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3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

   a. The refueling water tank contains not less than 350,000 gal. of water with a boron concentration of at least 2000 ppm.

   b. The boron injection tank contains not less than 1000 gal. of a 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F. Two channels of heat tracing, shall be available for the flow path. Valves 1821 and 1831 shall be open and valves 1822A and 1822B shall be closed, except during short period of time when they can be cycled to demonstrate their operability.

c. The four accumulators are pressured to at least 600 psig and each contain a minimum of 734 ft^3 and a maximum of 749 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

d. Three safety injection pumps together with their associated piping and valves are operable.

e. Two residual heat removal pumps together with their associated piping and valves are operable.

f. Two recirculation pumps together with their associated piping and valves are operable.

g. Two residual heat exchangers together with their associated piping and valves are operable.

h. Valves 856A, C, D and E, in the discharge header of the safety injection header are in the open position. Valves 856B and F, in the discharge header of the safety injection header are in the closed position. The hot leg valves (856B and F) shall be blocked closed by de-energizing the valve motor operators.

i. The four accumulator isolation valves shall be blocked open by de-energizing the valve motor operators.

j. Valve 1810 on the suction line of the high-head SI pumps and valves 882 and 744, respectively on the suction and discharge line of the residual heat removal pumps, shall be blocked open by de-energizing the valve-motor operators.

2. During power operation, the requirements of 3.3.A-1 may be modified to allow any one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.A-1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.A-1 are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occuring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[10] The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-6 of the FSAR.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 85°F).[4] In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at a rate consistant with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, three charcoal filters (and their associated recirculation fans) in operation, along with one containment spray pump and sodium hydroxide addition, will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values. These constitute the minimum safeguards for iodine removal, and are capable of being operated on emergency power with one diesel generator inoperable.

If off-site power is available or all diesel generators are operating to provide emergency power, the remaining installed iodine removal equipment (two charcoal filters and their associated fans, and one containment spray pump and sodium hydroxide addition) can be operated to provide iodine removal in excess of the minimum requirements. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

One of the five fan cooler units is permitted to be inoperable during power operation. This is an abnormal operating situation, in that the normal plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical.

However, because of the difficulty of access to make repairs, it is important on occasion to be able to operate temporarily without at least one fan-cooler. Compensation for this mode of operation, is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

The Component Cooling System is different from the system discussed above in that the pumps are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident.(6) During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards.(7)

A total of six service water pumps are installed, only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident.(8)

During the second phase of the accident, one additional service water pump on the non-essential header will be manually started to supply the minimum cooling water requirements for the component cooling loop.

The limits for the accumulators, and their pressure and volume assure the required amount of water injection following a loss-of-coolant accident. The values used for the accident analyses[9,10] are based on this accumulator water inventory plus the water within the connecting piping between the accumulator and first check valve.

Two full rated recombination systems are provided in order to control the hydrogen evolved in the containment following a loss-of-coolant accident. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the containment. Each of the systems is separate from the other and is provided with redundant features. Power supplies for the blowers and ignitors and separate, so that loss of one power supply will not effect the remaining system. Hydrogen gas is used as the externally supplied fuel. Oxygen gas is added to the containment atmosphere through a separate containment feed to prevent depletion of oxygen in the air below the concentration required for stable operation of the combustor (12%). The containment atmosphere sampling system consists of a sample line which originates in each of the containment fan cooler units. The fan and sampling pump head together are sufficient to pump containment air in a loop from the fan cooler through a containment penetration to a sample vessel outside the containment, and then through a second penetration to the sample termination inside the containment. The design hydrogen concentration for operating the recombiner is established at 2% by volume. Conservative calculations indicate that the hydrogen content within the containment will not reach 2% by volume until 13 days after a loss-of-coolant accident. There is therefore no need for immediate operation of the recombiner following an accident, and the quantity of hydrogen fuel stored at the site will be only for periodic testing of the recombiners.

The cable tunnel is equipped with two temperature controlled ventilation fans. Each fan has a capacity of 21,000 cfm and is connected to a 480v bus. One fan will start automatically when the temperature in the tunnel reaches 95°F. The second fan will start if the temperature in the tunnel reaches 100°F. Under the worst conditions, i.e. loss of outside power and all the Engineered Safety Features in operation, one ventilation fan is

capable of maintaining the tunnel temperature below 104°F. Under the same worst conditions, if no ventilation fans were operating, the natural air circulation through the tunnel would be sufficient to limit the gross tunnel temperature below a tolerable value of 140°F. However, in order to provide for ample tunnel ventilation capacity, the two ventilation fans are required to be operable when the reactor is made critical. If one ventilation fan is found inoperable, the daily testing of the other fan will ensure that cable tunnel ventilation is available.

Valves 856A, C, D and E are maintained in the open position during plant operation to assure a flow path for high-head safety injection during the injection phase of a loss-of-coolant accident. Valves 856B and F are maintained in the closed position during plant operation to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, the valve motor operators are de-energized to prevent spurious opening of these valves.

References

(1) FSAR Section 9
(2) FSAR Section 6.2
(3) FSAR Section 6.2
(4) FSAR Section 6.3
(5) FSAR Section 14.3.5
(6) FSAR Section 1.2
(7) FSAR Section 8.2
(8) FSAR Section 9.6.1
(9) FSAR Section 14.3
(10) WCAP 8399 "ECCS Acceptance Criteria Analysis, Indian Point Nuclear Generating Station Unit No. 2" September 1974, Westinghouse Non-Proprietary Class 3

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objective:

To ensure:

1. Core subcriticality after reactor trip,
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification:

3.10.1 Shutdown Reactivity

To assure that sufficient control rod reactivity worth is available for trip to bring the reactor to the reactivity shutdown shown in Figure 3.10-1, rod insertion limits shall be constructed consistent with Specification 3.10.4.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq (2.63/P) \times K(Z) \text{ for } P > .5$$
$$F_Q(Z) \leq (5.26) \times K(Z) \text{ for } P \leq .5$$
$$F^N_{\Delta H} \leq 1.65\ [1 + 0.2\ (1-P)]$$

where P is the fraction of full power at which the core is operating. K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

3.10.2.2.1 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.

3.10.2.2.2 The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified under Item 3.2.10.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated value equal to the ratio of the F_Q or $F_{\Delta H}^N$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.3 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. The target flux difference must be updated each effective full power month by linear interpolation using the most recent measured value and a value of 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.8 below, the indicated axial flux difference shall be maintained within a $\pm$ 5% band about the target flux difference (defines the band on axial flux difference).

3.10.2.5 At a power level greater than 90% of rated power,

3.10.2.5.1 If the indicated axial flux difference deviates from its target band, either the deviation shall be eliminated, or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

3.10.2.6 At a power level no greater than 90 percent of rated power,

3.10.2.6.1 The indicated axial flux difference may deviate from its +5% target band for a maximum of one hour (cumulative) in ay 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power.

3.10.2.6.2 If Item 3.10.2.6.1 is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

3.10.2.6.3 A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.7 At a power level no greater than 50 percent of rated power,

3.10.2.7.1 The indicated axial flux difference may deviate from its target band provided the flux difference does not exceed an envelope bounded by -31 percent and +31 percent at 50% power and increasing by -1% and +1% for each 2% of rated power below 50% power.

3.10.2.7.2 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than one hour (cumulative) out of the preceding 24 hour period.

3.10.2.7.3 The provisions of Item 3.10.2.7.2 need not be met during plant startup provided that prior to a power increase to a level greater than 90% of rated power is has been determined that the indicated axial flux difference has been within the target band for a minimum period of two hours.

3.10.2.8 The provisions of Item 3.10.2.4 through 3.10.2.7 need not be met on an interim basis provided that partial flux maps are taken in accordance with the following specifications:

3.10.2.8.1 A partial flux map shall be taken immediately upon violation of any of the provisions of Items 3.10.2.4 through 3.10.2.7 and at minimum 30 minutes intervals thereafter until such time as these same provisions have been satisfied for a preceding 2 hour period.

3.10.2.8.2 The normalized axial power distribution, $F_j(Z)$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of movable incore flux monitoring system.

3.10.2.8.3 At least two thimbles shall be operable for which $\overline{R}$ as defined in the bases has been determined.

3.10.2.8.4 At least two moveable detectors, with associated devices and readout equipment shall be operable for mapping $F_j(Z)$ in the above required thimbles.

3.10.2.8.5 If $F_j(Z)$ exceeds $[F_j(Z)]_s$ as defined in the basis by less than or equal to 4 percent, immediately reduce thermal power one percent of rated power for each percent by which $[F_j(Z)]_s$ is exceeded. Within the next two hour period, either reduce $F_j(Z)$ to within its limit or reduce thermal power to less than 50 percent of rated power.

3.10.2.8.6 If $F_j(Z)$ exceeds $[F_j(Z)]_s$ by greater than 4 percent, immediately reduce thermal power to less than 50 percent of rated power.

3.10.2.8.7 The full incore flux maps used to determine $\overline{R}$ and for monitoring $F_j(Z)$ shall be updated at least once per 30 effective full power days. The continued accuracy and representativeness of the selected thimbles shall be verified by using their latest flux maps no more than 30 effective full power days old to update the $\overline{R}$ for each representative thimble. The original uncertainty, σ, will not be updated, except as follows. If the absolute value of $\overline{R}_j - \overline{R}_{ij}$ exceeds the previous established $\overline{R}_j$ by $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map will be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

3.10.2.9 Operation with part length rods shall be restricted such that the part length rod bank does not cover any axial segment of the core for a period exceeding 18 out of any 30 equivalent full power days. Movement of part length rods shall be restricted by Figure 3.10-3.

3.10.2.10 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 <u>Quadrant Power Tilt Limits</u>

3.10.3.1 Whenever the indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a) Restrict core power level and reset the power range high flux setpoint two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

b) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be reset to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ratio exceeds 1.09 and there is simultaneous indication of a misaligned control rod, Specifications 3.10.3.1a, 3.10.3.1b, and 3.10.5 shall apply, otherwise the reactor shall be brought to the hot shutdown condition within 4 hours. If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt.

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02.

3.10.4 Rod Insertion Limits

3.10.4.1 The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin in Figure 3.10-1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits shown in Figure 3.10-4 or Figure 3.10-5 and the part length bank to the limits specified in Figure 3.10-3.

3.10.4.3 Control bank insertion shall be further restricted if the measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown.

3.10.4.4 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod inserted and part length rods fully withdrawn.

3.10.5 <u>Rod Misalignment Limitations</u>

3.10.5.1 If the indicated full-length or part-length rod cluster control assembly is misaligned from its bank demand position by more than 13 steps, then

a. apply Specification 3.10.3 if quadrant power tilt limits are exceeded.

b. realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.2 If Specification 3.10.3 is determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misalligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned rod cluster control is not realigned within 8 hours, the rod shall be declared inoperable.

3.10.6 <u>Inoperable Rod Position Indicator Channels</u>

3.10.6.1 If a rod position indicator channel is out of service then:

a. For operation between 50 percent and 100 percent of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first.

b. During operation below 50 percent of rating, no special monitoring is required.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3.10.6.3 If a full length or part length rod having a rod position indicator channel out of service, is found to be misaligned from 3.10.6.1a, above, then Specification 3.10.5 will be applied.

3.10.7 Inoperable Rod Limitations

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5 or 3.10.8.

3.10.7.2 Not more than one inoperable full length rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

3.10.7.3 The potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical

transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

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

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.10 Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.11 Notification

Any event requiring plant shutdown or trip setpoint reduction because of Specification 3.10 shall be reported to the Atomic Energy Commission within 30 days.

Basis

Design criteria have been chosen for Condition I and II events which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to conditions imposed for Condition I and II events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel factors are defined.

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

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

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

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on 2.5 times the normalized envelope of Figure 3.10-2 indicate a peak clad temperature of 2110°F corresponding to a 90°F margin to the 2200°F limit. Alternately, the LOCA analyses indicate a 2200°F F_Q limit of 2.63 times the normalized envelope of Figure 3.10-2.[1]

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

In the specified limit of $F^N_{\Delta H}$ there is a 15 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.65/1.15$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design basis including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design basis remain inviolate and identify operational anomolies which would, otherwise, affect these basis.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank damand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The full length and part length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above 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 reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that F_Q upper bound envelope of 2.32 times Figure 3.10-2 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

Since the F_Q upper bound envelope of 2.32 times Figure 3.10-2 is well below the FAC LOCA limit of 2.63, the specification permits occassional power operation with the indicated axial flux difference outside of the target band. In this case, the F_Q value of 2.32 may be exceeded and therefore frequent incore measurements of F_Q are required to assure that the F_Q LOCA limit of 2.63 times K (Z) in Figure 3.10-2.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ± 5 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 two methods for updating the target flux difference. Figure 3.10-6 shows a typical construction of the target flux difference band at BOL and Figure 3.10-7 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part 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 part power. Strict control of the flux difference if 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 low probability of a significant accident occuring 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, a limitation of one hour in any period of 24 hours is placed on operation outisde the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 to -14 percent (+11 percent to -11 percent indicated) increasing by $\pm$ 1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the $\pm$5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the boron system to position the full length control rods to produce the required indicated flux difference. It is accomplished, when part length control rods are used, by using the boron system to position the full length control rods in a desired range and simultaneously using the part length bank to control flux difference. The difference between these two methods is in the rate at which a return to full power can be accomplished when there is an increase in plant power demand.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cuase of the transients considered.

Above the power level of 50% rated power, additional flux shape monitoring is required if the ±5% target band has been exceeded more than 1 hour out of the last 24 hours. In order to assure that the total power peaking factor, F_Q, is maintained at or below the LOCA limited value of 2.63, the existing moveable incore instrumentation will be utilized. Thimbles are selected initially during startup physics tests so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor F_Q can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking factor results from a full core flux map and the axial peaking factor in a selected thimble.

The limiting value on the axial distribution function, $(F_j(Z))_L$, and $(F_j(Z))_S$ are derived as follows:

$$(F_Q(Z))_L = \frac{2.63}{P_L} \times K(Z)$$

$$(F_j(Z))_L = \frac{[F_Q(Z)]_L}{(\overline{R}_j)\ (1.03)}$$

$$(F_j(Z))_S = \frac{[F_j(Z)]_L}{(1+\sigma_j)\ (1.07)}$$

WHERE:

a. $P_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of THERMAL POWER.

c. K(Z) is the reduction in limit as a function of core elevation Z as determined from Figure 3.10-2.

d. $[F_j(Z)]_S$ is the operational limit.

e. $\overline{R}_j$, for thimble j, is determined from at least n-6 incore flux maps covering the full configuration of permissible rod patterns at a THERMAL POWER of at least 50% of rated power.

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

where

$$R_{ij} = \frac{F_{Q_i}^{meas}}{[F_{ij}(Z)]_{MAX}}$$

and $F_{ij}(Z)$ is the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of $F_{Q_i}^{meas}$.

f. σ_j is standard deviation of R_j and is derived from n flux maps from the relationship below, or 0.02, whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\overline{R}_j - R_{ij})^2\right]^{\frac{1}{2}}}{\overline{R}_j}$$

g. The factor 1.03 reduction in the [kw/ft] limit is the engineering uncertainty factor.

h. The factors $(1+\sigma_j)$ and 1.07 represent the margin between $[F_j(Z)]_L$ limit and the operational limit $[F_j(Z)]_S$. Since $(1+\sigma_j)$ is bounded by a lower limit of 1.02, there is at least 9% reduction in the operational limit. Operations are permitted in excess of the operational limit $\leq$4% while making power adjustment on a percent for percent basis.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation as this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cuase before a severe tilt could buildup. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely

cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level.

The two hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the cuase of the tilt condition. The measurements would include a full core physics map utilizing the moveable detector system. For a tilt condition $\leq$ 1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event a tilt condition of $\leq$ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for low power physics testing. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition.

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2% for each one-percent the tilt ratio exceeds 1.0) for the specified periods authorized to correct the rod misalignment.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. One percent shutdown is adequate except for steam break analysis, which requires more shutdown if the boron concentration is low. Figure 3.10-1 is drawn accordingly.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during these tests.

The rod position indicator channel is sufficiently accurate to detect a rod ± 7 inches away from its demand position. An indicated misalignment less than 13 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on

excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 13 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion

The required drop time to dashpot entry is consistent with safety analysis.

REFERENCE

1. WCAP-8399, "ECCS Acceptance Criteria Analysis, Indian Point Nuclear Generating Station Unit 2", September 1974, Westinghouse Non-Proprietary Class 3.

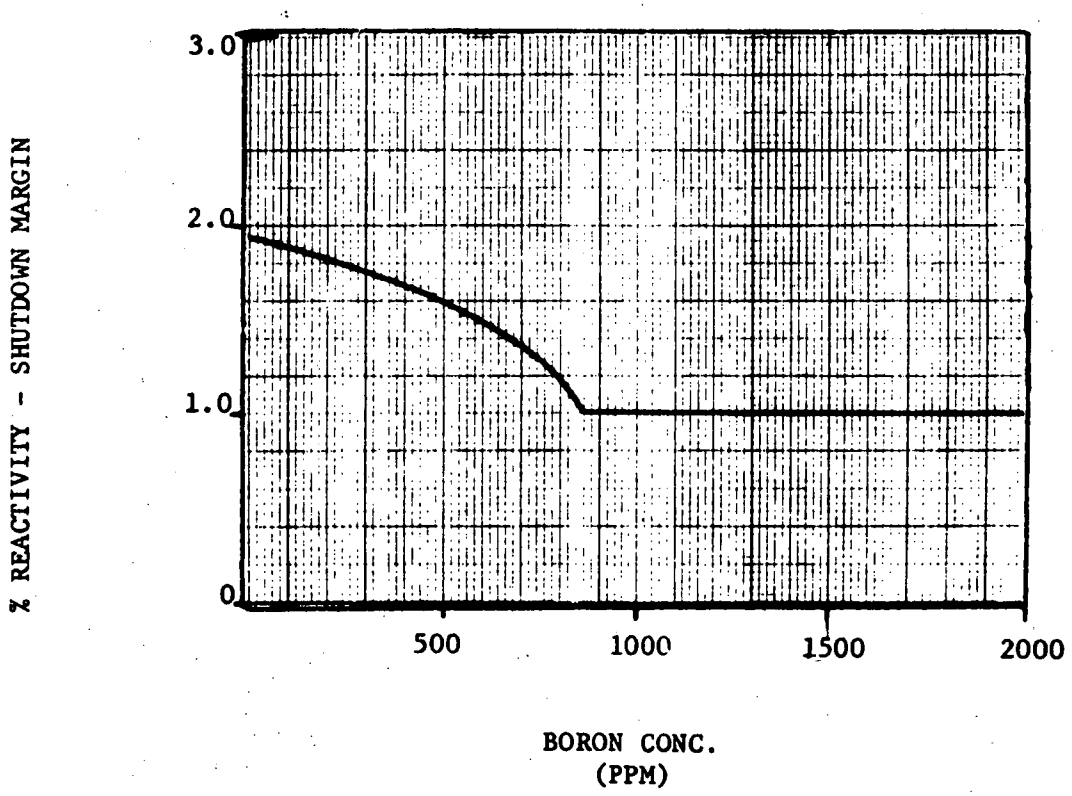

Figure 3.10-1

REQUIRED HOT SHUTDOWN MARGIN
vs
REACTOR COOLANT BORON CONCENTRATION

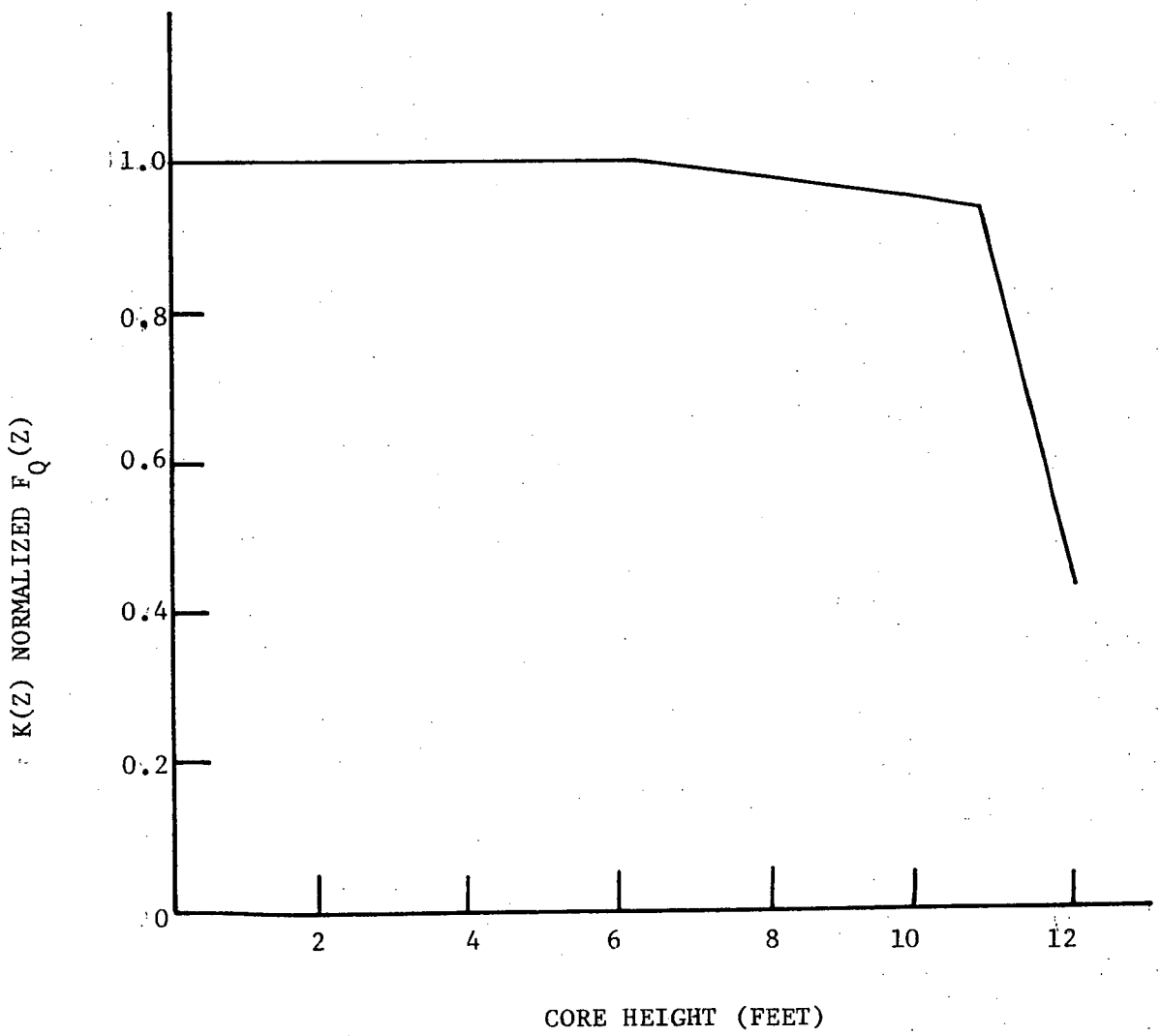

Figure 3.10-2 Hot Channel Factor Normalized Operating Envelope

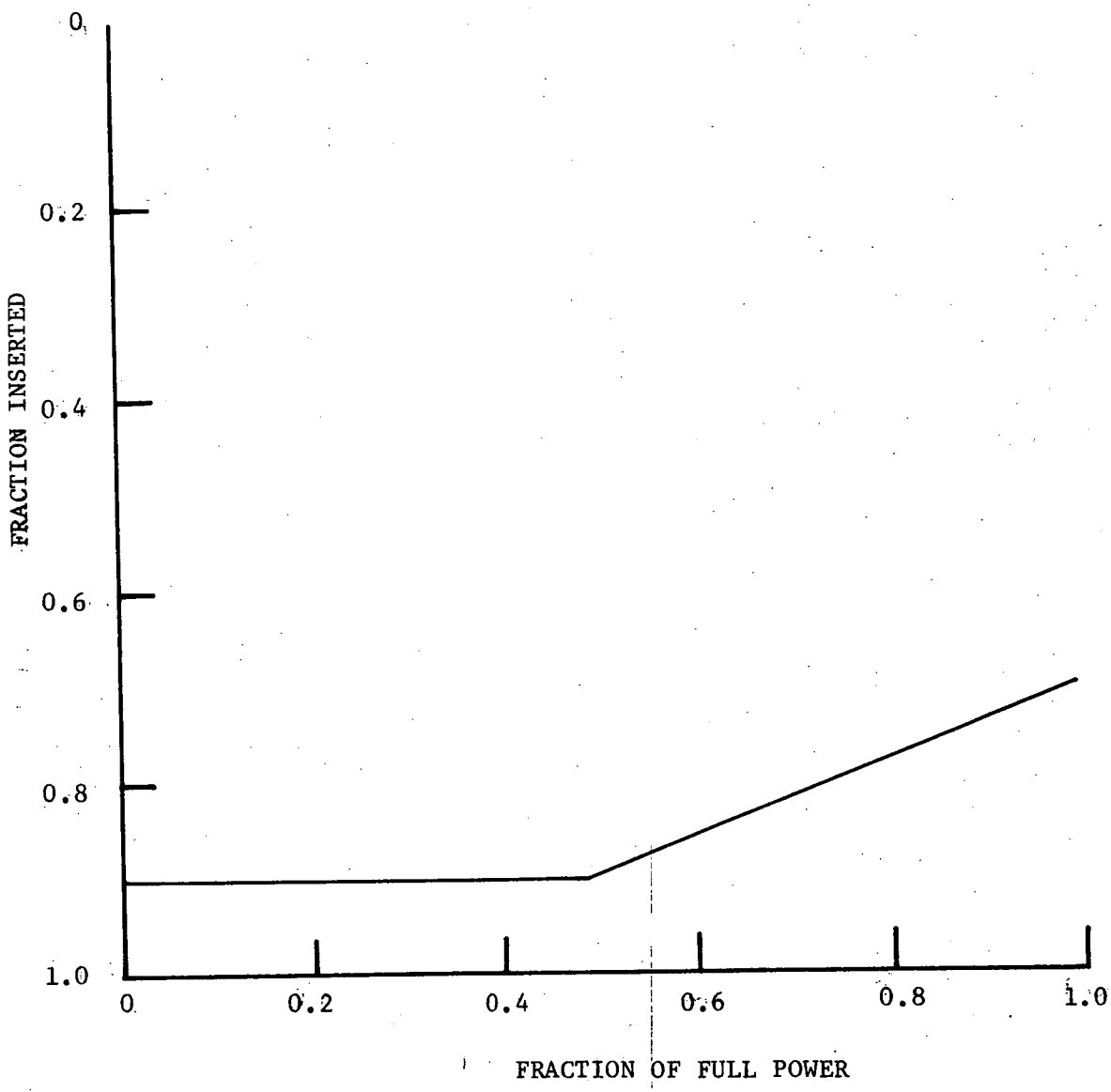

Figure 3.10-3 Part Length Rod Insertion Limit vs. Power

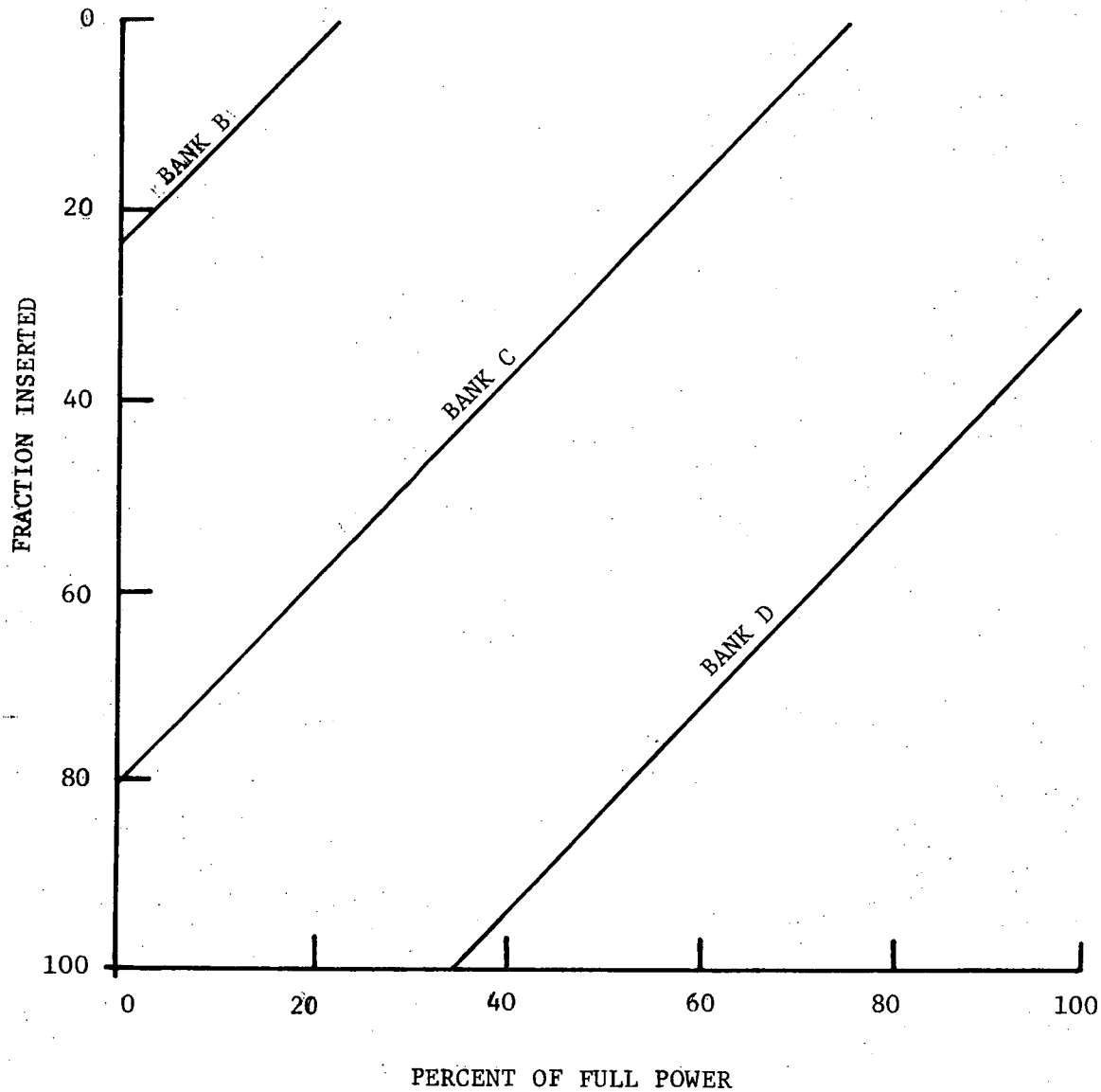

Figure 3.10-4 Insertion Limits 100 Step Overlap
Four Loop Operation

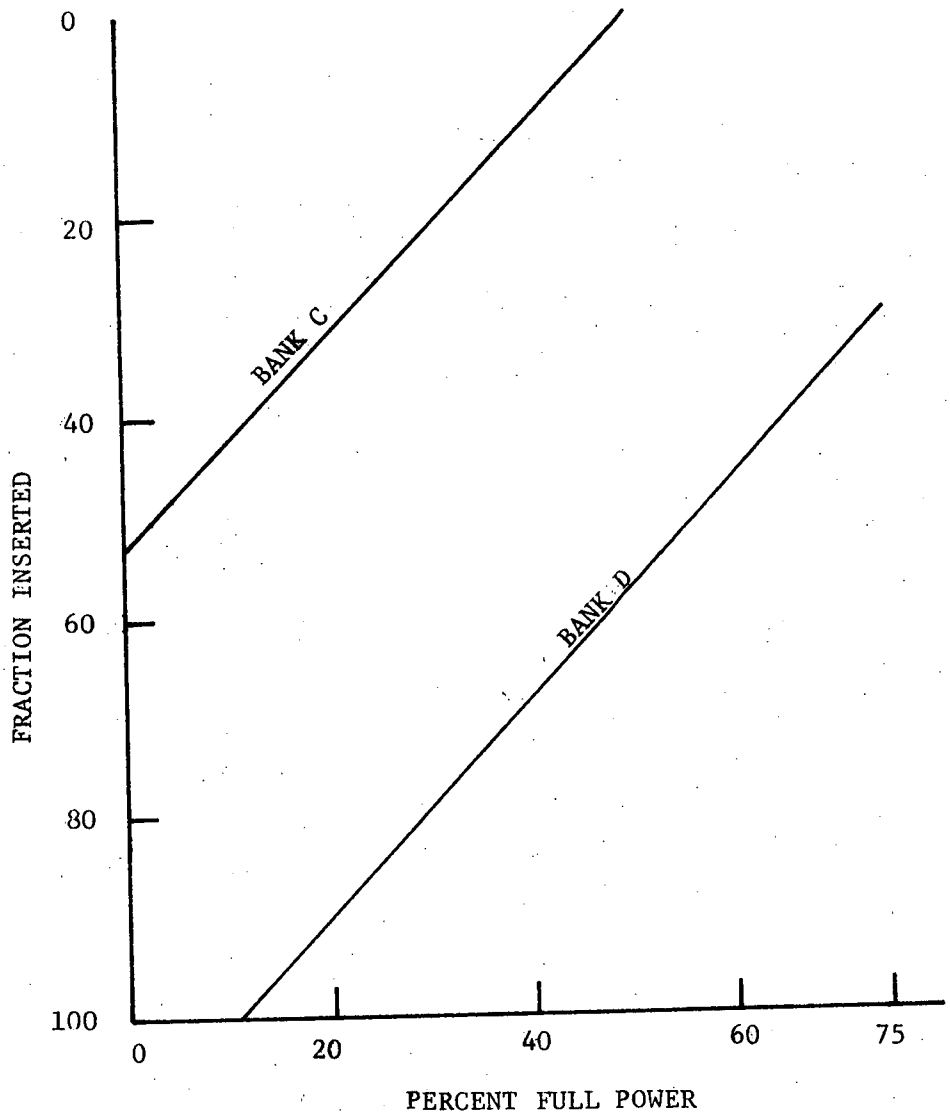

Figure 3.10-5 Insertion Limits 100 Step Overlap
3 Loop Operation

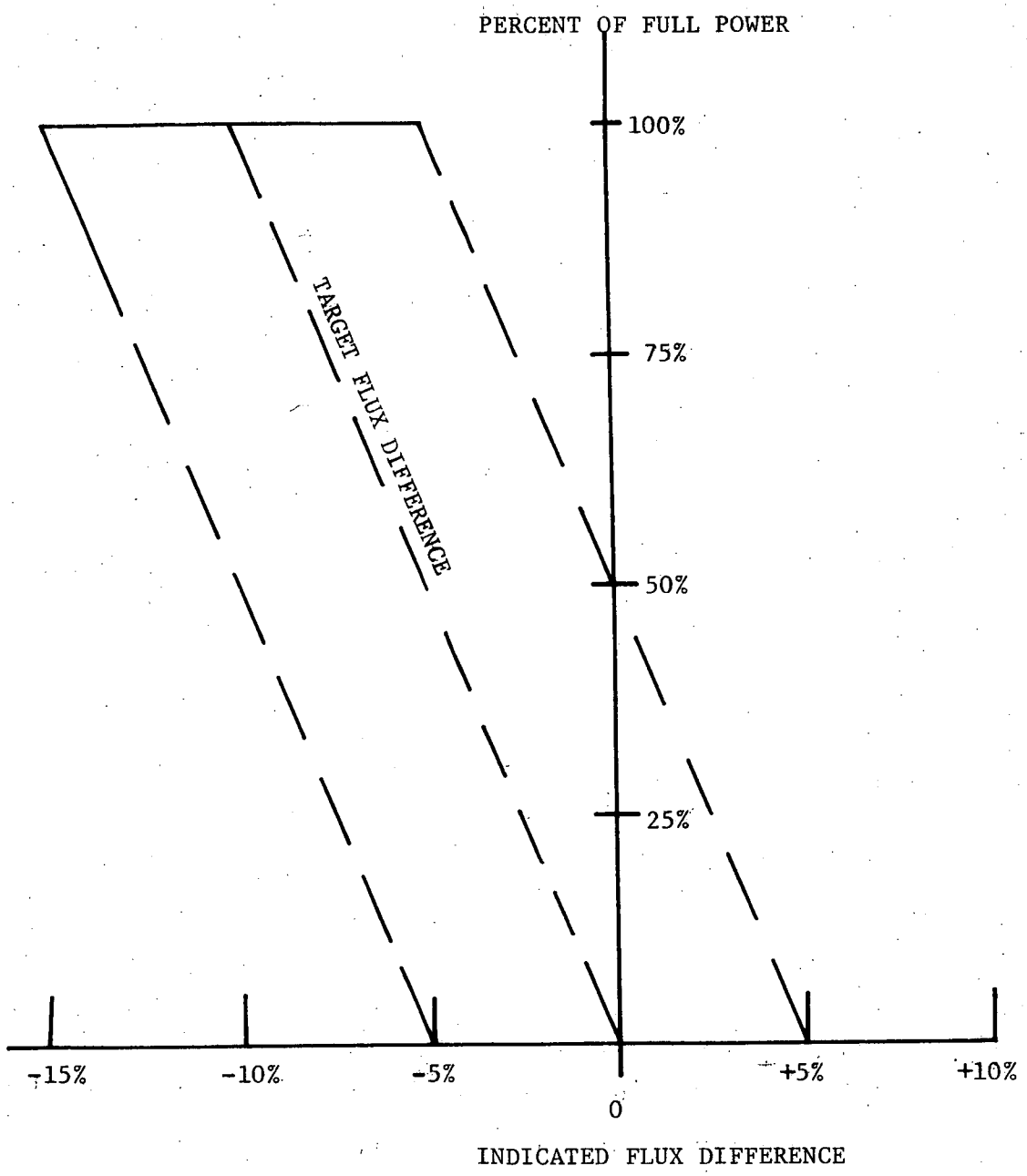

Figure 3.10-6 Target Band on Indicated Flux Difference as a Function of Operating Power Level (Typical for BOL)

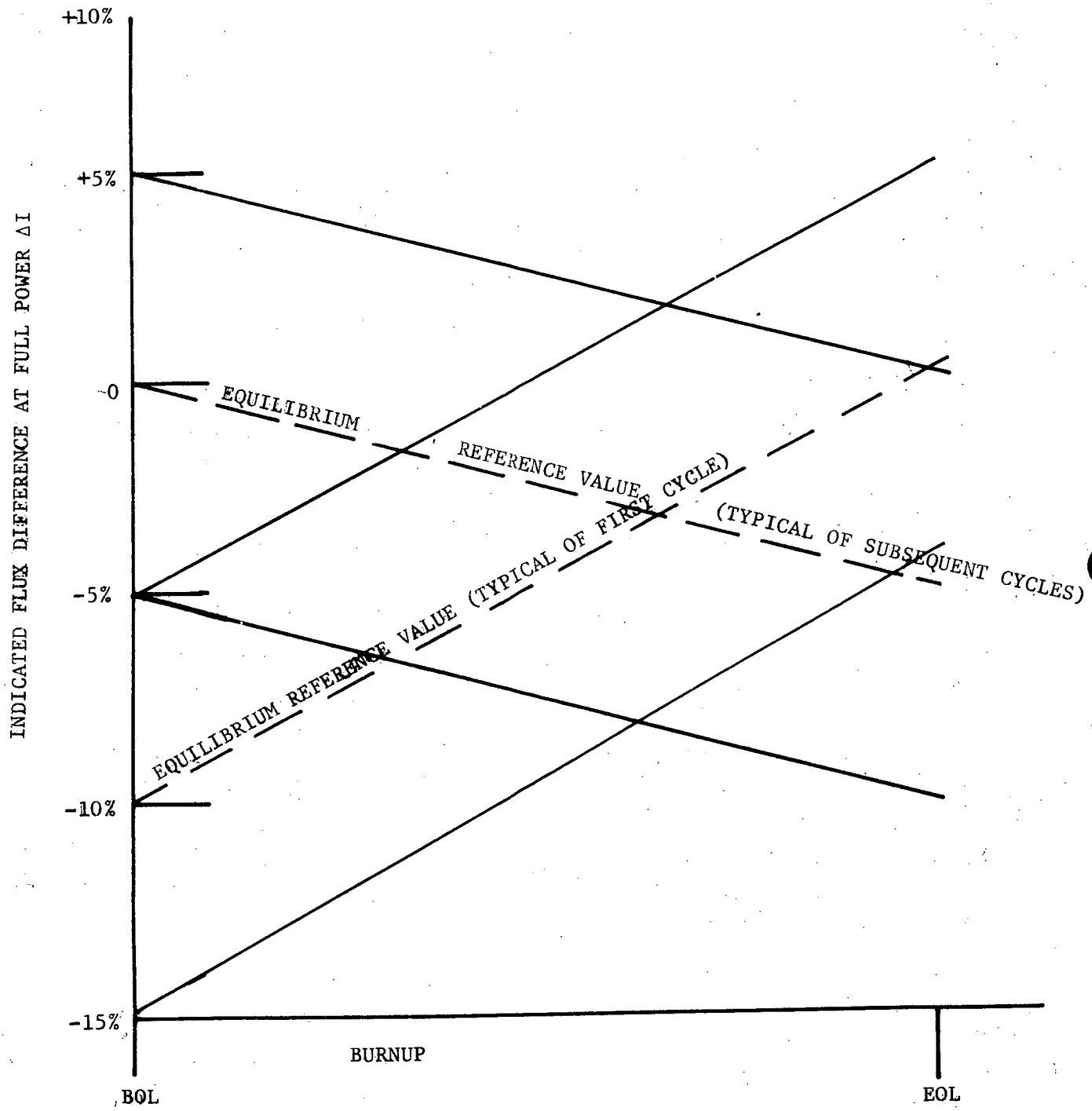

Figure 3.10-7 Permissible Operating Band on Indicated Flux Difference as a Function of Burnup (Typical)

ATTACHMENT B

The proposed changes to Sections 3.3 and 3.10 of the Technical Specifications set forth in Appendix A to Facility Operating License No. DPR-26 are requested as a result of the issuance of Commission's Final Acceptance Criteria for the Emergency Core Cooling System (ECCS). The analysis provided in WCAP-8399, "ECCS Acceptance Criteria Analysis, Indian Point Nuclear Generating Station Unit No. 2", dated September 1974 and included in this attachment demonstrates that the proposed changes to the Technical Specifications satisfy the new criteria.

There are no plant modifications required as a result of the proposed technical specification changes other than the two setpoint changes on the level alarms for the four accumulators. The proposed technical specification changes have been reviewed as required by the Consolidated Edison Nuclear Facilities Safety Committee and the Station Nuclear Safety Committee. Both Committees concur that these changes do not represent a significant hazards consideration.

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