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Fred Dacimo
Site Vice President
Administration

September 26, 2005

Re: Indian Point Unit 2
Docket 50-247
NL-05-107

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

SUBJECT: **Proposed Change to Indian Point 2 Technical Specifications Regarding LBLOCA Analysis Methodology**

REFERENCE: 1. Entergy letter (NL-05-058) to NRC; "Reanalysis of Large Break Loss of Coolant Accident Using ASTRUM", dated April 22, 2005.

Dear Sir:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc, (Entergy) hereby requests an amendment to the Operating License for Indian Point Nuclear Generating Unit 2 (IP2) to adopt the use of ASTRUM for the licensing basis analysis of the Large Break Loss of Coolant Accident (LBLOCA), as stated in Reference 1.

Entergy has evaluated the proposed change in accordance with 10 CFR 50.91 (a)(1) using the criteria of 10 CFR 50.92 (c) and Entergy has determined that this proposed change involves no significant hazards considerations, as described in Attachment 1. The proposed changes to the Technical Specifications are shown in Attachment 2.

A copy of this application and the associated attachments are being submitted to the designated New York State official.

Entergy requests approval of the proposed amendment by June 2006, to be implemented within 60 days. 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 9/26/2005

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

A001

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Change (markup)

cc: Mr. John P. Boska, Senior Project Manager, NRC NRR
Mr. Samuel J. Collins, Regional Administrator, NRC Region 1
NRC Resident Inspector's Office, Indian Point Unit 2
Mr. Peter R. Smith, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-05-107

ANALYSIS OF PROPOSED

TECHNICAL SPECIFICATION CHANGE REGARDING

USE OF ASTRUM FOR LBLOCA ANALYSIS

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

1.0 DESCRIPTION

This letter requests an amendment to Operating License DPR-26, Docket No. 50-247 for Indian Point Nuclear Generating Unit No. 2 (IP2).

The proposed amendment will revise the analysis method used for the Large Break Loss of Coolant Accident (LBLOCA) by incorporating the use of a new approach (ASTRUM) for the treatment of parameter uncertainties. The new approach is described in Westinghouse Topical Report WCAP-16009-P-A, approved by NRC in Reference 1.

Changes to the Technical Specifications to reflect the proposed use of ASTRUM in LBLOCA analyses consist of revisions to the list of references provided in Technical Specification Section 5.6.5, Core Operating Limits Report.

2.0 PROPOSED CHANGES

Technical Specification Section 5.6.5 (Core Operating Limits Report); three references in line item b.6 are replaced by a new reference, WCAP-16009-P-A. Refer to Attachment 2 for markup page.

3.0 BACKGROUND

The current methodology used for analysis of LBLOCA at IP2 is based on Westinghouse Topical Report WCAP-12945 and the plant-specific application of the methodology to IP2 as approved by NRC in 1997 (Reference 2). Since that time, the analyses have been updated to account for various plant change evaluations and model correction items in accordance with 10 CFR 50.46(a)(3). The most recent analysis update was performed as part of a stretch power uprate project in 2004 (Reference 3).

Entergy committed, in Reference 4, to perform a reanalysis of LBLOCA using the ASTRUM methodology. NRC subsequently approved (Reference 5) Westinghouse Topical Report WCAP-16009-P, which describes the ASTRUM methodology and Entergy performed the LBLOCA reanalysis as committed. Entergy reported the results of that analysis in Reference 6 and stated the intent to submit a license amendment request to formally adopt the ASTRUM methodology.

4.0 TECHNICAL ANALYSIS

The original application of the Westinghouse Best Estimate Methodology to Indian Point Unit 2 Nuclear Plant, approved by the NRC in 1997, employed the NRC approved 1996 Evaluation Model (Reference 2). Westinghouse recently underwent a program to revise the statistical approach used to develop the Peak Cladding Temperature (PCT) and oxidation results at the 95th percentile. This method is still based on the Code Qualification Document (CQD) methodology (Reference 7) and follows the steps in the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM evaluation model is documented in WCAP-16009-P-A (Reference 1).

This section summarizes the application of the Westinghouse ASTRUM Best Estimate Loss of Coolant Accident (BELOCA) evaluation model to the Indian Point Unit 2 Nuclear Plant for analysis of the large break LOCAs (LBLOCA). The analysis was performed in compliance with all the NRC conditions and limitations as identified in WCAP-16009-P-A.

The current WCOBRA/TRAC model for Indian Point Unit 2 is based on the methodology of WCAP-12945 and uses the current uprated power level of 3216 MWth. Use of the best estimate LBLOCA methodology for Indian Point 2 was initially approved by NRC in 1997 (Reference 2). The WCOBRA/TRAC noding that was developed at that time remains unchanged for the best estimate LBLOCA ASTRUM analysis. The ASTRUM best estimate LBLOCA analysis was performed for a full core of upgraded fuel. Table 1 lists the major plant parameter assumptions used in the analysis for Indian Point Unit 2. The axial power distribution envelope assumption is shown in Figure 1. Table 2 summarizes the results of the ASTRUM best estimate LBLOCA analysis, as previously reported in Reference 6. Table 3 contains a sequence of events for the limiting PCT transient. Based on these results, Indian Point Unit 2 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the Peak Clad Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with 95% confidence level. These parameters are needed to satisfy the 10 CFR 50.46 criteria with regard to PCT, LMO, and CWO. From these 124 calculations, Run 76 proved to be the limiting PCT transient and the limiting LMO transient, and Run 11 the limiting CWO transient.

The scatter plot presented on Figure 2 shows the effect of the effective break area on the final PCT. The effective break area is calculated by multiplying the discharge coefficient (C_D) with the sample value of the break area, normalized to the cold-leg cross sectional area. Figure 2 is provided because the break area is a significant contributor to the variation in PCT.

Figures 3 and 4 are presented to show the limiting cladding transient for each criterion. Figure 3 shows the predicted clad temperature transient at the PCT and LMO limiting elevation for Run 76 and Figure 4 presents the PCT trace for the CWO limiting transient from Run 11.

Table 1: Major Plant Parameter Assumptions Used in the Best Estimate Large Break LOCA Analysis for Indian Point Unit 2

Parameter	Value	Documentation
Plant Physical Description		
• SG Tube Plugging	≤10% **	UFSAR 14.3
Plant Initial Operating Conditions		
• Reactor Power	≤102% of 3216 MWt	UFSAR 14.3
• Peaking Factors	$F_Q \leq 2.5$ $F_{\Delta H} \leq 1.7$	UFSAR 14.3
• Axial Power Distribution	See Figure 1 **	UFSAR 14.3
Fluid Conditions		
• T_{AVG}	549 - 3.3 °F ≤ T_{AVG} < 572 + 3.3 °F (1) **	UFSAR 14.3
• Pressurizer Pressure	2250 - 25 psia ≤ P_{RCS} ≤ 2250 + 25 psia (2) **	UFSAR 14.3
• Reactor Coolant Flow	≥ 80700 gpm/loop **	UFSAR 14.3
• Accumulator Temperature	80 °F ≤ T_{ACC} ≤ 130 °F	UFSAR 14.3
• Accumulator Pressure	612.7 psia ≤ P_{ACC} ≤ 699.7 psia **	UFSAR 14.3
• Accumulator Water Volume	723 ft^3 ≤ V_{ACC} ≤ 875 ft^3	UFSAR 14.3
Accident Boundary Conditions		
• Single Failure Assumptions	Loss of one ECCS train	UFSAR 14.3
• Safety Injection Flow	Minimum	UFSAR 14.3
• Safety Injection Temperature	35 °F ≤ T_{SI} ≤ 110 °F **	UFSAR 14.3
• Safety Injection Initiation Delay Time	≤ 38 seconds (with offsite power) ≤ 45 seconds (without offsite power)	UFSAR 14.3
• Containment Pressure	Bounded (minimum)	UFSAR 14.3

(1) Include -3 (bias)*
(2) Include -3, +12 (bias)*

* Bias sign convention: "+" means indicated value is higher than actual and "-" means indicated value is lower than actual.

** The current version of the UFSAR does not contain this value, or range. However, this value or range will be updated in a subsequent UFSAR revision.

Table 2: Indian Point Unit 2 Best Estimate Large Break LOCA Results

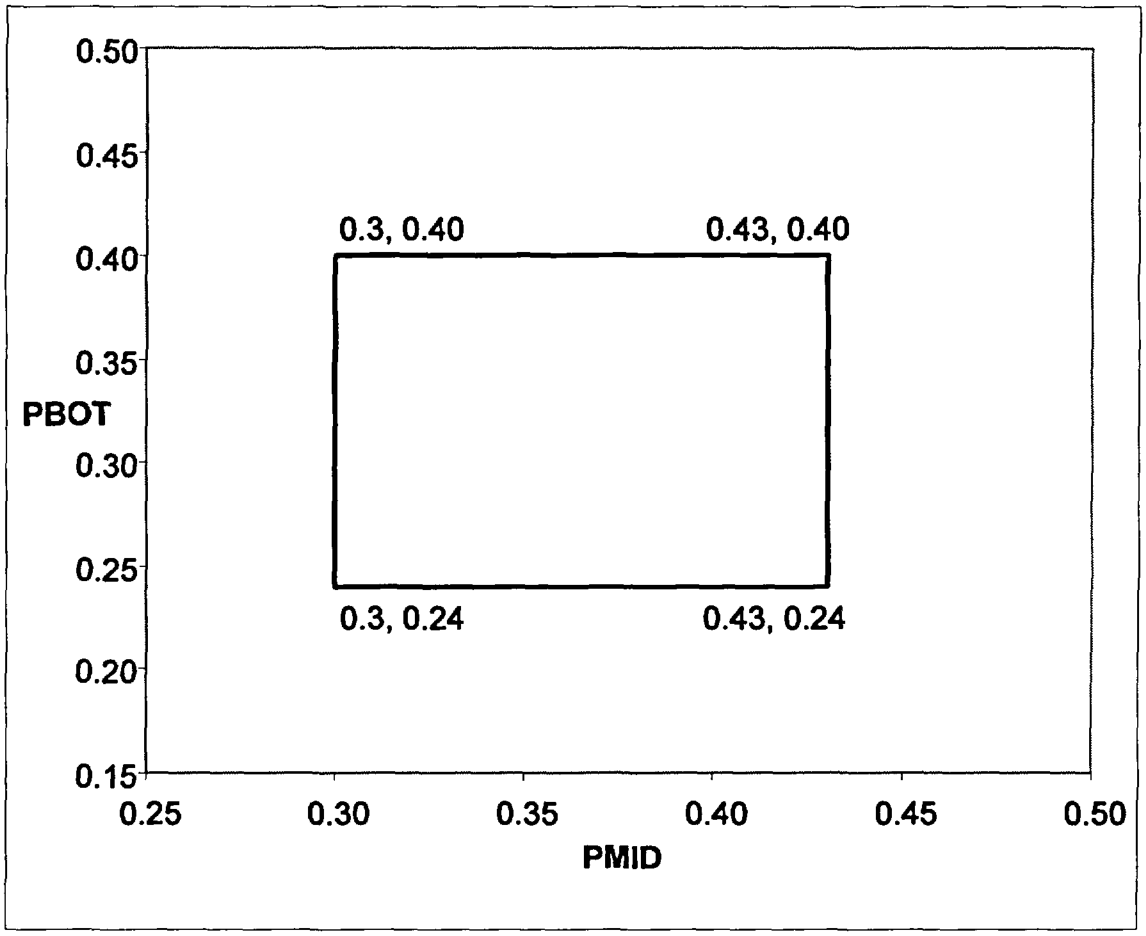
10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1962	< 2200
95/95 LMO [1] (%)	2.39	< 17
95/95 CWO [2] (%)	0.35	< 1
Coolable Geometry	Core remains coolable	Core remains coolable
Long Term Cooling	Core remains cool in long term	Core remains cool in long term

[1] Local Maximum Oxidation
[2] Core Wide Oxidation

Table 3: Indian Point Unit 2 Best Estimate Large Break LOCA Sequence of Events for Limiting PCT Transient

Event	Time (sec)
Start of Transient	0.0
Safety Injection Signal	6.0
Accumulator Injection Begins	10.0
End of Blowdown	28.0
Accumulator Empty	39.0
Bottom of Core Recovery	40.0
Safety Injection Begins	51.0
PCT Occurs	123.0
PCT Elevation Quench	330.0
End of Transient	500.0

Figure 1: Indian Point Unit 2 Best Estimate Large Break LOCA Analysis Axial Power Shape Operating Space Envelope



PBOT: integrated power fraction in the lower third of the core PMID: integrated power fraction in the middle third of the core

Figure 2: Indian Point Unit 2 Best Estimate Large Break LOCA Analysis HOTSPOT PCT vs. Effective Break Area Scatter Plot

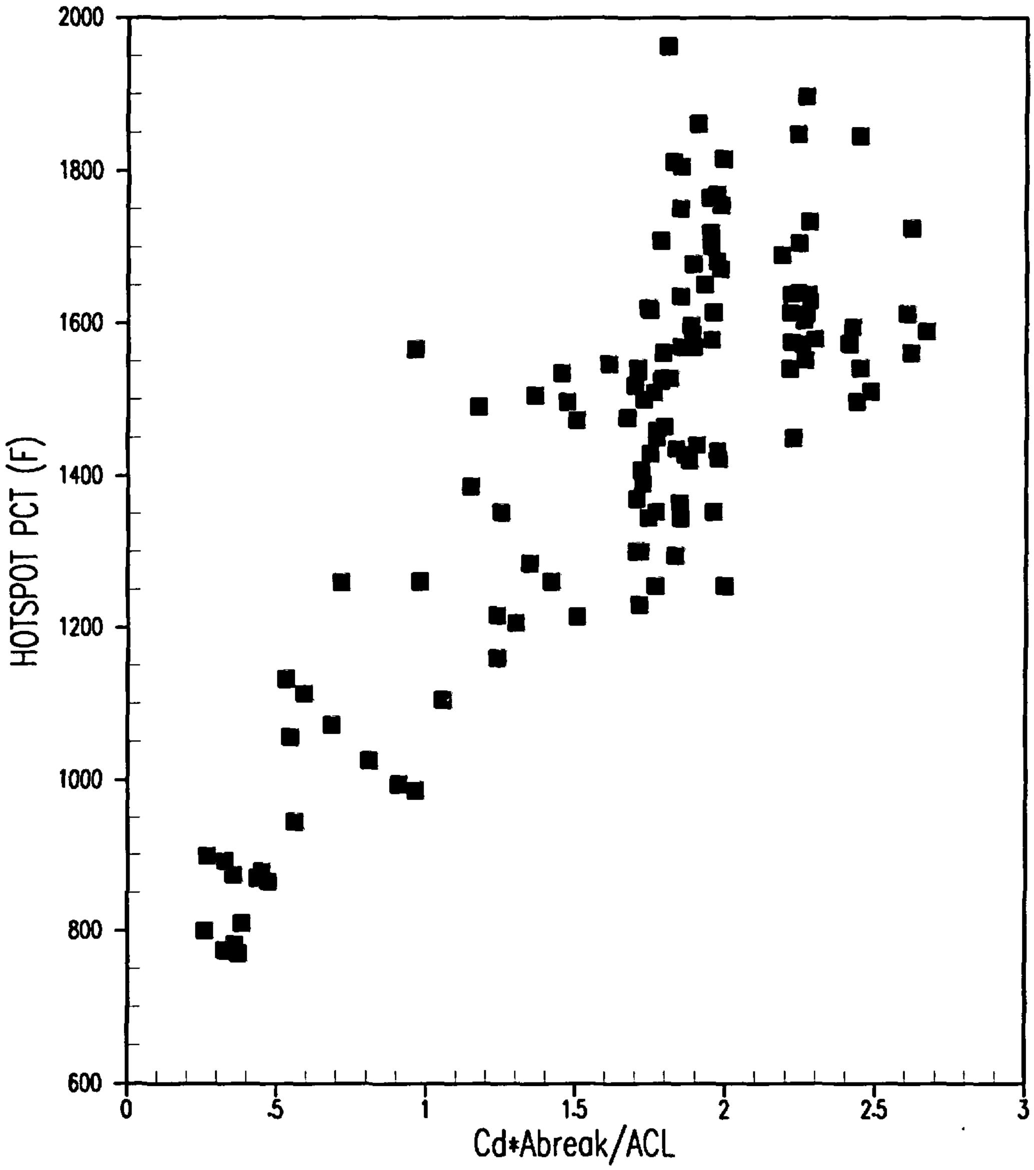


Figure 3: Indian Point Unit 2 Best Estimate Large Break LOCA Analysis HOTSPOT Clad Temperature Transient at the Limiting Elevation for the PCT and LMO Limiting Case

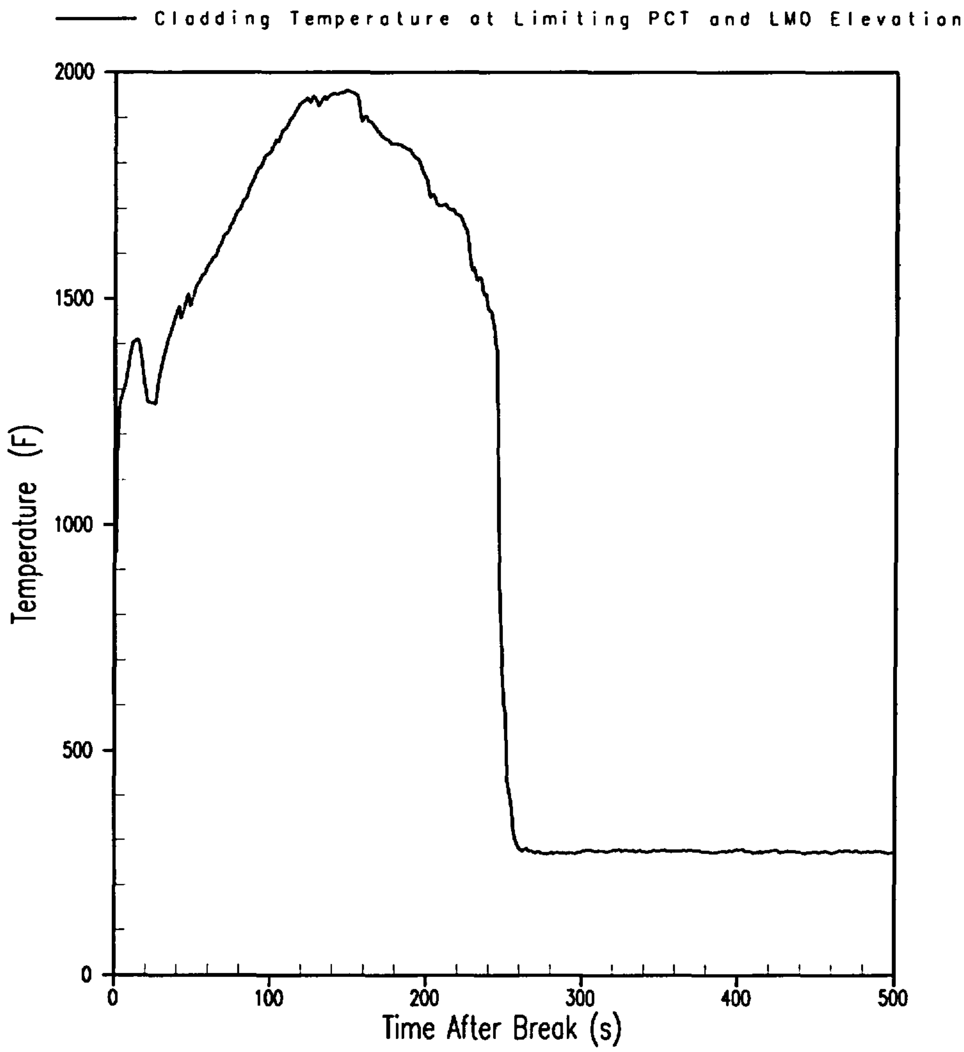
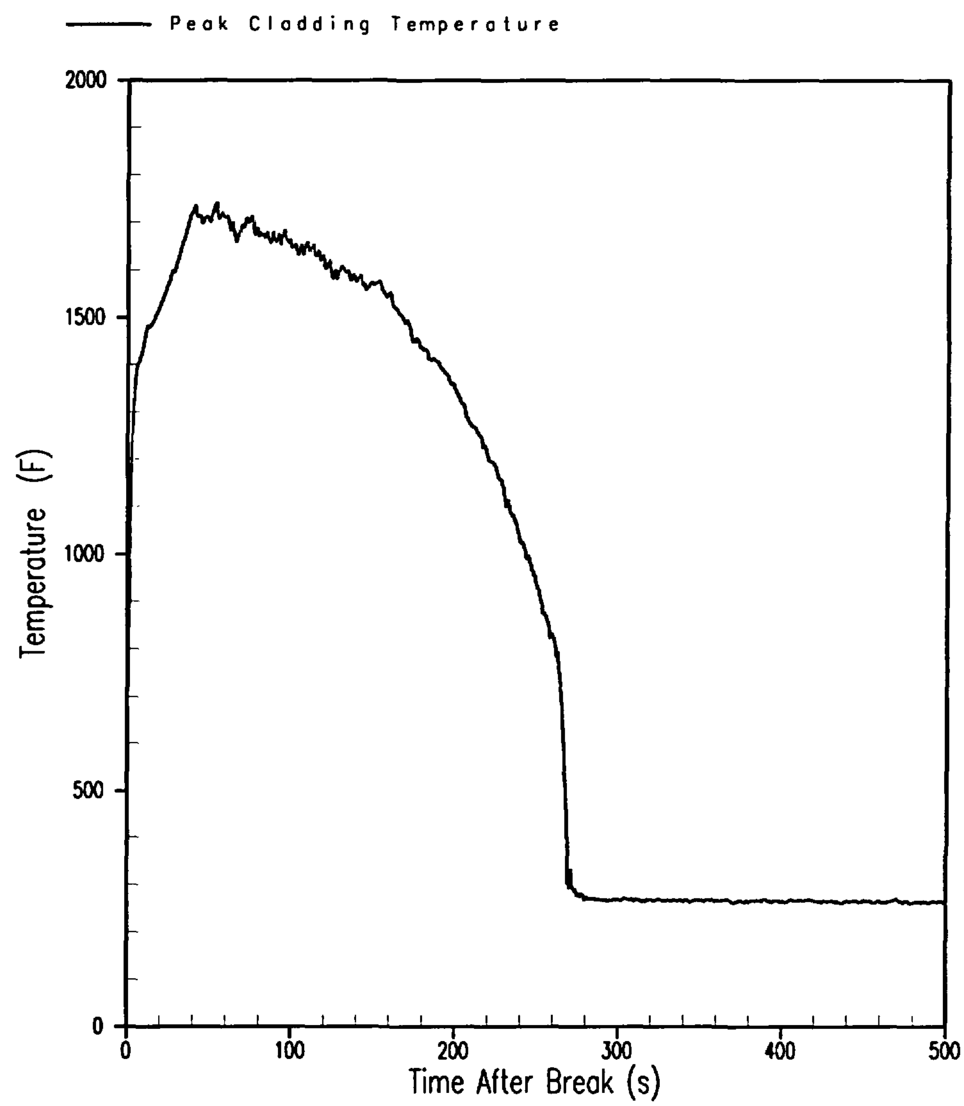


Figure 4: Indian Point Unit 2 Best Estimate Large Break LOCA Analysis WCOBRA/TRAC PCT Transient for the CWO Limiting Case



5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (Entergy) has evaluated the safety significance of the proposed changes regarding use of the ASTRUM methodology in the analysis of the Large Break Loss of Coolant Accident (LBLOCA) for Indian Point 2 (IP2) according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject changes do not involve a Significant Hazards Consideration as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change modifies the analysis methodology used to account for the variation in parameters that are used for the safety analysis of the LBLOCA. This proposed change has no effect on the design or operation of plant equipment. Use of the new methodology will revise the results of the current analysis, but there will be no change in initiating events for this accident scenario or the ability of the plant equipment or plant operators to respond.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not involve modifications to existing plant equipment or the installation of any new equipment. The proposed change only affects the analysis methodology that is used to evaluate the response of existing plant equipment to the LBLOCA scenario. Plant operating and emergency procedures that are in place for the LBLOCA scenario are also not being changed by this proposed amendment. This proposed change does not create new failure modes or malfunctions of plant equipment nor is there a new credible failure mechanism.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

The proposed license amendment revises the analysis methodology which is used to assess the impact of the LBLOCA scenario with respect to established acceptance criteria. Margins of safety for LBLOCA include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria and the associated margins of safety are not being changed. The evaluation of the LBLOCA scenario, using the proposed new methodology must still meet the existing established acceptance criteria.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements / Criteria

The applicable regulatory requirement for this license amendment request is 10 CFR 50.46, which includes requirements and acceptance criteria pertaining to the evaluation of emergency core cooling system (ECCS) performance.

This regulation includes the requirement that "... uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria ... there is a high level of probability that the criteria would not be exceeded."

The proposed license amendment requests approval to use the ASTRUM methodology (WCAP-16009) for the treatment of uncertainties in the inputs used for the LBLOCA analysis. There is no change being proposed to the analysis acceptance criteria specified in the regulations. NRC has reviewed WCAP-16009 and found it acceptable for referencing in licensing applications for Westinghouse and Combustion Engineering designed pressurized water reactors. WCAP-16009 is applicable to Indian Point 2 and the plant-specific application of the ASTRUM methodology to the IP2 LBLOCA analysis has been performed in accordance with the conditions and limitations of the topical report and the associated NRC Safety Evaluation.

5.3 Environmental Considerations

The proposed change to the IP2 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 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

NRC has reviewed and accepted the Westinghouse topical report (WCAP-16009) which describes the ASTRUM methodology and NRC is in the process of reviewing a plant-specific application (Docket 50-244; April 29, 2005) which includes use of ASTRUM.

7.0 REFERENCES

1. Nissley, M. E., et.al, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty (ASTRUM)," WCAP-16009-P-A, January 2005.

2. NRC letter to Consolidated Edison; Issuance of Amendment [188] for Indian Point Nuclear Generating Unit No. 2", dated March 31, 1997.

3. NRC letter to Entergy; "Issuance of Amendment [241] Re: 3.26 Percent Power Uprate (TAC MC1865)", dated October 27, 2004.

4. Entergy letter (NL-04-081) to NRC; "Proposed Schedule for Reanalysis of Large Break Loss of Coolant Accident", July 2, 2004.

5. NRC letter to Westinghouse; "Final Safety Evaluation for WCAP-16009-P, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty (ASTRUM)," November 5, 2004.

6. Entergy letter (NL-05-058) to NRC; "Reanalysis of Large Break Loss of Coolant Accident Using ASTRUM", April 22, 2005.

7. Bajorek, S. M., et. al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary).

ATTACHMENT 2 TO NL-05-107

MARKUP OF TECHNICAL SPECIFICATION PAGES

FOR PROPOSED CHANGES REGARDING

USE OF ASTRUM FOR LBLOCA ANALYSIS

AFFECTED PAGE

5.6-3

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

5.6 Reporting Requirements

NO CHANGES THIS PAGE – INFORMATION ONLY

5.6.3 Radioactive Effluent Release Report

- NOTE -

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not Used

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Technical Specification 2.1, Safety Limits (SL);
2. Technical Specification 3.1.1, SHUTDOWN MARGIN (SDM);
3. Technical Specification 3.1.3, Moderator Temperature Coefficient (MTC);
4. Technical Specification 3.1.5, Shutdown Bank Insertion Limits;
5. Technical Specification 3.1.6, Control Bank Insertion Limits;
6. Technical Specification 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);
7. Technical Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor;

INSERT REFERENCE 6 FOR SECTION 5.6.5.b:

WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," M. E. Nissley, et al., January 2005.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

9. Technical Specification 3.3.1, Reactor Protection System Instrumentation;

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

11. Technical Specification 3.9.1, Boron Concentration.

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

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

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

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

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

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

Replace with new Ref 6, see insert next page

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

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

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