Entergy

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Fred Dacimo
Site Vice President
Administration

August 12, 2004

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-04-100

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: **Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate (TAC MC1865)**

References:
1. NRC letter to Entergy Nuclear Operations, Inc; "Supplemental Request for Additional Information Regarding Stretch Power Uprate", dated June 30, 2004.
2. Entergy letter to NRC (NL-04-005); "Proposed Changes to Technical Specifications: Stretch Power Uprate Increase of Licensed Thermal Power (3.26%)", dated January 29, 2004.
3. Entergy letter to NRC (NL-04-086); "Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate", dated July 16, 2004.
4. Entergy letter to NRC (NL-04-095); "Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate", dated August 3, 2004.

Dear Sir:

This letter provides additional information, requested by the NRC in Reference 1, regarding the license amendment request submitted by Entergy Nuclear Operations, Inc (Entergy), in Reference 2. This response addresses LOCA transient questions 3, 4, and 5. Responses to the other questions were previously provided in References 3 and 4. This response also documents information provided to the staff during recent conference calls.

The requested additional information is provided in Attachment 1, except that two responses contain proprietary information. The proprietary and non-proprietary versions of those responses are provided in Attachments 2 and 3, respectively. The information provided in these attachments does not alter the conclusions of the no significant hazards evaluation that supports this license amendment request.

A001

The Westinghouse authorization letter, regarding proprietary information (CAW-04-1866, dated August 11, 2004), with the accompanying affidavit, Proprietary Information Notice, and Copyright Notice, is enclosed. As Attachment 2 contains information proprietary to Westinghouse Electric Company, it is supported 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 Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the 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 2.390 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-04-1866 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Kevin Kingsley at 914-734-6695.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 12 2004.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

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ATTACHMENT 1 TO NL-04-100

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING

PROPOSED LICENSE AMENDMENT REQUEST FOR

INDIAN POINT 2 STRETCH POWER UPRATE

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

Selected questions from NRC letter dated June 30, 2004:
(Note: Answers to other questions from the June 30, 2004 letter were provided in Entergy letters dated July 16, 2004 and August 3, 2004)

LOCA Transient Question 3:

See Attachments 2 and 3 for proprietary and non-proprietary responses, respectively.

LOCA Transient Question 4:

Provide the LBLOCA analysis results (tables and graphs, as appropriate) to the time that stable and sustained quench is established.

Response:

In order to demonstrate stable and sustained quench, the WCOBRA/TRAC calculation for the maximum local oxidation analysis was extended. Figure 1 shows the peak cladding temperatures for the five rods modeled in WCOBRA/TRAC. This figure indicates that quench occurs at approximately 275 seconds for the low power rod (rod 5), 400 seconds for the core average rods (rods 3 and 4), and 500 seconds for the hot rod (rod 1) and hot assembly average rod (rod 2). Once quench is predicted to occur, the rod temperatures remain slightly above the fluid saturation temperature for the remainder of the simulation. Figure 2 shows the collapsed liquid level in the four downcomer channels and shows steady behavior, with the level in each quadrant remaining near the bottom of the cold leg. By 600 seconds, bulk boiling in the downcomer has been terminated, and subcooling in the downcomer has been re-established. Figure 3 shows the collapsed liquid level in the four core channels and indicates a gradual increase in the core liquid inventory. This is consistent with the expected result based on the removal of the initial core stored energy and the gradual reduction in decay heat. Figure 4 shows the vessel liquid mass and indicates an increasing trend beginning at about 500 seconds. This indicates that the increase in inventory due to the pumped safety injection is more than offsetting the loss of inventory through the break. Based on these results, it is concluded that stable and sustained quench has been established for the Indian Point Unit 2 Large Break LOCA analysis.

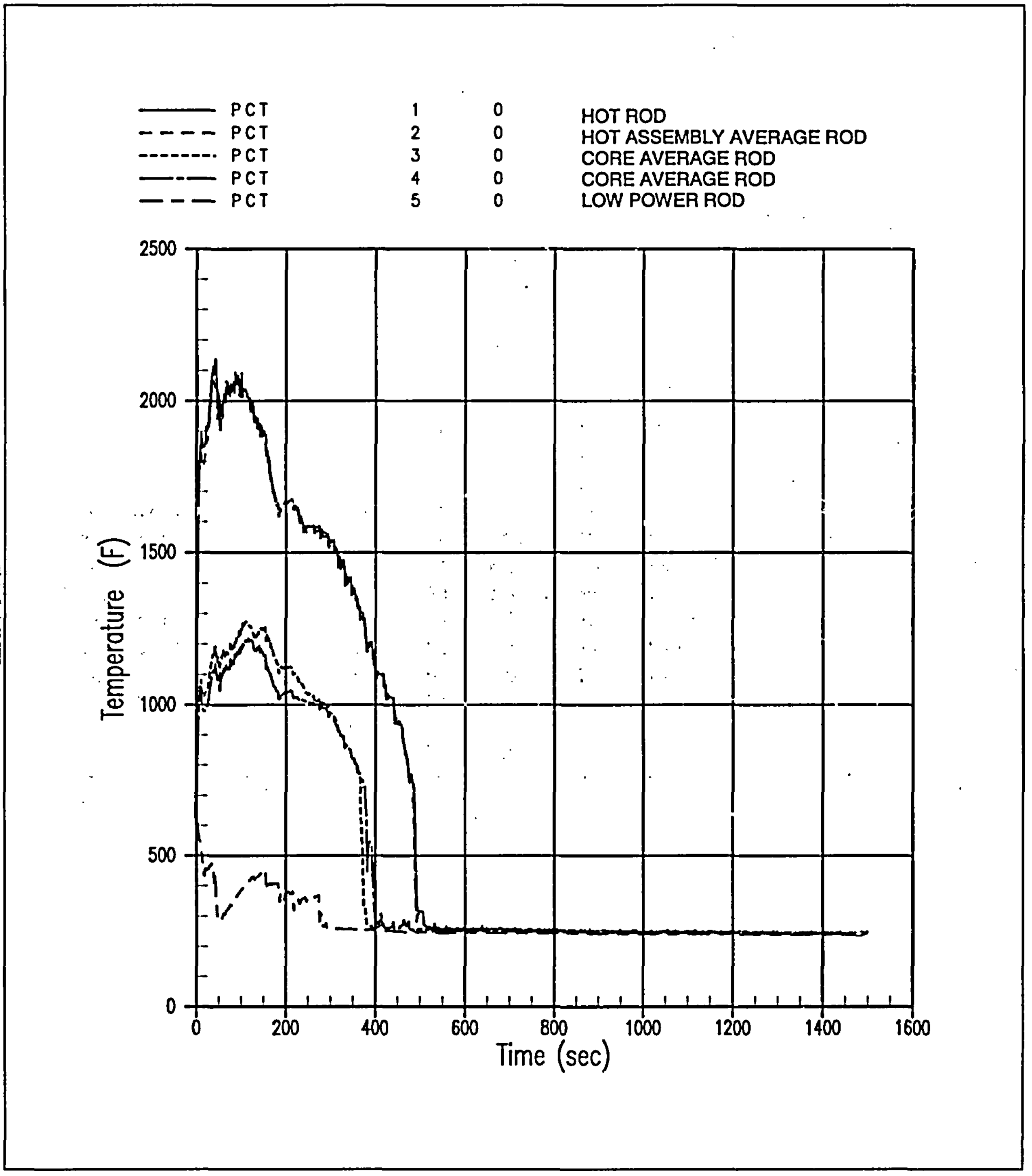

Figure 1 - Peak Cladding Temperatures

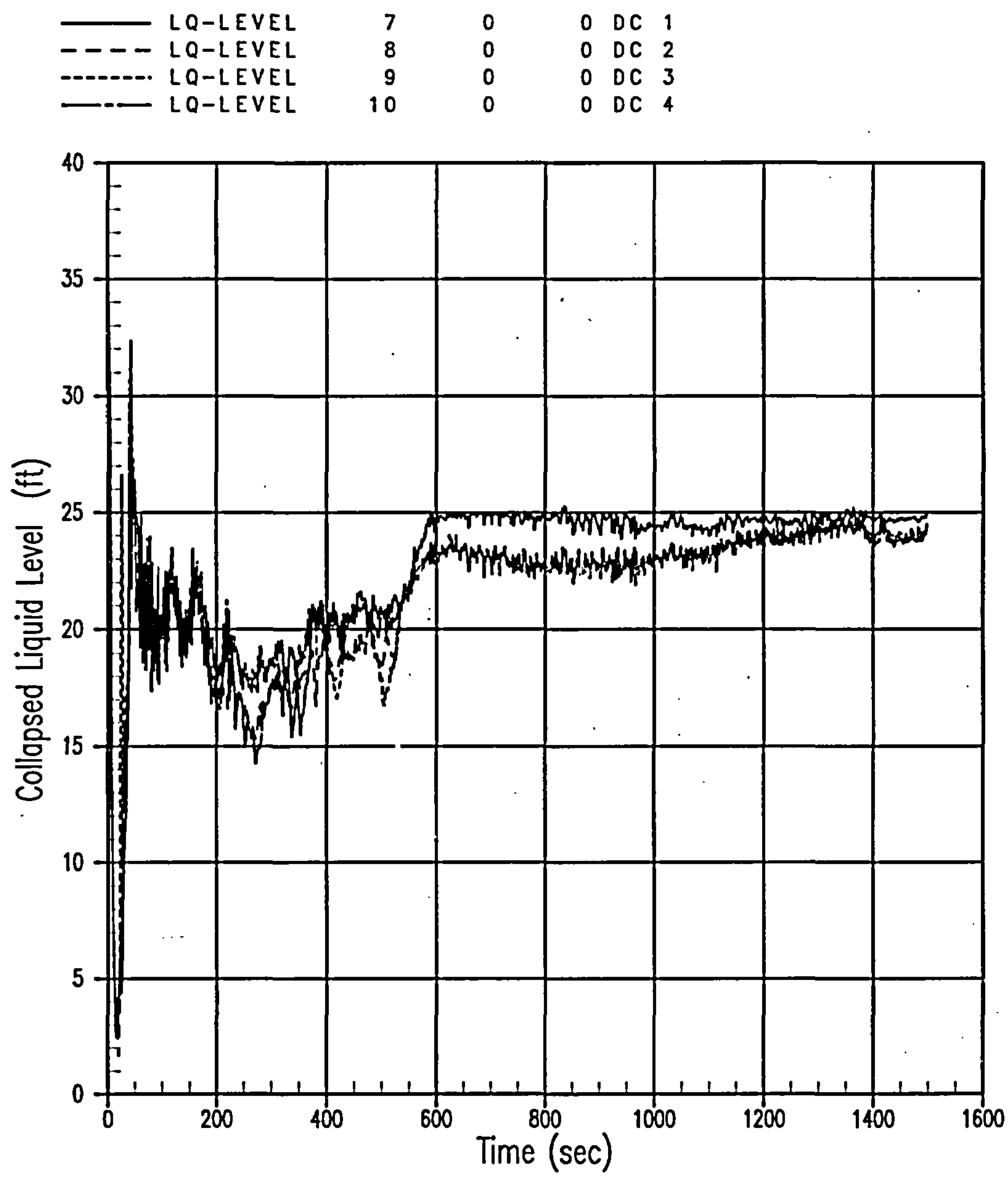

Figure 2 - Downcomer Collapsed Liquid Levels

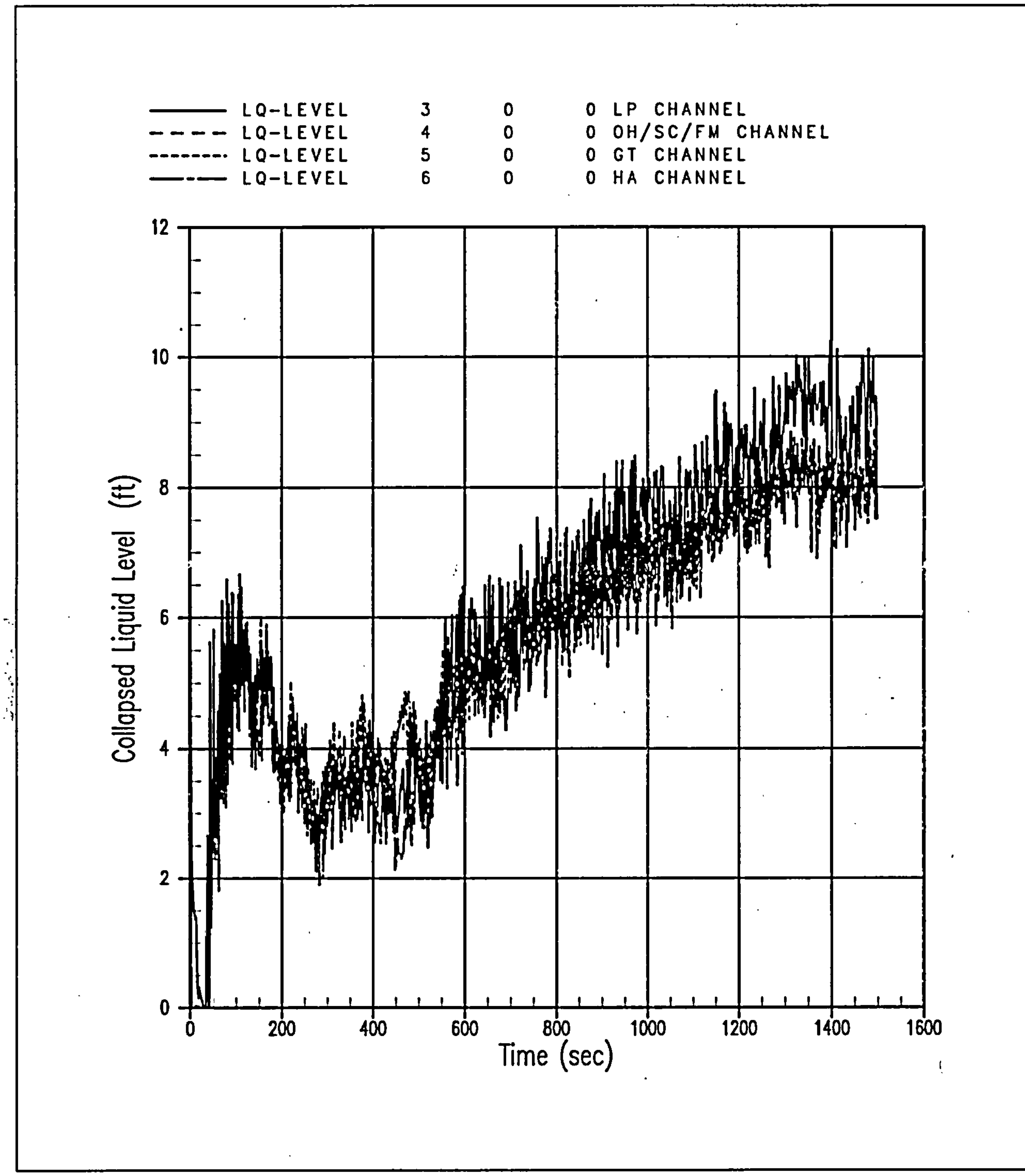

Figure 3 - Core Collapsed Liquid Levels

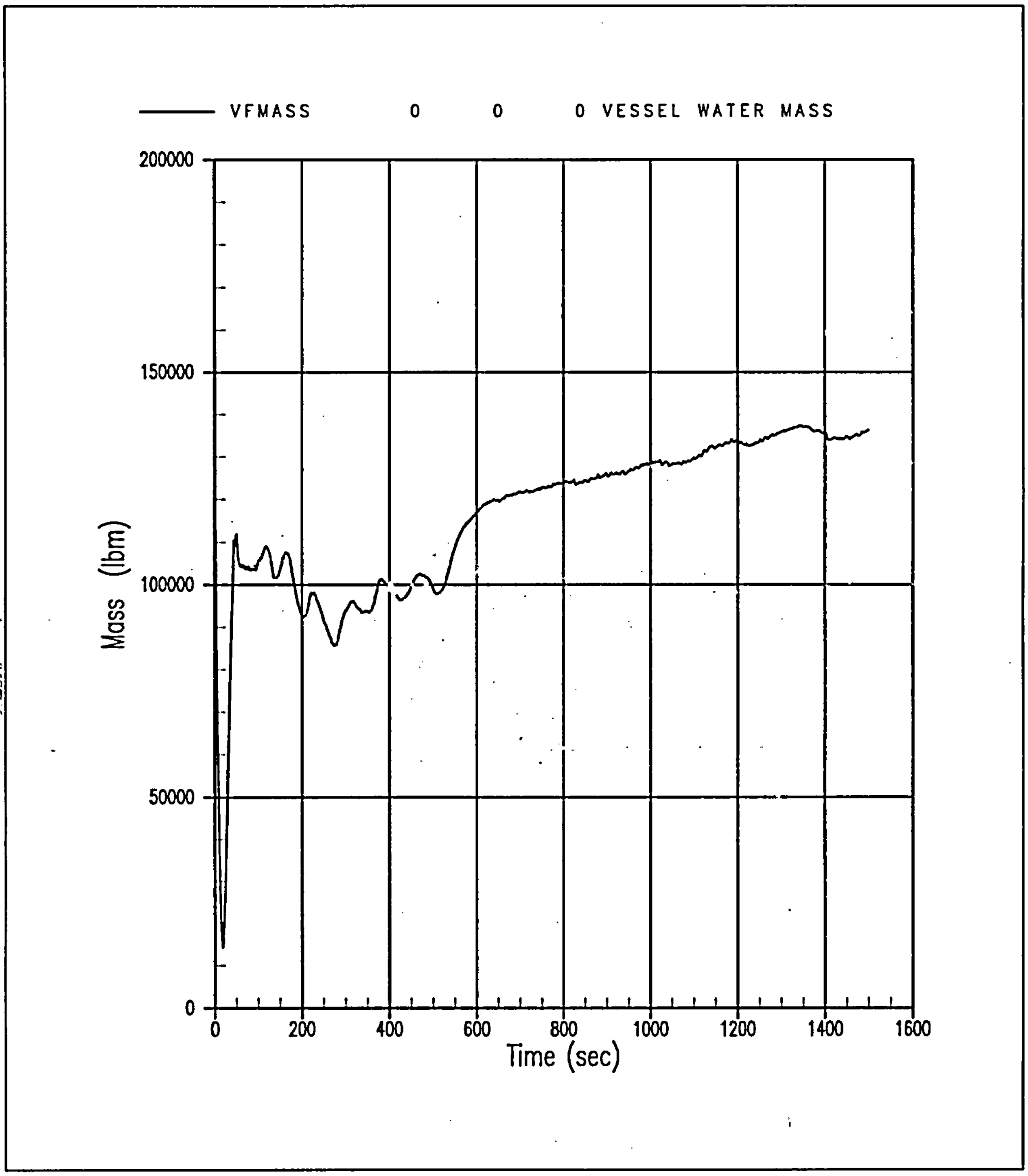

Figure 4 - Vessel Liquid Mass

LOCA Transient Question 5:

Tables 6.2-3 and 6.2.5 in the Application Report provide LBLOCA and SBLOCA analysis results for the IP2 SPU. Provide all results (peak clad temperature, maximum local oxidation and total hydrogen generation) for both LBLOCA and SBLOCA. For maximum local oxidation include consideration of both pre-existing and post-LOCA oxidation, cladding outside and post-rupture inside oxidation. Also include the results for fuel resident from previous cycles.

LOC-5 Response:

The results (peak clad temperature, maximum local oxidation and total hydrogen generation) for the IP2 LBLOCA and SBLOCA design basis analyses are provided in Table LOC-5-1 below. Additional information regarding the bases for the maximum local oxidation, including consideration of both pre-existing and post-LOCA oxidation, cladding outside and post-rupture inside oxidation is discussed below.

Large Break LOCA Pre-existing and Post-LOCA Oxidation:

The transient maximum local oxidation calculated for the Indian Point Unit 2 (IP2) large break LOCA analysis of record is 13.2 percent. Consistent with the NRC-approved methodology, this value was calculated using a LOCA transient whose nominal peak cladding temperature exceeds the 95th percentile value for both the first and second reflood peaks. This LOCA transient was manufactured by increasing the nuclear peaking factors by 5% for the most limiting power shape included in the PCT uncertainty analysis. The transient maximum local oxidation was predicted to occur at the burst elevation, such that the metal-water reaction occurred on both the inner and outer cladding surfaces. From the WCOBRA/TRAC transient, the nominal peak cladding temperature for this calculation was 2146°F. From the HOTSPOT oxidation calculation using the corresponding WCOBRA/TRAC transient boundary conditions, the nominal peak cladding temperature at the burst elevation was 2218°F, without the burst option turned on. The average HOTSPOT peak cladding temperature from the 1000 Monte Carlo calculations with the burst option turned on was 2312°F. These extreme conditions, in excess of the PCT acceptance criteria, were selected in order to get a very conservative oxidation assessment.

The maximum local oxidation was calculated for fresh fuel, at the beginning of the cycle. This *represents the maximum amount of transient oxidation that could occur at any time in life.* As burnup increases, the transient oxidation decreases for the following reasons:

1) The cladding creeps down towards the fuel pellets, due to the system pressure exceeding the rod internal pressure. This will reduce the average initial stored energy at the hot spot by several hundred degrees relatively early in the first cycle of operation. Accounting only for this change, which occurs early in the first cycle, reduces the transient oxidation significantly.

2) Later in life, the clad creep-down benefit still remains in effect. In addition, with increasing irradiation, the power production from the fuel will naturally decrease as a result of depletion of the fissionable isotopes. Reductions in achievable peaking factors in the burned fuel relative to the fresh fuel are realized before the middle of the second cycle of operation. The achievable linear heat rates decrease steadily from this point until the fuel is discharged, at which point the transient oxidation will be negligible.

The pre-transient oxidation increases with burnup, from zero at beginning of life (BOL) to a maximum value at the discharge of the fuel (end of life, or EOL). The design limit 95% upper bound value for each of the fuel designs that will be included in the SPU cores is < 15%. The actual upper bound values predicted for each of the fuel designs are well below this value, for each of the representative uprate reloads considered through the equilibrium cycle (including fuel currently resident and the upgrade fuel to be inserted for the SPU and subsequent cycles).

Based on the above discussion, the transient oxidation decreases from a very conservative maximum of 13.2% at BOL to a negligible value at EOL, while the pre-transient oxidation increases from zero at BOL to a very conservative maximum at EOL of <15%. Additional WCOBRA/TRAC and HOTSPOT calculations were performed at intermediate burnups, accounting for burnup effects on fuel performance data (primarily initial stored energy and rod internal pressure). These calculations support the conclusion that the sum of the transient and pre-transient oxidation remains below 15% at all times in life. This conclusion is applicable to each of the fuel designs that will be included in the SPU cores, and confirms IP2 conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

Small Break LOCA Pre-existing and Post-LOCA Oxidation:

As part of the IP2 SPU program, a new SBLOCA analysis was performed. The break spectrum that was analyzed yielded a maximum peak clad temperature of 1028°F for a 3 inch equivalent break diameter. The break spectrum results are summarized in Tables 6.2.4 and 6.2.5 of Reference 1.. Because of the low clad temperatures, fuel rod burst was not predicted to occur, and the maximum transient oxidation was only 0.02%. Because this is so low, the SBLOCA transient needs no further justification since the local oxidation limit will not be challenged even when the end of life initial (steady state) oxide layer is considered. This confirms IP2 conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

References

1. WCAP-16157-P, "Indian Point Nuclear Generating Unit No. 2, Stretch Power Uprate NSSS and BOP Licensing Report, January 2004

Table LOC-5-1 IP2 DESIGN BASIS ANALYSIS LOCA RESULTS		
	LBLOCA	**SBLOCA**
Peak Clad Temperature	2137°F (PCT$^{95\%}$)	1028°F
Maximum Local Oxidation	Pre-transient = 0% Transient = <13.2%	Pre-transient = 0% Transient = .02%
Total Hydrogen Generation	0.94%	<< 1%

Questions regarding pressure vessel materials per conference call of July 22, 2004, as follows:

Regarding prior response to PVM RAI 3a provided in NL-04-073:

1. When was the last time the Reactor Vessel nozzles were volumetrically examined?

Response:

The Reactor Vessel Nozzle welds (B-D) consist of the following:

8 - Nozzle-to-Vessel Welds (B3.90) RPVN1 thru RPVN8

8 - Nozzle Inside Radius Sections (B3.100) RPVN1(IR) thru RPVN8(IR)

The last volumetric inspection performed on these nozzle welds was March 1995.

Table 5.9-3 of WCAP-16157 (transmitted in the initial license amendment request, NL-04-005 dated January 29, 2004) includes the Fracture Integrity Evaluation Summary for the outlet nozzle-to-shell region.

2. Was the inspection technique qualified to ASME Section 8?

Response:

No. Ultrasonic examiners were qualified and certified to Level II or Level III in accordance with ASNT SNT-TC-IA 1984 Edition, as supplemented by the requirements of ASME Section XI, Subarticle IWA-2300 and Appendix VII.

3. What was the largest flaw?

Response:

There were no recordable indications.

4. Provide comparison of the inspection technique / qualification from 1995 to today, for Reactor vessel nozzles.

Response:

The applicability of inspections conducted in 1995 to the EPRI Performance Demonstration Initiative (PDI) is addressed in the following ASME technical paper:

"Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections" from Proceedings of ASME 2001 Pressure Vessels and Piping Conference (Atlanta, Ga).

The section entitled: "Nozzle Inner Radius Examination Capability from the inside surface", states that inspection capabilities were improved significantly in response to RG 1.150, in the time frame of 1983 through the late 80s, and that those techniques were used directly without change to meet the PDI requirements brought forth recently.

Therefore an exam of the reactor vessel conducted in 1995 would be using essentially the same techniques that would be used today, and called 'PDI qualified".

Regarding prior response to PVM RAI 4a provided in NL-04-073:

1. What is the temperature difference for the water entering the pressurizer shell for inadvertent aux. spray? Compare to the analysis of record. If the temperature difference between the SPU and current analysis increases, then provide the reanalysis.

Response:

The temperature difference (delta-T) considered in the original analysis for inadvertent auxiliary spray is 621ºF, which is the difference between pressurizer steam temperature of 653ºF (saturation temperature at 2250 psia) and a conservatively low spray temperature of 32ºF. These temperatures did not change due to IP2 Stretch Power Uprate. Moreover, the delta-T considered for inadvertent auxiliary spray (621ºF) is significantly larger than those encountered during other transients, and hence envelopes all transient changes for the IP2 Stretch Power Uprate. Therefore, no re-analysis is required.

Regarding prior response to PVM RAI 4d provided in NL-04-073:

1. When was the last time the pressurizer nozzles were volumetrically examined?

Response:

The pressurizer nozzles were last volumetrically examined during fabrication. The tests included radiography and magnetic particle test techniques used after fabrication. Procedures for performing the examinations are consistent with those established in the ASME Code Section III and are reviewed by qualified Westinghouse engineers.

The Indian Point 2 inservice inspection program provides for visual inspection of the pressurizer relief and safety inner radius nozzles PZRN-2, PZRN-3, PZRN-4, & PZRN-5, since they cannot be volumetrically examined. The pressurizer was designed and fabricated to Codes in effect during the late 1960's. The Codes at that time did not provide for full access for inservice inspection nor did they require a surface finish in the nozzle area suitable for volumetric examination. Performing a volumetric examination is impractical due to geometry and size of the nozzle and dose estimates of 3-5 rem/hr. The relief request in effect for the current ISI interval, allowing VT-1 visual examination is Relief Request No. 9 Rev.1 approved in NRC letter dated June 3, 1997.

Nozzles PZRN-2, PZRN-4, & PZRN-5 were visually examined on October 1997. Nozzle PZRN-3 visual examination is schedule to be performed by April 2006.

2. Was the technique equivalent to VIP-108?

Response:

No. The volumetric examination performed of the inner radius regions following fabrication is addressed in Section 2 the following reference:

"Technical Basis for Elimination of Nozzle Inner Radius Inspections (For Vessels Other Than The Reactor Vessel)" EDRE-SMT-99-110.

The nozzle is inspected 100 percent by a volumetric exam before the cladding is applied. Then, before the cladding is welded, a surface exam is applied, and then another surface exam is applied after the cladding is welded. These exams are generally followed by a baseline exam for Section XI.

3. What was the size of the largest flaw?

Response:

There were no recordable indications.

Questions regarding steam generator manway closure per conference call of July 29, 2004:

Regarding prior response to SG structural integrity RAI 1 provided in NL-04-073:

See Attachments 2 and 3 for proprietary and non-proprietary responses, respectively.

Regarding prior response to SG structural integrity RAI 3 provided in NL-04-073:

The licensee was asked to provide a table of primary stress calculation results for the shop welded plugs. The licensee responded by providing the table, which included a column for loading condition (design, operating, and test), a column of the calculated maximum stress intensities, and a column with the ASME Code limits. In the table for the "test" loading condition, the PL+Pb+Q maximum stress intensity term was calculated to be 41,962 psi. However, the ASME Code limit for this term was given as 34,950 psi. Apparently, the PL+Pb+Q maximum stress intensity calculated value is not within the ASME Code allowable value.

Explain why the calculation for PL+Pb+Q maximum stress intensity for the "test" loading condition is satisfactory, even though it exceeds the ASME Code requirement. Include a technical basis in your explanation.

Response:

When providing the Code limits for the various load conditions a factor of 0.5 was applied to each stress condition allowable. This 0.5 is a quality factor for a full penetration weld. In essence it reduces the permitted allowable stress. This reduction is directly applicable to Pm, and to Pm + Pb values. However, for PL+Pb+Q test condition, the value identified is actually a range between the primary hydrostatic test and the secondary hydrostatic test. The 0.5 quality factor is not required to be applied to the allowable value of 3 Sm for the range.

Therefore, the Table previously submitted should have read as follows:

1) Change PL+Pb+Q = 41,962 psi to [PL+Pb+Q]RANGE = 41,962 psi

2) ASME Code Limit for the PL+Pb+Q should be 3.0Sm = 3.0 x 23,300 = 69,900 psi

Revisions to Reference 6 in COLR reference list on Technical Specification page 5.6-3, per NRC comment. (see next page)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

9. Technical Specification 3.3.1, Reactor Protection System Instrumentation;

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

11. Technical Specification 3.9.1, Boron Concentration.

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

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

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

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

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

Insert A

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

Insert B

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

Insert C

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

Inserts for Technical Specification page 5.6-3:

Insert A: (for Ref 5)

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

Insert B: (for Ref 6)

WCAP-12945-P, "Code Qualification Document for Best Estimate LOCA Analysis", June 1993, as supplemented up to June 13, 1996 as follows:

- Westinghouse letter (N. J. Liparulo) to USNRC, "Re-Analysis Work Plans Using Final Best Estimate Methodology", NSD-NRC-96-4746, June 13, 1996, and
- USNRC letter (J.Harold) to Consolidated Edison Company (S. Quinn), "Issuance of Amendment [188] for Indian Point Nuclear Generating Unit No. 2 (TAC No. M96370)", March 31, 1997.

Insert C: (for Refs 7 through 11)

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

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

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

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

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

ATTACHMENT 3 TO NL-04-100

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING

PROPOSED LICENSE AMENDMENT REQUEST FOR

INDIAN POINT 2 STRETCH POWER UPRATE

Non-proprietary responses to LOCA Transient RAI 3 and SG structural integrity RAI 1

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

LOCA Transient Question 3:

The LOCA submittals did not address slot breaks at the top and side of the pipe. Justify why these breaks are not considered for the IP2 LBLOCA response

Response:

Break location, type and size are specifically considered for the IP2 LBLOCA transient simulations (Reference 1). This document concluded that the cold leg guillotine break is limiting for IP2. The uncertainties related to break location, type and size were included in the model uncertainties for the IP2 BELBLOCA PCT.

For Small Break LOCA (SBLOCA) events, the effects of break location have been generically evaluated as part of the application of the NOTRUMP Evaluation Model (Reference 2). This document concluded that a break in the Reactor Coolant System (RCS) cold leg was limiting. Additionally, the effects of break orientation were considered during the evaluation of Safety Injection in the Broken Loop and application of the COSI Condensation Model (Reference 3). This work concluded that a break oriented at the bottom of the RCS cold leg piping was limiting with respect to Peak Cladding Temperature (PCT).

While these references specifically address the short-term response to the LOCA break spectrum, the long-term effects associated with potential Reactor Coolant Pump (RCP) loop seal re-plugging core uncovery is addressed in the following.

A review of the analysis conditions associated with potential core uncovery due to loop seal re-plugging has previously been performed in Reference 4. Reference 4 documents the Westinghouse position with regards to the potential for Inadequate Core Cooling (ICC) scenarios following Large and Intermediate Break LOCAs as a result of loop seal re-plugging. Reference 4 concludes the following:

- *The reactor coolant system response following a LOCA is a dynamic process and the expected response in the long term is similar to the response that occurs in the short term. This short term response has been analyzed extensively through computer analysis and tests and is well documented.*
- *Consideration of the physical mechanisms for liquid plugging of the pump suction leg U-bend piping following large and intermediate break LOCA at realistic decay heat levels precludes quasi steady-state inadequate core cooling conditions.*
- *It is important to emphasize that the operator guidance provided in the Emergency Response Guidelines includes actions to be taken in the event of an indication of a challenge to adequate core cooling following a LOCA.*

A review of the key contributors associated with long-term loop seal plugging core uncovery scenarios, under LOCA conditions (specifically, extended-term SBLOCA conditions), was performed as part of Reference 5 including a review of pertinent experimental data.

a, c

From References 4 and 5 it can be concluded that post-LOCA core uncovery scenarios as a result of loop seal re-plugging do not constitute a significant concern to Indian Point Unit 2 plant safety.

References

1. WCAP-13837, Revision 1, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for Indian Point Unit 2 Nuclear Plant", S. B. Nguyen, M. Y. Young, December 1996.
2. WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code", S. D. Rupprecht, et al., 1986.

3. WCAP-10054-P Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", C. M. Thompson, et al., July 1997.
4. OG-87-37, "Westinghouse Owners Group (WOG) Post LOCA Long Term Cooling, Letter from Roger Newton (WOG) to Thomas Murley (NRC)", August 26, 1987.
5. NSD-NRC-97-5092, "Core Uncovery Due to Loop Seal Re-Plugging During Post-LOCA Recovery," Letter from N. J. Liparulo (W) to NRC, March, 1997.

<u>SG-1</u>: Attachment I, page 67 of 76 of the response to RAI Question 1:

The licensee's application stated in Table 5.6-2 that, under SPU operating conditions, the fatigue usage factor for the secondary manway bolts was [ ] ^{a,c} (the design limit is 1.0), and that the bolts would have to be replaced after 34 years of operation, or sooner. Table 5.6-2 also stated that the fatigue usage factor for secondary manway studs was [ ] ^{a,c}. The licensee was asked to provide a basis for the 34-year target for secondary manway bolt replacement, and to describe how the bolt replacement target would be incorporated into the plant maintenance procedures. The licensee responded that the IP-2 replacement steam generators use secondary manway studs, not secondary manway bolts. The staff notes that Table 5.6-2 from the application is confusing since it contains entries for both secondary manway bolts and secondary manway studs, but the licensee states in the RAI response that only secondary manway studs are applicable to IP-2's replacement steam generators. Based on the licensee's response, the staff concludes that Table 5.6-2 contains information that is not relevant to the IP-2 SPU application (i.e., the information regarding secondary manway bolts).

A. Confirm that you are using secondary manway studs, not secondary manway bolts, in your replacement steam generators, and that the fatigue usage factor for these components is [ ] ^{a,c}.

B. Provide a cross-sectional drawing, which shows how the secondary manway studs are positioned in the licensee's replacement steam generators. The drawing should include the important dimensions and design features of a secondary manway stud and its location relative to the adjoining components.

C. Indicate on the drawing the areas of highest stress on a secondary manway stud.

<u>SG-1 Response:</u>

The replacement steam generators were designed to accommodate either bolts or studs for secondary manway closure. Thus, values are provided for both in the LAR. The replacement steam generators were installed with studs for secondary manway closure. The fatigue usage value of [ ] ^{a,c} as stated in LAR Table 5.6-2 is correct. The attached sketch provides the dimensions and design features of the manway studs as requested. The area of highest stress is also identified as requested.

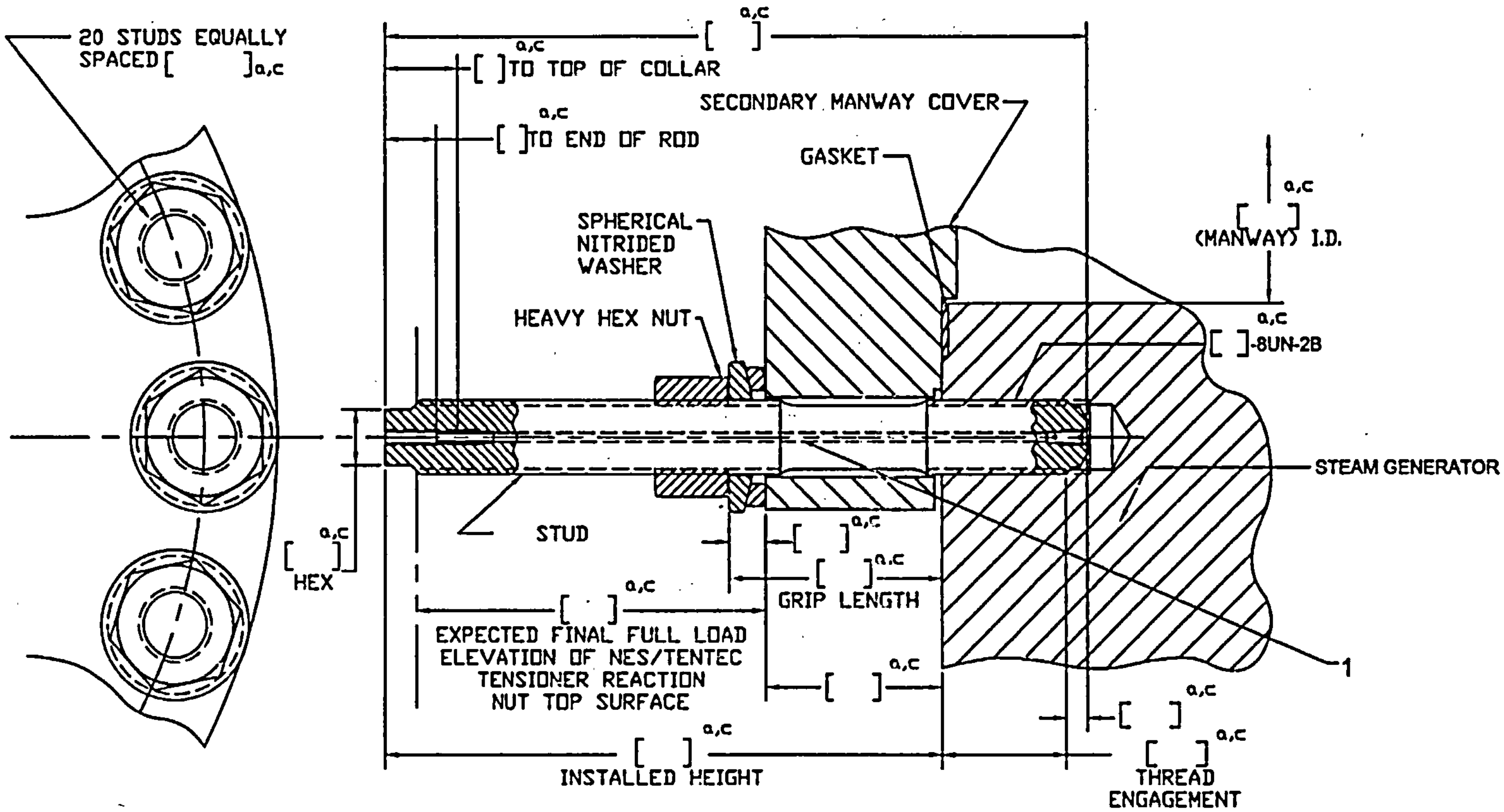

Figure SG-1-1 SG Manway Stud Configuration

[1] Maximum stress location corresponds to the threaded section of the stud.

Max stress (w/o uprate) = [ ] a,c ksi < 86.7 ksi (limit)
Max stress (w/uprate) = [ ] a,c ksi < 86.7 ksi (limit)

ENCLOSURE TO NL-04-100

Westinghouse authorization letter dated August 11, 2004 (CAW-04-1866), with the accompanying affidavit, Proprietary Information Notice, and Copyright Notice

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

Westinghouse

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Our ref: CAW-04-1866

August 11, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Transmittal PU2-W-04-030 (IPP-04-102), Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate Project, Westinghouse Responses to NRC LOCA-Related RAIs, August 11, 2004.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1866 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 2.390 of the Commission's regulations.

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

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1866, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: W. Macon
E. Peyton

bcc: R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)
S. Ira (WM F2D7) 1L, 1A
R. Laubham (ECE 419F) 1L, 1A
T. Timmons (ECE 406F) 1L, 1A
T. Gerlowski (ECE 413C) 1L, 1A
J. Stukus (ECE 419G) 1L, 1A
D. Morris (ENN) 1L, 1A
C. Jackson (ENN 1L, 1A
K. Kingsley (ENN) 1L, 1A
W. Wittich (ENN) 1L, 1A
J. Curry (ENN) 1L, 1A
J. Jawor (ENN) 1L, 1A

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

J. S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 11th day
of August, 2004

Sharon L. Fiori
Notary Public

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

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

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

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

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, 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 Attachment A to PU2-W-04-030, "Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate Westinghouse Responses to NRC LOCA-Related RAIs" (Proprietary) dated August 11, 2004, being transmitted by the Entergy Nuclear Northeast letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for the Indian Point Nuclear Generating Unit No. 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of Stretch Power Uprate License Amendment Request.

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

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

(b) Provide plant specific calculations.

(c) Provide licensing documentation support for customer submittals.

Further this information has substantial commercial value as follows:

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

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

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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