Attachment B

Technical Specification Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988

LIST OF FIGURES

Reactor Core Safety Limit-Four Loops In Operation	2.1-1
PORV Opening Pressure for Operation Less Than or Equal to 295°F	3.1.A-1
Maximum Presurizer Level with PORV's Inoperable and One Charging Pump Energized	3.1.A-2
Maximum Reactor Coolant System Pressure for Operation With PORV's Inoperable and One Safety Injection Pump and/or Three Charging Pumps Energized	3.1.A-3
Reactor Coolant System Heatup Limitations	3.1.B-1
Reactor Coolant System Cooldown Limitations	3.1.B-2
Limiting Fuel Burnup Versus Initial Enrichment	3.8-1
Required Hot Shutdown Margin vs. Reactor Coolant Boron Concentration	3.10-1
Hot Channel Factor Normalized Operating Envelope	3.10-2
Insertion Limits, 100 Step Overlap Four Loop Operation	3.10-3
Figure Deleted	3.10-4
Target Band on Indicated Flux Difference as a Function of Operating Power Level	3.10-5
Permissible Operating Band on Indicated Flux Difference as a Function of Burnup	3.10-6
Reactor Coolant System Heatup Limitation	4.3-1
Unrestricted Areas for Radioactive Gaseous and Liquid Effluents	5.1-1

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperaure during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figures 2.1-1 ___ ___

___ . 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 region 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 L-grid correlation for analysis of the LOPAR fuel, and the WRB-1 correlation for evaluation of the OFA. These DNB correlations have been developed to predict the DNB flux and location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB design basis is as follows: There must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that the DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin is maintained by performing DNB design evaluations to a higher DNBR value, called the Safety Limit DNBR. This margin is sufficient to cover applicable rod bow DNB penalties and provide margin for use in design and operational flexibility.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. These curves are based on a peak nuclear hot channel factor of 1.62 for the LOPAR fuel and a 1.65 for the OFA and a 1.55 cosine axial power shape.

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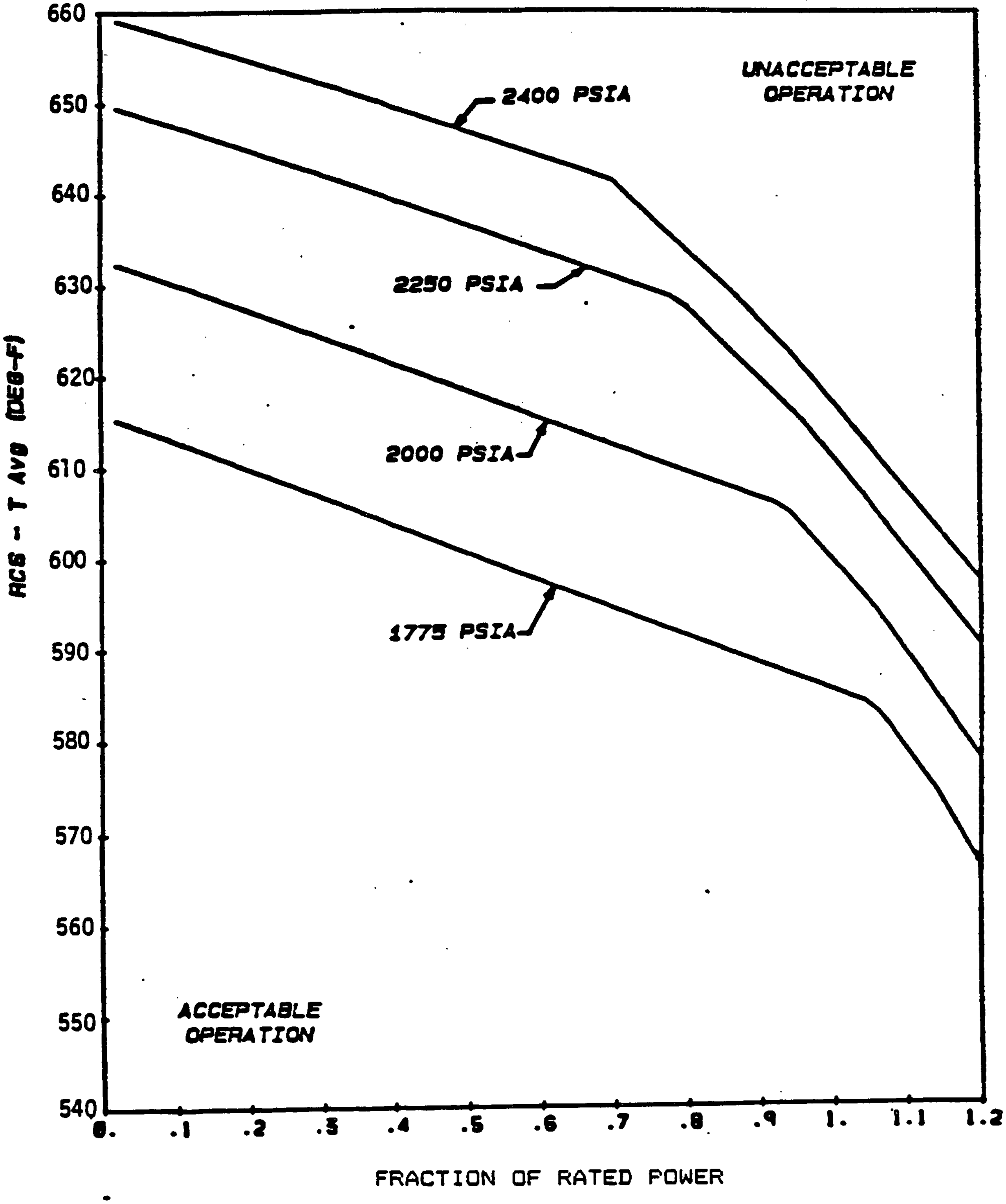

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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FIGURE 2.1-2

(3) Low pressurizer pressure - $\geq$1870 psig.

(4) Overtemperature ΔT

$$\Delta T \leq \Delta T_o \ [(K_1 - K_2 (T - T') + K_3 (P - P') - f(I)]$$

where:
ΔT = Measured ΔT by hot and cold leg RTDs, °F
ΔT_o $\leq$ Indicated ΔT at rated power

T = Average temperature, °F
T' = Design full power Tave at rated power, $\leq$567.7°F
P = Pressurizer pressure, psig
P' = 2235 psig
K_1 $\leq$ 1.25
K_2 = 0.022
K_3 = 0.00095

and f(ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -36% and +7%, f (ΔI) = 0, where q_t and q_b are percent RATED POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total POWER in percent of RATED POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -36%, the ΔT Trip Setpoint shall be automatically reduced by 2.14% of its value at RATED POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED POWER.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \ [K_4 - K_5 \frac{dT}{dt} - K_6 (T - T'')]$$

where:
ΔT = Measured ΔT by hot and cold leg RTDs, °F

ΔT_o $\leq$ Indicated ΔT at rated power

T = Average temperature, °F

T'' = Indicated full power T_{avg} at rated power $\leq$ 567.7°F

K_4 $\leq$ 1.074

K_5 = Zero for decreasing average temperature

K_5 $\geq$ 0.188, for increasing average temperature (sec/°F)

K_6 $\geq$ 0.0015 for $T \geq T''$; $K_6 = 0$ for $T < T''$

$\frac{dT}{dt}$ = Rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) $\geq$90% of normal indicated loop flow

(b) Low reactor coolant pump frequency - $\geq$ 57.5 cps

(7) Undervoltage - $\geq$ 70% of normal voltage

C. Other reactor trips

(1) High pressurizer water level - $\leq$ 92% of span

(2) Low-low steam generator water level - $\geq$ 5% of narrow range instrument span.

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

   A. The reactor trips on low pressurizer pressure, high pressurizer level, and low reactor coolant flow for two or more loops shall be unblocked when:

      1) Power range nuclear flux $\geq$ 10% of rated power, or

      2) Turbine first stage pressure $\geq$ 10% of equivalent full load.

   B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates $\leq$ 60% of rated power.

   C. The anticipatory reactor trip upon turbine trip shall be unblocked when the power range nuclear instrumentation indicates $\geq$ 35% of rated power.

3. The Control Rod Protection System, shall open the reactor trip breakers during RCS cooldown prior to T_{cold} decreasing below 350°F.

Basis

The high flux reactor trips provide redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.(1)

The power range nuclear flux reactor trip high set point protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.(2)(3)

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis.[4]

The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detector (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 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 Delta-T reactor trip prevents power density anywhere in the core from exceeding 118% of design power density,

and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[2]

The low flow reactor trip protects the core against DNB in the event of a loss of one or two reactor coolant pumps. The undervoltage reactor trip protects the core against DNB in the event of a loss of two or more reactor coolant pumps. The set points specified are consistent with the values used in the accident analysis.(8) The low frequency reactor coolant pump trip also protects against a decrease in flow. The specified set point assures a reactor trip signal by opening the reactor coolant pump breaker before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1600 ft^3 of water (39.75 ft above the lower instrument tap) corresponds to 92% of span. The specified set point allows margin for instrument error and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against postulated loss of feedwater accidents. The specified set point 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)

Specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point at which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 60% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below the safety limit DNBR's during normal operational transients.

A Turbine Trip causes a direct reactor trip, when operating at or above 35% power, in order to reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of this trip. Functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

The steam-feedwater flow mismatch trip does not appear in the specification as this setting is not used in the transient and accident analysis (FSAR Section 14).

To avoid mechanical interference due to thermal contraction between the fuel and the control rods, an automatic backup to manual tripping of the control rods is provided. Prior to T_{cold} decreasing below 350°F during RCS cooldown, the Control Rod Protection System will open the reactor trip breakers which unlatches the control rod drive shafts from the CRDMs.

References

(1) FSAR 14.1.1
(2) FSAR 14.1.2
(3) FSAR Table 7.4.2
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2
(7) FSAR 3.2.1
(8) FSAR 14.1.6
(9) FSAR 14.1.9

capability for removing decay heat; but single failure considerations require that at least two loops be operable. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant system.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

The specification that all reactor coolant pumps be operational during power operation is to assure that adequate core cooling will be provided. This flow will keep the minimum departure from nucleate boiling ratio above the safety limit DNBRs; therefore, cladding damage and release of fission products will not occur.

The Overpressure Protection System (OPS) is designed to relieve the RCS pressure for certain unlikely overpressure transients to prevent these incidents from causing the peak RCS pressure from exceeding 10CFR50, Appendix G limits. When the OPS is "armed" MOVs 535 and 536 are in the open position, and the PORVs will open upon receipt of the appropriate signal. This OPS arming can be accomplished either automatically by the OPS when the RCS is below a prescribed temperature or manually by the operator.

The OPS will be set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G limits for the following events:

1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature hotter than the RCS water temperature.

2. Letdown isolation with three charging pumps operating.

3. Startup of one safety injection pump.

4. Loss of residual heat removal causing pressure rise from heat additions from core decay heat or reactor coolant pump heat.

5. Inadvertant activation of the pressurizer heaters.

Consideration of the above events provides bounding PORV setpoints for other potential overpressure conditions caused by heat or mass additions at low temperature.

G. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW RATE

Specifications

The following DNB related parameters pertain to four loop steady-state operation at power levels greater than 98% of rated full power:

a. Reactor Coolant System $T_{ave} \leq 573.5°F$

b. Pressurizer Pressure $\geq$ 2206 psig

c. Reactor Coolant System Total Flow Rate $\geq$ 331,840 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Basis

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than the safety limit DNBRs.

The limits on reactor coolant system temperature, pressure and loop coolant flow represent those used in the accident analyses and are specified to assure that the values assumed in the accident analyses are not exceeded during steady-state four loop operation. Indicator uncertainties have not been accounted for in determining the DNB parameter limits on temperature and pressure.

Compliance with the specified ranges on reactor coolant system temperature and pressurizer pressure is demonstrated by verifying that the parameters are within their applicable ranges at least once each 12 hours.

Compliance with the specified range on Reactor Coolant System total flow rate is demonstrated by verifying the parameter is within it's range after each refueling cycle.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met.

1. Two charging pumps shall be operable.

2. The boric acid storage system shall contain a minimum of 6000 gallons of 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, and at least one boric acid transfer pump shall be operable.

3. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

4. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system.

C. During power operation, the requirements of 3.2.B may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.2.B within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to operable status within 24 hours.

2. The boric acid storage system (including the boric acid transfer pumps) may be inoperable provided the RWST is operable and provided that the boric acid storage system and at least one boric acid transfer pump is restored to operable status within 48 hours.

3. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

4. Both channels of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided at least one channel is restored to operable status within 48 hours, the required flow path is shown to be clear of blockage, and the second channel is restored to operable status within 7 days.

D. When RCS temperature is less than or equal to 295°F, the requirements of Table 3.1.A-2 regarding the number charging pumps allowed to be energized shall be adhered to.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank.

A third method will be to depressurize and use the safety injection pumps. There are three sources of borated water available for injection through 3 different paths.

(1) The boric acid transfer pumps can deliver the contents of the boric acid storage system to the charging pumps.

(2) The charging pumps can take suction from the refueling water storage tank. (2000 ppm boron solution). Reference is made to Technical Specification 3.3.A.

(3) The safety injection pumps can take their suction from the refueling water storage tank.

The quantity of boric acid in storage from either the boric acid storage system or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

Approximately 5700 gallons of the 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) of boric acid are required to meet cold shutdown conditions.

Thus, a minimum of 6000 gallons in the boric acid storage system is specified. An upper concentration limit of 13% (22,500 ppm of boron) boric acid in the boric acid storage system is specified to maintain solution solubility at the specified low temperature limit of 145°F. One of two channels of heat tracing is sufficient to maintain the specified low

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 storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

   b. Deleted

   c. The four accumulators are pressurized to at least 615 psig and each contains a minimum of 787.5 ft^3 and a maximum of 802.5 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 and heat exchangers together with their associated piping and valves are operable.

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

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases after initiating hot shutdown. Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to affect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and therefore in such a case the reactor is to be put into the cold shutdown condition.

Valves 1810, 744 and 882 are kept in the open position during plant operation to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As an additional assurance of flow passage availability, the valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves to take place. This additional precaution is acceptable since failure to manually re-establish power to close valves 1810 and 882, following the injection phase, is tolerable as a single failure. Valve 744 will not need to be closed following the 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. (9) 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.

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, and are based on the values used for the accident analysis.[9]

Two independent diverse systems are provided for removal of combustible hydrogen from the containment building atmosphere: (a) the hydrogen recombiners, and (b) the post accident containment venting system. Either of the two (2) hydrogen recombiners or the post accident containment venting system are capable of wholly providing this function in the event of a design basis accident.

Two full rated hydrogen 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 are separate, so that loss of one power supply will not affect 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 Post Accident Containment Venting System consists of a common penetration line which acts as a supply line through which hydrogen free air can be admitted to the containment, and an exhaust line, with parallel valving and piping, through which hydrogen bearing gases from containment may be vented through a filtration system.

The control room ventilation sytem is equipped with a toxic gas detection system consisting of redundant monitors capable of detecting chlorine, anhydrous ammonia, and hydrogen cyanide. These toxic gas detection systems are designed to isolate the control room from outside air upon detection of toxic concentration of the monitored gases in the control room ventilation system. The operability of the toxic gas detection systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored and the setpoint established for the monitors are based on the results described in the Indian Point Unit No. 2 Control Room Habitability Study dated May, 1981.

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 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 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 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. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

The specified quantities of water for the RWST include unavilable water (4687 gals) in the tank bottom, inaccuracies (6200 gals) in the alarm setpoints, and minimum quantities required during injection (246,000 gals)(10) and recirculation phases (80,000 gals).(10) The minimum RWST (i.e., 345,000 gals) provides approximately 8,100 gallons margin. The minimum RWST boron concentration ensures that the reactor core will remain subcritical during long term recirculation with all control rods fully withdrawn following a postulated large break LOCA.

The seven day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.(13) The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 47 psig.

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) Indian Point Unit No. 2 FSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

(11) FSAR Sections 6.5 and 6.6.

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMITS
1.	High Containment Pressure (Hi level)	Safety Injection	$\leq$ 2.0 psig
2.	High Containment Pressure (Hi-Hi level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 30 psig
3.	Pressurizer Low Pressure	Safety Injection	$\geq$ 1829 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	$\leq$ 150 psi
5.	High Steam Flow in 2/4 Steam Lines Coincident with Low Tavg or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	$\leq$ 40% of full steam flow at zero load $\leq$ 40% of full steam flow at 20% load $\leq$ 110% of full steam at full load $\geq$ 540°F Tavg $\geq$ 600 psig steam line pressure
6.	Steam Generator Water Level (low-low)	Auxiliary Feedwater	$\geq$ 5% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	$\geq$ 40% nominal voltage
8a.	480v Emergency Bus Undervoltage (Loss of Voltage)	---	220V + 100V, -20V 3 sec $\pm$ 1 sec
8b.	480v Emergency Bus Undervoltage (Degraded Voltage)	---	403V $\pm$ 5V 180 sec $\pm$ 30 sec

Amendment No.

3.6 CONTAINMENT SYSTEM

APPLICABILITY

Applies to the integrity of reactor containment

OBJECTIVE

To define the operating status of the reactor containment for plant operation

SPECIFICATION

A. CONTAINMENT INTEGRITY

1. The following requirements shall be satisfied: (a) whenever the reactor is above cold shutdown or (b) whenever the reactor vessel head is less than fully tensioned and, (i) the shutdown margin is $< 5\% \Delta k/k$: or, (ii) the boron concentration within the reactor is less than 2000 ppm.

   a. All non-automatic containment isolation valves which are not required to be open during accident conditions are closed and blind flanges installed where required. Those non-automatic containment isolation valves listed in Table 3.6-1 and any test connection valves which are located between containment isolation valves and which are normally closed with threaded caps or blind flanges installed, may be opened if necessary for plant operation or for testing and only as long as necessary to perform the intended function.

   b. All automatic containment isolation valves are either operable or in the closed position or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

   c. The equipment door is properly closed.

   d. At least one door in each personnel air lock is properly closed.

   e. The WC&PPS requirements of Specification 3.3.D are being satisfied.

   f. Containment leakage has been verified in accordance with the surveillance requirements of Specification 4.4.

2. The following additional requirements shall be satisfied during power operation:

B. Internal Pressure

If the internal pressure exceeds 2 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

C. Containment Temperature

The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.

BASIS

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if a Reactor Coolant System rupture were to occur.

The shutdown margins are selected based on the type of activities that are being carried out. The shutdown margin requirement of specification 3.8.B.2 when the head is off precludes criticality during refueling. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality at cold shutdown conditions.

Regarding internal pressure limitations, the containment calculated peak accident pressure of 47 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 8 psig.(1) The containment can withstand an internal vacuum of 2.5 psig.(2) The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material.(3)

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. During periods of normal plant operations requiring containment integrity, valves on this Table will be open either continuously or intermittently depending on requirements of the particular

6. The requirements for RHR pump and heat exchanger operability/operation in Specifications 3.8.A.3 and 3.8.A.4 may be suspended during maintenance, modification, testing, inspection, repair or the performance of core component movement in the vicinity of the reactor pressure vessel hot legs. During operation under the provisions of this specification, an alternate means of decay heat removal shall be available when the required number of RHR pump(s) and heat exchanger(s) are not operable. With no RHR pump(s) and heat exchanger(s) operating, the RCS temperature and the source range detectors shell be monitored hourly.

7. The reactor Tavg shall be less than or equal to 140°F.

8. Specification 3.6.A.1 shall be adhered to for reactor subcriticality and containment integrity.

B. With fuel in the reactor vessel and when:

i) the reactor vessel head is being moved, or
ii) the upper internals are being moved, or
iii) loading and unloading fuel from the reactor, or
iv) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed,

the following specifications (1) through (12) shall be satisfied:

1. Specification 3.8.A above shall be met.

2. The minimum boron concentration shall be $\geq$2000ppm and shall provide a shutdown margin $\geq$ 5% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

3. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

4. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 131 hours.

The shutdown margin requirements will keep the core subcritical. During refueling, the reactor refueling cavity is filled with borated water. The minimum boron concentration of this water is the more restrictive of either 2000ppm or else sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold shutdown condition with all rods inserted. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. Periodic checks of refueling water boron concentration ensure the proper shutdown margin. The specifications allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 131-hour decay time following plant shutdown and the 23 feet of water above the top of the reactor vessel flanges are consistent with the assumptions used in the dose calculations for fuel-handling accidents both inside and outside of the containment. The analysis of the fuel handling accident inside of the containment is based on an atmospheric dispersion factor (X/Q) of 5.1×10^{-4} sec/m^3 and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

The requirement that at least one RHR pump and heat exchanger be in operation ensures that sufficient cooling capacity is available to maintain reactor coolant temperature below 140°F, and sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR pumps and heat exchangers operable when there is less than 23 feet of water above the vessel flange ensures that a single failure will not result in a complete loss of residual heat removal capability. With the head removed and at least 23 feet of water above the flange, a large heat sink is available for core cooling, thus allowing adequate time to initiate actions to cool the core in the event of a single failure.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the refueling operation during changes in core geometry.

The fuel enrichment and burnup limits in Specification 3.8.C.1 assure the limits assumed in the spent fuel safety analyses will not be exceeded. Within this specification adjacent location means those four locations directly contacting the four sides (faces) of a fuel assembly but excludes those four locations which contact the four corners of a fuel assembly.

References

(1) FSAR Section 9.5.2

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.

Objectives:

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. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

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:

(a) $F^N_{\Delta H} \leq 1.62[1 + 0.3\ (1-P)]$

(b) For $\leq$ 25% steam generator tube plugging:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.64) \times K(Z)$ for $P \leq .5$

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.

b) there is no simultaneous indication of a misaligned control rod, reduce thermal power to less than 50% of rated thermal power within 2 hours and reduce the power range high flux trip setpoint to less than or equal to 55% of rated thermal power within the next 4 hours.

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 except as modified in specification 3.10.10.

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 3.10-3.

3.10.4.3 Control bank insertion shall be further restricted if:

a. 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,

b. A rod is inoperable (Specification 3.10.7).

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 control rod inserted.

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 fails to meet the requirements of 3.10.8.

3.10.7.2 Not more than one inoperable control 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 If any rod has been declared inoperable, then the potential eject worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform 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 control rod shall be no greater than 2.4 seconds from gripper release 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.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analyses. These related to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must be greater than the safety limit DNBRs in normal operation or in short term transients.

In addition to the above conditions, 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.

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

It should be noted that $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$.

The upper bound envelope of the total peaking factor (F_Q) of specification 3.10.2.1 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 the specified F_Q times the normalized envelope of Figure 3.10-2 indicate a peak clad temperature of less than 2200°F for the worst case double-ended cold leg guillotine break.[1]

When an 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 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H}$ 1.62/1.08. 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 thorugh 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.

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 bank 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 bank, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside 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 ± 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 ±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 by using the boron system to position the control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of less than the safety limit DNBRs 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 cause of the transients considered.

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.

<u>REFERENCE</u>

1. FSAR Section 14.3

FIGURE 3.10-2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE
(For S.G. Tube Plugging Levels up to 25%)

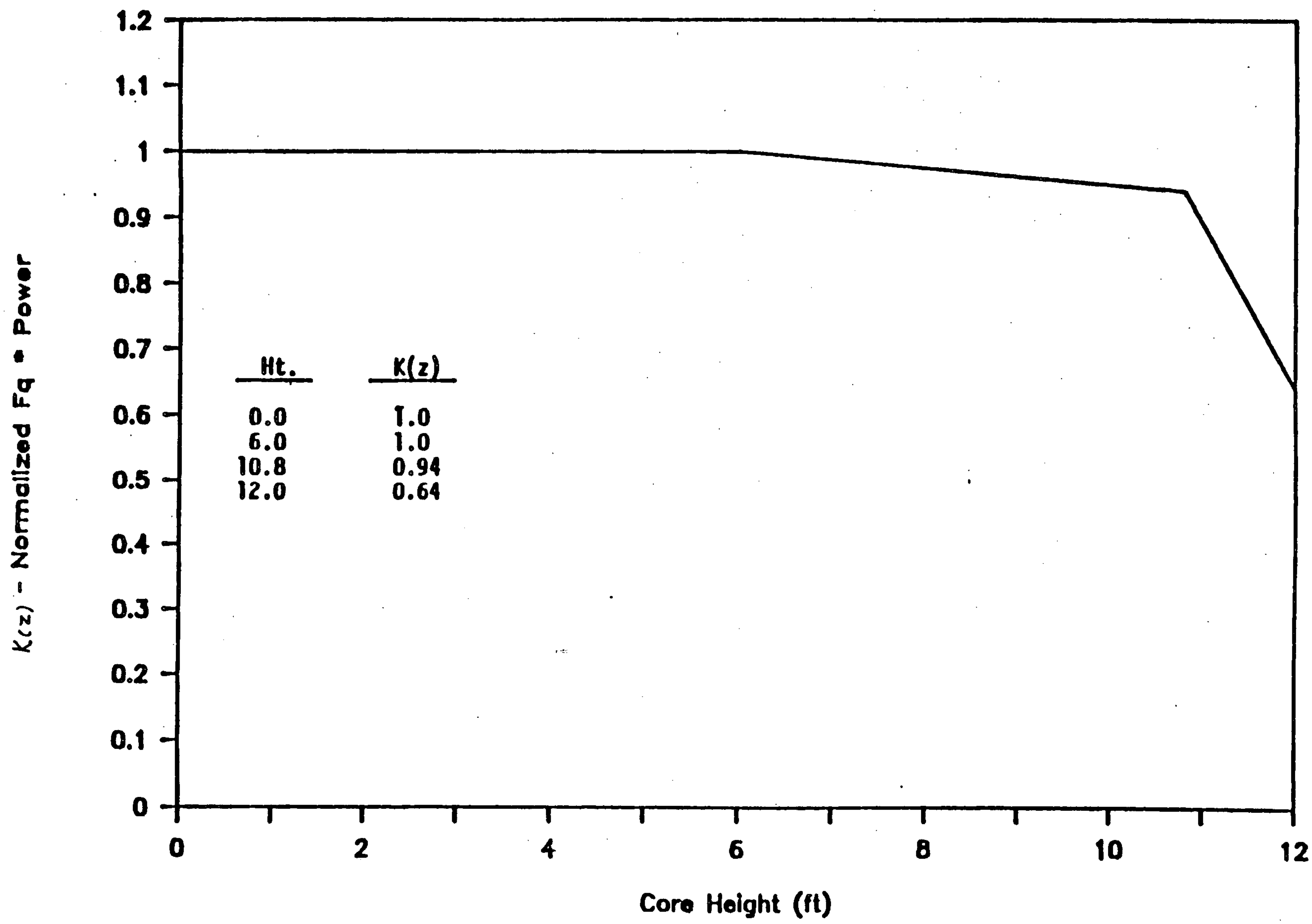
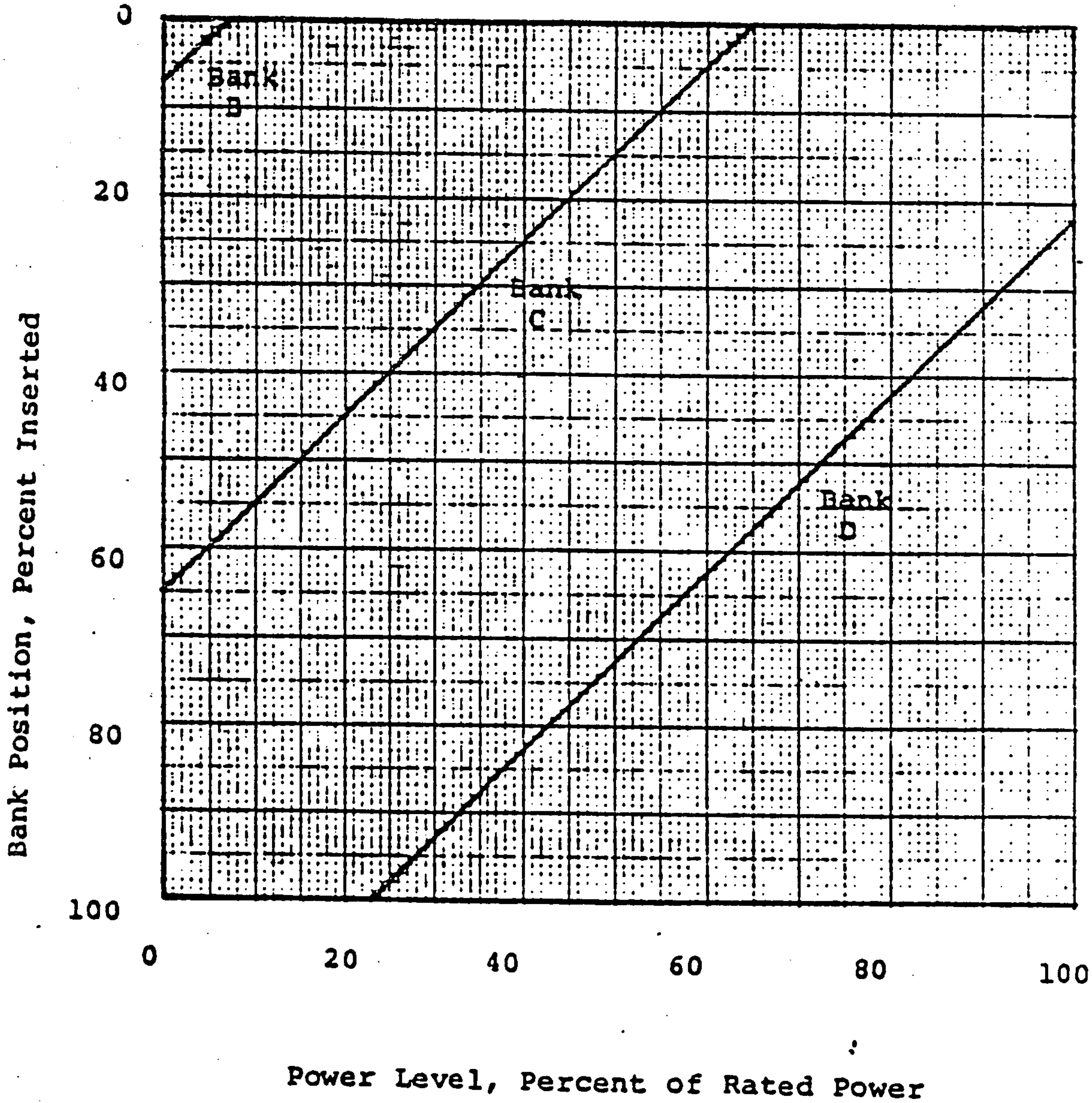


Figure 3.10-3

ROD BANK INSERTION LIMITS
(Four Loop Operation)
100 Step Overlap



Amendment No,

Figure 3.10-4

Intentionally
Deleted

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 system, 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 rated power if re-calibration requirements for excore axial off-set detection system, identified in Table 4.1-1, are not met.

Basis

The Movable In-core Instrumentation System[1] has six drives, six detectors, and 50 thimbles in the core. Each detector can be routed to sixteen or more 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 system, 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.

Amendment No.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, and with 0% Steam Generator tube plugging is 11,350 cubic feet.

References

(1) FSAR Section 3.2
(2) Deleted
(3) Deleted
(4) Deleted
(5) FSAR Sections 3.2
(6) FSAR Table 4.1-9

Attachment C

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 30, 1988