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September 13, 2002
IPN-02-073

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Indian Point Nuclear Generating Unit No.3
Docket No. 50-286
Reply to Request for Additional Information
Regarding Proposed License Amendment for
1.4% Measurement Uncertainty Recapture Power Uprate

REFERENCES:

1. NRC letter to Entergy Nuclear Operations, Inc; "Request for Additional Information, TAC NO. MB5297," dated August 26, 2002.

2. Entergy letter to NRC, IPN-02-041, "Request for License Amendment for 1.4% Measurement Uncertainty Recapture Power Uprate," dated May 30, 2002.

Dear Sir:

This letter provides the additional information requested by the NRC in Reference 1 regarding the license amendment request submitted by Entergy Nuclear Operations, Inc (ENO) in Reference 2. The additional information is provided in Attachment I.

The NRC also requested that the ENO response include a copy of WCAP-15824, which documents the methodology used to determine power calorimetric uncertainties. Accordingly, ENO is enclosing the following:

1. One copy of WCAP-15824-P, Rev 1, "Power Calorimetric Uncertainty for the 1.4 Percent Uprating of Indian Point Unit 3," dated May 2002. (Proprietary)

2. One copy of WCAP-15904, "Power Calorimetric Uncertainty for the 1.4 Percent Uprating of Indian Point Unit 3," dated September 2002. (Non-Proprietary)

Also enclosed is Westinghouse authorization letter dated September 11, 2002 (CAW-02-1551), with the accompanying affidavit, Proprietary Information Notice, and Copyright Notice. As Item 1 contains information proprietary to Westinghouse Electric Corporation, it is accompanied by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the

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Commission's regulations. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790 of the Commission's regulations.

Correspondence with respect to the copyright on proprietary aspects of the items listed above or the supporting affidavit should reference CAW-02-1551 and should be addressed to H. A. Sepp, Regulatory and Licensing Engineering, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

In addition to the information requested by the NRC, ENO is providing in Attachment II the following supplemental material and corrections to Reference 2:

- A markup page of the Facility Operating License to reflect the proposed new licensed power level. This page was inadvertently omitted from the Technical Specification and Bases markup pages (Attachment II of Reference 2).

- Revised summary of steam generator tube structural limits. The limits previously stated for the region of the anti-vibration bars (AVBs) were inadvertently based on the depth of the AVBs rather than the width. The limiting minimum wall requirement, previously defined for the straight leg region of the tubes, is unchanged. Although the new limit in the AVB region for the inner tubes is now equal to the limit for the straight leg region, operating experience has shown that tube-to-AVB wear occurs primarily for the outer row of tubes. A new page is provided to replace page 7-36 in Attachment III of Reference 2.

- Revised discussion regarding the Main Generator. The discussion previously stated that limit for 'leading MVARs' was a result of the instability limits shown on the generator capability curve. The discussion should state that this limit is a result of an excitation limiter setting shown on the generator capability curve. A new page is provided to replace page 9-3 in Attachment III of Reference 2.

The responses to the NRC questions and the supplemental information being provided does not change the conclusions of the no significant hazards evaluation previously provided in Reference 2. There are no new commitments identified in this letter. If you have any questions or require additional information, please contact Mr. Kevin Kingsley, NRR Project Manager at 914-734-5581.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 9/13/02

Very truly yours,

Robert J. Barrett
Vice President, Operations – IP3
Indian Point 3 Nuclear Power Plant

Attachments and Enclosures as stated

cc: next page

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ATTACHMENT I TO IPN-02-073

RESPONSE TO NRC QUESTIONS REGARDING

PROPOSED LICENSE AMENDMENT REQUEST FOR

1.4% MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

Question 1:

List the instrumentation uncertainty components used as inputs to the reactor power uncertainty calculation, including their associated measurement uncertainties and uncertainty values with respect to power. Discuss the methodology used. Provide data showing that the mathematical combination of these uncertainties is less than the stated 0.6% (with Caldon LEFM check flow elements installed) for IP3.

Response 1:

This information is contained in WCAP-15824, Rev.1, which is provided in the enclosure to this letter in response to Question 6.

Question 2:

In Attachment 3, Section 3.3, paragraph 8, the licensee states "If the plant experiences a power change of greater than 10% during the seven-day period, the permitted maximum power level would be reduced upon return to full power, in accordance with the power levels described below, since a plant transient may result in calibration changes of the alternate instrumentation."

a. Describe the significance of the 10% power change limit

b. Confirm that the stated 10% power changes do not involve power increases above the steady state power level at the time of Caldon LEFM check system becomes inoperable.

Response 2a:

The phrase "calibration changes of the alternate instrumentation" in Section 3.3 refers specifically to calibration changes of feedwater flow nozzles which can occur due to changes in the corrosion deposits (fouling) on the flow nozzle throat. Such changes can occur due to pH changes associated with feedwater temperature changes. Feedwater temperature changes associated with industry documented nozzle fouling and de-fouling are typically much greater than would occur with a 10% power change at or near full power. The 10% change is chosen to provide conservatism between the permitted power reduction and a significant power reduction from full-power conditions that is likely to change nozzle fouling. A significant power reduction is estimated to exceed a 20% power change.

Response 2b:

The stated 10% power changes do not involve power increases above 100% indicated power level by the LEFM Check System (maximum permitted power level) in any case. However, there would potentially be power increases above the steady-state power level at the time the Caldon LEFM Check System becomes inoperable, but only if the power level at the time the LEFM became inoperable is less than 100% indicated. For example, suppose that the plant is operating at 95% power when the LEFM Check System becomes inoperable and then the plant experiences a power reduction to 75% power and again approaches full power while operating on the alternate instrumentation. The plant would proceed to increase power past 95% indicated power using the alternate instruments, but would (by procedure) stop short of full indicated power. The plant would remain limited to the power levels described in Attachment 3 until the LEFM Check System again becomes operable. The plant would stop short of full indicated power but would in this case intentionally exceed the 95% power level at which the LEFM Check System had been operating.

Question 3:

In Attachment 3, Section 3.3, paragraph 4, the licensee states, "While recognizing that the accuracy of the alternate instruments may degrade over time, it is considered likely that any degradation as a result of nozzle fouling, drift and the like, would be imperceptible for the seven-day period as long as steady-state conditions persist." What is the measurement uncertainty, in terms of thermal power, of the alternate instruments, over the seven-day period?

Response 3:

The absolute uncertainty of the Barton Venturi nozzles is documented in Westinghouse's Revised Thermal Design Procedure (RTDP) Instrument Uncertainty Methodology dated March 1993 as 1.97%. The Venturi calorimetric was trended over Cycle 12 and found to be constant when compared to the LEFM calorimetric. A calorimetric was performed once a week with both the Barton Venturi nozzles and the LEFM concurrently. During the more than one-year period, the variation is relatively consistent. There was only approximately a 0.3% variation with no consistent drift in any direction. Therefore, if the LEFM fails, we could be relatively certain that a Barton calorimetric would be accurate for a seven-day period with an adjustment factor based on the preceding data.

Question 4:

In Attachment 3, Section 3.4, paragraphs 6, 7, and 9, the licensee discusses the control of software and hardware configuration of the Caldon LEFM Check system but does not mention other instrumentation that affect the power calorimetric. Provide a discussion of the control of software and hardware configuration of other plant instrumentation that affect the power calorimetric (See NRC RIS 2002-03 Item I.1.F).

Response 4:

In addition to the LEFM Check System, which is the source for Main Feedwater Mass Flows and Temperatures (4 individual monitored feedwater lines), the following additional instrumentation loops are employed for active input to the Secondary Side Calorimetric Algorithm:

1. Main Steam Pressures (3 instrument loops per line, total of 12 inputs)
2. Main Feedwater Pressures (one instrument loop per line, total of 4 inputs)
3. Steam Generator Blowdown Flows (one instrument loop per SG, total of 4 inputs)
4. Steam Generator Total Blowdown Flow (one instrument loop for all 4 SGs)
5. Reference NSSS T_{avg} (one instrument loop derived from Turbine 1st Stage Pressure)
6. NSSS Loop T_{avg} (one instrument loop per NSSS leg, total of 4 inputs)

The Pressure and Blowdown Flow signals are directly used in the computational functions that determine the thermal energy levels inherent in computing Reactor Power level. The NSSS temperatures, on the other hand, are not used for this purpose but instead provide confirmation that the NSSS operating point is stable and within a reasonable window of anticipated T_{avg} for the particular nominal plant power level.

The above loops are hardwired current-loop systems that deliver their respective values to the plant computer system for input to the calorimetric algorithm. In the case of Feedwater Pressures, the LEFM Check System also directly receives these hardwired inputs that are used by the LEFM's Main Feedwater Specific Gravity determination algorithm.

The Main Steam Pressure instrument loops, NSSS T_{avg} loops and various Steam Generator Blowdown instruments are currently monitored for drift via the IP3 Performance Monitoring Program that was instituted in response to NRC requirements for Surveillance Extension Programs. The new Main Feedwater Pressure instruments and the currently unincorporated components of the Steam Generator Blowdown instrument loops will be added to this program as part of implementation of the Appendix K Power Uprate.

The hardware configuration for all the above instrument systems is controlled via the plant's Modification Control and Design Control procedures which are designed to conform to 10CFR50 Appendix B. The software propagations and computational executions performed on these signals are controlled by Entergy's Software Quality Assurance Program (ENN-IT-104) which addresses development, testing, implementation, cataloging, modifying and retiring the plant's software based functions.

Question 5:

Attachment 3, Section 3.7, the licensee states, "The Caldon LEFM Check flow element calibrations were based upon Alden Research Laboratory (ARL) testing of a population of seven flow elements with identical inside diameters and dimensions." Were the flow elements tested in an IP3 plant-specific configuration at ARL? Provide the reference and results of the tests.

Response 5:

The seven flow elements cited in Section 3.7 of Attachment 3 were tested in straight pipe. These test configurations were not IP3 plant-specific configurations. Six of the seven flow elements were tested by Westinghouse at ARL, and are the same six 16-inch diameter flow elements whose testing is discussed in some detail in Appendix F of Caldon's NRC-approved report ER-80P. The seventh flow element was tested by Caldon in 1997 at ARL. The results of the testing for these meters are summarized as follows.

Spool Number	Measured Profile Factor
1	1.0026
2	1.0006
3	1.0014
4	1.0025
5	1.0011
6	1.0006
7	1.0019

Appendix F of ER-80P provides considerable technical justification that the Westinghouse program in the 1970's and 1980's proved that the profile factor for the four-path LEFM meter design (whose critical characteristics are unchanged in the Caldon LEFM Check System) is not very sensitive to varying piping geometry. Specifically, ER-80P demonstrates that in nearly all fluid system configurations the profile factor is within 0.1% to 0.3% of what it is in straight pipe. The only exceptions to this statement are associated with off-center swirling conditions, occurring downstream of closely-coupled non-planar bends. The piping bends upstream of the LEFM Check meters at IP3 are not closely-coupled, and therefore the profile factors for the meters at IP3 will fall within 0.30% of the straight pipe profile factor.

Over the past fifteen years Caldon has performed significant additional testing of four and eight-path meters associated with installations in nuclear power plants, and Caldon's testing has supported the conclusions of the Westinghouse program. Much of the relevant work is summarized in Caldon's report ER-262, "Effects of Velocity Profile Changes Measured In-Plant on Feedwater Flow Measurement Systems", which is currently under review by the NRC staff.

The profile factor uncertainty accounted for the meters at IP3 includes the following contributors:

- 0.25% uncertainty for the laboratory calibrations at Alden,
- 0.18% uncertainty for the statistical variation in the profile factor for the population of seven meters used as reference,
- 0.30% uncertainty to bound potential differences between the profile factors for the IP3 meters as they are located downstream of bends and therefore their profile factors will differ from the straight pipe profile factor by an amount bounded by 0.30%.

Because the profile factor uncertainty includes an explicit allowance of 0.30% for upstream hydraulic conditions, the profile factor uncertainty is bounded for this application.

Question 6:

Attachment 3, Section 4.1 references WCAP-15824, "Power Calorimetric for the 1.4% Uprating for Entergy Nuclear Indian Point Unit 3." Submit the referenced WCAP-15824 to NRC for staff review.

Response 6:

WCAP-15824 is provided in the enclosure to this letter.

Question 7:

The nuclear steam supply system operating point parameters for the power uprate conditions were calculated for a core power of 3,067.4 MWt. List the Final Safety Analysis Report (FSAR) Chapter 14 transients and accidents analyses which incorporate these uprate operating point parameters. For those that do not, provide justification that the current values used in the analyses are bounding.

Response 7:

Section 8 of Attachment 3 included Table 8-1 and associated text that delineate and explain how the various IP3 UFSAR Chapter 14 safety analyses events were classified. The basic classifications are "Affected" or "Unaffected" events according to the NRC guidance in RIS 2002-03. Unaffected events are those with existing analyses or record that bound plant operation at the 1.4% power level conditions. Therefore, new analyses were not necessary for "Unaffected" events (already listed in Table 8-1), and the NSSS operating parameter data were not explicitly incorporated into the existing licensing basis analyses for "Unaffected" events.

There were seven "Affected" non-LOCA events (already explicitly listed in Table 8-1). Three of the seven "Affected" events required reanalysis to address the potential effects of the 1.4% power uprate. These three events explicitly incorporate the 1.4% power uprate operating conditions and are discussed in detail in Section 8.3.3 of Attachment III. The potential effects of the 1.4% power uprate on the other four "Affected" events were evaluated as described in Section 8.3.4 of Attachment III. These evaluations did not explicitly incorporate the 1.4% power uprate operating conditions into the existing safety analyses. However, as described in Section 8.3.4, it has been determined by evaluation that the changes in the operating conditions have an insignificant effect on the results of the existing safety analyses of record.

Question 8:

Provide a quantitative discussion confirming that the low temperature overpressure protection relief valves have adequate relief capacity to remove the additional decay heat generated by the 1.4% power uprate such that there is no increase in peak pressure for this transient. Include a discussion of the NRC-approved methodology used to perform this analysis.

Response 8:

Cold Overpressure Mitigation System (COMS) events, also known as Low Temperature Overpressure Protection System (LTOPS) events, could potentially occur during cold shutdown operation (RCS temperature less than about 350ºF). The 1.4% power uprating is not changing any plant condition for which COMS/LTOPS is affected. For these events, the plant is in a shut down condition, so the uprating does not affect the plant response for these events.

The Low Temperature Overpressure Protection System (LTOPS) at IP3 is designed to respond to mass and heat injection events when the RCS is below the arming temperature (currently 319°F). This is well below the normal operating Tavg of 547°F and above. The Mass Injection event would be the result of inadvertent charging or safety injection, neither of which is related to decay heat rate. The Heat Injection event presumes the addition of latent secondary heat from the Steam Generator to the RCS subsequent to pump start; it is bounded by an allowed primary-to-secondary temperature difference of 64°F (TS 3.4.12) and is not related to decay heat.

The intent of COMS or LTOPS is to prevent violation of the reactor vessel Appendix G pressure-temperature limits. The Appendix G limit is not changing due to the 1.4% power uprating. Therefore, there is no effect on COMS/LTOPS due to the 1.4% power uprating.

Also, the calculation upon which the LTOPS setpoint is based (IP3-CALC-RCS-02444) explains in quantitative detail the derivation of the setpoint and all other low-temperature RCS limits. This calculation was previously submitted to the NRC to support IP3 Technical Specification Amendment No. 179.

Therefore, there is no aspect of the IP3 COMS/LTOPS or setpoints that is challenged in any way by a 1.4% increase in decay heat load.

Question 9:

With respect to the impacts of the proposed power uprate on the nuclear, thermal-hydraulic and fuel rod design analyses, list the NRC-approved codes and methodologies used for the design analyses discussed in Section 7.10. Confirm that all parameters and assumptions to be used for analyses described in Section 7.10 remain within any code limitations or restrictions.

Response 9:

The codes and methods used for the 1.4% power uprate design analyses discussed in Section 7.10 of the Attachment 3 are the same as those used for the following Westinghouse reload methodologies, previously approved by the NRC.

Reload Methodology:

- Davidson, S. L., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (proprietary) and WCAP-9272-NP-A (non-proprietary), July 1985.

Nuclear Design:

- Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

- Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

Fuel Rod Design:

- Slagle, W. H. (Editor), et al., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 (Proprietary) and WCAP-15064-NP-A, Revision 1 (Non-Proprietary), July 2000.

Thermal and Hydraulic Design:

- Sung, Y. et al, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A/WCAP-15306-NP-A, October 1999.

Confirmation of parameters and assumptions used for the fuel-related analyses described in Section 7.10 is performed in two phases:

Phase I consisted of the analyses performed for the non-LOCA statepoints that changed as a result of the 1.4% power uprate conditions. For the statepoints that did change, an analysis was performed to show that the new statepoints meet the current design basis for a mid-cycle uprate of IP3 Cycle 12. Standard Westinghouse core reload methodologies described in Reference 1 (above) are used for these analyses.

Phase II addresses the non-LOCA statepoints that do not change, but still need to be addressed for the change in core power conditions. As with the Phase I calculations, these statepoints are analyzed and addressed using standard Westinghouse core reload methods and codes. In order to minimize uncertainties with respect to fuel exposure, these analyses are typically performed approximately two months prior to the actual implementation date of the 1.4% power uprate. The results and conclusions are being summarized in a revision of cycle-specific Reload Safety Evaluation (RSE) document for the current cycle. These calculations will be subsequently repeated and summarized in the RSE's for future operating cycles.

Question 10:

Provide a more detailed anticipated transient without scram (ATWS) evaluation that is applicable to IP3 at power uprate conditions to demonstrate that the peak primary system pressure will not exceed the American Society of Mechanical Engineers Stress Level C limits of 3200 psig. Justify that the assumptions for the analyses are adequate as they relate to input parameters such as the initial power level, current fuel enrichment, moderator temperature coefficient (MTC), pressurizer safety and relief valves capacity, reactor coolant system volume, steam generator pressure, auxiliary feedwater (AFW) flow rate and its actuation delay time, and the setpoint for the ATWS mitigation system actuation circuit to actuate the AFW and trip the turbine. The submittal should include a discussion and applicable values of the unfavorable exposure time for the MTC assumed in the analyses. Explain why the TS value of MTC less than zero would assure the assumed MTC value in the ATWS analysis.

Response 10:

A detailed evaluation concerning anticipated transients without scram (ATWS) is presented in Section 8.3.6.8 of Attachment III. The level of detail included in Section 8.3.6.8 by Westinghouse is consistent with the level of detail provided in a response to a similar RAI that was generated by the NRC regarding the South Texas 1.4% power uprate project.

Section 8.3.6.8 discusses that with the current plant configuration and licensed power level for IP3, the peak RCS pressure for the limiting loss of load ATWS event is 2979 psia. Sensitivities performed for a generic four-loop PWR demonstrated that a 2% increase in power would result in a maximum 44 psi increase in the peak RCS pressure for the limiting loss of load ATWS event. Therefore, since a 2% increase in power would result in a maximum peak RCS pressure of 3023 psia, which is less than 3200 psig, a 1.4% increase in power should not challenge the stress limits.

Section 8.3.6.8 also includes a thorough discussion concerning the ATWS analysis assumptions for power level, moderator temperature coefficient (MTC), pressurizer relief and safety valve capacities, and the auxiliary feedwater (AFW) flow rate (including unfavorable exposure time).

Question 11:

Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and revision 1, NSAL 02-4 and NSAL 02-5, to document the problems with the Westinghouse designed steam generator (SG) water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15, 2002, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG measurements, affecting the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation, affecting the high-high level trip setpoint. NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. These analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how IP3 accounts for these uncertainties as documented in the NSALs in determining the SG water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the loss-of-coolant accident (LOCA) and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still limiting.

Response 11:

IP3 is operating with Westinghouse Model 44F steam generators. In comparison to other Westinghouse steam generator models, the Model 44F steam generator has a relatively large flow area through the mid-deck plate region of the steam generator and, as such, there is essentially no pressure drop across the mid-deck plate region of the Model 44F steam generators. This is shown for IP3 in the attachment to NSAL 02-3, Rev.1. Therefore, with respect to NSAL 02-3 and NSAL 02-5, IP3 is not affected and the current safety analyses remain limiting.

With respect to NSAL-02-4, a 15% void fraction above the mid-deck plate had to be considered for the IP3 Model 44F steam generators. Specifically, for the UFSAR Chapter 14 safety analyses, the issue described in NSAL 02-4 only potentially affected the Excessive Feedwater transient. Since the analysis for this transient assumes a conservatively large high-high steam generator water level safety analysis limit of 100% (narrow range span, NRS), in comparison to the nominal trip setpoint of 75% NRS, IP3 is not affected by NSAL 02-4 and the current safety analyses remain limiting.

Question 12:

While reviewing large-break loss-of-coolant accident (LBLOCA) models for power uprates, the NRC staff has recently found plants that require changes to their operating procedures because of inadequate hot leg switch-over times and boron precipitation modeling. Demonstrate that IP3 LBLOCA model continues to comply with 10 CFR 50.46 during the switch-over from the refueling water storage tank to the containment sump. Also, discuss how the analyses account for boric acid buildup during long-term core cooling. Discuss how the predicted time to initiate hot leg injection corresponds to the times in the IP3 operating procedures.

Response 12:

The methodology used to confirm post-LOCA long-term core cooling capabilities for IP3 establishes a post-LOCA Hot Leg Switchover (HLSO) time to support realignment of the recirculation safety injection (SI) flow from the cold legs to the hot legs. This realignment is required to preclude boron precipitation in the reactor vessel following a large-break LOCA. For a cold-leg break where injected SI water would boil off due to decay heat, the potential would exist for the boric acid solution in the reactor vessel to reach the boron precipitation point and impede core cooling flow. The Westinghouse emergency core cooling system (ECCS) long-term core cooling model is used to confirm the existence of a coolable core geometry by establishing HLSO times which ensure that boron precipitation does not occur.

The HLSO analytical model used as part of the long-term core cooling methodology is based on the following assumptions:

A boric acid concentration level is computed over time for a core-region mixing volume. Other than the steam exiting through the hot legs and the corresponding makeup SI entering through the lower plenum, there are no assumed flow paths in or out of the mixing volume. All boric acid entering the mixing volume remains in the mixing volume prior to initiation of hot leg recirculation. The water/boric acid solution is well mixed in the mixing volume region. The water/boric acid solution in the vessel is assumed to be at atmospheric conditions, at a temperature of 212°F. The mixing volume is calculated to be the volume of the collapsed voided mixture that would offset the downcomer static head above the active fuel region. The lower plenum volume and barrel baffle region volume are not included in the mixing volume.

The boric acid concentration limit is the experimentally determined boric acid saturation concentration with an additional four weight-percent margin factor. The calculation neglects any elevation of boiling temperature due to concentration of boric acid in the core or due to backpressure from containment.

The decay heat generation rate is based on the 1971 ANS Standard for a finite operating time. The decay heat generation includes a core power multiplier to address instrumentation uncertainty as identified by Section I.A of Appendix K.

The boron concentration of the make-up SI is a calculated sump mixed-mean boron concentration. The calculation of the sump mixed-mean boron concentration assumes maximum mass and maximum boron concentrations for significant boron sources and minimum mass and maximum boron concentration for significant dilution sources.

Once realigned to hot leg recirculation, boron precipitation is precluded and core cooling is assured by established minimum recirculation flow criteria for the hot legs, cold legs, or simultaneous hot and cold leg injection.

The IP3 long-term cooling analysis used the standard Westinghouse evaluation methodology described above. The analysis established a maximum required HLSO time of 23.4 hours post-LOCA to preclude boron precipitation (with 4 weight-percent boron margin to the solubility limit). Based on this HLSO time, the ECCS performance (cooling capability) at hot leg recirculation was also evaluated. All the minimum flow requirements were satisfied for a HLSO time of 14 hours. Since the core boil-off due to decay heat is greater at an earlier HLSO time, the minimum flow requirements are satisfied for a HLSO time > 14 hours. (Note that the prescribed Appendix K decay heat model is used for ECCS recirculation performance calculations.)

Based on the results of the evaluation discussed above, the IP3 emergency operating procedures require initiation of hot leg ECCS recirculation 14 hours following a large-break LOCA. The 14-hour switchover time requirement does not increase operator burden during LOCA mitigation and recovery and will provide an added measure of conservatism with respect to the long term cooling analysis.

Question 13:

Section 6.1.3 indicated that the post-LOCA containment sump temperature performance has been determined to be unaffected by the proposed power uprate. Provide a discussion to support the above conclusion.

Response 13:

The following parameters, related to the 1.4% power uprate, were used to evaluate the longer term, post-LOCA containment sump temperature at the time of switchover to recirculation:

Core Power with Uncertainty	
Current Analysis:	3085.5 MWt
1.4% Uprate Analysis:	3085.5 MWt
Vessel Inlet Temperature	
Current Analysis:	542.6°F
1.4% Uprate Analysis:	542.5°F
Vessel Outlet Temperature	
Current Analysis:	603.4°F
1.4% Uprate Analysis:	606.9°F
Steam Generator Pressure	
Current Analysis:	817 psia
1.4% Uprate Analysis:	762 psia

A comparison of the above parameters shows a slight increase in the RCS hot leg temperature of 3.5°F and a reduction in the steam generator secondary side pressure of 55 psi for the 1.4% uprate. This steam generator pressure reduction will result in reduction of secondary side energy that more than offsets the effect of the slight RCS hot leg temperature increase. Therefore, the calculated total energy available for release to the containment and the sump (according to the new 1.4% power uprate analysis) has been reduced relative to the current analysis.

Therefore, the 1.4% power uprate did not affect the current maximum sump temperature used for evaluating the component cooling water (CCW) system during post-LOCA recirculation.

Question 14:

For LOCA and non-LOCA transients and accidents that already assume 2% uncertainty in the current safety analysis, discuss the effects of the change of initial plant conditions for the power uprate on the results of these analyses.

Response 14:

The current UFSAR Chapter 14 safety analyses that bound the 1.4% power uprate already assume a 2% uncertainty on power. These events are delineated in Table 8-1 of Attachment 3. As discussed in Section 8 of Attachment 3, these transients did not require explicit re-analyses for the 1.4% uprate because the power level assumed in the current analyses (current core power level of 3025 MWt plus 2% uncertainty) is equivalent to the 1.4% uprated power of 3067.4 MWt plus 0.6% uncertainty. The other NSSS design parameters that changed are the vessel inlet (cold-leg) and outlet (hot-leg) temperatures and the steam pressure. However, the related changes to the hot-leg and cold-leg temperatures were an increase and decrease by about 0.4°F, respectively, and the steam pressure decreased by less than 8 psi. These condition changes were evaluated and determined to have an insignificant effect on the results of the safety analyses. The current safety analysis basis was maintained for design parameters such as RCS thermal design flow, reactor vessel Tavg, and steam generator tube plugging. Therefore, the 1.4% power uprate has no effect on the results of the current IP3 safety analyses that already assume a 2% uncertainty on power.

Question 15:

Section 8.3.4.2 indicated that for loss of flow and reactor coolant pump shaft seizure events were evaluated with respect to departure from nucleate boiling ratio (DNBR). The analysis showed that the DNBR design basis remains satisfied. Provide additional details of these evaluation/analysis including the calculated minimum DNBR for these events.

Response 15:

The loss of reactor coolant flow analyses yield statepoints for core heat flux, core mass flow rate, pressurizer pressure, and core inlet temperature. The core heat flux and mass flow statepoints are presented as fraction of nominal (FON) values. A sensitivity study performed by Westinghouse showed that for a 1.4% power uprate, there is a negligible effect on loss of reactor coolant flow statepoint values, including the core heat flux and core mass flow FON values. Only the actual nominal core heat flux would increase as a result of the 1.4% uprate. Therefore, the limiting loss of reactor coolant flow events were reanalyzed using the same statepoints, but with an increased nominal core heat flux.

The minimum DNBR values that were calculated for the complete loss of flow and reactor coolant pump (RCP) shaft seizure analyses, based on the WRB-1 DNB correlation, were 1.852 and 1.736, respectively. Compared to the minimum DNBR values calculated for the same statepoints with a current nominal core heat flux (based on a core power of 3025 MWt), the minimum DNBR values became slightly worse. However, the decrease in the minimum DNBR values was accommodated by existing margin to the DNBR safety analysis limit (SAL), and the results are still maintained above the DNBR SAL.

Question 16:

Section 8.3.6.5 of the report indicates that the excessive load increase event was evaluated to demonstrate that the DNBR design limit is met. Provide details of this evaluation.

Response 16:

According to Westinghouse methodology, evaluations were performed for cases at Beginning-of-Life (BOL) and End-of-Life (EOL) conditions with and without automatic rod control. These evaluations were performed by applying conservative bounding deviations in plant parameters to initial conditions for core power, average coolant temperature, and RCS pressure in order to generate limiting statepoints for each of the cases examined. The bounding statepoints were then compared to the revised core thermal safety limits for the 1.4% power uprate to determine if the statepoints violated the safety limit conditions. From this comparison, it was determined that the core thermal limits were not violated. Therefore, the results of this evaluation concluded that the minimum DNBR safety analysis limit was not violated for the 1.4% power uprate for any of the cases examined.

Question 17:

Provide a justification for the proposed changes in Table 3.3.2.1, Note C (from 110% full steam flow to 120% full steam flow).

Response 17:

The original IP3 Accident Analysis for Main Steamline Break - Core Response was performed with an input of 110% Steam Flow at 100% power. Current licensing basis analyses performed by Westinghouse are based on a break Steam Flow of 134% at 100% nominal power (re-evaluated for Appendix K Uprate conditions with a nominal full load Steam Flow ~1.4% higher).

With the implementation of the 1.4% power uprate and the higher associated full-load steam flow, and in order to provide adequate margin to the steam flow reactor trip setpoint, Entergy is reallocating a portion of the available margin in the safety analysis with a change in the associated Allowable Value ("AV" value in Tech Specs). This has no effect on the current licensing basis safety analysis and is therefore acceptable.

Question 18:

Provide the results of an evaluation of the impacts of the 1.4 percent power uprate on the ability of IP3 to cope with a Station Blackout event.

Response 18:

A Station Blackout (SBO) is defined as the complete loss of alternating current electric power to the essential and nonessential switchgear buses. (Loss of offsite electric power system, concurrent with a turbine trip and the unavailability of the onsite emergency AC power system.) A Station Blackout does not involve the loss of available AC power to buses fed by station batteries through inverters. The event is considered to be terminated upon the restoration of power to the essential switchgear buses from any source, including the alternate AC source that has been qualified as an acceptable coping mechanism. Station batteries have been sized to carry their expected shutdown loads following a plant trip and a loss of all AC power for a period of 2 hours. The IP3 coping period is 8 hours prior to Alternate AC (AAC) power capability.

IP3 has three Emergency Diesel Generators, two of which have sufficient capacity to supply the engineered safety features equipment in the event of a design basis accident concurrent with loss of offsite power. Additionally, the Appendix "R" diesel generator provides an additional permanently installed alternative AC power supply as part of an enhanced alternate shutdown capability.

The methodology and assumptions associated with the SBO analysis with regard to equipment operability are unchanged with the 1.4% power uprate. There is no change in the ability of the turbine-driven auxiliary feedwater pump, supplied with steam from the steam generators, to support reactor heat removal due to the 1.4% power uprate.

Systems associated with SBO not affected by the 1.4% power uprate include:

- Auxiliary Feedwater and Condensate Storage
- Safety Related Battery Capacity
- Compressed Air System
- Ventilation
- Containment Isolation
- Reactor Coolant Inventory

There are no expected changes to any of these systems due to the 1.4% power uprate. Therefore, there is no effect on the SBO program due to the uprate.

Question 19:

In Section 7.2.1 the licensee stated: "The calculated fluences used in this 1.4% power uprate evaluation comply with NRC Regulatory Guide [RG] 1.190". Explain how and why the calculated fluence satisfies the guidance in RG 1.190.

Response 19:

The following provides a description of the calculational methodology used to determine the neutron exposure of the reactor pressure vessel for the 1.4% mini-uprate program at IP3. The results of these plant specific transport calculations were compared with dosimetry results obtained from all in-vessel surveillance capsules withdrawn to date in order to demonstrate that the IP3 plant-specific analysis meets the 20% uncertainty criterion specified in Regulatory Guide 1.190. However, the calculation-to-measurement comparisons were not used in any way to modify the calculational results.

In performing the fast neutron exposure evaluations for the IP3 reactor, plant specific discrete ordinates transport calculations were carried out using the three-dimensional flux synthesis technique described in Section 1.3.4 of Regulatory Guide 1.190. This synthesis approach is summarized in the following equation:

$$\phi(r,\theta,z) = \phi(r,\theta) * \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

The transport calculations were carried out using the DORT two-dimensional discrete ordinates code Version 3.1 and the BUGLE-96 cross–section library. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature.

The r,θ model of the IP3 reactor geometry was based on octant symmetric geometry spanning 0° to 45°. In addition to the core, reactor internals, pressure vessel and primary biological shield, the model also included explicit representations of the surveillance capsules, the pressure vessel cladding, and the insulation located external to the pressure vessel.

From a neutronic standpoint, the inclusion of the surveillance capsules and associated support structure in the analytical model is significant. Since the presence of the capsules and structure has a marked impact on the magnitude of the neutron flux as well as on the relative neutron and gamma ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are properly accounted for in the analysis.

In developing the r,θ analytical model of the reactor, nominal design dimensions were employed for the various structural components. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The r,z model of the IP3 extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately one foot below the active fuel to approximately one foot above the active fuel. As in the case of the r,θ model, nominal design dimensions and full power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model.

The one-dimensional radial model used in the synthesis procedure consisted of the same radial geometry characteristic of the r,z model at the axial elevation of the core mid-plane. Thus, radial synthesis factors could be determined on a mesh-wise basis throughout the entire geometry.

The core power distributions used in the plant specific transport analysis for the IP3 reactor were taken from the appropriate fuel cycle design reports for each operating fuel cycle. The data extracted from the design reports included fuel assembly specific initial enrichments, beginning of cycle burnups and end of cycle burnups. Appropriate axial distributions were also extracted from the respective design reports.

In constructing the core source distributions from the fuel assembly specific enrichment and burnup data, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes; and from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

Table One, 'Fluence Calculation Methods,' provides a summary of how the IP3 analysis complies with the regulatory positions specified in Regulatory Guide 1.190.

Question 20:

In Section 7.7.5, "Regulatory Guide 1.121 Analysis," the licensee summarized the results of an analysis it performed to define the structural limits for various regions of the SG tube. The section refers to RG 1.121 which provides guidance on calculating the allowable tube repair limit (i.e., utilizing the structural limit, a growth allowance and eddy current measurement uncertainty allowance). However, the licensee did not conclude whether the revised structural limits affect the allowable tube repair limit currently documented in the TSs. Identify the appropriate tube repair limit, and discuss the basis for reaching this conclusion. If the tube repair limit currently documented in the TSs needs to be modified, submit an appropriate TS change.

Response 20:

The analysis discussed in Section 7.7.5 of Attachment 3 is a very conservative evaluation for the following reasons:

1. ASME Code minimum material properties are used, and;
2. A bounding assumption of uniform thinning is assumed.

The tensile properties of the IP3 tubes are known. The tube-weighted average yield strength is 49.5 ksi (RT) compared to the code minimum value of 40 ksi (100° F) used in the analysis. Similarly, the actual weighted average ultimate strength is 105.8 ksi compared to the code minimum value of 80 ksi used in the analysis. (Tube-weighted means that the specific value for each tube in each steam generator is used as a data point.) Therefore, the calculated structural limit is approximately 23% conservative.

For AVB wear, the assumption of uniform thinning is extremely conservative. Tests have shown that for AVB wear of approximately 0.4" in length, burst does not occur at wear depths greater than 90% throughwall in Alloy 600TT tubing. For the condition tested, analysis similar to the Reg Guide 1.121 Analysis discussed in Section 7.5.5 predicted a structural limit of about 75% throughwall using the same assumptions (code minimum allowables, uniform thinning). Therefore testing has confirmed that the analysis is very conservative.

For AVB wear, using background noise typical of the operating Model F (10-15 EFPY operation), the detectability of wear was shown to be about 8% TW for Alloy 600 AVBs about 0.3" wide. At this depth, all analysts detected 100% of indications. For steam generators with wider AVBs made of 405SS, it is expected that the detectable level of wear will be less than 8% throughwall.

There are no operational data on AVB wear growth for the second generation replacement steam generators that utilize the advanced AVB design incorporated in the IP3 steam generators, because no AVB wear has been observed. Typically, in the first generation steam generators and in the Model F steam generators a few tubes wear rapidly at the onset of operation. After two to four cycles of operation, high growth rates are no longer observed, and the growth rate continues to decline with addition operating time. The absence of any observed AVB wear in the IP3 steam generators indicates that the advanced design has successfully eliminated the potential for significant AVB wear, and has rendered the growth rate negligible.

Shallow wear at the FDB has been observed only in steam generators that have experienced pressure pulse cleaning; the wear is attributed to tube impact against the FDB during the cleaning operation. Re-inspection of the shallow wear in these steam generators has established that the wear is not progressing, i.e., has a zero growth rate. Detection of FDB and TSP wear is excellent, that is, 100% of indications greater than about 8-10% depth are detected. FDB wear has not been observed at IP3, and is not expected due to SG operation.

Wear at TSPs not related to foreign objects has been rarely observed. Insufficient data are available to determine the growth rate. It is estimated that the growth rate is less than that for typical AVB wear since the AVB wear mechanism (fluid-elastic excitation of the tubes) is more energetic than the turbulence induced tube motion at the TSPs. (Note that pre-heater turbulence induced wear is a non-applicable special case, since the pre-heater tubes were directly in the feedwater crossflow field). Therefore, a conservative estimate for TSP wear growth is about 4-6% throughwall per 24-month operating cycle.

Straight leg wear has been observed only as the result of a foreign object or due to maintenance tooling contacting tubes. Detection of freespan wear is excellent; generally better than AVB wear since foreign object wear tends to exhibit sharper discontinuities. Wear due to tool contact has no growth rate because the tools are not present during operation. Wear growth due to foreign object impingement cannot be predicted without prior history of the object. If a prior history of a foreign object is known and the object cannot be removed, the tube is plugged and appropriate steps are taken to protect against propagation (i.e., plugging the surrounding tubes). In some instances where a growth rate can be calculated based on prior inspection history, a tube may be retained in service if the end of cycle conservatively predicted wear does not exceed the structural limit.

It is concluded that the current Tech Spec repair limit of 40% throughwall is adequate for the following reasons:

1. The predicted structural limit is conservative by >20% due to the use of Code minimum properties in the Reg Guide 1.121 analysis.
2. Tests have demonstrated that the structural limit for AVB wear is significantly higher than the predicted limit.
3. AVB wear has not been observed in steam generators with the advanced AVB design.
4. The 100% detection threshold for all forms of wear is very low (≤ 8% TW)
5. Observed growth rates for operationally induced wear are small.
6. Growth rates for maintenance service related wear are zero.

Foreign object wear, when detected, is conservatively analyzed and proper disposition made relative to the applicable structural limit.

Question 21:

In Section 7.7.3, "Tube Wear," the licensee described the potential effects of the 1.4% power uprate on SG tube wear. Discuss the potential effects of the 1.4% power uprate on other modes of SG tube degradation (e.g., axial and/or circumferential cracking, pitting, etc.). Factor into your discussion the impact from the increase in primary system hot-leg coolant temperature.

Response 21:

The IP3 steam generators are Model 44F steam generators that utilize Alloy 690TT tubes. Comparative studies of the performance of Alloy 690TT, Alloy 600TT and Alloy 600MA have shown Alloy 690TT to have superior resistance to corrosion compared to Alloy 600 TT or MA. Plants utilizing Alloy 600TT have operated without evidence of PWSCC for over 15 EFPY. ODSCC was reported in a plant with Alloy 600TT tubing in May 2002 after about 9.7 EFPY operation. The cause for the ODSCC in this plant has not yet been confirmed. All other domestic steam generators with Alloy 600TT tubing have operated without evidence of cracking; the maximum operating time among these plants is over 15 EFPYs. These steam generators are operated at significantly higher temperature (~618°F). The estimated corrosion performance of Alloy 690TT is a factor of about 2-3 better than that of Alloy 600TT; therefore, at the normalization temperature (~618°F) a minimum operating period of at least 19 years would be expected prior to initiation of corrosion degradation. At the IP3 operating temperature of <601°F at the 1.4% power uprate condition, a significantly longer operating period prior to corrosion degradation is predicted.

Alloy 690TT was shown to be essentially immune to PWSCC in the qualification testing for the use of Alloy 690TT (EPRI NP-6997-SD, "Alloy 690 for Steam Generator Tubing Applications," October 1990). Consequently, the 0.5°F temperature increase due to the 1.4% power uprate will have negligible effect on the incidence of PWSCC in the IP-3 steam generators. Similarly, based on the Arrhenius equation to compare corrosion initiation times from the pre- and post-uprate temperatures, the change in the initiation time is negligible.

Pitting is principally a function of the chemical environment in which the tubes operate; thus the very small increase in temperature at the 1.4% power uprate conditions will insignificantly affect the potential for pitting of the tubes.

Question 22:

Discuss the impact the power uprate will have on the required frequency of SG tube inspections.

Response 22:

The inspection frequency is principally dictated by the current Technical Specification requirements and by the requirements of NEI 97-06. An increase in inspection frequency in terms of the number of steam generators inspected and the sample size of tubes inspected depends on the progression (if any) of degradation. None of the potential degradation mechanisms are significantly affected by the 1.4% power uprate conditions; therefore the required frequency of inspection is also not affected significantly by the planned power uprate.

Question 23:

Section 7.7.2, "Structural Integrity Evaluation," describes the impact of the power uprate on SG tube structural integrity. The licensee stated that the 1.4% power uprate structural evaluation was performed for 3082 MWt NSSS power and 0% SG tube plugging. Table 2.1 describes three different sets of design parameters, one of which relates to the 0% plugging assumption. Summarize the results of the structural evaluation performed for the other two sets of design parameters, or explain the basis for performing the SG tube structural integrity evaluation for only one set of design parameters.

Response 23:

The steam generator structural integrity evaluation performed to address the 1.4%, as described in Section 7.7.2 of the Application Report, addressed the current actual plant condition of 0% steam generator tube plugging. As explained in Section 2.0, the data in Sets 2 and 3 are conservatively based on tube plugging level assumptions that are being maintained solely for the purpose of supporting conservative IP3 UFSAR Chapter 14 safety analyses.

Question 24:

In reference to Section 7.7.2 of Attachment 3 to the amendment request, provide a summary evaluation of the flow-induced vibration for the SG U-bend tubes based on the increase in feedwater flow and the increase in pressure difference between the primary system pressure (unchanged at 2250 psi) and the decreased steam pressure for the proposed power uprate.

Response 24:

Analysis of the effects of the 1.4% power uprate condition on the S/G tubes has addressed both the straight leg and U-bend portions of the S/G. Limiting wear calculations have been performed where it was determined that the maximum projected rate of tube wear would increase no more than 4% over the current levels of wear. Since significant tube wear is not currently occurring at IP3, a 4% increase will not be significant. Should significant tube wear initiate sometime in the future, the rate of tube wear would be sufficiently small such that any tubes requiring repair would be detected during the normal eddy current inspection program.

With respect to the effect of increased primary-to-secondary side pressure difference, it should be noted that there is no direct correlation of flow-induced vibration with primary to secondary side pressure differences. The steam generator tubes respond primarily to the conditions associated with the secondary side since the forcing functions associated with the secondary side of the S/G dominate over any other effects. Any effects of primary-to-secondary side pressure difference are inherently considered in the analysis in that the secondary side conditions are defined by the total steam generator conditions such as steam pressure, flow rates, re-circulation, etc., and includes the primary to secondary side pressure difference.

Note that in some model steam generators particular consideration is given to the potential for high cycle fatigue of U-bend tubes. This phenomenon has been observed in tubes with carbon steel support plates where denting or a fixed tube support condition has been observed in the upper most plate. However, since the IP3 steam generator tube support plates are manufactured from stainless steel, there is no potential for the necessary boundary conditions (i.e. denting) to occur at the uppermost support plate. Hence, high cycle fatigue of U-bend tubes will not be an issue at IP3.

Question 25:

In reference to Section 12.2.5, "Safety-Related Motor Operated Valves," the licensee evaluated the effect of the proposed power uprate on the motor-operated valves program at IP3 for Generic Letter (GL) 89-10 and the GL 95-07 regarding pressure locking and thermal binding or safety-related power-operated gate valves. Provide a summary evaluation of the effects of the proposed power uprate on the response of GL 96-06 regarding overpressurization of isolated piping segment.

Response 25:

In September 1996, the NRC issued Generic Letter 96-06 to address (among other concerns) the concerns that thermally induced over-pressurization of isolated water-filled piping sections in containment could:

- Jeopardize the ability of accident mitigating systems to perform their safety functions
- Lead to a breach of containment integrity via bypass leakage

The accident pressure/temperature values are still bounded by design. Current accident analyses (LOCA) is bounding for the 1.4% power uprate condition, and there are no changes to CFC cooling water flow requirements or CFC accident conditions associated with the uprate. There is no increase in the possibility of over-pressurization of isolated segments of safety related piping inside containment, including penetrations, as a result of the 1.4% power uprate. Therefore, the GL-96-06 program is not affected by the uprate.

Question 26:

Provide a summary evaluation of the effect of the proposed power uprate on the design basis analysis for high energy line breaks, intermediate energy line breaks, jet impingement and pipe whip restraints.

<u>Response 26:</u>

High-energy lines as defined by IP3 are pipes in which, during normal plant operation, the fluid temperature exceeds 200°F or the pressure exceeds 275 psig. The piping outside of containment at IP3 that qualify for high-energy line break (HELB) analysis are listed in Attachment 2. (The current LOCA and MSLB analyses inside containment bound 1.4% power uprate conditions.) Information in Attachment 2 is organized by area of the plant. This breakdown of piping systems by area is consistent with the method of analysis for HELB used at IP3.

Lines in which 1.4% power uprate conditions are less severe then current operation (i.e., temperature and pressure decreases) are considered acceptable for the uprate. For systems that are considered "Unaffected" by the 1.4% power uprate, the current HELB analysis is also considered "Unaffected". The only location not bounded by these criteria is the Auxiliary Boiler Feed Pump Building El. 32ft.-6in. where feedwater temperature will increase slightly. This increase is minor (~1°F) and will not result in a change to analysis conclusions. As shown in Attachment 2, all current HELB analyses are bounding for uprate conditions and therefore acceptable for uprate. The HELB program analyses are not affected by the uprate.

The proposed 1.4% power uprate does not create any new high energy lines, and the existing HELB analysis for breaks outside containment will bound the uprate conditions. Therefore, no additional HELB evaluations on systems outside containment are required in support of the proposed power uprate. Since changes to operating temperatures, pressures and flow rates for applicable high and moderate energy piping systems are sufficiently small ($\leq$3%), the existing design basis for pipe break, jet impingement and pipe whip considerations remains valid for the 1.4% power uprate. Refer to Table Two for a summary listing of the high energy lines evaluated.

TABLE ONE - FLUENCE CALCULATION METHODS

(For response to question 19)

Summary of Compliance with the Regulatory Positions Specified in Regulatory Guide 1.190

Regulatory Position	NRC Staff Position	Westinghouse Approach
1.3	**Fluence Determination.** Absolute fluence calculations, rather than extrapolated fluence measurements, must be used for the fluence determination.	Fluence evaluations are based on absolute calculations using the discrete ordinates transport method and the 2D/3D-synthesis technique described in Section 1.3.4 of Regulatory Guide 1.190.
1.1.1	**Modeling Data.** The calculation modeling (geometry, materials, etc.) should be based on documented and verified plant specific data.	The calculation model is based on nominal dimensions obtained from plant specific design drawings. System operating temperatures and, hence coolant densities are treated on a fuel cycle specific basis.
1.1.2	**Nuclear Data.** The latest version of the Evaluated Nuclear Data File (ENDF/B) should be used for determining cross-sections. Cross-section sets based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable. When the recommended cross-section data change, the effect of these changes on licensee specific methodology must be evaluated and the fluence estimates updated when the effects are significant.	Discrete ordinates calculations make use of the BUGLE-96 ENDF/B-VI based cross-section library. The BUGLE-96 library is a 67 group (47 neutron, 20 gamma ray) data set produced specifically for light water reactor (LWR) applications by the Oak Ridge National Laboratory (ORNL). ENDF/B-VI represents the latest version of the Evaluated Nuclear Data File (ENDF/B).
	Cross-Section Angular Representation. In discrete ordinates transport calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed.	A P_5 scattering approximation was used in all of the cycle specific fluence calculations. The P_5 approximation exceeds the requirements of Regulatory Guide 1.190.

Regulatory Position	NRC Staff Position	Westinghouse Approach
1.1.2	**Cross-Section Group Collapsing.** The adequacy of the collapsed job library must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.	The BUGLE-96 library is itself a collapsed job library. The testing required by this Staff Position was performed by ORNL prior to the general release of the BUGLE-96 library and is included in the documentation package accompanying the library (RSIC Data Library Collection DLC-185).
1.2	**Neutron Source.** The core neutron source should account for local fuel isotopics and, where appropriate, the effects of moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.	Core power distribution data were applied on a cycle specific basis. In applying this data in the discrete ordinates analysis, the fission spectra, neutrons released per fission, and energy release per fission accounted for the presence of both uranium and plutonium fissioning isotopes.
1.2	**End-of-Life Predictions.** Predictions of the vessel end-of-life fluence should be made with a best estimate or conservative generic power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	Projections for future operation were based on fuel cycle data accounting for a power uprate.

Regulatory Position	NRC Staff Position	Westinghouse Approach
1.3.1	**Spatial Representation.** Discrete ordinates neutron transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of ~2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40 intervals per octant.	The spatial mesh used by Westinghouse in the discrete ordinates calculations exceeds the minimum requirements for radial and azimuthal mesh specified in Staff Position 1.3.1. An S_{16} angular quadrature set was used in all of the cycle specific fluence calculations. An S_{16} quadrature set exceeds the requirements of Regulatory Guide 1.190.
1.3.1	**Multiple Transport Calculations.** If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions must be demonstrated.	Not applicable. The "bootstrap" technique was not used in performing the discrete ordinates calculations.
1.3.2	**Point Estimates.** If the dimensions of the tally region or the definition of the average-flux region introduce a bias in the tally edit, the Monte Carlo prediction should be adjusted to eliminate the calculational bias. The average-flux region surrounding the point location should not include material boundaries or be located near reflecting, periodic, or white boundaries.	Not applicable. This staff position pertains to Monte Carlo calculations. The discrete ordinates approach was used in the Brunswick analysis.
1.3.2	**Statistical Tests.** The Monte Carlo estimated mean and relative error should be tested and satisfy all statistical criteria.	Not applicable. This staff position pertains to Monte Carlo calculations. The discrete ordinates approach was used in the Brunswick analysis.
1.3.2	**Variance Reduction.** All variance reduction methods should be qualified by comparison with calculations performed without variance reduction.	Not applicable. This staff position pertains to Monte Carlo calculations. The discrete ordinates approach was used in the Brunswick analysis.

Regulatory Position	NRC Staff Position	Westinghouse Approach
1.3.3	**Capsule Modeling.** The capsule fluence is extremely sensitive to the geometrical representation of the capsule geometry and internal water region, and the adequacy of the capsule representation and mesh must be demonstrated.	The capsule and associated structure were modeled explicitly in the discrete ordinates calculations. Adequacy of the modeling was tested by comparison with dosimetry from withdrawn surveillance capsules. Results were shown to be within the 20% uncertainty requirement specified in Regulatory Guide 1.190.
1.3.3	**Spectral Effects on RT_{NDT}.** In order to account for neutron spectrum dependence of RT_{NDT}, when it is extrapolated from the inside surface of the pressure vessel to the T/4 and 3T/4 vessel locations using the E > 1-MeV fluence, a spectral lead factor must be applied to the fluence for the calculation of ΔRT_{NDT}.	The current calculations for IP3 address this issue by including a calculation of iron atom displacement (dpa) distributions through the vessel wall. The dpa damage function accounts for the spectral shift toward lower energies with deeper penetration into the carbon steel vessel wall. In point of fact, these calculations are rarely used. In Regulatory Guide 1.99 Revision 2, an attenuation function intended to simulate the dpa distribution is provided. This built in function is usually used to determine values of RT_{NDT} at the T/4 and 3T/4 vessel locations.
1.3.5	**Cavity Calculations.** In discrete ordinates transport calculations, the adequacy of the S_8 angular quadrature used in cavity calculations must be demonstrated.	Not applicable. There were no cavity dosimetry comparisons associated with the IP3 fluence evaluations.

Regulatory Position	NRC Staff Position	Westinghouse Approach
1.4.1, 1.4.2, 1.4.3.	**Methods Qualification.** The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytical uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytical uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.	Measurement to calculation comparisons for the PCA Pressure Vessel Simulator described in NUREG/CR-6454 and the H. B. Robinson Unit 2 Pressure Vessel Benchmark described in NUREG/CR-6453 were completed. Likewise, an analytical sensitivity study assessing the uncertainty associated with important geometric, material density, and neutron source input parameters was performed. In combination, the results of these studies establish an overall calculational uncertainty of less than 20%.
1.4.3	**Fluence Calculational Uncertainty.** The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be $\leq$20% for RT_{PTS} and RT_{NDT} determination. In these applications, if the benchmark comparisons indicate differences greater than 20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.	The qualification of the methodology demonstrates an uncertainty of less than 20% (1 sigma). None of the benchmarking comparisons exceeded the 20% criteria specified in Staff Positions 1 and 1.4.3. Therefore, no adjustments to the calculated results have been required.

TABLE TWO
High Energy Lines – 1.4% Power Uprate Evaluation Summary
(For response to question 26)

No.	Location / High Energy Line	Acceptable for Uprate (Yes/No)	Justification – Comments
	Aux. Boiler Feed Pump Build., El. 18'-6"		
1	MS to Aux. Feed Pump Turbine	Yes	1.4% power uprate operating conditions are less severe then current (MS pressure and temperature decrease) and bounding conditions (i.e. hot shut down) are not affected.
	Aux. Feed Pump Build., El. 32'-6"		
2	Main Feedwater from Main Feed Pumps via Regulator Valves to Elevation 57'-6" and Containment Penetrations	Yes	Current analysis is based on full-open FRV; this flow will not change due to uprate. Predicted 1.4% power uprate feedwater temperature will increase slightly, this minor change (~1 F) will not impact analysis.
	Shield Wall Area, El. 43'-0"		
3	MS to Auxiliary Feed Pump Turbine	Yes	Refer to line 1 comment.
4	MS from Containment Penetrations to Turbine Building via Pipe Bridge	Yes	Refer to line 1 comment.
	Pipe Bridge/Tower		
5	MS and FW from Turbine Build. To MS and FW Isolation Area	Yes	Refer to line 1 comment.
	Turbine Building		
6	MS to HP Turbine from Containment	Yes	Refer to line 1 comment.
7	Boiler Feed lines from Boiler Feed Pumps to Containment	Yes	Refer to line 1 comment.
	Primary Auxiliary Building		
8	Letdown line	Yes	No change in letdown flow conditions.

No.	Location / High Energy Line	Acceptable for Uprate (Yes/No)	Justification – Comments
9	Charging line	Yes	No change in charging requirements or operation.
10	RCP Seal Water Injection lines	Yes	No change is seal water requirements.
11	SGBD lines	Yes	No Change in SGBD System.
12	SGBD and RCS Sample lines	Yes	No change in sample flow or requirements.
13	Auxiliary Steam lines	Yes	No change in auxiliary steam operation.
14	N2 Supply lines	Yes	No change in N2 system associated with the 1.4% power uprate.
	Pipe Penetration Area and Heat Exchanger Room		
15	SGBD lines from Containment Penetration to SGBD Flash Tank in PAB Pipe Penetration Area	Yes	No Change in SGBD System.
16	SGBD lines from Blowdown Headers in Pipe Penetration Area to Blowdown Recovery Heat Exchanger at the 15 Foot Elevation of PAB	Yes	No Change in SGBD System.
	Control Building		
17	Auxiliary Steam lines to Control Room Heating Coils	Yes	No change in auxiliary steam or heating requirements associated with the 1.4% power uprate.
18	Instrument Air line	Yes	Instrument Air not affected by the 1.4% power uprate.
	EDG Building		
19	Starting Air line from each DG Starting Air Tank	Yes	DG starting system / air requirements not affected by the 1.4% power uprate.
	Fuel Storage Building		
20	Auxiliary Steam lines to Building Space Heaters	Yes	No change in auxiliary steam or heating requirements associated with the 1.4% power uprate.

ATTACHMENT II TO IPN-02-073

SUPPLEMENTAL INFORMATION REGARDING

PROPOSED LICENSE AMENDMENT REQUEST FOR

1.4% MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

1. Proposed change (markup page) to Facility Operating License page 3.

2. Revision 1 of page 7-36 for Attachment III of Reference 2.

3. Revision 1 of page 9-3 for Attachment III of Reference 2.

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

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; and 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) <u>Maximum Power Level</u>

ENO is authorized to operate the facility at steady state reactor core Power levels not in excess of ~~3025~~ 3067.4 megawatts thermal (100% of rated power) — Amdt.17 8-18-78

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

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 212 are hereby incorporated in the License. ENO shall operate the facility in accordance with the Technical Specifications.

(3) <u>(DELETED)</u>

(4) <u>(DELETED)</u>

D. (DELETED) — Amdt. 46 2-16-83

E. (DELETED) — Amdt. 37 5-14-81

F. This amended license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter of May 2, 1975, to Consolidated Edison Company of New York, Inc., granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

G. ENO shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Indian Point 3 Nuclear Power Plant Physical Security Plan," with revisions submitted through December 14, 1987; "Indian Point 3 Nuclear Power Plant — Amdt. 81 6-6-88

TABLE 7-10 IP3 1.4% POWER UPRATE EVALUATION SUMMARY OF TUBE STRUCTURAL LIMITS		
Location / Wear Scar Length	**Parameter**	**Value**
Straight Leg	t_{min} (inch)	0.022
	Structural Limit (%)	56.0
Anti-Vibration Bar/0.5" (Tube Rows 16 – 45)	t_{min} (inch)	0.019
	Structural Limit (%)	62.0
Anti-Vibration Bar/0.75" (Tube Rows 1 – 15)	t_{min} (inch)	0.022
	Structural Limit (%)	56.0
Flow Distribution Baffle/0.75"	t_{min} (inch)	0.018
	Structural Limit (%)	64.6
Tube Support Plate/1.125"	t_{min} (inch)	0.021
	Structural Limit (%)	59.0

Note: Structural Limit $= [(t_{nom} - t_{min}) / t_{nom}] \times 100\%$

$t_{nom} = 0.050$ in

9.2 POWER BLOCK EQUIPMENT

The power block equipment consists of the Main Generator, the Isophase Bus Duct, the MTs, the UAT and the SAT, the switchyard, and the lines to the switchyard. Grid stability is also addressed with the power block equipment.

9.2.1 Main Generator

The nameplate rating of the main generator is 1,125.6 MVa (based on 75 psig hydrogen pressure), 0.90 power factor, 22 kV, three-phase, 60 Hz at 1800 rpm. The generator is supported by a number of systems, including cooling, lubrication and sealing systems.

The current operating point of the main generator is 1,027.2 MWe, taken for the current station heat balance. This operating point is at a power factor of 91.26. At the 1.4% power uprate, the main generator is anticipated to operate at an output of 1,041.2 MWe, based on the baseline heat balance for 101.4% of the current core thermal power. The generator capability curves show this operation is possible at a power factor of 92.54%. This operating point will allow production of up to 427.6 MVAR lagging. This operating point also allows acceptance of 170 MVARs leading. The leading limit is a result of an excitation limiter setting shown on the generator capability curve.

The cooling systems consist of stator water cooling and generator hydrogen cooling equipment, each sized to support the generator at nameplate rating. The generator lubrication systems provide lubrication to the bearings, and the seal oil systems maintain the generator atmosphere isolated from the outside ambient. These systems are supported by low voltage AC and DC components. These systems are sized to support the nameplate rating of the machine. Operation of the machine within the capability curve will not require additional capacity.

The exciter has the capability to support machine operation within its nameplate rating and within the capability curve of the machine for the leading and lagging case of VAR production.

Main Generator Protection

The applied Main Generator protection schemes are intended to limit machine damage for internal fault conditions and to prevent machine damage during abnormal operating or external fault conditions. A review of One Line Diagrams and Protective Relay Settings confirms that the applied schemes are dependent upon machine ratings and design parameters, and the design of the connected system. They are not affected by machine operation at the 1.4% power uprate conditions. For example, overlapping differential schemes provide machine protection for both internal (generator differential and unit differential schemes) and external (unit differential scheme) phase fault conditions. The schemes are not affected by load changes within the rated operating range of the machine. Ground fault protection schemes provided by ground over-voltage relays are designed and set based upon the system grounding design, and are independent of main generator output. Loss of excitation and negative sequence protection schemes that are included among the remaining main generator protection schemes are similarly unaffected by unit operation at the 1.4% power uprate conditions because the machine

ENCLOSURES TO IPN-02-073

1. **Westinghouse proprietary information authorization letter CAW-02-1551, dated September 11, 2002.**

2. **WCAP-15824-P, Rev 1, Power Calorimetric Uncertainty for the 1.4 Percent Uprating of Indian Point Unit 3," May 2002 (Proprietary version).**

3. **WCAP-15904, Power Calorimetric Uncertainty for the 1.4 Percent Uprating of Indian Point Unit 3," September 2002 (Non-Proprietary).**

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

Westinghouse

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Attention: Mr. Samuel J. Collins

Our ref: CAW-02-1551

September 11, 2002

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-15824, "Power Calorimetric for the 1.4% Uprating of Indian Point Unit 3", Revision 1, May 2002, (Proprietary) and WCAP-15940, "Power Calorimetric for the 1.4% Uprating of Indian Point Unit 3", Revision 1, September 2002, (Non-Proprietary)

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-02-1551 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 Operations, Inc.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-02-1551 and should be addressed to the undersigned.

Very truly yours,

J W Fasnacht /for
H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: G. Shukla/NRR
T. Carter/NRC (5E7)

CAW-02-1551

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY·

Before me, the undersigned authority, personally appeared J. W. Fasnacht, 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.

MARGARET L. GONANO COMMONWEALTH OF PENNSYLVANIA NOTARY PUBLIC

JW Fasnacht

J. W. Fasnacht, Manager
Integrated Plant Engineering Services

Sworn to and subscribed
before me this 11th day
of September, 2002

Margaret L. Gonano
Notary Public

Notarial Seal
Margaret L. Gonano, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 3, 2006
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Integrated Plant Engineering Services, 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 the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10CFR 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 the Westinghouse Electric Company LLC 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 10CFR 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 brackets, WCAP-15284, "Power Calorimetric Uncertainty For The 1.4-Percent Uprating of Indian Point Unit 3 (Proprietary), dated May 2002 for Indian Point Unit 3", being transmitted by the Entergy Nuclear Operations, Inc. and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use by Westinghouse Electric Company LLC for Indian Point Unit 3 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification for use of a Leading Edge Flow Meter (LEFM) for improved accuracy.

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

(a) Provide improved power calorimetric uncertainty with the use of a Leading Edge Flow Meter on the Feedwater Header.

(b) Establish appropriate procedures for calculation of calorimetric uncertainty based on the use of a Leading Edge Flow Meter on the Feedwater Header.

(c) Assist the customer to obtain NRC approval.

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.

(b) Westinghouse can sell support and defense of the calculation of power calorimetric uncertainty with Leading Edge Flow Meter on Feedwater Header.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology that 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 licensing support documentation 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) contained within parentheses 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).

CAW-02-1551

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