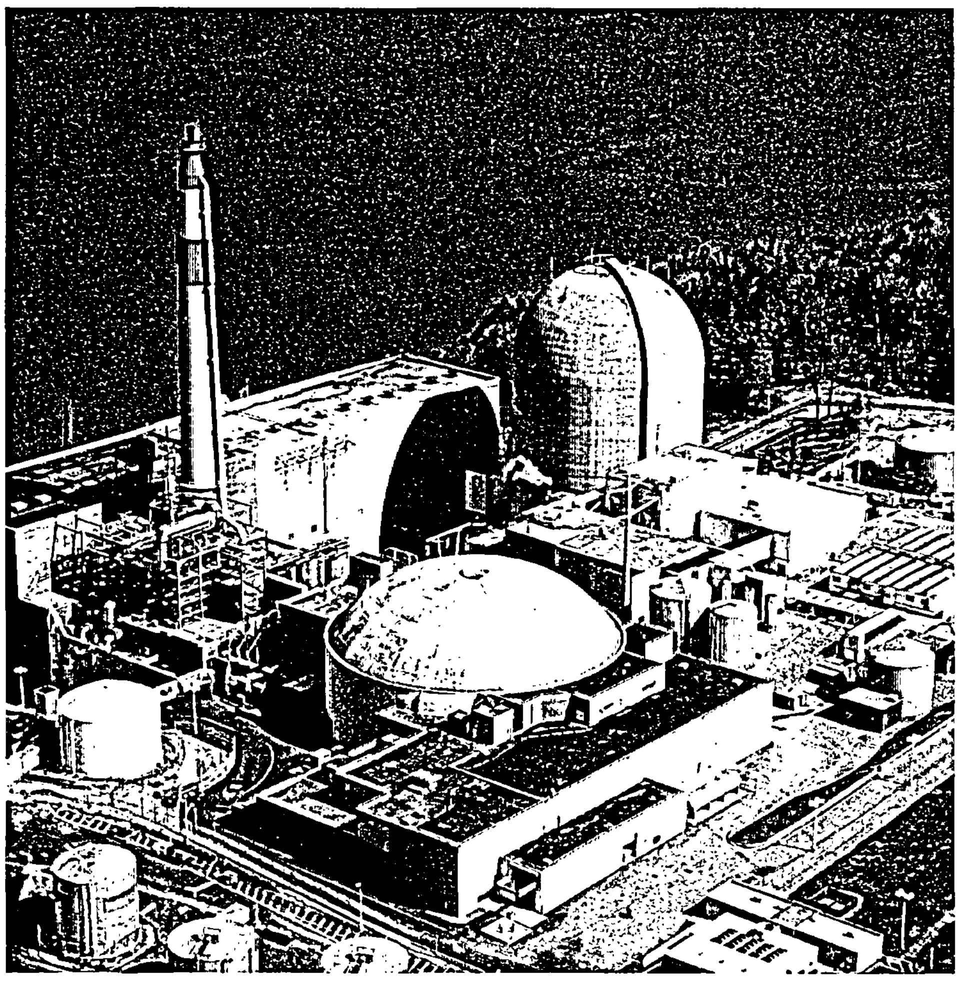
Entergy Nuclear Operations, Incorporated

Indian Point Nuclear Generating Unit No. 2



Stretch Power Uprate
License Amendment Request Package

ATTACHMENT I TO NL-04-005

ANALYSIS OF PROPOSED
TECHNICAL SPECIFICATION CHANGES REGARDING
INCREASE OF LICENSED THERMAL POWER, 3.26%

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-26, Docket No. 50-247 for Indian Point Nuclear Generating Unit No. 2 (IP2) to increase reactor thermal power to 3216 MWt. Although originally licensed to operate at 2758 MWt, IP2 was designed to be capable of operation at 3216.5 MWt. The engineered safety features were designed for a power level of 3216.5 MWt. The analyses presented in Attachment III have been performed in accordance with current regulatory standards and continue to demonstrate the plant capability at 3216 MWt.

The proposed changes to the IP2 Technical Specifications (TS) are based upon the application of a 3.26% Stretch Power Uprate (SPU) analysis. Entergy previously received authorization to increase Reactor Power at IP2 to 3114.4 MWt (May 2003) based upon analysis work that demonstrated a reactor calorimetric uncertainty of 0.6%. The analysis work contained in Attachment III is based upon the use of a traditional 2% calorimetric uncertainty. Using the 2% calorimetric uncertainty in place of the current 0.6% provides additional margin in safety analyses that are being revised for the SPU. This will also eliminate the existing requirement to reduce power when the Leading Edge Flow Meter is not available for performing the daily calorimetric determination of reactor thermal power.

Modifications are planned for the next refueling outage to accommodate the increase in power.

2.0 PROPOSED CHANGES

Facility Operating License:

Page 3; change Rated Thermal Power from 3114.4 MWt to 3216 MWt.

Technical Specifications:

1. Dose Equivalent I-131, Tech Spec Section 1.1

   This proposed change is not a result of the stretch uprate program. The updated definition is being proposed to be more consistent with the dose analysis methodology previously adopted by the Alternate Source Term license amendment (Amendment 211 issued July 27, 2000). There are no Bases for this Tech Spec section.

2. Rated Thermal Power (RTP), Tech Spec Section 1.1

   Previous value of 3114.4 MWt has been revised to 3216 MWt consistent with the analysis and evaluation in Attachment III. There are no Bases for this Tech Spec section.

3. Changes in Allowable Values in Table 3.3.1-1 (RPS Instrumentation)

- **Function 2.a** Power range neutron flux (high): Change allowable value from $\leq$ 112.6% RTP to $\leq$110.6%

- **Function 9** Reactor Coolant Flow – low: Change allowable value from $\geq$ 88.8% to $\geq$88.7%

- **Function 13** Steam Generator water level – low-low: Change allowable value from $\geq$ 3.7% to $\geq$3.4%

- **Function 14** Steam Generator water level – low: Change allowable value from $\geq$ 3.7% to $\geq$3.4%

- **Function 5, Note 1** Overtemperature ΔT: Change allowable value from 3.3% to 4.9% as stated below:

  "The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 4.9% ΔT span."

  Also, revise definition of T_o and T' parameters to clarify that loop specific values, not average values, are used.

- **Function 6, Note 2:** Overpower ΔT: Change allowable value from 2.3% to 2.4% as stated below:

  "The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 2.4% ΔT span."

  Also, revise the definition of T_o and T" parameters as stated in the proposed change for Note 1.

The above proposed changes in allowable values reflect updated uncertainty values for the affected instrument loops and/or changes in safety analysis limit assumptions used in analyses for the proposed stretch uprate conditions. The proposed change for Functions 13 and 14 addresses the industry operating experience regarding measurement uncertainty for steam generator water level.

There are no changes needed for Bases Section 3.3.1.

4. Changes in Allowable Values in Table 3.3.2-1 (ESFAS Instrumentation)

- **Function 1.f** High Steam Flow - Safety Injection, Coincident with Tavg – low: Change the allowable value from $\geq$ 540.75 F to $\geq$540.5 F.

- **Function 1.g** High Steam Flow - Safety Injection, Coincident with steamline pressure – low: Change the allowable value from ≥ 425.0 psig to ≥540.3 psig.

- **Functions 1.f, 1.g, 4.d, and 4.e, Note (b)** regarding turbine first stage pressure: Change allowable values as shown below:

  "Less than or equal to turbine first stage pressure corresponding to ~~53.7~~ 45.9% full steam flow below 20% load, and increasing linearly from ~~53.7~~ 45.9% full steam flow at 20% load to ~~110.8~~ 122.0% full steam flow at 100% load, and ~~110.8~~ 122.0% full steam flow above 100% load."

- **Function 4.d High Steam Flow - Steam Line Isolation, Coincident with Tavg – low:** Change the allowable value from ≥ 540.75 F to ≥540.5 F.

- **Function 4.e High Steam Flow – Steam Line Isolation, Coincident with Steam Line pressure – low:** Change the allowable value from ≥ 425.0 psig to ≥540.3 psig.

- **Function 5.b Feedwater Isolation, SG Water Level – high-high:** Change allowable value from ≤ 77.7% to ≤88.3%.

- **Function 6.b Auxiliary Feedwater, SG Water Level – low-low:** Change allowable value from ≥ 3.7% to ≥3.4%.

The above proposed changes in allowable values reflect updated uncertainty values for the affected instrument loops and/or changes in safety analysis limit assumptions used in analyses for the proposed stretch uprate conditions. The change for Functions 1.g and 4.d reflect an increase in the safety analysis limit from 400 psig to 513 psig for the steam line break analysis. The change for Function 5.b reflects an increase in the safety analysis limit from 80% to 90% steam generator water level.

A change to Bases Section 3.3.2 is needed to reflect the new values used in Note (b) regarding turbine first stage pressure.

5. Revise LCO and related SR limit for minimum RCS flow (TS 3.4.1)

This proposed change is not a result of the stretch uprate program. The existing TS value corresponds to a 'minimum measured flow' that includes uncertainty allowances. The proposed new value is the current RCS thermal design flow (TDF), which has not changed for SPU conditions. Updating the Tech Spec to show TDF versus minimum measured flow is consistent with WCAP-14483 for use of an expanded COLR, which was adopted during the recent conversion to the Standard Technical Specifications. Revisions to Bases Section 3.4.1 are provided for this change.

6. Revise LCO and SR limit for maximum pressurizer water level (TS 3.4.9)

   The proposed change reflects the pressurizer water level corresponding to the maximum value of Tavg (572 F) supported by stretch power analyses. Corresponding changes are proposed for Bases Section 3.4.9.

7. Revise SR for maximum boron concentration for accumulators (TS 3.5.1)

   Maximum boron concentration increased from 2500 ppm to 2600 ppm consistent with the analysis presented in Attachment III. The higher boron concentration provides increased flexibility in future core designs, such as reducing the amount of burnable poisons needed. There are no changes needed for Bases Section 3.5.1.

8. Revise SRs for RWST temperature and boron concentration (TS 3.5.4)

   The increase in maximum RWST temperature (from $\leq$ 100F to $\leq$ 110F) provides additional operational margin for RWST conditions that may be experienced in the summer months. The proposed new higher temperature is used in affected SPU safety analyses.

   RWST Boron Concentration range changed from $\geq$2000 ppm and $\leq$2500 ppm to $\geq$2400 ppm and $\leq$2600 ppm. As stated above for the change to accumulator boron concentration, these values provide additional flexibility for core design and are consistent with and supported by the analysis in Attachment III. Corresponding changes are needed for Bases Section 3.5.4.

9. Revise LCO for power limitations with inoperable MSSVs (TS 3.7.1)

   The proposed changes reflect new limits corresponding to the slightly higher steam flow at SPU conditions. No changes needed to Bases 3.7.1.

10. Revise references for the Core Operating Limits Report, COLR (TS 5.6)

   The proposed change updates the reference listing for SPU analyses. There are no Bases for this Tech Spec section.

3.0 BACKGROUND

Indian Point Nuclear Generating Unit No. 2 is currently licensed for a core thermal power rating of 3114.4 MWt. Approval is requested for an increase in the IP2 core thermal power to 3216 MWt, a 3.26% increase.

ENO evaluated the 3.26% increase on plant systems, components, safety analyses, and dose consequences. Results of these analyses and evaluations are contained in Attachment III of this license amendment submittal.

4.0 TECHNICAL ANALYSIS

Attachment III provided the detailed analyses and justification in the form of WCAP-16157-P, Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate NSSS and BOP Licensing Report. This comprehensive report summarizes the various evaluations and analyses of the potential effects of the 3.26% core power uprating on plant systems, components, safety analyses and dose consequences. The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Attachment III contains the necessary information to support regulatory conformance.

Analysis work in support of this 3.26% Stretch Power Uprate amendment request has been performed consistent with the NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002. This report is based upon the consideration of the guidance, scope of staff's review and information expected in an application as discussed in the draft EPU standard (Reference 3). This report does go beyond the content of the Measurement Uncertainty Recapture (MUR) guidance for power uprate as evidenced by the table of contents and the content of this report. The table of contents was based on an amalgam of several recent large uprates and the work scope that has been completed is designed to address the subject matter of those reports as well as the requests for additional information that have been issued for those and several other previous uprates. In particular, more detail has been provided for the accident analyses since many of the accident analyses have been re-performed to address the increased power level or to amend inputs and parameters to provide additional margin for operations.

The proposed TS changes are supported by applicable analyses and methodologies (as shown in Attachment III) valid for the IP2 nuclear unit. Through these analyses, it has been shown that all applicable acceptance criteria continue to be met.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) has evaluated the safety significance of the increase in the licensed core thermal power identified in the IP2 Technical Specifications and the associated Technical Specification changes according to the criteria of 10 CFR 50.92, "Issuance of Amendment," ENO has determined that the subject change does not involve a Significant Hazards Consideration as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The evaluations and analyses associated with this proposed change to core power level have demonstrated that all applicable acceptance criteria for plant systems, components, and analyses (including the Final Safety Analysis Report Chapter 14 safety analyses) will continue to be met for the proposed increase in licensed core thermal power for IP2. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or the operational performance of any potentially affected system, component or analysis. Therefore, the probability of an accident previously evaluated is not affected by this change. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its intended safety function. Further, the radiological dose evaluations in support of this power uprate effort show all acceptance criteria are met.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The evaluations of this proposed amendment show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for the proposed power increase in IP2 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analyses. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its safety function. Furthermore, the conditions and changes associated with the subject increase in core thermal power will neither cause initiation of any accident, nor create any new credible limiting single failure. The power uprate does not result in changing the status of events previously deemed to be non-credible being made credible. Additionally, no new operating modes are proposed for the plant as a result of this requested change.

Therefore, the subject increase in core thermal power level will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The evaluations associated with this proposed change show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for this proposed increase in IP2 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analysis. The subject power uprate will not adversely affect the ability of any safety-related system to meet its intended safety function.

Therefore, the subject increase in core thermal power will not involve a significant reduction in the margin of safety.

5.6 Applicable Regulatory Requirements / Criteria

The proposed change has been evaluated in accordance with NRC guidance provided in Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002. The analyses and evaluations completed to support the proposed increased in core thermal power demonstrate that acceptance criteria including those established by regulatory requirements continue to be met.

The affect of the new maximum power level on structures, systems, and components of the nuclear steam supply system and the balance-of-plant was evaluated to assure that applicable regulatory requirements and criteria are met. A description of the analyses and evaluations performed is provided in the Stretch Power Uprate Licensing Amendment Report provided with this application for amendment.

ENO has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than those technical specification changes requested in this submittal. Additionally, this change does not affect conformance with any General Design Criteria differently than described in the FSAR.

5.7 Environmental Considerations

The original license environmental evaluations were conducted at 3216.5 MWt (AEC SER dated 10/19/1970). The original license was issued for a rated thermal power of 2758 MWt and subsequent license amendments have been approved for increases in rated thermal power. Amendment 144 (NRC SER dated March 3, 1990) authorized an increase to 3071.4 MWt and Amendment 237 (NRC SER dated May 22, 2003) authorized an increase to 3114.4 MWt. In both of these cases, no environmental impact statement or environmental evaluation was required.

The proposed license amendment to increase rated thermal power to 3216 Mwt, and the related changes to the plant technical specifications do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The NRC has previously approved similar applications for Palo Verde 2 and Kewaunee (under review), and numerous MUR applications including Indian Point 2 and Indian Point 3.

7.0 REFERENCES

1. NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002.

2. Westinghouse WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated January 1983.

ATTACHMENT II TO NL-04-005

MARKUP OF TECHNICAL SPECIFICATION AND BASES

FOR PROPOSED CHANGES REGARDING

INCREASE OF LICENSED THERMAL POWER, 3.26%

- Facility Operating License, page 3
- Technical Specification pages:

Page 1.1-2	Page 3.4.1-1
Page 1.1-4	Page 3.4.1-2
Page 3.3.1-12	Page 3.4.9-1
Page 3.3.1-13	Page 3.4.9-2
Page 3.3.1-14	Page 3.5.1-2
Page 3.3.1-16	Page 3.5.4-2
Page 3.3.1-17	Page 3.7.1-1
Page 3.3.2-6	Page 3.7.1-3
Page 3.3.2-8	Page 5.6-3
Page 3.3.2-9	

- Technical Specification Bases pages:
  (for information only)

  Page B 3.3.2-12
  Pages B 3.4.1-1 thru -5
  Pages B 3.4.9-2 thru -4
  Page B 3.5.4-3

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 220 09-06-01

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility an steady state reactor core power levels not in excess of ~~3114.4~~ 3216 megawatts thermal. Amdt. 237 5-22-03

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 220, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

(3) The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:

1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details and Less Restrictive administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that ~~concentration~~ amount of I-131 ~~(microcuries/gram)~~ (Curies) that alone would produce the same ~~thyroid~~ committed effective dose equivalent (CEDE) dose as the quantity and isotopic mixture of I-130, I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~ Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988
Ē - AVERAGE DISINTEGRATION ENERGY	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 30 minutes, making up at least 95% of the total noniodine activity in the coolant.

1.1 Definitions

OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in UFSAR Chapter 13, "Tests and Operations," b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3114.4~~ 3216 MWt.

Table 3.3.1-1 (page 1 of 6)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11	≤ ~~112.6%~~ 110.6% RTP
b. Low	$1^{(b)},2$	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 29.6% RTP
3. Intermediate Range Neutron Flux	$1^{(b)}, 2^{(c)}$	2	F, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 31.4% RTP
4. Source Range Neutron Flux	$2^{(d)}$	2	H, I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 9.7 E5 cps
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	I, J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 9.7 E5 cps

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks

Table 3.3.1-1 (page 2 of 6)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12	Refer to Note 1
6. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12	Refer to Note 2
7. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1878 psig
b. High	1,2	3	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 2416 psig
8. Pressurizer Water Level - High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 96.9%
9. Reactor Coolant Flow - Low	1(e)	3 per loop	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ ~~88.8%~~ 88.7%
10. Reactor Coolant Pump (RCP) Breaker Position					
a. Single Loop	1(f)	1 per RCP	L	SR 3.3.1.14	NA
b. Two Loops	1(g)	1 per RCP	M	SR 3.3.1.14	NA

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

(g) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) Interlock

Table 3.3.1-1 (page 3 of 6)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
11. RCP Undervoltage (6.9 kV bus)	1(e)	1 per bus	K	SR 3.3.1.9 SR 3.3.1.10	≥ 4959.4 V (primary) ≥ 82.66 V (secondary)
12. RCP Underfrequency (6.9 kV bus)	1(e)	1 per bus	K	SR 3.3.1.9 SR 3.3.1.10	≥ 57.1 Hz
13. Steam Generator (SG) Water Level - Low Low	1,2	3 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ ~~3.7%~~ NR $\boxed{3.4\%}$
14. SG Water Level - Low	1,2	2 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ ~~3.7%~~ NR
Coincident with Steam Flow/Feedwater Flow Mismatch	1,2	2 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 3.88 E6 lbs/hr
15. Turbine Trip Low Auto Stop Oil Pressure	1(f)	3	N	SR 3.3.1.10 SR 3.3.1.14	≥ 26 psig
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	O	SR 3.3.1.14	NA

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 5 of 6)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following:

4.9%

The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than ~~3.3%~~ ΔT span.

$$\Delta T \le \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left[T - T'\right] + K_3(P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F (measured by hot leg and cold leg RTDs).
ΔT_0 is the ~~indicated~~ ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the ~~nominal~~ T_{avg} at RTP, $\le$ [*]°F.

loop specific
indicated

P is the measured pressurizer pressure, psig
P′ is the nominal RCS operating pressure, $\ge$ [*] psig

$K_1 \le$ [*]	$K_2 \ge$ [*]/°F	$K_3 \ge$ [*]/psig
$\tau_1 \ge$ [*] sec	$\tau_2 \le$ [*] sec	

$f_1(\Delta I)$ =	[*] {[*] + $(q_t - q_b)$}	when $q_t - q_b \le$ - [*]% RTP
	0% of RTP	when -[*]% RTP < $q_t - q_b \le$ [*]% RTP
	-[*] {$(q_t - q_b)$ - [*]}	when $q_t - q_b$ > [*]% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

*These values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (page 6 of 6)
Reactor Protection System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following:

The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than ~~2.3%~~ 2.4% ΔT span.

$$\Delta T \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s}{(1+\tau_3 s)} T - K_6 (T - T'') - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT,°F.
ΔT_0 is ~~the indicated~~ loop specific ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T" is the ~~nominal~~ indicated T_{avg} at RTP, ≤ [*]°F.

$K_4 \leq$ [*]	$K_5 \geq$ [*]/°F for increasing T_{avg} [*]/°F for decreasing T_{avg}	$K_6 \geq$ [*]/°F when $T > T^*$ [*]/°F when $T \leq T^*$

$\tau_3 \leq$ [*] sec
$f_2(\Delta I)$ = [*]

*These values denoted with [*] are specified in the COLR.

Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection					
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure - High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 8.6 psig
d. Pressurizer Pressure - Low	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 1801 psig
e. High Differential Pressure Between Steam Lines	1,2,3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 233.0 psid
f. High Steam Flow in Two Steam Lines	1,2,3	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(b)
Coincident with T_{avg} - Low	1,2,3	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	540.5 ≥ ~~540.75~~°F
g. High Steam Flow in Two Steam Lines	1,2,3	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(b)
Coincident with Steam Line Pressure - Low	1,2,3	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	540.3 ≥ ~~425.0~~ psig

(a) Above the Pressurizer Pressure interlock. 45.9%

(b) Less than or equal to turbine first stage pressure corresponding to ~~53.7%~~ full steam flow below 20% load, and increasing linearly from ~~53.7%~~ full steam flow at 20% load to ~~110.8%~~ full steam flow at 100% load, and ~~110.8%~~ full steam flow above 100% load.

45.9% 122.0%

Table 3.3.2-1 (page 3 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	1, $2^{(c)}$, $3^{(c)}$	2 per steam line	F	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	1, $2^{(c)}$, $3^{(c)}$	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure (High-High)	1, $2^{(c)}$, $3^{(c)}$	2 sets of 3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 28.6 psig
d. High Steam Flow in Two Steam Lines	1, $2^{(c)}$, $3^{(c)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(b)
Coincident with T_{avg} - Low	1, $2^{(c)}$, $3^{(c)}$	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	540.5 ≥ ~~540.75~~°F
e. High Steam Flow in Two Steam Lines	1, $2^{(c)}$, $3^{(c)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(b)
Coincident with Steam Line Pressure - Low	1, $2^{(c)}$, $3^{(c)}$	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	540.3 ≥ ~~425.0~~ psig

45.9%

(b) Less than or equal to turbine first stage pressure corresponding to ~~53.7%~~ full steam flow below 20% load, and increasing linearly from ~~53.7%~~ full steam flow at 20% load to ~~110.8%~~ full steam flow at 100% load, and ~~110.8%~~ full steam flow above 100% load.

(c) Except when all MSIVs are closed. 122.0%

Table 3.3.2-1 (page 4 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	1, 2(d), 3(d)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
b. SG Water Level - High High	1, 2(d), 3(d)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ ~~77.7% NR~~ 88.3%
c. Safety Injection	(d) Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
b. SG Water Level - Low Low	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ ~~3.7% NR~~ 3.4%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
d. Station Blackout (SBO) (Undervoltage Bus 5A or 6A)	1,2,3	Refer to LCO 3.3.5, "LOP DG Start Instrumentation," for requirements.			
e. Trip of Main Boiler Feedwater Pump	1(e), 2(e)	1 per MBFP	H	SR 3.3.2.6 SR 3.3.2.7	≥ 19.5 psig
7. ESFAS Interlocks Pressurizer Pressure	1,2,3	3	J	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 1980 psig

(d) Except when the main feedwater flowpath to each SG is isolated by a closed and deactivated automatic valve or a closed manual valve.

(e) Only required for MBFPs that are in operation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limit specified in the COLR,

b. RCS average temperature is less than or equal to the limit specified in the COLR, and

c. RCS total flow rate ≥ ~~331,840 gpm.~~ 322,800 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

- NOTE -

Pressurizer pressure limit does not apply during either:

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ ~~331,840 gpm.~~	12 hours
SR 3.4.1.4	- NOTE - Not required to be performed until 24 hours after ≥ 90% RTP. Verify by precision heat balance that RCS total flow rate is ≥ ~~331,840 gpm.~~	24 months

322,800 gpm and greater than or equal to the limit specified in the COLR.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ ~~60.6%~~ 65.1% and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW with each group powered from a different safeguards power train.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Pressurizer water level not within limit.	A.1	Be in MODE 3.	6 hours
	AND		
	A.2	Fully insert all rods.	6 hours
	AND		
	A.3	Place Rod Control System in a condition incapable of rod withdrawal.	6 hours
	AND		
	A.4	Be in MODE 4.	12 hours
B. One required group of pressurizer heaters inoperable.	B.1	Restore required group of pressurizer heaters to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ ~~60.6%.~~ 65.1%	12 hours
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	Verify borated water volume in each accumulator is $\geq$ 723 cubic feet and $\leq$ 875 cubic feet.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is $\geq$ 598 psig and $\leq$ 685 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is $\geq$ 2000 ppm and $\leq$ ~~2500~~ 2600 ppm.	31 days
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is $\geq$ 2000 psig.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is ≥ 40°F and ≤ ~~100~~ 110°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 345,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ ~~2000~~ 2400 ppm and ≤ ~~2500~~ 2600 ppm.	31 days
SR 3.5.4.4	Calibrate RWST level low low alarms to ensure the alarm setpoint is ≥ 74,200 gallons and ≤ 99,000 gallons.	92 days

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each MSSV.

CONDITION		REQUIRED ACTION	COMPLETION TIME
A. One or more steam generators with one MSSV inoperable.	A.1	Reduce THERMAL POWER to ≤ ~~60%~~ 57% RTP.	4 hours
B. One or more steam generators with two or three MSSVs inoperable.	B.1	Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours
	<u>AND</u>		
		- NOTE - Only required in MODE 1.	
	B.2	Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours

Table 3.7.1-1 (page ~~1~~ of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
4	~~59~~ 57
3	~~40~~ 38
2	~~21~~ 20

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. Technical Specification 3.2.3, Axial Flux Difference (AFD);

9. Technical Specification 3.3.1, Reactor Protection System Instrumentation;

10. Technical Specification 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits; and

11. Technical Specification 3.9.1, Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985;

2. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report", September 1974;

3. T.M. Anderson to K. Kniel (NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package;

4. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981, including Branch Technical Position CPB 4.3-1,Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981;

~~5. WCAP-10266-P-A Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using Bash Code", March 1987; and~~

Insert References from the next page.

6. WCAP-12945-P, Westinghouse "Code Qualification Document for Best Estimate LOCA Analyses", July, 1996.

~~7. Caldon, Inc. Engineering Report 80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System," Revision 0, March 1997, and Caldon, Inc. Engineering Report 160P, "Supplement to Topical Report ER 80P: Basis for a Power Uprate With the LEFM System," Revision 0, May 2000.~~

5. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989;

7. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

8. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995;

9. WCAP-10079-P-A, NOTRUMP A Nodal Transient Small Break and General Network Code," August 1985;

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985; and

11. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and Cosi Condensation Model," July 1997.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

One channel of T_{avg} per loop and one channel of low steam line pressure per steam line are required OPERABLE. For each parameter, the channels for all loops or steam lines are combined in a logic such that two channels tripped will cause a trip for the parameter. The Function trips on one-out-of-two high flow in any two-out-of-four steam lines if there is one-out-of-one low T_{avg} trip in any two-out-of-four RCS loops, or if there is a one-out-of-one low pressure trip in any two-out-of-four steam lines. Since the accidents that this event protects against cause both low steam line pressure and low T_{avg}, provision of one channel per loop or steam line ensures no single random failure can disable both of these Functions. The steam line pressure channels provide no control inputs. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate.

The Allowable Value for high steam flow is a linear function that varies with power level (Turbine first stage pressure). The function is a ΔP corresponding to approximately ~~53.7%~~ 45.9% of full steam flow between 0% and 20% load to ~~110.8%~~ 122.0% of full steam flow at 100% load. The nominal trip setpoint is similarly calculated.

With the transmitters located inside the containment (T_{avg}) or inside the auxiliary feed pump room (High Steam Flow), it is possible for them to experience adverse steady state environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). SLB may be addressed by Containment Pressure High (inside containment) or by High Steam Flow in Two Steam Lines coincident with Steam Line Pressure - Low, for Steam Line Isolation, followed by High Differential Pressure Between Two Steam Lines, for SI. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

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

Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.

limits are

The RCS pressure ~~limit is~~ consistent with operation within the nominal operational envelope. A lower pressure will cause the reactor core to approach DNB limits.

Because there may be differences between the loop temperatures, both the average of the loop Tavg values and the highest of the loop Tavg values are compared to their respective limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. ~~Because there may be differences between the loop temperatures, the loop with the highest indicated value of T_{avg} is assumed to be the RCS average temperature and the loop with the highest average temperature is compared to the acceptance criteria.~~ A higher 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 corresponds to that assumed for DNB analyses. A lower RCS flow will cause the core to approach DNB limits.

Flow rate indications are averaged to come up with a value for comparison to the limit.

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 LCO 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 LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the

BASES

APPLICABLE SAFETY ANALYSES (continued)

core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

consistent with

and include allowance for measurement uncertainty

The pressurizer pressure limit and RCS average temperature limit specified in the COLR are ~~based on~~ the analytical limits used in the safety analyses~~.~~ ~~Therefore, appropriate allowances for measurement and instrument uncertainty must be included when comparing the observed value with the analytical limits.~~

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, which is based on maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. ~~The RCS total flow rate limit of 331,840 gpm includes a measurement uncertainty of 2.8% associated with the performance of Reactor Coolant System Flow Calculation. This value does not include instrument uncertainty which must be accounted for when flow is monitored using control room instruments.~~

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location and have been adjusted for instrument error.

~~The numerical values for pressure and temperature specified in the COLR are the analytical limits used in the safety analyses. Unless otherwise specified in the COLR, these values do not include instrument uncertainty which must be accounted for when these parameters are monitored using control room instruments.~~

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate 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. In all other MODES, the power level is low enough that DNB is not a concern.

BASES

APPLICABILITY (continued)

A Note has been added to indicate 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 counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

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

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

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.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, 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.

BASES

SURVEILLANCE REQUIREMENTS

Replace the information on this page with the next page.

~~SR 3.4.1.1~~

~~SR 3.4.1.1 requires verification every 12 hours that pressurizer pressure is greater than or equal to the limit specified in the COLR. Pressurizer pressure indications in the control room are averaged to come up with a value for comparison to the limit. Unless otherwise specified in the COLR, the limit specified in the COLR does not include instrument uncertainty which must be accounted for when parameters are monitored using control room instruments. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

~~SR 3.4.1.2~~

~~SR 3.4.1.2 requires verification every 12 hours that RCS average temperature is less than or equal to the limit specified in the COLR. Because there may be differences between the loop temperatures, the loop with the highest indicated value of T_{avg} is assumed to be the RCS average temperature and the loop with the highest average temperature is compared to the acceptance criteria. Unless otherwise specified in the COLR, the limit specified in the COLR does not include instrument uncertainty which must be accounted for when parameters are monitored using control room instruments. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

~~SR 3.4.1.3~~

~~SR 3.4.1.3 requires verification every 12 hours that RCS total flow rate is greater than or equal to the limit specified in this LCO. RCS flow indications for each loop are averaged to come up with a value for comparison to the limit. The limit specified in this LCO does not include instrument uncertainty which must be accounted for when parameters are monitored using control room instruments. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.~~

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 24 months allows the installed RCS flow instrumentation to be calibrated and verifies 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, which may have caused an alteration of flow resistance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.4.1.4~~

~~Measurement of RCS total flow rate by performance of a precision calorimetric heat balance is required once every 24 months. The results are adjusted to account for an uncertainty of 2.8% associated with the performance of the flow measurement. These adjusted results are compared to the LCO limit to ensure that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.~~

~~The results are used as one of the inputs to the calibration of the installed RCS flow instrumentation.~~

~~The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.~~

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. UFSAR, Section 14.

BASES

BACKGROUND (continued)

Pressurizer heaters are powered from either the offsite source or the diesel generators (DGs) through the four 480 V vital buses as follows:

Safeguards Power Train 5A supports heater group 23 (485 kW);

Safeguards Power Train 6A supports heater group 24 (277 kW); and

Safeguards Power Train 2A/3A supports both:
- heater group 21 from Bus 3A (554 kW); and
- heater group 22 from Bus 2A (485 kW).

APPLICABLE SAFETY ANALYSES

71%

is based on the pressurizer program level at a full power Tavg of 572 Degrees F (65%) plus a 6.0%

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. For events that result in pressurizer insurge (e.g., loss of normal feedwater and the loss of load/turbine trip), the analyses assume that the limiting ~~nominal~~ value for the highest initial pressurizer level is ~~60.6%~~. This is an analytical limit and ~~does not include an~~ allowance for instrument error. For other events, the nominal value of pressurizer level is assumed because ~~the pressurizer level is automatically controlled and~~ the effect of initial pressurizer level on PCT is small (Ref. 1). Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

of the pressurizer program level at a full power Tavg of 572 Degrees F (65%) plus a 6.0% allowance for instrument error.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

71%

71%

The LCO requires the pressurizer to be OPERABLE with the actual water level less than or equal to ~~60.6%~~. This maximum pressurizer level of ~~60.6%~~ is the nominal level ~~that is used as the analytical limit for the initial condition in the accident analysis.~~ Pressurizer level indications in the control room are

BASES

LCO (continued)

A maximum allowance for instrument error of 5.9% (based on 2 channel measurement) applied to the analytical limit of 71% results in an indicated level that should not exceed 65.1%.

averaged to come up with a value for comparison to the limit. ~~An additional margin of approximately 5%, should be allowed for instrument error (i.e., the indicated level should not exceed 55.6%).~~

Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW. Each of the two groups of pressurizer heaters must be powered from a different DG to ensure that the minimum required capacity of 150 kW can be energized during a loss of offsite power condition assuming the failure of a single DG. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The value of 150 kW has been demonstrated to be adequate to maintain RCS pressures control.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

BASES

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that the redundant heater group is still available and the low probability of an event during this period. Pressure control may be maintained during this time using the remaining heaters.

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The LCO requires that the actual pressurizer water level less than or equal to ~~60.6%~~ 71%. Pressurizer level indications in the control room are averaged to come up with a value for comparison to the limit. An additional margin of approximately ~~6%~~ 5.9%, should be allowed for instrument error (i.e., the indicated level should not exceed ~~55.6%~~ 65.1%). The Frequency of 12 hours

BASES

APPLICABLE SAFETY ANALYSES (continued)

The specified minimum water quantity for the RWST (345,000 gallons) includes the minimum quantity required for the injection phase (246,000 gallons) for accident mitigation, the minimum quantity of water required during the recirculation phase (60,000 gallons) for accident mitigation, and a sufficient quantity of water (39,000 gallons) to allow for instrument inaccuracies, additional margin, and for water that is unavailable from the bottom of the tank.

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.

2600

The upper limit on boron concentration of ~~2500~~ ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 40°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. Additionally, the lower limit on RWST temperature limits thermal shock to the ECCS injection nozzles and the reactor pressure vessel. The upper temperature limit of ~~100~~°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment. 110

Following a LOCA, switchover from the injection phase to the recirculation phase must occur before the RWST empties to prevent damage to the pumps and a loss of cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment to support recirculation pump suction.

ATTACHMENT III TO NL-04-005

WCAP-16157-P (Proprietary)

Indian Point Nuclear Generating Unit No. 2
Stretch Power Uprate
NSSS and BOP Licensing Report

NOTE: Attachment III report is not included with packages for non-proprietary distribution. WCAP-16157-NP (Non-Proprietary) is provided in lieu of Attachment III

ENCLOSURES TO NL-04-005

A. Westinghouse authorization letter dated January 27, 2004 (CAW-04-1778), with the accompanying affidavit, Proprietary Information Notice, and Copyright Notice

B. WCAP-16157-NP, "Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate NSSS and BOP Licensing Report," dated January 2004. (Non-Proprietary)

Enclosure B is included in lieu of Attachment III for packages issued to non-proprietary distribution.

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

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Our ref: CAW-04-1778

January 27, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16157-P, "Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate NSSS and BOP Licensing Report" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1778 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Entergy Nuclear Northeast.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1778, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: D. Holland
B. Benney
E. Peyton

CAW-04-1778

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 27th day
of January, 2004

Sharon L. Fiori
Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16157-P, "(Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate NSSS and BOP Licensing Report)" (Proprietary) dated January 2004, being transmitted by the Entergy Nuclear Northeast letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for the Indian Point Nuclear Generating Unit No. 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of Stretch Power Uprate License Amendment Request.

This information is part of that which will enable Westinghouse to:

(a) Provide information in support of plant power uprate licensing submittals.

(b) Provide plant specific calculations.

(c) Provide licensing documentation support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation associated with power uprate licensing submittals.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations, evaluations, analyses and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

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