Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601
Tel 914 272 3200
Fax 914 272 3205

Michael R. Kansler
Senior Vice President &
Chief Operating Officer

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IPN-02-052

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

SUBJECT: Indian Point Nuclear Generating Unit No. 3
Docket No. 50-286
Request for Plant Specific Approval Of Best Estimate Large Break LOCA Analysis and Proposed Changes To Technical Specifications To Utilize For Indian Point 3

Dear Sir:

Enclosed is an application for amendment to the Indian Point Power Plant (IP3) Technical Specifications. This License Amendment Request (LAR) requests approval to apply the Westinghouse generic Best Estimate Large Break LOCA Analysis methodology using the WCOBRA/TRAC computer code to Indian Point 3 and requests an Amendment to the Indian Point 3 Technical Specifications to allow use of the methodology.

Approval of the Westinghouse generic Best Estimate Large Break Analysis methodology is documented in the NRC's Safety Evaluation Report found in the Topical Report WCAP-12945-P-A, "Code Qualification Document for Best Estimate Loss-of-Coolant Analysis", dated March, 1998. A plant specific analysis of the Indian Point 3 plant has been performed using the approved methodology.

This application for amendment to the Indian Point 3 Technical Specifications seeks to amend Technical Specification 5.6.5.b, "Core Operating Limits Report (COLR)". Attachment I to this application contains the analysis of the request and Attachment II provides the markup pages for the proposed changes to the Technical Specifications.

These changes are being made to incorporate the best estimate approach into the licensing basis for the Indian Point 3 large break LOCA analyses in accordance with 10 CFR 50.46, Regulatory Guide 1.157, and the Westinghouse "Code Qualification Document For Best Estimate LOCA Analysis," WCAP-12945-P-A, Volumes I-V.

The Best Estimate LOCA methodology will be incorporated into the Indian Point 3 licensing basis and the UFSAR will be updated to present the inputs and results for the Indian Point Unit 3 Best Estimate Large Break LOCA analysis as per 10 CFR 50.71(e). The COLR will also be updated via plant procedures.

A001

This submittal contains no new commitments. If you have any questions, please contact Ms. Charlene Faison at 914-272-3378.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 6/25/02

Very truly yours,

Michael R. Kansler
Senior Vice President and
Chief Operating Officer

Attachments

cc: Mr. Patrick D. Milano, Project Manager
Project Directorate I,
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, D.C. 20555

Mr. Hubert J. Miller
Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
Indian Point 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. William M. Flynn
New York State Energy, Research and
Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Dept. of Public Service
3 Empire Plaza
Albany, NY 12223

ATTACHMENT I TO IPN-02-052

ANALYSIS OF THE PROPOSED CHANGES

TO TECHNICAL SPECIFICATION 5.6.5.b

REGARDING APPROVAL AND USE OF BEST ESTIMATE LARGE

BREAK LOCA ANALYSIS

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286
DPR-64

1.0 DESCRIPTION

This License Amendment Request (LAR) to the Indian Point Unit 3 Technical Specifications (TS) proposes to amend TS 5.6.5.b, "Core Operating Limits Report (COLR)". This submittal requests approval to apply the Westinghouse generic Best Estimate Large Break LOCA Analysis methodology using the WCOBRA/TRAC computer code to Indian Point 3. It also requests an Amendment to the Indian Point 3 Technical Specifications to allow use of the methodology.

The specific changes to the TS are noted on the marked up copy of the applicable TS pages provided in Attachment II.

2.0 PROPOSED CHANGE

TS 5.6.5 identifies the applicable references for the analytical methods used to determine the core operating limits and specifies that they shall be reviewed and approved by the NRC. Reference 3a identifies the Westinghouse topical report that documents the approved large break LOCA analysis methodology. With the introduction of the Best Estimate Large Break LOCA analysis methods, this reference needs to be revised to reflect the new approved WCAP (WCAP-12945-P-A).

3.0 BACKGROUND

Westinghouse submitted to the NRC and received approval to use a best estimate large break LOCA methodology. Approval of the Westinghouse generic Best Estimate Large Break Analysis methodology is documented in the NRC's Safety Evaluation Report found in the Topical Report WCAP-12945-P-A, "Code Qualification Document for Best Estimate Loss-of-Coolant Analysis", dated March, 1998. A plant specific analysis of the Indian Point 3 plant has been performed using the approved methodology.

These changes are being made to incorporate the best estimate approach into the licensing basis for the Indian Point 3 large break LOCA analyses in accordance with 10 CFR 50.46, Regulatory Guide 1.157, and the Westinghouse "Code Qualification Document For Best Estimate LOCA Analysis," WCAP-12945-P-A, Volumes I-V. UFSAR changes will be made in accordance with 50.71(e).

Table 1 lists the plant specific parameters used in the Indian Point 3 plant specific analysis and the location of the documentation of the values and ranges used for the parameters.

Table 2 presents the 50th and 95th percentile Peak Clad Temperature (PCT) for Indian Point 3, maximum cladding oxidation, maximum hydrogen generation, and cooling results.

4.0 TECHNICAL ANALYSIS

A best estimate large break loss of coolant accident (LOCA) analysis has been performed for Indian Point 3 using the approved Westinghouse best estimate methodology contained in WCAP-12945-P-A. All plant specific parameters used in the analysis are bounded by the models and correlations contained in the generic methodology. Therefore, the Indian Point 3 analysis conforms to 10 CFR 50.46 and Appendix K, and meets the intent of Regulatory Guide 1.157.

The conclusions of the analysis are that there is a high level of probability that:

1) The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2) The calculated total oxidation of the cladding (maximum cladding oxidation) will no-where exceed 0.17 times the total cladding thickness before oxidation.

3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4) The calculated changes in core geometry are such that the core remains amenable to cooling.

5) After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Therefore, Entergy Nuclear Operations, Inc. has concluded that adopting the Best Estimate Large Break LOCA methodology for Indian Point 3 and making the proposed TS changes to allow implementation of this change will not adversely affect the health and safety of the public.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No.

No physical changes are being made by this change. The proposed changes involve use of the Best Estimate Large Break LOCA analysis methodology and associated TS changes. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10CFR 50.46 paragraph b, that is it meets the five criteria listed in Section II of this evaluation. No other accident is potentially affected by this change.

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 previously analyzed?

Response: No.

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of existing plant equipment. All plant systems will perform equally during the response to a potential accident.

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

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

Response: No.

It has been shown that the analytic technique used in the analysis more realistically describes the expected behavior of the Indian Point 3 reactor system during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties have been analyzed to provide assurance that the most severe postulated loss of coolant accidents were calculated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46 paragraph b) are met.

Based on the above, Entergy 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 proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the SAR. The information in the other sections of this analysis demonstrate that the proposed change does not impact 10CFR 50.46.

5.3 Environmental Considerations

Entergy has evaluated this proposed TS SR amendment request against the Criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Entergy has determined that this proposed amendment request meets the eligibility criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 5.1 of this Evaluation, the proposed TS change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents of handling of solid radioactive waste; nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this TS change.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes to the operation or configuration of the facility that impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of radioactive waste, nor will this TS amendment proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative radiation exposure resulting from this TS SR change.

6.0 PRECEDENCE

The Westinghouse Best Estimate Large Break LOCA Analysis methodology was approved by the NRC's Safety Evaluation Report which is found in the Topical Report WCAP-12945-P-A, "Code Qualification Document for Best Estimate Loss-of-Coolant Analysis", dated March, 1998.

Indian Point Unit 2 is one of the plants that submitted a similar license amendment and received NRC approval.

7.0 REFERENCES

1) Topical Report WCAP-12945-P-A, "Code Qualification Document for Best Estimate Loss-of-Coolant Analysis", dated March 1998.

Table 1

Major Plant Parameter Assumptions Used In the BE LOCA Analysis For IP3 And Where They Will Be Documented

Parameter	Value	Documentation
Plant Physical Description		
Steam Generator Tube Plugging	$\leq$ 30%	UFSAR 14.3.3
Plant Initial Operating Conditions		
Reactor Power	$\leq$102% of 3025 MWt	UFSAR 14.3.3
Peaking Factors	$F_q \leq 2.50$, $F_{\Delta H} \leq = 1.70$	UFSAR 14.3.3
Axial Power Distribution	Figure 1	UFSAR 14.3.3
Fluids Conditions		
T_{avg}	$556.1 \leq T_{avg} \leq 580.1$°F	UFSAR 14.3.3
Pressurizer Pressure	$2150 \leq P_{RCS} \leq 2350$ psia	UFSAR 14.3.3
Reactor Coolant Flow	$\geq$ 80,900 gpm/loop	UFSAR 14.3.3
Accumulator Temperature	$\leq$ 110 °F	UFSAR 14.3.3
Accumulator Pressure	$555 \leq P_{acc} \leq 715$ psia	UFSAR 14.3.3
Accumulator Volume	$715 \leq V_{acc} \leq 875$ ft3	UFSAR 14.3.3
Accident Boundary Conditions		
Single Failure Assumptions	1 Train of SI Pumps	UFSAR 14.3.3
Safety Injection Flow	Table 3	UFSAR 14.3.3
Safety Injection Temperature	$35 \leq T_{SI} \leq 120$°F	UFSAR 14.3.3
Safety Injection Initiation Delay Time	$\leq$ 15 sec No LOOP	
	$\leq$ 35 sec LOOP	UFSAR 14.3.3
Containment Pressure	Figure 2	UFSAR 14.3.3
Transient Results		
Peak Clad Temperature versus Time		
For Reference Case	Figure 3	UFSAR 14.3.3
Sequence of Events For Reference Case	Table 4	UFSAR 14.3.3

Table 2

IP3 BEST ESTIMATE LARGE BREAK LOCA RESULTS

	Value	Criteria
50th Percentile PCT (°F)*	1764	N/A
95th Percentile PCT (°F)*	2158**	2200
Maximum Cladding Oxidation (%)*	5.6	17
Maximum Hydrogen Generation (%)*	.65	1
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

* Documented in WCAP-14820 and calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

** The final licensing basis result including all evaluations is a 95th percentile calculation of 2158°F [2128°F (MONTECF with 1.4% uprate) + 5°F (PLOW) + 20°F (COCO) + 5°F (BENT ALIGNMENT PINS)]

Table 3
Large Break LOCA Total Injected SI Flow

RCS Pressure (psig)	Flow Rate (ft^3/sec)
0	7.82
20	6.78
40	5.71
60	3.98
80	3.40
90	2.62
100	1.95
110	1.59
200	1.52

Table 4
Large Break LOCA Time Sequence of Events
Reference Case

Event	Time (sec)
Time of break	0
SI signal (pressurizer reaches 1648.7 psia)	5.6
Accumulators start delivery	10
SI delivery starts (15 sec. delay)	20.6
Bottom of core recovers	35
Accumulator water to nitrogen injection period	41-50
Peak cladding temperature 1863°F (hot rod,8.2 ft. elevation)	41

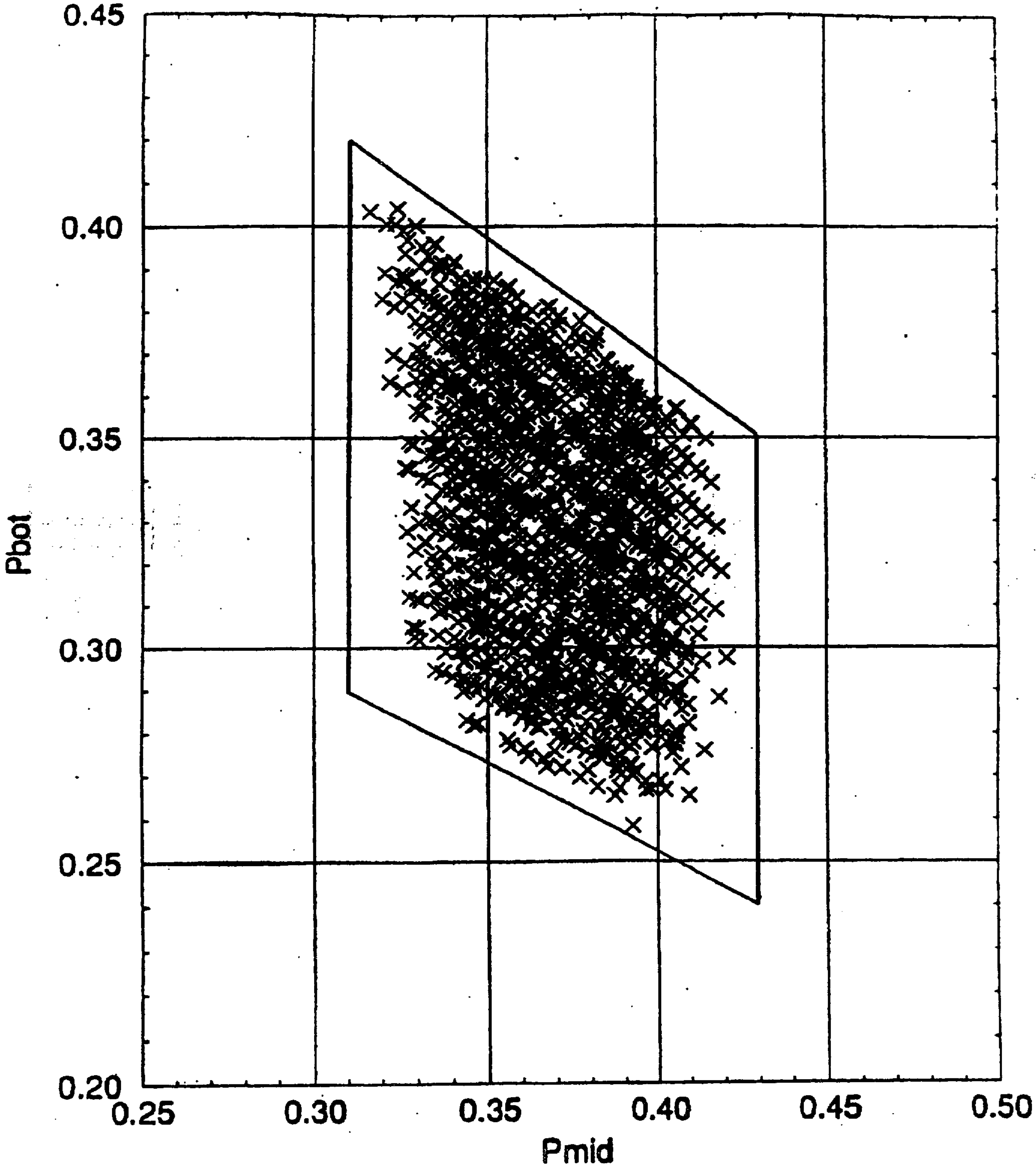

Figure 1 IP3 Pbot/Pmid Axial Power Distribution Limits

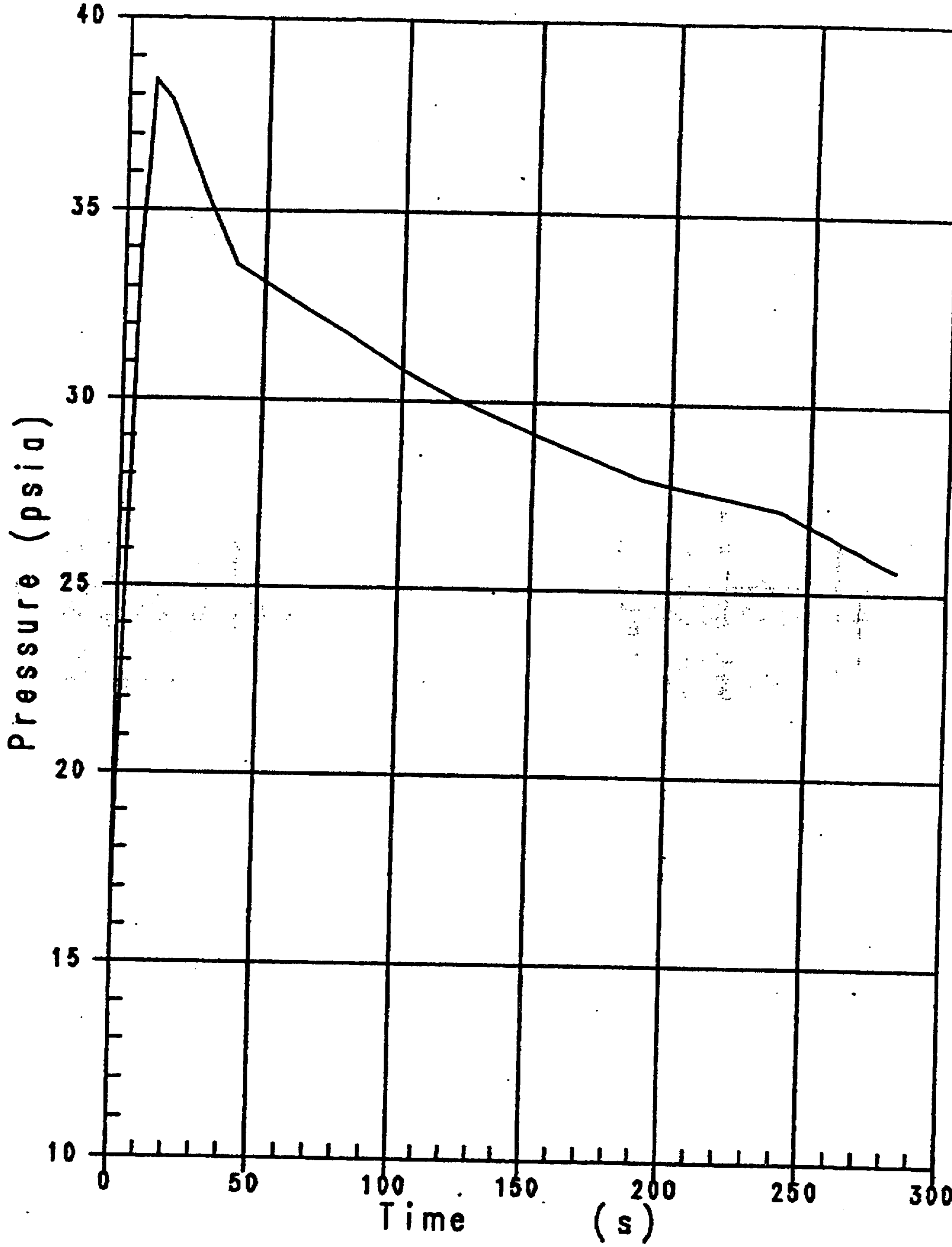

Figure 2 Containment Backpressure for IP3

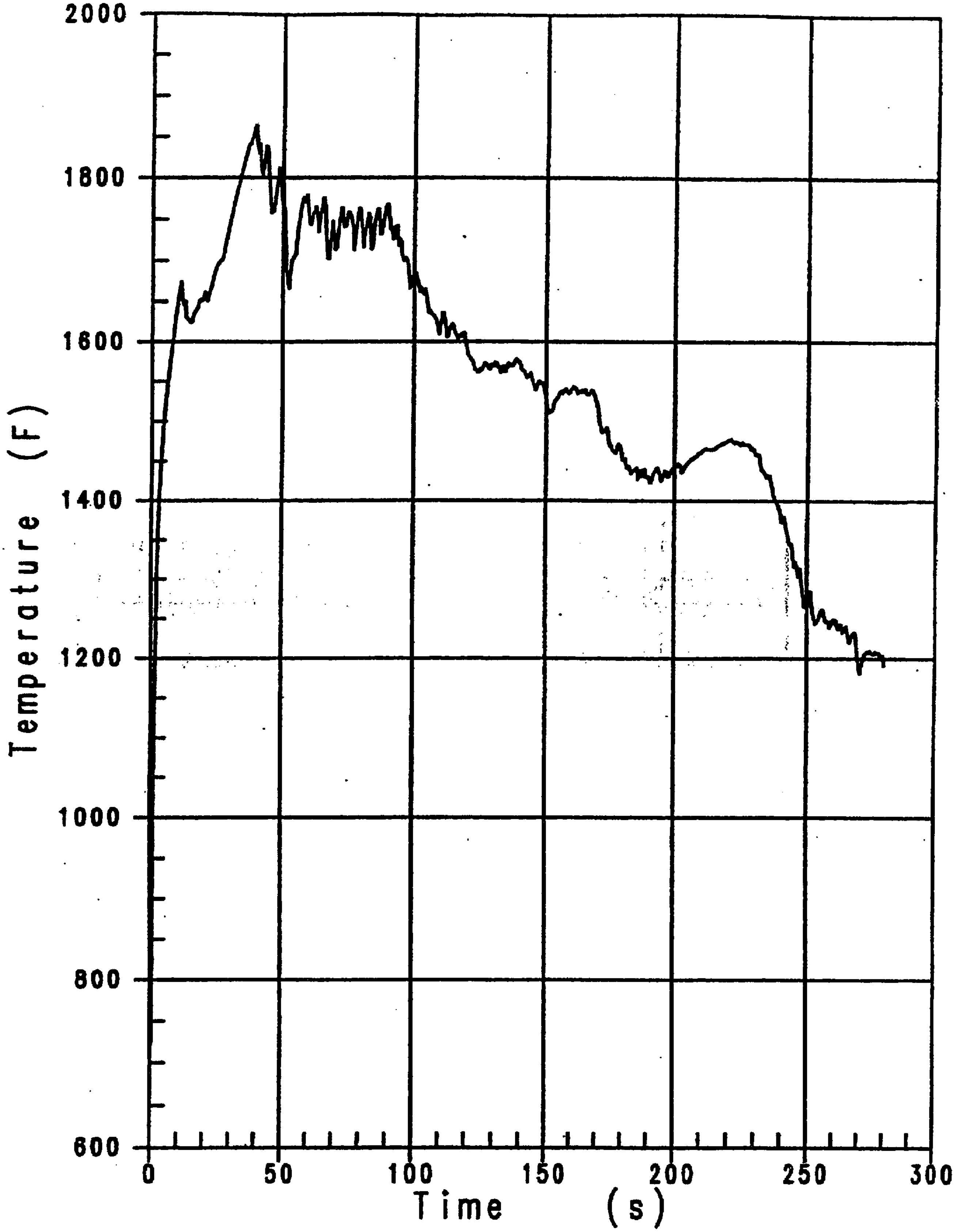

Figure 3 Peak Cladding Temperature for Reference Transient - IP3

ATTACHMENT II TO IPN-02-052

MARKUP PAGES FOR THE PROPOSED CHANGES

TO TECHNICAL SPECIFICATION 5.6.5.b

REGARDING APPROVAL AND USE OF BEST ESTIMATE LARGE

BREAK LOCA ANALYSIS

ENTERGY NUCLEAR OPERATIONS, INC.
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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

3a. ~~WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));~~
WCAP-12945-P-A, Volume 1 (Revision 2)and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

3b. ~~WCAP - 9561 P A ADD. 3, Rev. I, "BART A - 1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS, SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL, "JULY 1986 (W Proprietary).(Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));~~
NOT USED

3c. ~~WCAP - 10266 P A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary). (Specification 3.2.1 Heat Flux Hot Channel Factor (FQ(Z)));~~
NOT USED

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));

3e. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z))); and

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided for each reload cycle to the NRC.

5.6.6 NOT USED