Entergy

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Fred Dacimo
Site Vice President
Administration

December 15, 2004

Re: Indian Point Unit 3
Docket No. 50-286
NL-04-155

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

Subject: **Supporting Information for License Amendment Request Regarding Indian Point 3 Stretch Power Uprate (TAC MC 3552)**

Reference:

1. Entergy Letter NL-04-069 to NRC; "Proposed Changes to Technical Specifications: Stretch Power Uprate (4.85%) and Adoption of TSTF-339", dated June 3, 2004.
2. Entergy Letter NL-04-145 to NRC; "Supporting Information for License Amendment Request Regarding Indian Point 3 Stretch Power Uprate (TAC MC 3552)," dated November 18, 2004

Dear Sir:

Entergy Nuclear Operations, Inc (Entergy) is submitting additional information to support NRC review of the stretch power uprate (SPU) license amendment request (Reference 1) for Indian Point 3 (IP3). This additional information, based on NRC staff questions regarding the uprate request for Indian Point 2, is being provided as discussed during a meeting with NRC on September 14, 2004. This letter supplements the Reference 2 letter and covers the balance of questions regarding uprate request for Indian Point 2.

Attachment 1 is a summary listing of those RAIs that are being addressed in this letter. The responses to the RAIs are provided in Attachment 2, except for responses that contain proprietary information. The proprietary responses and the corresponding non-proprietary version of those responses are provided in Attachments 3 and 4, respectively.

As Attachment 3 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

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with 10 CFR 2.390 of the Commission's regulations. Westinghouse authorization letter dated December 9, 2004 (CAW-04-1927), with the accompanying affidavit, Proprietary Information Notice, and Copyright Notice is provided in Enclosure A.

Correspondence with respect to the copyright on proprietary aspects of the items listed above or the supporting affidavit should reference CAW-04-1927 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.

The additional supporting information provided in this letter does not alter the conclusions of the no significant hazards evaluation that supports the subject license amendment request. There are no new commitments being made 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 12/15/2004.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

Attachment 1: Summary Listing of RAI Responses Regarding Stretch Power Uprate License Amendment Request for Indian Point 3
Attachment 2: Additional Information for IP3 SPU License Amendment Request, Based on NRC RAIs Issued for IP2 SPU
Attachment 3: Additional Information for IP3 SPU License Amendment Request, Based on NRC RAIs Issued for IP2 SPU (with Proprietary Information)
Attachment 4: Additional Information for IP3 SPU License Amendment Request, Based on NRC RAIs Issued for IP2 SPU (non-Proprietary version of Attachment 3)
Enclosure A: Westinghouse Withholding Request for Attachment 3 Proprietary Information
cc: next page

cc: Mr. Patrick D. Milano, Senior Project Manager
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ATTACHMENT 1 TO NL-04-155

SUMMARY LISTING OF RAI RESPONSES

REGARDING STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

FOR INDIAN POINT 3

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

No.	RAI	Review Area	From Letter	IP3 Response
1	NL-04-073-FP-1	Fire Protection	NL-04-073	Att 2 - Non-Proprietary
2	NL-04-073-FP-2	Fire Protection	NL-04-073	Att 2 - Non-Proprietary
3	NL-04-073-FP-3a	Fire Protection	NL-04-073	Att 2 - Non-Proprietary
3	NL-04-073-FP-3b	Fire Protection	NL-04-073	Att 2 - Non-Proprietary
3	NL-04-073-FP-3c	Fire Protection	NL-04-073	Att 2 - Non-Proprietary
4	NL-04-073-EL-1	Electrical	NL-04-073	Att 2 - Non-Proprietary
5	NL-04-073-IC-1	Instrumentation and Controls	NL-04-073	See letter NL-04-145
6	NL-04-073-IC-2	Instrumentation and Controls	NL-04-073	See letter NL-04-145
7	NL-04-073-IC-3	Instrumentation and Controls	NL-04-073	See letter NL-04-145
8	NL-04-073-IC-4	Instrumentation and Controls	NL-04-073	See letter NL-04-145
9	NL-04-073-IC-5	Instrumentation and Controls	NL-04-073	See letter NL-04-145
10	NL-04-073-IC-6	Instrumentation and Controls	NL-04-073	Not Applicable
11	NL-04-073-IC-7	Instrumentation and Controls	NL-04-073	See letter NL-04-145
12	NL-04-073-PVM-1a	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
12	NL-04-073-PVM-1b	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
13	NL-04-073-PVM-2	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
14	NL-04-073-PVM-3a	Pressure Vessel Materials	NL-04-073	Not Applicable
14	NL-04-073-PVM-3b	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
14	NL-04-073-PVM-3c	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
15	NL-04-073-PVM-4a	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
15	NL-04-073-PVM-4b	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
15	NL-04-073-PVM-4c	Pressure Vessel Materials	NL-04-073	Att 2 - Non-Proprietary
15	NL-04-073-PVM-4d	Pressure Vessel Materials	NL-04-073	See letter NL-04-145
16	NL-04-073-RSA-1	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
17	NL-04-073-RSA-2a	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
17	NL-04-073-RSA-2b	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
18	NL-04-073-RSA-3	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
19	NL-04-073-RSA-4	Reactor Systems and Analyses	NL-04-073	Not Applicable
20	NL-04-073-RSA-5	Reactor Systems and Analyses	NL-04-073	Not Applicable
21	NL-04-073-RSA-6	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
22	NL-04-073-RSA-7	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
23	NL-04-073-RSA-8	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145

No.	RAI	Review Area	From Letter	IP3 Response
24	NL-04-073-RSA-9a	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
24	NL-04-073-RSA-9b	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
25	NL-04-073-RSA-10a	Reactor Systems and Analyses	NL-04-073	Not Applicable
25	NL-04-073-RSA-10b	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
25	NL-04-073-RSA-10c	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
25	NL-04-073-RSA-10d	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
25	NL-04-073-RSA-10e	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
26	NL-04-073-RSA-11	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
27	NL-04-073-RSA-12a	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
27	NL-04-073-RSA-12b	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
28	NL-04-073-RSA-13a	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
28	NL-04-073-RSA-13b	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
29	NL-04-073-RSA-14	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
30	NL-04-073-RSA-15	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
31	NL-04-073-RSA-16	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
32	NL-04-073-RSA-17a	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
32	NL-04-073-RSA-17b	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
32	NL-04-073-RSA-17c	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
33	NL-04-073-RSA-18	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
34	NL-04-073-RSA-19	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
35	NL-04-073-RSA-20	Reactor Systems and Analyses	NL-04-073	See letter NL-04-145
36	NL-04-073-ENV-1	Environmental Considerations	NL-04-073	Not Applicable
37	NL-04-073-ENV-2	Environmental Considerations	NL-04-073	Not Applicable
38	NL-04-073-ENV-3	Environmental Considerations	NL-04-073	Att 2 - Non-Proprietary
39	NL-04-073-FAC-1a	Flow Accelerated Corrosion Program	NL-04-073	See letter NL-04-145
39	NL-04-073-FAC-1b	Flow Accelerated Corrosion Program	NL-04-073	See letter NL-04-145
39	NL-04-073-FAC-1c	Flow Accelerated Corrosion Program	NL-04-073	See letter NL-04-145
39	NL-04-073-FAC-1d	Flow Accelerated Corrosion Program	NL-04-073	See letter NL-04-145
39	NL-04-073-FAC-1e	Flow Accelerated Corrosion Program	NL-04-073	See letter NL-04-145
40	NL-04-073-PCP-1a	Protective Coatings Program	NL-04-073	See letter NL-04-145
40	NL-04-073-PCP-1b	Protective Coatings Program	NL-04-073	See letter NL-04-145
40	NL-04-073-PCP-1c	Protective Coatings Program	NL-04-073	See letter NL-04-145

No.	RAI	Review Area	From Letter	IP3 Response
41	NL-04-073-SG-1	Steam Generator Structural Integrity Evaluation	NL-04-073	Not Applicable
42	NL-04-073-SG-2a	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
42	NL-04-073-SG-2b	Steam Generator Structural Integrity Evaluation	NL-04-073	Not Applicable
42	NL-04-073-SG-2c	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
43	NL-04-073-SG-3a	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
43	NL-04-073-SG-3b	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
44	NL-04-073-SG-4	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
45	NL-04-073-SG-5	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
46	NL-04-073-SG-6	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
47	NL-04-073-SG-7	Steam Generator Structural Integrity Evaluation	NL-04-073	See letter NL-04-145
48	NL-04-073-DOS-1	Dose Assessments	NL-04-073	See letter NL-04-145
49	NL-04-073-DOS-2	Dose Assessments	NL-04-073	See letter NL-04-145
50	NL-04-073-DOS-3	Dose Assessments	NL-04-073	See letter NL-04-145
51	NL-04-073-DOS-4	Dose Assessments	NL-04-073	See letter NL-04-145
52	NL-04-073-DOS-5	Dose Assessments	NL-04-073	See letter NL-04-145
53	NL-04-086-FDF-1	Fuel Design Features and Components	NL-04-086	See letter NL-04-145
54	NL-04-086-FDF-2	Fuel Design Features and Components	NL-04-086	See letter NL-04-145
55	NL-04-086-FDF-3	Fuel Design Features and Components	NL-04-086	See letter NL-04-145
56	NL-04-086-FDF-4	Fuel Design Features and Components	NL-04-086	See letter NL-04-145
57	NL-04-086-FDF-5	Fuel Design Features and Components	NL-04-086	See letter NL-04-145
58	NL-04-086-FDF-6	Fuel Design Features and Components	NL-04-086	Not Applicable
59	NL-04-095-LOC-1	LOCA Transients	NL-04-095	See letter NL-04-145
60	NL-04-095-LOC-2	LOCA Transients	NL-04-095	Not Applicable
	NL-04-100-LOC-3	LOCA Transients	NL-04-100	See NL-04-100-LOC-3
	NL-04-100-LOC-4	LOCA Transients	NL-04-100	See NL-04-100-LOC-4
	NL-04-100-LOC-5	LOCA Transients	NL-04-100	See NL-04-100-LOC-5
61	NL-04-095-NFS-1	NSSS Fluid Systems	NL-04-095	See letter NL-04-145
62	NL-04-095-MDT-1	Mechanical Equipment Design Transients	NL-04-095	Not Applicable
63	NL-04-095-PS-1	Piping and Supports	NL-04-095	Att 2 - Non-Proprietary
64	NL-04-095-GIP-1	Generic Issues and Programs	NL-04-095	See letter NL-04-145
65	NL-04-095-GIP-2	Generic Issues and Programs	NL-04-095	Not Applicable
66	NL-04-095-GIP-3	Generic Issues and Programs	NL-04-095	See letter NL-04-145

No.	RAI	Review Area	From Letter	IP3 Response
67	NL-04-095-GIP-4	Generic Issues and Programs	NL-04-095	See letter NL-04-145
68	NL-04-095-GIP-5	Generic Issues and Programs	NL-04-095	See letter NL-04-145
69	NL-04-095-GIP-6	Generic Issues and Programs	NL-04-095	See letter NL-04-145
70	NL-04-095-GIP-7	Generic Issues and Programs	NL-04-095	See letter NL-04-145
71	NL-04-095-GIP-8	Generic Issues and Programs	NL-04-095	See letter NL-04-145
72	NL-04-095-GIP-9	Generic Issues and Programs	NL-04-095	Not Applicable
73	NL-04-095-GIP-10	Generic Issues and Programs	NL-04-095	See letter NL-04-145
74	NL-04-095-GIP-11	Generic Issues and Programs	NL-04-095	Att 2 - Non-Proprietary
75	NL-04-095-GIP-12	Generic Issues and Programs	NL-04-095	Att 2 - Non-Proprietary
76	NL-04-095-GIP-13	Generic Issues and Programs	NL-04-095	Att 2 - Non-Proprietary
77	NL-04-095-GIP-14	Generic Issues and Programs	NL-04-095	Att 2 - Non-Proprietary
78	NL-04-100-LOC-3	LOCA Transients	NL-04-100	Att 3, 4 - Proprietary
79	NL-04-100-LOC-4	LOCA Transients	NL-04-100	Att 2 - Non-Proprietary
80	NL-04-100-LOC-5	LOCA Transients	NL-04-100	Att 2 - Non-Proprietary
81	NL-04-100-PVM-3a -1	Pressure Vessel Materials	NL-04-100	See letter NL-04-145
81	NL-04-100-PVM-3a -2	Pressure Vessel Materials	NL-04-100	See letter NL-04-145
81	NL-04-100-PVM-3a -3	Pressure Vessel Materials	NL-04-100	See letter NL-04-145
81	NL-04-100-PVM-3a -4	Pressure Vessel Materials	NL-04-100	See letter NL-04-145
82	NL-04-100-PVM-4a -1	Pressure Vessel Materials	NL-04-100	See letter NL-04-145
82	NL-04-100-PVM-4d -1	Pressure Vessel Materials	NL-04-100	Att 2 - Non-Proprietary
82	NL-04-100-PVM-4d -2	Pressure Vessel Materials	NL-04-100	Att 2 - Non-Proprietary
82	NL-04-100-PVM-4d -3	Pressure Vessel Materials	NL-04-100	Att 2 - Non-Proprietary
83	NL-04-100-SG-1	Steam Generator Structural Integrity Evaluation	NL-04-100	Not Applicable
84	NL-04-100-SG-3	Steam Generator Structural Integrity Evaluation	NL-04-100	Not Applicable
85	NL-04-121-NRC-1	Mechanical Equipment Design Transients	NL-04-121	See letter NL-04-145
86	NL-04-121-NRC-2	Piping and Supports	NL-04-121	Att 2 - Non-Proprietary
87	NL-04-121-NRC-3	LOCA Transients	NL-04-121	See letter NL-04-145
88	NL-04-121-NRC-4	Steam Generator Structural Integrity Evaluation	NL-04-121	See letter NL-04-145
89	NL-04-121-NRC-5	NSSS Fluid Systems	NL-04-121	See letter NL-04-145
90	NL-04-121-NRC-6	Pressure Vessel Materials	NL-04-121	See letter NL-04-145
91	NL-04-121-NRC-7	Reactor Systems and Analyses	NL-04-121	Att 2 - Non-Proprietary
92	NL-04-121-NRC-8	Pressure Vessel Materials	NL-04-121	Att 2 - Non-Proprietary

ATTACHMENT 2 TO NL-04-155

ADDITIONAL INFORMATION FOR IP3 SPU LICENSE AMENDMENT REQUEST

BASED ON NRC RAIs ISSUED FOR IP2 SPU

(Refer to Attachments 3 and 4 for other responses involving proprietary information)

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

Non-Proprietary

Question NL-04-073-FP-1:

In NRR RS-001, Revision 0, "Review Standard for Extended Power Uprates," Attachment 2 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that "... power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's application should confirm that these elements are not impacted by the extended power uprate..."

Section 10.1, "Fire Protection (10CFR50 Appendix R) Program," of application report (Attachment III to the January 29 letter) does not address these items. At a minimum, provide a statement to address each of these items.

Response NL-04-073-FP-1:

IP3 SPU results in increased decay heat generation following plant trips. The RHR Cooldown Analysis for SPU, documents that cold shutdown is achieved and maintained within 72 hours. It should be noted that the subject analysis includes a specific "Appendix R" cooldown case that uses only the limited equipment set credited in the IP3 Appendix R Safe-Shutdown Model. The updated cooldown analysis and evaluation addressing SPU confirms that cold shutdown can be achieved and maintained using this same limited equipment set, inclusive of the additional burden associated with SPU. Appendix R program administrative controls are unchanged. The elements of the program such as Fire Suppression; Fire Barriers; Fire protection responsibilities of plant personnel are unchanged. Procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown are unaffected and the radiological release resulting from a fire is also unchanged.

Question NL-04-073-FP-2:

In NRR RS-001, Attachment 2 to Matrix 5, states that "... where licensees rely on less than full capability systems for fire events..., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability ... The licensee should identify the impact of the power uprate on the plant's post-fire safe shutdown procedures."

Section 10.1, of application report does not address the items above. As a minimum, provide a statement to address each of these items.

Response NL-04-073-FP-2:

The evaluation of the IP3 Fire Protection Program was conducted to determine the effect of SPU on the program. There are no modifications required by the SPU to the plant equipment

Non-Proprietary

used for post-fire safe shutdown. There are minor changes required for the procedures. The procedures are capable of being used to achieve post-fire safe shutdown as shown by the response to item FP-3b and as noted in section 4.1.3 of the IP3 SPU Licensing Report.

The analysis and evaluations for the Appendix R cooldown show that the plant is maintained and cooled to 200°F with RCS pressure below the RCS Safety Valve setpoint, with level in the pressurizer, with positive subcooling and with decay heat being removed. Based on the analysis and evaluations for the Appendix R cooldown, the fuel remains covered and therefore fuel design limits are not exceeded and there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Additional detail regarding the Appendix R cooldown analysis and evaluation is provided in the response to question 3b.

Alternate Shutdown Capability

The normal sources of auxiliary ac power at IP3 during plant operation are both off-site power and three emergency diesel generators. If these sources are disabled by fire, the safe-shutdown loads can be supplied by an alternate diesel generator. As addressed in the Indian Point Unit 3 UFSAR, Section 9.6.2.5, "Safe Shutdown Capability in Case of Fire," there are two alternate shutdown schemes credited in compliance with 10CFR50 Appendix R, Section III.G.3, that utilize an alternate diesel generator (referred to herein as the "Appendix R diesel generator"): (1) a scheme that makes use of local control stations in the Auxiliary Feedwater (AFW) Pump Room, Primary Auxiliary Building (PAB), and the Auxiliary Boiler Feedwater Pump Building to effect shutdown following a fire that requires safe shutdown from outside the Control Room, and (2) a scheme that makes use of the Appendix R diesel generator aligned to the 480V vital buses to ensure safe shutdown from the Control Room.

The Appendix R diesel generator (DG) is a dedicated 2500 kw diesel generator located in its own enclosure in the yard area. AC power generated by the Appendix R DG can be supplied to 6.9 kv buses 5 and 6. These buses in turn feed 6.9 kv buses 1 and 3, which supply 480V to buses 312 through 313 through step-down transformers. Supporting services for the Appendix R ac power source are independent of the supporting equipment used by the emergency diesel generators (e.g., service water, 125V dc control power, starting air, and fuel oil).

The alternative power system, as described above, is designed to be independent and sufficiently isolated from the existing emergency power system to ensure the availability of power to the safe shutdown equipment of concern in the event of fires in the Control and Diesel Generator Buildings. In case of a fire affecting certain portions of the PAB and Electrical Tunnels which could disable emergency diesel generator auxiliaries, the Appendix R DG can be used to power the 480V vital buses to ensure safe shutdown from the Control Room.

The local control station in the PAB is provided with indication of pressurizer level, RCS pressure, and source range neutron flux. Operators at this location will control RCS boration and makeup with the charging pumps. The local control station in the AFW Pump Room is provided with indication of steam generator water level and pressure, pressurizer level, RCS pressure, and RCS loop 31 hot and cold leg temperature. The local control station for the Steam Generator atmospheric relief valves is located in the Auxiliary Feedwater Pump Building.

The SPU does not affect the above-described alternate shutdown schemes. There are no modifications required by the SPU to the plant equipment used for post-fire safe shutdown.

Non-Proprietary

Evaluation of Appendix R DG load requirements under SPU conditions shows that there are no significant load increases that would affect the conclusions of the existing Appendix R DG load analysis.

Question NL-04-073-FP-3:

Section 10.1 of Attachment III (WCAP-16157-P) to the License Amendment Request, states that "for the SPU, the steam generator dryout time provides adequate time for the operator to supply feedwater to the secondary side of the steam generator. The Appendix R plant cooldown analysis under SPU conditions shows that IP2 complies with the Appendix R requirement that cold shutdown be achieved within 72 hours after reactor trip following a fire."

a. Provide a discussion, including numerical values, of the change, if any, in steam generator dry-out time as a result of the SPU, and reference to the calculations performed to determine there is adequate time for the required operator action.

b. Provide a discussion, including numerical values, of the change, if any, in time to achieve cold shutdown as a result of the SPU, and reference to the calculations performed to determine that it can be achieved within the required time frame.

c. Provide corresponding references, including appropriate extracts from the Updated Final Safety Analysis Report (UFSAR), plant-specific Appendix R evaluation, etc., that justify these claims.

Response NL-04-073-FP-3a:

The Indian Point Fire Protection Plan states that the steam generators would not dryout in 30 minutes. For the Stretch Power Uprate, the steam generator dry out time was predicted using the RETRAN code and an IP3 plant-specific calculation. The initiating event was a Loss of all AC Power to the Station Auxiliaries. The analysis conservatively assumed an initial power level of 102% of 3216 MWt and a minimum initial SG level of 35%. Decay heat was based on the 1979 version of ANS 5.1 and includes a 2 sigma uncertainty. The results of this analysis showed that the steam generators would boil dry after approximately 39 minutes.

To assure continued natural circulation and removal of decay heat by steaming to the atmosphere, auxiliary feedwater should be injected prior to the steam generator dryout. This ability was demonstrated by timed field walkdowns, which showed that auxiliary feedwater could be injected well within 30 minutes.

Response NL-04-073-FP-3b:

For purposes of Appendix R cooldown analysis, the RHR cooldown analysis for Appendix R conditions is discussed in Section 4.1.3 of WCAP-16212-P and documents the cooldown from the RHR cooldown initiation to achieving cold shutdown with in the Appendix R requirement of 72 hours. The evaluation of a natural circulation cooldown from normal operating temperature (NOT) to RHR cooldown initiation conditions at 350°F is discussed below.

Non-Proprietary

Natural Circulation Cooling Analysis (NOT to 350°F)

To demonstrate that the stretch power uprate (SPU) does not adversely affect the natural circulation cooling capability of the IP3 plant, an evaluation for post-fire safe shutdown was performed. The evaluation considered only on the limited equipment set available for the IP3 Appendix R safe-shutdown conditions. It was based on the scheme that makes use of local control stations in the Auxiliary Feedwater (AFW) Pump Room, Primary Auxiliary Building (PAB), and the Auxiliary Boiler Feedwater Pump Building to effect shutdown following a fire that requires safe shutdown from outside the Control Room. Following plant trip and control room evacuation, the plant is cooled by steam relief from the Main Steam Safety Valves. RCPs are tripped and pressurizer PORVs and pressurizer heaters are assumed unavailable. One motor-driven AFW pump feeding 2 Steam Generators with manual flow control is credited after 30 minutes. One charging pump is assumed after 60 minutes to provide RCS makeup from the RWST and to increase RCS boron concentration. Charging flow is manually controlled. Plant cooldown at 25°F/hr is commenced 4 hours after reactor trip. Steam Generator Atmospheric Relief valves are manually operated to control the cooldown. A total delay of 8 hours is assumed to allow the upper head to cool or "soak" before depressurizing to the RHR cut-in pressure. As per the ERG generic analysis, this upper-head soak delay is included to allow the upper-head region sufficient time to cool due to the assumed loss of control rod drive mechanism (CRDM) fans.

The SPU evaluation concluded that the RCS pressure would be stabilized at 375 psia (360 psig) with Thot < 350°F in all hot legs and at the core exit at approximately 28 hours after reactor trip. From this condition, RHR cooling can be initiated to cool the RCS to < 200°F within 72 hours.

RHR Cooling Analysis (350°F to 200°F)

The SPU affects the plant cooldown time(s) since core power, and therefore the decay heat increases. The plant cooldown calculation was performed at a core power of 3216 MWt to support the SPU. The RCS heat capacity and the other RHR heat loads were explicitly considered in these analyses. The analysis was performed to confirm that the RHR and CCW systems continue to meet their design basis functional requirements and performance criteria for plant cooldown under the uprated power conditions. The two-train system alignment was considered to address the design capability in the *Indian Point Unit 3 Updated Final Safety Analysis Report* (UFSAR). In addition, a cooldown analysis was performed to support the worst-case scenario for the 10CFR50 Appendix R (Reference 4) fire safe shutdown analysis.

The following considerations were applied to these cooldown analyses:

- The CCW and RHR HX data assumes 5-percent tube plugging, as was used for the previous cooldown analyses of record (AOR).
- The design service water temperature of 95°F was assumed. For the Appendix R cooldown, the CCWS supply temperature is limited to 125°F.
- Various CCWS auxiliary heat loads and the RCS heat capacity were included in the normal cooldown cases and the Appendix R plant cooldown case. These heat loads, along with an increase in the spent fuel pool heat load (assuming a full SFP of fuel that has operated at 3216 MWt) were used in the cooldown analysis.

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- Decay heat curves based on 24-month fuel cycles were used.
- Service water (SW) flow rates for Appendix R cooldown were varied to minimize SW flow demand while meeting the Appendix R criteria as shown in Table NL-04-073-FP-1.

The Appendix R/safe shutdown cases continue to meet the 72-hour time limit for cold shutdown. For these cases, the minimum CCW HX service water flow to meet the time 72 hour cooldown time limit criterion was determined as shown in Table NL-04-073-FP-1.

Acceptable RHR cooldown performance is provided at the SPU conditions for normal plant cooldown and the limiting Appendix R/safe shutdown cases, based on the service water flows shown in Table NL-04-073-FP-1.

Table NL-04-073-FP-1
SPU Cooldown Analyses Results

Cases	Cooldown Time to 140°F (hrs. after shutdown)	Cooldown Time to 200°F (hrs. after shutdown)	RHR Initiation Time @350°F (hrs. after shutdown)(1)	Total SW Flow (gpm)
A. App. R, Enhanced CCW UA/U, 5700 gpm SW Flow	N/A	64.8(2)	29.0	5700
B. App. R, Enhanced CCW UA/U, SW Flow Minimized to Meet 72-hr. Cooldown Time	N/A	71.8	29.0	4700
C. App. R, Original Design SSC UA/U, SW Flow Minimized to Meet 72-hr. Cooldown Time	N/A	71.9	29.0	5324
D. Same as A without SFP Heat Load	N/A	58.0(2)	29.0	5700
E. Same as B without SFP Heat Load	N/A	71.8	29.0	3596
F. Same as C. Without SFP Heat Load	N/A	72.0	29.0	3918

Notes:

1. The 29-hour cut-in time for the Appendix R cases, limited by the CCWS supply temperature, is also indicative of the cut-in time assumed in the radiological consequences analyses of accidents with secondary side releases (that is, SGTR).
2. These cases increase the component cooling water return piping temperature compared to the previous 1.4% MUR Appendix R analysis. Previous Appendix R cases had a maximum return temperature of 173°F, and the temperature for Case D is 188°F, which remains bounded by post-LOCA conditions.

Appendix R Cooldown analysis and evaluation demonstrate that IP3 can be cooled from the normal operating temperature to the RHR initiation conditions using a natural circulation cooling

process in 29 hours and from the RHR initiation condition to cold shutdown within the requirement of 72 hours.

Response NL-04-073-FP-3c:

The Indian Point Unit 3 UFSAR, Table 9.3-2, "Residual Heat Removal Loop Component Data," documents the 72 hour requirement regarding time after plant shutdown to reach cold shutdown conditions for Appendix R fire scenarios. This table also documents the time after plant shutdown that shutdown cooling is initiated. As indicated in the response to Question FP-3b, under SPU conditions the time after plant shutdown that RHR shutdown cooling is initiated is 29 hours.

As addressed in the responses to Questions FP-3a and FP-3b, plant specific analysis and evaluation were performed to show that IP3 is capable of achieving cold shutdown conditions within 72 hours after reactor trip following a fire.

Alternate Shutdown Capability

The normal sources of auxiliary ac power at IP3 during plant operation are both off-site power and three emergency diesel generators. If these sources are disabled by fire, the safe-shutdown loads can be supplied by an alternate diesel generator. As addressed in the Indian Point Unit 3 UFSAR, Section 9.6.2.5, "Safe Shutdown Capability in Case of Fire," there are two alternate shutdown schemes credited in compliance with 10CFR50 Appendix R, Section III.G.3, that utilize an alternate diesel generator (referred to herein as the "Appendix R diesel generator"): (1) a scheme that makes use of local control stations in the Auxiliary Feedwater (AFW) Pump Room, Primary Auxiliary Building (PAB), and the Auxiliary Boiler Feedwater Pump Building to effect shutdown following a fire that requires safe shutdown from outside the Control Room, and (2) a scheme that makes use of the Appendix R diesel generator aligned to the 480V vital buses to ensure safe shutdown from the Control Room.

The Appendix R diesel generator (DG) is a dedicated 2500 kw diesel generator located in its own enclosure in the yard area. AC power generated by the Appendix R DG can be supplied to 6.9 kv buses 5 and 6. These buses in turn feed 6.9 kv buses 1 and 3, which supply 480V to buses 312 through 313 through step-down transformers. Supporting services for the Appendix R ac power source are independent of the supporting equipment used by the emergency diesel generators (e.g., service water, 125V dc control power, starting air, and fuel oil).

The alternative power system, as described above, is designed to be independent and sufficiently isolated from the existing emergency power system to ensure the availability of power to the safe shutdown equipment of concern in the event of fires in the Control and Diesel Generator Buildings. In case of a fire affecting certain portions of the PAB and Electrical Tunnels which could disable emergency diesel generator auxiliaries, the Appendix R DG can be used to power the 480V vital buses to ensure safe shutdown from the Control Room.

The local control station in the PAB is provided with indication of pressurizer level, RCS pressure, and source range neutron flux. Operators at this location will control RCS boration and makeup with the charging pumps. The local control station in the AFW Pump Room is provided with indication of steam generator water level and pressure, pressurizer level, RCS

pressure, and RCS loop 31 hot and cold leg temperature. The local control station for the atmospheric dump valves is located in the Auxiliary Feedwater Pump Building.

The SPU does not affect the above-described alternate shutdown schemes. There are no modifications required by the SPU to the plant equipment used for post-fire safe shutdown. Evaluation of Appendix R DG load requirements under SPU conditions shows that there are no significant load increases that would affect the conclusions of the existing Appendix R DG load analysis.

Entergy procedure 3-AOP-SSD-1 Revision 2 is the procedure for Post-Fire safe shutdown operations. This procedure has been reviewed for SPU and only minor changes are required for SPU.

Conclusions

The Indian Point Unit 3 UFSAR, Table 9.3-2, "Residual Heat Removal Loop Component Data," documents the 72 hour requirement regarding time after plant shutdown to reach cold shutdown conditions for Appendix R fire scenarios. This table also documents the time after plant shutdown that shutdown cooling is initiated. As indicated in the response to Question FP-3b, under SPU conditions the time after plant shutdown that RHR cooling is initiated is 29 hours.

As addressed in the responses to Questions FP-3a and FP-3b, plant specific analysis and evaluation were performed to show that IP3 is capable of achieving cold shutdown conditions within 72 hours after reactor trip following a fire.

Question NL-04-073-EL-1:

Address the compensatory measures that the licensee would take to compensate for the depletion of the nuclear unit megavolt-ampere reactive (MVAR) capability on a grid-wide basis.

Response NL-04-073-EL-1:

See Entergy letter NL-04-156 for the response to this question.

Question NL-04-073-PVM-4:

Table 5.9-5 of the application report indicates a flaw depth of 0.50-inch for safety and relief nozzle (corner) and 0.15-inch for upper shell meet the fracture toughness requirements of Appendix G of the ASME Code (NOTE: Table 5.9-5 indicates K_I/K_{IR} is 0.94 for the safety and relief nozzle (corner) and 1.0 for the upper shell).

a. Describe the analysis that determined a 0.50-inch flaw depth for the safety and relief nozzle (corner) and a 0.15-inch flaw depth for the upper shell will meet the fracture toughness requirements of Appendix G of the ASME Code.

b. Identify whether the analysis satisfies the requirements of Article G-2220 of Section XI of the ASME Code. Does the analysis for the safety and relief nozzles and upper shell satisfy these structural factors?

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c. Describe the non-destructive examination technique which will be utilized to inspect the safety and relief nozzles and upper shell.

d. Provide the data, a description of the analysis, and the probability of detection of flaws with a depth of 0.50-inch for the safety and relief nozzle and 0.15-inch for the upper shell.

Response NL-04-073-PVM-4a:

See letter NL-04-145 for response.

Response NL-04-073-PVM-4b:

See letter NL-04-145 for response.

Response NL-04-073-PVM-4c:

As noted in NL-04-145 response to NL-04-073-PVM-4a, the revised calculations for the pressurizer nozzles demonstrate that the postulated flaw size meets the requirements of Appendix G (1/4t or 1 inch).

Safety and Relief Nozzle:

The IP3 Pressurizer has three Code Safety Inner Radius Nozzles (20IR, 21IR, and 22IR) and one Power Operated Relief Inner Radius Nozzle (23IR). These nozzles are ASME Section XI, Code Category B-D, Item B3.120. These nozzles require volumetric examinations per ASME Section XI, 1989 Code Edition. However, for the Third 10-year Interval, which ends in July 2009, Entergy submitted Relief Request 3-16 to perform a remote visual (VT-1) with color capability on each of the nozzle inner radius sections. The NRC approved this relief request on April 22, 2003 (TAC No. MB4766).

Upper Shell:

The IP3 Pressurizer shell has 9 circumferential welds (1, 3, 5, 7, 9, 11, 13, 15, 17) and 8 longitudinal welds (2, 4, 6, 8, 10, 12, 14, 16). For the purposes of this discussion, welds 16, and 17 will be considered the upper shell welds since weld 17 is the uppermost circumferential weld and weld 16 is its intersecting longitudinal weld. These welds are the welds required to be inspected by ASME Section XI, Table IWB-2500-1, Code Category B-B, Item B2.11 and B2.12. Table IWB-2500-1, Category B-B, Note 4 requires the volumetric coverage stipulated by Figures IWB-2500-1 and 2 be performed on 100% of the Code Class 1 circumferential welds and the adjoining 1 foot section of the longitudinal welds. The upper circumferential (17) and longitudinal (16) welds are enclosed in a biological and missile shield and are completely inaccessible for volumetric examination (NDE). Therefore, for the Third 10-year Interval, which ends in July 2009, Entergy submitted Relief Request 3-14 to perform a visual examination (VT-2) for leakage during system pressure tests performed each refueling outage in accordance with IWB-2500, Category B-P and Code Case N-498-1. The NRC approved this relief request on April 22, 2003 (TAC No. MB4766).

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Response NL-04-073-PVM-4d:

See letter NL-04-145 for response.

Question NL-04-073-ENV-3:

Section 5.7 states that the current power uprate qualifies for a categorical exclusion under 10 CFR 51.22(c)(9). Provide the environmental evaluation performed for the proposed power uprate in accordance with Appendix B of the facility operating license. The response should include a discussion of the radiological and non-radiological impacts of the proposed uprate.

Response NL-04-073-ENV-3:

The environmental evaluation of the impact of the IP3 Stretch Power Uprate (SPU) is provided in the IP3 SPU Licensing Attachment III, Sections 6.11 and 11. The evaluation concludes that the proposed license amendment to increase rated thermal power to 3216 MWt and the related changes to the plant technical specifications do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9).

The radiological analysis for annual radwaste effluent releases estimates the impact of uprate on normal operation offsite doses using scaling techniques. The system parameters for uprated conditions used in the analysis reflect the flow rates and coolant masses at a NSSS power level of 3230 MWt and a core power level of 3280.3 MWt. The evaluation utilizes offsite doses based on an average 5 yr set of organ and whole body doses calculated from effluent reports for the years 1998 through 2002 extrapolated to 100% availability at the associated average annual core power level. Releases occurring during periods of Unit shutdown are conservatively lumped with operational releases and included in the doses scaled for 100% availability.

The qualitative assessment is based on methodology and equations found in NUREG-0017 Rev. 1 (Ref 1), and a comparison of the change in power level and in plant coolant system parameters (e.g., reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, steam generator moisture carryover, etc.) for both pre-uprate and uprate conditions. To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group of radioisotopes pertinent to the release pathway is applied to the average doses previously determined as representative of operation at pre-uprate conditions (at 100% availability) to estimate the maximum potential increase in effluent doses due to the uprate and demonstrate that the estimated off-site doses following uprate, although increased, will continue to remain below regulatory limits.

The criteria used in the evaluation include a liquid and gaseous radwaste systems' design capable of maintaining normal operation offsite releases and doses within the requirements of 10CFR50, Appendix I (Ref. 2) following power uprate. (Note that actual performance and operation of installed equipment, and reporting of actual offsite releases and doses continues to be controlled by the requirements of the Technical Specifications and the Offsite Dose Calculation Manual.)

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The non-radiological impact of the IP3 SPU to 3216 MWt was reviewed and evaluated considering the information contained in the Final Environmental Statement (FES) (Ref. 3) for the station. Section 1 of Appendix B of the Facility Operating License requires environmental concerns identified in the FES that relate to water quality matters to be regulated by way of the State Pollutant Discharge Elimination System (SPDES) permit (Ref. 4) limits. The Indian Point SPDES restrictions on discharge temperatures and discharge flow rates for the station were evaluated along with the flow limits set forth in IP3 SPDES Consent Order (Ref. 5).

The criteria used in the evaluation required that the environmental impacts associated with the proposed changes be within the existing regulatory release permits.

Uprate Evaluation

Radiological Effects
The power uprate has no significant impact on the expected annual radwaste effluent releases/doses (i.e. all doses remain a small percentage of allowable Appendix I doses) as summarized below.

1. Expected Reactor Coolant Source Terms

The requested SPU is an increase of 4.85% in reactor power and the source term would increase by the same amount. However, based on a comparison of base vs. power uprate input parameters, and the methodology outlined in NUREG 0017, the effective factor increase in dose depending on chemical group of isotopes released, ranges between 1.11 to 1.12. Note that the maximum expected increase in the reactor coolant source due to the uprate is well within the uncertainty of the existing (NUREG 0017 based) expected reactor coolant isotopic inventory used for radwaste effluent analyses.

2. Estimated Impact on Effluent Doses due to Uprate

Gaseous Effluents

	Dose
Gamma Air (mrad)	3.74E-04
Beta Air (mrad)	7.60E-04
Iodine and Particulate (mrem)	8.22E-04

Liquid Effluents

	Dose
Organ Dose (mrem)	3.00E-03
Adult Total Body (mrem)	1.22E-03

The estimated doses due to uprate are presented above and are a fraction of that allowable under 10CFR50 Appendix I.

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3. Solid Radioactive Waste

Though solid radwaste is not specifically addressed in 10 CFR 50, Appendix I, for completeness relative to radwaste assessments, the impact of core uprate on solid radwaste generation is summarized below.

For a "new" facility, the estimated volume and activity of solid waste is linearly related to the core power level. However, for an existing facility that is undergoing power uprate, the volume of solid waste would not be expected to increase proportionally, since the power uprate neither appreciably impacts installed equipment performance, nor does it require drastic changes in system operation or maintenance. Only minor, if any, changes in waste generation volume are expected. However, it is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity bounded by maximum increase in power.

Therefore, following uprate, the liquid and gaseous radwaste effluent treatment system will remain capable of maintaining normal operation offsite doses within the requirements of 10 CFR 50 Appendix I. Only minor, if any, changes in solid waste generation volume are expected.

Non-Radiological Effects

The IP3 FES that was approved by the AEC in February 1975 for a maximum calculated thermal power of 3,216.5 MWt envelops the SPU condition. Increased heat rejection to the plant systems is expected to result in a nominal calculated increase in discharge temperature to the Hudson River. This temperature increase falls within the applicable SPDES permit thermal limits for Indian Point.

Final Environmental Statement (FES)

The environmental issues associated with the issuance of an operating license for Indian Point Unit 3 were originally evaluated in the Indian Point Unit 3 FES that was approved by the AEC in February 1975. The AEC approved Final Environmental Statement (FES) relates to operation of Indian Point Nuclear Generating Plant Unit No. 3 (Volume 1, page I-1 Section I) and has addressed plant operation up to a maximum calculated thermal power of 3,216.5 MWt. The SPU does not significantly change the types or the amount of any effluents that may be released offsite that have not already been evaluated and approved in the FES for a power rating of 3,216.5 MWt. Since the AEC approved FES has already addressed plant operation up to a maximum calculated thermal power of 3,216.5 MWt, the SPU has been determined to not significantly impact the FES.

State Pollutant Discharge Elimination System (SPDES) Permit and Consent Order Flows

The State Pollutant Discharge Elimination System (SPDES) permit places restrictions on discharge temperatures and discharge flow rates to the river for the station. The Indian Point SPDES restrictions on discharge temperatures and discharge flow rates for the station were evaluated along with the flow limits set forth in Indian Point 3 Consent Order.

IP3 operation at the SPU power level of 3216 MWt will increase the exhaust steam flow and

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duty of the main condenser and, therefore, increase the heat load rejected by the Circulating Water System (CWS). The SPU evaluation assumes the existing CWS pumps are not modified and continue to operate at the same flow rates. Heat load increases due to SPU in the Normal and Emergency Service Water System (SWS) will also result in increase in the SWS discharge temperature.

The SPDES permit has the following limitations that regulate the discharge temperature:

> The maximum discharge temperature at station DSN001 shall not exceed 43.3°C (110°F)
>
> and
>
> Between April 15 and June 30 the daily average discharge temperature at station DSN001 shall not exceed 34°C (93.2°F) for an average of more than 10 days per year during the term of the permit beginning with 1981; provided that in no event shall the daily average discharge temperature at Station DSN001 exceed 34°C (93.2°F) on more than 15 days between April 15 and June 30 in any year.

The Station's discharge temperatures were evaluated using the heat balance model (PEPSE). The temperature rise across each condenser from the model was tuned based on plant data from July 28, 2003. In addition, State Consent Order flows were used as input to the PEPSE model. Additional conservatism was added to the calculated temperature to account for miscellaneous plant cooling to determine plant discharge temperature. Plant historic data for the river water inlet temperature was iterated to predict the maximum plant discharge temperatures.

Based on conservative maximum plant discharge temperatures and the existing administrative controls imposed on plant operation, it is concluded that the station will remain capable of meeting SPDES permit limits at SPU conditions.

References

1. NUREG 0017, Rev. 1, April 1985, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors"

2. Code of Federal Regulations Title 10, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion As Low As Reasonably Achievable for Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents".

3. Final Environmental Statement Related to Operation of Indian Point Nuclear Generating Plant Unit No. 3, Consolidated Edison Company of New York, Inc. Docket No. 50-286, February 1975

4. New York State Department of Environmental Conservation, State Pollutant Discharge Elimination System (SPDES) Discharge Permit, 11/90

5. Fourth Amended Stipulation of Settlement and Judicial Consent Order, Index No. 6570-91, RJI No. 0191-ST3251

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Question NL-04-095-LOC-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 NL-04-095-LOC-3:

The response this question contains proprietary information. The proprietary and non-proprietary versions of the response are provided as response NL-04-100-LOC-3 in Attachments 3 and 4 of this letter, respectively.

Question NL-04-095-LOC-4:

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

Response NL-04-095-LOC-4:

See response NL-04-100-LOC-4.

Question NL-04-095-LOC-5:

Tables 6.2-3 and 6.2.5 in the Application Report provide LBLOCA and SBLOCA analyses 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, and cladding outside and post-rupture inside oxidation. Also include the results for fuel resident from previous cycles.

Response NL-04-095-LOC-5:

See response NL-04-100-LOC-5.

Question NL-04-095-PS-1:

In Section 9.9.3 of the Application Report, the justifications provided on page 9.9-3 for not evaluating the piping and support systems where the increase in temperature, pressure and flow rate are less than 5 percent of the current rated design basis condition are qualitative and nonspecific. For instance, the licensee stated that these increases are some what offset by conservatism in analytical methods used. The licensee also indicated that conservatism may include the enveloping of multiple thermal operating conditions.

Provide the technical basis for not evaluating these piping and support systems. The technical justifications should be based on specific quantitative assessment or intuitively conservative deduction. Also, discuss how the flow effects on the transient loads, which may increase non-proportional to the ratio of flow rate change, are considered (see page 9.9.2).

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Response NL-04-095-PS-1:

All piping systems with change factors greater than 1.0 were evaluated to document pipe stress and support system acceptability.

The method used to qualify the main steam piping involved detailed computer analysis of the piping system. Although operating temperatures and pressures at SPU conditions were bounded by the existing data considered in the design basis piping evaluations, the main steam piping was evaluated using detailed computer analysis in order to reconcile an approximate 6 percent flow rate increase that results due to SPU conditions. These detailed evaluations were performed to assess the potential increase in fluid transient stresses and loads resulting from a turbine stop valve (TSV) closure event.

A summary of revised main steam system stress levels corresponding to SPU conditions is provided in Table 1. The results presented include existing stress levels (i.e., pre-uprate), revised pipe stress levels for SPU conditions, allowable stress for the applicable loading condition, and the resulting design margin for each piping analysis that was evaluated to reconcile SPU conditions. The design margin provided is based on the ratio of the calculated stress divided by the allowable stress.

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Table 1 Stress Summary at SPU Conditions					
Piping Analysis Description	**Loading Condition**	**Existing Stress (psi)**	**SPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin**
Main Steam Line 1 (Inside Containment)	DL + LP + TSV	12,410	12,587	21,000	0.60
Main Steam Line 2 (Inside Containment)	DL + LP + TSV	11,833	11,993	21,000	0.57
Main Steam Line 3 (Inside Containment)	DL + LP + TSV	12,812	13,234	21,000	0.63
Main Steam Line 4 (Inside Containment)	DL + LP + TSV	12,649	12,811	21,000	0.61
Main Steam Lines 1, 2, 3 and 4 (Outside Containment)	Thermal expansion	18,489	19,171	19,950	0.96

Notes:
1. Loading condition "DL + LP + TSV" corresponds to the combination of stresses due to deadweight + pressure + turbine stop valve effects.
2. Stress Ratio reported is based on the ratio of SPU stress divided by the allowable stress.

For the remaining piping systems with thermal and pressure change factors greater than 1.0, these piping systems (i.e., condensate, feedwater, extraction steam, feedwater heaters vents and drains, and moisture separator and reheater drains systems) were evaluated using computer analyses, as well as performing a field walkdown of the piping systems.

The results presented in Tables 2 through 5 contain stress data for the critical portions of the Condensate, Feedwater, Extraction Steam and Feedwater Heater Vent & Drains Systems. The results provided include existing stress levels (i.e., pre-uprate), revised pipe stress levels for SPU conditions, allowable stress for the applicable loading condition, and the resulting design margin for each piping analysis that was evaluated to reconcile SPU conditions. The design margin provided is based on the ratio of the calculated SPU stress divided by the allowable stress.

Table 2 Condensate System Stress Summary					
Piping Analysis Description	Loading Condition	Existing Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Design Margin
Heaters 35A/B/C to FW Pumps	Thermal	15,540	15,668	22,500	0.70

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Table 3 Feedwater System Stress Summary					
Piping Analysis Description	Loading Condition	Existing Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Design Margin
Feedwater to SG 31	DL + LP	6,532	7,162	17,500	0.41
Feedwater to SG 32	DL + LP	8,095	8,725	17,500	0.50
Feedwater to SG 33	DL + LP	7,569	8,199	17,500	0.47
Feedwater to SG 34	DL + LP	7,532	7,982	17,500	0.46

Table 4 Extraction Steam System Stress Summary					
Piping Analysis Description	Loading Condition	Existing Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Design Margin
Extraction Steam to Heaters 33A/B/C	DL + LP	1,734	1,780	15,000	0.12
Extraction Steam to Heaters 33A/B/C	Thermal	4,620	4,727	22,500	0.21

Table 5 FW Heater Vents and Drains System Stress Summary					
Piping Analysis Description	Loading Condition	Existing Stress (psi)	SPU Stress (psi)	Allowable Stress (psi)	Design Margin
Heaters 34A/B/C to Heaters 33A/B/C	DL + LP	1,808	1,825	15,000	0.12
Heaters 34A/B/C to Heaters 33A/B/C	Thermal	13,308	13,544	22,500	0.60

In addition to the detailed evaluations that were performed of the critical portions of the Condensate, Feedwater, Extraction and Feedwater Heater Vents and Drains Systems described above, a turbine building plant walkdown of these piping systems was also performed to review the individual piping layouts and associated pipe support configurations. The purpose of these piping system walkdowns was to assess the adequacy of the installed piping deadweight spans and to review the existing thermal flexibility of the piping systems. The overall assessment from the walkdowns performed concluded that the existing piping that was observed was adequately supported and contained adequate flexibility to accommodate the small pressure and temperature changes resulting from SPU. Piping systems were determined to be adequately supported if the piping was supported by vertical supports, rod hangers or

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spring hangers, such that piping spans were consistent with the guidance presented in ASA B31.1-1955, Code for Pressure Piping. Piping systems were determined to have adequate flexibility if the following attributes were observed:

- Piping lengths and offsets were consistent with simplified industry methods of determining flexibility (for example, nomographs).
- There were no non-integral or integrally welded piping anchors installed.
- There was a sufficient and reasonable number of piping elbows installed providing thermal flexibility.

Hence, based on the detailed evaluations of the critical portions of these systems along with the additional plant walkdowns that were performed, it is concluded that these piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature and pressure effects resulting from SPU conditions.

Question NL-04-095-GIP-11:

Section 10.8.4, "SPU Equipment Qualification Evaluation," states that accident temperatures outside containment in the steam and feedline penetration area have been reanalyzed and result in higher temperatures, and that all equipment outside containment required for accident response have been justified as qualified.

Discuss the evaluation of any safety-related pumps and valves located in the steam and feedline penetration area, and the impact on their performance from higher temperature due to SPU conditions.

Response NL-04-095-GIP-11:

The equipment types in the main steam and feedline penetration area on the EQ list are ASCO solenoid valves, Namco limit switches, Westinghouse and Buchanan terminal blocks, and associated cables manufactured by GE PVC and Rockbestos Firewall III Cable, CONAX Connectors, and Fisher E/P Transducers. There are no EQ pumps in this area. The EQ valves evaluated are the ASCO valves.

The equipment was evaluated using the thermal analysis of the components for a 1.4 square foot MSLB header break downstream from the Main Steam Isolation Valves, summer building ventilation configuration for the outdoor louvers and 102% SPU power.

The results are presented in Graph 1 for the ASCO solenoid valves. The temperature of the ASCO case and the coil are very close. The coil is only energized for 20 seconds to perform the safety function of tripping the MSIVs, so there is little heat generated within the component.

As shown on Graph 1, the temperature of the ASCO coil and case remain below the qualification test temperature. The qualification testing for the ASCO valves included a pre-test accident soak to assure the ASCOs reached the test chamber temperature.

The cables that are associated with the ASCO solenoid valves are installed in conduit. These cables were also thermally analyzed. Graph 2 indicates that the cables remain below their qualification temperature.

Non-Proprietary

1. ASCO solenoid valves are qualified for both the 10-minute and the 15-minute operator response time. The peak temperature of the 15-minute operator response time is 350.749°F. The ASCO test report (Reference 1) demonstrates that the ASCO solenoid valves have been tested to temperatures enveloping this peak temperature.

2. Buchanan terminal blocks are qualified for both the 10-minute and the 15-minute operator response time because the qualification test envelopes the accident profile. The maximum thermal lag temperature of the terminal blocks is 322.3°F. The qualification test peaks at 346°F.

3. The Westinghouse terminal blocks are not qualified for the 322.3°F temperature (15 minute response), but are qualified for the 10-minute operator response time with the peak thermal lag temperature of 279°F compared to the qualification test temperature of 295°F (Reference 2).

4. GE Flamenol PVC cable is shown to be qualified for the 10-minute operator response time. The peak temperature of the 15-minute operator response time is 362.676°F (1.4 SF break in winter with 15-minute operator response). The peak test temperature is 370°F. The time over the qualification curve can be shown to reduce the time of the qualification temp of 350°F long term by only 748 seconds. The impact is an 8% reduction of thermal life at 350°F but no impact on the overall transient being enveloped by the qualification test. Therefore the GE Cable is considered also qualified for the 15-minute operator response time.

5. The Rockbestos Firewall III cable is qualified to 674°F (Reference 3).

6. Conax conduit seals are qualified for both the 10-minute operator response time and the 15-minute operator response time. The peak thermal lag temperature of the Conax is 329.876°F (1.4 SF break in winter with 15 minute operator response time). The qualification test peaks at 380°.

7. Namco limit switches are qualified for the 10-minute operator response time. The peak temperature of the 15 minute operator response time of 335.313°F (1.4 SF break in winter with 15 minute operator response) exceeds the existing qualification test of 315°F. However, the following qualification test reports yield higher temperatures: QTR 157, Rev 1 peaks at 364°F and QTR 155 peaks at 341°F. Also, the length of time that the Namco exceeds the test temperature of 315°F is 1777 seconds and results conservatively in reducing the thermal life at 315°F slightly (4%) which does not affect the test enveloping the accident temperature. Therefore, the Namco limit switches are considered also qualified for the 15-minute operator response time.

8. The Fisher E/P Transducers were already evaluated for the higher Pre-SPU power level and were found acceptable.

<u>Summary</u>: All of the equipment, except the Westinghouse terminal blocks, is qualified for accident conditions for the longer 15-minute operator response time. All of the equipment is qualified for the 10-minute operator response time.

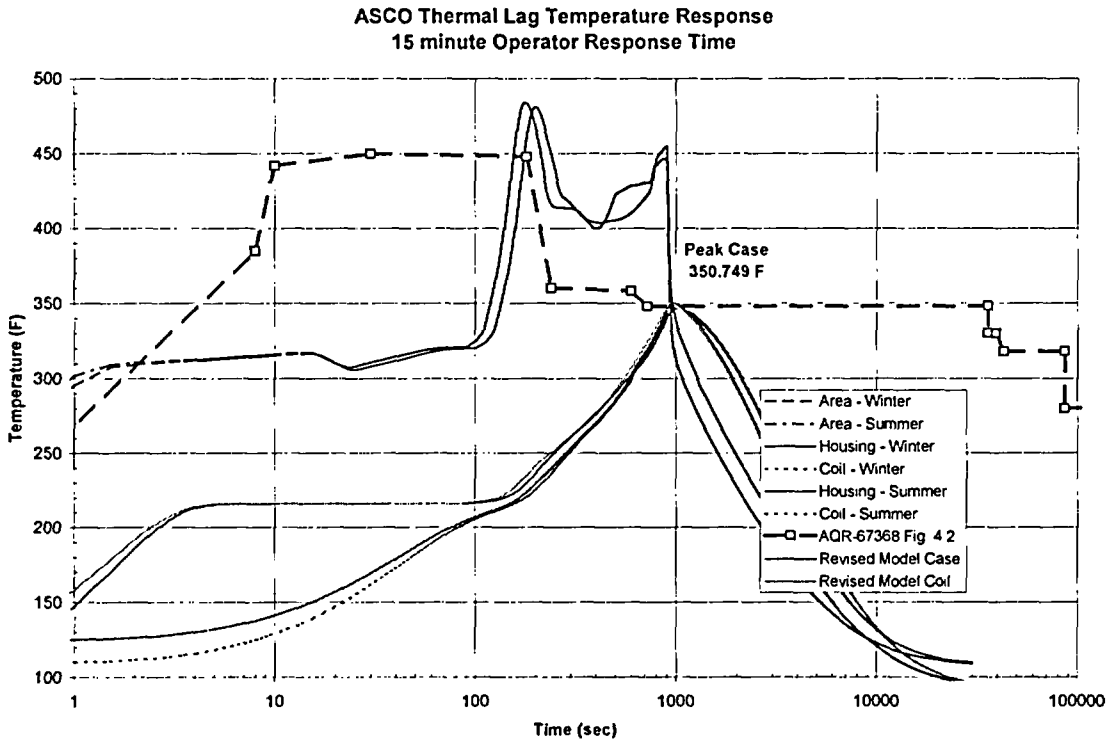
Non-Proprietary

References:

1. ASCO Test Report, EQ-QR 03.02.01, AQR-67368, Rev. 1
2. EQ file EQ-SE-17.01.01, Westinghouse Terminal Blocks
3. Wyle Qualification Report 4795R01, 12/24/2002, "Environmental Qualification Extension of Rockbestos Firewall III XLPE and GE Flamenol PVC Cables for use in Entergy Nuclear Northeast Indian Point Energy Center Unit 3"

Non-Proprietary

Graph 1 for NL-04-095-GIP-11



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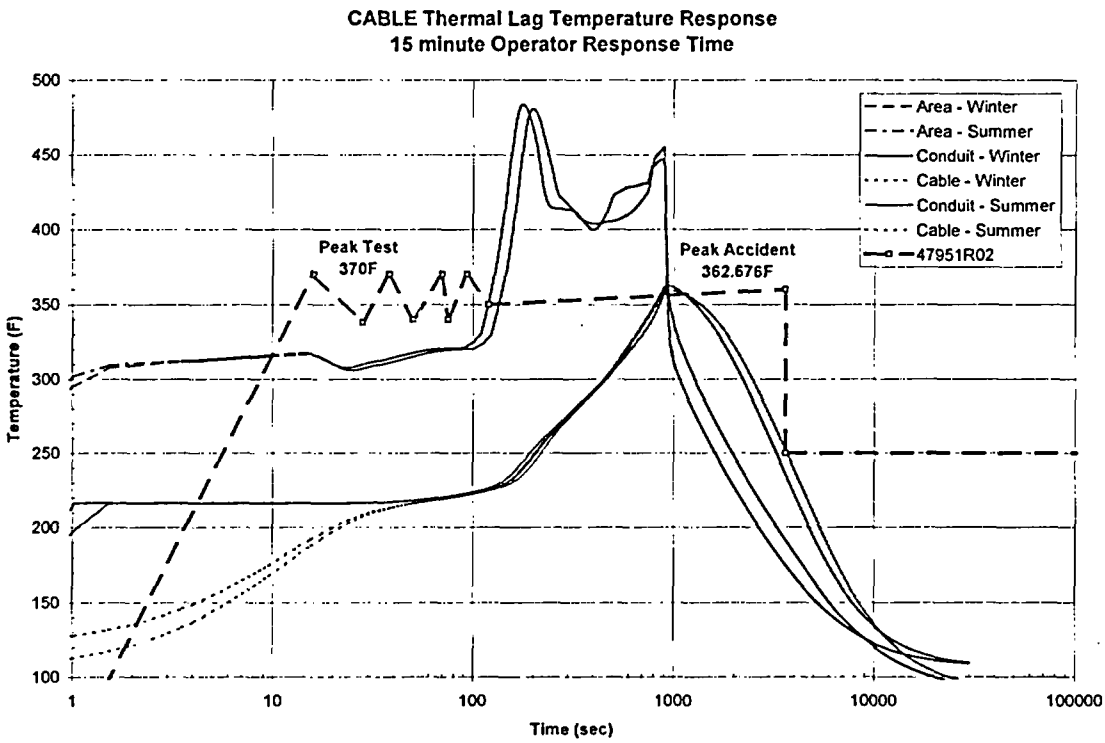
Graph 2 for NL-04-095-GIP-11



Non-Proprietary

Question NL-04-095-GIP-12:

Section 10.10, "Generic Letter 95-07," states that the effect of the SPU on the current pressure locking and thermal binding evaluation was reviewed, and that the SPU does not introduce any increased challenge for thermal binding and/or pressure locking and does not effect the results and conclusions of the current evaluation.

Discuss, with examples, the evaluation of the effect of the SPU on the potential for thermal binding and pressure locking of safety-related POVs, including consideration of increased ambient temperatures in applicable locations.

Response NL-04-095-GIP-12:

Based on recognition of the potential for pressure locking, nineteen motor-operated gate valves were field modified prior to initial startup to eliminate the potential for pressure locking. All of these valves except one have a drilled hole in the valve disc. One valve has an external vent line from the valve bonnet to the high pressure side of the valve.

Results of the screening of safety-related motor-operated gate valves identified 18 motor-operated valves (MOVs) that required a detailed evaluation for susceptibility to pressure locking, and 8 MOVs that required a detailed evaluation for susceptibility to thermal binding. Subsequent to this screening, 5 of the 18 MOVs identified as susceptible to pressure locking were modified to install a hole in the valve disc to eliminate the potential for occurrence of pressure locking.

Screening of gate valves with attached hydraulic / pneumatic actuators identified two air-operated valves (AOVs) potentially susceptible to pressure locking. These valves are parallel-disc gate valves and therefore are not susceptible to thermal binding.

The following is a summary of the current evaluations / key parameters and impact of the SPU on these evaluations / parameters for MOVs and AOVs subject to pressure locking. The evaluations considered two types of pressure locking: hydraulically induced pressure locking (HIPL) and thermally induced pressure locking (TIPL).

1. Pressure locking of RHR Pump Discharge Isolation Valve (MOV):

   a) HIPL: This valve may be required to be opened following transfer from cold leg to hot leg recirculation during a LOCA event. The evaluation considers pressure trapped in the valve bonnet under both small break and large break LOCA conditions. Under SPU conditions, the time interval for transferring from cold leg to hot leg recirculation is being changed from 14 hours to 6.5 hours (Section 6.2). For the small break LOCA case, credit is taken in the evaluation for bonnet depressurization during the time interval between cold leg and hot leg recirculation. The above change in time interval would result in a small differential pressure between the bonnet pressure and the downstream line pressure under SPU conditions, as opposed to zero differential pressure under pre-SPU conditions. However, the large break LOCA case remains bounding in the evaluation, since pressure trapped in the valve bonnet, which is based on shutoff head

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of the RHR pumps, is conservatively assumed not to depressurize, and the upstream and downstream line pressures are conservatively assumed to be zero. The shutoff head of the RHR pumps is not affected by the SPU.

b) TIPL: The valve is located outside Containment in the Pipe Penetration Area. Maximum temperature of this area does not change during a LOCA from the normal maximum ambient temperature. The valve is thermally insulated. During a large break LOCA, the valve is potentially cooled to RWST temperature. Evaluation shows that, for the scenario where there is thermal addition due to increase in the ambient temperature, bonnet pressure decays at a faster rate than it increases by thermal addition, and therefore there is no pressure increase due to thermal addition. The SPU does not affect this evaluation.

2. Pressure locking of PORV Block Valves (MOVs):

a) HIPL: Pressure trapped in valve bonnet is based on pressurizer safety valves relief setpoint, which is not affected by the SPU.

b) TIPL: Valves are not required to open for mitigation of LOCA or HELB in Containment; therefore, an assessment of TIPL under accident conditions is not required under the scope of GL 95-07.

Regarding operation of these valves during low RCS temperatures in conjunction with the Overpressure Protection System: assuming steam was trapped as a result of prior closure to isolate a leaking PORV during power operation, thermally induced pressure locking would not be of concern since the trapped steam would be cooling down. The SPU does not affect this evaluation.

3. Pressure locking of Safety Injection Pump #31 Discharge Isolation Valves (MOVs):

a) HIPL: Pressure trapped in valve bonnet is based on the discharge pressure of the recirculation pumps. This condition is bounded by the conditions evaluated in the TIPL evaluation, discussed below.

b) TIPL: These valves are required to be opened following transfer from cold leg to hot leg recirculation during a LOCA event. The evaluation assumes water trapped in bonnet of the valves heats up from RWST temperature (35°F) to maximum ambient temperature at the location of the valves outside containment (85°F). The maximum pressure in the bonnet of the valves includes the thermally induced pressure from the bonnet fluid temperature change plus pressure trapped in the bonnet based on discharge pressure of the recirculation pumps. Under SPU conditions, the time interval for transferring from cold leg to hot leg recirculation is being changed from 14 hours to 6.5 hours (Section 6.2). However, the evaluation conservatively assumes the valves heat up to the maximum ambient temperature and that there is no depressurization of the pressure trapped in the bonnet during this time interval. Accordingly, the change in the time interval for transferring from cold leg to hot leg recirculation does not affect the evaluation results. Also, the SPU does not impact the temperature parameters used in the evaluation and does not impact recirculation pump head.

Non-Proprietary

4. Pressure locking of Safety Injection Pump #32 Discharge Isolation Valves (MOVs):

   a) HIPL: Pressure trapped in valve bonnet is based on the developed head of the recirculation pumps plus the safety injection pumps. Safety injection pump head is not affected by the SPU. Pump head for the recirculation pumps is not affected by the SPU.

   b) TIPL: The fluid passing through the valves during safety injection is from the RWST, resulting in cooling the valves to RWST temperature (35°F). If the valves are re-opened during recirculation, fluid in the bonnets is presumed to be at ambient temperature (85°F), thus resulting in an increase in bonnet pressure and differential pressure across the disc. The thermally induced pressure locking analysis for these valves shows that the actuators are capable of opening the valves, with margin. The SPU does not affect this evaluation.

5. Pressure locking of Containment Spray Pump Discharge Isolation Valves (MOVs):

   a) HIPL: The only pressure source to pressurize the bonnets of these valves is the containment spray suction supply, which is the head of the RWST plus elevation difference between the tank and the valves. This pressure is bounded by the requirement for the valves to open against full shutoff head of the containment spray pumps. Therefore, HIPL of these valves is ruled out.

   a) TIPL: These valves experience minimal temperature gradients. Sometime after containment spray is initiated, the temperature of these valves may drop due to fluid from the RWST. However, once the valves are closed, there is no requirement to open them. Therefore, TIPL is not a concern for these valves.

6. Pressure locking of Low Head to High Head Recirculation Stop Valves (MOVs):

   a) HIPL: These valves are fitted with Isolation Valve Seal Water System (IVSWS) nitrogen supply to accommodate thermal expansion of water trapped in the bonnet. However, an evaluation was performed to address the scenario of leakage of nitrogen from the IVSWS supply line. In this evaluation, pressure trapped in the bonnet is based on the setpoint of the RHR heat exchanger outlet safety valves, which is not affected by the SPU.

   b) TIPL: These valves are located in the PAB Pipe Penetration Area, where the ambient temperature can be 105°F during normal operation, as well as post-LOCA. The area temperature may rise above this value during a HELB, but these valves are not required to open during an HELB. During the injection phase of a LOCA, RWST water circulating in the line upstream of the valves will tend to cool them, reducing bonnet pressure. As the event continues, gradual reheating results in the bonnet temperature returning to the ambient range until the signal to open. Therefore, TIPL is not a concern for these valves.

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7. Pressure locking of Boron Injection Tank Outlet Isolation Valves (MOVs):

   a) HIPL:

   SI Actuation following a LOCA

   The normal positions of SI-MOV-1835A & B were changed utilizing the 10 CFR 50.59 (Reference 1) process from normally closed to normally open to eliminate the potential for the valves to pressure lock when required to open for this event.

   Post-LOCA Cold Leg and Hot Leg Recirculation Phases

   These valves are maintained open in the post-LOCA cold leg and hot leg recirculation phases, and therefore HIPL is not a concern.

   b) TIPL: As addressed above, these valves are normally open and maintained in the open position post-LOCA, and therefore TIPL is not a concern.

8. Pressure locking of AFW Pump Turbine Steam Supply Isolation Valves (AOVs):

   a) HIPL: If the valves close due to a steam line break in the AFW Pump Room, steam will be trapped in the valve bonnet. However, the plant must be cooled down below 350°F to effect repairs to the line, which would significantly reduce the pressure in the bonnet as the steam condensed to water. This will allow for re-opening the valve. The SPU does not affect this evaluation.

   b) TIPL: If the valves close due to high temperature in the AFW Pump Room due to a fire, and it is desired to open the valves, the valve need only open against the normal differential pressure of main steam on the upstream side and turbine backpressure on the downstream side. This is considered a normal operating requirement for the valve. No thermal addition to pressure in the bonnet will be experienced from external sources, since the process fluid is at a much higher temperature than the maximum ambient room temperature. In addition, procedural guidance specifies equalizing pressure across the valves prior to opening. The SPU does not affect this evaluation.

The following is a summary of the current evaluations and impact of the SPU on these evaluations for MOVs subject to thermal binding (TB). For thermal binding evaluations, the Westinghouse Owners Group has developed additional criteria for determining susceptibility based on temperature change: For flexible wedge gate valves, only temperatures above 200°F, and temperature changes above 100°F are considered significant for thermal binding.

1. Thermal binding of PORV Block Valves:

   Although these flexible wedge gate valves are potentially susceptible to thermal binding, they are considered acceptable in the current condition based on: (1) The maximum differential temperature between closing and subsequent opening these valves would experience is 150°F, which, although it exceeds the 100°F temperature change criteria identified above, is not large, (2) Based on an 18 plant survey, no occurrences of thermal binding of these valves were reported

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over many years of operation, (3) High conductivity of valve materials, and insulation of the valves and adjacent piping, minimize temperature differences which may contribute to thermal binding, (4) the valve body and wedge are both stainless steel having nearly identical thermal expansion coefficients, and (5) Past performance history of these valves during plant cooldowns has been satisfactory. The SPU does not affect this evaluation.

2. Thermal binding of Safety Injection Pump #31 Discharge Isolation Valves:

   These flexible wedge gate valves are potentially susceptible to thermal binding. However, thermal binding is not a concern for these valves based on the following: During closure of these valves after safety injection, the valve temperatures will not exceed 120°F. When the valves are required to open to transfer from cold leg to hot leg recirculation, the valves and the fluid at the valves will be at ambient temperature (maximum of 85°F). The low temperature at closure, and the relatively minor temperature difference between closure and opening are both well within the temperature criteria for thermal binding susceptibility noted above. The SPU does not affect this evaluation.

3. Thermal binding of RHR Heat Exchangers #31 and #32 Outlet Isolation Valves:

   These valves (two per HX) are open and energized during normal power operation, safety injection, low head recirculation, and RHR operation. The valves are closed, post-LOCA, to initiate high head cold leg or hot leg recirculation. If closed during high head cold leg recirculation, operating procedures direct the operator to open one HX pair to establish low head recirculation if RCS pressure decreases sufficiently. However, the valves are not designed with the intent to ensure opening after closure while mitigating an accident. The valves' control mechanism has been modified to control valve closing via geared limit switches to minimize seating forces. The motors are de-energized prior to the valves' discs swinging, thus preventing the valves from being wedged too tightly.

   The worst case accident conditions for thermal binding occurs during a small break LOCA. The valves remain open during the injection phase, but are subsequently closed to support high head cold leg recirculation. The subject valves are not expected to undergo any significant cooling during this event, post closure. In addition, one pair of the valves would be required to be cycled open and closed to determine when RCS pressure has decreased sufficiently to facilitate low head recirculation. Combined, these effects and those discussed above preclude the valves from thermally binding. Furthermore, long-term cooling can be achieved without re-opening these valves, post-LOCA.

   The SPU does not affect the above evaluation.

Reference:

1. 10 CFR 50.59, "Changes, Tests, and Experiments."

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Question NL-04-095-GIP-13:

Section 10.15.4, "Startup Testing," states that power escalation will be controlled by a specific procedure that includes controls for power escalation, hold points, and data collection requirements. Section 10.15.4 also states that a vibration monitoring activity will be initiated to monitor plant response at various power levels.

Discuss the plans for power escalation including specific hold points and duration, inspections, and plant walkdowns. Also, discuss the vibration monitoring activity including data collection methods and locations, baseline vibration measurements, and planned data evaluation.

Response NL-04-095-GIP-13:

This information has been included in Section 10.15.4 of the IP3 SPU LAR.

Question NL-04-095-GIP-14:

Discuss the evaluation of potential flow vibration effects resulting from SPU conditions for reactor pressure vessel internals, and steam and feedwater systems and their associated components, including impact on structural capability and performance during normal operations, anticipated transients (initiation and response), and design-basis conditions; and preparation for responding to the potential occurrence of loose parts as a result of the power uprate.

Response NL-04-095-GIP-14:

- **Reactor Vessel Internals**

Flow induced vibrations (FIV) of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies is to assure the structural integrity and reliability of the reactor internals components. These efforts have included in-plant tests, scale model tests, tests in fabricators' shops, bench tests of components, and various analytical investigations. The results of scale model and in-plant tests indicate that the vibrational behavior of 2-, 3-, and 4-loop plants is essentially similar; the results obtained from each of the tests complement one another and make possible a better understanding of the flow induced vibration phenomena.

As described in References 1 and 2, Westinghouse performed a comprehensive instrumented reactor internals testing program at the Indian Point Unit 2 plant. This test program included heatup and cooldown as well as operation with 1, 2, 3, and 4 reactor coolant pumps, including starting and stopping transient operations. The initial program was performed without the core present (Reference 1). A subsequent program was performed with the core in place (Reference 2). The results of this program were used to develop theories and concepts related to reactor internals vibration under various operating conditions as well as to assess the fatigue and stress effects of operational vibrations. The testing performed at Indian Point 2 included the acquisition of data during hot functional testing (without core present) and subsequently with the core installed. The results of this comprehensive testing program showed that the vibrational response of the reactor internals is small and that adequate margins of safety exist with regard to flow induced vibration.

Non-Proprietary

To address the SPU program at IP3 an evaluation was performed to show that the vibration characteristics of reactor internals do not change significantly and the structural adequacy of the reactor internals in regards to FIV is not impaired.

The reactor internal components that are generally addressed for FIV consists of lower internals (core barrel, thermal shield support flexures, thermal shield support bolts and dowel pins) and upper internals (guide tubes). The current design temperature range between T_{cold} and T_{hot} is 58.6°F and changes to 63.4 with the implementation of SPU at IP3.

This SPU design condition will slightly alter T_{cold} and T_{hot} fluid densities, which will slightly change the forces, induced by flow. The corresponding T_{cold} and T_{hot} fluid densities will increase by about 2%.

Evaluations performed for the SPU conditions show that the FIV loads on the guide tubes and the upper support columns increases by about 6% and the impact on the lower internals is negligible. Benchmark tests of guide tubes and upper support columns together with previous FIV analysis for similar 4-loop reactors has shown that a large margin exists in regards to calculated stresses versus the code allowable. Therefore, the effect on the FIV on the reactor internals is considered negligible or essentially non-existent for the SPU conditions at the IP3 plant.

References:

1. WCAP-7879-P-A, "Four Loop PWR Internals Assurance and Test Program", July 1972.
2. WCAP-7879-AD1, "Four Loop PWR Internals Assurance and Test Program Addendum 1, IPP-2 Reactor Internals Vibration with-Core Testing Program", October 1972.

- **Steam Generator**

Steam generator tube vibration and wear are addressed in Section 5.6.6 of the LAR.

- **Steam and Feedwater Systems and Their Associated Components**

The main steam and feedwater piping systems and their associated components will be evaluated for potential flow vibration effects resulting from SPU conditions. These piping systems will be included in the piping vibration monitoring plan to be performed in support of SPU. The piping vibration monitoring plan will identify the specific piping locations for monitoring, the monitoring methods to be used (e.g. accelerometers, hand held devices), as well as acceptance criteria to determine piping vibration acceptability.

Refer to response for Generic Issues and Programs Question 3 for additional details related to the overall piping vibration monitoring plan

- **Response to the potential occurrence of loose parts as a result of the power uprate.**

Entergy has procedures in place for the control of and exclusion of foreign objects during maintenance activities, including during outages. These procedures have been successful in controlling foreign objects. Entergy has installed metal impact monitors to detect the occurrence of loose parts or foreign objects in the reactor coolant system. Detection of unusual signals

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from the metal impact monitors triggers investigations and evaluations to determine the source of the signals and to take corrective actions if that is needed.

Question NL-04-100-LOC-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 NL-04-100-LOC-3:

The response this question contains proprietary information. The proprietary and non-proprietary versions of the response are provided as response NL-04-100-LOC-3 in Attachments 3 and 4 of this letter, respectively.

Question NL-04-100-LOC-4:

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

Response NL-04-100-LOC-4:

In order to demonstrate stable and sustained quench, the WCOBRA/TRAC calculation from the maximum local oxidation analysis for Indian Point Unit 3 was extended beyond the rod quench time.

Figure 1 shows the peak cladding temperatures for the five rods modeled in WCOBRA/TRAC. This figure indicates that quench occurs at about 260 seconds for the low power rod (Rod 5), about 300-320 seconds for the core average rods (Rod 3 and 4), and about 400 seconds for the hot rod (Rod 1) and hot assembly average rod (Rod 2). Once quench is predicted to occur, the rod temperatures remain steady and slightly above the fluid saturation temperature for the remainder of the simulation.

Figure 2 shows the collapsed liquid level in the four downcomer quadrants and shows that increasing level trend is established at the end of the transient, with the level in each quadrant about 5 feet below the bottom of the cold leg and rising.

Figure 3 shows the collapsed liquid level in the core channels. As seen on Figure 3, a trend of decreasing core collapsed liquid levels is established between 130 and 380 seconds, due to downcomer boiling. During this period, the downcomer collapsed liquid levels also tend to decrease (Figure 2). Later in the transient, a reverse trend of stable and gradual increase of core inventory and downcomer levels is observed, due to the adequate SI injection rate. This is consistent with the expected result based on the removal of the initial core stored energy and the gradual reduction in decay heat.

Figures 4 shows the collapsed liquid level established in the upper plenum. It is evident that liquid pool is established in the upper plenum and maintained until the end of the transient, with the level approaching the bottom of the hot legs.

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Figure 5 shows the vessel liquid mass and indicates an increasing trend beginning at about 340 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 3 Large Break LOCA analysis.

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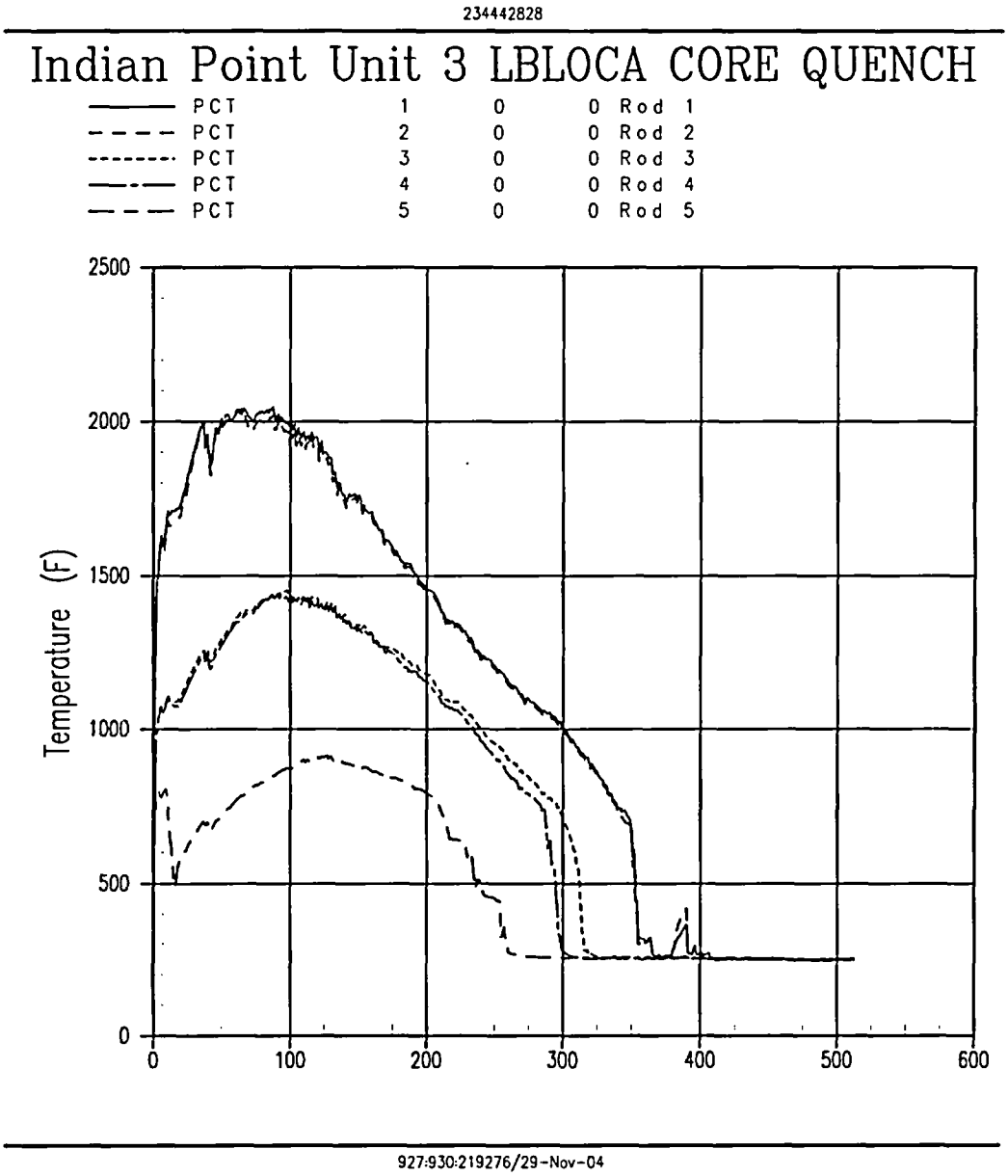

Figure 1 - Peak Cladding Temperatures

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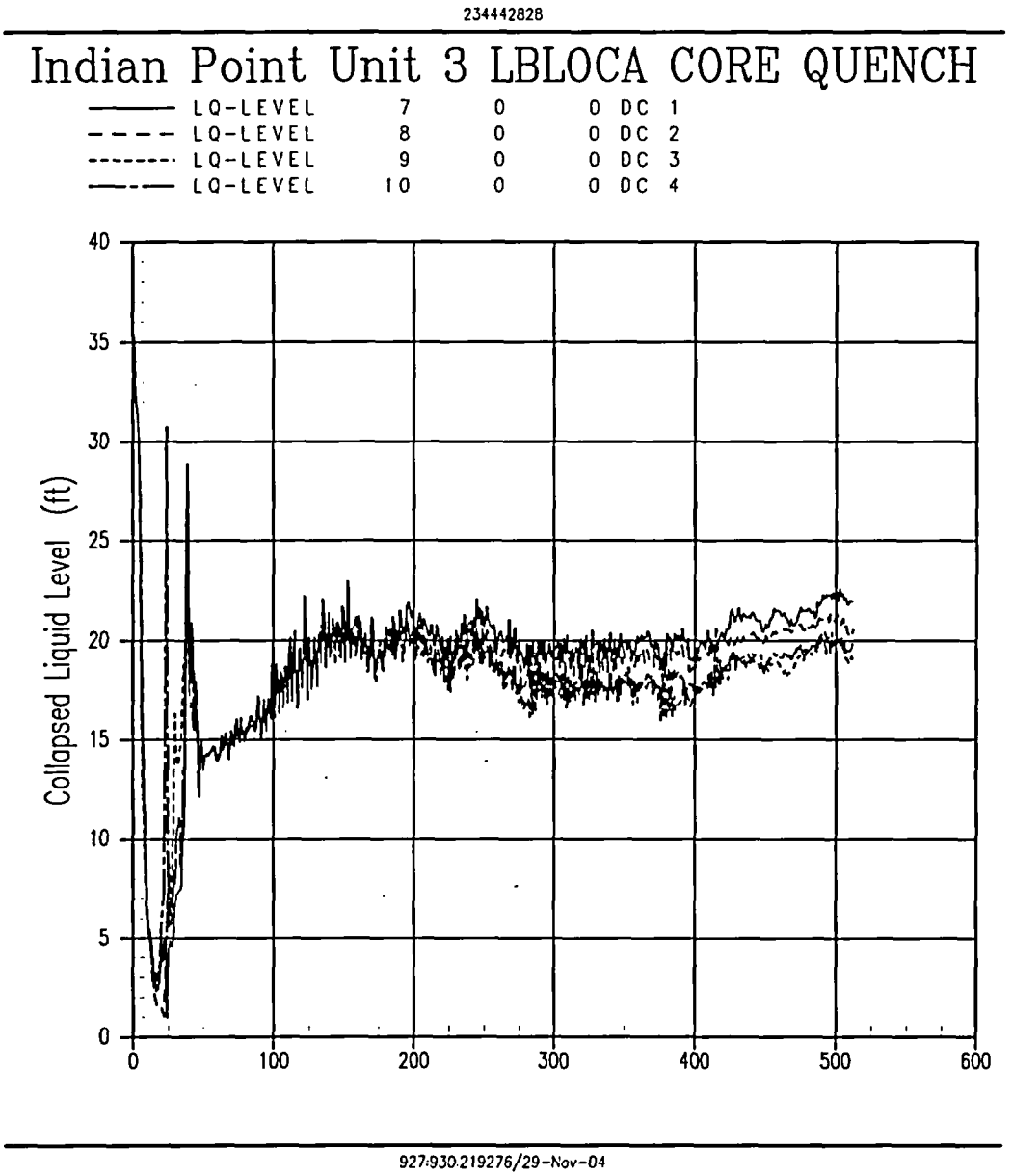

Figure 2 - Downcomer Collapsed Liquid Levels

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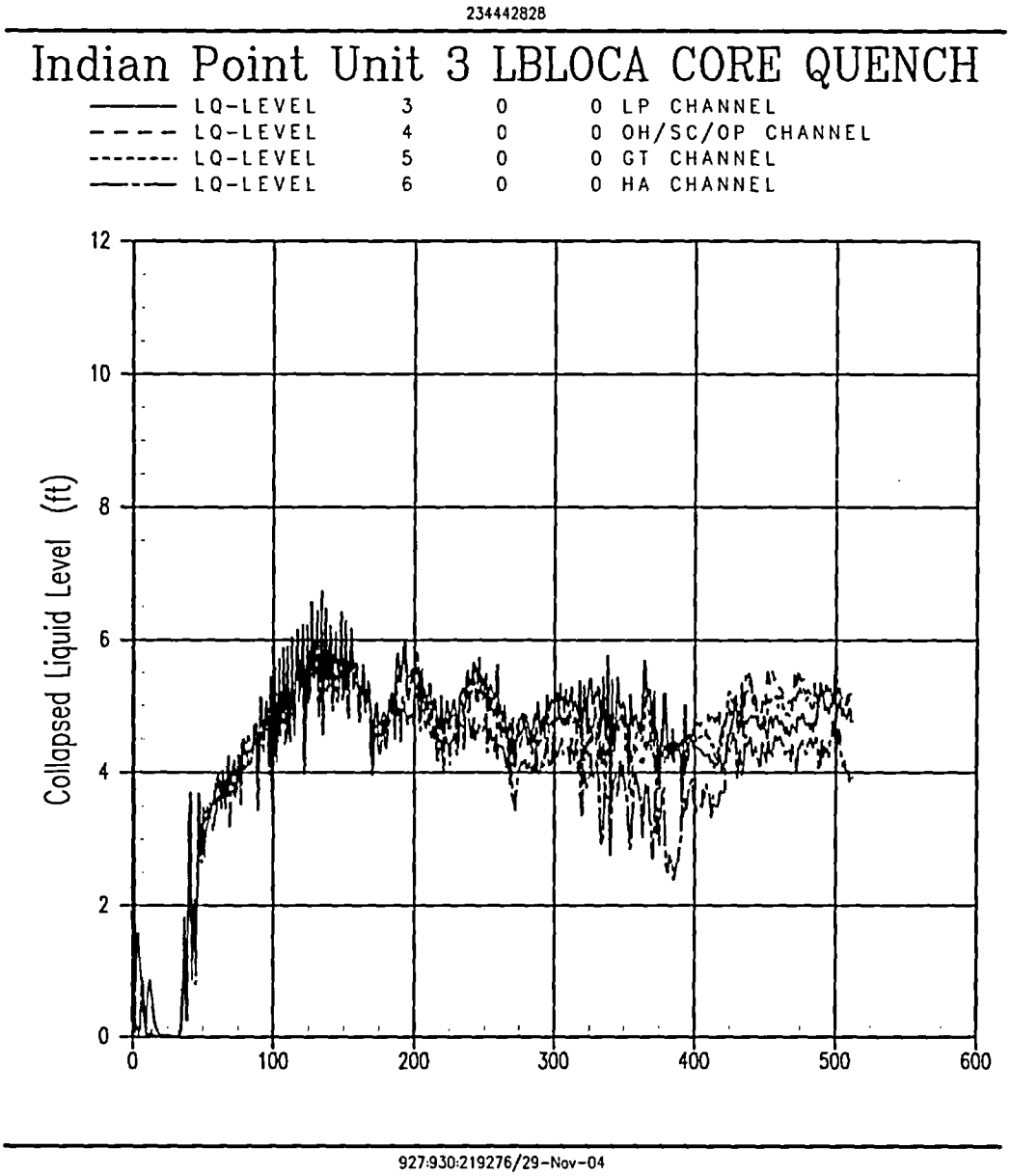

Figure 3 - Core Collapsed Liquid Levels

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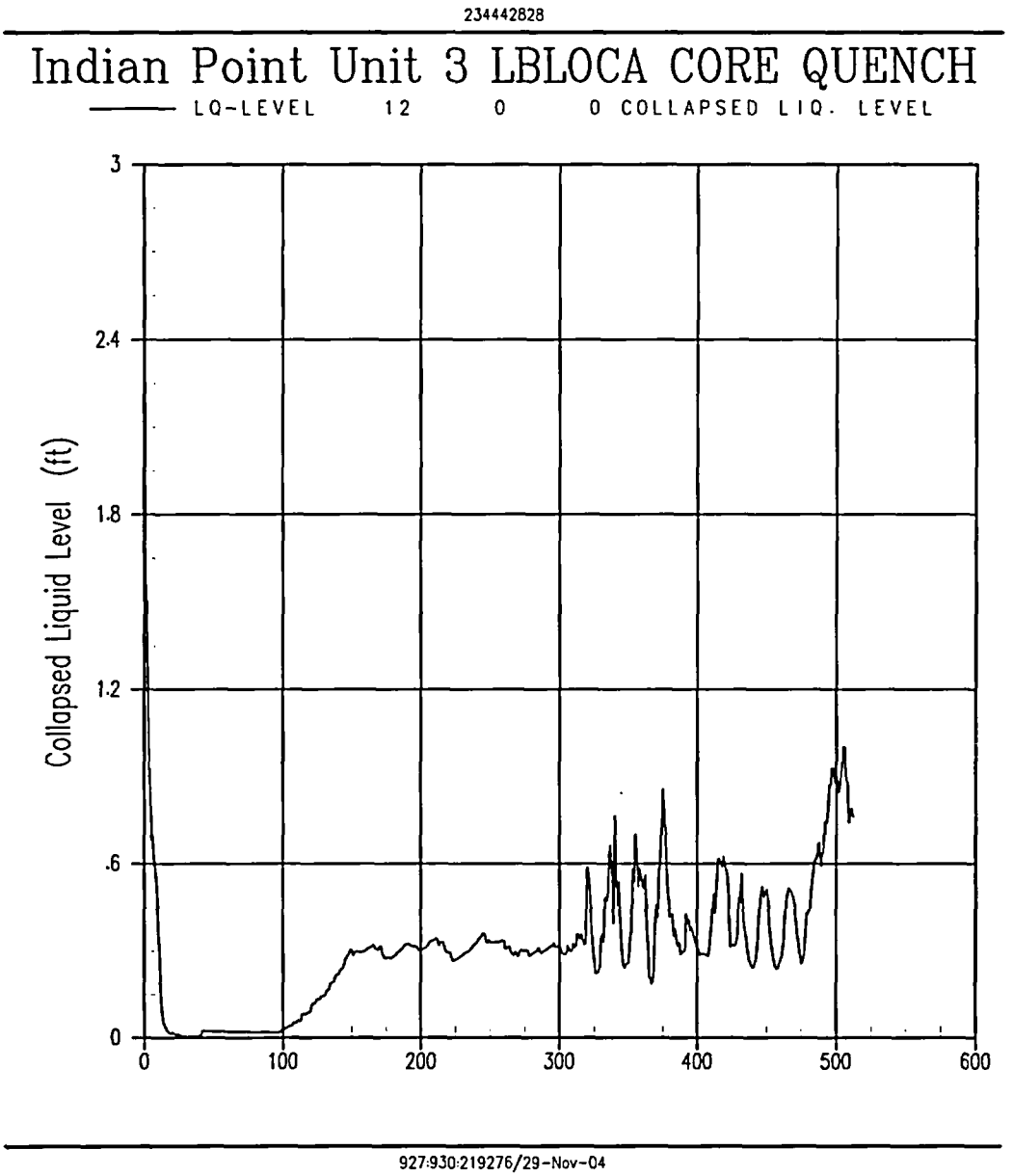

Figure 4 – Upper Plenum Collapsed Liquid Level (All channels)

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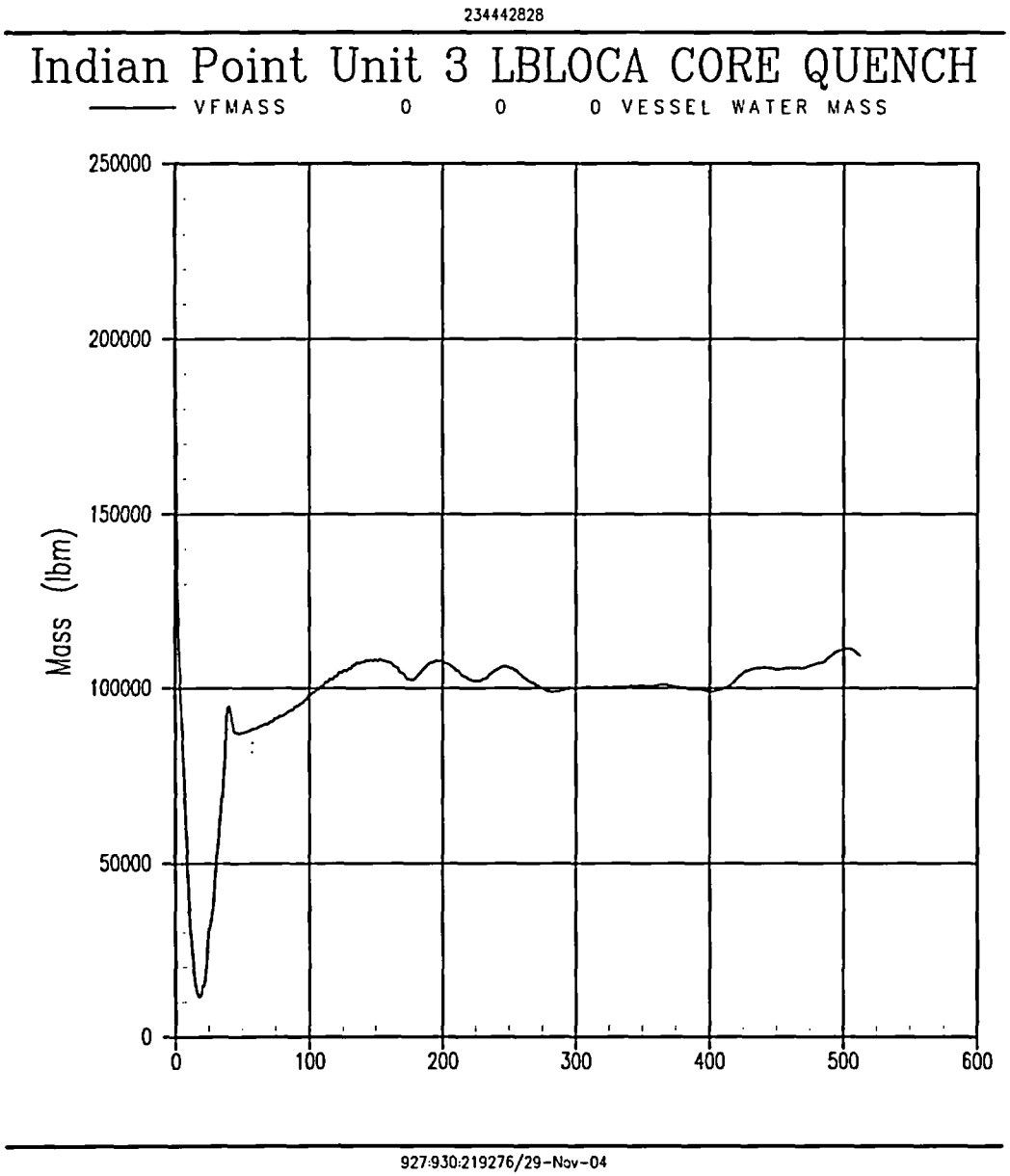

Figure 5 - Vessel Liquid Mass

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Question NL-04-100-LOC-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.

Response NL-04-100-LOC-5:

The results (peak clad temperature, maximum local oxidation and total hydrogen generation) for the Indian Point Unit 3 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 3 (IP3) large break LOCA analysis of record is 7.6 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. 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.

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 expected to be well below this value.

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Based on the above discussion, the transient oxidation decreases from a very conservative maximum of 7.6% 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 <16%. Additional WCOBRA/TRAC and HOTSPOT calculations were performed at an intermediate burnup, 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 16% 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 IP3 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 IP3 SPU program, a new SBLOCA analysis was performed. The break spectrum that was analyzed yielded a maximum peak clad temperature of 1543 °F for a 3 inch equivalent break diameter. The break spectrum results are summarized in Tables 6.2-2 and 6.2-3 of Reference 1. Because of the low clad temperatures, fuel rod burst was not predicted to occur, and the maximum transient oxidation was only 1.04%. 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 IP3 conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

References

1. WCAP-16212-P, "Indian Point Nuclear Generating Unit No. 3, Stretch Power Uprate NSSS and BOP Licensing Report," J. R. Stukus, et al., June 2004.

Table LOC-5-1 IP3 DESIGN BASIS ANALYSIS LOCA RESULTS		
	LBLOCA	**SBLOCA**
Peak Clad Temperature	1944°F (PCT95%)	1543°F
Maximum Local Oxidation	Pre-transient = 0% Transient = <7.6%	Pre-transient = 0% Transient = 1.04%
Total Hydrogen Generation	0.620%	<< 1%

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

Question NL-04-100-PVM-4d-1:

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

Response Question NL-04-100-PVM-4d-1:

As noted in NL-04-145 response to NL-04-073-PVM-4a, the revised calculations for the pressurizer nozzles demonstrate that the postulated flaw size meets the requirements of Appendix G (1/4t or 1 inch). Therefore this RAI is not applicable.

Question NL-04-100-PVM-4d-2:

Was the technique equivalent to VIP-108?

Non-Proprietary

Response NL-04-100-PVM-4d-2:

As noted in NL-04-145 response to NL-04-073-PVM-4a, the revised calculations for the pressurizer nozzles demonstrate that the postulated flaw size meets the requirements of Appendix G (1/4t or 1 inch). Therefore this RAI is not applicable.

Question NL-04-100-PVM-4d-3:

What was the size of the largest flaw?

Response NL-04-100-PVM-4d-3::

As noted in NL-04-145 response to NL-04-073-PVM-4a, the revised calculations for the pressurizer nozzles demonstrate that the postulated flaw size meets the requirements of Appendix G (1/4t or 1 inch). Therefore this RAI is not applicable.

Question NL-04-121-NRC Item 2:

The Entergy response for Piping and Supports Question 1, in letter NL-04-095 dated August 3, 2004, provides a stress summary table for main steam piping. Please provide similar quantitative results for evaluations performed for other balance-of-plant (BOP) piping systems.

Response NL-04-121- NRC Item 2:

Stress summary tables for the other critical balance of plant (BOP) piping systems have been included in the response to PS-1.

Question NL-04-121-NRC Item 7:

During a CVCS malfunction to induce a boron dilution transient, Entergy chose to use a mixing volume that is equal to the RHR and RCS volumes. This appears to be non-conservative. The staff feels that the transient involves, conservatively, only diluted water from the primary water storage tank is injected into the cold leg through the charging lines at maximum letdown rate. This flow would then only mix with the volume of water in the cold leg and downcomer and lower plenum provided the RCPs were on. If they are not on, then there is less justification for mixing and it may be a dilute slug entering the core to cause a local power spike. The staff questions why the licensee is assuming RHR and RCS volume as the mixing volumes.

Response NL-04-121-NRC Item 7:

The CVCS malfunction event is discussed in WCAP-16212, Licensing Report Section 6.3.5. The question is best addressed by plant mode and the operation of the Reactor Coolant Pumps and the RHR System.

Modes 1, 2, 3: One or more Reactor Coolant Pumps are in service and thus adequate mixing is assured.

Non-Proprietary

Modes 4 and 5: At least one Reactor Coolant Pump is in service on shutdowns until Reactor Coolant System temperature is less than approximately 170°F. The RHR System is placed in service when the Reactor Coolant System temperature is less than approximately 350°F thus assuring adequate mixing. Similarly, during startup, the RHR System is in service and a Reactor Coolant Pump is placed in service while Reactor Coolant System temperature is less than 200°F. In addition, the Westinghouse Interim Operating Procedure was developed specifically for these modes, addressing the potential effects of a "dilution front" and a limited active mixing volume, and has been incorporated in plant procedures.

In addition, for modes 4 and 5, at the pressures in the Reactor Coolant System associated with RHR operation (less than 450 psig) letdown flow is limited to 120 gpm. Second, only two charging pumps (90 gpm each) are permitted to be available due to low temperature over pressurization restrictions.

Mode 6: At least one RHR pump (providing a minimum flow rate of 1000 gpm) is in service except during short periods. This flow rate is considered adequate for mixing in the lower plenum. The actual flow from one RHR pump would be much higher than 1000 gpm. While the CVCS Malfunction event has been analyzed in the refueling mode, it is administratively precluded. Plant procedures require that the valve in the boron addition/dilution path be placed in manual and closed upon shutting down the last Reactor Coolant Pump. Thus in Mode 6 (Refueling), plant procedures preclude a dilution event.

Based on the above, Entergy concludes that adequate mixing for the active RCS volumes is available or that administrative controls preclude boron dilution.

The time to reach criticality for the CVCS malfunction event, Modes 1, 2 and 6, is calculated based on the following equation.

Cb(t) = Cbi * e ^ [-(mdil / M) * t]
Where:
Cb(t) = boron concentration of the system as a function of time
Cbi = initial boron concentration of the system
mdil = mass flow rate of diluent
M = initial mass of the system
t = time
In using this equation, it is assumed that the system has a constant mass and that the concentration of the diluent is equal to zero.

Question NL-04-121-NRC Item 8:

Section 5.10.4 of the Stretch Power Uprate Licensing Report (WCAP-16212) provides an estimated increase in PWSCC susceptibility of 22 percent for the reactor pressure vessel head penetrations as a result of the stretch power uprate. An increase of greater than 20 percent is considered by the NRC staff to be significant. Please provide additional information regarding the estimated increase in PWSCC susceptibility and is there a plan for RPV head replacement.

Also, Section 5.10.4 of the Stretch Power Uprate Licensing Report provides an estimated increase in PWSCC susceptibility of 9 percent for the RV hot leg nozzle weld as a result of SPU. How will the 9 percent increase be accommodated in the future?

Non-Proprietary

Response NL-04-121-NRC Item 8:

The approach used in Section 5.10.4, was to estimate a relative effect of PWSCC susceptibility by estimating the temperature change in the upper head region based on a conservatively wide range of operating temperatures that correspond to a full-power programmed T_{avg} range from 549°F to 572°F. The resulting temperature increase of 5.3°F was evaluated using the crack initiation probability methodology described in Reference 2 of Section 5.10.

In practice, Entergy is required to establish RPV head inspection requirements in accordance with NRC Order EA-03-009. The Order provides for a time-at-temperature methodology to determine the effective degradation years (EDY) value that is used to determine the inspection category. Based on the current plant operating history and cycle-specific temperature data, the projected EDY value increase is 11.8%. As required by the NRC Order, Entergy will recalculate the EDY value to establish the inspection requirements for each refueling outage using plant data for each operating cycle.

Entergy is assessing options to mitigate the effects of PWSCC on continued plant operation. One possible option involves a modification that would result in reduction of the upper head temperature. Entergy is also assessing eventual replacement of the reactor vessel head.

A similar assessment of PWSCC susceptibility for the RCS hot leg nozzle welds was performed. Although the NRC Order does not establish EDY categories and inspection requirements for these locations, Entergy is required to inspect these areas in accordance with ASME Section XI and the IP3 Inservice Inspection Program. Also, Entergy is participating in industry programs that monitor operating experience and develop recommendations, including augmented inspections. MRP has recently issued recommendations (MRP 2003-039, dated January 20, 2004) that include visual inspection of the Alloy 600 (vessel head-to-pipe) welds within the next two refueling outages.

ATTACHMENT 4 TO NL-04-155

ADDITIONAL INFORMATION FOR IP3 SPU LICENSE AMENDMENT REQUEST

BASED ON NRC RAIs ISSUED FOR IP2 SPU

Non-Proprietary version of responses containing proprietary information (from Westinghouse transmittal PU3-W-04-161)

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

Question NL-04-100-LOC-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 NL-04-100-LOC-3:

Break location, type and size are specifically considered for the IP3 LBLOCA transient simulations. This analysis concluded that the cold leg guillotine break is limiting for IP3. The uncertainties related to break location, type and size were included in the model uncertainties for the IP3 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 1). 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 2). 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 3. Reference 3 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 3 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 4 including a review of pertinent experimental data.

a, c

From References 3 and 4 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 3 plant safety.

References

1. WCAP-11372-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code", S. D. Rupprecht, et al., 1986.
2. WCAP-10081-NP Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and Improved Condensation Model", C. M. Thompson, et al., October, 1995.
3. OG-87-37, "Westinghouse Owners Group (WOG) Post LOCA Long Term Cooling, Letter from Roger Newton (WOG) to Thomas Murley (NRC)", August 26, 1987.
4. 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.

ENCLOSURE A TO NL-04-155

Westinghouse authorization letter dated December 8, 2004 (CAW-04-1927), with the accompanying affidavit, Proprietary Information Notice, and Copyright Notice

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

Westinghouse Proprietary Class ~~2~~ 3

Westinghouse

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Our ref: CAW-04-1927

December 9, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse IP3 SPU Application (WCAP-16212-P) Responses to IP2 RAIs Listed as "Later" in NL-04-145, December 9, 2004

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1927 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-04-1927, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Enclosures

cc: B. Benney
L. Feizollahi

CAW-04-1927

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed
before me this 9th day
of December, 2004

Sharon L. Fiori
Notary Public

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

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

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 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.

(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 "Westinghouse IP3 SPU Application (WCAP-16212-P) Responses to IP2 RAIs listed as "later" in NL-04-145, December 9, 2004" (Proprietary), 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. 3 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).

COPYRIGHT NOTICE

Westinghouse

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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Our ref: CAW-04-1923 Rev. 1

November 17, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Transmittal PU3-W-04-153 (INT-04-203), Indian Point Nuclear Generating Unit No. 3 Stretch Power Uprate Project, Westinghouse Responses to RAIs, November 16, 2004.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1923 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-04-1923, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Enclosures

cc: W. Macon
E. Peyton

CAW-04-1923

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed
before me this 16th day
of November, 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 an 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.

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(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.

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

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(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 PU3-W-04-153, "Indian Point Nuclear Generating Unit No. 3 Stretch Power Uprate Westinghouse Responses to RAIs" (Proprietary) dated November 16, 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. 3 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:

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Further this information has substantial commercial value as follows:

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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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