June 6, 1989

Docket No. 50-286

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 71938)

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 20, 1989, as supplemented February 2, February 15, May 5, and May 23, 1989.

The amendment revises The Technical Specifications to reflect changes proposed as a result of the transition to Westinghouse 15 X 15 Vantage 5 fuel from Westinghouse 15 X 15 low-parasitic (LOPAR) assemblies and 15 X 15 optimized Fuel Assemblies (OFAS). The enclosed Safety Evaluation also addresses the RTD Bypass Manifold Elimination modification.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY

Joseph D. Neighbors, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 86 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

[AMEND IP3 71938]

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cc

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Power Authority of the State
of New York

Indian Point Nuclear Generating
Unit No. 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated January 20, 1989, as supplemented February 2, February 15, May 5, and May 23, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald S. Brinkman for

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects, I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 6, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-2	1-2
2.1-1	2.1-1
2.1-2	2.1-2
Figure 2.1-1	Figure 2.1-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-4	2.3-4
2.3-5	2.3-5
2.3-6	2.3-6
3.1-2	3.1-2
3.1-3a	3.1-3a
3.1-3b	3.1-3b
3.1-3c	
3.2-1	3.2-1
3.2-4	3.2-4
3.3-2	3.3-2
3.3-15	3.3-15
3.6-1	3.6-1
3.6-2	3.6-2
3.6-3	3.6-3
3.8-1	3.8-1
3.8-3	3.8-3
3.8-4	3.8-4
3.8-5	3.8-5
3.8-6	3.8-6
3.10-1	3.10-1
3.10-2	3.10-2
3.10-5	3.10-5
3.10-8	3.10-8
3.10-9	3.10-9
3.10-10	3.10-10
3.10-11	3.10-11
3.10-I3	3.10-13
Figure 3.10-2	Figure 3.10-2
5.3-2	5.3-2

1.2.3 Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

1.2.4 Power Operation Condition

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

1.2.5 Refueling Operation Condition

When the reactor is subcritical by at least 5% $\Delta k/k$ and T_{avg} is $\leq 140^{o}F$ and core alterations are being made with the head completely unbolted.

1.3 REFUELING OUTAGE

An outage in which core alterations are performed in order to compensate for fuel burnup.

1.4 CORE ALTERATION

The addition, removal, relocation or other movement of fuel, controls, or installed equipment or material in a reactor core, except for functions normally performed during conventional reactor operation in accordance with intended design of equipment, such as control rod or instrument detector movement or performance of flux scans.

1.5 OPERABLE

Properly installed in the system and capable of performing the intended functions in the intended manner as verified by testing and tested at the frequency required by the Technical Specifications. Implicit in this definition shall be the assumption that all necessary attendant controls, electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

2.0 Safety Limits and Limiting Safety System Settings

2.1 Safety Limits, 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 vessel inlet temperature and power level is at any time above the appropriate pressure line.

Basis

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation for Westinghouse Optimized fuel. This relation 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 DNB design basis is as follows: There must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I (normal operation and operational transients) and Condition II (events of moderate frequency) 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% probability with 95% confidence that 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 applicable DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. The 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.

The curves of Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for 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.

The calculation of these limits includes:

1. $F^N_{\Delta H}$ of 1.56

2. an equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24%, (2)

3. a reactor coolant system total flow rate of greater than or equal to 332,240 gpm as measured at the plant,

4. a reference cosine with a peak of 1.55 for axial power shape.

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

$$F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1-P)]$$

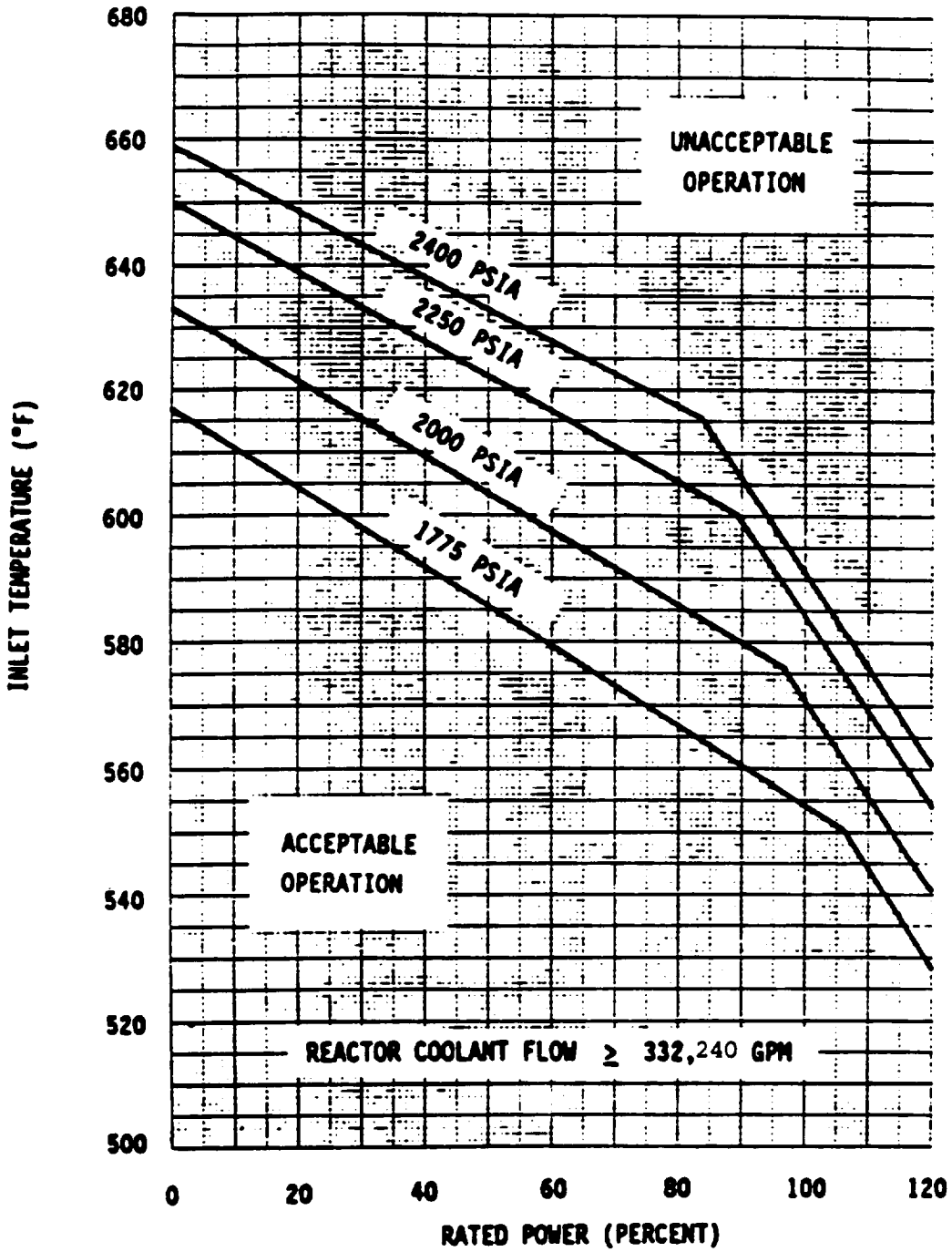
Where P is the fraction of Rated Thermal Power.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.6% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (Figure 3.10-4) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

<u>References</u>

1. FSAR Section 3.2.2.



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWT

Amendment No. 86

FIGURE 2.1-1

ΔT_o ≤ Measured full power ΔT for the channel being calibrated, °F

T_{avg} = Average Temperature for the channel being calibrated, °F (input from instrument racks)

T' = Measured full power T_{avg} for the channel being calibrated, °F

P = Pressurizer pressure, psig (input from instrument racks)

P' = 2235 psig (i.e., nominal pressurizer pressure at rated power)

K_1 ≤ 1.35

K_2 = 0.0212

K_3 = 0.000981

K_1 is a constant which defines the overtemperature 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 setpoint 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(ΔI) = 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 defined above such that:

(a) for $q_t - q_b$ below 6 percent, f(ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.6 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta To\ (K_4 - K_5 \frac{dT_{avg}}{dt} - K_6(T_{avg} - T'))$$

where

$\Delta T_o \leq$ measured full power ΔT for the channel being calibrated, oF

T_{avg} = measured average temperature for the channel being calibrated, ^oF (input from instrument racks)

T' = measured full power T_{avg} for the channel being calibrated, ^oF (can be set no higher than 573.3 ^oF)

K_4 ≤ 1.073

K_5 = 0 for decreasing average temperature

≥ 0.175 sec/^oF for increasing average temperature

K_6 = 0 for $T \leq T'$

≥ 0.00116 for $T > T'$

K_4 is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature term is 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}

$\frac{dTavg}{dt}$ = 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

C. Other reactor trips

(1) High pressurizer water level - ≤ 92% of span.

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

(3) Anticipatory reactor trip upon turbine trip.

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

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

(1) Power range nuclear flux ≥ 10% of rated power, or

(2) Turbine first stage pressure ≥ 10% of equivalent full load.

The reactor trip on turbine trip may be blocked at power levels ≥10% during turbine overspeed surveillance testing.

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates ≤ 50% of rated power. The single loop loss-of-flow trip setpoint is hereafter referred to as P-8.

Basis

The high flux reactor trip provides 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 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 detectors (about 3.5 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 values of the constants K_1, K_2 and K_3 are determined during the design of the core for operation with all reactor loops in service. The values are then specified for the reactor protection system manufacturer and for calibration. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and the applicable safety limit DNBR will not be violated.

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 (via the overall gain in the rate controller) 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 values of the constants K_4, K_5, and K_6 are determined during the design of the core and the reactor protection system. The values are then specified for the reactor protection system manufacturer and for calibration.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. Fuel temperature decreases due to cladding creepdown with burnup and consequential reduction of pellet-cladding gap. Thus overpower limits become less restrictive as fuel burnup proceeds.

The T' values represent the measured full power T_{avg} for the overtemperature and overpower Delta-T equations. T' must correspond to the indicated full power T_{avg}, and may only be set as high as 573.3° F if the plant operates at the design full power T_{avg}. Reducing T' for a lower (than design) full power T_{avg} assures that the overtemperature and overpower delta-T setpoint are decreased for any increase in T_{avg} above the indicated loop full power T_{avg}.

The constants delta-T_o and T' for each overtemperature and overpower protection channel are set in accordance with the measured delta-T and T_{avg} at rated power existing in the loop from which the process inputs for a particular protection channel are supplied. This is done to account for loop to loop differences in delta-T and T_{avg} which may exist as a result of asymmetric steam generator tube plugging.

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. This 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 points 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 the P-8 setpoint for four-loop operation, 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 applicable safety limit during normal operational transients.

A turbine trip causes a direct reactor trip, when operating at or above 10% power. This anticipatory trip will operate in advance of the pressurizer high pressure reactor trip to reduce the peak Reactor Coolant System pressure. No credit was taken in the accident analyses for operation of this trip. (10)

The turbine and steam-feedwater flow mismatch trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14).

Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is above the cold shutdown condition except during reactor coolant system hydrostatic tests and/or safety valve settings.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ±1% allowance for error.

3. Pressurizer Heaters

Whenever the reactor is above the hot shutdown condition, the pressurizer shall be operable with at least 150 kw of pressurizer heaters.

a. With less than 150 kw of pressurizer heaters operable, restore the required inoperable heaters within 72 hours or be in at least hot shutdown within an additional 6 hours.

4. Power Operated Relief Valves

Whenever the reactor coolant system is above 400°F, the power operated relief valves (PORVs) shall be operable or their associated block valves closed.

a. If the block valve is closed because of an inoperable PORV, the control power for the block valves must be removed.

b. If the above conditions cannot be satisfied within 1 hour, be in at least hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

5. Power Operated Relief Block Valves

Whenever the reactor coolant system is above 400°F, the motor operated block valves shall be operable or closed.

a. If the block valve is inoperable, the control power is to be removed.

b. If the above conditions cannot be satisfied within 1 hour be in at least hot shutdown within the following 30 hours.

6. Reactor Coolant System T_{avg}

During steady state operation, the maximum indicated T_{avg} shall not exceed 578.3°F.

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.

The limit on maximum indicated T_{avg} provides assurance that RCS temperatures are maintained within the normal steady state envelope of operation assumed in the safety analyses performed to support the Vantage 5 fuel reloads with asymmetric tube plugging among steam generators. A maximum full power T_{cold} of 547.9°F (including control deadband and measurement uncertainties) was assumed in these safety analyses restricting the maximum indicated T_{avg} to 578.3°F assures that a T_{cold} of 547.9°F is not exceeded at a measured flow of 332,240 gpm when considering asymmetric tube plugging among steam generators.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of a least one reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirments of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the 10 CFR 50, Appendix G, limits. "Arming" means that the motor operated valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to 326°F or manually by the control room operator.

Due to the rate of energy transferred to the RCS, when the RCP is started, the resultant rate of temperature rise and the pressure increase are strongly dependent on the temperature difference between the RCS and the steam generators. The presence of a pressurizer bubble provides for a more moderate pressure increase. The bubble size is sufficient to prevent the RCS from going water solid for 10 minutes during which time operator action will terminate the pressure transient. Pressurizer level refers to indicated level and includes instrument uncertainty. The preventive measures for a Mass Input initiating event (i.e., SI pump or charging pumps) as well as the Heat Input initiating event are described in References (3), (4) and (5)). (Also refer to Specification 3.3.A.8. Safety Injection and Residual Heat Removal Systems). The OPS need not be operable when the RCS temperature is less than 326°F if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. This opening is adequate to relieve the worst case analyzed.

The OPS arming temperature of 326°F permits the performance of an RCS hydrostatic test (see Fig. 4.3-1) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above 370°F. This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature 370°F, the pressurizer safety valves will preclude violation of the 10 CFR 50, Appendix G, curves. In addition, the OPS need not be operable upon satisfying the conditions of Specification 3.1.A.8.b. (3) which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification 3.1.A.8.b(3) also places restrictions on the number of SI pumps capable of feeding the RCS (see Specification 3.3.A.8). An SI pump can be rendered incapable of feeding the RCS is, for example, its switch is in the trip pull-out position, or if at least one valve in the flow path from the SI pump to the RCS is closed and locked (if manual) or de-energized (if motor operated).

This section has also been revised in accordance with the results of tests conducted on the capsule "T" specimens (Reference 6).

References

1) FSAR Section 14.1.6
2) FSAR Section 14.1.8
3) Letter dated 10/25/78 "Summary of Changes to IP-3 Plant Operating Procedures in Order to Preclude RCS Overpressurization"
4) Letter dated 2/28/76 "Conceptual Design of the Reactor Coolant Overpressure Protection System" and response to NRC questions.
5) IP-3 Low Temperature Overpressurization Protection System Analysis, NYPA Report dated 8/24/84.
6) WCAP-9491 "Analysis of Capsule T from IP-3 Reactor Vessel Radiation Surveillance Program", J.A. Davidson, S.L. Anderson, W. T. Kaiser, April 1979.

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 brought above the cold shutdown condition unless the following requirements are met:

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable one of which shall be operating to recirculate the contents of the Boron Injection Tank.

3. The boric acid storage system shall contain a minimum of 5000 gallons of 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. 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 and a recirculation flow path between the boric acid storage system and the Boron Injection Tank.

5. The appropriate boric acid storage tank level indicator(s) and the Boron Injection Tank recirculation flow indicator shall be operating.

6. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system to the Reactor Coolant System.

Continuous recirculation between the boric acid storage system and the boron injection tank, and operability of the heat tracing circuit of the recirculation line insures that a flow path exists from the boric acid storage system to the boron injection tank.

A minimum volume of 5000 gallons of the 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) of boric acid are required to meet cold shutdown conditions. An upper concentration limit of 13% (22,735 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 channel of heat tracing is sufficient to maintain the specified low temperature limit. The second channel of heat tracing provides backup for continuous plant operation when one channel is inoperable. Should both channels of heat tracing become inoperable, the reactor will be shutdown and can easily be borated before the line temperature is reduced near the boric acid precipitive temperature.

The city water system is used as a source of water for emergency cooling of the charging pumps and as a source of flush water to remove concentrated boric acid from the piping between the outlet of the boric acid storage tanks and the inlet to the charging pumps in the unlikely event of a complete loss of electrical power and/or a complete loss of service water resulting from turbine missiles.

<u>References</u>

1) FSAR - Section 9.2
2) FSAR - Section 6.2

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within the next 20 hours.

3. The reactor coolant system T_{avg} shall not exceed 350°F unless the following requirements are met:

a. The refueling water storage tank contains a minimum of 346,870 gallons of water at a boron concentration of at least 2000 ppm.

b. The boron injection tank contains 900 gallons of a boric acid solution of 11-1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) at a temperature of at least 145°F. Two channels of heat tracing shall be operable for that portion of the flow path bounded by the boron injection tank inlet and outlet motor operated valves and the recirculation flow path to and from the boric acid tanks.

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 775 ft^3 and a maximum of 800 ft^3 of water at a boron concentration of at least 2000 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies de-energized whenever the reactor coolant system pressure is above 1000 psig.

Assuming the reactor has been operating at full rated power, 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 effect repairs.

Failure to complete repairs within 1 hour 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.

The limits for the Boron Injection Tank, Refueling Water Storage Tank, and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis. (9) (13)

The specified quantities of water for the RWST include unavailable water (4687 gals) in the tank bottom, inaccuracies (1406 gals) in the alarm setpoints, and minimum quantities required during injection (246,000 gals) (3) and recirculation phases (80,000 gals). (4) The minimum RWST (e.g., 346,870 gals) provides approximately 13,370 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 four accumulator isolation valves (894 A.B,C,D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phases of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator de-energized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating statue of the reactor containment for plant operation.

Specification

A. Containment Integrity

1. The containment integrity (as defined in 1.10) shall not be violated unless the reactor is in the cold shutdown condition. However, those non-automatic valves listed in Table 3.6-1, may be opened if necessary for plant operation and only as long as necessary to perform the intended function.

2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin equal to or greater than the requirements of specification 3.8.D.

3. If the containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within one hour or the reactor shall be in the hot shutdown condition within six hours and in cold shutdown condition within the next 30 hours.

B. Internal Pressure

If the internal pressure exceeds 2.5 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.

D. Containment Vent and Purge System

The reactor shall not be taken above the cold shutdown condition unless the containment vent isolation valves (PCV - 1190, -1191, -1192) are closed or limited to a maximum valve opening angle of 60° (90° = full open) by mechanical means.

The reactor shall not be taken above the cold shutdown condition unless the containment purge supply and exhaust isolation valves (FCV-1170, -1171, -1172, -1173) are closed.

If the above conditions cannot be met within one hour, the reactor shall be in the hot shutdown condition within six hours and in the cold shutdown condition within the next 30 hours.

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 on the type of activities that are being carried out. The shutdown margin requirement of specification 3.8.D when the vessel head bolts are less than fully tensioned precludes criticality during refueling. When the reactor head is not to be removed, the specified shutdown margin of 1% $\Delta k/k$ precludes criticality at cold shutdown conditions.

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 6.4 psig. (1) The containment can withstand an internal vaccum of 3 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 temperatures of the containment liner is well above the NDT + 30°F criterion for the linear material. (3)

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function.[4] 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 protection, safeguards or essential service systems. Those valves to be open intermittently are under administrative control and are open only as long as necessary to perform their intended function. In all cases, however, the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

The opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to a maximum opening angle of 60°. The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown.

REFERENCES

(1) FSAR - Volume 7, Response to Question 14.6
(2) FSAR - Appendix 5A, Section 3.1.8
(3) FSAR - Section 5.1.1.1
(4) FSAR - Section 5.2

3.8 Refueling, Fuel Handling and Storage

Applicability

Applies to operating limitations during refueling, fuel handling, storage operations, and when heavy loads are moved over the reactor when the head is removed.

Objective

To ensure that no incident could occur during refueling, fuel handling, and storage operations that would adversely affect public health and safety.

Specification

A. During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met:

1. The equipment door and at least one door in each personnel air lock shall be properly closed. When the closure plate with a personnel door that prevents direct air flow from the containment is used, it shall be properly closed.

2. At least one isolation valve shall be operable, locked closed or blind flanged in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

3. Radiation levels in the containment and spent fuel storage areas shall be monitored continously.

4. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service.

5. At least one residual heat removal pump and heat exchanger shall be operating except during those core alternations in which the residual heat removal flow interferes with component positioning.

6. During reactor vessel head removal and while loading and unloading fuel in the reactor, T_{avg} shall be < 140°F.

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

a. No. 31 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

b. No. 32 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

c. The water level in the refueling cavity above the top of the reactor vessel flange is equal to or greater than 23 feet.

B. If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

C. During fuel handling and storage operations, the following conditions shall be met:

1. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used.

2. The spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, no loads in excess of 2,000 pounds shall be moved over any region of the spent fuel pit.

3. During periods of spent fuel cask or fuel storage building cask crane movement over the spent fuel pit, or during periods of spent fuel movement in the spent fuel pit when the pit contains irradiated fuel, the pit shall be filled with borated water at a concentration of >1000 ppm.

4. Whenever movement of irradiated fuel in the spent fuel pit is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated in the storage rack.

5. Hoists or cranes utilized in handling irradiated fuel shall be deadload tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the fuel handling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test prior to fuel handling.

6. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the periods of inoperability.

7. Fuel assemblies to be stored in the spent fuel pit can be categorized as either Category 1, 2 or 3 based on burnup and initial enrichment as specified in Figure 3.8-1. Category 2 fuel shall be loaded into the spent fuel pool storage locations in a checkerboard fashion with the intermediate storage locations containing Category 1 fuel, non-fuel materials or left empty. Unless restricted by the above, Category 1 or 3 fuel can be stored in any location in the spent fuel pool.

D. When any fuel assemblies are in the reactor vessel and the reactor vessel head bolts are less than fully tensioned, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A shutdown margin greater than or equal to 5% $\Delta K/K$

or

b. A boron concentration of greater than or equal to 1900 ppm.

The required boron concentration will be verified by chemical analysis daily. With the requirements of the above specification not satisifed, immediately suspend all operations involving core alterations or positive reactivity changes and initiate boration to return to the more restirctive of the limits above.

Basis

The equipment and general procedures to be utilized during refueling, fuel handling, and storage are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling, fuel handling, reactor maintenance or storage operations that would result in a hazard to public health and safety. [1] Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin indicated will keep the core subcritical. During refueling the reactor refueling cavity is filled with approximately 342,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. Periodic checks of refueling water boron concentration and residual heat removal pump operation insure the proper shutdown margin. The requirement for direct communications allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

The minimum boron concentration of this water is the more restrictive of either 1900 ppm or else sufficent to maintain the reactor subcritical by at leat 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.

In addition to the above safeguards, interlocks are utlized 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 120-hour decay time following the subcritical condition and the 23 feet of water above the top of the reactor pressure vessel flange is consistent with the assumptions used in the dose calculation for the fuel-handling accident.

The waiting time of 162 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR. The calculations confirming this are based on an inlet river tempertaure of 92°F, service water flow to the component cooling heat exchangers of 7000 gpm (FSAR) and component cooling flow to the Spent Fuel Pit heat exchanger of 2800 gpm (FSAR).

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45 day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. Fuel Storage Building isolation is actuated upon receipt of a signal from the area high activity alarm or by manual operation. The emergency ventilation bypass assembly is manually isolated, using manual isolation devices, prior to movement of any irradiated fuel. This ensures that all air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers. The ventilation system is tested prior to all fuel handling acitivities to ensure the proper operation of the filtration system.

When fuel in the reactor is moved before the reactor has been subcritical for at least 365 hours, the limitations on the containment vent and purge system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The limit to have at least two means of decay heat removal operable ensures that a single failure of the operating RHR System will not result in a total loss of decay heat removal capability. With the reactor head removed and 23 feet of water above the vessel flange, a large heat sink is available for core cooling. Thus, in the event of a single component failure, adequate time is provided to initiate diverse methods to cool the core.

The minimum spent fuel pit boron concentration and the restriction of the movement of the spent fuel cask over irradiated fuel were specified in order to minimize the consequences of an unlikely sideways cask drop.

Fuel assemblies whose initial enrichment is greater than 3.5 w/o U-235 but less than or equal to 4.3 w/o can be stored in the spent fuel pool providing they are placed in a checkerboard array with fuel whose initial enrichment and burnup are sufficient to ensure that K_{eff} is less than 0.95 with no soluble boron present. This is ensured by categorizing the fuel whose initial enrichment is greater than 3.5 w/o U-235 but less than or equal to 4.3 w/o and whose burnup is below the curve of Figure 3.8-1 as Category 2. This fuel can be stored by checkerboarding with Category 1 fuel which is defined as fuel whose initial enrichment and burnup place it on or above and to the left of th curve in Figure 3.8-1. Category 3 fuel which is less than or equal 3.5 w/o U-235 and below the curve of Figure 3.8-1 cannot be used in the checkerboard with Category 2 fuel. Any Category 1 or 3 fuel can continue to be stored on a repeating x-y array with other non-Category 2 fuel. For the purpose of storing Category 2 fuel, non-fuel material or empty locations can be utilized in place of Category 1 fuel.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated in the bridge rails make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit. Thus, it will be possible to handle the spent fuel cask with the 40-ton hook and to move new fuel to the new fuel elevator with a 5-ton hook, but it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5-ton hook of the fuel storage building crane.

Dead load tests and visual inspection of the hoists and cranes before handling irradiated fuel provide assurance that the hoists or cranes are capable of proper operation.

References

(1) FSAR - Section 9.5.2

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distribution and to 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:

$$F_Q(Z) \leq (2.32/p) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (4.64) \times K(Z) \text{ for } P \leq 0.5$$

$$F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1-P)]$$

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.

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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 When $F_{\Delta H}^{N}$ is measured, no additional allowances are necessary prior to comparison with the limits of section 3.10.2. An error allowance of 4% has been included in the limits of section 3.10.2 If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power 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 for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux differences 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 3.10.2.7 below, the indicated axial flux difference of all but one operable excore channel shall be maintained within a ± 5% band about the target flux difference.

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. 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.

3.10.4 <u>Rod Insertion Limits</u>

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.

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 Control rod 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.

3.10.9 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 Reactivity Balance

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Fuel Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicated reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3.10.11 Notification

Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

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 analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR 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

F^E_Q 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 $F^N_{\Delta H}$.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined consistent with Appendix K criteria and is satisfied for OFA transition mixed cores (3) 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 analysis based on this upper bound normalized envelope of Figure 3.10-2 demonstrates a peak clad temperature not greater than 2049°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 movable 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 an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.62/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

(e.g. rod misalignment) affect $F_{\Delta H}^{N}$, 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 $F_{\Delta H}^{N}$ 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 is less readily available. When a measurement of $F_{\Delta H}^{N}$ is taken, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

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 bases remain inviolate and identify operational anomolies which would, otherwise, affect these bases.

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 demand position. An indicated misalignment limit of 12 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 control rod 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 $F_{\Delta H}^{N}$ 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 referenced 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.

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

-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 FSAR Section 14.1 events, the core is protected from overpower and a minimum DNBR of the applicable safety limit DNBR by an automatic protection system. Compliance with operating procedures is assumed as a precondition for FSAR Section 14.1 events. However, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

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 cause before a severe tilt could build up. 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

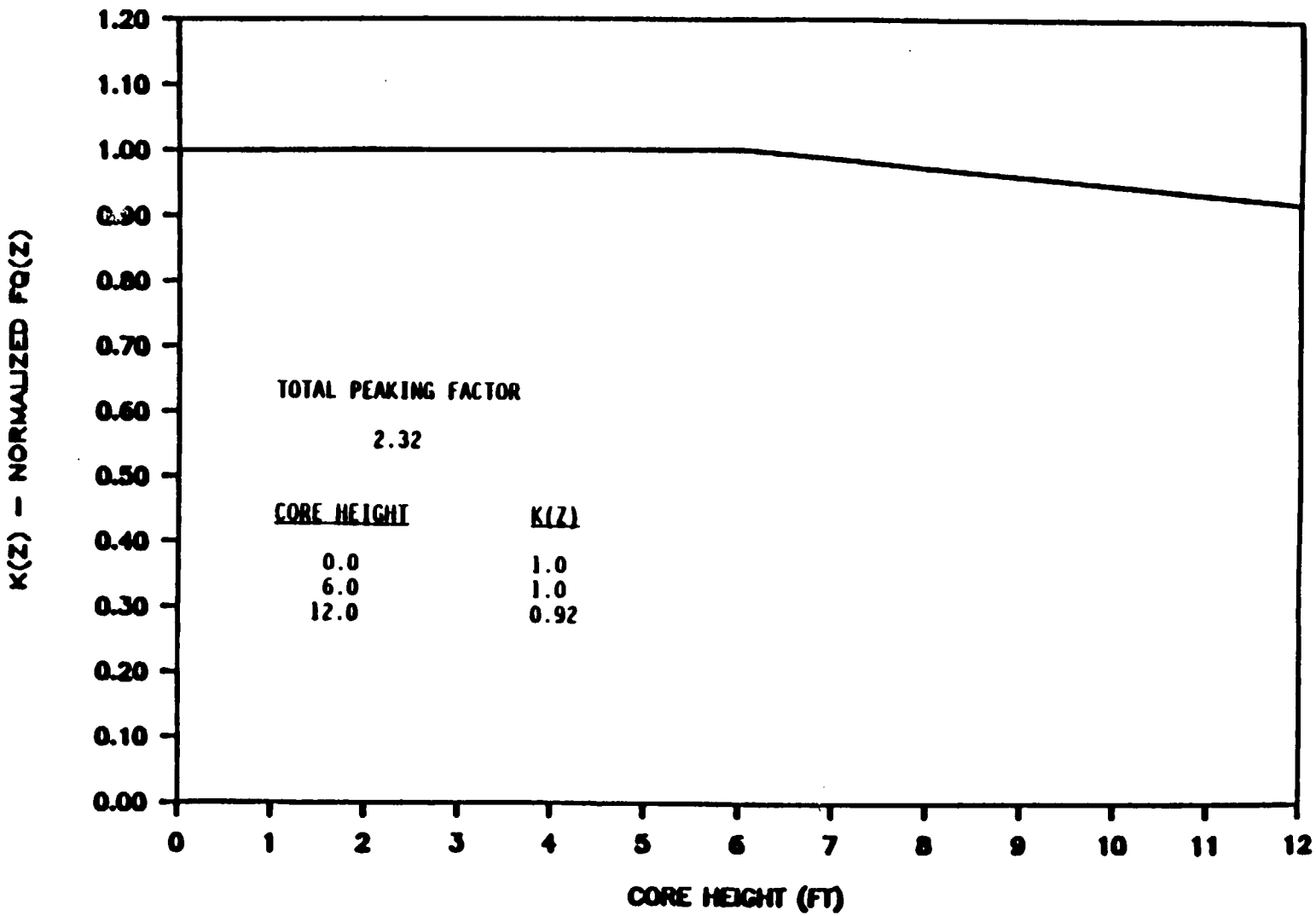

Figure 3.10-2 Hot Channel Factor Normalized Operating Envelope

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel. (5)

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements. (6)

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions and with 0% equivalent steam generator tube plugging, is 11,522 feet.

References

(1) FSAR Section 3.2.2

(2) FSAR Section 3.2.1

(3) FSAR Section 3.2.1

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-9

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated January 20, 1989 (Ref. 1) as supplemented and amended by letters dated February 2, 1989 (Ref. 2) February 15, (Ref. 3), May 5 (Ref. 4), and May 23, 1989 (Ref. 5), the Power Authority of the State of New York (the licensee) made application to amend the Technical Specifications of the Indian Point Nuclear Generating Unit No. 3 (IP-3). The proposed changes will allow for transition to Westinghouse 15x15 VANTAGE 5 fuel and for the RTD (Resistance Temperature Detector) Bypass Manifold Elimination modification.

Indian Point Unit 3 operated Cycle 6 with a transition fueled core of Westinghouse 15x15 low-parasitic (LOPAR) assemblies and 15x15 optimized fuel assemblies (OFAs). Subsequent cycles will be refueled with Westinghouse 15x15 VANTAGE 5 fuel. IP-3 will operate with transition cores containing VANTAGE 5 and OFA fuel until an all VANTAGE 5 core is achieved.

The licensee is planning to eliminate the resistance temperature detector (RTD) bypass manifold. To replace the bypass manifold, thermowell mounted RTDs will be installed directly into the hot and cold leg loop piping. Experience at other Westinghouse plants indicates that using thermowell mounted RTDs minimizes system leakage and thus decreases the maintenance and housekeeping activities associated with the leakage.

The 15x15 VANTAGE 5 fuel assembly is a modification of the current 15x15 LOPAR and the OFA designs. Except for the 15x15 fuel array, no intermediate flow mixer (IFM) grids and the use of the debris filter bottom nozzle (DFNB), the Indian Point Unit 3 15x15 VANTAGE 5 fuel assembly is the same as the standard 17x17 fuel assembly defined in Reference 6, "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A. The major design features of the 15x15 VANTAGE 5 design are (1) Integral Fuel Burnable Absorbers (IFBA), (2) Reconstitutable Top Nozzle, (3) Extended Burnup, (4) Axial Blankets, and (5) Debris Filter Bottom Nozzle. The bottom nozzle will be fabricated from stainless steel rather than Inconel. The size and pattern of the flow holes has been designed to minimize passage of debris particles large enough to cause damage while providing sufficient flow area, comparable pressure drop and continued structural integrity of the nozzle.

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The licensee's report will serve as a reference safety evaluation/analysis for the region by region transition from Cycle 7 to the core fueled with all 15x15 VANTAGE 5 fuel. The transition and VANTAGE 5 core evaluation/analysis were performed at core thermal power level of 3025 MWt. The following assumptions were made in the safety evaluation: (1) a full power $F_{\Delta H}$ of 1.62 (with uncertainties) for both transition and full VANTAGE 5 cores, (2) an increase in the maximum F_Q to 2.32, and (3) up to a 24 percent asymmetric or uniform steam generator tube plugging.

The standard Westinghouse reload design methods described in Reference 7 were used for this evaluation. The licensee reviewed all the non-LOCA accidents to address any impact from the V-5 fuel. The V-5 design features that were considered were:

Reconstitutable Top Nozzle
Debris Filter Bottom Nozzle
Axial Blanket
Integral Fuel Burnable Absorbers
Extended Burnup Fuel Assembly Design
Extreme Low Leakage Loading Pattern

The analysis assumptions that were different from those currently used for IP-3 are:

Increased $F_{\Delta H}$
Improved Thermal Design Procedure (ITDP)
Standardized Chamfer Pellet Design
Refueling Shutdown Margin
Reduced Auxiliary Feedwater Flow

The large break LOCA analysis for IP-3 applicable to a full core of 15x15 V-5 fuel was performed using the NRC approved 1981 Evaluation Model with BART/BASH. The small break LOCA was analyzed using the NOTRUMP code.

The Technical Specification changes for the transition to VANTAGE 5 fuel and the RTD bypass manifold elimination modification were included in Reference 1.

2.0 EVALUATION

2.1 Mechanical Design

The VANTAGE 5 fuel assembly was designed to be compatible with the OFAs, reactor internals interfaces, the fuel handling equipment and the refueling equipment. The design basis and design limits are essentially the same as those for the OFA designs.

2.1.1 Fuel Rod Performance

Fuel rod performance for all fuel rod designs was shown to satisfy the NRC SRP fuel rod design bases on a region by region basis. The V-5 fuel is designed to achieve a higher burnup and to operate with a higher $F_{\Delta H}$ limit than the OFA. Fuel performance evaluations have been completed for each fuel region which demonstrate that the design criteria is satisfied for all fuel rod types under planned operating conditions.

2.1.2 Grid Assemblies Reconstitutable Top Nozzle, Fuel Rods and Axial Blankets

The design bases and evaluations of these features were given in Reference 6, which has been approved by the staff.

2.1.3 Debris Filter Bottom Nozzle

The debris filter bottom nozzle (DFBN) was introduced in order to reduce the possibility of fuel rod damage due to debris induced fretting. Tests were performed to measure pressure drop and demonstrate structural integrity. The results show that this design is totally compatible with the current design.

2.1.4 Mechanical Compatibility of Fuel Assemblies

Based on the evaluation of the VANTAGE 5/OFA design difference and the evaluation of the DFBN, it was concluded that the two designs are mechanically compatible with each other.

The staff has reviewed the fuel rod design for future cores of IP-3 and concludes that it is acceptable because (1) approved codes are used for the evaluations (2) all applicable criteria are satisfied, and (3) the results include consideration of the increased power peaking factor and increased fuel duty.

2.2 Nuclear Design

Evaluations for transition and equilibrium cycle VANTAGE 5 cores in general and for IP-3 have shown that the impact of using V-5 fuel does not cause a significant change in physics parameters beyond the normal range of variation seen from cycle to cycle. Methods and core models used in the IP-3 reload transition are the standard approved ones. There is increased emphasis on three dimensional nuclear models because of the axially heterogeneous nature of the V-5 fuel when axial blankets and reduced length burnable absorbers are used.

Power distributions and peaking factors show slight changes as a result of the use of axial blankets and reduced length burnable absorbers, in addition to normal variations from different loading patterns. Technical Specification changes are proposed for increased $F_{\Delta H}$ and F_Q limits. The increased $F_{\Delta H}$ limit will allow a loading pattern design with lower leakage which in turn allows longer cycles. The increased F_Q limit will provide greater flexibility with regard to the axial heterogeneous cores (blankets and short burnable absorbers). As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied.

The staff has reviewed the nuclear design and concludes that it is acceptable because (1) approved codes and methodologies have been used, (2) acceptable core parameters have been obtained, and (3) appropriate changes to the Technical Specifications have been proposed.

2.3 Thermal and Hydraulic Design

The calculational methods used for the thermal-hydraulic analysis of the transition to all V-5 fuel includes use of the Improved Thermal Design Procedures (ITDP) and the THINC-IV Code. The core was analyzed at 3025 MWt with a 574.7°F vessel average temperature and an $F_{\Delta H}$ limit of 1.62 (including uncertainties). The minimum required DNBR design limit value for the OFA and VANTAGE 5 fuel analyses are 1.33 for thimble cold wall cells and 1.34 for typical cell. The safety analysis DNBR limit of 1.60 for thimble and typical cells for the OFA and V-5 fuel was used in the safety analyses. The existing 16.2 percent margin between the design and safety analysis DNBR limits includes a small (less than 3 percent) amount for flexibility in the design. The fuel temperature used in the safety analysis for V-5 fuel was evaluated using the same methods as those used to evaluate the OFA fuel.

The staff has reviewed the thermal hydraulic design of the reference core and concludes that it is acceptable because (1) approved codes and methodologies have been used, (2) rod bow penalty is offset by the DNBR margin available in the safety limit DNBR, and (3) all of the current thermal-hydraulic design criteria are satisfied.

2.4 Non-LOCA Accidents

The Indian Point Unit 3 licensing basis as reported in the FSAR includes analyses or evaluation of the 15 non-LOCA accidents. The licensee has reviewed all the non-LOCA accidents to find the impact of using the V-5 fuel. The eleven transients which were impacted by one or more of the V-5 design features or modified safety analysis assumptions as listed in Section 1 of this report were reanalyzed. The other four accidents were discussed as to why the V-5 had no impact on them. The accidents that were reanalyzed are:

(1) Uncontrolled Control Rod Withdrawal from a Subcritical Condition
(2) Uncontrolled Control Rod Assembly Withdrawal at Power
(3) Rod Cluster Control Assembly (RCCA) Drop

(4) Chemical and Volume Control System Malfunction
(5) Loss of Reactor Coolant Flow
(6) Loss of External Electrical Load
(7) Loss of Normal Feedwater
(8) Excessive Heat Removal Due to Feedwater System Malfunction
(9) Excessive Load Increase Incident
(10) Loss of all AC Power to the Station Auxiliaries
(11) Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

The uncontrolled control rod withdrawal from a subcritical condition event was analyzed assuming a reactivity insertion greater than that corresponding to two consecutive banks moving with a 100 percent overlap at maximum speed in the maximum worth region. The methods and assumptions used in the analysis were consistent with those used in the FSAR. The results showed that the DNBR remains above the limiting value, and the fuel and clad temperature remain well below the limiting values. Thus, all the applicable safety criteria for the event have been met.

The uncontrolled control rod assembly withdrawal at power results in increased core heat flux and an increase in reactor coolant temperature. Unless terminated, the coolant temperature rise could result in DNB. The event was reanalyzed at a range of power levels from 10 percent to 100 percent and various reactivity insertion rates for both maximum and minimum reactivity feedback cases. The high neutron flux and overtemperature delta-T trips terminated the event. The minimum value of DNBR was always larger than the safety analysis limit value. Thus the DNB design basis has been met.

The rod cluster control assembly drop is initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core. The acceptance criteria for this event is that no fuel failures occur. This transient was analyzed using current approved methodology for turbine runback plants. For all cases analyzed the DNBR criterion was met and thus the applicable safety criteria is satisfied.

The chemical and volume control system malfunction events (boron dilution events) were analyzed during refueling, during startup and at power. The refueling event would require more than 30 minutes before the reactor could go critical. For both the during startup and the at power boron dilution events, more than 15 minutes are available after trip before return to criticality. Thus the design basis for the boron dilution events has been met.

All the loss of flow transients of the FSAR were reevaluated to determine the effect of increased peaking factors and other changes associated with the V-5 fuel transition. Also the analysis considered the effects of steam generator tube plugging for both uniform and asymmetric conditions. The results of the analysis verified that the DNBR remains above the limiting value for the complete loss of flow transient. Loss of a single pump with all loops in service was also analyzed and the results were less severe than the complete loss of coolant flow. Thus the DNB design basis has been met for this event.

The locked rotor event was reanalyzed considering the transition to V-5 fuel at a nominal power of 3025 MWt. The peak reactor coolant pressure was 2568 psia and peak clad temperature was 1845°F. The maximum zirconium steam reaction at the core hot spot was 0.4 percent by weight. The most limiting case showed no rods in DNB. Thus all the safety criteria are met.

The loss of external electrical load and/or turbine trip events were reanalyzed. The results show that loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. The maximum pressures are within the design limits and the reactor protection system maintains the integrity of the core. DNBR is maintained above the safety analyses limit. Thus the design basis for the event are met.

The loss of normal feedwater results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. The analysis has been performed to show that the auxiliary feedwater (AFW) system is adequate to remove stored and residual heat. Results of the analysis show that the minimum AFW capacity of 340 gpm is sufficient to prevent pressurizer filling. This assures that the RCS is not overpressurized and that all safety criteria are met.

The excessive heat removal due to feedwater system malfunction transient is analyzed with the LOFTRAN code. The results and conclusions of the uncontrolled control rod withdrawal from a subcritical condition bound those for the excessive heat removal due to a feedwater system malfunction at no-load conditions. The analysis of the excessive feedwater addition initiated from full power with and without automatic rod control shows no violation of the DNBR limit value. Thus the applicable safety criteria for excessive heat removal due to feedwater system malfunction are met.

The excessive load increase incident is analyzed using the LOFTRAN computer code. Cases are analyzed for both BOC and EOC with both minimum moderator reactivity feedback and maximum moderator reactivity feedback. In all cases

the plant rapidly reaches a stablized condition and the DNBR remains above the safety analysis limit. Thus the applicable safety criteria are met.

The loss of all AC power to station auxiliaries was analyzed using the LOFTRAN code. The results of the analysis show that the minimum AFW capacity of 340 gpm is sufficient to prevent pressurizer filling and any subsequent water relief through the pressurizer relief and safety valves. The analysis also demonstrated that sufficient long term heat removal capability exists to prevent fuel and clad damage. Thus all safety criteria are met.

The rod ejection accident was reanalyzed to ensure that the fuel rod enthalpy, melt and clad temperature criteria would not be violated by the increased peaking factor and other changes associated with V-5 fuel. For the four conditions (EOC-HFP, EOC-HZP, BOC-HFP and BOC-HZP) analyzed the clad average temperatures were all below 2700°F, the fuel enthalpies were below 360 BTU/lb and the fuel melt <10 percent criteria was met in each case. Thus the safety criteria for rod ejection were met.

The staff has evaluated the licensee's evaluation and analysis of the non-LOCA events, using the revised safety analysis assumptions associated with the VANTAGE 5 fuel and concludes that they are acceptable because (1) approved methodologies and computer codes have been used, and (2) all applicable safety criteria have been met.

2.5 Large and Small Break LOCA Analysis

2.5.1 Large Break LOCA

The large break LOCA analysis was performed for the 15x15 VANTAGE 5 fuel. The licensee presented the results which showed that the acceptance criteria were all met. The effects of axial blankets, reconstitutable top nozzle, and the extended burnup features on large break LOCA have been generally evaluated and do not result in changes to the large break LOCA results. The Integrated Fuel Burnable Absorbers (IFBA) feature was specifically evaluated for IP-3. It was determined that no transition core penalty was needed because there is no hydraulic mismatch between the VANTAGE 5 and OFA fuel designs.

2.5.2 Small Break LOCA

Again since no hydraulic mismatch exists, no transition core penalty due to potential flow maldistribution was required. The small break LOCA analysis for IP-3 resulted in a peak clad temperature of 1711°F. The maximum local metal-water reaction was 0.741 percent and the total metal-water reaction was less than 0.3 percent for all cases analyzed. All other acceptance criteria were met.

2.6 Reduced Auxiliary Feedwater

The licensee performed analysis and evaluation to examine the effects of a reduction in the auxiliary feedwater on the following accidents:

Large Break LOCA
Small Break LOCA
Containment Integrity
Steam Generator Tube Rupture

The results showed that for both small and large break LOCA the peak clad temperatures are within the acceptance criteria.

Sensitivity analyses showed that a reduction in auxiliary feedwater flow rate up to 75 percent has a negligible effect on calculated peak containment pressure. The current calculated limiting peak pressure is 39.39 psig and the design pressure is 47.0 psig. Therefore the effect of auxiliary feedwater flow rate will have an insignificant effect on containment integrity.

The licensee stated in its submittal that it has made the following two changes in the assumptions previously used for the steam generator tube rupture accident analysis described in the Indian Point Unit 3 FSAR Section 14.2.4:

(1) An Auxiliary Feedwater (AFW) flow rate is decreased to 300 gpm from 400 gpm by the motor driven AFW pumps and the turbine driven AFW pump was eliminated. This change increased the amount of the reactor primary coolant discharged to the affected steam generator to 95,000 lbs from 75,000 lbs as stated in the FSAR.

(2) The tube uncovery will persist for the entire 30 minute duration of the break flow and assumed an iodine partition factor of 1.0 along with a steam flashing fraction of 1.0. Therefore, all of the primary coolant entering the affected steam generator will be flashed to steam and all of the iodine activity in the coolant is assumed to become airborne. No tube uncovery was assumed in the Indian Point Unit 3 FSAR.

Using the above two changes and assumptions listed in the FSAR Section 14.2.4.4, we calculated the offsite doses due to the steam generator tube rupture accident (see Table 1). As shown in the table, the calculated doses are within the 10 CFR Part 100 dose reference values used as an acceptance criterion in 1973 (the time frame of the original SER). Therefore, we find that the licensee's proposed license amendment to use Westinghouse VANTAGE 5 fuel in the area of the offsite radiological consequences of design basis accidents is acceptable.

2.7 Conditions on Use of VANTAGE 5 Fuel

During the staff review of the VANTAGE 5 fuel designs in WCAP-10444-P-A, the staff identified several conditions to be resolved for use of VANTAGE 5 fuel. Our review of the conditions listed in the safety evaluation of WCAP-10444-P-A follows.

2.7.1 Statistical Convolution Method

In our SER on WCAP-10444, we stated that the statistical method should not be used in VANTAGE 5 for evaluating the fuel rod shoulder gap. The licensee stated that worst case fabrication tolerances and fuel rod and assembly growth were used to determine the initial fuel rod to nozzle growth gaps. We consider this acceptable.

2.7.2 Seismic and LOCA Loads

In our SER on WCAP-10444, we stated that for each plant application, it must be demonstrated that the fuel assembly will maintain its coolable geometry under combined seismic and LOCA loads. The licensee stated that an evaluation has been performed and that the grid load results show that for both OFA and V-5 fuel all SRP criteria are met. Thus we consider this acceptable.

2.7.3 Irradiation Demonstration Program

In our SER on WCAP-10444 we required that an irradiation program be performed to confirm the V-5 fuel performance. In addition to the demonstration of 17x17 V-5 fuel, demonstration fuel rods of the 15x15 design were irradiated in Turkey Point 3 and 4. Turkey Point 4 currently operates with a fuel reload of fuel with V-5 features. We consider this acceptable.

2.7.4 Improved Thermal Design Procedure (ITDP)

In our SER on WCAP-10444, we stated that those restrictions in approving the use of Westinghouse improved thermal design procedure (ITDP) should be applied to the V-5 fuel design. The licensee submitted a report WCAP-12128 Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for New York Power Authority Indian Point Unit 3 (Ref. 3). Since the restrictions were followed, we thus consider this acceptable.

2.7.5 DNBR Limit

In our SER on WCAP-10444 we stated that data and analysis are required for 14x14 or 15x15 fuel with an appropriate DNBR limit. Since IP-3 is using V-5 fuel without intermediate flow mixer (IFM) grids, the WRB-1 correlation with a DNBR limit of 1.17 is appropriate. We conclude that this is acceptable.

2.7.6 Transition Core Penalty

In our SER on WCAP-10444 we stated that separate analysis will be required to determine a transitional mixed core penalty. The licensee has addressed this issue and since there is no hydraulic mismatch, a transitonal core penalty due to potential flow mismatch is not required. We find this acceptable.

2.7.7 Higher Peaking Factor

In our SER on WCAP-10444 we stated that specific plant analysis should be performed to demonstrate that the DNBR limit will not be violated with higher $F_{\Delta H}$ and F_Q. The licensee's analysis shows that the DNBR limits are met with the higher peaking factors. This is acceptable.

2.7.8 Reactor Coolant Pump Shaft Seizure

In our SER on WCAP-10444, we stated that the mechanistic approach in determining the fraction of fuel failures during the reactor coolant pump shaft seizure accident was unacceptable. The ITDP was used for the rods-DNBR calculation. The most limiting case yields no rods in DNB based on the 95/95 DNBR limit. Based on the acceptable fuel failure criterion of 95/95 DNBR limit, we conclude that the reactor coolant pump shaft seizure accident is satisfactorily addressed for the V-5 fuel.

2.8 RTD Bypass Manifold Elimination

The licensee submitted by letter dated February 2, 1987 (Ref. 2) WCAP-12009, Revision 1 RTD Bypass Elimination Licensing Report for Indian Point Unit 3 and clarifying Information relating to RTD bypass manifold elimination.

Currently, the hot and cold leg RTDs are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTDs and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTDs. The RTDs are located in manifolds and are directly inserted into the reactor coolant bypass without thermowells. Each RTD manifold (one hot leg and one cold leg per reactor coolant loop) contains two narrow range RTDs. Flow into each bypass loop is provided by three scoops located at 120° intervals around the hot leg and a tap into the corresponding cold leg.

Each loop's pair of RTDs (one in the hot leg and one in the cold leg) is used to provide inputs for protection system function based on the average loop temperatures (Tavg) and the loop differential temperature (ΔT). Each loop's pair of RTDs is also used to provide inputs for control systems functions based on the average loop temperature and the loop differential temperature.

The hot leg temperature measurement on each loop will be accomplished with three fast response, narrow range, single element RTDs mounted in thermowells located within the existing RTD bypass manifold scoops. A hole will be drilled through the end of each scoop so that water will flow in through the existing holes in the leading edge of the scoop, past the RTD, and out through the new hole. This modification will not affect the single wide range RTD currently installed near the entrance of each steam generator. This RTD will continue to provide the hot leg temperature used to monitor reactor coolant temperature during startup, shutdown, and post accident conditions.

One fast response, narrow range, dual-element RTD will be located in each cold leg at the discharge of the reactor coolant pump (RCP). This will be the replacement for the cold leg RTD located in the bypass manifold. Temperature streaming in the cold leg is not a concern due to the mixing action of the RCP. For this reason, only one RTD is required. This RTD will measure the cold leg temperature which is used to calculate the average loop temperature and the loop differential temperature. One element of the RTD will be considered active and the other element will be held in reserve as a spare. As before, this modification will not affect the single wide range RTD in each cold leg currently installed at the discharge of the reactor coolant pump. This RTD will continue to provide the cold leg temperature used to monitor reactor coolant temperature during startup, shutdown and post accident conditions.

The hot leg RTD measurements will be electronically averaged within the process protection system. The averaged T_{hot} signal will then be used with the T_{cold} signal to calculate reactor coolant ΔT and Tavg which are used in the reactor control and protection systems. The temperature averaging methodology and the treatment of a failed hot leg RTD is the same as has been implemented at Westinghouse modified plants incorporating the RTD Bypass Elimination. This methodology averages the three hot leg RTDs to obtain a single, accurate, hot leg temperature. This average hot leg temperature is used in place of the present manifold-mounted hot leg RTD.

Existing control board ΔT and Tavg indicators and alarms will provide the means of identifying RTD failures. In addition, the hot leg averaging electronics will provide an alarm in the event a hot leg RTD fails high or low. The spare cold leg RTD element provides sufficient spare capacity to accommodate a single cold leg RTD failure per loop. Failure of a hot leg RTD is handled by defeating the failed signal and rescaling the electronics to average the remaining signals. The electronics will be programmed to automatically reject a failed hot leg RTD and compute the average of the remaining two valid hot leg RTDs. The appropriate bias value is then manually

applied to this average. This bias has been previously calculated manually during normal operations. The channel is immediately declared inoperable and placed in trip until the bias value has been applied.

The objectives of this review were to confirm that the reactor trip and engineered safety features actuation systems satisfy the requirements of the acceptance criteria and guidelines applicable to the protection system and will perform their safety function during all plant conditions for which they are required. Since our review indicates that the modified system does not functionally change the reactor trip or the engineered safety features actuation systems, the staff's original evaluation conclusion for these systems remain valid.

The modifications have been reviewed to evaluate conformance to the requirements of IEEE 279-197 criteria. Following is a discussion of design basis requirements in conformance to pertinent I&C criteria:

(1) The single failure criterion continues to be satisfied by this change because the independence of redundant protection sets is maintained.

(2) The changes will continue to maintain the capability of the protection system to initiate a reactor trip during and following natural phenomena credible to the plant site to the same extent as the existing system.

(3) Channel independence and electrical separation is maintained because the Protection Set circuit assignments continue to be Loop 1 Circuits input to Protection Set I; Loop 2 to Protection Set II; Loop 3 to Protection Set III, and Loop 4 to Protection Set IV with appropriate observance of field wiring interface criteria to assure the independence.

On the basis of the above evaluation, it is concluded that the compliance of Indian Point Unit 3 to IEEE 279-1971 is acceptable.

The licensee plans to use the Foxboro Microspec 200 microprocessor in the RDT Bypass System. This will consist of the GATE and CALC modules. We have discussed the licensee's verification and validation (V&V) program for these modules and conclude that it is acceptable subject to the confirmation of the acceptability of the V&V program in accordance with IEEE-7.4.3.2. The licensee has committed to provide a description of its qualification program in a confirmatory letter.

Other areas of concern are flow measurement uncertainty, the response time, and the use of Leading Edge flowmeter (ILEFM).

(1) The new fast response RTD measurement system has been designed to perform within the response time envelope of the original RTD system. The analysis and evaluation of transients and accidents therefore continue to assume a 6.0 second delay in accounting for the overtemperature and overpower Delta-T channel response. We find this acceptable.

(2) The licensee provided flow measurement analysis that accounts for changes due to the RTD bypass removal. The results of the analysis determined the flow measurement uncertainty of 2.6 percent. This calculation was based on the standard Westinghouse methodology previously approved on earlier submittals and the use of ITDP. This flow uncertainty has been factored into the ITDP and is being used with the Cycle 7 reload. The Technical Specification on minimum flow has been changed to reflect this value. Thus we find the flow measurement uncertainty acceptable.

(3) The licensee plans to use a leading edge flowmeter as the primary measurement of feedwater flow for the precision flow calorimetic prior to startup from each refueling. These flow meters have been successfully used on several other plants. Thus we find their use acceptable.

Based on our evaluation and the licensee's statement that the new Foxboro Microspec 200 modules and RTD have been qualified to the criteria discussed in WCAP-12037, "RTD BYPASS ELIMINATION LICENSING REPORT FOR INDIAN POINT UNIT 3", we find the plant modifications to eliminate the RTD bypass manifold and install fast response RTDs directly into the reactor coolant system hot and cold legs to be acceptable.

2.9 Technical Specifications

(1) Section 1.2.5, 3.6.A.2, 3.6 Basis, 3.8.A, 3.8.D and 3.8 Basis. These sections and basis were changed to revise the refueling shutdown margin from requiring 10 percent delta k/k to the more restrictive of 5 percent delta k/k or 1900 ppm. This change will provide additional fuel management flexibility while still providing adequate shutdown margin. It is similar to the requirements in the Standard Technical Specifications and to specifications the staff has been approving for sometime.

(2) 2.1 Basis, Figure 2.1-1, Section 2.3.1.B.4, Section 2.3.1.B.5. These changes are due to the use of the Improved Thermal Design Procedure (ITDP). As a result of using ITDP the reactor core safety limits are revised. These limits are different due to convoluting the plant uncertainties directly into the safety limit DNBR value.

(3) 2.1 Basis, 2.3 Basis, 3.10 Basis. These changes replaced the "design" DNBR with "Safety Limit" DNBR, revised the $F_{\Delta H}$ value, deleted the reference to the Fuel Densification Report and increased the RCS total flow. The change from "design" to "safety" reflects the use of ITDP. The fuel densification report is no longer applicable. The new value of RCS total flow rate reflects thermal design flow plus measurement uncertainties.

(4) Section 2.3.2.B. This change removes reference to three loop operation. Safety analyses have not been performed by Westinghouse for three loop operation and thus three loop operation is no longer an option.

(5) Section 3.1.A.6 and 3.1 Basis. This change increases T_{ave} to 578.3°F, the value used for the analysis.

(6) Section 3.2.B.3 and 3.2 Basis. These changes increase the boric acid storage system volume requirement from 4400 to 5000 gallons. This will provide fuel management flexibility while maintaining shutdown margin.

(7) Section 3.3.A.3.C. This change decreases the accumulator volume limits, which is consistent with the LOCA analysis.

(8) Section 3.10.2.1, 3.10.2.2.2, 3.10.2.4. These changes deal with the new values for F_Q = 2.32 and $F_{\Delta H}$ = 1.62 and deleting the requirement to add measurement error to measured $F_{\Delta H}$. All the analyses were done with the new values which included the measurement uncertainty.

(9) Figure 3.10.2. This changes the K(Z) curve to reflect the new F_Q used in the revised LOCA analysis.

(10) Section 5.3.B. These changes deleted the RCS design temperature and pressure statement, and added that the RCS volume represents 0 percent tube plugging. They were made to reflect the RCS parameters used for VANTAGE 5 fuel.

These changes are acceptable because they reflect the design changes made to the plant and the assumptions used in the safety analyses.

3.0 SUMMARY

We have reviewed the request by the licensee to operate Indian Point Unit 3 with the VANTAGE 5 fuel and the RTD bypass manifold elimination modification. Based on this review, we conclude that appropriate material was submitted and that normal operation and the transients and accidents were reevaluated/reanalyzed. We find the methods used in this reanalysis and the results obtained acceptable. The Technical Specifications submitted reflect the necessary modification for the operation of the future IP-3 reloads and are acceptable. Overall, the licensee's request is therefore acceptable.

The notice for the request for amendment, dated January 20, 1989, was published in the Federal Register on March 8, 1989 (54 FR 9927). The February 15, 1989 letter provided supporting information and did not provide any changes to the Technical Specifications. The May 5, 1989 letter provided a change to the Technical Specifications that revised the minimum reactor coolant system flow rate. This change was necessary because a 2.6% measurement uncertainty was not properly applied. This change is not significant (0.07% change in value) and was not noticed. The May 23, 1989 letter provided new Technical Specification pages but did not change any Technical Specifications but corrected mistakes in the previously proposed Technical Specifications.

4.0 REFERENCES

1. Letter, John C. Brons (New York Power Authority) to NRC, January 20, 1989.

2. Letter, John C. Brons (New York Power Authority) to NRC, February 2, 1989.

3. Letter, John C. Brons (New York Power Authority) to NRC, February 15, 1989.

4. Letter, John C. Brons (New York Power Authority) to NRC, May 5, 1989.

5. Letter, John C. Brons (New York Power Authority) to NRC, May 23, 1989.

6. "Reference Core Report: VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

7. Davidson, S. L. (Ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9171-P-A 1985.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be

endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 6, 1989

7.0 PRINCIPAL CONTRIBUTORS

M. Chatterton
J. Lee
S. Rhow

Table 1

OFFSITE DOSES DUE TO SGTR ACCIDENT

	EAB		LPZ	
	Thyroid	Whole Body	Thyroid	Whole Body
	(rem)		(rem)	
Staff Calculation	38	<1	14	<1
Licensee Calculation	21	<1	7.7	<1
10 CFR 100 Limit	300	25	300	25