July 15, 1997

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M97482)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 175 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated December 23, 1996, as supplemented February 26, 1997, May 12, 1997, June 16, 1997, July 2, 1997, and July 11, 1997. The amendment changes the TSs to allow for the use of VANTAGE+ fuel for Indian Point Unit 3 fuel cycle 10.

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

Sincerely,

ORIGINAL SIGNED BY:

George F. Wunder, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 175 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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ISSUANCE OF AMENDMENT NO. 175 TO FACILITY OPERATING LICENSE NO. DPR-28

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 15, 1997

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cc w/encls: See next page

James Knubel
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 175
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 December 23, 1996, as supplemented February 26, 1997, May 12, 1997, June 16, 1997, July 2, 1997, and July 11, 1997, 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:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 175, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W Dromerick

Alexander W. Dromerick, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 15, 1997

ATTACHMENT TO LICENSE AMENDMENT NO.175

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
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
2.3-5	2.3-5
3.1-36	3.1-36
3.1-37	3.1-37
3.1-38	3.1-38
3.8-2	3.8-2
3.8-6	3.8-6
3.10-9	3.10-9
3.10-10	3.10-10
3.10-16	3.10-16
4.3-4	4.3-4
4.13-1	4.13-1
5.3-1	5.3-1
5.3-2	5.3-2

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. The safety limits represent a design requirement for establishing the trip setpoints identified in Technical Specification 2.3. Technical Specification 3.1.H, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," provide more restrictive limits to ensure that the safety limits are not exceeded.

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 correlations which have 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 Design DNBR limit.

In meeting the DNB criterion, uncertainties in operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analyses limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

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. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at Rated Thermal Power (RTP) specified in the COLR.

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 375,600 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:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} \ (1 + PF_{\Delta H} \ (1-P))$$

Where P is the fraction of Rated Thermal Power.

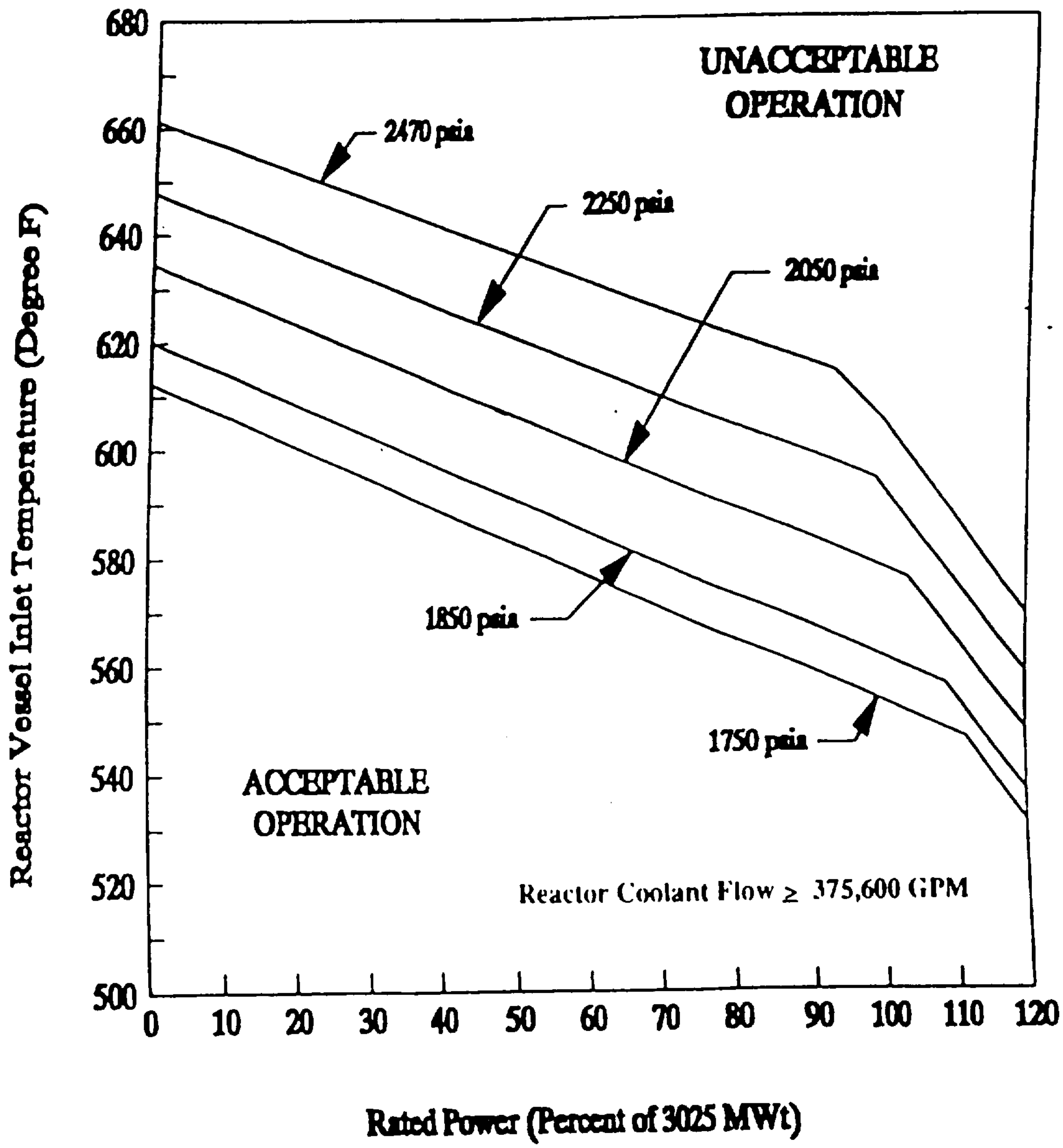
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.9% 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 (specified in the COLR) 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.

REACTOR CORE SAFETY LIMITS

This curve does not provide allowable limits for normal operation.
(See Technical Specification 3.1.H for DNB limits)



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt

Pressures and temperatures do not include allowance for instrument error.

FIGURE 2.1-1

ΔT_o $\leq$ 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 $\leq$ 1.20

K_2 = 0.0273

K_3 = 0.0013

K_1 is a constant which defines the overtemperature ΔT trip margin during steady state operation if the temperature, pressure, and $f(\Delta I)$ 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 setpoint to pressurizer pressure.

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

$f(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are defined above such that:

(a) for $q_t - q_b$ between -6.75 percent and +6.9 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +6.9 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 3.333 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ is more negative than -6.75 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 4.000 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left(K_4 - K_5 \frac{dTavg}{dt} - K_6(T_{avg} - T')\right)$$

where:

ΔT_o $\leq$ measured full power ΔT for the channel being calibrated, °F

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

T' = measured full power T_{avg} for the channel being calibrated, °F (can be set no higher than 570.3 °F)

K_4 $\leq$ 1.073

K_5 = 0 for decreasing average temperature

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

K_6 = 0 for $T \leq T'$

$\geq$ 0.00134 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 - $\geq$ 57.2 cps

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

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 value for K_1 includes an allowance for instrument channel uncertainty, and therefore is a nominal trip setpoint. K_2 and K_3 are analytical limits, and do not require an allowance for instrument channel uncertainty. 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 setpoints meet this requirement. [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 value for K_4 includes an allowance for instrument channel uncertainty, and therefore is a nominal trip setpoint. K_5 and K_6 are analytical limits, and do not require an allowance for instrument channel uncertainty.

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 570.3°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}.

3.1 Reactor Coolant System (RCS)

H. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits*

Specification

1. During the POWER OPERATION CONDITION, RCS DNB parameters for pressurizer pressure and RCS average temperature shall be within the limits specified below:

   a. Pressurizer pressure ≥ 2205 psig;

   b. Maximum indicated T_{avg} ≤ 571.5°F; and

2. At the POWER OPERATION CONDITION with four reactor coolant pumps running, the RCS DNB parameter for RCS total flow rate shall be within the following limit:

   RCS total flow rate ≥ 375,600 gpm.

3. The pressurizer pressure limit of Specification 3.1.H.1 does not apply during:

   a. THERMAL POWER ramp > 5% RTP per minute; or

   b. THERMAL POWER step > 10% RTP.

4. If pressurizer pressure, RCS average temperature, or RCS total flow rate are not in accordance with Specifications 3.1.H.1, 3.1.H.2, or 3.1.H.3, then, immediately verify that the safety limits of Specification 2.1 have not been exceeded and, within 2 hours, restore the RCS DNB parameter(s) to within limits.

5. If pressurizer pressure and/or RCS average temperature are not restored to within limits within 2 hours, be in the HOT SHUTDOWN CONDITION within 6 hours.

6. If RCS total flow rate is not restored to within the limits of Specification 3.1.H.2 within 2 hours, bring THERMAL POWER to ≤ 10% RTP within 6 hours and ensure operation is in accordance with Specification 3.1.A.1.e.

Surveillance Requirements

Reference Technical Specification Table 4.1-1, Items 4, 5, and 7, and Section 4.3.B.

Bases

Background
These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS

* Current DNB analysis contains adequate margin for Cycle 10. Prior to achieving criticality in Cycle 11, the DNB analysis must be reviewed and approved by NRC staff.

pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure and temperature limits are consistent with operation within the nominal operational envelope. A lower pressure will cause the reactor core to approach DNB limits. A higher RCS average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit bounds that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

Applicable Safety Analyses
The requirements of this Specification represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this Specification will result in meeting the applicable DNBR criteria. Changes to the unit that could affect these parameters must be assessed for their effect on the DNBR criteria.

Specification
Specifications 3.1.H.1 and 3.1.H.2 specify limits on the monitored process variables (pressurizer pressure, RCS average temperature, and RCS total flow rate) to ensure that the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The RCS total flow rate limit of 375,600 gpm allows for a measurement uncertainty of 2.9% associated with the performance of Reactor Coolant System Flow Calculation required by Technical Specification 4.3.B. Because the flow instrumentation provides flow indication based on a percentage of full flow, the 375,600 gpm is converted into a percentage of full flow to accomodate the verification that RCS total flow is within limits during channel checks.

The pressurizer pressure limit of 2205 psig allows for measurement uncertainty and instrument error. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.

The limit on maximum indicated RCS average temperature 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 + fuel reloads with asymmetric tube

3.1.H (continued)

plugging among steam generators. A maximum full power T_{cold} of 547.7°F (including control deadband and measurement uncertainties) was assumed in these safety analyses. A T_{avg} of 578.3°F assures that a T_{cold} of 547.7°F is not exceeded at a measured flow of $\geq$ 375,600 gpm when considering asymmetric tube plugging among steam generators for DNB considerations. However, T_{avg} will be controlled to a maximum indicated T_{avg} of 571.5°F which assures consistency with analyses for post-LOCA containment integrity.

Applicability
During the POWER OPERATION CONDITION, the limits on pressurizer pressure and RCS coolant average temperature must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. For the same reason, during the POWER OPERATION CONDITION with four reactor coolant pumps running, the limit on RCS flow rate must be maintained. In all other operating conditions, the power level is low enough that DNB is not a concern.

Specification 3.1.H.3 indicates that the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counter productive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit 2.1, "Safety Limits, Reactor Core." Those limits are less restrictive than the limits of this specification but violation of a Safety Limit merits stricter, more severe required action. Should a violation of Specification 3.1.H.1 occur, the operator must check whether or not a Safety Limit has been exceeded.

Actions
RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within specification limits, action must be taken to restore the parameter(s).

The 2 hour completion time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience for Westinghouse plants.

If the required action of Specification 3.1.H.4 is not met within the associated completion time, the plant must be brought to a mode in which Specification 3.1.H.1 does not apply. To achieve this status, the plant must be brought to at least the HOT SHUTDOWN CONDITION within 6 hours. The reduced power condition eliminates the potential for violation of the accident analysis bounds. The completion time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

9. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 145 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 421* hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal absorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than 76 assemblies are to be discharged from the reactor, those assemblies in excess of 76 shall not be discharged earlier than 267 hours after shutdown.

10. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

11. Hoists or cranes utilized in handling irradiated fuel shall be dead- load tested before 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 refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test and prior to fuel handling. A test of interlocks and overload cutoff devices on the manipulator shall also be performed.

12. 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 period of inoperability.

13. To ensure redundant decay heat removal capability, at least two of the following requirements shall be met:

* Movement of irradiated VANTAGE + fuel assemblies before the reactor has been subcritical for ≥550 hours requires operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers.

The waiting time of 267 hours required following plant shutdown before unloading more than 76 assemblies 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 temperature of 95°F, consistent with the FSAR assumptions[2].

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 activities 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 421 hours (See footnote on page 3.8-2), 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.

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

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

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

An upper bound envelope of F_Q^{RTP} specified in the COLR times the normalized peaking factor axial dependence of K(Z) specified in the COLR 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, K(Z), specified in the COLR demonstrates that the peak clad temperature is below the 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 $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq F_{\Delta H}^{RTP}/1.04$, where $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^N$ limit at Rated Thermal Power specified in the COLR. 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 $F_{\Delta H}^{N}$ 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 anomalies 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.

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

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The assumed control rod drop time in the safety analysis is 2.7 seconds, consisting of 1.80 seconds for normal rod drop time plus additional margin which includes a seismic allowance. The required control rod drop time in Section 3.10.8 is therefore consistent with that assumed in the safety analysis.

<u>REFERENCE</u>

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program," August 1975
2. FSAR Appendix 14C
3. Letter from J.P. Bayne to S.A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."

4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

B. Reactor Coolant System Flow Calculation

Specification

Once every 24 months, prior to exceeding 24 hours of continuous operation with THERMAL POWER ≥ 90% RTP, verify by flow calculation that RCS total flow rate is ≥ 375,600 gpm.

Basis

Measurement of RCS total flow rate by performance of a flow calculation once every 24 months verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered or steam generator tubes have been plugged, which may have caused an alteration of flow resistance.

This specification allows for placement of the unit in the best condition for performing the Surveillance Requirement. The specification allows the Surveillance Requirement to be performed within 24 hours after THERMAL POWER ≥ 90% RTP. This is appropriate because a flow calculation performed with the plant ≥ 90% RTP will ensure that instrument inaccuracies are consistent with those assumed in the accident analyses. The Surveillance shall be performed within 24 hours of continuous operation at or above 90% RTP.

4.13 Containment Vent and Purge System

Applicability

This specification applies to the surveillance requirements of the containment vent and purge system during normal operations and when reactor fuel is anticipated to be moved before the reactor has been subcritical for at least 421* hours.

Objective

To verify the operability of the containment vent and purge system.

Specification

The following surveillance shall be performed as stated.

A. Isolation Valves

1. Each month verify that the containment purge supply and exhaust isolation valves are closed during operation above cold shutdown.

2. At least once per 24 months verify that the mechanical stops on the containment vent isolation valve (PCV-1190, -1191, -1192) actuator is limited to the valve opening angle to 60° (90° = full open).

B. HEPA Filters and Charcoal Absorbers

If fuel movement is to take place before the reactor has been subcritical for at least 421* hours, the containment vent and purge system shall be demonstrated operable as follows:

1. Within 18 months prior to fuel movement and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank within 18 months prior to fuel movement, or (2) after structural maintenance on the HEPA filter or charcoal adsorber housing within 18 months prior to fuel movement, which could effect system operation:

   a. Verify that the charcoal adsorbers remove $\geq$ 99% of halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at the operating flow $\pm$ 10%.

   b. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place while operating the ventilation system at the operating flow rate $\pm$ 10%.

2. Within 18 months prior to fuel movement and after every 720 hours of system operation, subject a representative sample of carbon from the charcoal adsorbers to a laboratory analysis and verify within 31 days a removal efficiency of $\geq$ 90% for radioactive methyl iodine at an operating air flow velocity $\pm$ 20% per test 5.b in Table 2 of Regulatory Guide 1.52, March 1978.

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* Movement of irradiated VANTAGE + fuel assemblies before the reactor has been subcritical for ≥550 hours requires operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers.

5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 89 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.[1]

2. The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.[2]

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 5.0 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borosilicate glass clad with stainless steel.[4] Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

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 cubic feet.

Basis

Deleted

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. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 175 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 December 23, 1996, as supplemented February 26, 1997, May 12, 1997, June 16, 1997, July 2, 1997, and July 11, 1997, the Power Authority of the State of New York (PASNY) requested an amendment to change the Technical Specifications (TSs) for Indian Point Nuclear Generating Unit No. 3 (IP3). PASNY requested a revision to several sections of Appendix A of IP3 TSs to accommodate the transition from VANTAGE 5 fuel (without intermediate flow mixing grids) to VANTAGE+ with PERFORMANCE+ Westinghouse fuel features for the upcoming cycle 10. The February 26, 1997, May 12, 1997, June 16, 1997, July 2, 1997, and July 11, 1997, letters contained information necessary to support the transition to VANTAGE+ fuel and did not change the initial proposed no significant hazards determination.

The VANTAGE+ fuel design is a modification of the NRC-approved Westinghouse VANTAGE-5 fuel assembly design, for which IP3 is currently licensed. Some new product features (such as intermediate flow mixing (IFM) grids, low pressure drop mid-grids, ZIRLO clad and tubing materials, annular axial blankets, and variable pitch plenum springs) are being implemented at IP3 starting with cycle 10. Changes are being made to pertinent TS parameters as required by the transition to VANTAGE+ fuel.

2.0 EVALUATION

2.1 Mechanical Design

The IP3 cycle 10 will be a mixed transition core. The 193 assembly core will be loaded with 83 15x15 VANTAGE+ with PERFORMANCE + features fuel assemblies, while the remainder will be 15X15 VANTAGE-5 fuel assemblies. The VANTAGE+ fuel, designed to be mechanically compatible with the VANTAGE-5 fuel, is based upon a modification of the NRC-approved Westinghouse VANTAGE-5 fuel assembly design. The VANTAGE+ features are the ZIRLO IFM grids, low pressure drop mid-grids, ZIRLO clad and tubing materials, and variable pitch plenum springs. The VANTAGE+ fuel mechanical design was approved for licensing applications in Topical Report WCAP-12610 up to a rod average burnup of 60 GWD/MTU. The staff has concluded that the VANTAGE+ fuel is acceptable for the IP3 cycle 10 reload, because from a fuel performance standpoint, the VANTAGE+ fuel is essentially no different from the VANTAGE-5 fuel.

2.2 Nuclear Design

Westinghouse analyzed key safety parameters for IP3 and found few differences between the two fuel types. The changes are typical of the normal cycle-to-cycle variations as loading patterns change. Westinghouse's analyses showed that the VANTAGE+ fuel design bases were satisfied and that all safety limits characteristics of the VANTAGE-5 (without IFMs) fuel design, also apply to the VANTAGE+ fuel design. The analysis also showed that the margins to key safety parameters limits were not reduced by using the VANTAGE+ fuel design rather than the VANTAGE-5 (without the IFMs) fuel design. The nuclear design methodology was not affected by the transition to the VANTAGE+ fuel design, nor was the IP3 final safety analysis report (FSAR) nuclear design basis. We conclude that the VANTAGE+ fuel nuclear design is acceptable for use at IP3 in the cycle 10 reload, because safety analyses conducted using VANTAGE+ fuel design has been shown to be acceptable.

2.3 Thermal and Hydraulic Design

The currently approved thermal and hydraulic analyses to support the 15X15 VANTAGE-5 Optimized Fuel Assembly (OFA) also supports the 15X15 VANTAGE+ fuel. The Westinghouse rod bundle critical heat flux (CHF) correlation, WRB-1, predicts critical heat flux in rod bundles based on subchannel local fluid conditions. This correlation was initially approved for the standard 14X14, 15X15 and the 17X17 standard Westinghouse fuel. Evolution of the standard 17X17 and 15X15 fuel have been developed by Westinghouse and their behavior simulated by using an NRC approved scaling technique. This scaling technique was validated for all four of the different 17X17 fuel types, but not for the 15X15 OFA and the VANTAGE+ (w/IFMs) fuel. No testing was conducted to verify that the scaling technique applied to the 15X15 standard fuel; however, cycle 10 analyses has shown that there is substantial departure from nucleate boiling ratio (DNBR) margin. Consequently, until such time as fuel tests are conducted on the 15X15 VANTAGE+ (w/IFMs) to validated the scaling technique and the applicability of the WRB-1 correlation, is acceptable for the upcoming cycle 10 only. Also, DNB analyses must be submitted to the staff for review and approval prior to cycle 11.

2.3.1 Non-Loss-of-Coolant Accident (LOCA) Safety Analysis

Using the same methods that were used in the previous IP3 fuel reloads, the licensee reanalyzed all the FSAR Chapter 14 non-LOCA accidents which are pertinent to the transition from the Westinghouse 15x15 VANTAGE-5 fuel (w/o IFMs) to Westinghouse 15x15 VANTAGE+ fuel for the IP3. The licensee reanalyzed all those accidents which are affected by one or more of the VANTAGE+ design features. Some of the assumptions used in the reanalysis differ from those currently used: the revised thermal design procedure (RTDP), revised overtemperature delta T (OT delta T) and overpower delta temperature reactor trip (OP delta T) setpoints, increased $F_{\Delta H}$ and F_Q, and increased uncertainties for the 24 month reload cycles. The TS rod drop time limit of 1.8 will remain unchanged, but the safety analysis value for the rod drop time will be increased from 2.4 to 2.7 to obtain a more conservative analysis. These assumptions along with the DNB correlation, are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR limit will be acceptable. The reanalysis showed that the transition from the 15x15 VANTAGE-5 (w/o IFMs) to 15x15 VANTAGE+ fuel can be accommodated without exceeding the margin to the applicable FSAR safety analysis limits. The NRC staff reviewed the licensee's results and finds this analysis acceptable.

2.3.2 LOCA Safety Analysis

Large-break LOCA using approved codes, methodologies, and parameters, were performed to demonstrate the continued conformance of IP3 to the acceptance criteria of 10 CFR 50.46. All analyses considered the VANTAGE+ fuel, including the following features: extended burnup, the optimized fuel rod plenum spring, low cobalt top and bottom nozzles, debris resistance oxide coating, ZIRLO guide thimble and instrumentation tubes, enriched annular pellets in axial blankets, and the protective bottom grid with elongated bottom end plug. An additional feature of the 15x15 VANTAGE+ fuel, the integral fuel burnable absorber (IFBA), was specifically analyzed for IP3. The analyses show that the use of 15x15 VANTAGE+ fuel with IFBA, will decrease large-break LOCA peak clad temperatures (PCTs).

The licensee performed the small-break LOCA analysis in conjunction with the use of the VANTAGE+ fuel. Hydraulic differences between the VANTAGE-5 (w/o IFMs) and VANTAGE+ fuel assemblies were determined not to be a factor. Thus, the analyses showed that both LOCA analyses for transition to the VANTAGE+ fuel for IP3 remain in compliance with the requirements of 10 CFR 50.46 and are therefore acceptable.

2.3.3 Fuel Handling Accident

The staff performed an independent dose analysis to determine if the use of VANTAGE+ fuel in cycle 10 at IP3 would result in any change to the offsite doses at the low population zone (LPZ) or exclusion area boundary (EAB). VANTAGE+ fuel has a larger design radial peaking factor (1.7) than the VANTAGE-5 fuel currently used in the reactor core. The staff's analysis utilized licensee provided data, except for the atmospheric dispersion factors for the LPZ and EAB, which were obtained from the staff's original safety evaluation report for IP3, dated September 21, 1973. On the basis of this analysis, the staff determined that HEPA and charcoal filtration must be used during movement of spent VANTAGE+ fuel for the first 550 hours following reactor shutdown in order not to exceed the staff's dose limit for a fuel handling accident of 75 rem (25 percent of the 10 CFR Part 100 limit of 300 rem) to the thyroid at the EAB. After 550 hours, spent VANTAGE+ fuel can be moved without HEPA and charcoal filtration.

Table 1 presents the results of the staff's confirmatory dose calculation. Table 2 lists the assumptions used in the staff's confirmatory calculation.

Table 1

CALCULATED RADIOLOGICAL CONSEQUENCES

Exclusion Area Boundary	Dose	SRP Acceptance Criteria
Whole Body	0.28 rem	6 rem
Thyroid	74.9 rem	75 rem

Table 2

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES

Parameters		
Power Level, Mwt		3025
Number of Fuel Rods Damaged		225
Total Number of Rods		43,425
Shutdown Time, hours		550
Power Peaking Factor		1.70
Fission Product Release Duration*		2 hrs
Fission-Product Release Fractions (%)*		
Iodine	10	
Noble Gases		30
Pool Decontamination Factors*		
Iodine	100	
Noble Gases		1
Iodine Forms(%)		
Elemental		75
Organic	25	
Core Fission Product Inventories per TID-14844		
Receptor Point Variables		
Exclusion Area Boundary**		
Atmospheric Dispersion Factor, X/Q (sec/m^3)		
0-2 hours	1.8×10^{-3}	

* Regulatory Guide 1.25
** Staff Safety Evaluation for Indian Pt. 3 (September 21, 1973)

2.4 Technical Specification Changes

In order to support the use of VANTAGE+ fuel, it is necessary to change the following sections of the TS:

Section 2.1 - Safety Limits - Reactor Core

The curves that determine acceptable reactor vessel inlet temperature as a function of reactor power have been changed to accommodate the transition to VANTAGE+ fuel. The curves are consistent with the thermal and hydraulic analyses discussed in Section 2.3 above and are acceptable.

The basis of this section was changed to reflect these changes.

Section 2.3 - Limiting Safety System Settings, Protective Instrumentation

Changes to parameters in the Overtemperature delta T and Overpower delta T protective circuitry were changed to accommodate the transition to VANTAGE+ fuel. These changes are consistent with the nuclear, thermal-hydraulic, and safety analyses discussed in Section 2.3 above and are acceptable.

The basis of this section was changed to reflect these changes.

Section 3.1 - Reactor Coolant System

The reactor coolant flow was changed from 385,400 gpm to 375,600 gpm.

A footnote requiring the licensee to receive staff approval of DNB analyses before startup for Cycle 11 was added.

The maximum allowable cold leg temperature was changed from 547.9 degrees Fahrenheit to 547.7 degrees Fahrenheit.

These changes are consistent with the thermal and hydraulic design analyses discussed in Section 2.3 above. The staff has found that these TS changes support the transition to VANTAGE+ and are acceptable.

Section 3.8 - Refueling, Fuel Handling and Storage

Spent fuel cannot be moved until the reactor has been subcritical for 421 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. In addition, irradiated VANTAGE+ fuel cannot be moved until the reactor has been subcritical for ≥ 550 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. These changes will allow a greater decay time before fuel is handled without the containment purge and vent system lined up to the HEPA and charcoal filters. These changes are consistent with the analyses discussed in Section 2.3.3 above and are acceptable.

Section 3.10 - Control Rod and Power Distribution Limits

A reference to a maximum peak clad temperature of 2049 degrees Fahrenheit was removed. This number was the result of earlier analyses. The current analyses show that the peak clad temperature will not exceed 2200 degrees Fahrenheit and this is consistent with the requirements of 10 CFR 50.46. The staff finds this acceptable.

A typographical error on page 3.10-10 was corrected.

The Bases were changed to state that assumed rod drop time was increased from 2.40 seconds to 2.7 seconds. This change is consistent with Section 2.3.1 of the safety evaluation.

Section 4.3-4 - Reactor Coolant System Testing

The reactor coolant system flow was changed from 385,400 gpm to 375,600 gpm. These changes are consistent with the thermal and hydraulic design analyses discussed in Section 2.3 above.

Section 4.19 - Containment Vent and Purge System

Spent fuel cannot be moved until the reactor has been subcritical for 421 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. In addition, irradiated VANTAGE + fuel cannot be moved until the reactor has been subcritical for ≥ 550 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. These changes will allow a greater decay time before fuel is handled without the containment purge and vent system lined up to the HEPA and charcoal filters. This change is consistent with the analyses discussed in section 2.3.3 above and is acceptable.

Section 5.3 - Reactor

A reference to fuel assemblies W51 and W06 each containing one stainless steel filler rod during Cycle 9 and Cycle 10 has been removed. These fuel assemblies have been replaced with new VANTAGE+ assemblies. This change is consistent with the transition to VANTAGE+ fuel. The change is acceptable.

The basis of this section was changed to reflect the current configuration.

2.5 Summary

The staff has reviewed the information submitted for the cycle 10 operation of IP3. The mechanical design, the nuclear design, the thermal-hydraulic design, and the transient and accident analyses are acceptable. Since the scaling technique for the Critical Heat Flux (CHF) correlation, WRB-1, has not been tested for the 15X15 fuel, the staff concludes that the transition from the VANTAGE-5 (w/o IFMs) to the VANTAGE+ (w/IFMs) fuel design is acceptable for the upcoming cycle 10 only. The proposed TS changes for the cycle 10 reload represent the necessary modifications for this cycle.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 6578). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Attard

Date: July 15, 1997

5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 89 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.[1]

2. The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.[2]

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 5.0 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borosilicate glass clad with stainless steel.[4] Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

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 cubic feet.

Basis

Deleted

References

(1) FSAR Section 3.2.2

(2) FSAR Section 3.2.1

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(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-9

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 175 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

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2.0 EVALUATION

2.1 Mechanical Design

The IP3 cycle 10 will be a mixed transition core. The 193 assembly core will be loaded with 83 15x15 VANTAGE+ with PERFORMANCE + features fuel assemblies, while the remainder will be 15X15 VANTAGE-5 fuel assemblies. The VANTAGE+ fuel, designed to be mechanically compatible with the VANTAGE-5 fuel, is based upon a modification of the NRC-approved Westinghouse VANTAGE-5 fuel assembly design. The VANTAGE+ features are the ZIRLO IFM grids, low pressure drop mid-grids, ZIRLO clad and tubing materials, and variable pitch plenum springs. The VANTAGE+ fuel mechanical design was approved for licensing applications in Topical Report WCAP-12610 up to a rod average burnup of 60 GWD/MTU. The staff has concluded that the VANTAGE+ fuel is acceptable for the IP3 cycle 10 reload, because from a fuel performance standpoint, the VANTAGE+ fuel is essentially no different from the VANTAGE-5 fuel.

2.2 Nuclear Design

Westinghouse analyzed key safety parameters for IP3 and found few differences between the two fuel types. The changes are typical of the normal cycle-to-cycle variations as loading patterns change. Westinghouse's analyses showed that the VANTAGE+ fuel design bases were satisfied and that all safety limits characteristics of the VANTAGE-5 (without IFMs) fuel design, also apply to the VANTAGE+ fuel design. The analysis also showed that the margins to key safety parameters limits were not reduced by using the VANTAGE+ fuel design rather than the VANTAGE-5 (without the IFMs) fuel design. The nuclear design methodology was not affected by the transition to the VANTAGE+ fuel design, nor was the IP3 final safety analysis report (FSAR) nuclear design basis. We conclude that the VANTAGE+ fuel nuclear design is acceptable for use at IP3 in the cycle 10 reload, because safety analyses conducted using VANTAGE+ fuel design has been shown to be acceptable.

2.3 Thermal and Hydraulic Design

The currently approved thermal and hydraulic analyses to support the 15X15 VANTAGE-5 Optimized Fuel Assembly (OFA) also supports the 15X15 VANTAGE+ fuel. The Westinghouse rod bundle critical heat flux (CHF) correlation, WRB-1, predicts critical heat flux in rod bundles based on subchannel local fluid conditions. This correlation was initially approved for the standard 14X14, 15X15 and the 17X17 standard Westinghouse fuel. Evolution of the standard 17X17 and 15X15 fuel have been developed by Westinghouse and their behavior simulated by using an NRC approved scaling technique. This scaling technique was validated for all four of the different 17X17 fuel types, but not for the 15X15 OFA and the VANTAGE+ (w/IFMs) fuel. No testing was conducted to verify that the scaling technique applied to the 15X15 standard fuel; however, cycle 10 analyses has shown that there is substantial departure from nucleate boiling ratio (DNBR) margin. Consequently, until such time as fuel tests are conducted on the 15X15 VANTAGE+ (w/IFMs) to validated the scaling technique and the applicability of the WRB-1 correlation, is acceptable for the upcoming cycle 10 only. Also, DNB analyses must be submitted to the staff for review and approval prior to cycle 11.

2.3.1 Non-Loss-of-Coolant Accident (LOCA) Safety Analysis

Using the same methods that were used in the previous IP3 fuel reloads, the licensee reanalyzed all the FSAR Chapter 14 non-LOCA accidents which are pertinent to the transition from the Westinghouse 15x15 VANTAGE-5 fuel (w/o IFMs) to Westinghouse 15x15 VANTAGE+ fuel for the IP3. The licensee reanalyzed all those accidents which are affected by one or more of the VANTAGE+ design features. Some of the assumptions used in the reanalysis differ from those currently used: the revised thermal design procedure (RTDP), revised overtemperature delta T (OT delta T) and overpower delta temperature reactor trip (OP delta T) setpoints, increased $F_{\Delta H}$ and F_Q, and increased uncertainties for the 24 month reload cycles. The TS rod drop time limit of 1.8 will remain unchanged, but the safety analysis value for the rod drop time will be increased from 2.4 to 2.7 to obtain a more conservative analysis. These assumptions along with the DNB correlation, are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR limit will be acceptable. The reanalysis showed that the transition from the 15x15 VANTAGE-5 (w/o IFMs) to 15x15 VANTAGE+ fuel can be accommodated without exceeding the margin to the applicable FSAR safety analysis limits. The NRC staff reviewed the licensee's results and finds this analysis acceptable.

2.3.2 LOCA Safety Analysis

Large-break LOCA using approved codes, methodologies, and parameters, were performed to demonstrate the continued conformance of IP3 to the acceptance criteria of 10 CFR 50.46. All analyses considered the VANTAGE+ fuel, including the following features: extended burnup, the optimized fuel rod plenum spring, low cobalt top and bottom nozzles, debris resistance oxide coating, ZIRLO guide thimble and instrumentation tubes, enriched annular pellets in axial blankets, and the protective bottom grid with elongated bottom end plug. An additional feature of the 15x15 VANTAGE+ fuel, the integral fuel burnable absorber (IFBA), was specifically analyzed for IP3. The analyses show that the use of 15x15 VANTAGE+ fuel with IFBA, will decrease large-break LOCA peak clad temperatures (PCTs).

The licensee performed the small-break LOCA analysis in conjunction with the use of the VANTAGE+ fuel. Hydraulic differences between the VANTAGE-5 (w/o IFMs) and VANTAGE+ fuel assemblies were determined not to be a factor. Thus, the analyses showed that both LOCA analyses for transition to the VANTAGE+ fuel for IP3 remain in compliance with the requirements of 10 CFR 50.46 and are therefore acceptable.

2.3.3 Fuel Handling Accident

The staff performed an independent dose analysis to determine if the use of VANTAGE+ fuel in cycle 10 at IP3 would result in any change to the offsite doses at the low population zone (LPZ) or exclusion area boundary (EAB). VANTAGE+ fuel has a larger design radial peaking factor (1.7) than the VANTAGE-5 fuel currently used in the reactor core. The staff's analysis utilized licensee provided data, except for the atmospheric dispersion factors for the LPZ and EAB, which were obtained from the staff's original safety evaluation report for IP3, dated September 21, 1973. On the basis of this analysis, the staff determined that HEPA and charcoal filtration must be used during movement of spent VANTAGE+ fuel for the first 550 hours following reactor shutdown in order not to exceed the staff's dose limit for a fuel handling accident of 75 rem (25 percent of the 10 CFR Part 100 limit of 300 rem) to the thyroid at the EAB. After 550 hours, spent VANTAGE+ fuel can be moved without HEPA and charcoal filtration.

Table 1 presents the results of the staff's confirmatory dose calculation. Table 2 lists the assumptions used in the staff's confirmatory calculation.

Table 1

CALCULATED RADIOLOGICAL CONSEQUENCES

Exclusion Area Boundary	Dose	SRP Acceptance Criteria
Whole Body	0.28 rem	6 rem
Thyroid	74.9 rem	75 rem

Table 2

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES

Parameters		
Power Level, Mwt		3025
Number of Fuel Rods Damaged		225
Total Number of Rods		43,425
Shutdown Time, hours		550
Power Peaking Factor		1.70
Fission Product Release Duration*		2 hrs
Fission-Product Release Fractions (%)*		
Iodine	10	
Noble Gases		30
Pool Decontamination Factors*		
Iodine	100	
Noble Gases		1
Iodine Forms(%)		
Elemental		75
Organic	25	
Core Fission Product Inventories per TID-14844		

Receptor Point Variables

Exclusion Area Boundary**

Atmospheric Dispersion Factor, X/Q (sec/m^3)	
0-2 hours	1.8×10^{-3}

* Regulatory Guide 1.25
** Staff Safety Evaluation for Indian Pt. 3 (September 21, 1973)

2.4 Technical Specification Changes

In order to support the use of VANTAGE+ fuel, it is necessary to change the following sections of the TS:

Section 2.1 - Safety Limits - Reactor Core

The curves that determine acceptable reactor vessel inlet temperature as a function of reactor power have been changed to accommodate the transition to VANTAGE+ fuel. The curves are consistent with the thermal and hydraulic analyses discussed in Section 2.3 above and are acceptable.

The basis of this section was changed to reflect these changes.

Section 2.3 - Limiting Safety System Settings, Protective Instrumentation

Changes to parameters in the Overtemperature delta T and Overpower delta T protective circuitry were changed to accommodate the transition to VANTAGE+ fuel. These changes are consistent with the nuclear, thermal-hydraulic, and safety analyses discussed in Section 2.3 above and are acceptable.

The basis of this section was changed to reflect these changes.

Section 3.1 - Reactor Coolant System

The reactor coolant flow was changed from 385,400 gpm to 375,600 gpm.

A footnote requiring the licensee to receive staff approval of DNB analyses before startup for Cycle 11 was added.

The maximum allowable cold leg temperature was changed from 547.9 degrees Fahrenheit to 547.7 degrees Fahrenheit.

These changes are consistent with the thermal and hydraulic design analyses discussed in Section 2.3 above. The staff has found that these TS changes support the transition to VANTAGE+ and are acceptable.

Section 3.8 - Refueling, Fuel Handling and Storage

Spent fuel cannot be moved until the reactor has been subcritical for 421 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. In addition, irradiated VANTAGE+ fuel cannot be moved until the reactor has been subcritical for ≥ 550 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. These changes will allow a greater decay time before fuel is handled without the containment purge and vent system lined up to the HEPA and charcoal filters. These changes are consistent with the analyses discussed in Section 2.3.3 above and are acceptable.

Section 3.10 - Control Rod and Power Distribution Limits

A reference to a maximum peak clad temperature of 2049 degrees Fahrenheit was removed. This number was the result of earlier analyses. The current analyses show that the peak clad temperature will not exceed 2200 degrees Fahrenheit and this is consistent with the requirements of 10 CFR 50.46. The staff finds this acceptable.

A typographical error on page 3.10-10 was corrected.

The Bases were changed to state that assumed rod drop time was increased from 2.40 seconds to 2.7 seconds. This change is consistent with Section 2.3.1 of the safety evaluation.

Section 4.3-4 - Reactor Coolant System Testing

The reactor coolant system flow was changed from 385,400 gpm to 375,600 gpm. These changes are consistent with the thermal and hydraulic design analyses discussed in Section 2.3 above.

Section 4.19 - Containment Vent and Purge System

Spent fuel cannot be moved until the reactor has been subcritical for 421 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. In addition, irradiated VANTAGE + fuel cannot be moved until the reactor has been subcritical for $\geq$ 550 hours unless the containment building purge and vent system is lined up to discharge through both HEPA and charcoal filters. These changes will allow a greater decay time before fuel is handled without the containment purge and vent system lined up to the HEPA and charcoal filters. This change is consistent with the analyses discussed in section 2.3.3 above and is acceptable.

Section 5.3 - Reactor

A reference to fuel assemblies W51 and W06 each containing one stainless steel filler rod during Cycle 9 and Cycle 10 has been removed. These fuel assemblies have been replaced with new VANTAGE+ assemblies. This change is consistent with the transition to VANTAGE+ fuel. The change is acceptable.

The basis of this section was changed to reflect the current configuration.

2.5 Summary

The staff has reviewed the information submitted for the cycle 10 operation of IP3. The mechanical design, the nuclear design, the thermal-hydraulic design, and the transient and accident analyses are acceptable. Since the scaling technique for the Critical Heat Flux (CHF) correlation, WRB-1, has not been tested for the 15X15 fuel, the staff concludes that the transition from the VANTAGE-5 (w/o IFMs) to the VANTAGE+ (w/IFMs) fuel design is acceptable for the upcoming cycle 10 only. The proposed TS changes for the cycle 10 reload represent the necessary modifications for this cycle.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 6578). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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