ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

RELATED TO

SAFETY LIMIT, REACTOR CORE

AND

CONTROL ROD AND POWER DISTRIBUTION LIMITS,

SHUTDOWN REACTIVITY

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

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 temperature 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 Figure 2.1-1 for four-loop operation. 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 DNB "L" grid geometry correlation.[3] The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure 2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.30. The area where clad integrity is assured is below these lines.

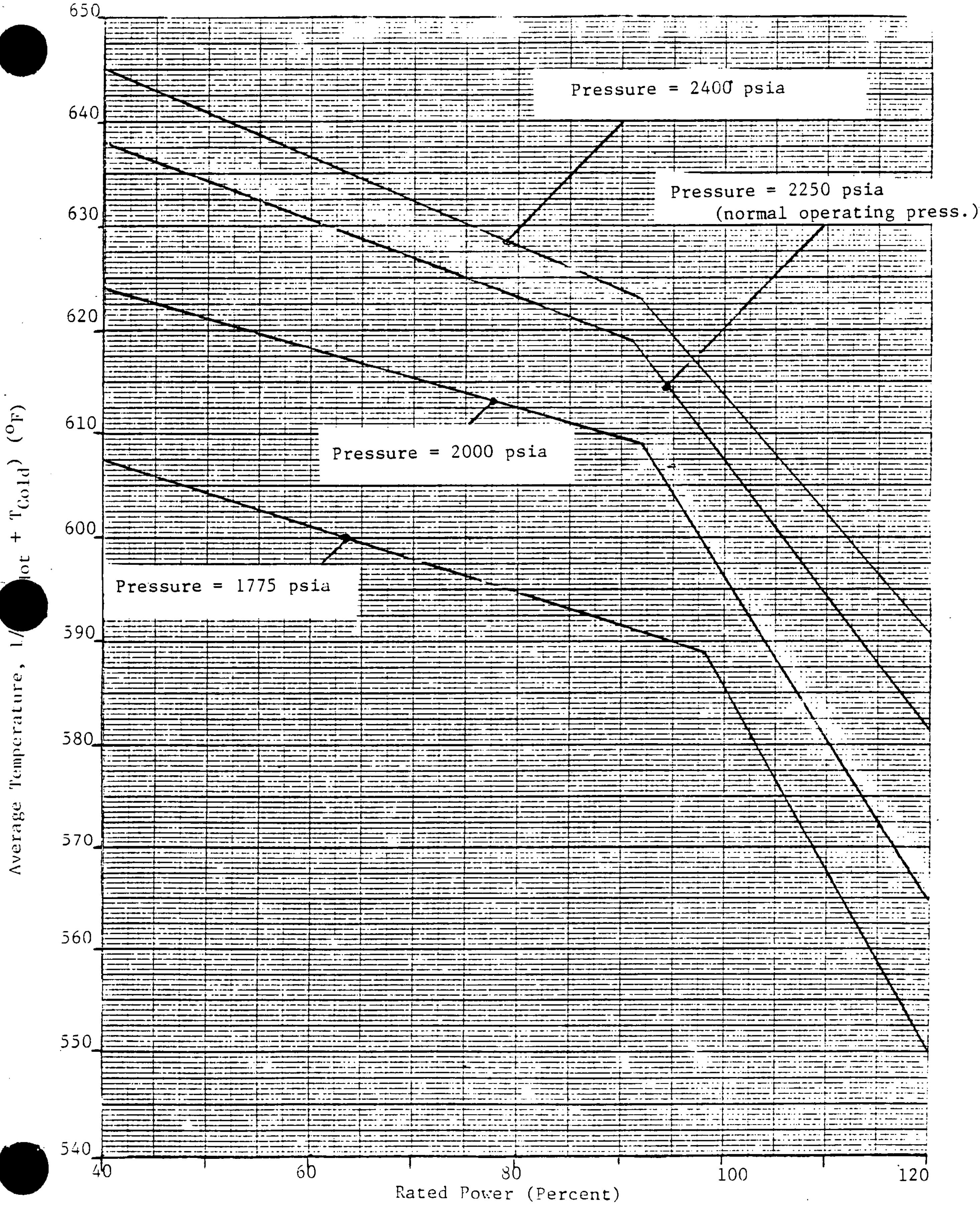
The calculation of these limits includes an $F^N_{\Delta H}$ of 1.55, DNB penalties for increased pellet eccentricity, local power spikes, 8% uncertainty in $F^N_{\Delta H}$, up to 24% steam generator tube plugging, and a reference cosine with a peak of 1.55 for axial power shape.[3]

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H}$ = 1.55 [1 + 0.2 (1-P)] where P is the fraction of rated power.[3]

The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits for four loop operation as dictated by Figure 3.10-4, insures that the DNBR is always greater at partial power than at full power.[3]

Indian Point Unit 3
Reactor Coolant Flow > 323600 GPM
24 Percent Tube Plugging



100 Percent Rated Power is Equivalent to 3025 MW_{th}

Amendment No.

Figure 2.1-1

D E L E T E D

Figure 2.1-2. Core Limits - Three Loop Operation

Amendment No.

ΔT_o = Indicated ΔT at rated power, $\leq 63.5^oF$

T_{avg} = Average Temperature, ^oF

T' = Indicated T_{avg} at nominal conditions at rated power, 574.7^oF

P = Pressurizer pressure, psig

P' = Indicated nominal pressurizer pressure at rated power = 2235 psig

$K_1 \leq 1.135$

$K_2 = 0.0114 \pm .00057$

$K_3 = 0.00066 \pm .0000033$

K_1 is a constant which defines the over temperature ΔT trip margin during steady state operation if the temperature, pressure and f (ΔI) terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT set point to T_{avg}

K_3 is a constant which defines the dependence of the overtemperature ΔT set point to pressurizer pressure.

ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.

$f(\Delta I)$ = 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, where q_t and q_b are as defined above such that:

(a) for $q_t - q_b$ within -20, + 10 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds + 10 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 6.0 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -20 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T') - f(\Delta I)\right]$$

where

ΔT_o = indicated ΔT at rated power, (100% full power measured ΔT, no greater than 63.5° F.)

T_{avg} = average temperature, °F

T' = indicated T_{avg} at nominal conditions at rated power, $\leq$ 574.7°F

K_4 $\leq$ 1.089

K_5 = 0 for decreasing average temperature

$\geq$ 0.175 sec/°F for increasing average temperature

K_6 = 0 for $T \leq T'$

$\geq$ 0.00116 for $T > T'$

K_4 is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature and the $f(\Delta I)$ terms are zero.

K_5 is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.

K_6 is a constant which defines the dependence of the overpower ΔT setpoint to T_{avg}.

$f(\Delta I)$ = as defined above.

$\frac{dT_{avg}}{dt}$ = 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 — $\geq$ 55.0 cps

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

3 LIMITING CONDITIONS FOR OPERATION

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System; operational components; heatup, cooldown, criticality, activity, chemistry and leakage.

Objective

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

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

   a. At least one reactor coolant pump or one residual heat removal pump in the Residual Heat Removal System when connected to the Reactor Coolant System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   b. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

   c. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. 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.

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 reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approval for less than four loop operation at power levels above 10% rated power has been granted by the Commission. (See license condition 2.C.(3))

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the ~~maximum~~ surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

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:

$$F_Q(Z) \leq (2.14/P) \times K(Z) \text{ for } P > 0.5$$
$$F_Q(Z) \leq (4.28) \times K(Z) \text{ for } P \leq 0.5$$
$$F^N_{\Delta H} \leq 1.55 \, [1 + 0.2 \, (1-P)]$$

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.

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 2.14 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined consistent with Appendix K criteria and is satisfied by 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 this upper bound normalized envelope of Figure 3.10-2 demonstrate a peak clad temperature of 1995°F, which is below peak clad temperature limit of 2200°F.[2]

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 $F_{\Delta H}^{N}$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^{N} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape 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 2.14 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.

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

the 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

D E L E T E D

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

Amendment No. ~~14~~

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

SAFETY LIMIT, REACTOR CORE

AND

CONTROL ROD AND POWER DISTRIBUTION LIMITS,

SHUTDOWN REACTIVITY

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

Section I - Description of Modification to the Technical Specifications

The proposed changes to the Technical Specifications are shown in Attachment I. Sections 2.1, 2.3, 3.1 and 3.10 of the Technical Specifications have been revised. These proposed changes delete references to three loop operation and incorporate the results of a recent ECCS reanalysis which modifies the reactor core limits.

The Westinghouse ECCS reanalysis report is based on a steam generator tube equivalent plugging level of twenty-four (24) percent and is enclosed as Attachment III to this submittal.

Section II - Purpose of Modification to the Technical Specifications

The purpose of the modification is to revise the IP-3 Technical Specifications so as to comply with the ECCS reanalysis for up to 24% steam generator equivalent tube plugging. References to three loop operating have been deleted from the Technical Specifications since IP-3 is restricted to 10% power or less for less than four loop operation.

Section III - Impact of the Change to the Technical Specifications

The proposed changes to the Technical Specifications do not require modifications of any system or subsystem. This change will permit plant operations with a higher percentage of steam generator tubes plugged. This requires a lower heat flux peaking factor value to meet peak clad temberature requirements, based on the approved Westinghouse 1981 model. The modification as proposed will not impact the ALARA or Fire Protection Program at IP-3, nor will it adversely impact the environment.

Section IV - Implementation of the Modification to the Technical Specifications

The proposed changes to the Technical Specifications are included as Attachment I to this letter.

Section V - Conclusion

The incorporation of these modifications: a) will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report: b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; and d) does not constitute an unreviewed safety question.

Section VI - References

(a) IP-3 FSAR

(b) IP-3 SER

ATTACHMENT III

EMERGENCY CORE COOLING SYSTEM REANALYSIS FOR STEAM GENERATOR TUBE PLUGGING LEVEL OF TWELVE PERCENT

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

The loss of Coolant Accident (LOCA) has been reanalyzed for Indian Point Unit III with 24% S.G. tubes plugged. The following information amends Safety Analysis Report section on Major Reactor Coolant System Pipe Ruptures. The results are consistent with acceptance criteria provided in reference 1.

The description of the various aspects of the LOCA analysis is given in WCAP-8839[2]. The individual computer codes which comprise the Westinghouse Emergency Core Cooling System (ECCS) evaluation model are described in detail in separate reports[3-6] along with code modifications specified in references 7, 9, 10, 11, 12, 13 and 14. The analysis presented here was performed with the 1981 version of the evaluation model which includes modifications delineated in reference 16.

Results

The analysis of the loss of coolant accident is performed at 102 percent of the licensed core power rating. The peak linear power and total core power used in the analysis are given in Table 2. Since there is margin between the value of peak linear power density used in this analysis and the value of the peak linear power density expected during plant operation, the peak clad temperature calculated in this analysis is greater than the maximum clad temperature expected to exist.

Table 1 presents the occurence time for various events throughout the accident transient.

Table 2 presents selected input values and results from the hot fuel rod thermal transient calculation. For these results, the hot spot is defined as the location of maximum peak clad temperatures. That location is specified in Table 2 for each break analyzed. The location is indicated in feet which presents elevation above the bottom of the active fuel stack.

Table 3 presents a summary of the various containment systems parameters and structural parameters which were used as input to the COCO computer code[6] used in this analysis.

Tables 4 and 5 present reflood mass and energy releases to the containment, and the broken loop accumulator mass and energy release to the containment, respectively.

The results of several sensitivity studies are reported[3]. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Figures 1 through 17 present the transients for the principle parameters for the break sizes analyzed. The following items are noted:

Figures 1A - 3C: Quality, mass velocity and clad heat transfer coefficient for the hotspot and burst locations

Figures 4A - 6C: Core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet to the pressure just beyond the core outlet

Figures 7A - 9C: Clad temperature, fluid temperature and core flow. The clad and fluid temperatures are for the hot spot and burst locations

Figures 10A - 11C: Downcomer and core water level during reflood, and flooding rate

Figures 12A - 13C: Emergency core cooling system flowrates, for both accumulator and pumped safety injection

Figures 14A - 15C: Containment pressure and core power transients

Figures 16, 17: Break energy release during blowdown and the containment wall condensing heat transfer coefficient for the worst break

Containment Pressure Relief

Branch Technical Position CSB6-4 states that the evaluation of a containment pressure relief system design should include "an analysis of the reduction in containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination." An analysis has been performed for Indian Point Unit 3 based on the limiting FAC analysis case (DECLG break, C_D = 0.4) which was obtained using the 1981 Westinghouse Evaluation Model.

Valves in the containment pressure relief system will close shortly after the beginning of a postulated LOCA transient based on the response to the containment isolation signal. The containment pressure relief at Indian Point Unit 3 consists of a single 10-inch pressure relief line.

This line is conservatively represented in the analysis by the following model:

1. The frictional resistance associated with duct entrance and exit bases, filters, duct work bends and skin friction has not been considered.
2. Fan coastdown effects are ignored.
3. Steady-state flow is immediately established through the purge system ducts at the inception of the LOCA.
4. A 3.5 second valve closure time is considered. No credit is taken for the reduction in flow area with time as the valve moves towards the fully closed position.

A mixture of steam and air will pass through the containment pressure relief lines during the time that the isolation valves are assumed to remain open. The effects of varying the exhaust gas composition have been investigated by considering two extreme cases, air flow exclusively and steam flow exclusively. For the purposes of this analysis it was conservatively assumed that critical flow will be established thru the pressure relief lines at the inception of the LOCA and will be maintained until valve closure time.

Equation (4.18) in reference (17) was used to calculate the critical flow of air thru the maximum available area (10" diameter/line). Figure 14 of reference (18) was used to establish the critical flow rate of steam through the pressure relief lines. The total mass released during the time in which the valves are assumed to be open is calculated as 247.5 lbs. of air or 178.5 lbs. of steam.

The reduction in containment pressure from the calculated mass loss is less than 0.1 psi in the case of either air flow or steam flow. A containment pressure reduction of this magnitude on the calculated peak clad temperature (PCT) is expected to be minor (less than 1.0°F).

If consideration of the effects of containment pressure relief on LOCA is applied to the 24% tube plugging case (FQ=2.14), no additional reduction in peaking is necessary.

Conclusions - Thermal Analysis

For breaks up to and including the double ended severance of a reactor coolant pipe, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46.[1] That is:

1. The calculated peak clad temperature does not exceed $2200^{\circ}F$ based on a total core peaking factor of 2.20

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircalloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 10CFR50.46. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., Massie, H. W., And Zordan, T. A., "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, July 1974.

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary Version), WCAP-8306 (Non-Proprietary Version), June 1974.

4. Bordelon, F. M., Et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary Version), WCAP-8305 (Non-Proprietary Version), June 1974.

5. Kelly, R. D., et al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)." WCAP-8170 (Proprietary Version), WCAP-8171 (Non-Proprietary Version), June 1974.

6. Bordelon, F. M., and Murphy E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

7. Bordelon F. M., et al., "The Westinghouse ECCS Evaluation Model: Supplementary Information," WCAP-8471 (Proprietary Version), WCAP-8472 (Non-Proprietary Version), January 1975.

8. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary Version), WCAP-8356 (Non-Proprietary Version), July 1974.

9. "Westinghouse ECCS Evaluation Model, October, 1975 Versions," WCAP-8622 (Proprietary Version), WCAP-8623 (Non-Proprietary Version), November, 1975.

10. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassalo of the Nuclear Regulatory Commission, letter number NS-CE-924, January 23, 1976.

11. Kelly, R. D., Thompson, C. M., et. al., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation With One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166, February, 1978.

12. Eicheldinger C., "Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-A (Non-Proprietary Version), February, 1978.

13. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-1981, Nov. 1, 1978.

14. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-2014, Dec. 11, 1978.

15. Letter from T. M. Anderson of Westinghouse Electric written to Darrell G. Eisenhut of the Nuclear Regulatory Commission, letter number NS-TMA-2165, December 16, 1979.

16. NS-TMA-2448

17. Shapiro, A. H. The Dynamils and Thermodynamics of Compressible Fluid Flow, Volume 1, p. 85.

18. 1967 ASME Steam Tables, p. 301.

[Westinghouse Letter to NRC]