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October 6, 2005

Re: Indian Point Unit No. 3
Docket No. 50-286
NL-05-109

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 No. 3**

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 Unit 3 between January 1, 2004 and April 7, 2005, 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)

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cc: see next page

cc: Mr. John P. Boska, Senior Project Manager, Section 1
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U.S. Nuclear Regulatory Commission

Mr. Samuel J. Collins
Regional Administrator, Region 1
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Resident Inspector's Office
Indian Point Unit 3 Nuclear Power Plant
U.S. Nuclear Regulatory Commission

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

ATTACHMENT 1 TO NL-05-109

50.59 REPORT LISTING

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

50.59 REPORT LISTING

50.59 Evaluation Number	Rev. No.	Unit 3 – 2005 Report 50.59 EVALUATION TITLE
96-3-126-SWS	0	Hydrogen Cooler Service Water Vent Piping
97-3-325-RCS	0	Steam Generator Tube Rupture Re-Evaluation
EVAL-99-3-063-NIS	0	Defeat Overpower Delta T (OPΔT) and Over Temperature Delta T (OTΔT) Turbine Runbacks
00-3-010	1	RCS Vacuum Refill
00-3-092	2	Mansell Level Monitoring System MLMS
01-3-005-RCS	0	Gas Injection to Assist RCS Draindown
01-3-022	0	Instrument Bus 34 Inverter Replacement
01-3-080	0	Control Room Kitchen Upgrade
EVL-02-3-070-HC	0	Allowance for Additional Aluminum in Containment
EVL-02-3-115-SG	0	Use of Nozzle Dams in the Indian Point 3 Steam Generator Lines
EVL-02-3-123-MS	0	Evaluation of Steam Break Outside Containment for EQ Purposes
EVL-03-3-011-RCS	0	Reactor Vessel Head Stud Detensioning and Removal, Installation and Tensioning
EVL-03-3-013-RCS	0	Thermal Controls to Support Rapid Core Offload
EVL-03-3-027-MTG	0	Post Trip Turbine Overspeed Test
EVL-03-3-095-CM	0	Operation of the Fuel Storage Building Overhead Crane with Ventilation Inoperable and in Consideration of Heavy Load Handling
04-0561-TM-00-RE	0	Evaluation of EDG Operability During Flange Weld Repairs of EDG Discharge Header Valve SWN-55
04-1066-MD-00-RE	0	Fuel Design Change to Improve Grid-to-Rod Fuel Fretting Margin
05-299-MD-00-RE	0	IP3 Cycle 14 Core Reload Design
05-364-MM-00-RE	0	Install Isolation Valve and Associated Fill Valve in ¾"-SI-1501R Line #31

ATTACHMENT 2 TO NL-05-109

50.59 SUMMARY OF CHANGES, TESTS AND EXPERIMENTS

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

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Changes, Tests, and Experiments

50.59 Evaluation Number	Rev. No.	TITLE

96-3-126-SWS	0	Hydrogen Cooler Service Water Vent Piping

This evaluation is for the installation of the piping and valves which would allow the venting of Service Water from the main turbine generator hydrogen coolers from an accessible location on elevation 15' of the Turbine Building. This vent is located below the 53' floor elevation and is difficult to access. The piping material for this modification will comply with the requirements of Specification No. TS-MS-027, Rev. 0. The design of the piping supports has been analyzed in accordance with ANSI B31.1 for deadweight loads. All supports are adequately designed in accordance with the applicable codes and standards.

97-3-325-RCS	0	Steam Generator Tube Rupture Re-Evaluation

An evaluation was performed to bound up to 60 minutes to terminate the primary to secondary break flow for a Steam Generator Tube Rupture (SGTR). Westinghouse has determined, based on a plant specific evaluation, that inclusion of charging flow, along with the increased operator action time for break flow termination, will affect the total integrated break flow during the event, and the resultant offsite dose levels. However, sufficient margin exists to the regulatory acceptance criteria for the radiological consequences of a SGTR event such that the small expected dose increase will easily be accommodated by regulatory limits. An assessment in the form of a setpoint relaxation study allows revision of the affected Emergency Operating Procedures (EOPs) without formally changing the Indian Point 3 (IP3) design basis.

EVAL-99-3-063-NIS	0	Defeat Overpower Delta T (OPΔT) and Over Temperature Delta T (OTΔT) Turbine Runbacks

This evaluation was for the permanent removal of the Overpower Delta Temperature (OPΔT) and Overtemperature Delta Temperature (OTΔT) runback by lifting wires on the associated relays in the Central Control Room (CCR) rack G1. The existing feature of blocking "Auto rod withdrawal" will be maintained. New components/cables will not be added. All installation work is in rack G1 in the CCR. Westinghouse has performed analysis for the condition of defeat of runback. This analysis has concluded that sufficient margin to safety (i.e., DNB>1.3) will be maintained. The accident analysis demonstrates that consequences of a dropped rod accident will be within the margin to safety (DNB>1.3) for this accident even without a turbine runback. Credit is not taken in the FSAR for turbine runback.

00-3-010	1	RCS Vacuum Refill

Revision 1 of this evaluation added clarification as to how Reactor Coolant System (RCS) level indication, during mid-loop operation, using ultrasonic devices will be qualified. Reference to ASME Section XI has been deleted because this code does not apply to measuring fluid level in pipe. Two new manual UT devices will be used that have been qualified using approved procedures, certified personnel, and a mock-up demonstration to prove accurate level indication can be measured.

00-3-092	2	Mansell Level Monitoring System (MLMS)

Revision 2 of this evaluation was to review the MLMS low point pressure transducer location on a low point connection off the Reactor Coolant System hot leg as well as the original points off the intermediate leg piping. This allows additional flexibility for installing the low point transducers.

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01-3-005-RCS	0	Gas Injection to Assist RCS Draindown

This evaluation was to review the performance of Reactor Coolant System (RCS) drain down using gas injection to assist with the removal of water from the Steam Generator U-tubes. Following RCS cool down and depressurization operations, the RCS must be drained for Steam Generator tube inspections and other maintenance. Assisting removal of water from the Steam Generator U-tubes is accomplished by injecting non-condensable gas (nitrogen/air) to replace the liquid held up in the U-tubes due to the inability to vent the U-tubes until the Steam Generator Channel Heads are uncovered. Injection of non-condensable gas facilitates a smooth contiguous drain down operation as opposed to the chugging that occurs with lowering level below the channel head.

01-3-022	0	Instrument Bus 34 Inverter Replacement

This evaluation is for the replacement of the Instrument Bus 34 Inverter. The existing Instrument Bus 34 Westinghouse 7.5 KVA Inverter is aging and does not incorporate an automatic static transfer feature. Transferring power supplies for the Instrument Bus requires a "break before make" operation resulting in potential plant transients. The replacement 7.5 KVA Static Inverter manufactured by Solidstate Controls Incorporated (SCI) will replace the existing unit and incorporates an internal static transfer switch allowing uninterrupted automatic and manual bus transfers.

01-3-080	0	Control Room Kitchen Upgrade

This evaluation is for the enlarging and modernizing the kitchen in the back of the Central Control Room (CCR). The existing kitchen is actually referred to as a Pantry on early plant drawings. The kitchen will be moved into the CCR locker room area which is no longer used as a locker room. The lockers that are displaced will be relocated to the old pantry, the current kitchen. The existing HVAC system is adequate and satisfactory for the proposed kitchen. This modification will not install any combustibles in the CCR. No wood will be used in the construction, the cabinets will be stainless steel, and the new floor covering and suspended ceiling will meet the required flame spread criteria.

EVL-02-3-070-HC	0	Allowance for Additional Aluminum in Containment

This evaluation is for establishing new limits for aluminum in the Vapor Containment (VC) building by increasing the total allowable margin as described in the FSAR from 25 lb and 25 ft^2 to 408 lb and 1842 ft^2. The revised figures are taken from a Westinghouse analysis which was prepared in 2000 to support modifications to the Control Rod Drive Mechanism (CRDM) fan blades. The increase in allowable aluminum inventory has no significant effect on plant response to a LB-LOCA. The presence of the additional aluminum shortens the required time to activate the Hydrogen Recombiners from 10 days to 5.5 days. These times are very long with respect to the expected time at which hydrogen monitoring is established, which is less than seven hours. This evaluation also addresses the effect of zinc on post-LOCA hydrogen generation. The FSAR originally reviewed the effects of zinc and dismissed them as inconsequential. The FSAR is

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updated to incorporate a 1992 Con Edison Evaluation for Indian Point Unit 2 which shows that zinc buildup is not an issue until approximately 5.5 days post-LOCA, by which time hydrogen recombination will have already begun.

EVL-02-3-115-SG	0	Use of Nozzle Dams in the Indian Point 3 Steam Generator Lines

This evaluation is for the use of nozzle dams in the Indian Point 3 Steam Generator Lines. These devices will allow full flooding in the reactor cavity and fuel handling activity to be performed while one or more steam generators are open on the primary side. This evaluation considers the impact of nozzle dam installation on design basis accidents and other off-normal operating conditions. It considers the possibility of nozzle failure and the contingencies that must be in place to address such an event.

EVL-02-3-123-MS	0	Evaluation of Steam Break Outside Containment for EQ Purposes

This evaluation revises FSAR Section 6.F.3.E, "Effects of Pipe Breaks in the Steam and Feedline Penetration Area" to reflect the revised Main Steam Line Break (MSLB) analysis and the calculations performed to verify equipment operability. A series of errors in the original MSLB analysis, both conservative and non-conservative, led to the need to completely reanalyze the accident. When the work was complete, the peak accident temperatures exceeded the maximum Environmental Qualification (EQ) temperature for the EQ equipment in the feedline penetration area designed to respond to the MSLB. Therefore, thermal lag calculations were prepared and/or new test data was procured for the applicable equipment to demonstrate qualification to the higher temperature. It was shown that all EQ equipment in the vicinity of the feedline penetrations is capable of satisfying its accident mitigation function in the post-accident environment.

EVL-03-3-011-RCS	0	Reactor Vessel Head Stud Detensioning and Removal, Installation and Tensioning

This evaluation is for revising the procedures for Reactor Vessel Head Stud detensioning and removal, installation and tensioning. The temperature to commence detensioning changed from above 60°F to above 70°F. The number of sets of studs to be detensioned changed from 27 to 25. Initial detensioning pressure changed from 7100 psi to a range of 5250 - 8600 psi. The revised procedure requires only one and one-third full tensioning pass versus one and two-third full tensioning pass. Based on three tensioners, the pressures used are increased from a range of 5000 - 8100 psi to 7150 - 8800 psi. Closure stud elongation tolerance band changed from 0.048" - 0.054" to Level 1 and 2 bands. Level 1 band establishes an elongation range of 0.048" - 0.058". Level 2 band permits individual stud elongation to be between 0.042" - 0.048" provided the average elongation of five adjacent studs (the out of tolerance stud and two nearest neighbor studs on both sides) remains within the Level 1 acceptance band. Tensioning of closure studs will be performed only when vessel, head flange and stud temperatures are at or above 70°F instead of 60°F and if the temperature differential between the stud and head flange exceeds 5°F, the minimum or maximum elongation limit will be adjusted. The change of

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stresses in the studs, vessel flange and head flange are in compliance with ASME Section III BPVC limits.

EVL-03-3-013-RCS	0	Thermal Controls to Support Rapid Core Offload

This evaluation reviews the associated administrative controls and the change from 145 hours subcritical to 84 hours subcritical to begin core offload. Time limits for core offload are tied to the thermal limits of Spent Fuel Pit (SFP) heat removal and to the calculated dose subsequent to a postulated fuel handling accident. The change is dependent upon the issuance of Technical Specification amendment adopting the Alternate Source Term for Fuel Assembly Drop. The core offload thermal analysis is based on the last assembly; i.e., when the last assembly is removed from the core, and that the SFP heat load cannot exceed 35 MBTU/hr. The administrative controls can be information provided in a formal calculation, computer run or engineering report. Reactor Engineering is responsible for working with Outage Management throughout the planning process to ensure an offload strategy is prepared that will support the schedule and maintain the SFP within its design limits.

EVL-03-3-027-MTG	0	Post Trip Turbine Overspeed Test

This evaluation supports implementation of Procedure 3PT-V21A, "Turbine Overspeed Test Post-Trip", which uses decay and pump heat to perform the turbine Overspeed test. The turbine will be tested with reactor trip breakers closed in order to open the Turbine Stop Valves with its normal protection features in place. The reactor trip breakers can be closed with the reactor subcritical since the Rod Drive Motor Generator sets will be secured. With the reactor trip breakers closed, 20/AST & 20/ASB remain available to trip the turbine providing normal protection from Steam Generator carry over and unbalanced steam flow. In this test configuration, normal logic remains available for opening LP Steam Dumps on turbine trip to dissipate steam energy to the Condenser. Reactivity insertion events, either through dilution or cooldown, is protected by boration to the 400 °F equivalent prior to procedure initiation, inherent margin in the shutdown boration curve, maintenance of the safety injection low pressure signal during the test, manual termination at 520 °F, and assurance of adequate steam by requiring the test to be done approximately 30 minutes after reactor trip.

EVL-03-3-095-CM	0	Operation of the Fuel Storage Building Overhead Crane with Ventilation Inoperable and in Consideration of Heavy Load Handling

This evaluation provides a review for movement of the overhead crane in the Fuel Storage Building whenever a heavy load is being carried and whenever the building ventilation system is inoperable. The impact of fuel damage has been significantly reduced under the new analytical rules provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants". The conclusions of this analysis, which have been incorporated into the design basis via Technical Specification Amendment 215, justify relaxation of the present restrictions, provided that administrative controls are maintained. Furthermore, the necessary administrative controls required for transport of loads >2000 lbs across the SFP, in accordance with existing commitments and NUREG-0612 "Control of Heavy

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Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36", have been established.

04-0561-TM-00-RE	0	Evaluation of EDG Operability During Flange Weld Repairs of EDG Discharge Header Valve SWN-55

This evaluation is for reviewing three Temporary Alterations to provide alternate Service Water (SW) return flow paths for the Emergency Diesel Generator (EDG) coolers. This Temporary Alteration to all three EDGs was required to facilitate the flange weld repair of EDG SW outlet valve SWN-55 during Refueling Outage 13. The Temporary Alterations on each EDG installs a QA Cat. 1 Jacket Water Cooler outlet head cover with a 4 inch manual globe valve and discharge hose, route the return of each EDG's Jacket Water Cooler to it's 24 inch room floor drain, and provide continuous SW flow to each EDG Cooler set. The Temporary Alteration can only be placed in service when the plant is shutdown for a minimum of five days, and be in Mode 6 (Refueling), and the Reactor Vessel Closure heads removed. Five days after shutdown is necessary to achieve an acceptable reduction in core decay heat, and the head must be off the reactor vessel to allow for gravity drain in the event of a loss of the Residual Heat Removal pumps.

04-1066-MD-00-RE	0	Fuel Design Change to Improve Grid-to-Rod Fuel Fretting Margin

This evaluation is for fuel design change to improve grid-to-rod fuel fretting margin, while minimizing effects on other thermal-hydraulic, mechanical, or safety performance factors of the current design. The 15x15 Upgrade design contains 1) ZIRLO structural mid-grids with the same number and vane shape as the 15x15 V5H design (current design) but re-distributed to create a balanced vane pattern and twice the rod contact length, 2) ZIRLO Integrated Flow Mixer (IFM) grids with the same balanced vane pattern as that developed for the mid-grid with over three times the contact length of the V5H IFMs, and 3) tube-in-tube ZIRLO guide thimbles. There is no change to the fuel assembly length, fuel assembly envelope or fuel rod design, relative to the current 15x15 V5H design. Thus, the 15x15 Upgrade fuel assembly design is considered a direct replacement for the current 15x15 V5H/OF A/V+ designs with the appropriate plant-specific evaluations/analysis supporting it's acceptability.

05-299-MD-00-RE	0	IP3 Cycle 14 Core Reload Design

This evaluation reviews the IP3 Cycle 14 fuel, core and cycle design. This evaluation was accomplished utilizing the methodology described in WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology". The rated thermal power is planned to be increased from 3067.4 to 3180 MWt. The safety analysis results performed in support of the power increase have been incorporated in this reload evaluation. Cycle 14 will be the first cycle to use the 15x15 Upgraded fuel assembly design in the core for Regions 16A and 16B. The NRC-approved Fuel Criteria Evaluation Process (FCEP) was used to address the design modifications for the 15x15 upgraded fuel assemblies. The FCEP was used to justify additional fuel rod

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average burnup exposure of up to 2,000 MWD/MTU beyond the currently licensed limit of 60,000 MWD/MTU. All fuel design criteria continue to be satisfied for Cycle 14. The evaluation is consistent with the evaluation/analysis in the FSAR and those performed in support of the uprating.

05-364-MM-00-RE	0	Install Isolation Valve and Associated Fill Valve in ¾"-SI-1501R Line #31

Leakage of nitrogen saturated water past SI Test Valve 839H may cause N2 gas to buildup in the SI Pump suction and discharge lines. ER-05-3-039 installs an isolation valve assembly in the section of the ¾" test line (# 31) downstream of valves 839H and 839G to stop the leakage. Line # 31 can not be isolated and drained during plant operations; therefore, freeze seals are used to isolate the section of line # 31 where the valve assembly is to be installed. The valve assembly consists of an isolation valve with a fill valve and closure cap. The fill valve is required to pressurize the piping before the freeze seals are thawed. This evaluation reviews the loss of Line # 31 testing functions on SI check valve leakage when the isolation valve is closed. It evaluates the impact on the safety analysis of a ¾" line break, should an SI Actuation occur during the installation of the ¾" valve after cutting Line # 31, but prior to complete installation.