Entergy

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Re: Indian Point Unit Nos. 1 and 2
Docket Nos. 50-003 and 50-247
NL-05-068

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

Subject: **10 CFR 50.59 (d) Report for Indian Point Unit Nos. 1 and 2**

Dear Sir:

Pursuant to 10 CFR 50.59 (d)(2), enclosed please find a summary report (Attachment 1) of the changes, tests and experiments implemented at Indian Point Units 1 and 2 between November 28, 2002 and November 23, 2004, or utilized in support of the UFSAR update. The summaries of Safety Evaluations (SEs) and 50.59 Evaluations (REs) set forth in the report represent the changes in the facilities, changes in procedures and tests and experiments implemented pursuant to 10 CFR 50.59. Attachment 2 provides a summary of these evaluations implemented for the period defined above.

There are no new commitments made by Entergy contained in this letter. If you have any questions, please contact me at (914) 734-6668.

Very truly yours,

Patric W. Conroy

Patric W. Conroy
Manager, Licensing
Indian Point Energy Center

Attachment 1 (50.59 Report Listing)
Attachment 2 (50.59 Summary of Changes, Tests and Experiments)

cc: see next page

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cc: Mr. Patrick D. Milano, Senior Project Manager
Project Directorate I
Division of Licensing Project Management
U.S. Nuclear Regulatory Commission

Mr. Samuel J. Collins
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission

Resident Inspector's Office
Indian Point Unit 2 Nuclear Power Plant
U.S. Nuclear Regulatory Commission

Mr. Paul Eddy
New York State Dept. of Public Service

ATTACHMENT 1 TO NL-05-068

50.59 REPORT LISTING

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 1 AND 2
DOCKET NOS. 50-003 AND 50-247

50.59 Evaluation Number	Rev. No.	Units 1 and 2 – 2005 Report 50.59 EVALUATION TITLE
96-67-MD	0	IP2 Fan Cooler Unit (FCU) Charcoal Filter Removal
96-200-MD	0	IP-Elimination Spray Additive System
96-200-MD	1	IP-Elimination Spray Additive System
02-0344-PR-02-RE	2	Increase T_{avg} from 559°F to 562°F
04-0732-MD-00-RE	0	Installation of a Temporary Roll-Up Door on the Containment Equipment Hatch
04-0732-MD-00-RE	1	Installation of a Temporary Roll-Up Door on the Containment Equipment Hatch
04-0918-PR-00-RE	0	Use of Predicted temperature from Westinghouse Analysis CN-CRA-00-087 instead of saturation temperature for peak containment LOCA temperature
04-1269-MD-00-RE	0	Develop New Fuel Design – Westinghouse 15x15 Upgraded Fuel Design
04-1311-MD-00-RE	1	IP2 Cycle 17 Reload Core Design Change
04-1649-EV-00-RE	0	To Allow Operation with a Maximum Increase in T_{avg} of 3°F, i.e., from 562°F to 565°F
04-1649-EV-00-RE	1	To Allow Operation with a Maximum Increase in T_{avg} of 3°F, i.e., from 562°F to 565°F
05-0326-MM-00-RE	0	Install Isolation Valve and Associated Fill Valve in ¾" SI-1501R Line # 31

ATTACHMENT 2 TO NL-05-068

50.59 SUMMARY OF CHANGES, TESTS AND EXPERIMENTS

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 1 AND 2
DOCKET NOS. 50-003 AND 50-247

50.59 Summary of
Changes, Tests, and Experiments

50.59 Evaluation Number	Rev. No.	TITLE
96-67-MD	0	IP2 Fan Cooler Unit (FCU) Charcoal Filter Removal

This evaluation is for removing the absolute (HEPA) filters, the charcoal filters and associated fire protection and detection equipment from each of the five containment fan cooler units. A study has been performed demonstrating the adequacy of the design basis post accident dose using the revised source term methodology of NUREG 1465. The results of that study demonstrate that off-site and Control Room dose limits are maintained within all regulatory limits without reliance on the HEPA and charcoal filters currently associated with the containment Fan Cooler System. This study has been documented in WCAP-14542, "Evaluation of the Radiological Consequences from a Loss of Coolant Accident at Indian Point Nuclear Generating Station Unit No. 2 Using NUREG-1465 Source Term Methodology".

96-200-MD	0	IP-Elimination Spray Additive System

This evaluation is for the removal of the Spray Additive System and the installation of a passive basket system. When the Spray Additive System is eliminated, the pH must be adjusted by alternate means to assure that the Iodine removed by the spray is retained in solution. Adjusting the pH will be accomplished using Trisodium Phosphate contained in a passive basket system installed in the post Loss of Coolant Accident (LOCA) flooded region of the Containment.

96-200-MD	1	IP-Elimination Spray Additive System

Revision 1 of this evaluation provides clarification that maximum pH will be 9.0 assuming all Trisodium Phosphate (TSP) is dissolved. Additionally, corrosion considerations for NSSS EQ and non-NSSS EQ equipment were evaluated to be insignificant. Consideration to prevent inadvertent loss of TSP is also addressed to include revisions to the Containment Close-Up Procedure. Also, consideration is provided to revise the Emergency Operating Procedure to eliminate those steps no longer applicable to the equipment retired and removed.

02-0344-PR-02-RE	2	Increase T_{avg} from 559°F to 562°F

This evaluation is to increase Tavg from 559 deg F to 562 deg F. An increase in the nominal operating Tavg will result in an improved thermal efficiency of the turbine and an increase in the Mwe generation capability of the unit. The major Revision 2 change is to maintain T' (loop specific indicated Tavg at rated thermal power) at 569 deg F in the OTDT reactor trip.

04-0732-MD-00-RE	0	Installation of a Temporary Roll-Up Door on the Containment Equipment Hatch

Revision 0 of this evaluation provided justification to install a manually operated roll-up door on the Vapor Containment (VC) equipment hatch for use during outages. The evaluation confirmed that the door may be installed and used in accordance with the requirements of the plant's technical specifications and technical requirements manual provided that the door can only be used with either (1) reactor cavity water level at least 14 feet above the reactor vessel flange, or (2) the core configuration consists of at least 72 unirradiated fuel assemblies (reload core) with the Reactor Coolant System elevation greater than 66 feet.

50.59 Summary of
Changes, Tests, and Experiments

50.59 Evaluation Number	Rev. No.	TITLE
04-0732-MD-00-RE	1	Installation of a Temporary Roll-Up Door on the Containment Equipment Hatch

Revision 1 of this evaluation provides clarification that the manually operated roll-up door on the Vapor Containment (VC) equipment hatch is sufficient to provide containment closure requirements for reactor vessel head and upper internals transport. Revision 0 of this evaluation provided justification to install a manually operated roll-up door on the Vapor Containment (VC) equipment hatch for use during outages. The evaluation confirmed that the door may be installed and used in accordance with the requirements of the plant's technical specifications and technical requirements manual provided that the door can only be used with either (1) reactor cavity water level at least 14 feet above the reactor vessel flange, or (2) the core configuration consists of at least 72 unirradiated fuel assemblies (reload core) with the Reactor Coolant System elevation greater than 66 feet.

04-0918-PR-00-RE	0	Use of Predicted temperature from Westinghouse Analysis CN-CRA-00-087 instead of saturation temperature for peak containment LOCA temperature

This evaluation is for the change to use the predicted temperature (plus margin as required) in lieu of saturation temperature for a Loss Of Coolant Accident (LOCA) in the containment and confirms that the environmental qualification (EQ) equipment in this area will perform its accident mitigation functions in the resultant harsh environment. The predicted temperature is based on Westinghouse Containment Response Analysis, which is the Analysis of Record for Indian Point 2 (UFSAR Section 14.3.5.1.1). The revised curves remain below the currently qualified temperature of 287 deg. F, and will not prevent any EQ equipment from performing its designated function during a design basis LOCA in the containment.

04-1269-MD-00-RE	0	Develop New Fuel Design – Westinghouse 15x15 Upgraded Fuel Design

This evaluation is for the fuel design change for IP2 as documented by Westinghouse in 50.59 evaluation number EVAL-03-129. This document has been reviewed and determined to be valid for IP2. This evaluation was based on the analysis done in WCAP-16156-P, "Indian Point 2 Stretch Power Uprate NSSS Engineering Report". Submittal to the NRC for approval of this new fuel design is not required based on the Westinghouse Fuel Criteria Evaluation Process (FCEP) which is detailed in WCAP-12488-A. This process requires only notification of the NRC by Westinghouse which was done in letter LTR-NRC-04-8 dated February 6, 2004.

04-1311-MD-00-RE	1	IP2 Cycle 17 Reload Core Design Change

This evaluation is for the Indian Point Unit 2 (IP2) Cycle 17 reload core design change. The IP2 Cycle 17 reactor core is comprised of 120 VANTAGE+ fuel assemblies and 73 15 x 15 Upgrade fuel assemblies. The Cycle 17 core loading configuration features a low leakage pattern. During Cycle 16/17 refueling, 81 fuel assemblies will be discharged and replaced with 41 fresh Region 19A assemblies (4.00 w/o U-235), 32 fresh Region 19B assemblies (4.40 w/o U-235) and 8 twice burned reinsert assemblies from Region 16B. Fuel assembly S-45 in core location B-13 has been reconstituted with two stainless steel filler rods in locations A-15 and K-15. Fuel

50.59 Summary of
Changes, Tests, and Experiments

50.59 Evaluation Number	Rev. No.	TITLE

assembly S-48 in core location N-14 has been reconstituted with a stainless steel filler rod in location G-14. It should be noted that the Region 16 and 17 assemblies contain 6-inch axial blankets enriched to 2.600 w/o U-235, and the Region 18 and 19 assemblies contain 8-inch axial blankets enriched to 3.200 w/o U-235. Cycle 17 will be operated with fuel assembly thimble plugging devices inserted in locations that do not have a rod cluster control assembly (RCCA), secondary source or wet annular burnable poison rod assembly (WABA) component.

04-1649-EV-00-RE	0	To Allow Operation with a Maximum Increase in T_{avg} of 3°F, i.e., from 562°F to 565°F

Following Stretch Power Uprate, turbine inlet pressure may be lower than predicted. This evaluation is to allow operation with a maximum increase in Tavg of 3 deg F from 562 deg F to 565 deg F. While this evaluation justifies a maximum increase in Tavg of 3 deg F, Tavg may only need to be increased by 1 deg F or 2 deg F as necessary. The increase in the nominal operating Tavg will result in an increase in turbine inlet pressure resulting in an improved thermal efficiency of the turbine and an increase in the Mwe generation capability of the unit.

04-1649-EV-00-RE	1	To Allow Operation with a Maximum Increase in T_{avg} of 3°F, i.e., from 562°F to 565°F

Revision 1 of the evaluation identifies limitations associated with a temporary Tavg increase to 565 deg F prior to full implementation such as limited duration for data gathering, setpoint changes, and procedures all to ensure the plant design basis is not compromised. It also clarifies potential non-conservative effects resulting from Overtemperature-Overpower delta-T recalibration. There is an interval between the time that OPDT and OTDT are readjusted and the time that Tavg is actually increased. During this period, the OT/OPDT setpoints will be non conservative for K2[T – T'] and K6[T – T"] terms, however a 2.5 deg F setpoint increase is acceptable.

05-0326-MM-00-RE	0	Install Isolation Valve and Associated Fill Valve in ¾" SI-1501R Line # 31

This evaluation is for the installation of modification ER-05-2-030, Revision 0, "Install Isolation Valve on ¾"-SI-1501R Line #31 Downstream of valves 839H & 839G" to include a review of the loss of Line #31 testing functions on SI check valve leakage when the isolation valve is closed, and the impact on the safety analysis of a ¾" line break, should a Safety Injection (SI) Actuation occur during the installation of the ¾" valve after cutting Line #31, but prior to complete installation. Potential failures have been analyzed to show that the Emergency Core Cooling System (ECCS) Systems are capable of performing their intended design functions during modification installation.