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NL-09-135

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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**
Indian Point Unit No. 3
Docket No. 50-286
License No. DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.59 (d)(2), enclosed please find a 50.59 report listing and summary report of the changes, tests and experiments implemented at Indian Point Unit 3 between March 31, 2007, and April 15, 2009 or utilized in support of the UFSAR update. The 50.59 Evaluations 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) 734-6710.

Sincerely,

RW/as

Attachment 1 – 50.59 Report Listing
Attachment 2 - 50.59 Summary of Changes, Tests and Experiments

cc: see next page

IE47
NRR

cc: Mr. John P. Boska, NRR Senior Project Manager
Mr. Samuel J. Collins, Regional Administrator, Region 1
Mr. Paul Eddy, Public service Commission
Mr. Francis J. Murray, Jr., President & CEO NYSERDA
NRC Resident Inspector's Office

ATTACHMENT 1 TO NL-09-135

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 – 2009 Report 50.59 EVALUATION TITLE
08-3001-00-EVAL	0	Partial UFSAR Revision in Support of Resolution of GSI-191 (EC-2813)
09-3001-00-EVAL	0	Evaluation of Motor Driven Auxiliary Feedwater (MDAFW) Pump Branch Line Flow Imbalance of up to 150 GPM Difference Between Two Steam Generators (EC-14175)
09-3002-00EVAL	0	TS Bases 3.6.5 Explanation of Initial Containment Temperature Assumed in the LOCA Analysis of Record for Peak Cladding Temperature (PCT)

ATTACHMENT 2 TO NL-09-135

50.59 SUMMARY OF CHANGES, TESTS AND EXPERIMENTS

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

50.59 Summary of Changes, Tests and Experiments

50.59 Evaluation No.	Rev. No.	TITLE
08-3001-00-EVAL	0	Partial UFSAR Revision in Support of Resolution of 191 (EC-2813)

Brief Description of the Change, Test or Experiment:

Revise the Unit 3 UFSAR to incorporate the 30 day mission time input assumption to be used in the assessments of the functional performance of the recirculation equipment following a Loss Of Coolant Accident (LOCA) with respect to the new methodology for evaluating Pressurized Water Reactors (PWR) sump performance under NRC Generic Letter GL-2004-02 and Generic Safety Issue GSI-191.

Summary of the associated 10 CFR 50.59 Evaluation

Due to the issuance of GSI-191 / GL 2004-02, we need to validate the new generic NRC methodology for calculating and evaluating post LOCA debris generation, transport and effects in PWRs. This results in a new mechanistically based evaluation of sump and recirculation phase performance by all PWR licensees. Many input assumptions, calculational methodologies, and effects analysis are being changed from original plant licensing bases. The original IP3 licensing basis for sump performance was deterministically based. Post-LOCA recirculation equipment was required to perform in a post-LOCA borated and buffered solution.

With respect to evaluating the impacts of the new GL-2004-02 assumptions on post-LOCA recirculation equipment, the NRC addressed the issue of mission time in the generic Safety Evaluation Report (SER) on NEI 04-07. Westinghouse WCAP-16406 provided a generic assessment of the effects of the new GSI-191 debris assumptions on systems downstream of the sumps. It also established a generic mission time of 720 hours (ie: 30 days) specifying that any mission times less than this needed to be supported by plant specific calculation. IP3 will be using the 30 day mission time for the specific evaluations and analyses necessary to respond to the GSI-191 and GL-2004-02 requirements. The UFSAR will be updated to reflect the use of a 30 day mission time along with many other new methodology / input assumption changes to be addressed separately for purposes of responding to GSI-191 and GL-2004-02.

50.59 Evaluation No.	Rev. No.	TITLE
09-3001-00-EVAL	0	Evaluation of Motor Driven Auxiliary Feedwater (MDAFW) Pump Branch Line Flow Imbalance of up to 150 GPM Difference Between Two Steam Generators (EC-14175)

<u>Brief Description of the Change, Test or Experiment:</u>

This evaluation reviewed the UFSAR Chapter 14 analysis for MDAFW pump branch line flow imbalances up to 150 gpm difference between two steam generators.

<u>Summary of the associated 10 CFR 50.59 Evaluation</u>

When a motor driven auxiliary feedwater pump discharge pressure runback controller actuates, an asymmetrical flow distribution to the associated two steam generators (SG) is created. Although the SGs receive different flow rates, the total flow to both SGs meets the total minimum required accident flow from the pump. A MDAFW pump branch line flow asymmetry of up to 150 gpm was reviewed for impact on the UFSAR Chapter 14 analyses. Most analyses were either not impacted or could accommodate the asymmetry. The limiting cases for the following transients were reanalyzed:

- Loss of Normal Feedwater (LONF),
- Loss of Non-Emergency AC Power LOAC),
- The near best estimate versions of the LONF and LOAC transients,
- Anticipated Transients Without Scram (ATWS),
- The Mass & Energy releases for Main Steam Line Break (MSLB) Outside of Containment.

These reanalyses concluded that all acceptance criteria continue to be met and the conclusions presented in the IP3 UFSAR remain valid.

50.59 Evaluation No.	Rev. No.	TITLE
09-3002-00-EVAL	0	TS Bases 3.6.5 Explanation of Initial Containment Temperature Assumed in the LOCA Analysis of Record for Peak Cladding Temperature (PCT)

Brief Description of the Change, Test or Experiment:

IP3 Technical Specification (TS) Bases 3.6.5 was clarified to explain the initial containment temperature assumed in the LOCA analysis of record for peak cladding temperature (PCT).

Summary of the associated 10 CFR 50.59 Evaluation

Operations found it challenging to maintain 90°F containment temperature at power in the colder months and so the impact on LOCA due to the reduction in initial containment temperature from 90°F to 80°F is discussed in this evaluation. TS Bases 3.6.5 specifies the allowable range for containment initial temperature to be between 50°F and 130°F. The change to the initial containment temperature for input to LOCA PCT analysis from 90°F to 80°F is still within this range. The only impact of this change is a slight increase in the PCT. This change is to an analytical model input only with no change to any structure, system or component in the plant.