different physical models for describing the DNB phenomenon, but all resultant correlations are highly empirical in nature. The evolution of these correlations is given by Tong [4.4-2, 4.4-3], including the W-3 correlation which is in wide use in the pressurized water reactor (PWR) industry.

As testing methods progressed to the use of rod bundles, instead of single channels, it became apparent that the bundle average flow conditions cannot be used in DNB correlations. As outlined by Tong [4.4-4] test results showed that correlations based on average conditions were not accurate predictors of DNB heat flux. This indicated that a knowledge of the local subchannel conditions within the bundle is necessary.

It is also noted that in this power capability evaluation, there has not been any change in the design basis. The reactor is designed to a minimum DNBR greater than or equal to the design limit DNBR as well as no fuel centerline melting during normal operation, operational transients and faults of moderate frequency. 18

~~All DNB analyses performed for this application have included a DNBR multiplier of .88 in accordance with the results of 17 x 17 geometry DNB tests [4.4-5].~~ delete

Fuel densification has been considered in the DNB and fuel temperature evaluations utilizing the methods and models described in detail in Reference 4.4-6.

In order to determine the local subchannel conditions, the THINC [4.4-7] computer code was developed. In the THINC Code, a rod bundle is considered to be an array of subchannels each of which includes the flow area formed by four adjacent rods. The subchannels are also divided into axial steps such that each may be treated as a control volume. By solving simultaneously the mass, energy, and momentum equations, the local fluid conditions in each control volume are calculated. The W-3 correlation, developed from single channel [illegible] applied to rod bundles by using the subchannel local fluid conditions calculated by the THINC code.

Insert D

It was shown by Tong [4.4-4] that the above approach yielded conservative predictions particularly in rod bundles with mixing vane grid spacers. Hence a correction factor was developed to adapt the W-3 correlation, (which was developed based on single channel data), to rod bundles with spacer grids. This correction factor, termed the "modified spacer factor", was developed as a multiplier on the W-3 correlation.

The modified spacer factor was developed from rod bundle DNB test results conducted in the Westinghouse high pressure water loop at Columbia University. These tests were conducted on non-uniform axial heat flux test sections to determine the DNB performance of a low parasitic, top-split mixing vane grid design, hereafter referred to as the "R" grid. A description of this test program and a summary of the results are given below. The grid to be used in the 17 x 17 fuel assembly will be similar in design to the "R" grid.

"R" grid rod bundle DNB tests, References [4.4-8] and [4.4-9], were conducted over a wide range of simulated reactor conditions applicable to this plant. These conditions were:

Axial grid spacing	20 in. 26 in. and 32 in.
Local DNB quality	-15 percent to +15 percent
Local mass velocity	1.6×10^6 to 3.7×10^6 $lb_m/hr\text{-}ft^2$
Local inlet temperature	440°F to 620°F
Pressure	1490 to 2440 psia
Local heat flux	0.3×10^6 to 1.1×10^6 $BTU/hr\text{-}ft^2$
Axial heat flux distribution	Non-uniform (Cos u and u Sin u)
Heated length	8 ft. and 14 ft.
Heater rod O.D.	0.422 in.

The experimental program consisted of a DNB test series for both an all and/or partial channel surface heated condition in a 16 rod bundle arranged in a 4 x 4 array. A radial power profile was simulated by operating the central 4 rods of the bundle at 15 percent higher power than the other rods. Two test series were conducted on 26 in. axial grid spacing: 1) all channel surface heated condition (typical cell), 2) partial channel surface heated condition (thimble cold wall cell).

For the thimble cold wall test series, one of the central four heater rods was replaced by an unheated rod. The simulated unheated thimble was made up of a thin steel rod over which were placed ceramic cylinders with an outer diameter equal to the thimble outer diameter. These thimbles are attached to the grid in the same manner as in the reactor core using a sleeve which is brazed into the grid and then bulged out above and below the grid to connect to the thimble.

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The WRB-1 (Reference A) correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data with better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of variables is:

Pressure	:	$1440 \leq P \leq 2490$ psia
Local Mass Velocity	:	$0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft^2-hr
Local Quality	:	$-0.2 \leq X_{loc} \leq 0.3$
Heated Length, Inlet to CHF Location	:	$L_h \leq 14$ feet
Grid Spacing	:	$13 \leq g_{sp} \leq 32$ inches
Equivalent Hydraulic Diameter	:	$0.37 \leq d_e \leq 0.60$ inches
Equivalent Heated Hydraulic Diameter	:	$0.46 \leq d_h \leq 0.58$ inches

Figure 4.4-1 shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

The WRB-1 correlation is applicable for the South Texas 17X17 fuel assembly with "R" mixing vane grid design.

(NOTE TO HLEP - "Reference A" appears on this page)

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These rod bundle DNB data have been analyzed and a modified spacer factor, References 4.4-8 and 4.4-9, has been developed to conservatively incorporate the "R" mixing vane grid benefit for both typical and cold wall cells. This modified spacer factor is:

$$F'_S = \left(\frac{P}{225.895}\right)^{0.5} (1.445 - 0.0371\ L)\ [e^{(x+0.2)^2} - 0.73]$$

$$+ K_S \frac{G}{10^6} \left(\frac{TDC}{0.019}\right)^{0.35} \qquad (4.4\text{-}1)$$

where:

- P = the primary system pressure, psia
- L = the total heated core length, ft.
- x = the local quality expressed in fractional form
- G = the local mass velocity, $lb_m/hr\text{-}ft^2$
- TDC = the thermal diffusion coefficient
- K_S = the axial grid spacing coefficient which has the following values:

Grid Spacing, in.	K_S
32	0.027
26	0.046
20	0.066

Figure 4.4-1 shows all the "R" grid typical cell data. Figure 4.4-2 shows all the "R" grid thimble cell data. The predicted heat flux in Figures 4.4-1 and 4.4-2 incorporates the modified spacer factor per Equation (4.4-2) for typical cells and Equation (4.4-3) for thimble cold wall cells:

$$q''_{PRED} = q''_{DNB,N\ W\text{-}3} \times F'_S \qquad (4.4\text{-}2)$$

where:

$q''_{DNB,N\ W\text{-}3}$ is the predicted non-uniform DNB heat flux using the W-3 correlation as described in Reference 4.4-10.

$$q''_{PRED} = q''_{DNB,N,CW\ W\text{-}3} \times F'_S \qquad (4.4\text{-}3)$$

where:

$q''_{DNB,N,CW\ W\text{-}3}$ is the predicted non-uniform DNB heat flux for a cell having a cold (unheated) wall evaluated with the W-3 cold wall correlation described in Subsection 4.4.2.2.2 and Reference 4.4-3.

DELETE

F'_c as defined in Equation (4.4-1) is the same in both Equations (4.4-2) and (4.4-3) for both typical and thimble cold wall cells.

Effect of 17 x 17 Geometry on DNB

A test program similar to the one described above was conducted at the Westinghouse pressure water loop at Columbia University. In this test program, DNB data was obtained for 17 x 17 fuel assembly geometry, Reference 4.4-5, in a 5 x 5 rod bundle array.

Test results were obtained for typical cells (all walls heated) in 8 ft and 14 ft bundles with uniform axial heat flux and for 14 ft typical and thimble cold wall cells with non-uniform axial heat flux. All bundles were for mixing vane spacings of 22 in. (greater spacing than this design). 18

The data obtained were analyzed with the existing "R" grid DNB correlation described above to determine the effect on DNB of the 17 x 17 fuel assembly geometry. Plots of ratio of measured to predicted DNB heat flux versus flow parameters show that the "R" grid correlation properly accounts for local fluid parameters. However, the "R" grid correlation consistently overpredicts the DNB heat flux. Hence, a multiplier of .88 on the modified spacer factor F'_s is required to correctly predict the magnitude of the DNB heat flux for 17 x 17 geometry.

Figure 4.4-3 shows the 17 x 17 data obtained in this test program. The predicted heat flux includes the 0.88 multiplier on the modified spacer factor, Equation (4.4-1), as noted above.

As stated in Subsection 4.4.1.1 Westinghouse has chosen the design criterion that DNB will not occur at a 95 percent probability with a 95 percent confidence level.

In order to meet this criterion, a limiting value of DNBR is determined by the method of Owen [4.4-11]. Owen has prepared tables which give values of K such that "at least a proportion P of the population is greater than $\bar{X}/F-Ks$ with confidence γ," where $\bar{X}/F$ and s are the sample mean and standard deviation, respectively. When this method was carried out using the data on Figure 4.4-3, the results indicated that a reactor core using ~~this geometry~~ these fuel geometries may operate with a minimum DNBR of ~~1.00~~ 1.17 and satisfy the design criterion. ~~However, as stated in Section 4.4.1.1, a minimum DNBR of 1.00 is conservatively used.~~ 18

4.4.2.2.2 Definition of Departure from Nucleate Boiling Ratio (DNBR): The DNB heat flux ratio (DNBR) as applied to this design when all flow cell walls are heated is:

$$DNBR = \frac{q''_{DNB,\,N}}{q''_{loc}} \qquad (4.4\text{-}4)$$

delete

where:

Insert { $q''_{DNB,\,N}$ is the heat flux predicted by the applicable DNB correlation

For the W-3 correlation,

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F} \qquad (4.4\text{-}5)$$

and $q''_{DNB,EU}$ is the uniform DNB heat flux as predicted by the W-3 DNB correlation, Reference 4.4-10 all flow cell walls are heated.

F is the flux shape factor to account for nonuniform axial heat flux distributions, Reference 4.4-10, with the "C" term modified as in Reference 4.4-3.

~~F'_s is the modified spacer factor defined by Equation (4.4-1) in Section 4.4.2.2.1 and using an axial grid spacing coefficient, K_s = 0.059, and a thermal diffusion coefficient (TDC) of 0.059, based on the 22 in. grid spacing data previously described. Since the actual grid spacing is 19.8 in., the modified spacer factor is conservative since the DNB performance was found to improve and TDC increases as axial grid spacing is decreased, References 4.4-8 and 4.4-12. The TDC value for 20 in. grid spacing (approximately the same spacing as this design) is 0.061.~~ 18

Delete

q''_{loc} is the actual local heat flux.

The DNB heat flux ratio as applied to this design when a cold wall is present is:

$$DNBR = \frac{q''_{DNB,N,CW} \times F'_s}{q''_{loc}} \qquad (4.4\text{-}6)$$

Delete

where:

$$q''_{DNB,N,CW} = \frac{q''_{DNB,EU,Dh} \times CWF}{F} \qquad (4.4\text{-}7)$$

where:

$q''_{DNB,EU,Dh}$ is the uniform DNB heat flux as predicted by the W-3 cold wall DNB correlation, Reference 4.4-3, when not all flow cell walls are heated (thimble cold wall cell).

$$CWF\ [4.4\text{-}3] = 1.0 - Ru\ [13.76 - 1.372e^{1.78v} + 4.732 \left(\frac{G}{10^6}\right)^{0.0535} \qquad (4.4\text{-}8)$$

$$- 0.0619 \left(\frac{P}{1000}\right)^{0.14} - 8.509 Dh^{0.107}]$$

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and Ru = 1 - De/Dh

~~F'_s defined by Equation (4.4-1) in Section 4.4.2.2.1 is the same as used for typical cell.~~

~~Values of minimum DNB provided in Table 4.4-1 are the limiting values obtained by applying the above two definitions of DNBR to the appropriate cell (typical cell with all walls heated, or a thimble cold wall cell with a partial heated wall condition).~~ 36 33

The procedures used in the evaluation of DNB margin for this application show that the calculated minimum DNBR for the peak rod or rods in the core will be above ~~1.30~~ during Class I and II incidents, even when all the engineering hot 19

the design limit DNBR

Insert E

For the WRB-1 correlation,

$$Q''_{DNB,\ N} = \frac{Q''_{WRB-1}}{F} \qquad (4.4\text{-}8a)$$

where F is the same flux shape factor that is used with the W-3 correlation.

Insert F

The safety analysis for South Texas cores maintained sufficient margin between the safety analysis DNBR limit and the design DNBR limit to accomodate full and low flow DNBR penalties identified in Reference 4.4-85 with the incorporation of the L^2/I scaling factor (I = fuel rod bending moment of inertia, L = span length) to account for 17x17 XL span lengths.

1. Pellet diameter, density and enrichment |18

Design values employed in the THINC analysis related to the above fabrication variations are based on applicable limiting tolerances such that these design values are met for 95 percent of the limiting channels at a 95 percent confidence level. Measured manufacturing data on Westinghouse 17 x 17 fuel show the tolerances used in this evaluation are conservative. The |18
effect of variations in pellet diameter, enrichment and density is employed in the THINC analysis as a direct multiplier on the hot channel enthalpy |18
rise.

2. Inlet Flow Maldistribution

The consideration of inlet flow maldistribution in core thermal performances is discussed in Section 4.4.4.2.2. A design basis of 5 percent reduction in coolant flow to the hot assembly is used in the THINC-IV analysis.

3. Flow Redistribution

The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the non-uniform power distribution is inherently considered in the THINC analysis for every operating condition which is evaluated.

4. Flow Mixing

The subchannel mixing model incorporated in the THINC Code and used in reactor design is based on experimental data [4.4-17] discussed in Section 4.4.4.5.1. The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

4.4.2.2.5 Effects of Rod Bow on DNBR: The phenomenon of fuel rod bowing, as described in Reference 4.4-84, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained from measured plant operating parameters (such as $F_{\Delta H}$ or core flow), which are less limiting than those required by the plant safety analysis, can be used to offset the effect of rod bow.

Replace with Insert F

~~The safety analysis for South Texas cores maintained sufficient margin (3.3 percent) to accommodate full and low flow DNBR penalties identified in Reference 4.4-85 with the incorporation of the L^2/I scaling factor (I = fuel rod bending moment of inertia, L = span length) to account for 17 X 17 XL span lengths. A design limit DNBR of 1.30 vs. 1.28, a grid spacing coefficient (K_s) of .059 vs. .066, and a thermal diffusion coefficient (TDC) of .059 vs. .061 (used for modified spacer factor F'_s only) are examples of conservatism utilized in the safety analysis.~~ 43 15

The maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 33,000 MWd/MTU. At burnups

greater than 33,000 MWd/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required (Reference B). |

4.4.2.3 Linear Heat Generation Rate. The core average and maximum Linear Powers are given in Table 4.4-1. The method of determining the maximum Linear Powers is given in section 4.3.2.2.

4.4.2.4 Void Fraction Distribution. The calculated core average and the hot subchannel maximum and average void fractions are presented in Table 4.4-3 for operation at full power with design hot channel factors. The void fraction distribution in the core at various radial and axial locations is presented in Reference 4.4-18. The void models used in the THINC-IV computer code are described in Section 4.4.2.7.3. Normalized core flow and enthalpy rise distributions are shown on Figures 4.4-5 through 4.4-7.

4.4.2.5 Core Coolant Flow Distribution. Assembly average coolant mass velocity and enthalpy at various radial and axial core locations are given below. Coolant enthalpy rise and flow distributions are shown for the 1/3 core height elevation on Figure 4.4-5, and 2/3 core height elevation on Figure 4.4-6 and at the core exit on Figure 4.4-7. These distributions are for the full power conditions as given in Table 4.4-1 and for the radial power density distribution shown on Figure 4.3-7. The THINC Code analysis for this case utilized a uniform core inlet enthalpy and inlet flow distribution. No orificing is employed in the reactor design.

4.4.2.6 Core Pressure Drops and Hydraulic Loads.

4.4.2.6.1 Core Pressure Drops: The analytical model and experimental data used to calculate the pressure drops shown in Table 4.4-1 are described in Section 4.4.2.7. The core pressure drop includes the fuel assembly, lower core plate, and upper core plate pressure drops. The full power operation pressure drop values shown in Table 4.4-1 are the unrecoverable pressure drops across the vessel, including the inlet and outlet nozzles, and across the core. These pressure drops are based on the best estimate flow for actual plant operating conditions as described in Section 5.1.1. This section also defines and describes the thermal design flow (minimum flow) which is the basis for reactor core thermal performance and the mechanical design flow (maximum flow) which is used in the mechanical design of the reactor vessel internals and fuel assemblies. Since the best estimate flow is that flow which is most likely to exist in an operating plant, the calculated core pressure drops in Table 4.4-1 are based on this best estimate flow rather than the thermal design flow. |53

Uncertainties associated with the core pressure drop values are discussed in Section 4.4.2.9.2.

4.4.2.6.2 Hydraulic Loads: The fuel assembly hold down springs, Figure 4.2-2, are designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events with the exception of the turbine overspeed transient associated with a loss of external load. The hold down springs are designed to tolerate the possibility of an over deflection associated with fuel assembly liftoff for this case and provide contact between the fuel assembly and the lower core plate following this |18

Tests of the primary coolant loop flow rates will be made (see Subsection 4.4.5.1) prior to initial criticality to verify that the flow rates used in the design, which were determined in part from the pressure losses calculated by the method described here, are conservative.

4.4.2.7.3 Void Fraction Correlation: There are three separate void regions considered in flow boiling in a PWR as illustrated on Figure 4.4-8. They are the wall void region (no bubble detachment), the subcooled boiling region (bubble detachment) and the bulk boiling region.

In the wall void region, the point where local boiling begins is determined when the clad temperature reaches the amount of superheat predicted by Thom's [4.4-22] correlation (discussed in Subsection 4.4.2.7.1). The void fraction in this region is calculated using Maurer's [4.4-27] relationship. The bubble detachment point, where the superheated bubbles break away from the wall, is determined by using Griffith's [4.4-28] relationship.

The void fraction in the subcooled boiling region (that is, after the detachment point) is calculated from the Bowring [4.4-29] correlation. This correlation predicts the void fraction from the detachment point to the bulk boiling region.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality.

4.4.2.8 Thermal Effects of Operational Transients. DNB core safety limits are generated as a function of coolant temperature, pressure, core power and axial power imbalance. Steady-state operation within these safety limits insures that the minimum DNBR is not less than [redacted]. Figure 15.0-1 shows [redacted] limit lines and the resulting Overtemperature ΔT trip lines (which become part of the Technical Specifications), plotted as ΔT, versus T_{avg} for various pressures. This system provides adequate protection against anticipated operational transients that are slow with respect to fluid transport delays in the primary system. In addition, for fast transients (e.g., uncontrolled rod bank withdrawal at power incident [Section 15.4.2]) specific protection functions are provided as described in Section 7.2 and the use of these protection functions are described in Chapter 15.

4.4.2.9 Uncertainties in Estimates.

4.4.2.9.1 Uncertainties in Fuel and Clad Temperature: As discussed in Subsection 4.4.2.11, the fuel temperature is a function of crud, oxide, clad, gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to the inpile thermocouple measurements, References 4.4-30 through 4.4-36, by out-of-pile measurements of the fuel and clad properties, Reference 4.4-37 through 4.4-48 and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is presented in Reference 4.4-6.

flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the THINC-IV program. Inspection of the DNB correlation (Subsection 4.4.2.2 and Reference 4.4-10 shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The THINC-IV Code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on subchannel basis, regardless of where the flow blockage occurs. In Reference 4.4-49 it is shown that for a fuel assembly similar to the design, THINC-IV accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 in. downstream of the blockage. With the reactor operating at the nominal full power conditions specified in Table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching a minimum DNBR [redacted] less than the design limit DNBR. 18

From a review of the open literature it is concluded that flow blockage in "open lattice cores" similar to the cores cause flow perturbations which are local to the blockage. For instance, Ohtsubol [4.4-81] et al., show that the mean bundle velocity is approached asymptotically about 4 in. downstream from a flow blockage in a single flow cell. Similar results were also found for 2 and 3 cells completely blocked. Basmer [4.4-82], et al., tested an open lattice fuel assembly in which 41 percent of the subchannels were completely blocked in the center of the test bundle between spacer grids. Their results show the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about 5 in. for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or, in essence, the complete recovery length. Thus, local flow blockages within a fuel assembly have little effect on subchannel enthalpy rise. The reduction in local mass velocity is then the main parameter which affects the DNBR. If the plants were operating at full power and nominal steady state conditions as specified in Table 4.4-1, a reduction in local mass velocity greater than 58 percent would be required to reduce the DNBR [redacted] to the design limit DNBR. The above mass velocity effect on the DNB correlation was based on the assumption of fully developed flow along the full channel length. In reality a local flow blockage is expected to promote turbulence and thus would likely not effect DNBR at all. 18

Coolant flow blockages induce local crossflows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high cross-flow component. Fuel rod vibration could occur, caused by this crossflow component, through vortex shedding or turbulent mechanisms. If the crossflow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The crossflow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the crossflow (jet flow or not). These limits are greater than those for vibratory fuel rod wear. Crossflow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow induced vibration is considered in the fuel rod fretting evaluation (Section 4.2).

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Add

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A. Motley, F. E., Hill, K. W., Cadek, F. F. and Shefchek, J., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," ~~WCAP-8762, July, 1976~~ (Proprietary) and WCAP-8763, July, 1976 (Non-Proprietary). WCAP-8762-P-A, July 1984

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

Reference C

Letter from A. C. Thandani (NRC) to W. J. Johnson (Westinghouse), January 31, 1984, Subject: Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P/9227-NP, "Reactor Core Response to Excessive Secondary Steam Releases"

(NOTE TO HLP: Reference A appears in Insert D of the revised text.

Reference B appears on page 4.4-12 of revised text

Reference C appears in Insert B of revised text

These references & associated text should be appropriately numbered in the text and reference sections of the FSAR following HLP revised case approval.)

Errata, 5/5/89

TABLE 4.4-1

THERMAL AND HYDRAULIC COMPARISON TABLE

Design Parameters	W. B. McGuire Units 1 and 2	South Texas Unit 1	South Texas Unit 2
Reactor Core Heat Output, MWt	3,411	3,800	3,800
Reactor Core Heat Output, 10^6 Btu/hr	11,641	12,969	12,969
Heat Generated in Fuel, %	97.4	97.4	97.4
System Pressure, Nominal, psia	2,250	2,250	2,250
System Pressure, Minimum Steady State, psia	2,220	2,220	2,220
Minimum DNBR at Nominal Conditions			
Typical Flow Channel	2.08	~~2.16~~ 2.24	~~2.16~~ ~~2.19~~
Thimble (Cold Wall) Flow Channel	1.74	~~1.77~~ 2.07	~~1.77~~ ~~1.79~~
Minimum DNBR for Design Transients	≥1.30	~~≥1.30~~ 1.17	~~≥1.30~~ 1.17
DNB Correlation	"R" (W-3 with Modified Spacer Factor)	"R" (W-3 with Modified Spacer Factor) WRB-1	"R" (W-3 with Modified Spacer Factor) WRB-1
Coolant Flow			
Total Thermal Flow Rate, 10^6 lb_m/hr	140.3	~~139.6~~ 141.3	141.3
Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	134.0	~~133.2~~ 135.0	135.0
Effective Flow Area for Heat Transfer, ft^2	51.1	51.1	51.1
Average Velocity Along Fuel Rods, ft/sec	16.7	~~16.2~~ 16.9	~~16.7~~ 16.9
Average Mass Velocity, 10^6 lb_m/hr-ft^2	2.62	~~2.61~~ 2.64	2.64

CHANGE NOTICE 1348

STP FSAR

TABLE 4.4-1 (Continued)

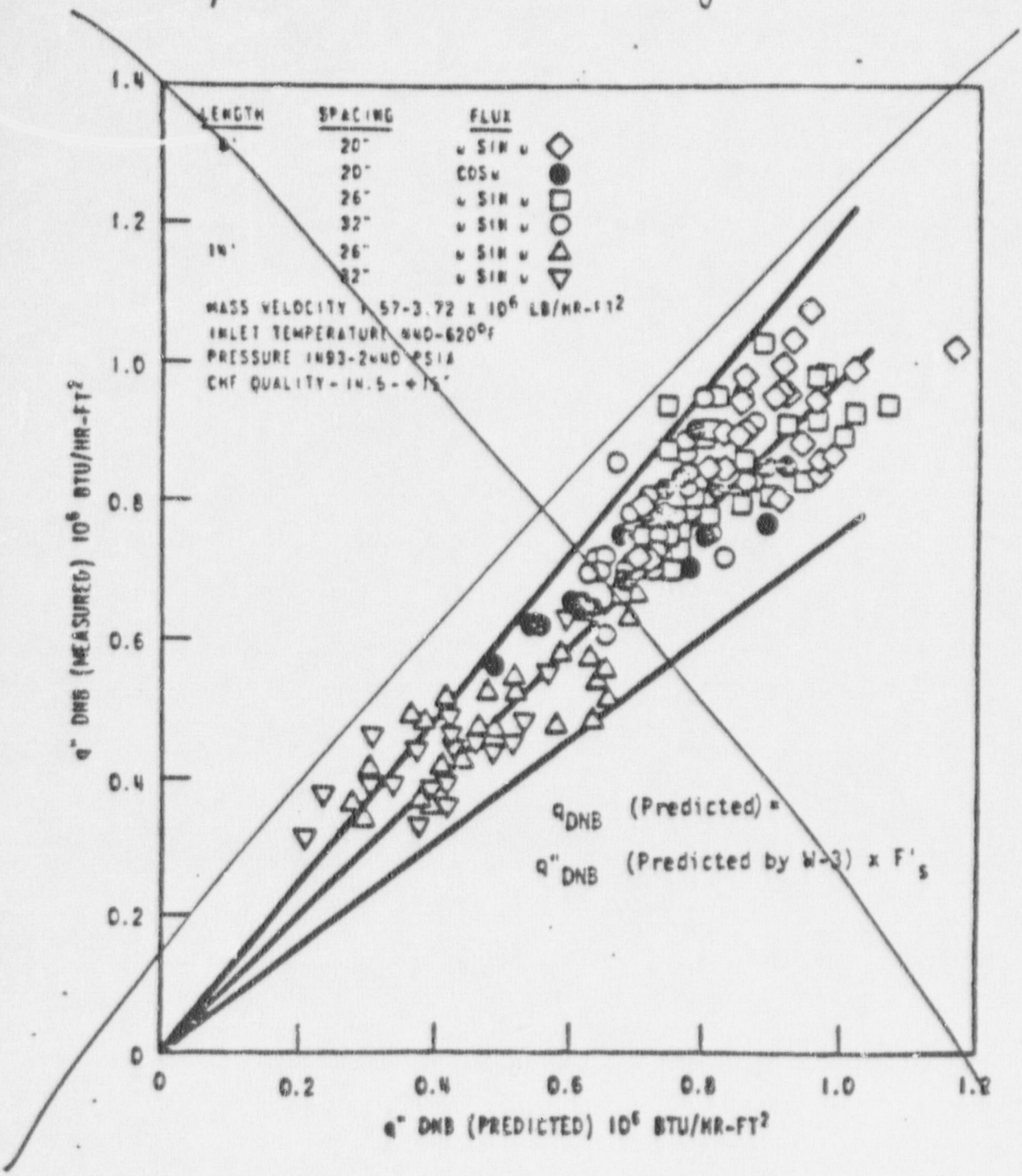
THERMAL AND HYDRAULIC COMPARISON TABLE

Design Parameters	W. B. McGuire Units 1 and 2	South Texas Unit 1	South Texas Unit 2	
Coolant Temperature				
Nominal Inlet, °F	558.1	~~560.2~~ 560.4	560.4	
Average Rise in Vessel, °F	60.2	~~66.2~~ 65.2	65.2	18
Average Rise in Core, °F	62.7	~~68.2~~ 67.9	67.9	
Average in Core, °F (Based on Avg. Enthalpy)	592.1	~~596.2~~ 596.5	596.5	44
Average in Vessel, °F	588.2	593.0	593.0	18
Heat Transfer				
Active Heat Transfer, Surface Area, ft^2	59,700	69,700	69,700	
Average Heat Flux, Btu/hr-ft^2	189,800	181,200	181,200	
Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	440,200 (a1)	453,100 (a2)	453,100 (a2)	
Average Linear Power, kW/ft	5.44	5.20	5.20	
Peak Linear Power for Normal Operation, kW/ft	12.6 (a1)	13.0 (a2)	13.0 (a2)	
Peak Linear Power Resulting from Overpower Transients/Operators Errors (assuming a maximum overpower of 118%), kW/ft (b)	18.0	18.0	18.0	
Peak Linear Power for Prevention of Centerline Melt, kW/ft (c)	▶18.0	▶18.0	▶18.0	
Power Density, kW per liter of core (d)	104.5	96.8	96.8	
Specific Power, kW per kg Uranium (d)	39.6	36.6	36.6	

CHANGE NOTICE 1348

6274-78

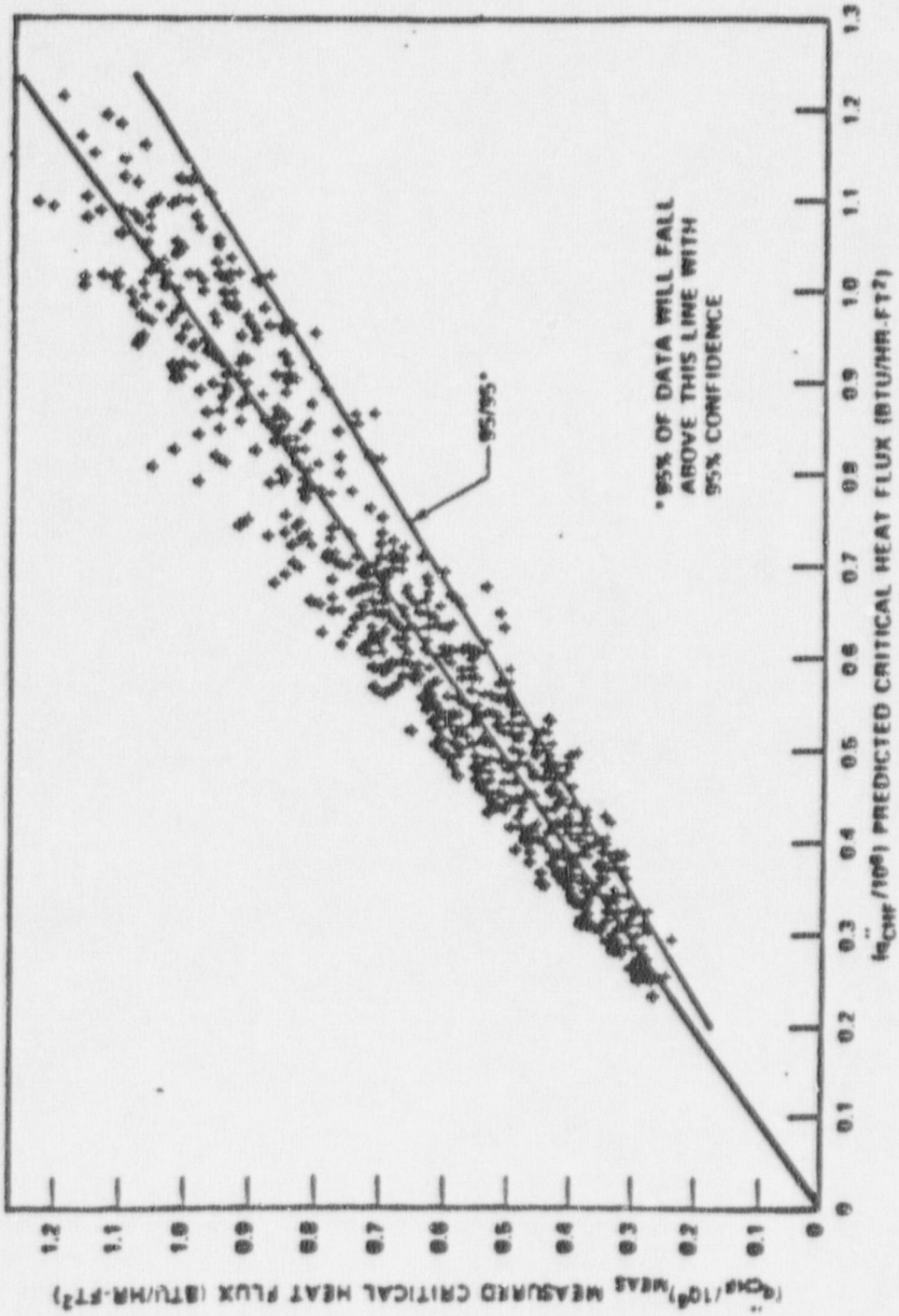
Replace with New Fig. 4.4-1



SOUTH TEXAS PROJECT
UNITS 1 & 2

Comparison of "R" Grid Data for Typical Cell (Reference [42])
Figure 4.4-1.

New Fig. 4.4-1



SOUTH TEXAS PROJECT
UNITS 1 & 2

Figure 4.4-1

Measured versus Predicted Critical Heat Flux – WRB-1 Correlation

8274-71

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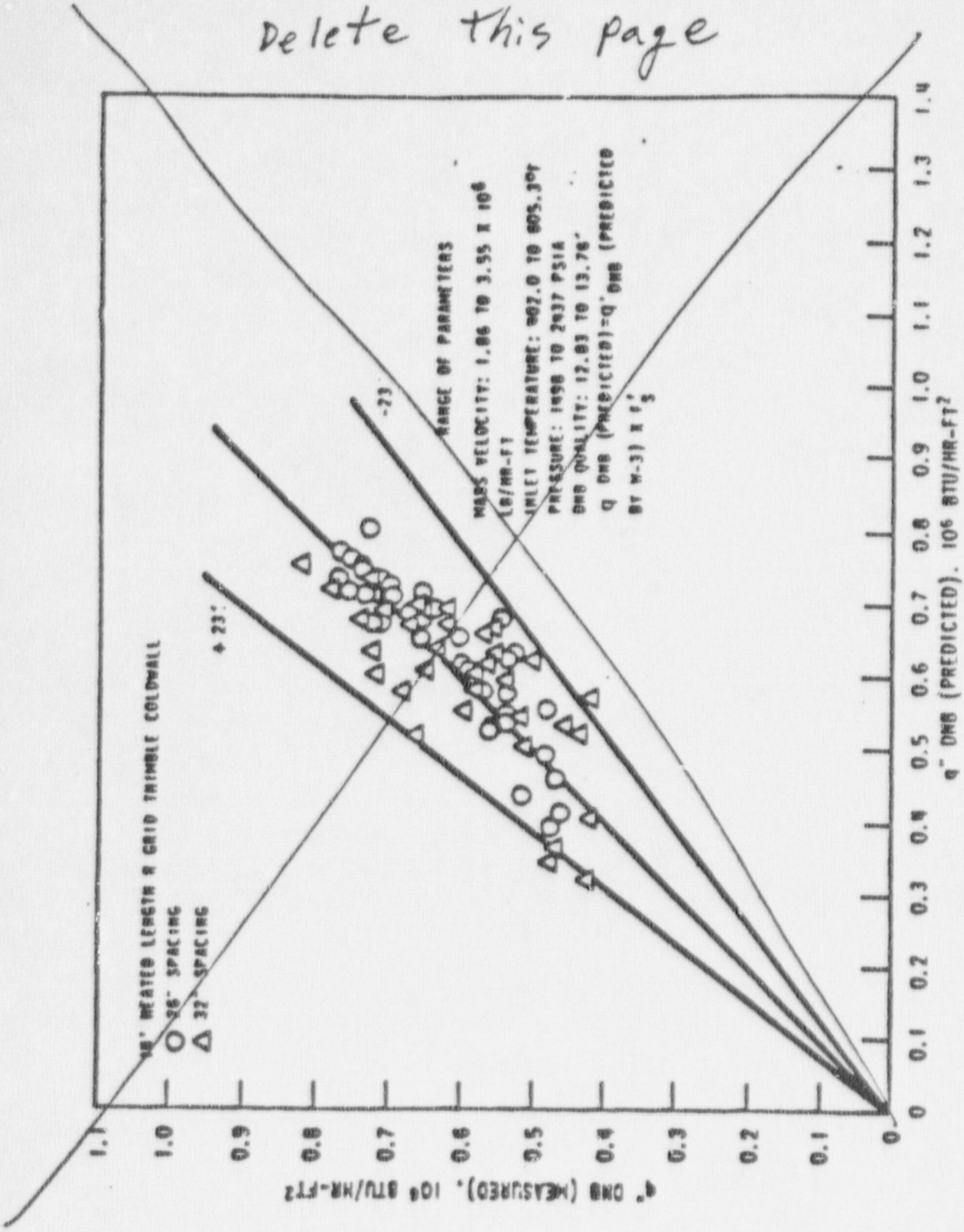

SOUTH TEXAS PROJECT
UNITS 1 & 2

Comparison of All "R" Grid Data for Thimble Cells
Figure 4.4-2.

12,373-14

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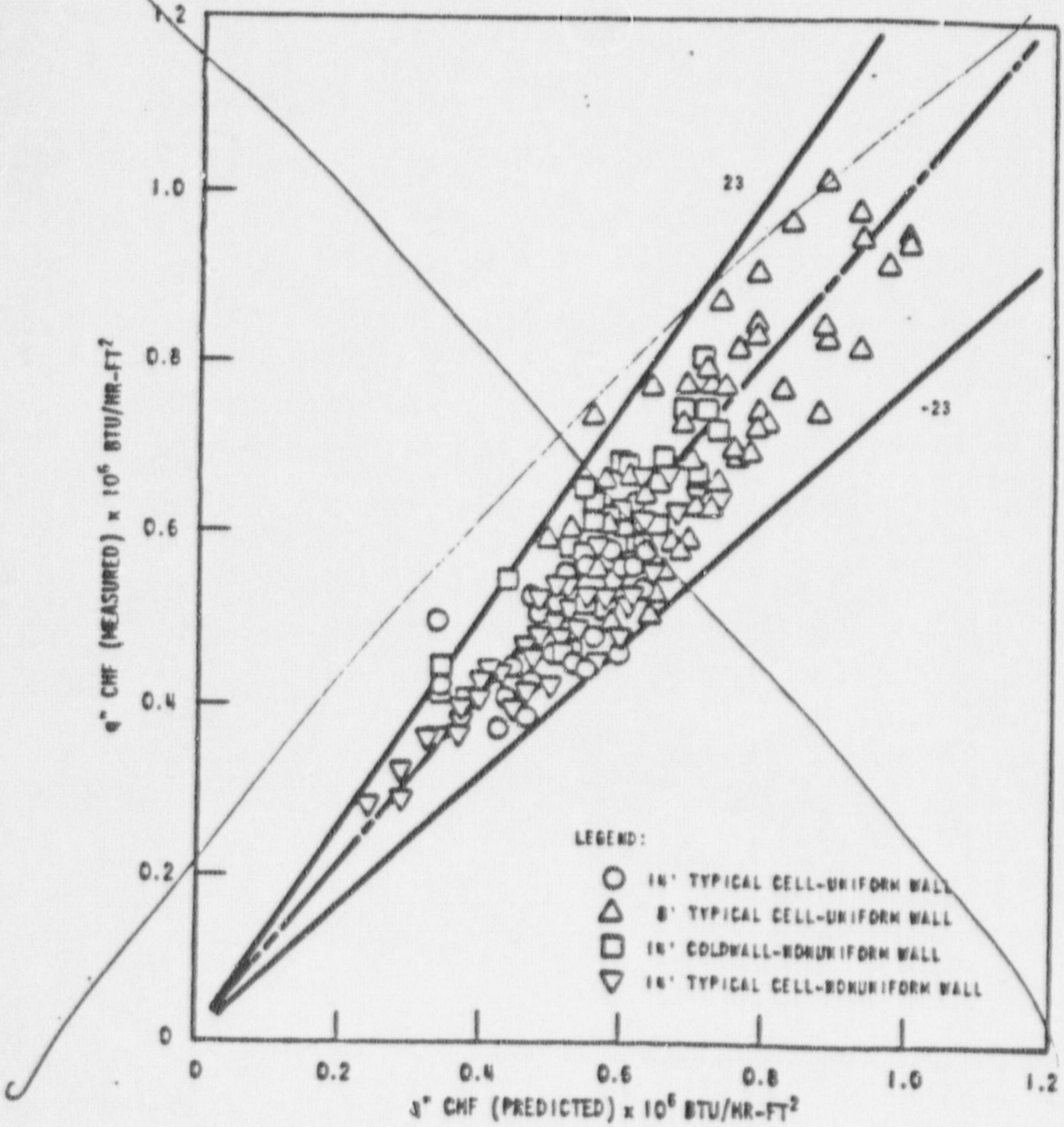


SOUTH TEXAS PROJECT
UNITS 1 & 2

Comparison of Measured to Predicted 17 x 17 DNB Data (Reference [2])
Figure 4.4-3.

Chapter 15

up to 2,000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures).

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 15.0-8.

15.0.10.6 THINC. The THINC code is described in Section 4.4.

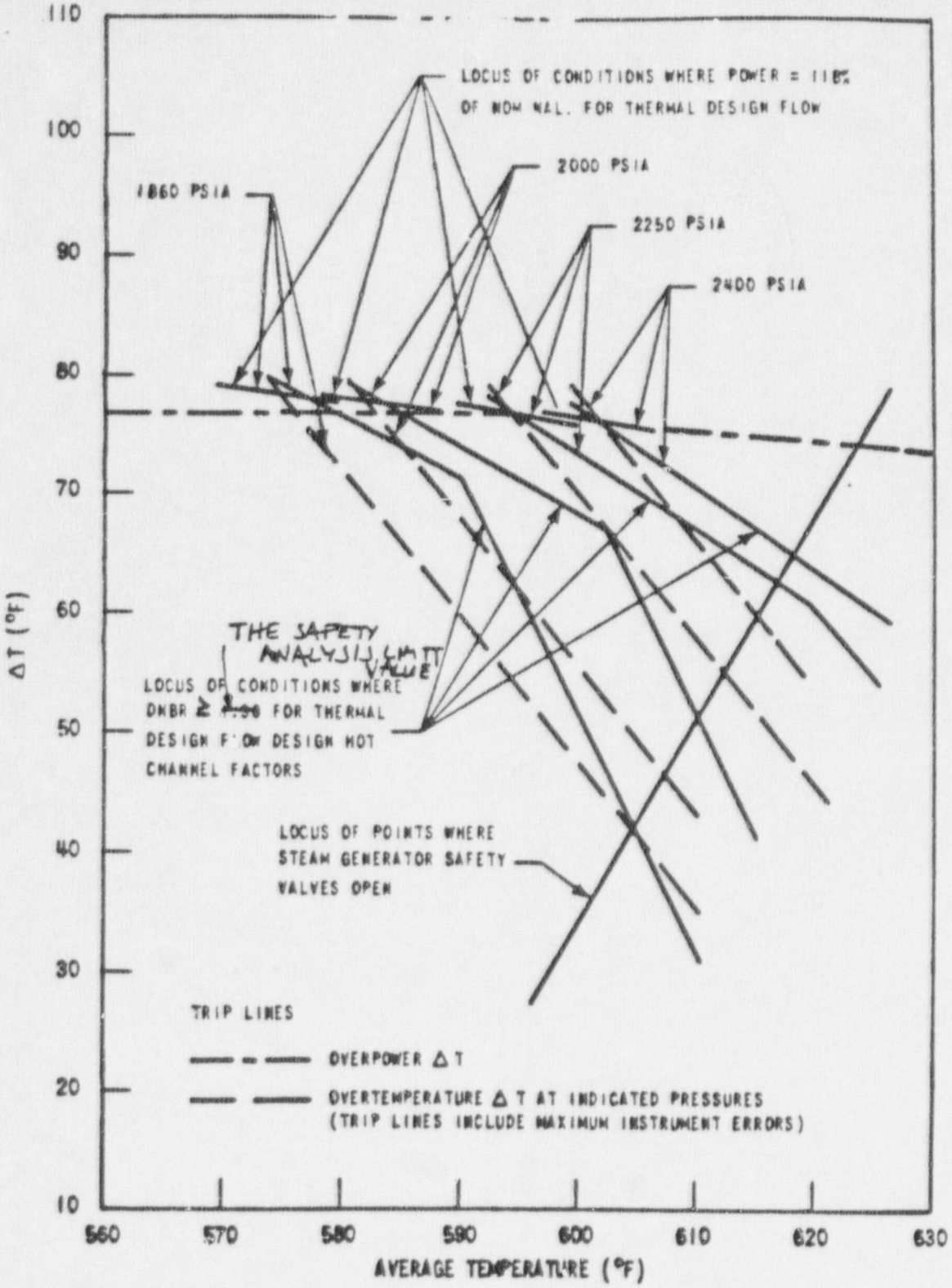
15.0.11 Summary of Accident Results 2 Q211.7

For all Condition II transients analyzed in the FSAR, the calculated minimum DNBR is greater than ~~1.30~~ the safety analysis limit value (see Section 4.4). For each of these transients, the peak RCS pressure is less than the safety limit of 110 percent of design pressure (2750 psia) and there is no failed fuel as a result of the transient. For all of the applicable Condition III transients, the minimum DNBR is greater than ~~1.30~~ the safety analysis limit value and there is no failed fuel, except for a single RCCA withdrawal at full power. For this transient, the upper bound of the number of fuel rods experiencing DNBR ~~<1.30~~ less than the safety analysis limit value is 5 percent of the total rods in the core. All of the applicable Condition III transients experience a peak RCS pressure less than 2750 psia. 60

All the applicable Condition IV transients analyzed in the FSAR have a minimum DNBR ~~>1.30~~ greater than or equal to the safety analysis limit value except LOCA, locked reactor coolant pump rotor, and rod ejection. For these three transients, the amount of failed fuel is ≤100 percent, ≤10 percent and ≤10 percent, respectively. Major rupture of a steam line experiences ≤5 percent failed fuel. All other applicable transients have no failed fuel. All of the applicable Condition IV transients experience a peak RCS pressure less than 2750 psia. 60 60 60

Block diagrams identifying protection sequences for Chapter 15 events are provided in Figures 15.0-7 through 15.0-31. 2 Q211.6

15.4.3.1-1



Amendment 18, 5/1/81

SOUTH TEXAS PROJECT UNITS 1 & 2

Figure 15.0-1.
Illustration of Overtemperature and Overpower Δ T Protection

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events has been postulated which could result in an increase in heat removal from the Reactor Coolant System (RCS) by the secondary system. Analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following RCS cooldown events are presented:

1. Feedwater system malfunction causing a reduction in feedwater temperature (Section 15.1.1).

2. Feedwater system malfunction causing an increase in feedwater flow (Section 15.1.2).

3. Excessive increase in secondary steam flow (Section 15.1.3).

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the Main Steam System (Section 15.1.4).

5. Spectrum of steam system piping failures inside and outside Containment (Section 15.1.5).

The above are considered to be American Nuclear Society (ANS) Condition II events, with the exception of a major steam system pipe break, which is considered to be ANS Condition IV event (Section 15.0.1).

15.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description. Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary system of the RCS. The overpower-overtemperature protection (neutron overpower, and overtemperature and overpower ΔT trips) prevents any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than ~~1.30~~. the safety analysis limit value. 43 18

A reduction in feedwater temperature could be caused by the accidental opening of a feedwater bypass valve which diverts flow around one of the high pressure feedwater heaters, or by the accidental closing of the extraction steam block valves or nonreturn valves to the high pressure feedwater heaters. (The deaerator will attenuate any upstream disturbances, i.e., low pressure heaters out of service, loss of extraction, steam, etc.) In the event of an accidental opening of a bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. If the extraction steam valves are accidentally closed, a more gradual, though greater, reduction in feedwater inlet temperature to the steam generators will occur. At power, this increased subcooling will create a greater load demand on the RCS. 43 57

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and, thus, a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease so the transient is less severe than the full power case.

15.1.1.4 Conclusions. The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Section 15.1.2) and the increase in secondary steam flow event (Section 15.1.3). Based on results presented in Section 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met. There are no radiological consequences of this event.

15.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description. Additions of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary system and of the RCS. The overpower-overtemperature protection (neutron overpower, and overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than ~~1.30~~ the safety analysis limit value. 43 18

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and, thus, a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. 43

Continuous addition of excessive feedwater is prevented by the steam generator high-high water level signal, which initiates feedwater isolation. The high-high steam generator water level signal initiates a turbine trip which then initiates a reactor trip. 43 57

An increase in normal feedwater flow is classified as an ANS Condition II event, a fault of moderate frequency (See Section 15.0.1).

Plant systems and equipment, which are available to mitigate the effects of the accident, are discussed in Section 15.0.8 and listed in Table 15.0-6.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-7. 2 Q211.6

15.1.2.2 Analysis of Effects and Consequences.

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code LOFTRAN (Ref. 15.1-1). This code simulates a multiloop system with neutron kinetics, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. 43

A control system malfunction or operator error is assumed to cause a feedwater control valve to open fully. Two cases are analyzed as follows:

automatic rod control mode results in a similar, but less limiting, transient. The rod control system is not required to function for an excessive feedwater flow event. 57

The calculated sequence of events for this accident is shown in Table 15.1-1. 43

When the steam generator water level in the faulted loop reaches the high-high level setpoint, all feedwater isolation valves and feedwater control valves are automatically closed and the SG feed pumps are tripped. This prevents continued addition of feedwater. In addition, a turbine trip is initiated. 43

Following turbine trip, the reactor will be automatically tripped directly due to turbine trip. If in manual rod control, the ensuing transient would then be similar to a turbine trip event as analyzed in Section 15.2.3 resulting in an overtemperature ΔT signal. If the reactor were in the automatic control mode, the control rods would be inserted at the maximum rate following turbine trip. 57

Transient results, see Figures 15.1-1 and 15.1-2, show the core heat flux, pressurizer pressure, Tavg and DNBR as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. The DNBR does not drop below ~~limit~~ the safety analysis limit value. Following the reactor trip, the plant approaches a stabilized condition; standard plant shutdown procedures may then be followed to further cool down the plant. 18

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant, hence the peak value does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). The peak fuel temperature will thus remain below the fuel melting temperature.

The transient results show that departure from nucleate boiling (DNB) does not occur at any time during the excessive feedwater flow incident; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

15.1.2.3 Radiological Consequences. There are only minimal radiological consequences from this event. The turbine trip causes a reactor trip and heat is removed from the secondary system through the steam generator power-operated relief valves (PORVs) or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences are less severe than the steam line break analyzed in Section 15.1.5.3. 43

15.1.2.4 Conclusions. The results of the analysis show that the DNBRs encountered for an excessive feedwater addition at power are at all times above ~~limit~~ the limit value, hence, the DNB design basis as described in Section 4.4 is met. Additionally, it has been shown that the reactivity insertion rate which occurs at no-load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition analysis. The radiological consequences of this event are not limiting. 18

2. Reactor control in manual with maximum moderator reactivity feedback.

3. Reactor control in automatic with minimum moderator reactivity feedback.

4. Reactor control in automatic with maximum moderator reactivity feedback.

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and, therefore, the least inherent transient capability. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For all cases, a small (absolute value) Doppler coefficient of reactivity is assumed (see Figure 15.0-2).

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters. Initial operating conditions are assumed at extreme values consistent with the steady-state full power operation allowing for calibration and instrument errors. This assumption results in minimum margin to core DNB at the start of the accident.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Normal reactor control systems and ESF systems are not required to function. The RTS is assumed not to be operable in order to show that DNBR criteria will be satisfied in the absence of reactor trip. No single active failure will prevent the RTS from performing its intended function when required. 43 18 43

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required as a safety feature. 18

Results

The calculated sequence of events for the excessive load increase incident is shown in Table 15.1-1.

Figures 15.1-3 through 15.1-6 illustrate the transient with the reactor in the manual control mode. For the minimum moderator feedback case, there is a slight power increase, the average core temperature decreases, and the DNBR increases slightly. For the maximum moderator feedback, manually controlled case there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above ~~1.30~~ the safety analysis limit value. 18 18

Figures 15.1-7 through 15.1-10 illustrate the transient assuming the reactor is in the automatic control mode. Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure so that there are only small changes in these parameters, as is also the case for the maximum moderator feedback case with manual control. For both of these cases, the minimum DNBR remains above ~~1.30~~ the safety analysis limit value. 18

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power. | 2

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip is not assumed for the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increased steam flow. If reactor trip were assumed, the analysis results would be less severe. | 18 | 18

Since DNB does not occur at any time during the excessive load increase transients, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

15.1.3.3 Conclusions. The analysis presented above shows that for a ten percent step load increase, the DNBR remains above the safety analysis limit value; ~~1.30,~~ thus, the DNB design basis as described in Section 4.4 is met. The plant reaches a stabilized condition rapidly following the load increase. | 18

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System

15.1.4.1 Identification of Causes and Accident Description. The most severe core conditions for an accidental depressurization of the main steam system result from an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Section 15.1.5.

The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The increased energy removal from the RCS causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly (RCCA), with offsite power available, and assuming a single failure in the ESF, there is no consequential damage to the core or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve. | 2

Accidental depressurization of the secondary system is classified as an ANS Condition II event (see Section 15.0.1).

The following systems provide the necessary protection for an accidental depressurization of the main steam system;

1. Safety injection actuation from either: | 2

   a. Two out of four low pressurizer pressure signals

5. In computing the steam flow, the Moody Curve (Ref. 15.1-3) for f(L/D) = 0 is used. 43

6. Perfect moisture separation in the steam generator is assumed.

Results

The calculated time sequence of events for this accident is listed in Table 15.1-1.

The results presented are a conservative indication of the events which would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figures 15.1-12 and 15.1-13 show the transient results for a steam flow of 292 lbs/sec. at 1300 psia from one steam generator.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve.

Safety injection is initiated by low pressurizer pressure. Boron solution at 2,500 ppm enters the RCS providing sufficient negative reactivity to maintain significant margin to core design limits. The cooldown for the case shown on Figures 15.1-12 and 15.1-13 as a result of the conservative analysis is more rapid than the case of steam release from all steam generators through one steam dump, relief, or safety valve. The calculated transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about five minutes, the neglected stored energy will have a significant effect in slowing the cooldown. 1 3 43 2

15.1.4.3 Radiological Consequences. There are only minimal radiological consequences from this event. The inadvertent opening of a single steam dump, relief or safety valve can result in steam release from the secondary system. Since no fuel damage is postulated to occur from this transient, the radiological consequence are less severe than the steam line break analyzed in Section 15.1.5.3. 43

15.1.4.4 Conclusions. The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the Main Steam System, the minimum DNBR remains well above the limit~~ing~~ value and no system design limits are exceeded. 3

Safety analysis

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside Containment

15.1.5.1 Identification of Causes and Accident Description. The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator 2

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Except as discussed above, normal reactor control systems and ESF systems are not required to function. Several cases are presented in which pressurizer spray and power-operated relief valves are assumed, but the more limiting cases where these functions are not assumed are also presented. 57

The RTS may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference 15.2-2.

Results

The transient responses for a turbine trip from 102 percent of full power operation are shown for four cases: two cases for minimum moderator feedback and two cases for maximum moderator feedback (Figures 15.2-1 through 15.2-8). For the minimum moderator feedback cases, the core has the least negative moderator coefficient of reactivity. For the maximum moderator feedback cases, the moderator temperature coefficient has its highest absolute value. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-1 and 15.2-2 show the transient responses for the turbine trip with minimum moderator feedback, assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the turbine bypass. The reactor is tripped on the high pressurizer pressure signal. The minimum departure from nucleate boiling ratio (DNBR) remains well above ~~1.30~~ the safety analysis limit value. The pressurizer safety valves are actuated and maintain primary system pressure below 110 percent of the design value. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint. 57 18

Figures 15.2-3 and 15.2-4 show the responses for the turbine trip with maximum moderator feedback. All plant parameters are the same as the above and the reactor is tripped on the high pressurizer pressure signal. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively. The pressurizer safety valves are not actuated for this case. 62

The turbine trip accident was also studied assuming the plant to be initially operating at 102 percent of full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or turbine bypass. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2-5 and 15.2-6 show the transients with minimum moderator feedback. The neutron flux remains essentially constant at 102 percent of full power until the reactor is tripped. The DNBR increases throughout the transient. In this case the pressurizer safety valves are actuated and maintain system pressure below 110 percent of the design value. 43

Figures 15.2-7 and 15.2-8 show the transients with maximum moderator feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient, and the pressurizer safety valves are actuated to limit primary pressure.

Reference 15.2-4 presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Radiological Consequences. There are only minimal radiological consequences associated with this event, therefore, this event is not limiting. The radiological consequences resulting from atmospheric steam dump are less severe than the steam line break event discussed in Section 15.1.5.

15.2.3.4 Conclusions. Results of the analyses, including those in Reference 15.2-4, show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The DNBR remains above the safety analysis limit value for all cases analyzed; thus, the DNB design basis as described in Section 4.4 is met. The above analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. 1

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

The inadvertent closure of main steam isolation valves would cause a turbine trip and other consequences as described in Section 15.2.5 below.

15.2.5 Loss of Condenser Vacuum and Other Events Causing a Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Section 10.2. A loss of condenser vacuum would preclude the use of turbine bypass to the condenser; however, since turbine bypass is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to loss of condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Section 10.2, are covered by Section 15.2.3. Possible overfrequency effects due to a turbine overspeed condition are discussed in Section 15.2.2.1 and are not a concern for this type of event. 4 4

Emergency operating procedures following a main feedwater line rupture require the operator to isolate feedwater flow spilling from the ruptured feedwater line and to control the RCS temperature which also prevents the pressurizer from filling. 32 Q211.74 54

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-13. 2 Q211.06

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

No reactor control systems with the exception of the pressurizer PORVs, are assumed to function. The operation of the PORVs serves to worsen the transient via minimizing the saturation temperature and therefore minimizing the margin to subcooling. It also allows a greater discharge of mass from the primary system, thus maximizing the liquid volume in the pressurizer. The RTS is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system. 54

Only one auxiliary feedwater pump is assumed to function following receipt of an initiating signal. Following initiation, the auxiliary pump is assumed to deliver 540 gal/min of auxiliary feedwater to one intact steam generator. 32

Results 32 Q211.74

Calculated plant parameters following a major feedwater line rupture are shown on Figures 15.2-11 through 15.2-24. Results for the case with offsite power available are presented on Figures 15.2-11 through 15.2-17. Results for the case where offsite power is lost are presented on Figures 15.2-18 through 15.2-24. The calculated sequence of events for both cases analyzed is listed in Table 15.2-1.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented on Figures 15.2-12 and 15.2-15 (with offsite power available) and Figures 15.2-19 and 15.2-22 (without offsite power) show that pressures in the RCS and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip on low-low steam generator water level. Pressure then decreases, due to the loss of heat input. Coolant expansion occurs due to reduced heat transfer capability in the steam generators; the pressurizer power-operated relief valves open to maintain RCS pressure at an acceptable value. 43

DNBR remains above ~~1.30~~ the safety analysis limit value at all times during the transients, as shown on Figures 15.2-17 and 15.2-24; thus, the DNB design basis as described in Section 4.4 is met. 18

The reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the AFW system.

The major difference between the two cases analyzed can be seen in the plots of hot and cold leg temperatures, Figures 15.2-13 and 15.2-14 (with offsite

Results

Figures 15.3-1 through 15.3-4 show the transient response for the loss of one reactor coolant pump with four loops in operation. Figure 15.3-4 shows the DNBR to be always greater than ~~1.30~~ the safety analysis limit value. 18

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values. 54

The calculated sequence of events is shown in Table 15.3-1. The affected reactor coolant pump will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed. 54

15.3.1.3 Radiological Consequences. A partial loss of reactor coolant flow from full load would result in a reactor and turbine trip. Assuming, in addition, that the condenser is not available, atmospheric steam dump may be required.

There are only minimal radiological consequences associated with this event. Therefore this event is not limiting. The radiological consequences resulting from atmospheric steam dump are less severe than the steam line break event analyzed in Section 15.1.5 since fuel damage as a result of this transient is not postulated.

15.3.1.4 Conclusions. The analysis shows that the DNBR will not decrease below ~~1.30~~ the safety analysis limit value at any time during the transient. Thus, the DNB design basis as described in Section 4.4 is met. 18

The radiological consequences of this event are not limiting.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description. A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly. 43

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator. When a generator, turbine, or reactor trip occurs, without an electrical fault, the generator circuit breaker automatically opens and back-feed of off-site power occurs through the main transformer and unit auxiliary transformer. Thus, the pumps will continue to supply coolant flow to the core. 43 54

undervoltage or underfrequency. One variation between this analysis and that of the previous section is that the RCCA insertion time to dashpot entry is 2.58 seconds. This is a conservative insertion time under the reduced flow conditions that exist when the RCCAs are inserted for this transient. 55

Results

Figures 15.3-9 through 15.3-12 show the transient response for the loss of power to all reactor coolant pumps with four loops in operation. The reactor is again assumed to be tripped on undervoltage signal. Figure 15.3-12 shows the DNBR to be always greater than ~~1.30~~ the safety analysis limit value. 54

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events is shown in Table 15.3-1. The reactor coolant pumps will continue to coastdown, and natural circulation flow will eventually be established, as demonstrated in Section 15.2.6. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed. 54

15.3.2.3 Radiological Consequences. A complete loss of reactor coolant flow from full load results in a reactor and turbine trip. Assuming, in addition, that the condenser is not available, atmospheric steam dump would be required. The quantity of steam released would be the same as for a loss of offsite power.

There are only minimal radiological consequences associated with this event. Therefore, this event is not limiting. Since fuel damage is not postulated, the radiological consequences resulting from atmospheric steam dump are less severe than the steam line break, discussed in Section 15.1.5.

15.3.2.4 Conclusions. The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below ~~1.30~~ the safety analysis limit value at any time during the transient. Thus, the DNB design basis as described in Section 4.4 is met. 18

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description. The accident postulated is an instantaneous seizure of a reactor coolant pump rotor such as is discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low reactor coolant flow signal. 43

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon turbine trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge 43

Figures 15.4-1 shows the neutron flux transient. The neutron flux does not overshoot the nominal full power value.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a heat flux much 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 15.4-3 shows the response of the hot spot fuel and cladding temperature. The hot spot fuel average temperature increases to a value below the nominal full power hot spot value. The minimum DNBR at all times remains above ~~1.30~~ the safety analysis limit value. | 18

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Radiological Consequences. There are no radiological consequences associated with an uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition event since radioactivity is contained within the fuel rods and RCS within design limits.

15.4.1.4 Conclusions. In the event of a RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR which is well above the limit~~ing~~ value. ~~of 1.30.~~ Thus, the DNB design basis as described in Section 4.4 is met. | 18

safety analysis

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description. Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generators lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the Reactor Trip System (RTS) is designed to terminate any such transient before the DNBR falls below ~~1.30~~ the safety analysis limit value. | 43

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Section 15.0.1.

The automatic features of the RTS which prevent core damage following the postulated accident include the following: | 43

1. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

4. A high pressurizer pressure reactor trip is actuated if any two out of four pressure channels exceed the setpoint. This set pressure is less than the set pressure for the pressurizer safety valves. 43

5. A high pressurizer water level reactor trip is actuated if any two out of four level channels exceed the setpoint. 43

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

1. High neutron flux (one out of four);

2. Overpower ΔT (two out of four); and,

3. Overtemperature ΔT (two out of four).

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7. Figure 15.0-1 presents allowable reactor coolant loop average temperatures and ΔTs for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals ~~1.30~~ the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than ~~1.30~~ this limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints). 43

15.4.2.2 Analysis of Effects and Consequences.

Method of Analysis

This transient is analyzed by the LOFTRAN code (Ref. 15.4-3). This code simulates the neutron kinetics, RCS, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated on Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

The transient response for a slow RCCA withdrawal from full power is shown on Figures 15.4-7 through 15.4-9. Reactor trip on overtemperature ΔT occurs after a longer period. Again, the minimum DNBR is greater than ~~1.30~~ the safety analysis limit value. 18

Figure 15.4-10 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is always greater than ~~1.30~~ the safety analysis limit value. 18

Figures 15.4-11 and 15.4-12 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below ~~1.30~~ the safety analysis limit value. 18

The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to the minimum feedback case in Figure 15.4-11, for example, it is noted that:

1. For high reactivity insertion rates (i.e., between approximately 3×10^{-4} Δk/sec and 1.0×10^{-3} Δk/sec) reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNBRs during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNBR during the transient thus decreases with decreasing insertion rate. 62

2. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured RCS average temperature and pressure. This trip circuit is described in detail in Chapter 7; however, it is important in this context to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the RCS in response to power increases.

3. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (e.g., at approximately 3×10^{-4} Δk/sec reactivity insertion rate). 62

   For reactivity insertion rates between approximately 3×10^{-4} Δk/sec and approximately 5×10^{-5} Δk/sec the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of 62

15.4.2.4 Conclusions. The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates (i.e. the minimum value of DNBR is always larger than ~~1.30~~ the safety analysis limit value). Thus, the DNB design basis as described in Section 4.4 is met. |62 |43

15.4.3 Rod Cluster Control Assembly Misoperation |43

15.4.3.1 Identification of Causes and Accident Description. RCCA misoperation accidents include:

1. One or more dropped RCCAs within the same group; |30
2. A dropped RCCA bank;
3. Statically misaligned RCCA; |43
4. Withdrawal of a single RCCA. |30

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion or immobility. |30 |43 |53 |43 |53

The dropped RCCA, dropped RCCA bank, and statically misaligned RCCA events are classified as American Nuclear Society (ANS) Condition II incidents (incidents of moderate frequency) as defined in Section 15.0.1. However, the single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below. |53

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures (probability for single random failure is on the order of 10^{-4}/year; refer to Section 7.7.2.2) or multiple serious operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is very low. The consequences, however, may include slight fuel damage. Thus, consistent with |43 |43 |53 |57 |53

curves during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than ~~1.30~~ the safety analysis limit value.

Reactivity addition for the inactive loop startup accident case is due to the decrease in core water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the cooler water entering the core from the hot leg side (colder temperature side prior to the startup of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.4-16.

The calculated sequence of events for this accident is shown in Table 15.4-1. The transient results illustrated on Figures 15.4-16 through 15.4-20 indicate that a stabilized plant condition, with the reactor tripped, is approached at 30 seconds. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.3 Radiological Consequences. There are only minimal radiological consequences associated with startup of an inactive reactor coolant loop at an incorrect temperature. Therefore, this event is not limiting. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences associated with this event are less severe than the steam line break event, as discussed in Section 15.1.5.

15.4.4.4 Conclusions. The transient results show that the core is not adversely affected. The DNBR remains above the safety analysis limit value ~~1.30~~ throughout the transient; thus, the DNB design basis as described in Section 4.4 is met. 43

15.4.5 A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate

Not applicable to South Texas.

15.4.6 Chemical and Volume Control System (CVCS) Malfunction That Results in
a Decrease in Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description. Reactivity can be added to the core by feeding unborated water into the RCS via the CVCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The CVCS is designed to limit the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner. 57

15.6.1.2 Analysis of Effects and Consequences. The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN (Ref. 15.6-1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator (SG), and SG safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Plant characteristics and initial conditions are discussed in Section 15.0.3. In order to give conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions are made.

1. Initial conditions of maximum core power and reactor coolant temperature (+4.7°F uncertainty) and minimum reactor coolant pressure (-46 psi uncertainty) are assumed. This results in the minimum initial margin to departure from nucleate boiling (DNB) (see Section 15.0.3). 57 | 62

2. A least negative moderator temperature coefficient of reactivity is assumed. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

3. A large (absolute value) Doppler coefficient of reactivity is assumed (Fig. 15.0-2) so that the resultant amount of positive feedback is conservatively high. This retards any power decrease due to moderator reactivity feedback.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode in order to hold the core at full power longer and thus delay the trip. This is a worst-case assumption; if the reactor were in manual control, an earlier trip could occur on low pressurizer pressure. The RTS functions to trip the reactor on the appropriate signal. No single active failure will prevent the RTS from functioning properly. | 45

Results

The system response to an inadvertent opening of a pressurizer safety valve is shown on Figures 15.6-1 through 15.6-3. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on overtemperature ΔT. The pressure decay transient and average temperature transient following the accident are given on Figure 15.6-2. Pressure drops more rapidly after core heat generation is reduced via the trip, and then slows once saturation temperature is reached in the hot leg. The DNBR transient is shown on Figure 15.6-3; DNBR remains above ~~1.30~~ the safety analysis limit value throughout the transient. | 45

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown in Table 15.6-1.

15.6.1.3 Radiological Consequences. An inadvertent opening of a pressurizer safety or relief valve releases primary coolant to the pressurizer relief tank; however, even assuming a direct release to the Containment atmosphere, the radiological consequences of this event would be substantially less

Question 440.62N

Provide the values of the moderator and Doppler coefficients of reactivity used in the loss of normal feedwater/loss of offsite power analysis and verify their conservatism.

Response

The acceptance criteria used for these transients are a) the pressurizer was not permitted to become water solid and b) the minimum departure from nucleate boiling ratio (DNBR) must be greater than the limit value. ~~1.30~~ Thus, the moderator and Doppler coefficients of reactivity were selected to yield conservative results. This was done by selecting the moderator and Doppler coefficients of reactivity to maximize core power which in turn maximizes the volume expansion. The pertinent coefficients are listed in Table 15.0-2.

Question 440.62N

Provide the values of the moderator and Doppler coefficients of reactivity used in the loss of normal feedwater/loss of offsite power analysis and verify their conservatism.

Response

The acceptance criteria used for these transients are a) the pressurizer was not permitted to become water solid and b) the minimum departure from nucleate boiling ratio (DNBR) must be greater than the limit value ~~1.30~~ safety analysis. Thus, the moderator and Doppler coefficients of reactivity were selected to yield conservative results. This was done by selecting the moderator and Doppler coefficients of reactivity to maximize core power which in turn maximizes the volume expansion. The pertinent coefficients are listed in Table 15.0-2.

Chapter 7

7.2.1.2.1 Generating Station Conditions: Generating station conditions requiring a reactor trip are the following:

1. DNBR approaching the design basis limit (Chapters 4 and 15).

2. Power density (kW/ft) approaching rated value for ANS Condition II faults (see Chapter 4 for fuel design limits).

3. Reactor Coolant System (RCS) overpressure creating stresses approaching the limits specified in Chapter 5.

7.2.1.2.2 Generating Station Variables: The following variables are required to be monitored to provide reactor trips (see Table 7.2-1):

1. Neutron flux

2. Reactor coolant temperature

3. RCS pressure (pressurizer pressure)

4. Pressurizer water level

5. Reactor coolant flow

6. RCP operational status (voltage and frequency)

7. Steam generator water level (density compensated) |43

8. Turbine generator operational status (trip fluid pressure and stop valve position)

7.2.1.2.3 Spatially Dependent Variables: The reactor coolant temperature measurement is the only spatially dependent variable. |54

7.2.1.2.4 Limits, Margins, and Setpoints: The parametric values that will require reactor trip are given in Chapter 15. Chapter 15 proves that the setpoints to be used in the Technical Specifications are conservative. |43

The setpoints for the various functions in the RTS were analytically determined so that the operational limits so prescribed will prevent fuel rod clad damage and loss of integrity of the RCS as a result of any ANS Condition II incident (anticipated malfunction). As such, during any ANS Condition II incident, the RTS limits the following parameters to:

1. Minimum DNBR - Design Limit DNBR (as discussed in Section 4.4.1) |43

2. Maximum system pressure - 2,750 psia

3. Fuel rod maximum linear power for determination of protection setpoints - 18.0 kW/ft

TECH SPECS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the ~~W-3 R Grid~~ correlation. The ~~W-3 R Grid~~ DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

WRB-1

WRB-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% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.~~

Replace with Insert A

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.

1.27

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.52 and a reference cosine with a peak of 1.61 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.52\ [1 + 0.3\ (1-P)]$$

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 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

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Insert A

The minimum value of DNBR during steady state, normal operational transients, and anticipated transients is limited to 1.17, the DNBR design limit of the WRB-1 correlation. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur. Margin is maintained by meeting a DNBR value of 1.27 in the safety analyses.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than 1.30

the design limit

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BASES

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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

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

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement (395,000 gpm) and the requirement on $F^N_{\Delta H}$ guarantees that the DNBR used in the safety analysis will be met. 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 $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limit. A measurement error of 4% for $F^N_{\Delta H}$ has been allowed for in the determination of the design DNBR value.

~~Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 3.3% DNBR completely offset any rod bow penalties. This margin includes the following:~~

~~a. Design limit DNBR of 1.30 vs 1.28,~~

~~b. Grid Spacing (K_s) of 0.059 vs 0.065, and~~

~~c. Thermal Diffusion Coefficient (for use in modified spacer factor) of 0.059 vs 0.061.~~

~~The applicable values of rod bow penalties are explained in FSAR Section 4.4.2.2.5.~~

Replace with Insert B

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3/4.4 REACTOR COOLANT SYSTEM

the design limit

BASES

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 6 hours.

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.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The boron dilution analysis assumed a common RCS volume, and maximum dilution flow rate for MODES 3 and 4, and a different volume and flow rate for MODE 5. The MODE 5 conditions assumed limited mixing in the RCS and cooling with the RHR system only. In MODES 3 and 4, it was assumed that at least one reactor coolant pump was operating. If at least one reactor coolant pump is not operating in MODE 3 or 4, then the maximum possible dilution flow rate must be limited to the value assumed for MODE 5.

The operation of one reactor coolant pump (RCP) 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.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY 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 504,950 lbs per hour of saturated steam at the valve setpoint of 2500 psia. 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 safety valves are OPERABLE, an operating RHR loop, connected to the

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BASES

3/4.2.5 DNB PARAMETERS (Continued)

initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 598°F and the indicated pressurizer pressure value of 2200 psig are provided assuming that the readings from four channels will be averaged before comparing with the required limit. The flow requirement (395,000 gpm) includes a measurement uncertainty of 3.5%.

greater than or equal to the design 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.

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Insert B

Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses (1.27) and the design limit (1.17) to offset the rod bow penalty and other penalties which may apply.