2) Cold Shutdown

The reactor is in the cold shutdown condition when the reactor has a shutdown margin of at least 1% $\Delta k/k$ and reactor coolant temperature is $\leq 200^{o}F$.

3) Refueling Shutdown

The reactor is in the refueling shutdown condition when the reactor is subcritical by at least 10% $\Delta k/k$ and T_{avg} is $\leq 140^{o}F$. A refueling shutdown refers to a shutdown to move fuel to and from the reactor core.

4) Shutdown Margin

Shutdown margin is the instantaneous amount of reactivity by which the reactor core would be subcritical if all withdrawn control rods were tripped into the core but the highest worth withdrawn RCCA remains fully withdrawn. If the reactor is shut down from a power condition, the hot shutdown temperature should be assumed. In other cases, no change in temperature should be assumed.

h. Power Operation

The reactor is in power operating condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates greater than 2% of FULL power.

i. Refueling Operation

Refueling operation is any operation involving movement of core components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

j. Rated Power

Rated power is here defined as a steady state reactor core output of 1518.5 MWT.

k. Thermal Power

Thermal power is defined as the total core heat transferred from the fuel to the coolant.

l. Degree of Redundancy

Degree of redundancy is defined as the difference between the number of operable channels and the minimum number of channels which when tripped will cause an automatic shutdown.

m. Reactor Critical

The reactor is said to be critical when the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

n. Low Power Operation

The reactor is in the low power operating condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates less than or equal to 2% of FULL power.

o. Fire Suppression Water System

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source; pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard post indicating valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

p. Full Power

Full power is defined as 100% of rated power when the RCS flow is $\geq$ 178000 gpm. When RCS total flow is < 178000 gpm, full power is defined to be 91% of rated power.

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 when RCS Total Flow rate $\geq$ 178000 gpm, and Figure 15.2.1-2 when RCS Total Flow rate < 178000 gpm. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

Basis:

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNB ratio of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure 15.2.1-1 and 15.2.1-2 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNB ratio is not less than 1.30. The area of safe operation is below these lines. The safety limits curves have been revised to allow for heat flux peaking effects due to fuel densification and flattened fuel cladding sections.

Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of the curves shown in Figures 15.2.1-1 and 15.2.1-2. These curves are based on an $F^N_{\Delta H}$ of 1.58, cosine axial flux shape, and a DNB analysis as described in Section 4.3 of WCAP-8050, "Fuel Densification, Point Beach Nuclear Plant Unit 1 Cycle 2" (including the effects of fuel densification and flattened cladding).

Figures 15.2.1-1 and 15.2.1-2 also include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H} = 1.58 \{1 + 0.2 (1-P)\}$ where P is a fraction of FULL power when $P < 1.0$ $F^N_{\Delta H} = 1.58$ when $P \geq 1.0$.

The effects of rod bow have been included in the determination of a conservative value for $F^N_{\Delta H}$. Rod bow effects of up to 14.9% DNBR are offset by credits available from the design limit DNBR, pitch reduction, design thermal diffusion coefficient and the fuel densification power spike, which were previously approved.*

The hot channel factors are also sufficiently large to account for the degree of malpositioning of full-length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached. The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30.

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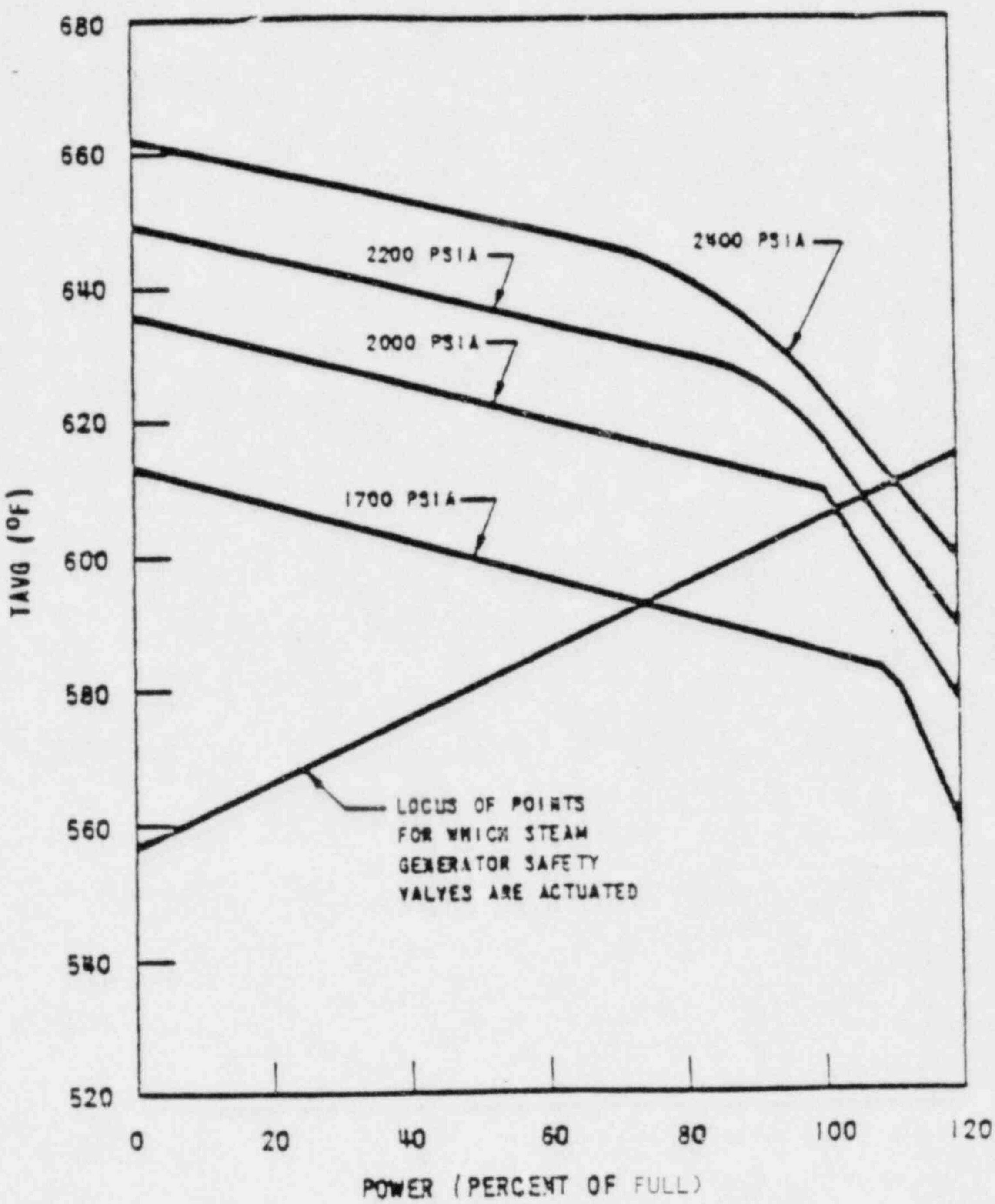
* Memorandum from D. F. Ross and D. G. Eisenhut, USNRC, to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977.



TOTAL RCS FLOW ≥ 178000 GPM

Figure 15.2.1-1 Core DNB Safety Limits
Point Beach Units 1 and 2

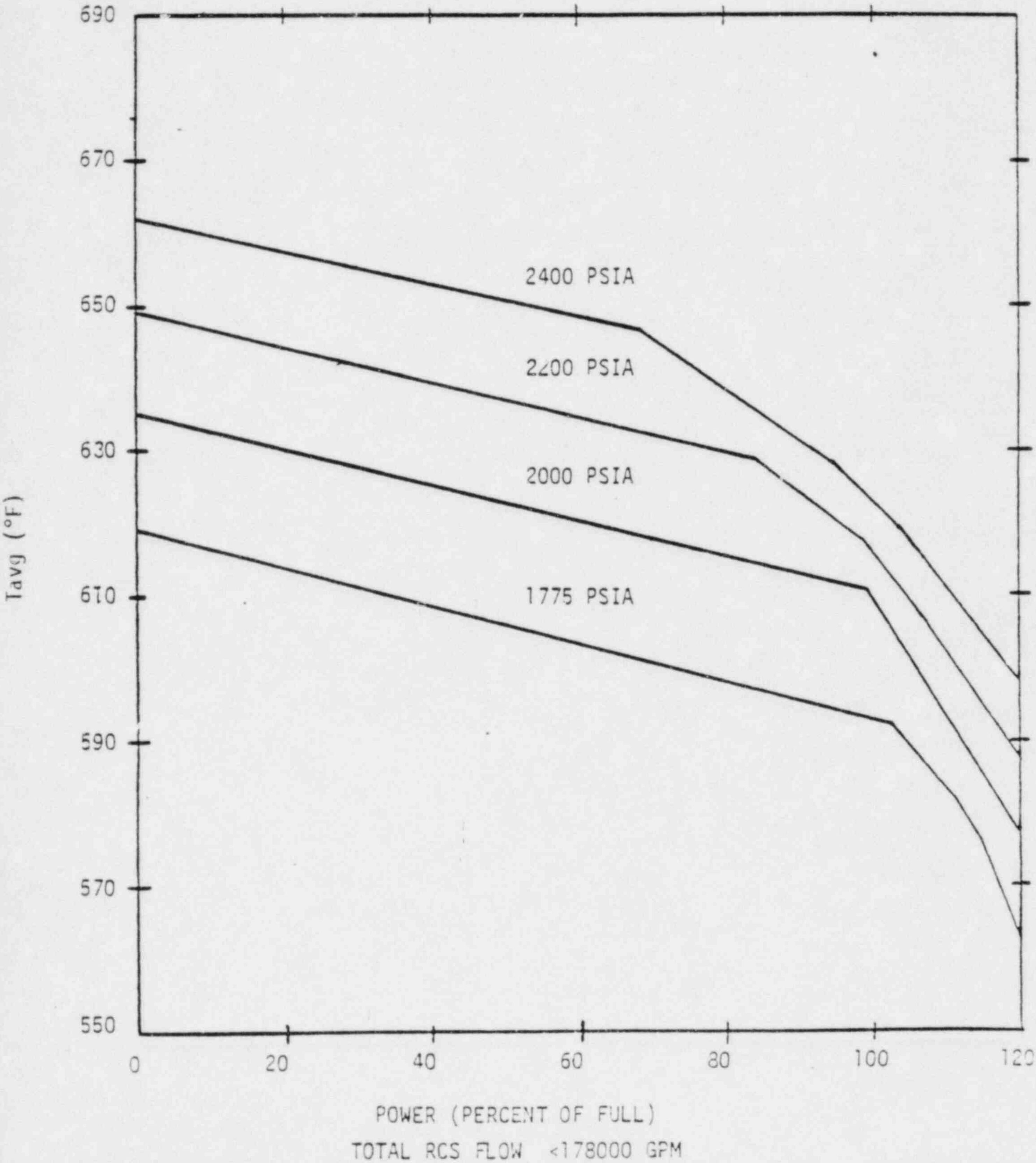


FIGURE 15.2.1-2 CORE DNB SAFETY LIMITS

POINT BEACH UNITS 1

15.2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability:

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

Objective:

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

Specification:

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, source range - within span of source range instrumentation.

      (2) High flux, intermediate range - ≤ 40% of FULL power.

      (3) High flux, power range (low set point) - ≤ 25% of FULL power.

   B. Core limit protection

      (1) High flux, power range (high setpoint)

      ≤ 108% of FULL power

      (2) High pressurizer pressure - ≤ 2385 psig.

(3) Low pressurizer pressure - $\geq$ 1790 psig for operation at 2000 psia primary system pressure

(4) Overtemperature ΔT

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

where

ΔT_o = indicated ΔT at FULL power, °F

T = average temperature, °F

T' = 574.2 °F

P = pressurizer pressure, psig

P' = 2235 psig

K_1 $\leq$ 1.30 for operation at 2000 psia primary system pressure

K_2 = 0.0150

K_3 = 0.000791

$\tau 1$ = 25 sec.

$\tau 2$ = 3 sec.

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of FULL power, such that:

(a) for $q_t - q_b$ within -17, +9 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +9 percent the ΔT trip set point shall be automatically reduced by an equivalent of two percent of FULL power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent the ΔT trip setpoint shall be automatically reduced by an equivalent of two percent of FULL power.

[1.3. (5)] Overpower ΔT

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

where

ΔT_o = indicated ΔT at FULL power, °F

T = average temperature, °F

T' = 574.2

K_4 $\leq$ 1.089 of FULL power

K_5 = 0.0262 for increasing T

= 0.0 for decr T

K_6 = 0.00123 for $T \geq T'$

= 0.0 for $T < T'$

τ_3 = 10 sec.

$f(\Delta I)$ as defined in (4) above,

(6) Undervoltage - $\geq$ 75% of normal voltage

(7) Low indicated reactor coolant flow per loop- $\geq$90% of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point $\geq$57.5 cps

(b) Low voltage set point $\geq$75% of normal voltage

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

A. The "at power" reactor trips (low pressurizer pressure, high pressurizer level, and low reactor coolant flow for both loops) shall be unblocked when:

(1) Power range nuclear flux $\geq$ 9% ($\pm$1%) of FULL power or

(2) Turbine Load $\geq$ 10% of full load turbine pressure.

B. The single loss of flow trip shall be unblocked when the power range nuclear flux $\geq$ 50% of FULL power.

C. The power range high flux level low range trip, and intermediate range high flux level trip shall be unblocked when power is $\leq$ 9% ($\pm$1%) of FULL power.

D. The source range high flux reactor trip shall be unblocked when the intermediate range flux is $\leq 10^{-10}$ amperes.

power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown on Figure 15.2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6] [7]

The overpower, overtemperature and pressurizer pressure system setpoints have been revised to include effect of reduced system pressure operation (including the effects of fuel densification). The revised setpoints as given above will not exceed the revised core safety limits as shown in Figure 15.2.1-1 and 15.2.1-2.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[4]

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis.[8] The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency setpoint, 57.5 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value. The high pressurizer water level reactor trip protects the pressurizer safety valves agains water relief. The specified setpoint allows adequate operating instrument error[2] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[9]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed. Specifications 15.2.3.2.A(1) and 15.2.3.2.C have ±1% tolerance to allow for a 2% deadband of the P10 bistable which is used to set the limit of both items.

Sustained operation with only one pump will not be permitted above 10% Full power. If a pump is lost while operating between 10% and 50% of Full power, an orderly and immediate reduction in power level to below 10% of Full power is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.

References

(1) FSAR 14.1.1	(4) FSAR 14.3.1	(7) FSAR 3.2.1
(2) FSAR, Page 14-3	(5) FSAR 14.1.2	(8) FSAR 14.1.9
(3) FSAR 14.2.6	(5) FSAR 7.2, 7.3	(9) FSAR 14.1.11

15.3 LIMITING CONDITIONS FOR OPERATION

15.3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

Specification:

1. Coolant Pumps

   a. At least one reactor coolant pump or the residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   b. When the reactor is critical and above 1% Full power, except for natural circulation tests, at least one reactor coolant pump shall be in operation.

   c. (1) Reactor power shall not be maintained above 10% of FULL power unless both reactor coolant pumps are in operation.

      (2) If either reactor coolant pump ceases operating, immediate power reduction shall be initiated under administrative control as necessary to reduce power to less than 10% of FULL power.

2. Steam Generator

   a. One steam generator shall be operable whenever the average reactor coolant temperature is above 350°F.

because of the low pressurizer volume and because pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Part 1 of the specification requires that a sufficient number of reactor coolant pumps be operating to provide core cooling in the event that a loss of flow occurs. The flow provided in each case will keep DNBR well above 1.30 as discussed in FFDSAR Section 14.1.9. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Heat transfer analyses (1) show that reactor heat equivalent to 10% of FULL power can be removed with natural circulation only; hence, the specified upper limit of 1% FULL power without operating pumps provide a substantial safety factor.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at setpoint. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat is removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization. Part 1 c(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 10% and 50% of FULL power. Above 50% FULL power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0 which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.(2)

A PORV is defined as OPERABLE if leakage past the valve is less than that allowed in Specification 15.3.1.D and the PORV has met its most recent channel test as specified in Table 15.4.1-1. The PORVs operate to relieve, in a controlled manner, reactor coolant system pressure increases below

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown during FULL power operation:

1. T_{AVG} shall be maintained:
   $\leq$ 578°F when RCS total flow $\geq$ 178000 gpm
   $\leq$ 576.9°F when RCS total flow < 178000 gpm

2. Reactor coolant system pressure shall be maintained:
   $\geq$ 1955 psig during operation at 2000 psia

3. Reactor Coolant System Total Flow Rate $\geq$ (.95) x 178,000

Basis:

Although the operational limitations above require reactor coolant system total flow be maintained above a minimum rate, no direct means of measuring absolute flow during operation exist. However, during initial startup reactor coolant flow was measured and correlated to core ΔT. Therefore monitoring of ΔT may be used to verify the above minimum flow requirement is met. If a change in steady state full power ΔT greater than 3°F is observed, the actual flow measurements will be taken.

The eight main steam safety valves have a total combined rated capability of 6,664,000 lbs/hr. The total rated steam flow is 6,620,000 lbs/hr, therefore eight (8) main steam safety valves will be able to relieve the total full steam flow if necessary.

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured for each unit by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves or atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from a unit. The minimum amount of water in the condensate storage tanks is the amount needed for 25 minutes of operation/unit, which allows sufficient time for operator action.

An unlimited supply is available from the lake via either leg of the plant service water system for an indefinite time period.

15.3.5 INSTRUMENTATION SYSTEM

Operational Safety Instrumentation

Applicability:

Applies to plant instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

A. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 15.3.5-1.

B. For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at FULL power shall be permitted. to continue in accordance with Tables 15.3.5-2 through 15.3.5-4.

C. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Tables 15.3.5-2 through 15.3.5-4, Operator Action when minimum operable channels unavailable.

D. The accident monitoring instrumentation channels in Table 15.3.5-5 shall be operable. In the event the number of channels in a particular sub-system falls below the minimum number of operable channels given in Column 2, operation and subsequent operator action shall be in accordance with Column 3.

Basis:

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features(1).

No.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
11.	Turbine Trip	3	2	2	1		Maintain <50% of FULL power
12.	Steam Flow - Feed Water Flow mismatch	2/loop	1/loop	1/loop	1/loop		Maintain hot shutdown
13.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop		Maintain hot shutdown
14.	Undervoltage 4 KV Bus	2/bus	1/bus (both buses)	1/bus	--		Maintain hot shutdown
15.	Underfrequency 4 KV Bus	2/bus	1/bus (both buses)	1/bus	--		Maintain hot shutdown

NOTE: When block condition exists, maintain normal operation.

FP. = FULL power

* Not Applicable

** One additional channel may be taken out of service for zero power physics testing.

TABLE 15 3.5-5

INSTRUMENT OPERATING CONDITIONS FOR INDICATIONS

O.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNEL	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
1.	PORV Position Indicator	1/Valve	1/Valve	If the operability of the PORV position indicator cannot be restored within 48 hours, shut the associated PORV Block Valve.
2.	PORV Block Valve Position Indicator	1/Valve	1/Valve	If the operability of the PORV Block Valve Position Indicator cannot be restored within 48 hours, shut and verify the Block Valve shut by direct observation or declare the Block Valve inoperable.
3.	Safety Valve Position Indicator	1/Valve	1/Valve	If the operability of the Safety Valve Position Indicator cannot be restored within seven days, be in at least Hot Shutdown within the next 12 hours.
4.	Reactor Coolant System Subcooling	1	1	If the operability of a subcooling monitor cannot be restored or a backup monitor made functional within 48 hours, be in at least Hot Shutdown within the next 12 hours.
5.	Auxiliary Feedwater Flow Rate*	1	1	If the operability of the auxiliary feedwater flow rate indicator cannot be restored within 48 hours, be in hot shutdown within 12 hours.
6.	Control Rod Misalignment as Monitored by On-Line Computer	1	1	Log individual rod positions once/hr., after a load change >10% of full power or after >30 inches of control motion.

*Applies to presently installed combination of auxiliary feedwater pump discharge flow indicators and auxiliary feedwater flow to steam generator indicators.

A.2 Under abnormal conditions including Black Plant startup, one reactor may be made critical providing the following conditions are met:

a. One 345 KV transmission line is in service; or the gas turbine is operating.

b. The 345/13.8 KV and the 13.8/4.16 KV station auxiliary transformers associated with the unit to be taken critical are in service; or the associated 13.8/4.16 KV station auxiliary transformer is in service and the gas turbine is operating.

c. Reactor power level is limited to 50% FULL power until 2 or more transmission lines are restored to service.

d. 480 Volt buses B03 and B04 for the unit to be taken critical are energized.

e. 4160 Volt buses A03, A04, A05, and A06 for the unit to be taken critical are energized.

f. A fuel supply of 11,000 gallons is available; and both diesel generators are operable.

g. Both batteries and DC systems are operable.

B.1 During power operation of one or both reactors, the requirements of 15.3.7.A.1 may be modified to allow the following arrangements of systems and components:

a. If the 345 KV lines are reduced to only one, any operating reactor(s) must be promptly reduced to, and limited to, 50% FULL power. If all 345 KV lines are lost, any operating reactor(s) will be reduced to supplying its auxiliary load, until one or more 345 KV transmission lines are again available.

b. If both 345/13.8 KV auxiliary transformers are out of service and only the gas turbine is operating, only one reactor will remain operating and it will be limited to 50% FULL power. The second reactor will be placed in the hot shutdown condition.

c. If the 13.8/4.16 KV auxiliary transformers are reduced to only one, the reactor associated with the out of service transformer must be placed in the hot shutdown condition.

d. Either bus A03 or A04 may be out of service for a period not exceeding 7 days provided both diesel generators are operable and the associated diesel generator is operating and providing power to the engineered safeguard bus normally supplied by the out of service bus.

e. One diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure operability and the engineered safety features associated with this diesel generator shall be operable.

f. One battery may be inoperable for a period not exceeding 24 hours provided the other battery and two battery chargers remain operable with one charger carrying the DC load of the inoperable battery's DC supply system.

Basis

This two unit plant has four 345 KV transmission line interconnections. A 20 MW gas turbine generator and two 2850 KW diesel generators are installed at the plant. All of these energy sources will be utilized to provide depth and reliability of service to the Engineered Safeguards equipment through redundent station auxiliary power supply systems.

If only one 345KV transmission line is in service to the plant switchyard, a temporary loss of this line would result in a reactor trip(s) if the reactor(s) power level were greater than 50% FULL power. Therefore, in order to maintain continuity of service and the possibility of self-sustaining operations, if only one 345KV transmission line is in service to any operation reactor(s), the power level of the affected reactor(s) will be limited to 50% FULL power.

If both 345/13.8KV station auxiliary transformers are out of service, only one reactor will be operated. The gas turbine will be supplying power to operate the safeguards auxiliaries of the operating reactor and acts as a backup supply for the unit's normal auxiliaries. Therefore, to prevent overloading the gas turbine in the event of a reactor trip, the maximum power level for the operating reactor will be limited to 50% FULL power. These conservative limits are set to improve transmission system reliability only and are not dictated by safety system requirements.

References

FSAR Section 8

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and to core power distribution limits.

Objective

To insure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical rod cluster control assembly (RCCA) ejection, and (3) an acceptable core power distribution during power operation.

Specification

A. Bank Insertion Limits

1. When the reactor is critical, except for physics tests and control rod exercises, the shutdown banks shall be fully withdrawn.
2. When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. Exceptions to the insertion limit are permitted for physics tests and control rod exercises.
3. The shutdown margin shall exceed the applicable value as shown in Figure 15.3.10-2 under all steady-state operating conditions from 350°F to FULL power. An exception to the stuck RCCA component of the shutdown margin requirement is permitted for physics tests.
4. Except for physics tests a shutdown margin of at least 1% Δk/k shall be maintained when the reactor coolant temperature is less than 350°F.
5. When the reactor is in the hot shutdown condition or during any approach to criticality, except for physics tests, the critical rod position shall not be lower than the insertion limit for zero power. That is, if the control rods were withdrawn in normal sequence with no other reactivity change, the reactor would not be critical until the control banks were above the insertion limit.

B. Power Distribution Limits

1. a. Except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \le \frac{(2.32)}{P} \times K(Z)$ for $P > .5$
$F_Q(Z) \le 4.64 \times K(Z)$ for $P \le .5$
RCS Total Flowrate $\ge$ 178000

$F_Q(Z) \le \frac{(2.52)}{P} \times K(Z)$ for $P > .5$
$F_Q(Z) \le 5.04 \times K(Z)$ for $P \le .5$
RCS Total Flowrate < 178000

$F^N_{\Delta H} \le 1.58 \times \{1 + 0.2\ (1-P)\}$

Where P is the fraction of FULL power at which the core is operating, K(Z) is the function in Figure 15.3.10-3 and Z is the core height location of F_Q.

b. Following a refueling shutdown prior to exceeding 90% of FULL power and at effective FULL power monthly intervals thereafter, power distribution maps using the moveable incore detector system shall be made to confirm that the hot channel factor limits are satisfied. The measured hot channel factors shall be increased in the following way:

(1) The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

(2) The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error.

c. If a measured hot channel factor exceeds the FULL power limit of Specification 15.3.10.B.1.a, the reactor power and power range high setpoints shall be reduced until those limits are met. If subsequent flux mapping cannot, within 24 hours, demonstrate that the full power hot channel factor limits are met, the overpower and overtemperature ΔT trip setpoints shall be similarly reduced and reactor power limited such that Specification 15.3.10.B.1.a above is met.

2. a. The target flux difference as defined in the basis shall be measured at least quarterly. A target flux difference update value shall be determined monthly by measurement, or by linear interpolation between the last measured value and 0% at end of cycle life (that is when the boron concentration in the coolant is zero ppm), or by extrapolation of the last three measured points. The target flux difference and its associated alarm setpoints need not be updated if the update value for FULL power target flux difference is within ±0.5% of the presently employed FULL power target flux difference value.

b. Except for physics testing, excore detector calibration (including recovery), or as modified below, the indicated axial flux difference shall be maintained within a range of +6 and -9 percent of the target flux difference. This is defined as the target band.

c. At a power level greater than 90 percent of FULL power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent of FULL power.

d. At a power level no greater than 90 percent of FULL power,

(1) The indicated axial flux difference may deviate from its +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% FULL power and increasing by -1% and +1% for each 2% of

FULL power below 90%. If the cumulative time exceeds one hour in any 24 hour period, then the reactor power shall be reduced immediately to no greater than 50% FULL power and the high neutron flux setpoint reduced to no greater than 55% of FULL power.

(2) A power increase to a level greater than 90% of FULL power is contingent upon the indicated axial flux difference being within its target band.

e. At a power level no greater than 50 percent of FULL power,

(1) The indicated axial flux difference may deviate from its target band.

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

f. Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 15.3.10.B.2.c or the flux difference-time requirement of 15.3.10.B.2.d(1). If the alarms are temporarily out-of-service, the axial flux difference shall be noted and conformance with the limits assessed every hour for the first 24 hours, and half-hourly thereafter.

3. Except for physics tests, whenever the indicated quadrant power tilt exceeds 2% the tilt condition shall be eliminated within two hours or the following actions shall be taken:

   a. Reduce core power level and the power range high flux setpoint two percent of rated values for every percent of indicated quadrant power tilt.

   b. If the tilt is not corrected within 24 hours, but the hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission. Return to FULL power is permitted, providing the hot channel factors are not exceeded.

   c. If the design hot channel factors for FULL power are exceeded or not determined within 24 hours, the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% FULL power for every percent of quadrant power tilt.

   d. The excore nuclear instrumentation system serves as the primary quadrant power tilt alarm. If the alarm is not functional for two hours, backup methods of assuring that the quadrant power tilt is acceptable shall be used. These methods include hand calculations, incore thermocouples using either a computer or manual calculations or incore detectors.

   e. When one power range channel is inoperable and thermal power is greater than 75% of FULL power, the quadrant power tilt shall be confirmed as acceptable by use of the movable incore detectors at least once per 12 hours.

C. Inoperable Rod Cluster Control Assembly (RCCA)

   1. An RCCA shall be considered inoperable if one or more of the following occurs:

D. Misaligned or Dropped RCCA

1. If the rod position indicator channel is functional and the associated RCCA is more than 7.5 inches indicated out of alignment with its bank and cannot be aligned when the bank is between 215 steps and 30 steps, then unless the hot channel factors are shown to be within design limits as specified in Section 15.3.10.B-1 within eight (8) hours, power shall be reduced to less than 75% of FULL power. When the bank position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the allowable indicated misalignment is 15 inches.

2. To increase power above 75% Full power with an RCCA more than 7.5 inches indicated out of alignment with its bank when the bank position is between 215 steps and 30 steps, an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 15.3.10.B. When the bank position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the allowable indicated misalignment is 15 inches.

3. If it is determined that the apparent misalignement or dropped RCCA indication was caused by rod position indicator channel failure, sustained power operation may be continued if the following conditions are met:

   a. For operation between 10% power and FULL power, the position of the RCCA(s) with the failed rod position indicator channel(s) will be checked indirectly by core instrumentation (excore detectors, and/or thermocouples, and/or moveable incore detectors) every shift and after associated bank motion exceeding 24 steps in one direction.
   b. For operation below 10% of FULL Power, no special monitoring is required.

E. RCCA Drop Times

1. At operating temperature and full flow, the drop time of each RCCA shall be no greater than 1.8 seconds from the loss of stationary gripper coil voltage to dashpot entry.

anomalies which would, otherwise, affect these bases.

Axial Power Distribution

The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the FULL power equilibrium value of axial offset (axial offset = ΔI/fractional power).

The FULL power target flux difference is defined as that indicated flux difference of the core in the following condition: equilibrium xenon (little or no oscillation) and with the full-length rod control rod bank more than 190 steps withdrawn (i.e., the normal full power position). Values for all other core power levels are obtained by multiplying the FULL power value by the factional power. At zero power the target flux differance is 0%. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of +6 and -9 percent ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides three methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during reduced power operation. This is because xenon distribution control at reduced power is not as significant as the control at FULL power and allowance has been made in predicting the heat flux peaking factors for less strict control at reduced power. Strict control of the flux difference is not possible during certain physics tests or during required periodic excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations. This is acceptable due to the increased core monitoring performed as part of the tests and low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band. However, to simplify the specification for operation up to 90% of FULL power, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This insures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band.

For normal operation and anticipated transients, the core is protected from overpower and minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a pre-condition; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant Tilt

The excore detectors are somewhat insensitive to disturbances near the core center such as misaligned inner control rods. It is therefore possible that a five percent tilt might actually be present in the core when the excore detectors respond with a two percent indicated quadrant tilt. On the other hand, they are overly responsive to disturbances near the periphery.

in a deliberate manner without undue pressure on the operating personnel because of the unusual techniques to be used to accommodate the reactivity changes associated with the shutdown.

Misaligned RCCAS

The various control rod banks (shutdown banks and control banks, A, B, C, and D) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Direct information on rod position indication is provided by two methods: A digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position. The rod position indicator channel has a demonstrated accuracy of 5% of span ($\pm$7.2 inches). Therefore, an analysis has been performed to show that a misalignment of 15 inches cannot cause design hot channel factors to be exceeded. A single fully misaligned RCCA, that is, an RCCA 12 feet out of alignment with its bank, does not result in exceeding core limits in steady-state operation at power levels less than or equal to rated power. In other words, a single dropped RCCA is allowable from a core power distribution viewpoint. If the misalignment condition cannot be readily corrected, the specified reduction in power to 75% of FULL power will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The eight (8) hour permissible limit on rod misalignment at rated power is short with respect to the probability of an independent accident. Because the rod position indicator system may have a 7.5 inch error when a misalignment of 15 inches is occurring, the Specification allows only a 7.5 inch indicated misalignment. However, when the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the consequences of a misalignment are much less severe. The differential worth of an individual RCCA is less, and the resultant purturbation on power distributions is less than when the bank is in its high differential worth region. At the top and bottom of the core, an indicated 15 inch misalignment may be representing an actual misalignment of 22.5 inches.

The failure of an LVDT in itself does not reduce the shutdown capability of the rods, but it does reduce the operator's capability for determining the position of that rod by direct means. The operator has available to him the excore detector recordings, incore thermocouple readings and periodic incore flux traces for indirectly determining rod position and flux tilts should the rod with the inoperable LVDT become malpositioned. The excore and incore instrumentation will not necessarily recognize a misalignment of 15 inches because the concommitant increase in power density will normally be less than 1% for a 15 inch misalignment. The excore and incore instrumentation will, however, detect any rod misalignment which is sufficient to cause a significant increase in hot channel factors and/or any significant loss in shutdown capability. The increased surveillance of the core if one or more rod position indicator channels is out-of-service serves to guard against any significant loss in shutdown margin or margin to core thermal limits.

The history of malpositioned RCCA's indicates that in nearly all such cases, the malpositioning occurred during bank movement. Checking rod position after bank motion exceeds 24 steps will verify that the RCCA with the inoperable LVDT is moving properly with its bank and the bank step counter. Malpositioning of an RCCA in a stationary bank is very rare, and if it does occur, it is usually gross slippage which will be seen by external detectors. Should it go undetected, the time between the rod position checks performed every shift is short with respect to the probability of occurrence of another independent undetected situation which would further reduce the shutdown capability of the rods.

Any combination of misaligned rods below 10% FULL power will not exceed the design limits. For this reason, it is not necessary to check the position of rods with inoperable LVDT's below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

15.3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability:

Applies to the operability of the movable detector instrumentation system.

Objective:

To specify functional requirements on the use of the in-core instrumentation systems for the recalibration of the excore axial off-set detection system.

Specification:

A. A minimum of 2 thimbles per quadrant and sufficient movable in-core detectors shall be operable during re-calibration of the excore axial off-set detection system.

B. Power shall be limited to 90% of FULL power if the calibration requirements for excore axial off-set detection system, identified in Table 15.4.1-1, are not met.

Basis:

The Movable In-Core Instrumentation System[1] has four drives, four detectors, and 36 thimbles in the core. The A and B detectors can be routed to eighteen thimbles. The C and D detectors can be routed to twenty-seven thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the ex-core detectors.

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

ATTACHMENT A

SAFETY EVALUATION FOR

REDUCED THERMAL DESIGN

FLOW STUDY

POINT BEACH NUCLEAR PLANT

UNIT 1

I. INTRODUCTION AND PURPOSE

This safety evaluation has been performed to address the non-LOCA safety considerations in allowing Point Beach Unit No. 1 to operate with significant steam generator tube plugging. Tube plugging in sufficient numbers may result in three effects:

- Reactor coolant flow is reduced due to increased steam generator flow resistance.
- The primary flow and steam generator heat transfer area are reduced. Thus to maintain guaranteed steam flow, T_{avg} must be increased or steam pressure reduced.
- Primary reactor coolant mass inventory is reduced.

The impact of higher steam generator tube plugging levels of up to 24 percent on the non-LOCA accident analyses presented in Chapter 14 of the FDSAR has been assessed. The basic approach used was to identify the important parameters for each accident, determine which of these parameters were affected by the higher steam generator tube plugging levels, and then determine how the impacted parameters affected the accident analysis. The resulting impacts were determined by either evaluating the accident to qualitatively demonstrate that the accident is not limiting or by reanalyzing the affected accident (if the accident was found to be limiting or very sensitive to the impact of higher steam generator tube plugging levels). The evaluations were consistent with the following assumptions:

Maximum core thermal power, MWt	1381.8
Thermal design flow, gpm/loop	84,500
S.G. tube plugging level, percent	24
T_{avg} at 100 percent of maximum allowed power, °F	572.86
ΔT at 100 percent of maximum allowed power, °F	55.5
RCS pressure, psia	2000
$F_{\Delta H}$	1.58
$T_{no\,load}$, °F	547
F_Q maximum	2.52

II. ACCIDENT ANALYSIS

The impact of reduced power and flow with respect to operation at 2000 psia, on the non-LOCA accident analyses presented in the Point Beach FSAR has been assessed. In general, all of the transients are sensitive to initial power level, steady state primary flow, and changes in system temperature and pressure. A study was made of each currently applicable accident analysis to identify margins to safety limits which could be used to offset penalties due to reduced primary flow. Reduction in system power is a benefit in DNB calculations and more than offsets the flow and T_{avg} (relative to reduced power) penalties.

The most recently applicable analysis used in this report is indicated by the reference number after each title.

Uncontrolled RCCA Withdrawal From a Subcritical Condition[1]

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the negative reactivity feedback of the Doppler power coefficient. The power excursion causes a heatup of the moderator. However, since the power rise is rapid and is followed by an immediate reactor trip, the moderator temperature rise is small. Thus, nuclear power response is primarily a function of the Doppler power coefficient.

The reduction in primary coolant flow is the primary impact which influences this accident. The reduced primary coolant flow results in a decreased core heat transfer coefficient which in turn results in a faster fuel temperature increase than reported in the most recent analysis.[1] The fast temperature increase would result in more Doppler feedback thus reducing the nuclear power heat flux excursion, as presented in Reference 1, which would partially compensate for the flow reduction. Therefore, the nuclear transient is only moderately sensitive to the impact of steam generator tube plugging.

The most recent analysis(1) shows that for a 90 x 10^{-5} Δk/sec reactivity insertion rate, the peak heat flux achieved is 76 percent of nominal. This is conservative for the higher plugging situation for the reasons stated above. The resultant peak fuel average temperature was 772°F. A 5 percent reduction in flow and the associated reduction in core heat transfer coefficient would degrade heat transfer from the fuel by a maximum 5 percent and increase the rise in peak fuel and clad temperature by a maximum of 5 percent. Therefore, the fuel and clad temperatures would be less than ~784°F and ~617°F, respectively, for the present evaluation. These values are still significantly below fuel melt (4900°F) and zirconium-H_2O reaction (1800°F) limits, and the impact of increased steam generator tube plugging, up to 24 percent would not result in a violation of safety limits.

Uncontrolled RCCA Withdrawal at Power(2)

An uncontrolled control rod assembly withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature (Section 14.1.2 of FSAR). Reduced flows result in less margin to DNB. Reduced thermal power results in more margin to DNB. In addition, the reduced primary flow will increase loop transit time which could require new values of lead/lag time constants to be determined for the overtemperature ΔT set point equation. Thus to assure adequate core protection the Reactor Core Thermal and Hydraulic Safety Limits have been recalculated consistent with the reduction in RCS flow and thermal power. The resulting overtemperature ΔT protection limits were essentially unchanged. Based on the current overtemperature ΔT limits, new core limits, reduced RCS flow and reduced rated power, the accident has been reanalyzed to verify the adequacy of protection setpoints and the lead/lag time constants.

Method of Analysis

The transient was reanalyzed employing the same digital computer code and assumptions regarding initial conditions and instrumentation and setpont errors used for the FSAR, including:

1. Power levels equal to 102 percent, 62 percent, and 12 percent of 1381.8 MWT;

2. Average temperature 4°F above T_{avg} corresponding to the initial power level;

3. Pressure (1970 psia) 30 psi below nominal;

4. Reactor trip on high nuclear flux at 118 percent of 1381.8 MWT with trip delay of 0.5 seconds; and

5. The setpoints for the overtemperature ΔT reactor trip function are those which presently appear in the Technical Specification currently for 2000 psia operation, with allowances for instrumentation errors. A trip delay time of 2.0 seconds was used.

6. Nominal flow is 84,500 gpm/loop.

7. No credit is taken for the high pressurizer water level and high pressure reactor trips.

Results

Figures 1 through 3 show the minimum DNBR as a function of reactivity insertion rate for 102 percent, 62 percent and 12 percent of full power.

Conclusions

These results demonstrate that the conclusions presented in the FSAR are still valid. That is, the core and reactor coolant system are not adversely affected since nuclear flux and overtemperature ΔT trips prevent the minimum DNB ratio from falling below 1.30 for this incident. Thus the current setpoint equation and reduction in rated power compensate for the reduction in thermal design flow.

Malpositioning of the Part Length Rods[2]

A malposiitiong of a part length rod accident need not be addressed since the part length rods have been removed from the core.

Rod Cluster Control Assembly (RCCA) Drop[3]

The drop of a Control Rod Assembly results in a step decrease in reactivity which produces a similar reduction in core power, thus reducing the coolant average temperature. The highly negative moderator temperature coefficient (-40 pcm/°F) assumed in the analysis results in a power increase (overshoot) above the turbine power runback value causing a temporary imbalance between core power and secondary power extraction capability. The effect of 5 percent reduction in initial RCS flow would be a smaller reduction in coolant average temperature and less of a power overshoot. Statepoints were evaluated consistent with a 5 percent reduction in flow and a 9 percent reduction in power. The reduction in power results in additional DNB margin. The resulting DNB evaluation showed that the DNBR limit of 1.30 can be more than accommodated with margin in the current cycle. The conclusions in the FSAR remain valid.

Chemical and Volume Control System Malfunction[2]

For a boron dilution incident during refueling or startup, while the reactor is subcritical, Section 14.1.4 of the FSAR shows that the operator has sufficient time to identify the problem and terminate the dilution before the reactor becomes critical. Tube plugging has no effect on the analysis at refueling conditions or cold shutdown conditions since only the reactor vessel and RHR system volumes are considered. For a dilution during startup, the effective volume of primary coolant in the steam generator tubes has been reduced by ~24% (~323 ft^3). Thus the volume of the reactor coolant (excluding the pressurizers) is reduced from 5253 ft^3 to 4930 ft^3. However, the minimum dilution time has been recalculated for refueling and startup assuming a minimum boron concentration of 1800 ppm, as opposed to 2000 ppm assumed in the FSAR. This will result in a shorter time to dilute to the maximum critical boron concentration of

1130 ppm at refueling and 1600 ppm at startup. The minimum time required for the reactor to become critical at refueling and startup has been calculated to be 74 minutes and 23.9 minutes respectively. Thus adequate time is available for the operator to recognize and terminate the dilution flow from refueling and startup conditions.

For dilution at power, it is necessary that the time to lose shutdown margin be sufficient to allow identification of the problem and termination of the dilution. As in the dilution during startup case, the RCS volume reduction due to steam generator tube plugging must be considered. The effective reactivity addition rate is a function of the reactor coolant temperature and boron concentration. The reactivity insertion rate calculated is based on a conservatively high value for the expected boron concentration at power (1400 ppm) as well as a conservatively high charging flow rate capacity (181.5 gpm). The reactor is assumed to have all rods out in either automatic or manual control. With the reactor in manual control and no operator action to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. After reactor trip there is at least 15.1 minutes for operator action prior to return to criticality. The boron dilution transient in this case is essentially the equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 1.15 pcm/sec and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Prior to reaching the Overtemperature ΔT reactor trip the operator will have received an alarm on Overtemperature ΔT and turbine runback.

With the reactor in automatic control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in rod insertion limit and axial flux alarms. The minimum time to lose the 1 percent $\Delta k/k$ shutdown margin required at beginning-of-life would be greater than 15.1 minutes. The time would be significantly longer at end-of-life due to the low initial boron concentration and 2.77 percent $\Delta k/k$ shutdown margin.

Rupture of a Steam Pipe[2]

The steamline break transient is analyzed for hot zero power, end of life conditions (Section 14.2.5 of the FSAR) for the following cases:

- Hypothetical Break (steam pipe rupture)

  Inside Containment with and without power

  Outside Containment with and without power

- Credible Break (Safety valve opening)

A steamline break results in a rapid depressurization of the steam generators which causes a large reactivity insertion to the core via primary cooldown. The acceptance criteria for this accident is that no DNB must occur following a return to power. This limit, however, is highly conservative since steam line break is classified as a Condition IV event. As such, the occurrence of DNB in small regions of the core (~5 percent) would not violate NRC acceptance criteria.

The impact of increased levels of steam generator tube plugging would affect the accident principally due to the reduced flow, reduced RCS inventory, and reduced heat transfer coefficient. These impacts would result in changed cooldown and feedback reactivity characteristics such that the return to power as shown in the previous analysis would be slightly conservative with respect to the lower initial flow conditions. In addition, the time of Safety Injection actuation would be unaffected by flow conditions for the Hypothetical Breaks. This coupled with the slightly slower return to power would result in a reduction in peak average power for the cases with and without power and indicate results conservative with respect to the current analysis.

However, as this is a limiting accident with respect to available DNB margin at reduced pressure, the limiting cases were reanalyzed and limiting statepoints evaluated.

Method of Analysis

Analysis methods and assumptions used in the reanalysis were consistent with those employed in the most recent safety analysis. These assumptions included:

1) Minimum shutdown margin equal to 2.77 percent.

2) The most negative moderator temperature coefficient for the rodded core at end of life.

3) The rod having the most reactivity stuck in its fully withdrawn position.

4) One train of safety injection fails to function as designed.

Results

The minimum value of the DNBR for the hypothetical breaks was greater than the 1.30 limit. Results for the credible break confirmed that the core remained subcritical throughout the transient. Table I presents the core parameters for the 4 hypothetical break cases used in DNB evaluations. Figures 4 through 7 present the transient results for those cases summarized in Table I. Figure 8 presents the transient results for the credible break.

Conclusions

The steamline rupture accident has been shown to meet the DNB design basis for the hypothetical breaks and remains subcritical for the credible breaks for the 24 percent tube plugging.

Startup of an Inactive Reactor Coolant Loop[2]

Startup of an idle reactor coolant pump results in the injection of relatively cold water into the core. This accident need not be addressed due to Technical Specifications restrictions which prohibit power operation with a loop out of service. However, a brief discussion of the impact of the new operating conditions is included. The analysis in FSAR Section 14.1.5 shows that the reactor does not trip and reaches a peak power of 21 percent full power. The lower loop flow would result in a slightly lower reactivity insertion rate, resulting in a lower peak heat flux. Therefore results presented in the FSAR would be conservative.

Reduction in Feedwater Enthalpy Incident[2]

The addition of excessive feedwater and inadvertent opening of the feedwater bypass valve are excessive heat removal incidents which result in a power increase due to moderator feedback. Increased levels of steam generator tube plugging would impact this analysis principally due to the reduced flow.

Section 14.1.6 of the FSAR presents two cases. The first case assumes a zero moderator coefficient, which is used to demonstrate inherent transient attenuation capability during a feedwater reduction. A reduction in flow will have a negligible effect on stability since the reactivity insertion is identical to the FSAR case due to the zero moderator temperature coefficient. DNB is not a consideration for this case since DNBR's do not fall below the steady state value. This is due to the relatively large reduction in T_{avg}. The reduction in flow is more than compensated by the reduction in nominal power, resulting in an increase in the initial steady state DNBR. In addition, as discussed in the FSAR, the reactor will trip on low pressure trip.

The second case assumes a large negative moderator coefficient. The reduction in thermal design flow will result in a slower cooldown, and therefore the reactivity insertion rate will be less than in the FSAR analysis. The integral reactivity insertion due to moderator temperature reduction will be less than the FSAR case, thus producing a lower peak nuclear power. The reduction in nominal power results in a net increase in steady state DNBR.

Protection for this accident is provided by the overpower-overtemperature ΔT protection. The adequacy of this protection was verified in the rod withdrawal at power reanalysis.

Excessive Load Increase Incident[2]

An excessive load increase incident is defined as a rapid increase in steam generator flow that causes a power mismatch between the reactor core power and the steam generator load demand. Four cases were analyzed in the FSAR, Section 14.1.7. A 10 percent step load increase was analyzed for manual and automatic control, at beginning of life (BOL) and end of life (EOL). As in the Feedwater Malfunction Accident, reduced flow is the principal impact on this accident due to increased levels of steam generator tube plugging.

The worst case results (automatic control-EOL) indicate that with no trip actuation, steady state conditions are reached with a minimum DNBR of > 1.30. The reduction in thermal design flow will result in a slower cooldown, and therefore a lower reactivity insertion rate. The integral reactivity insertion due to moderator temperature will be less than the FSAR case, thus producing a lower peak nuclear power. The reduction in nominal power results in a net increase in steady state DNBR.

Protection for this accident is provided by the overpower-overtemperature ΔT protection. The adequacy of this protection was verified in the rod withdrawal at power reanalysis.

Loss of Reactor Coolant Flow/Locked Rotor[2]

As demonstrated in the FSAR, Section 14.1.8, the most severe loss of flow transient is caused by the simultaneous loss of electrical power to both reactor coolant pumps. The reduced thermal power results in a net increase in initial steady state DNB ratio. The increased steam generator tube bundle resistance has an extremely small impact on the flow coastdown during the critical first few seconds of the transient. Therefore, the reactor trip on low reactor coolant flow will be generated at approximately the same time in the transient. With the higher initial DNB ratio and same time to trip, a loss of flow event from these conditions will result in more margin to the DNB limit. This was verified by evaluation of the statepoints consistent with a 5 percent reduction in flow and 9 percent reduction in power. The resulting DNB evaluation showed that the DNBR limit of 1.30 can be more than accommodated with the margin in the current cycle. The conclusions in the FSAR remain valid.

The FSAR shows that the most severe Locked Rotor Accident is an instantaneous seizure of a reactor coolant pump rotor at 100 percent power. Following the incident, reactor coolant system temperature rises until shortly after reactor trip.

The impact on the Locked Rotor Accident of increased steam generator tube plugging will be primarily due to the reduced flow. These impacts will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. The flow coastdown in the affected loop due to the Locked Rotor is so rapid that the time of reactor trip (low flow setpoint is reached) is essentially identical to most recent analyses. The nuclear power and heat flux responses will be somewhat lower due to reduced thermal power. The reduction in power also results in reduced initial hot spot values. This would offset the slight increase in fuel and clad temperatures effect of reduced flow. Consequently, the expected peak fuel and clad temperatures would remain the same as results of the currently applicable analysis.

It is estimated that the peak pressure will not increase above the previous value due to reduced power, however the maximum calculated value was 2778 psia based on 2250 psia operation plus 30 psia uncertainty. This is significantly below the pressure at which vessel stress limits are exceeded. In addition, this is conservative as noted by the conclusions of the FSAR. The 24 percent reduction in steam generator tubes would result in approximately a 8 percent reduction in primary mass which decreases the heat capacity of the RCS by the same amount. This would not result in higher peak temperatures or pressures since the peak values are reached in considerably less than one loop transport time constant.

Thus operation at reduced flow will not cause safety limits to be exceeded for a locked rotor accident.

Loss of External Electrical Load[2]

The result of a loss of load is a core power level which momentarily exceeds the secondary system power extraction causing an increase in core water temperature.

The impact of increased levels of steam generator tube plugging would be again principally due to the reduced flow and the decreased RCS mass inventory. Two cases, analyzed for both beginning and end of life conditions, are presented in Section 14.1.9 of the FSAR:

a. Reactor in automatic rod control with operation of the pressurizer spray and the pressurizer power operated relief valves; and

b. Reactor in manual rod control with no credit for pressurizer spray or power operated relief valves.

The FSAR analysis results in a peak pressurizer pressure of 2514 psia following reactor trip. A reduction in loop flow and RCS mass inventory will result in a more rapid pressure rise than is currently shown. The effect will be minor, however, since the reactor is tripped on high pressurizer pressure. Thus, the time to trip will be decreased which will result in a lower total energy input to the coolant.

The minimum transient DNBR evaluated at reduced pressures of 2000 psia minus 30 psia uncertainty (DNBR is more limiting at reduced pressure) will result in a net increase due to reduced power. However, this transient is bounded by the Uncontrolled Rod Withdrawal at Power transient.

Loss of Normal Feedwater/Station Blackout[2]

This transient is analyzed to determine that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that the pressurizer must not be filled with water. The effect of reducing initial core flow would be a larger and more rapid heatup of the primary system. The resulting coolant density change would increase the volume of water in the pressurizer. The analyses in FSAR Section 14.1.10 and 14.1.11 show that the peak pressurizer volume reached is 780 ft^3 on an approximate 250 ft^3 change in volume. This result was due to a ~ 26°F change in coolant average temperature. Using the highly conservative assumption that the average temperature would increase 50 percent due to flow reductions, this would result in a maximum increase of less than 125 ft^3 in liquid volume. This is still below the 1000 ft^3 capacity of the pressurizer, thus no reanalysis is necessary. In addition, due to the relatively long duration of the transient following trip, the results are highly sensitive to residual (decay) heat generation. Residual heat generation is directly proportional to initial power level preceding the trip. At reduced power, the total energy input to the system be likewise reduced.

Rupture of a Control Rod Drive Mechanism Housing, RCCA Ejection[1]

The rupture of a control rod mechanism housing which allowed a control rod assembly to be rapidly ejected from the core would result in a core thermal power excursion. This power excursion would be limited by the Doppler reactivity effect as a result of the increased fuel temperature and would be terminated by a reactor trip activated by high nuclear power signals.

The rod ejection transient is analyzed at full power and hot zero power for both beginning and end of life conditions (Section 14.2.6 of the FSAR). Reduced core flow is the primary impact resulting from increased levels of steam generator tube plugging. This impact would result in a reduction in heat transfer to the coolant which would increase clad and fuel peak temperatures.

Reanalysis was performed using the conservative ejected rod worths and post ejection peaking factors assumed in the latest analyses which are above the calculated Point Beach reload values. Reanalysis was performed to show that the increase in initial F_Q from 2.47 in the previous analysis to 2.52 is still acceptable.

Method of Analysis

Analysis methods and assumptions used in the reanalysis were consistent with those employed in the most recent analysis and FSAR 14.2.6. The calculation of the transient is performed in two stages, first an average core calculation and then a hot region calculation. The average core is analyzed to determine the average power generation with time including the various total core feedback effects, i.e. Doppler reactivity and moderator density reactivity. Enthalpy and temperature transients in the hot spot are determined by adding a multiple of the average core energy generation to the hotter rods and performing a transient heat-transfer calculation. The asymptotic power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. The DNB time is not calculated. DNB is conservatively assumed to occur near the start of the transient.

Results

The analysis results and inputs are summarized in Table II. The conditions at the hot spot fuel rod do not exceed the limiting fuel criteria[4]. The conclusions of the FSAR, therefore are still valid.

III Conclusions

To assess the effect of non-LOCA accident analyses on operation of Point Beach Unit 1, with significant levels of steam generator tube plugging, a safety evaluation was performed.

The transients and/or statepoints were analyzed for rod ejection, steamline break, boron dilution, dropped rod and loss of flow. In addition, an evaluation was performed to identify the effect of the reduced operating conditions (power, flow and pressure) on the remaining transients and to quantify margins available to affect penalties. Those accidents that are sensitive to higher operating pressures were addressed also. Based on this evaluation, operation at these reduced conditions and a maximum 24 percent effective steam generator tube plugging level will not result in violation of safety limits for the transients evaluated at either 2000 psia or 2250 psia operation.

REFERENCES

1. Davidson, S. L., Editor, "Reload Safety Evaluation Point Beach Nuclear Plant Unit 1, Cycle 9A," December 1980.

2. Final Safety Analysis Report Point Beach Nuclear Plant, Units Number 1 and 2.

3. Davidson, S. L. Editor, "Reload Safety Evaluation Point Beach Nuclear Plant Unit 2, Cycle 9A," May 1982.

4. Risher, D. H. Jr., "An evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP-7588, Revision 1-A, January, 1975.

TABLE I

CORE PARAMETERS USED IN STEAM LINE BREAK DNB ANALYSIS

	Outside Break With Power				Outside Break Without Power				Inside Break With Power				Inside Break Without Power			
Time (Sec.)	72.4	75.2	78.4	81.6	104.8	108.0	111.2	114.0	58.4	61.2	64.8	67.6	94.4	97.6	102.4	107.2
Core Inlet Temperature (°F)																
Loop A	403	401	399	398	361	357	355	352	356	355	353	351	295	292	288	285
Loop B	454	451	448	446	489	489	488	487	436	434	430	427	509	508	507	507
RCS Flow (percent of nominal)	100	100	100	100	10.1	9.8	9.6	9.4	100	100	100	100	10.9	10.6	10.2	9.8
Heat Flux (percent of 1381.8 Mwt)	21.8	22.9	24.2	18.5	10.4	10.5	10.8	10.5	39.3	41.0	42.9	34.6	17.0	17.2	17.3	16.9
RCS pressure (psia)	695	694	691	689	1039	1045	1053	1061	656	656	655	653	1008	1020	1027	1030

TABLE II

SUMMARY OF ROD EJECTION ANALYSIS PARAMETERS AND RESULTS

	BOL	BOL	EOL	EOL
Power Level, percent	0	102	0	102
Ejected rod worth, percent $\Delta\rho$	0.91	0.34	0.95	0.42
Delayed neutron fraction, percent	0.0049	0.0049	0.0046	0.0046
F_Q before rod ejection	2.52	-	2.52	-
F_Q after rod ejection	11.2	5.03	13.7	5.10
Number of operating pumps	1	2	1	2
Maximum fuel pellet center temperature, °F	3504	4543	3719	4458
Maximum fuel pellet average temperature, °F	3095	3492	3289	3392
Maximum clad average temperature, °F	2440	2129	2550	2071
Fuel Pellet Melting, percent	0.0	0.0	0.0	0.0
Maximum fuel enthalpy (btu/lb)	231.4	266.7	248.7	257.8

Figure 1
ROD WITHDRAWAL FROM 102% POWER

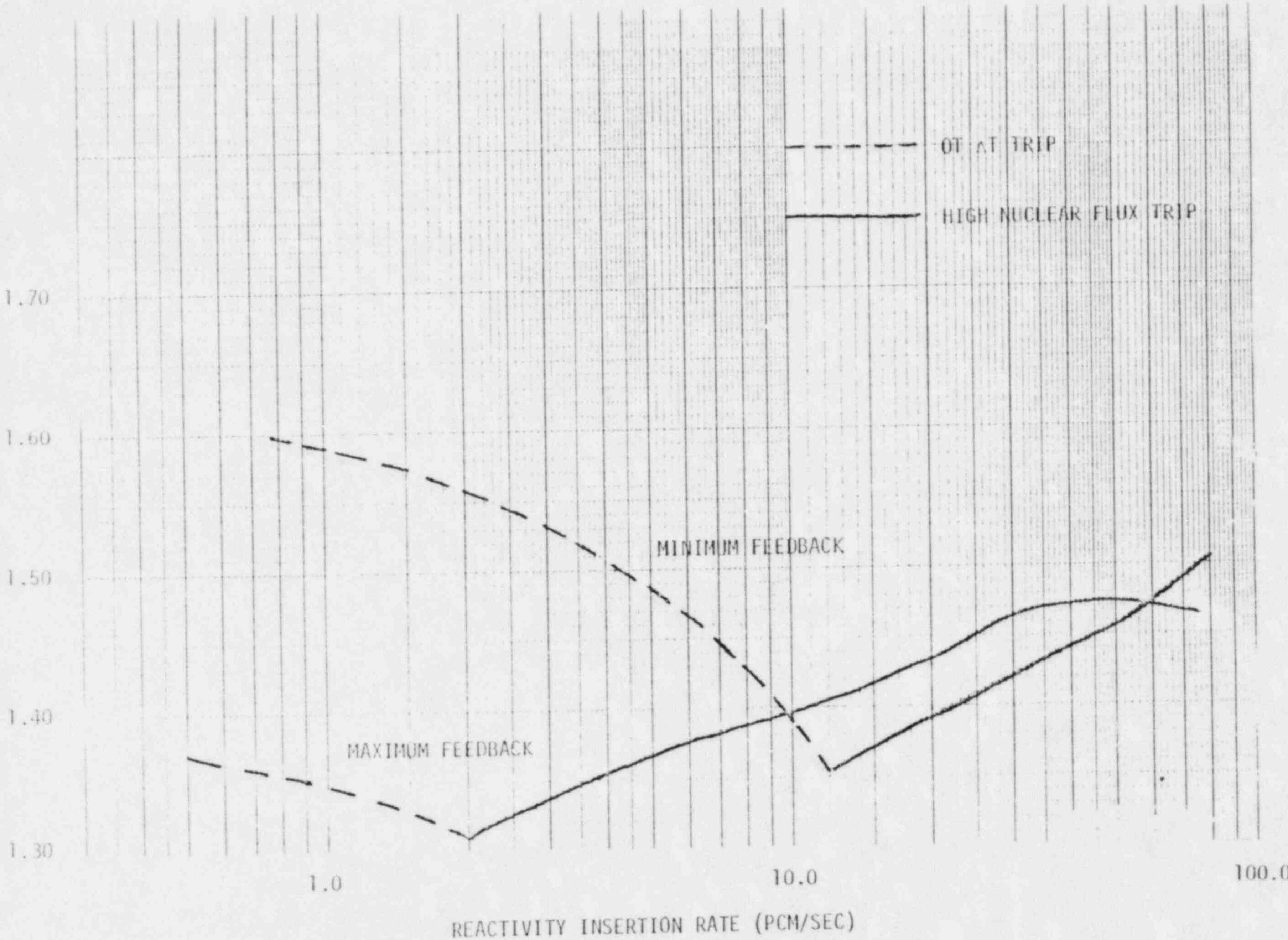


FIGURE 2

ROD WITHDRAWAL FROM 62% POWER

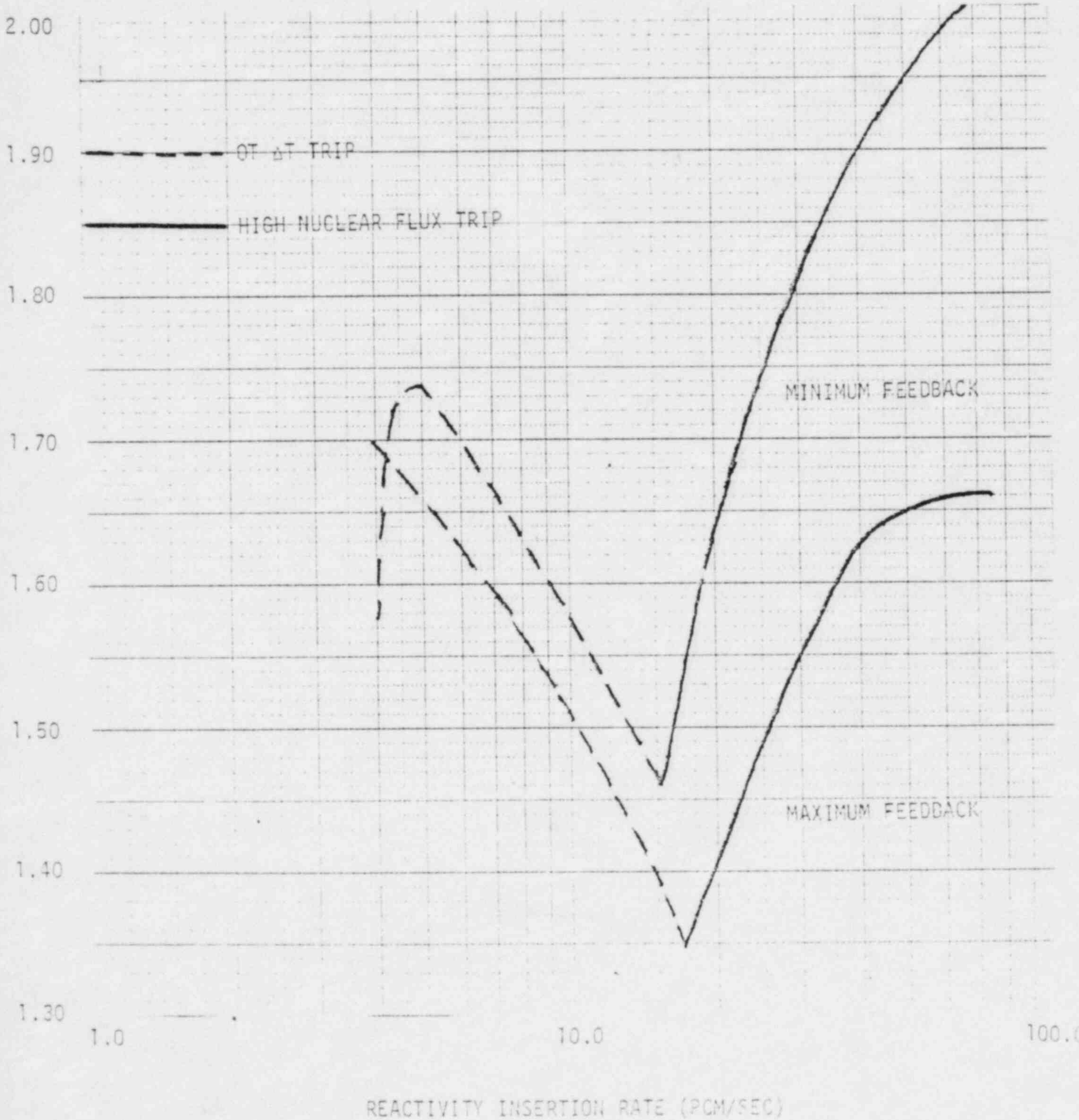
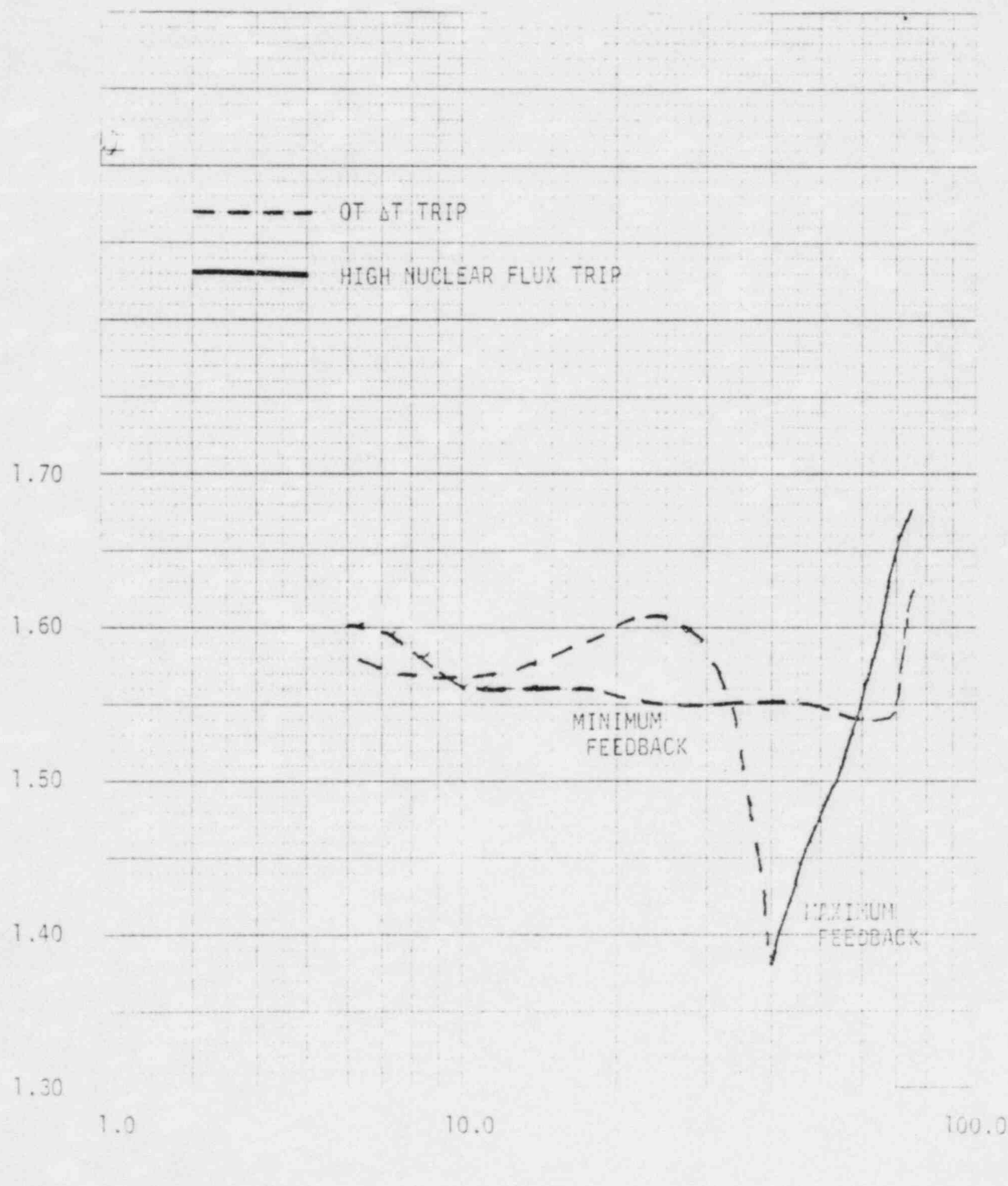


FIGURE 3

ROD WITHDRAWAL FROM 12% POWER



STEAMLINE BREAK OUTSIDE THE CONTAINMENT
(DOWNSTREAM OF FLOW MEASURING NOZZLE)
OUTSIDE POWER AVAILABLE

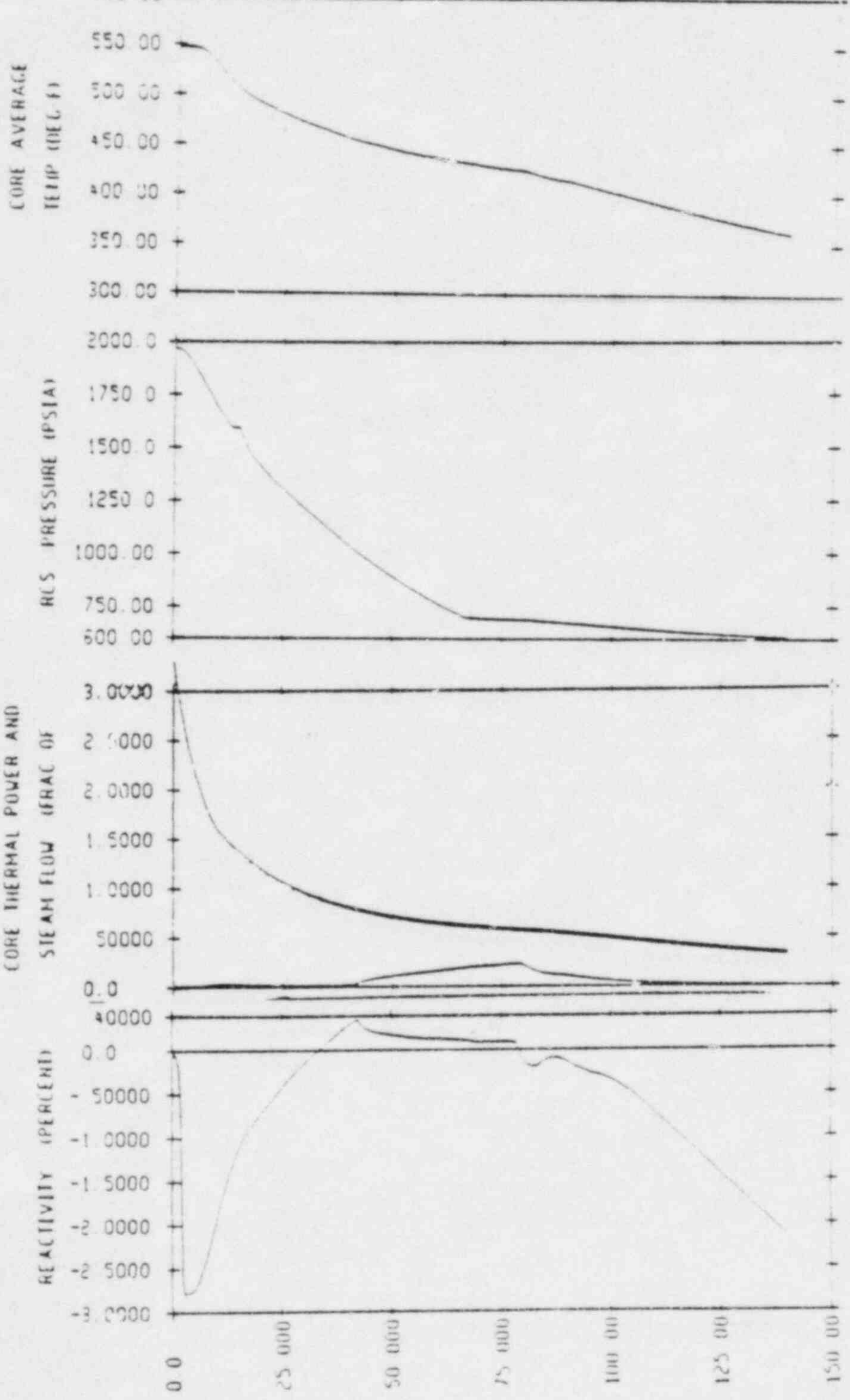


FIGURE 4

STEAMLINE INSIDE THE CONTAINMENT
(AT EXIT OF STEAM GENERATOR)
OUTSIDE POWER AVAILABLE

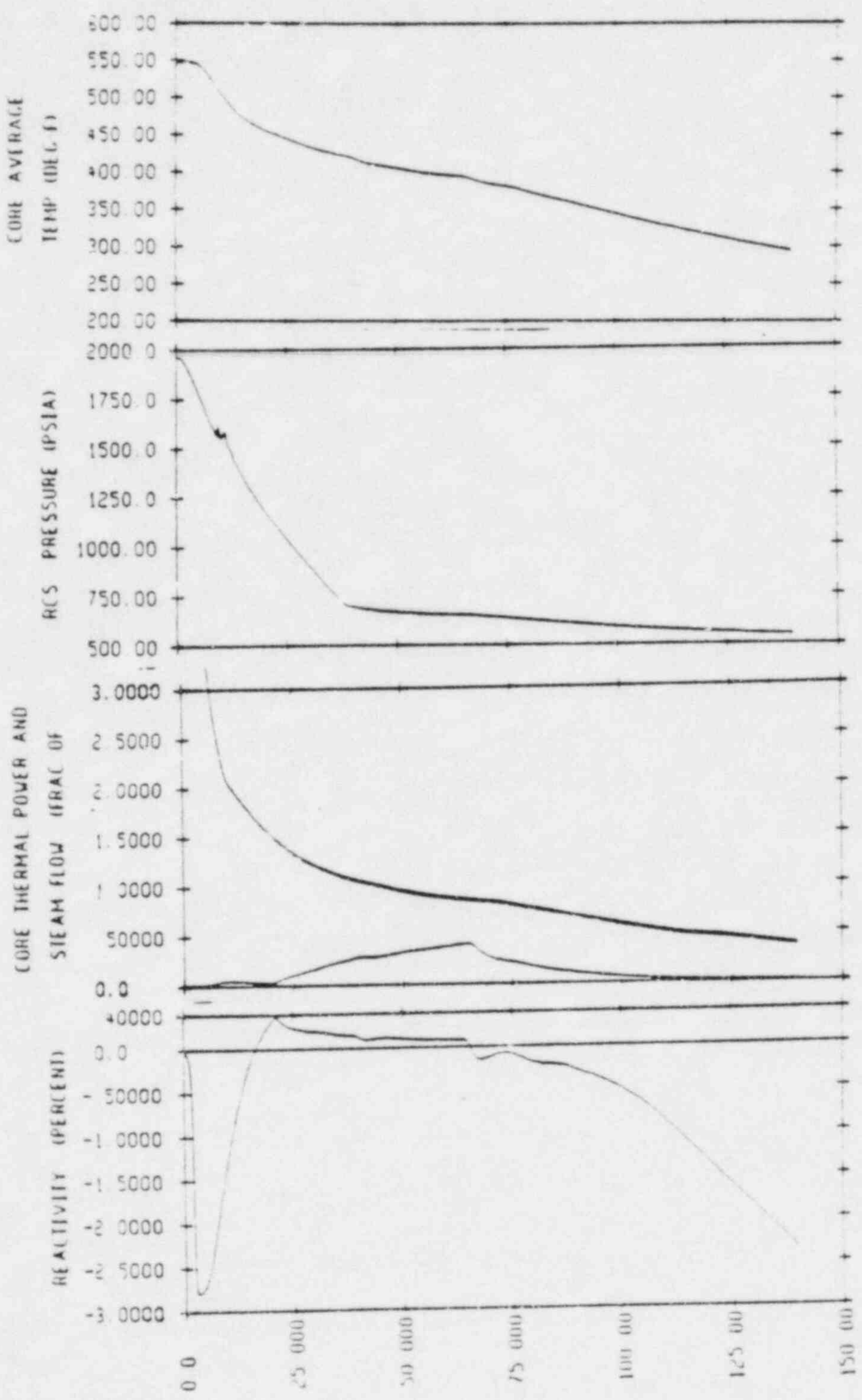


FIGURE 5

STEAMLINE BREAK OUTSIDE THE CONTAINMENT
(DOWNSTREAM OF THE FLOW MEASURING NOZZLE)
LOSS OF OUTSIDE POWER AT T = 0

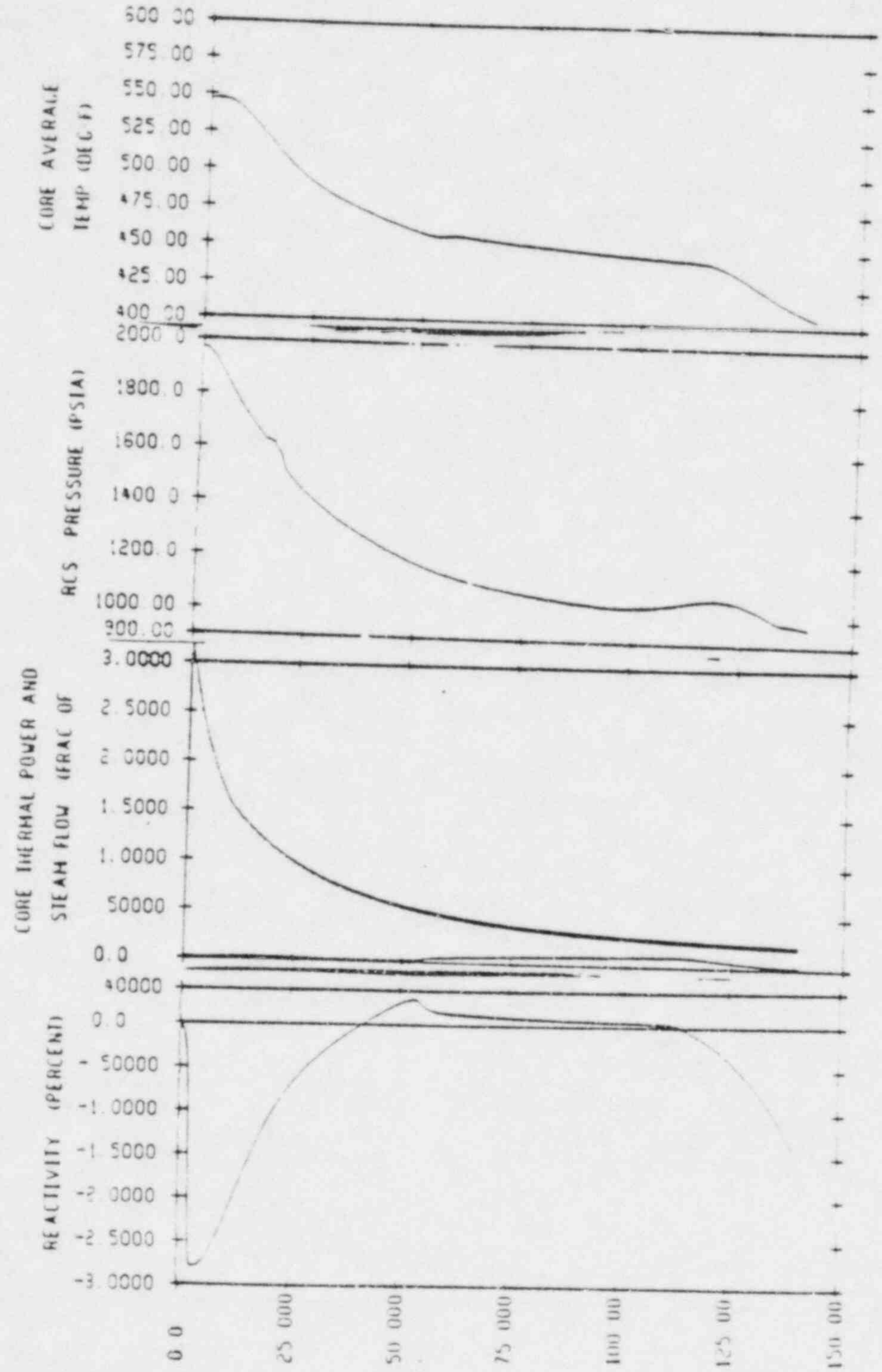


FIGURE 6

STEAMLINE BREAK INSIDE THE CONTAINMENT
(AT EXIT OF STEAM GENERATOR)
LOSS OF OUTSIDE POWER AT T = 0

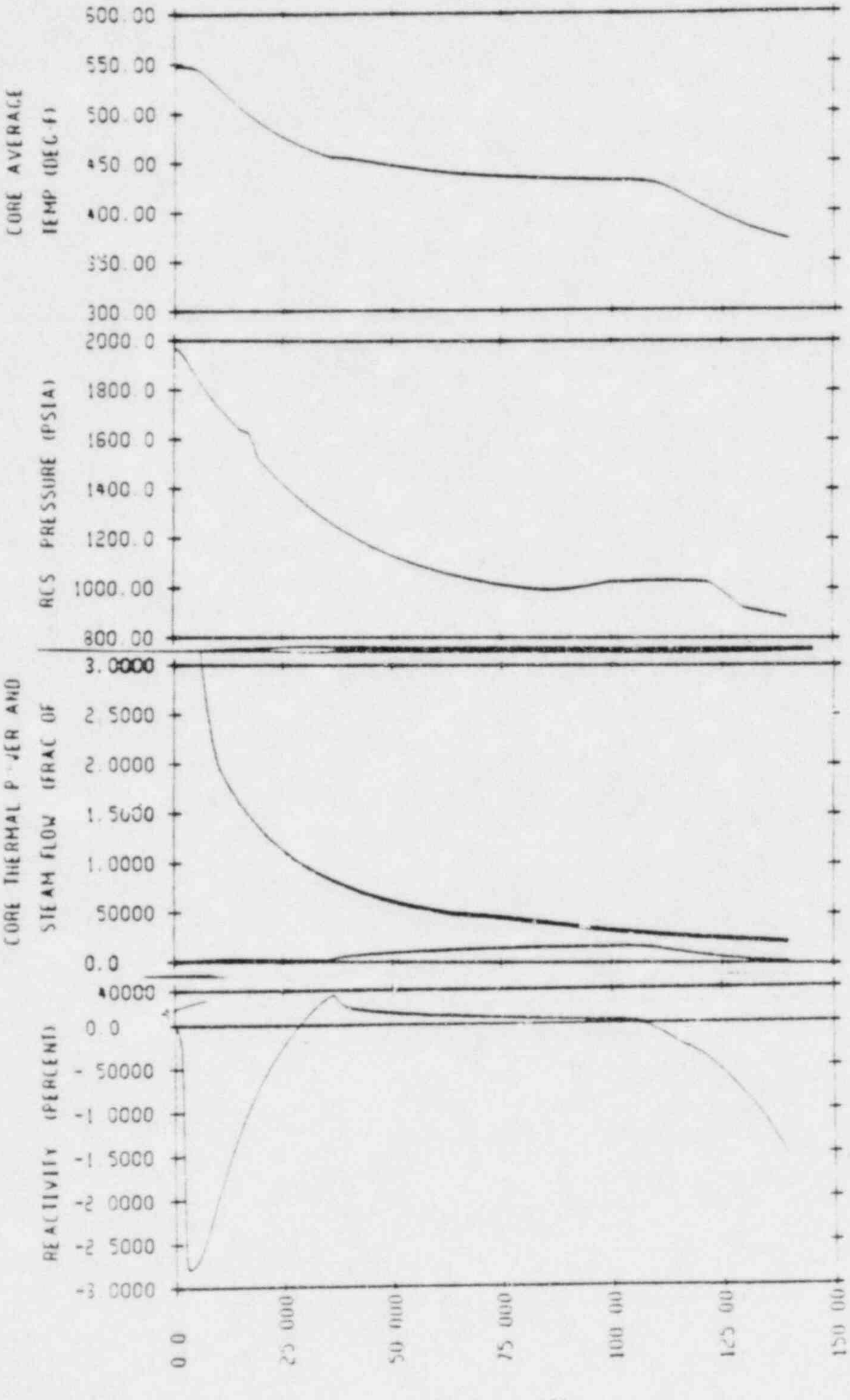


FIGURE 7

STEAM BREAK EQUIVALENT TO ONE
STEAM GENERATOR SAFETY VALVE

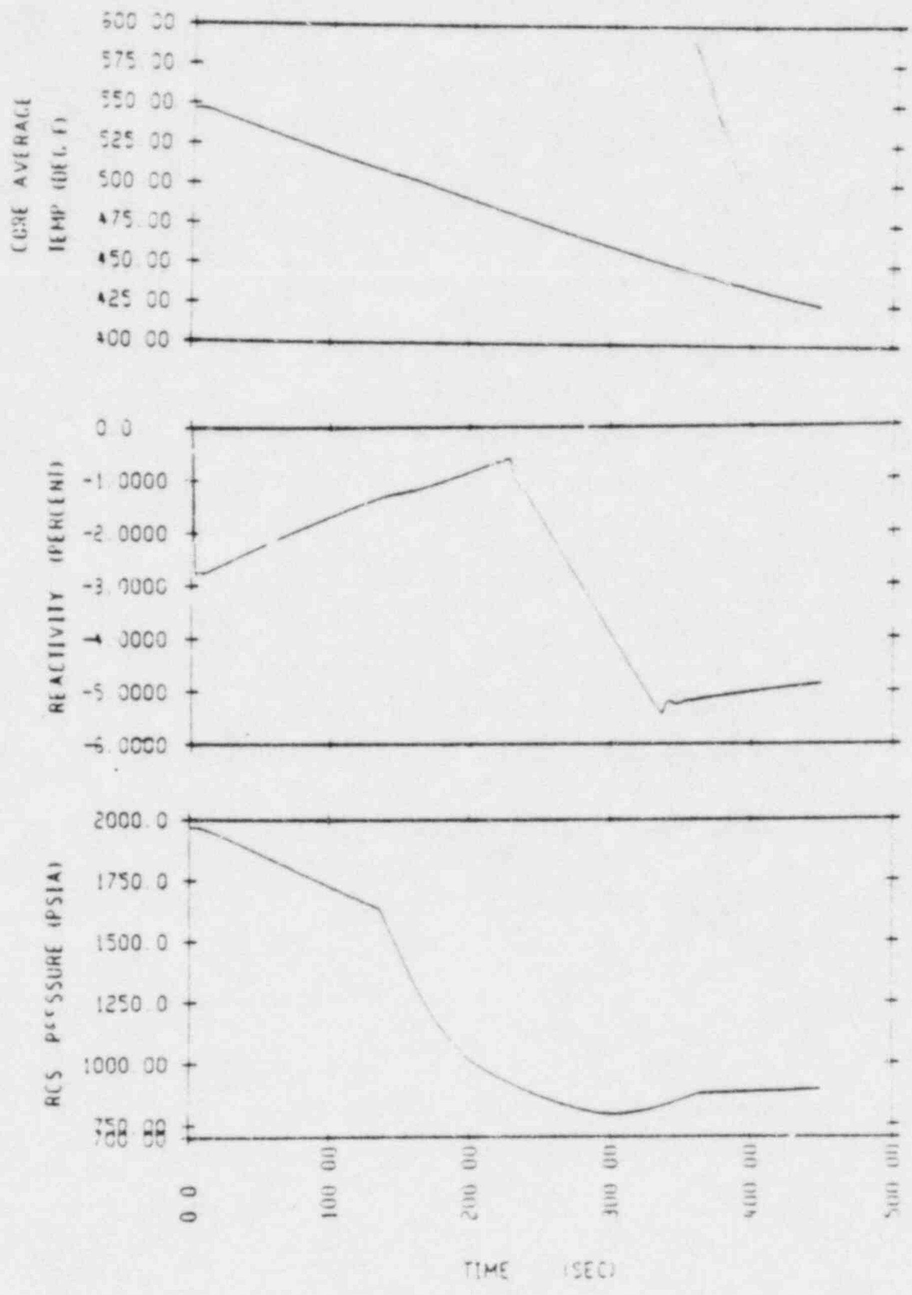


FIGURE 8