WESTINGHOUSE CLASS 3

WCAP-13170
Revision 1

SAFETY ASSESSMENT FOR THE INDIAN POINT UNIT 3 FUEL ASSEMBLIES WITH ZIRLO™ CLAD FUEL RODS

Revision 2

January 1992

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FAL-92-053

9201140246 49 pp.

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1.0 INTRODUCTION AND BACKGROUND

1.1 Introduction

The New York Power Authority (NYPA) plans to insert Westinghouse fuel assemblies containing fuel rods fabricated with the advanced zirconium alloy cladding material ZIRLO™ into the Indian Point Unit 3 Cycle 9 core and beyond. These fuel assemblies will have fuel rods fabricated with ZIRLO™ cladding to obtain additional operational benefit from the cladding's improved corrosion resistance. NYPA made the transition to VANTAGE 5 fuel in Indian Point Unit 3 for Cycle 7 as described in the submittal to the NRC dated January 20, 1989 (IPN-89-007). Indian Point Unit 3 is currently operating in its second cycle with Westinghouse VANTAGE 5 fuel.

This report will show, based on both evaluations and analyses, that no unreviewed safety questions exist as a result of inserting ZIRLO™ clad fuel rods into the Indian Point Unit 3 reactor core. This report will also show that the subsequent proposed changes to the Indian Point Unit 3 Technical Specifications will not involve significant hazard considerations.

1.2 Background

Westinghouse has developed a new zirconium based fuel rod clad alloy, known as ZIRLO™, to enhance fuel reliability and achieve extended burnup. This alloy provides significant improvement in fuel rod clad corrosion resistance and dimensional stability under irradiation. ZIRLO™ cladding corrosion resistance has been evaluated in long-term, out-of-pile tests over a wide range of temperatures (up to 600°F in water tests, up to 932°F in steam tests). Additional tests have also been conducted in lithiated water environments. The improved corrosion resistance of ZIRLO™ cladding has also been demonstrated to very high burnups in the BR-3 reactor.

A conditional licensing approval for the use of this advanced alloy cladding in two demonstration fuel assemblies for the North Anna Unit 1 reactor core was given in a USNRC letter dated May 13, 1987. The USNRC granted an exemption[(1)] from the provision of 10CFR50.46, 10CFR50.44 and 10CFR51.52 with respect to the use of the North Anna demonstration fuel assemblies with the advanced cladding material, ZIRLO™. The information required to support the licensing basis for the implementation of the ZIRLO™ clad fuel rods in Indian Point Unit 3 is given in References 2 and 3. The fuel assemblies will be utilized in Indian Point Unit 3, beginning with Cycle 9, scheduled to start in the second quarter of 1992.

1.3 Areas Assessed

The following areas have been assessed during the safety evaluation process: chemical/mechanical properties, neutronic performance, thermal and hydraulic performance, cladding performance under non-LOCA conditions, and cladding performance under LOCA conditions. These areas are discussed in detail in Section 3.0.

Reference 6 addresses the VANTAGE 5 design and its application to a 17x17 fuel assembly. The VANTAGE 5 design may be applied to other fuel assembly arrays (14x14, 15x15) where such applications are evaluated on a plant specific basis and licensed in accordance with NRC requirements. Indian Point Unit 3 has been licensed for 15x15 VANTAGE 5 as noted in Section 1.1. Subsequently, the applicable models and methods employed to address the 15x15 VANTAGE 5 design have been licensed for Indian Point Unit 3. The principal difference between the Indian Point Unit 3 Region 11 fuel and the licensed 15x15 VANTAGE 5 fuel is the use of ZIRLO™ cladding. The use of ZIRLO™ cladding does not alter the previously licensed models and methods of Reference 6 with the exception of the LOCA model and methodology as noted in Section 3.6 of this report. The revised LOCA model and methodology were used as the basis to evaluate the effects of the change in cladding material as described in Section 3.6. These evaluations have shown that the present LOCA related design bases and limits remain valid. Where the models and methods of Reference 6 are not affected by ZIRLO™ cladding, Indian Point Unit 3 plant specific evaluations and analyses have also shown that the current design bases and limits remain valid.

2.0 LICENSING BASIS

2.1 Acceptance Criteria Basis

Based on the design criteria[2], the acceptance criteria for this safety assessment is specified in Reference 3.

2.2 Proposed Technical Specification Change

The Indian Point Unit 3 Technical Specification[4], Design Features Section 5.3.1.A includes the following text (superscripted references in the quoted sections are not included):

1. "The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods,[1] except during Cycle 8 operation. For Cycle 8 operation only, fuel assembly T53 will contain two stainless steel filler rods in place of two fuel rods."

2. "The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.[2]"

3. "Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 4.5 weight percent of U-235."

4. "Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borosilicate glass clad with stainless steel.[4] Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control."

5. "There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.[5]"

In order to allow for the insertion of fuel rods clad with ZIRLO™ alloy in the eighty fuel assemblies, the following revision to Technical Specification[4] Design Features Section 5.3.1.A is proposed (superscripted references in the quoted sections are not included):

1. "The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.[1]"

2. "The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.[2]"

3. "Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 4.5 weight percent of U-235."

4. "Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borosilicate glass clad with stainless steel.[4] Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control."

5. "There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.[5]"

In addition to the above listed change, a change to Section 6.9.1.6 of Indian Point Unit 3 Technical Specifications is recommended. This change adds references to the Small Break Loss of Coolant Accident (LOCA) Evaluation Model. These references are three topicals which describe the methodology used to support the analysis for the Heat Flux Hot Channel Factor in the Core Operating Limits Report. Insert the following three topicals in Section 6.9.1.6 as noted:

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary).

(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3e. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary).

(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).

(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3.0 SAFETY EVALUATION

3.1 Previous Irradiation Experience

Fuel rods fabricated with ZIRLO™ cladding have been previously irradiated in a foreign reactor (BR-3 reactor) at linear power levels up to 17 kw/ft, and burnups significantly greater than those planned for the Indian Point Unit 3 fuel assemblies. Corrosion and hydriding data obtained on the ZIRLO™ cladding were compared with the reference Zircaloy-4 cladding of fuel rods irradiated as controls in the same test assemblies. Based on the irradiation results of the test assemblies in the foreign reactor, the Indian Point Unit 3 ZIRLO™ cladding waterside corrosion and hydriding will be significantly less than that expected for the Zircaloy-4 clad fuel rods. The irradiation test results substantiate a lower clad irradiation growth (Δ L/L) and creepdown for the ZIRLO™ cladding compared to Zircaloy-4 cladding.

Two demonstration fuel assemblies, containing ZIRLO™ clad fuel rods, began irradiation in the North Anna Unit 1 reactor during June 1987. The ZIRLO™ clad fuel rods achieved over 21,000 MWD/MTU burnup in their first cycle (completed during February 1989). Visual inspection during refueling showed no abnormalities. One demonstration assembly with ZIRLO™ clad fuel rods underwent a second cycle of irradiation and achieved over 37,000 MWD/MTU burnup (completed January 1991). Visual inspection of the two cycle ZIRLO™ clad fuel rods during refueling showed no abnormalities. Cladding corrosion measurements showed that the reduced corrosion obtained with the ZIRLO™ clad rods was significantly better than that anticipated on the basis of licensing basis evaluations. The present and future irradiation results are and will be considered in the design of the fuel rods with ZIRLO™ cladding to assure that all fuel rod design bases are satisfied for the planned irradiation life of the Indian Point Unit 3 fuel assemblies.

3.2 Chemical/Mechanical Properties

The chemical composition (see Table 1) of the ZIRLO™ clad fuel rods in the Indian Point Unit 3 fuel assemblies is similar to Zircaloy-4 except for slight reductions in the content of tin (Sn), iron (Fe), and zirconium (Zr) and the elimination of chromium (Cr). ZIRLO™ cladding also contains a nominal amount of niobium (Nb). These small composition changes are responsible for the improved corrosion resistance compared to Zircaloy-4. The physical and mechanical properties are very similar to Zircaloy-4 while in the same metallurgical phase. However, the temperatures at which the metallurgical phase changes occur are different for Zircaloy-4 and ZIRLO™ cladding (Appendix A of Reference 2). These differences are considered in the evaluations discussed below for cladding behavior under non-LOCA and LOCA conditions. Further aspects of the ZIRLO™ cladding performance under LOCA conditions are given in Reference 2. Evaluations have been performed using the NRC approved fuel rod performance code(5) to verify that the fuel rod design bases and design criteria are met for assemblies containing ZIRLO™ clad fuel rods. The fuel rod design bases, criteria and models, which are affected by the use of ZIRLO™ cladding are described in Reference 2.

3.3 Neutronic Performance

The design and predicted nuclear characteristics of fuel rods with ZIRLO™ cladding are similar to those of VANTAGE 5 design(6). The evaluations have shown(2) that the nuclear design bases are satisfied for fuel rods with ZIRLO™ cladding and that the use of ZIRLO™ cladding will not affect the standard nuclear design analytical models and methods to accurately describe the neutronic behavior of fuel rods with ZIRLO™ cladding. The safety limit characteristics of the VANTAGE 5 fuel design(6) are not affected.

3.4 Thermal and Hydraulic Performance

The thermal and hydraulic design bases for fuel rods with ZIRLO™ cladding are identical to those of the VANTAGE 5 design(6). Since the use of the ZIRLO™ clad fuel does not cause changes affecting the parameters which are major contributors in this area (i. e., DNB, core flow, and rod bow), the design bases of the VANTAGE 5 design(6) remain valid.

3.5 Cladding Performance Under Non-LOCA Conditions

The two non-LOCA accidents potentially affected by the use of ZIRLO™ cladding are the Locked Rotor/Shaft Break and RCCA Ejection Accidents. For the Locked Rotor/Shaft Break Accident, it was determined that the ZIRLO™ cladding results in a very small increase in peak clad temperature[2]. However, the effect on the metal-to-water reaction rate is negligible when compared to Zircaloy-4. Sufficient margin exists in the Indian Point Unit 3 safety analysis to accommodate the small PCT increase (approximately 2°F). For the RCCA Ejection Accident, the ZIRLO™ cladding results in a negligible benefit in both the fraction of fuel melting at the hot spot, and the fuel peak stored energy when compared to the results for Zircaloy-4. Thus, the conclusions in the Indian Point Unit 3 FSAR[7] for the two affected non-LOCA accidents remain valid.

3.6 Cladding Performance Under LOCA Conditions

The Loss Of Coolant Accident (LOCA) analyses and evaluations addressing the use of VANTAGE 5 fuel in Indian Point Unit 3 were performed in 1988 using the 1981 Evaluation Model with BART/BASH (Large Break LOCA) and the NOTRUMP Evaluation Model (Small Break LOCA) submitted in January 1989[13]. Modifications to those evaluation models for use in the analyses of fuel with ZIRLO™ cladding have been identified and reported in Reference 2. The modifications include changes to incorporate the effects of ZIRLO™ cladding specific heat, high temperature creep (swelling), burst temperature, burst strain and assembly blockage.

3.6.1 Evaluation Methodology

Reference 2 describes modifications to some portions of the codes which comprise the 1981 Evaluation Model with BART/BASH and the NOTRUMP Evaluation Model used for the analysis of Large Break and Small Break LOCA, respectively. The reference calculations provided primarily in Appendix G of Reference 2 were performed using the described version of the rod heat-up codes and calculating the Peak Clad Temperature for the reference plant assuming a full core of ZIRLO™ clad fuel.

For Large Break LOCA, a single break reanalysis was performed to demonstrate continued conformance to the acceptance criteria of 10CFR50.46 for a core containing ZIRLO™ fuel with or without Integrated Fuel Burnable Absorbers. This analysis duplicated the methods employed for the reference plant analysis as described in Appendix G of Reference 2 for the consideration of ZIRLO™ cladding. The results of this analysis are contained in Appendix A of this report.

The Large Break LOCA reanalysis was performed for the limiting $C_D = 0.4$ discharge coefficient. This discharge coefficient was previously determined to result in the highest calculated Peak Clad Temperature based on the most recent, approved licensing basis Large Break LOCA analysis for Indian Point Unit 3. These spectrum calculations were performed in support of the transition to 15x15 VANTAGE 5 (without IFMs) fuel(13) using the 1981 Evaluation Model with BART/BASH and are currently documented in the Indian Point Unit 3 FSAR(7). The limiting Large Break LOCA $C_D = 0.4$ case represents the most limiting case of all design basis LOCA events, including the Small Break LOCA spectrum, as evidenced by the FSAR(7).

To provide a comparison basis, a base case using models and data appropriate for zircaloy-4 cladding was analyzed. This case was performed with the 1981 Evaluation Model with BART/BASH, including Evaluation Model changes previously reported to the NRC(14)(15) under the reporting requirements of 10CFR50.46. These changes are highlighted to indicate that the models used include such coding and modeling changes. Westinghouse believes that these modeling changes are acceptable, but notes that they were not part of the approved evaluation model. The version of the evaluation model used for the Indian Point Unit 3 Large Break reanalysis includes all such changes and thus represents the latest acceptable methodology employed by Westinghouse for these evaluation models. Based on the studies identified in Appendix G of Reference 2, only the rod heat-up portion of the transient is significantly affected by the ZIRLO™ clad related changes. Thus, the reanalysis to assess the effects of ZIRLO™ clad was performed by rerunning the rod heat-up calculation using the LOCBART code. Specifically the version of LOCBART described in Reference 2 and updated for the modeling of ZIRLO™ cladding material characteristics was employed. By performing an Indian Point Unit 3 specific Large Break LOCA reanalysis, and sensitivities based on the Indian Point Unit 3 Large Break hydraulic transient, the effects of the conservative modeling of the Indian Point Unit 3 containment design are conservatively included in the evaluation. Due to the differences in zircaloy-4 and ZIRLO™ clad strain characteristics, a study of the effects of fuel burnup was also performed for the ZIRLO™ clad fuel to demonstrate the limiting time in fuel life as required by 10CFR50, Appendix K. The results of all these runs are summarized in Section 3.6.2 of this report, and additional detail is provided in Appendix A. Specifics of the ZIRLO™ cladding evaluation for Large Break LOCA are provided in Section 3.6.2.

For Small Break LOCA, a complete break spectrum reanalysis was performed to demonstrate continued conformance to the acceptance criteria of 10CFR50.46 for a core containing ZIRLO™ fuel with or without Integrated Fuel Burnable Absorbers. This analysis duplicated the methods employed for the reference plant analysis as described in Appendix G of Reference 2 for the consideration of ZIRLO™ cladding. The results of this analysis are contained in Appendix B of this report.

The Small Break LOCA reanalysis was performed for 4, 6, and 8 inch equivalent diameter break sizes for Zircaloy-4 fuel. As in the analysis of record, the 6 inch break was found to result in the highest calculated Peak Clad Temperature for Indian Point Unit 3. This break size was then analyzed for ZIRLO™ clad fuel. These spectrum calculations were performed with 15x15 VANTAGE 5 (without IFMs) fuel (as in Reference 14) using the NOTRUMP Evaluation Model.

To provide a comparison basis, a base case using models and data appropriate for Zircaloy-4 cladding was analyzed. This case was performed with the NOTRUMP Evaluation Model, including Evaluation Model changes previously reported to the NRC[15][16] under the reporting requirements of 10CFR50.46. These changes are highlighted to indicate that the models used include such coding and modeling changes. Westinghouse believes that these modeling changes are acceptable, but notes that they were not part of the approved evaluation model. The version of the evaluation model used for the Indian Point Unit 3 Small Break reanalysis includes all such changes and thus represents the latest acceptable methodology employed by Westinghouse for these evaluation models. Based on the studies identified in Appendix F of Reference 2, only the rod heat-up portion of the transient is significantly affected by the ZIRLO™ clad related changes. Thus, the reanalysis to assess the effects of ZIRLO™ clad was performed by rerunning the rod heat-up calculation using the LOCTA-IV code. Specifically the version of LOCTA-IV described in Reference 2 and updated for the modeling of ZIRLO™ cladding material characteristics was employed. Due to the differences in Zircaloy-4 and ZIRLO™ clad strain characteristics, a plant specific study of the effects of fuel burnup was also performed for both Zircaloy-4 and the ZIRLO™ clad fuel. The results of all these runs are summarized in Section 3.6.3 of this report, and additional detail is provided in Appendix B. Specifics of the ZIRLO™ cladding evaluation for Small Break LOCA are provided in Section 3.6.3.

3.6.2 Large Break Evaluation

The evaluation for the effects of ZIRLO™ on Large Break LOCA PCT is based on a plant specific reanalysis of the limiting Indian Point Unit 3 Large Break LOCA case, including the effects of

ZIRLO™ cladding, as described above. The results of this reanalysis are provided in Appendix A. These results demonstrate that the limiting Large Break LOCA PCT for Indian Point Unit 3 is 1894 °F* for non-IFBA ZIRLO™ clad fuel at the beginning of life for the limiting $C_D = 0.4$ break.

To conservatively ensure that the use of ZIRLO™ cladding will satisfy the requirements of 10CFR50.46 with respect to Large Break LOCA, the effects of other known issues will be included prior to the evaluation of regulatory compliance. Temporary and permanent PCT assessments which continue to apply to the Large Break LOCA PCT will be considered in conjunction with the limiting Large Break LOCA results identified above. It is this cumulative PCT that is then compared to the 2200 °F Acceptance Criteria limit of 10CFR50.46 to confirm continued regulatory compliance.

The implementation of ZIRLO™ cladding will begin with Indian Point Unit 3 Cycle 9. The reload process for Cycle 9 will also include implementation of the Power Shape Sensitivity Model (PSSM)[17] for Indian Point Unit 3. Successful implementation of PSSM results in a reduction of 100 °F in the calculated Large Break LOCA Peak Clad Temperature by removal of the temporary 100 °F PCT penalty assessed to address the potential for top-skewed power distributions to be more limiting than the analyzed chopped cosine power distribution. The efforts for the implementation of PSSM for Indian Point Unit 3 Cycle 9 have been completed based on the preliminary core design, and indicate that the 100 °F PCT reduction for Cycle 9, and future cycles, is appropriate. These efforts will be confirmed for the final core design. Therefore, the +100 °F temporary PCT assessment addressing the potential for limiting top-skewed power distributions has not been included in confirming that the implementation of ZIRLO™ cladding will continue to satisfy the requirements of 10CFR50.46 with respect to Large Break LOCA.

Recent hydraulic testing has revealed that the hydraulic resistance for the 15x15 VANTAGE 5 without IFMs fuel may be as much as 10% higher than previously considered. This increase in the hydraulic resistance will affect the results of the Large Break LOCA analysis that forms part of the licensing basis for Indian Point 3. The potential effect on the Indian Point 3 licensing basis analysis has been conservatively evaluated based on the results of sensitivity studies performed using Zion as a representative 4-loop 15x15 plant. The evaluation provided a temporary PCT assessment of

* All Large Break Peak Clad Temperatures reported here include a + 1 °F assessment which accounts for the effects of containment purge.

+40 °F for the Indian Point Unit 3 Large Break LOCA analysis. This increase, when applied to the current Indian Point Unit 3 licensing basis limiting Large Break LOCA analysis PCT (Cycle 8) and combined with the effects of previously reported Evaluation Model changes and other evaluations, resulted in an overall Large Break LOCA PCT less than the 10CFR50.46 limit of 2200 °F. Pending resolution of this issue, the +40 °F Large Break LOCA ΔPCT will continue to be applied to Indian Point Unit 3 in evaluating conformance to the requirements of 10CFR50.46. Therefore, this PCT assessment will also be included in the evaluation of the acceptability of the implementation of ZIRLO™ cladding for Indian Point Unit 3 Cycle 9 and subsequent cycles.

Based on the analysis results identified here and in Appendix A, the limiting Large Break LOCA PCT, including the effects of ZIRLO™ cladding, implementation of PSSM, and the PCT assessment for increased fuel hydraulic resistance, when combined with the effects of all other currently applicable PCT assessments, resulted in an overall Large Break LOCA PCT of 1974 °F. Therefore, the implementation of ZIRLO™ cladding will continue to satisfy the requirements of 10CFR50.46 with respect to Large Break LOCA.

3.6.3 Small Break Evaluation

The evaluation for the effects of ZIRLO™ on Small Break LOCA PCT is based on a plant specific reanalysis of the limiting Indian Point Unit 3 Small Break LOCA case, including the effects of ZIRLO™ cladding, as described above. This reanalysis consisted of a spectrum of break sizes for Zircaloy-4 fuel. The limiting break size was then analyzed for ZIRLO™ clad fuel. The results of this reanalysis are provided in Appendix B. These results demonstrate that the limiting Small Break LOCA PCT for Indian Point Unit 3 is 1470 °F for non-IFBA Zircaloy-4 clad fuel at the beginning of life for the limiting 6 inch break.

To conservatively ensure that the use of ZIRLO™ cladding will satisfy the requirements of 10CFR50.46 with respect to Small Break LOCA, the effects of other known issues will be included prior to the evaluation of regulatory compliance. Temporary and permanent PCT assessments which continue to apply to the Small Break LOCA PCT will be considered in conjunction with the limiting Large Break LOCA results identified above. It is this cumulative PCT that is then compared to the 2200 °F Acceptance Criteria limit of 10CFR50.46 to confirm continued regulatory compliance.

The implementation of ZIRLO™ cladding will begin with Indian Point Unit 3 Cycle 9. Recent hydraulic testing has revealed that the hydraulic resistance for the 15x15 VANTAGE 5 without IFMs fuel may be as much as 10% higher than previously considered. This increase in the hydraulic resistance will affect the results of the Small Break LOCA analysis that forms part of the licensing basis for Indian Point 3. The effect on the Indian Point 3 licensing basis analysis has been determined by including the increased hydraulic resistances in the reanalysis described above. Also included in this reanalysis are several corrections to account for various temporary and permanent PCT assessments which the Indian Point 3 licensing basis PCT is currently including as separate assessments. The final result indicates the conservative nature of the assessments to date, particularly the effect of improved convergence criteria. These plant specific reanalyses show that the Indian Point 3 licensing basis limiting Small Break LOCA analysis PCT (Cycle 8) combined with other evaluations, resulted in an overall Small Break LOCA PCT less than the 10CFR50.46 limit of 2200 °F. Therefore, this PCT assessment will also be included in the evaluation of the acceptability of the implementation of ZIRLO™ cladding for Indian Point Unit 3 Cycle 9 and subsequent cycles.

Based on the analysis results identified here and in Appendix B, the limiting Small Break LOCA PCT, including the effects of ZIRLO™ cladding, increased fuel hydraulic resistance, Small Break LOCA clad burst effects**, and all other currently applicable PCT assessments, resulted in an overall Small Break LOCA PCT of 1505.5 °F. Therefore, the implementation of ZIRLO™ cladding will continue to satisfy the requirements of 10CFR50.46 with respect to Small Break LOCA.

3.6.4 Conclusions

The results of studies performed to assess the effects of ZIRLO™ clad on LOCA PCT have demonstrated continued conformance to all the 10CFR50.46 acceptance criteria. The Large and Small Break PCTs will remain less than 2200 °F with the implementation of ZIRLO™ for Indian Point Unit 3 Cycle 9 and subsequent cycles.

** The effects of fuel burnup were examined through actual Evaluation Model calculations described in Section 3.6.1 and Appendix B to account for fuel rod burst effects, and are included in the results reported in Section 3.6.3.

TABLE 1

NOMINAL COMPOSITION OF ZIRLO™ AND ZIRCALOY-4 CLADDING

Element	Zircaloy-4 (wt %)	ZIRLO™ (wt %)
Sn	1.6	1.0
Fe	0.21	0.1
Cr	0.1	0.0
Nb	0.0	1.0
Zr	> 97.0	> 97.0

4.0 ASSESSMENT OF NO UNREVIEWED SAFETY QUESTIONS (10CFR50.59 Screening Criteria)

The use of the fuel assemblies, containing fuel rods clad with ZIRLO™ material, has been determined not to involve an unreviewed safety question as defined in 10CFR50.59. The basis for this determination is as follows:

1. The probability of occurrence of an accident previously evaluated in the Indian Point Unit 3 FSAR[7] will not be increased with the use of ZIRLO™ clad fuel rods. The clad integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO™ clad fuel rod improves corrosion performance and dimensional stability. Therefore, the probability of occurrence of an accident previously evaluated in the FSAR[7] has not increased.

2. The consequences of an accident previously evaluated in the Indian Point Unit 3 FSAR[7] will not be increased with the use of ZIRLO™ clad fuel rods. The ZIRLO™ clad material is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus, clad integrity is maintained. Since the dose predictions presented in the FSAR[7] are not sensitive to the fuel rod cladding material changes specified in this report, the radiological consequences of accidents previously evaluated in the Indian Point Unit 3 FSAR[7] remain valid.

3. The use of ZIRLO™ clad fuel rods will not create the possibility of an accident of a different type than any previously evaluated in the Indian Point Unit 3 FSAR[7]. The fuel assemblies containing the ZIRLO™ clad fuel rods will satisfy the same design bases[2],[6],[9],[10] as that used for fuel assemblies in the other fuel regions. All design and performance criteria will continue to be met and no new single failure mechanisms have been created. Therefore, the possibility of an accident of a different type than any previously evaluated in the FSAR[7] has not been created.

4. The use of ZIRLO™ clad fuel rods, in compliance with the methodology established in Reference 2, will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[7]. No new performance requirements are being imposed on any system or component such that any design criteria will be exceeded. No new modes or limiting single failures have been created with the ZIRLO™ clad design. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR[7] has not increased.

5. The consequences of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[7] will not be increased with the use of ZIRLO™ clad fuel rods. The dose predictions presented in the FSAR[7] are not sensitive to the fuel rod cladding material. The use of ZIRLO™ cladding material does not change the performance requirements on any system or component such that any design criteria will be exceeded. No new modes or limiting single failures have been created with the ZIRLO™ clad design. Therefore, the radiological consequences of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[7] remain valid.

6. The use of ZIRLO™ clad fuel rods will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the Indian Point Unit 3 FSAR[7]. All original design and performance criteria continue to be met, and no new failure modes have been created for any system, component, or piece of equipment. No new single failure mechanisms have been introduced. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR[7] has not been created.

7. The use of the fuel assemblies, containing fuel rods clad with ZIRLO™ material, will not reduce the margin of safety as defined in the basis for any Technical Specification[4]. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the Technical Specifications[4]. For each cycle reload core, these fuel assemblies will be specifically evaluated using standard reload design methods[11] and approved fuel rod design models and methods[2],[5],[12]. The Indian Point Unit 3 VANTAGE 5 reload design and safety analysis limits will apply. This will include considerations of the core physics analysis peaking factors and core average linear heat rate effects. Therefore, the margin of safety as defined in the

Bases to the Indian Point Unit 3 Technical Specifications(4) and VANTAGE 5 Licensing Amendment Request(13) is not reduced.

Based on the information presented above, it can be concluded that the ZIRLO™ clad fuel rods will perform as well as or better than fuel rods clad with Zircaloy-4 and therefore, using ZIRLO™ cladding does not constitute an unreviewed safety question as defined by 10CFR50.59 (a)(2).

5.0 NO SIGNIFICANT HAZARDS CRITERIA EVALUATION (10CFR50.92 Screening Criteria)

The use of the fuel assemblies, containing fuel rods clad with ZIRLO™ material, has been determined not to involve a significant hazards consideration as defined in 10CFR50.92. The basis for this determination is as follows:

1) The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased. The VANTAGE 5 fuel assemblies containing ZIRLO™ clad fuel rods meet the same fuel assembly and fuel rod design bases as VANTAGE 5 fuel assemblies in the other fuel regions. In addition, the 10CFR50.46 criteria will be applied to the ZIRLO™ clad fuel rods. The use of these fuel assemblies will not result in a change to the proposed Indian Point Unit 3 VANTAGE 5 reload design and safety analysis limits[13]. The ZIRLO™ clad material is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO™ clad fuel rod improves corrosion resistance and dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel rod cladding material changes as specified in this report, the radiological consequences of accidents previously evaluated in the safety analyses remain valid. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2) The possibility for a new or different type of accident from any accident previously evaluated is not created, since the VANTAGE 5 fuel assemblies containing ZIRLO™ clad fuel rods will satisfy the same design bases[2],[6],[9],[10] as that used for VANTAGE 5 fuel assemblies in the other fuel regions. Since the original design criteria is being met, the ZIRLO™ clad fuel rods will not be an initiator for any new accident. All design and performance criteria will continue to be met and no new single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alterations to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3) The margin of safety is not significantly reduced, since the VANTAGE 5 fuel assemblies containing ZIRLO™ clad fuel rods do not change the proposed Indian Point Unit 3 VANTAGE 5 reload design and safety analysis limits[13]. The use of these fuel assemblies containing fuel rods with ZIRLO™ cladding alloy will take into consideration the normal core operating conditions allowed in the Technical Specifications[4]. For each cycle reload core, these fuel assemblies will be specifically evaluated using standard reload design methods[11] and approved fuel rod design models and methods[2],[3],[5],[12]. This will include consideration of the core physics analysis peaking factors and core average linear heat rate effects. In addition, the 10CFR50.46 criteria will be applied each cycle to the ZIRLO™ clad fuel rods. Analyses or evaluations will be performed each cycle to confirm that 10CFR50.46 will be met. Therefore, the margin of safety as defined in the Bases to the Indian Point Unit 3 Technical Specifications[4] and VANTAGE 5 Licensing Amendment Request[13] is not significantly reduced.

Based upon the preceding information, it has been determined that the proposed change, amending the fuel rod clad material description to Zircaloy-4 or ZIRLO™ in the Technical Specifications Design Features Section, 5.3.1.A, does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change to ZIRLO™ meets the requirements of 10CFR50.92(c) and does not involve a significant hazards consideration.

6.0 CONCLUSIONS

The Indian Point Unit 3 Technical Specifications ensure that the plant operates in a manner that provides acceptable levels of protection for the health and safety of the public. The Technical Specifications are based upon assumptions made in the safety and accident analyses, including those relating to the core design. This ensures adequate margin to the regulated acceptance criteria for the accident analyses. Since it has been concluded that the core design parameters and assumptions utilized in the accident analyses are appropriate with consideration for the introduction of ZIRLO™ clad fuel rods, the conclusions in the Indian Point Unit 3 FSAR are valid. Therefore the regulated margin of safety as defined in the Bases of the Technical Specifications is not affected by the use of ZIRLO™ cladding in Indian Point Unit 3.

Based on the acceptance criteria as specified in Section 2.1 of this report, and the evaluations and analyses results as specified in Section 3.0 of this report, it has been demonstrated in Section 4.0 of this report that no unreviewed safety question, as defined in 10CFR50.59, exists. The Technical Specification changes specified in Section 2.2 result in no significant hazards consideration, as defined in 10CFR50.92.

7.0 REFERENCES

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2. Davidson, S. L., and Nuhfer, D. L. (Ed.), "VANTAGE + Fuel Assembly Report," WCAP-12610, June 1990.

3. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse),"Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report'", TAC. NO. 77258, July 01, 1991.

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11. Davidson, S. L. (Ed.), et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (Proprietary), July 1985.

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14. Letter from J. C. Brons (New York Power Authority) to NRC Document Control Desk, "Indian Point 3 Nuclear Power Plant ECCS Evaluation Models," IPN-90-008, February 14, 1990, Docket Number 50-286.

15. Letter from R. E. Beedle (New York Power Authority) to NRC Document Control Desk, "Indian Point 3 Nuclear Power Plant Report of ECCS/Evaluation Model Changes," IPN-91-028, July 26, 1991, Docket Number 50-286.

16. Letter from S. F. Stimac (Commonwealth Edison) to Dr. Thomas E. Murley (NRC), "Zion Station Units 1 and 2 Application for Amendment to Facility Operating Licenses DPR-39 and DPR-48 and Appendix A, Technical Specifications," July 10, 1991, Docket Numbers 50-295 and 50-304.

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APPENDIX A

REVISED LARGE BREAK LOCA ANALYSIS FOR INDIAN POINT UNIT 3 FUEL INCLUDING THE EFFECTS OF ZIRLO™ CLADDING

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1.0 Introduction and Background

This section provides a single break reanalysis for Indian Point Unit 3 and includes the effects of the ZIRLO™ clad fuel on the Large Break LOCA accident analysis. Specifically, the Large Break LOCA analysis for Indian Point Unit 3 was performed for the 15x15 VANTAGE 5 fuel design. The following design features were addressed in these analyses:

1) axial blankets,
2) currently residing zircaloy-4 fuel rod cladding, and
3) the reload ZIRLO™ fuel rod cladding, with or without Integral Fuel Burnable Absorbers (IFBA).

In addition, the following assumptions were made:

1) $F_{\Delta H} = 1.62$,
2) $F_Q = 2.32$, and
3) Fuel temperatures and pressures specific to the fuel cladding being analyzed (zircaloy-4 or ZIRLO™ cladding).

Only the limiting double-ended cold leg guillotine (DECLG) break with a discharge coefficient of $C_D=0.4$ was analyzed in the Large Break LOCA analysis. This break was previously determined to be limiting for the Large Break[1]. The intent of this analysis was to demonstrate conformance with the emergency core cooling system (ECCS) Acceptance Criteria as set forth in 10CFR50.46 for Indian Point Unit 3, with the effects of ZIRLO™ cladding included.

The LOCA analysis performed with the Westinghouse 1981 Large Break ECCS Evaluation Model with BART/BASH, including previously approved modifications for the analysis of ZIRLO™ cladding, meet the requirements of the 10CFR50.46 ECCS Acceptance Criteria. For the worst Large Break case (C_D=0.4, DECLG), this analysis resulted in a Peak Clad Temperature (PCT) of 1894 °F* at an F_Q of 2.32 for the ZIRLO™ clad fuel. As a point of reference, the same analysis (C_D = 0.4) for Zircaloy-4 clad fuel resulted in a Peak Clad Temperature of 1891 °F at the same F_Q.

2.0 LOCA Background

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analysis reported here, a major pipe break (Large Break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 square foot (ft^2). This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis.

The ECCS Acceptance Criteria for the LOCA results are described in 10CFR50.46 as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F,

2. The calculated total oxidation of the cladding shall nowhere 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 shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling, and

5. After any calculated successful operation of the ECCS, the calculated core temperature

* All Large Break Peak Clad Temperatures reported here include a + 1 °F assessment which accounts for the effects of containment purge.

shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in the performance of the ECCS following a LOCA.

The Large Break LOCA analysis was performed for Indian Point Unit 3 assuming a full core of 15x15 VANTAGE 5 fuel without IFMs. Separate cases were modeled assuming a full core of fuel utilizing Zircaloy-4 cladding, and a full core of fuel utilizing ZIRLO™ cladding. The Large Break analysis for Zircaloy-4 cladding utilized NRC approved 1981 Westinghouse Large Break ECCS Evaluation Model with BART/BASH. The Large Break analysis for ZIRLO™ cladding utilized a modified version of the NRC approved 1981 Westinghouse Large Break ECCS Evaluation Model with BART/BASH. Modifications were made to the Large Break Evaluation Model computer codes to represent the ZIRLO™ cladding as discussed in Reference 2. A double-ended-cold-leg guillotine (DECLG) break with a discharge coefficient (C_D) of 0.4 was analyzed.

It is noted that the ZIRLO™ specific metal-water reaction discussed in Reference 2 was not used in Large Break LOCA analysis. Instead, the Baker-Just equation as discussed in Appendix K to 10CFR50 was utilized. A chopped cosine power shape was used in the Large Break analysis. The effect of fuel burnup on the results of the Large Break LOCA accident was specifically considered.

3.0 Description of Large Break LOCA Accident

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These actions will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

However, no credit is taken in the LOCA analysis for boron content of the injection water. In addition, the insertion of control rods to shutdown the reactor is neglected in the Large Break analysis, and

2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the Auxiliary Feedwater System. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves and also initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to approximately 615 psia, the accumulators begin to inject borated water into the reactor coolant loops. Since the loss of offsite power is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends after the RCS pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the bypassing of emergency core cooling water injected into the RCS are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill (bottom of core recovery time) until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and subsequently the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

The ECCS pumps continue to supply water to the reactor coolant system during the long term-cooling portion of the transient. Core temperatures are reduced to long-term steady state levels associated with dissipation of residual heat. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the engineered safety features sump by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The Containment Spray System continues to operate to further reduce containment pressure. Approximately 8.2 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

4.0 Method of Analysis - Large Break LOCA Evaluation Model

A description of the Large Break LOCA analysis methodology is given in References 3, 4, and 5. These documents describe the major phenomena modeled in the computer codes, the interface between the computer codes, and the features of codes which ensure compliance with the limits defined by the ECCS Acceptance Criteria. The blowdown depressurization of the RCS is calculated with the SATAN-VI computer code. The WREFLOOD, COCO, and BASH computer codes calculate the RCS response during the refilling and reflooding of the reactor vessel. The LOCBART computer code is used to calculate the fuel cladding temperature and metal-water reaction of the hottest rod in the core.

The mechanistic core heat transfer model in the BART code is incorporated into the analysis to replace the core heat transfer normally performed by LOCTA-IV.

Thermal hydraulic parameters from the reflood portion of the transient and fuel rod conditions from LOCTA are input to the BART code. BART then calculates conditions within the hot assembly rod at all times following the bottom of core recovery (BOC). After the BART calculation, additional LOCTA calculations are performed in which the heat transfer coefficient on the hot rod during reflood is taken directly from the BART calculation.

The requirements of Appendix K to 10CFR50, regarding specific model features, were met by selecting models which provide a significant overall conservatism in the analysis, including the use of the Baker-Just metal water reaction rate equation in the analysis of the ZIRLO™ clad fuel. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K to 10CFR50.

A sensitivity study was specifically performed, as described in Reference 2, to ensure that the effects of the ZIRLO™ clad fuel did not result in a more severe hydraulic transient. The conclusions of this study indicated that only the clad heat-up portion of the transient is significantly affected by the ZIRLO™ clad fuel related changes. Consequently, the LOCA analysis results modeling the ZIRLO™ clad presented herein were determined using only the LOCBART computer code in the Westinghouse ECCS Evaluation Model with BART/BASH.

5.0 Results - Large Break LOCA

Based on the results of the LOCA sensitivity studies, Reference 3, the limiting Large Break was found to be the double-ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the Large Break ECCS performance analysis. Calculations were performed for a break discharge coefficient (C_D) of 0.4. Important characteristics of the reflood transient are depicted in Figures 1 through 3. The results of the calculations modeling zircaloy-4 fuel are summarized in Table 1 and depicted in Figures 4 through 7. The analysis of ZIRLO™ cladding were performed using the models developed specifically for the analysis of ZIRLO™ clad fuel (with the exception of the metal water reaction rate equation). The results of these calculations are summarized in Tables 2 and 3, and are depicted in Figures 8 through 10.

For the C_D = 0.4 DECLG case analyzed, transients of the following parameters are presented:

Figure 1 Reflood Transient Core Inlet Velocity
Figure 2 Reflood Transient Core & Downcomer Levels
Figure 3 Pumped ECCS Flow (Reflood)
Figure 4 Zircaloy-4 Clad Average Temperature-Hot Rod (Burst Location)
Figure 5 Zircaloy-4 Clad Average Temperature-Hot Rod (Peak Location)
Figure 6 Zircaloy-4 Hot Spot Fluid Temperature (Burst Location (- -) Peak Location (—))
Figure 7 Zircaloy-4 Core Heat Transfer Coefficient (Peak Location)
Figure 8 ZIRLO™ Clad Average Temperature-Hot Rod (Peak and Burst Location)
Figure 9 ZIRLO™ Hot Spot Fluid Temperature (Peak and Burst Location)
Figure 10 ZIRLO™ Core Heat Transfer Coefficient (Peak Location)

The maximum PCT calculated for the Large Break is 1894 °F, for ZIRLO™ clad fuel, which is less than the 10CFR50.46 ECCS Acceptance Criteria limit of 2200 °F. The maximum local metal-water reaction is below the embrittlement limit of 17 percent as required by 10CFR50.46. The total metal-water reaction is less than 1 percent, as compared with the 1 percent criterion of 10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided. The burnup application of the ZIRLO™ clad fuel rods demonstrates that the beginning of life remains the most limiting for the Large Break Peak Clad Temperature calculation.

6.0 Conclusions

An analysis was performed using the Westinghouse 1981 Large Break ECCS Evaluation Model with BART/BASH for the analysis of Zircaloy-4 clad fuel, and modified versions of the Westinghouse 1981 Large Break ECCS Evaluation Model with BART/BASH for the analysis of ZIRLO™ clad fuel. Modifications were made to the models to represent the ZIRLO™ cladding properties, as described in Reference 2. The analysis presented in this section shows that the LOCA PCT response for the ZIRLO™ clad fuel rods is similar to that obtained for the Zircaloy-4 clad fuel rods.

The results of burnup sensitivity studies concluded that the beginning of life is limiting for the Large Break application of ZIRLO™ clad fuel rods. In all cases, ample margin to the 10CFR50.46 limits was demonstrated.

The results of this analysis demonstrate that for Large Break LOCA, the emergency core cooling system will meet the acceptance criteria as presented in 10CFR50.46.

7.0 References

1. Letter from J. C. Brons, (New York Power Authority) to NRC Document Control Desk, "Indian Point Unit 3 Nuclear Power Plant, Proposed Changes to Technical Specifications Regarding the Transition to Westinghouse 15x15 VANTAGE 5 Fuel and RTD Bypass Manifold Elimination Modifications," IPN-89-007, January 20, 1989, Docket Number 50-286.

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TABLE 1

Zircaloy-4 Plant Specific Large Break Results

	Zircaloy-4
Hot Rod Limiting Elevation (ft)	6.25
PCT (°F)	1891
PCT Time (sec)	71.96
Oxidation Thickness (%)	4.63
Hot Rod Burst Elevation (ft)	6.00
PCT (°F)	1861
PCT Time (sec)	71.68
Burst Time (sec)	47.29
Rod Pressure at Burst (psi)	580
Burst Temperature (°F)	1588
Heat-up Rate (°F/sec)	11.5
Burst Strain (%)	173.6
Hot Assembly Burst Elevation (ft)	6.25
Burst Time (sec)	57.59
Blockage (%)	53.1
Core Wide Zircaloy-4/Water Reaction (%)	< 1

TABLE 2

ZIRLO™ Plant Specific Large Break Results

	ZIRLO™
Hot Rod Limiting Elevation (ft)	6.25
PCT (°F)	1894
PCT Time (sec)	71.91
Oxidation Thickness (%)	4.65
Hot Rod Burst Elevation (ft)	6.25
PCT (°F)	1894
PCT Time (sec)	71.91
Burst Time (sec)	46.29
Rod Pressure at Burst (psi)	646
Burst Temperature (°F)	1575
Heat-up Rate (°F/sec)	15.0
Burst Strain (%)	135.2
Hot Assembly Burst Elevation (ft)	6.00
Burst Time (sec)	55.16
Blockage (%)	26.2
Hot Assembly Burst Elevation (ft)	< 1

TABLE 3

Sequence of Events

$C_D = 0.4$

Event	Time (sec)
Break occurs	0
Reactor trip signal	0.495
Safety injection signal	0.96
Start accumulator injection	18.3
End of Blowdown	37.686
Bottom of Core Uncovery	51.19
Accumulator empty	61.82
Start pumped ECC injection	25.96
End of bypass	37.686

Figure 1

Reflood Transient Core Inlet Velocity

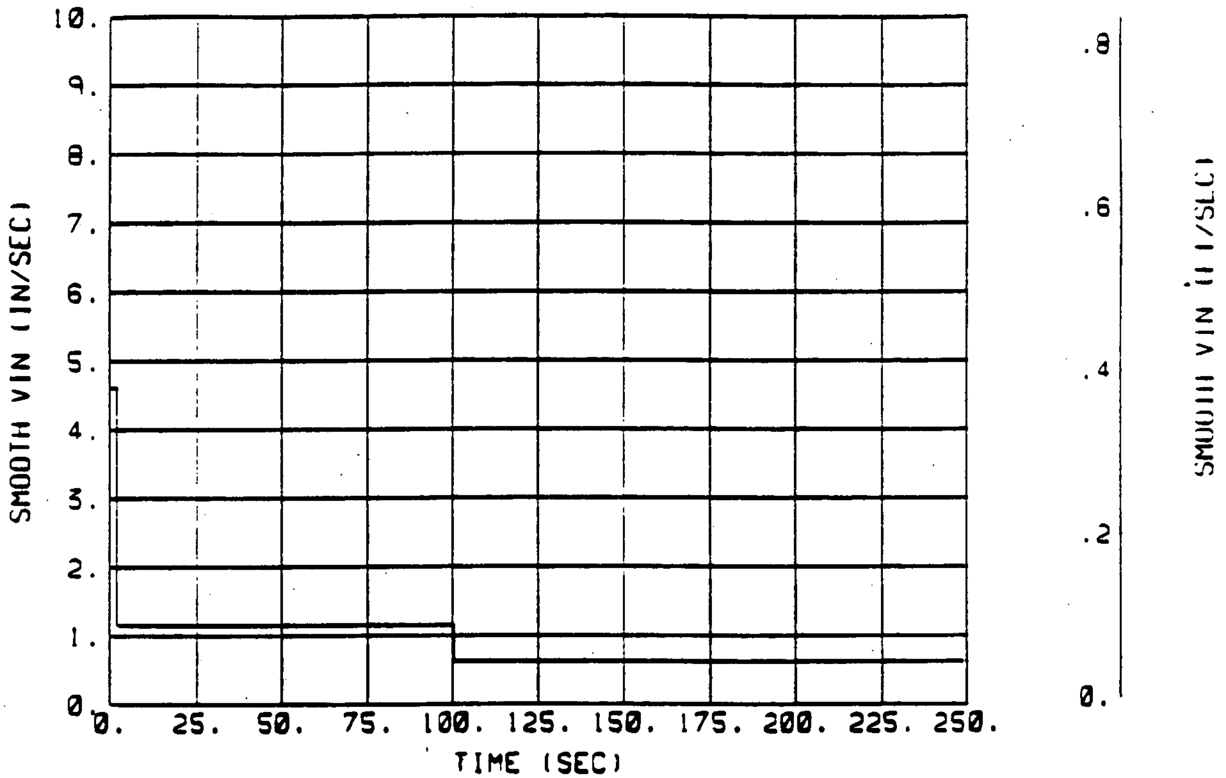


Figure 2

Reflood Transient Core and Downcomer Levels

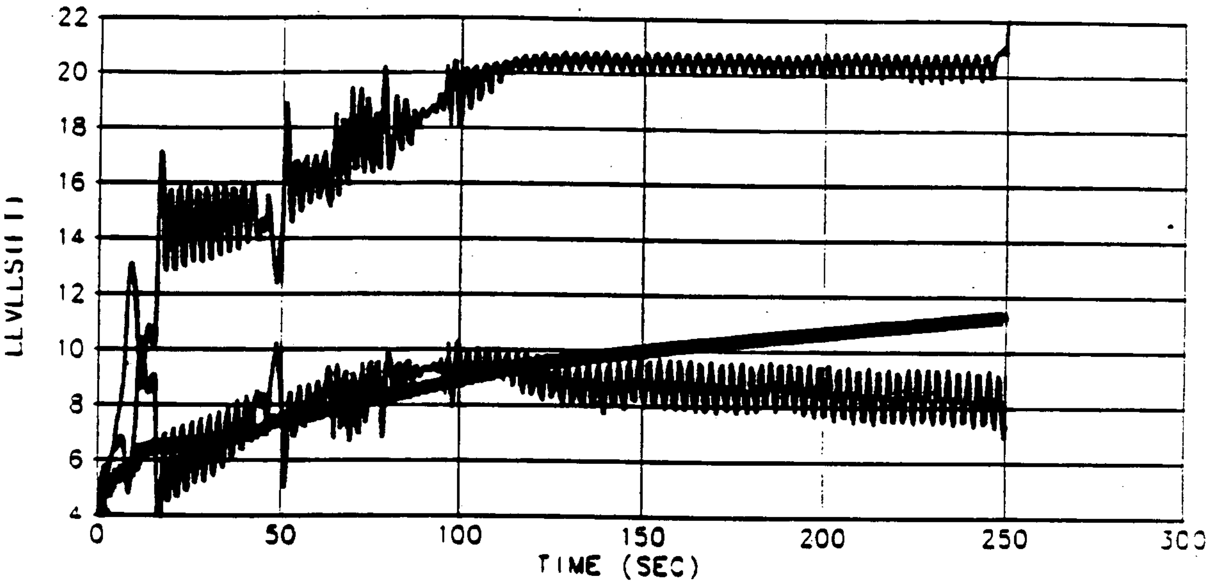


Figure 3

Pumped ECCS Flow (Reflood)

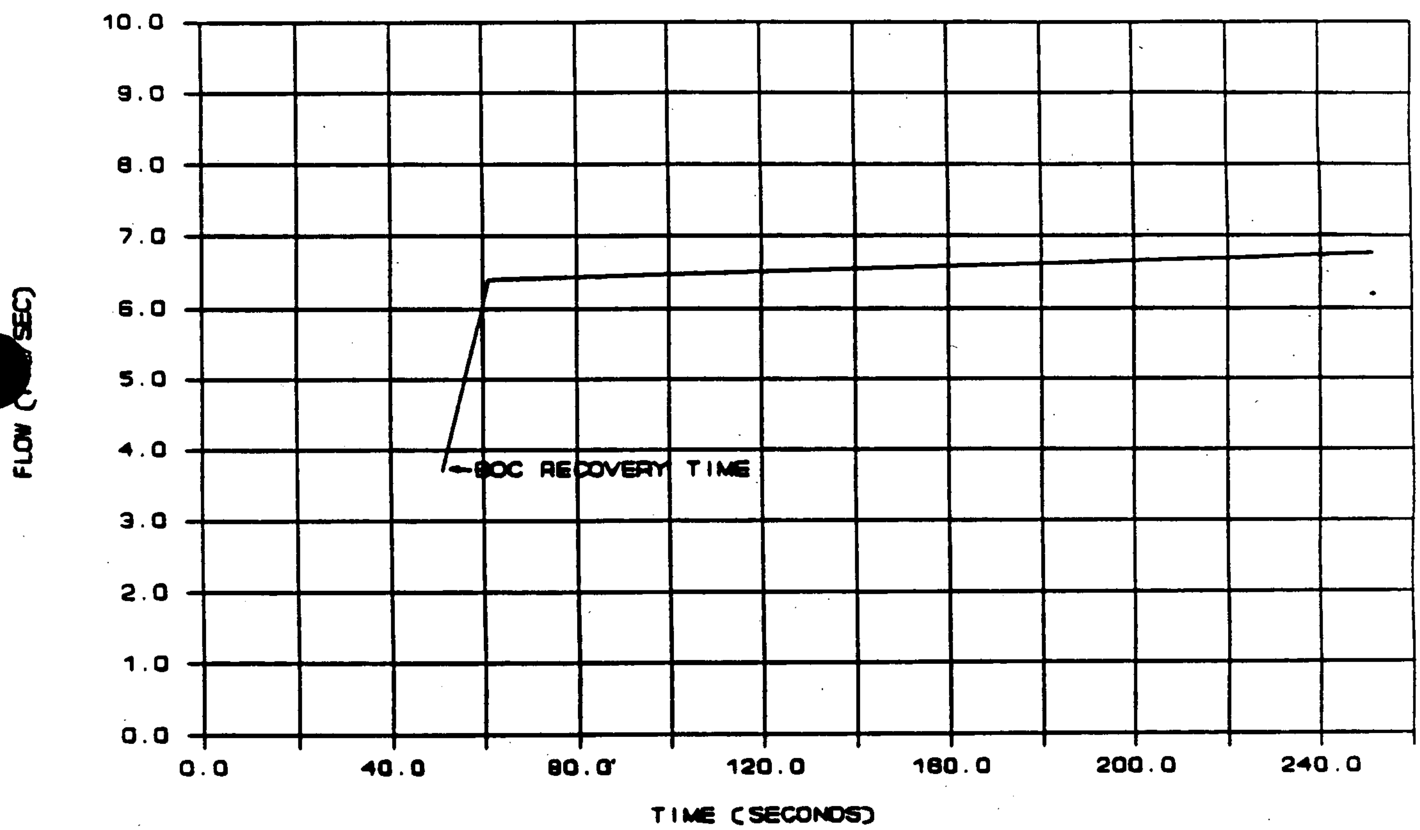


Figure 4

Zircaloy-4 Clad Average Temperature-Hot Rod (Burst Location)

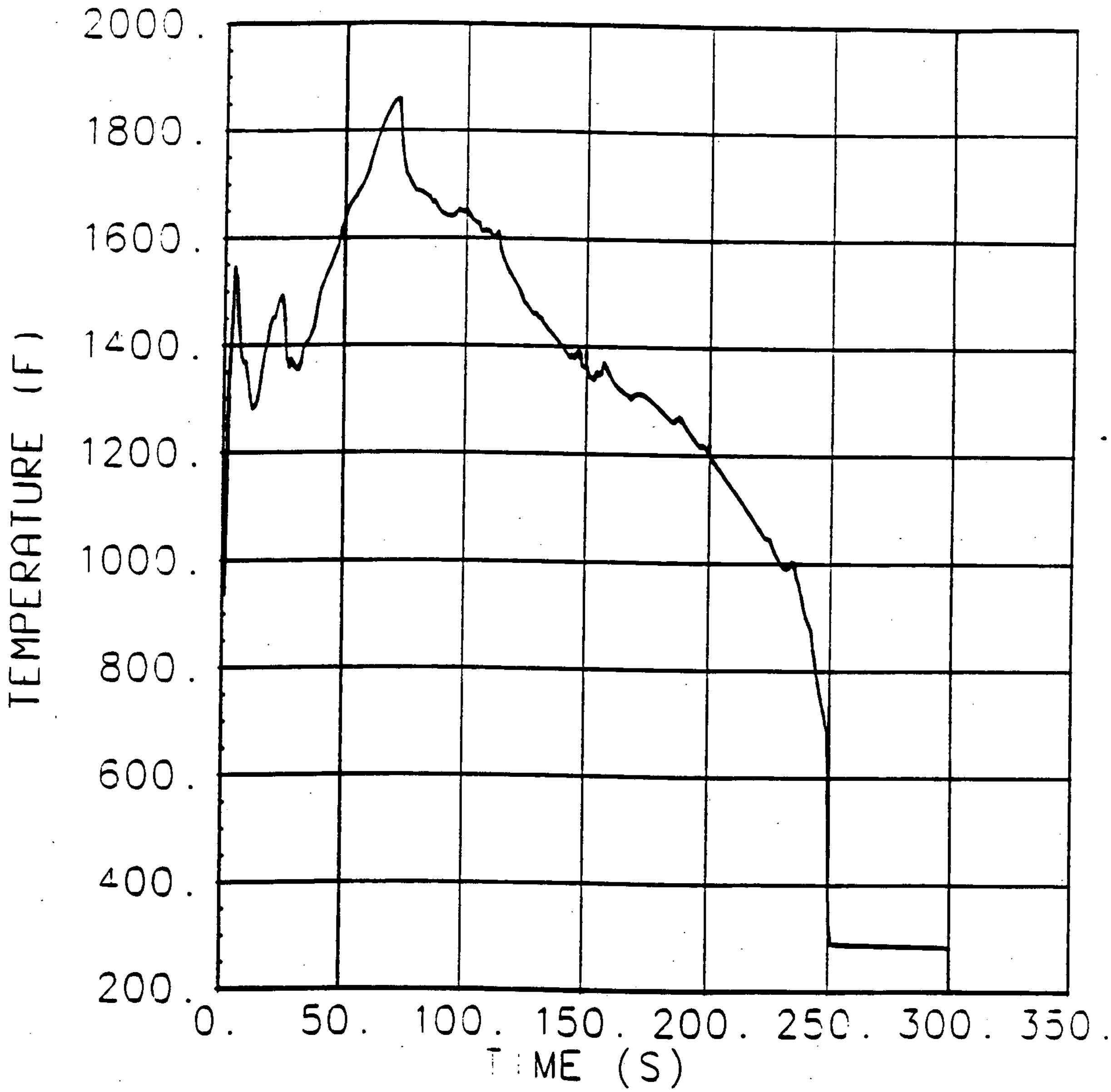


Figure 5

Zircaloy-4 Clad Average Temperature-Hot Rod (Peak Location)

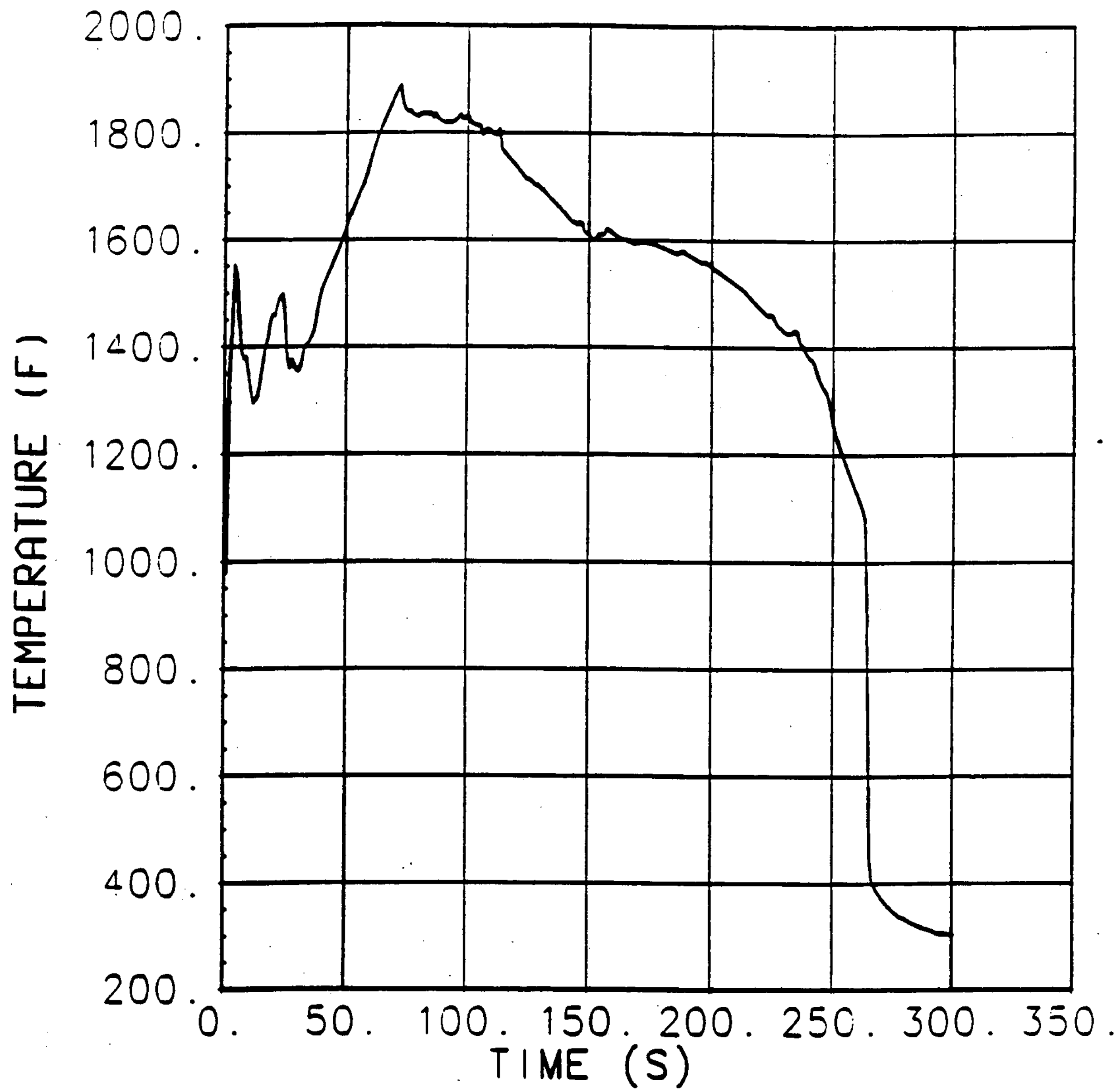


Figure 6

Zircaloy-4 Hot Spot Fluid Temperature (Burst Location (- -) Peak Location (—))

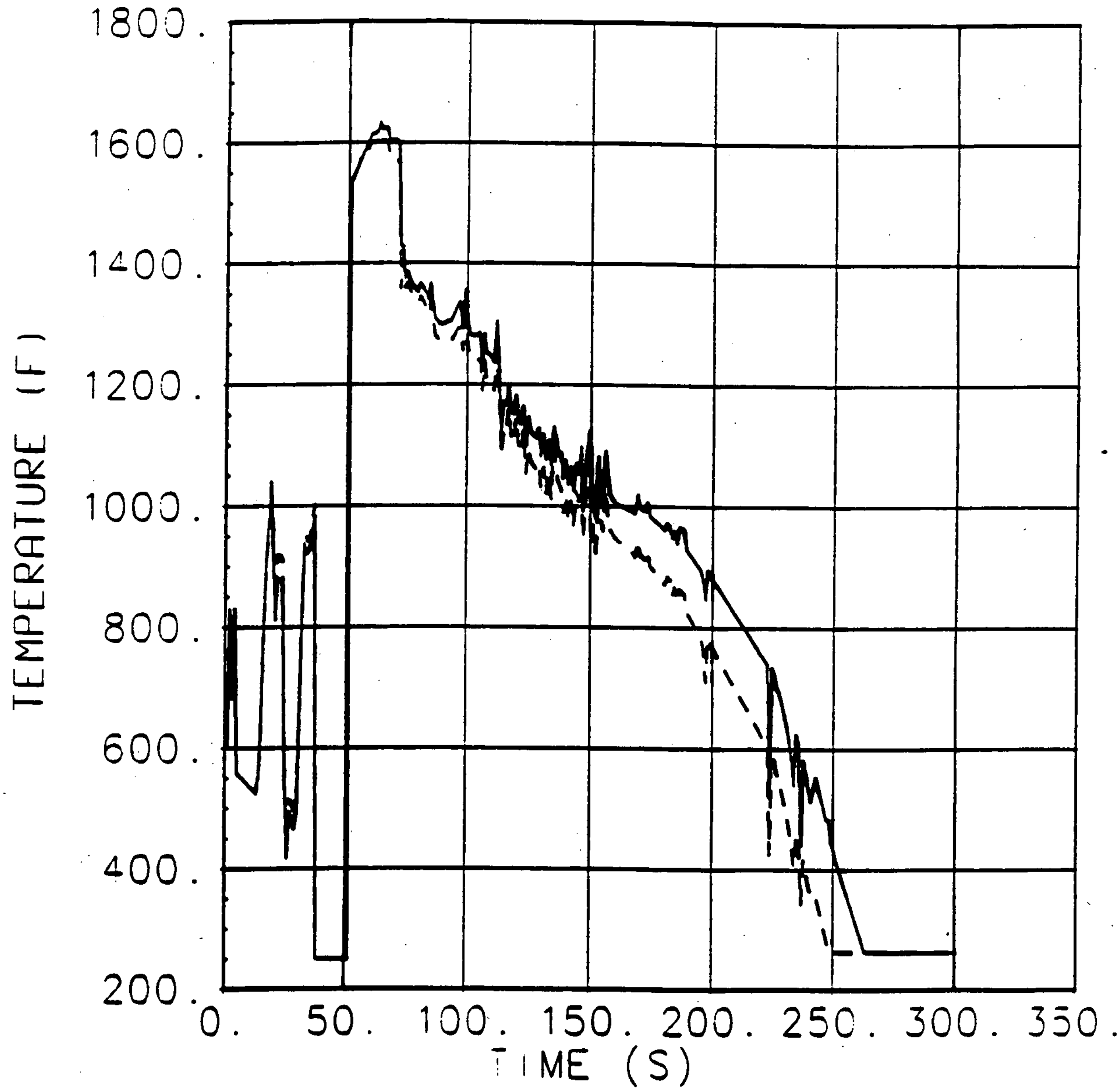


Figure 7

Zircaloy-4 Core Heat Transfer Coefficient (Peak Location)

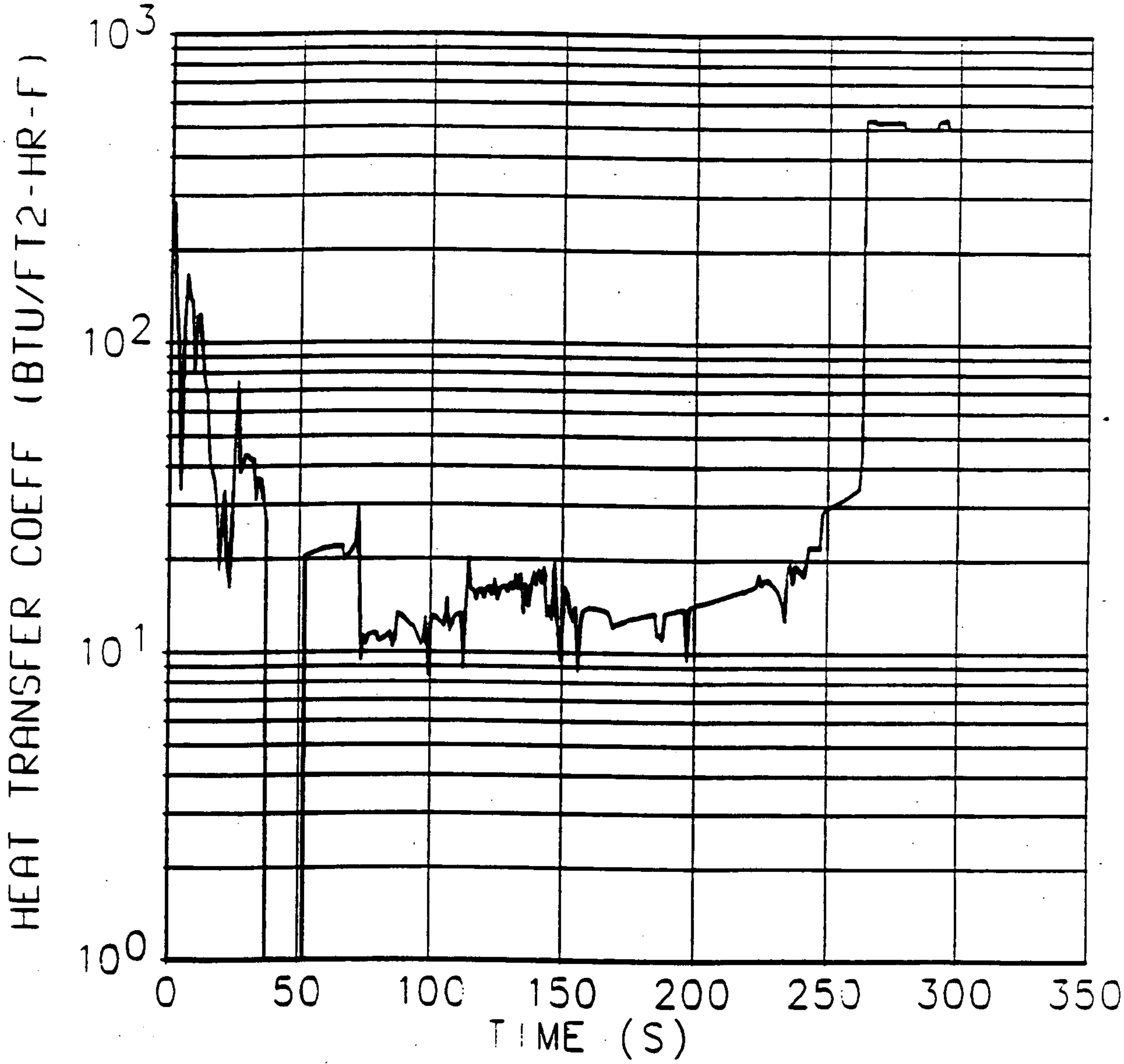


Figure 8

ZIRLO™ Clad Average Temperature-Hot Rod (Peak and Burst Location)

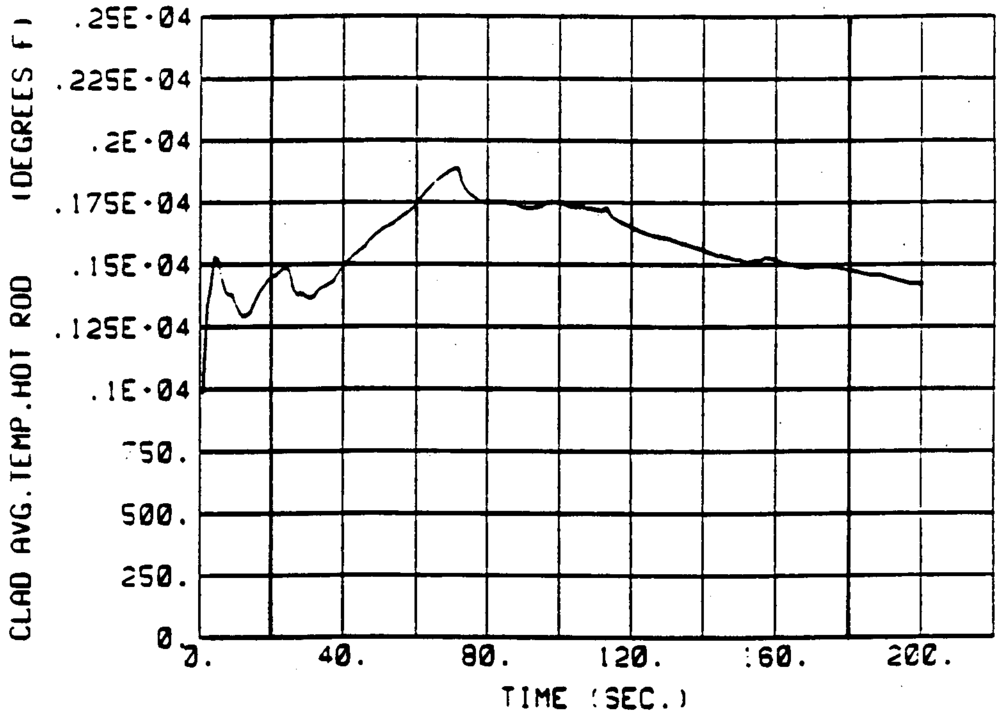


Figure 9

ZIRLO™ Hot Spot Fluid Temperature (Peak and Burst Location)

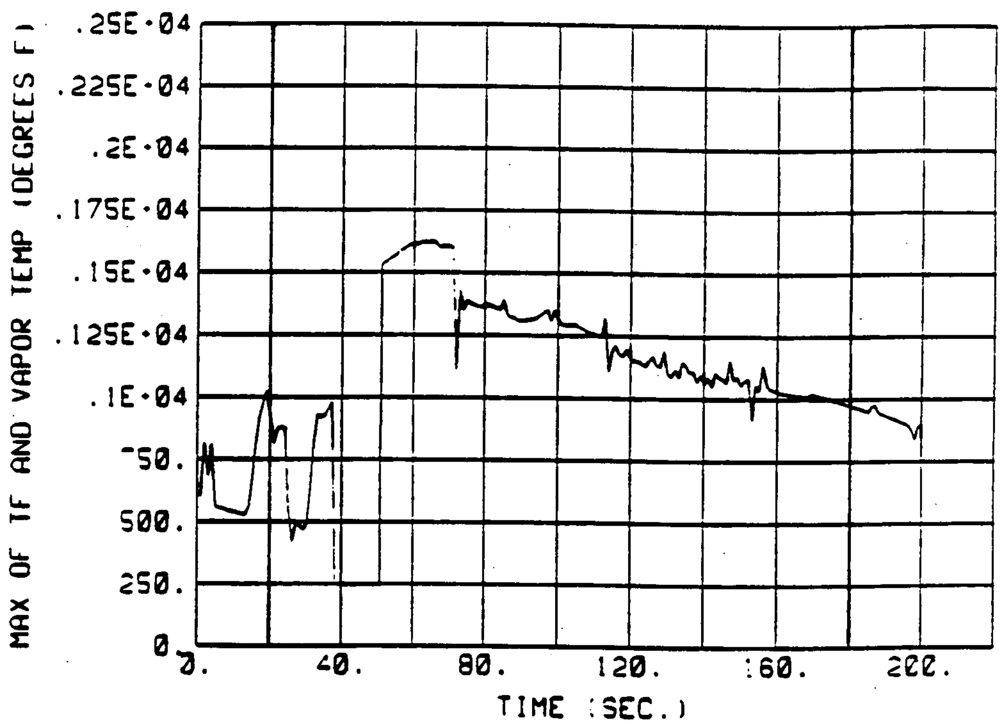
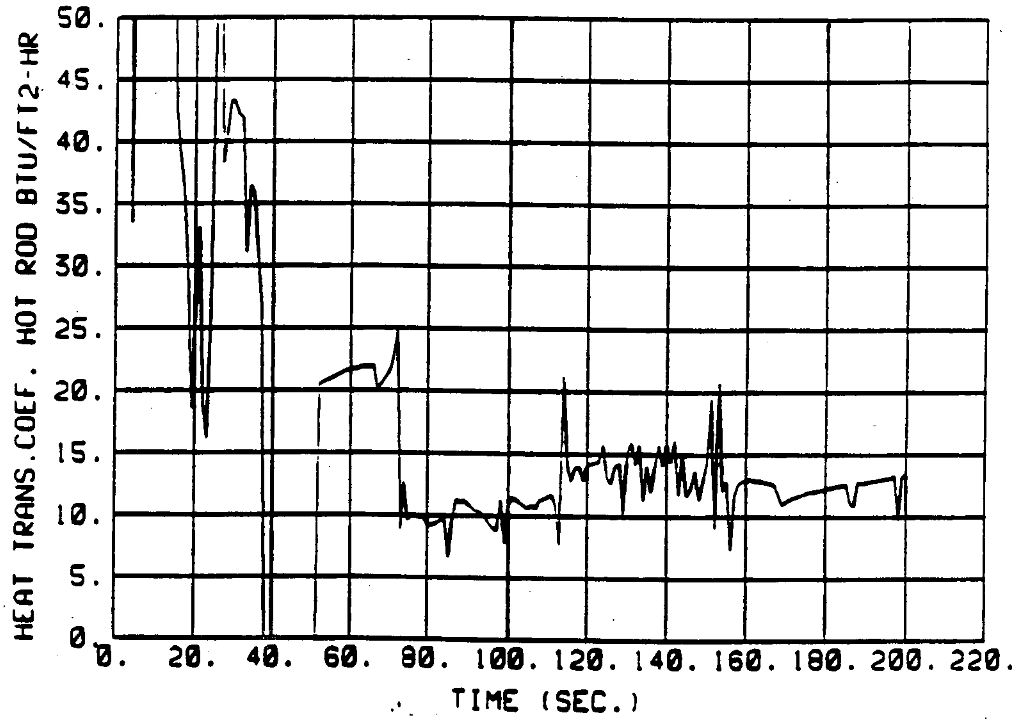


Figure 10

ZIRLO™ Core Heat Transfer Coefficient (Peak Location)



APPENDIX B

REVISED SMALL BREAK LOCA ANALYSIS FOR INDIAN POINT UNIT 3 FUEL INCLUDING THE EFFECTS OF ZIRLO™ CLADDING

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1.0 Introduction and Background

This section provides a complete break spectrum reanalysis for Indian Point Unit 3 and includes the effects of the ZIRLO™ clad fuel on the Small Break LOCA accident analysis. Specifically, the Small Break LOCA analysis for Indian Point Unit 3 was performed for the 15x15 VANTAGE 5 fuel without IFMs design. The following design features were addressed in these analyses:

1) axial blankets,
2) currently residing zircaloy-4 fuel rod cladding, and
3) the reload ZIRLO™ fuel rod cladding, with or without Integral Fuel Burnable Absorbers (IFBA).

In addition, the following assumptions were made:

1) $F\Delta H = 1.62$,
2) $F_Q = 2.42$ to conservatively bound $F_Q = 2.32$, and
3) Fuel temperatures and pressures specific to the fuel cladding being analyzed (Zircaloy-4 or ZIRLO™ cladding).

This reanalysis consisted of a spectrum of break sizes for Zircaloy-4 fuel. The limiting break size was then analyzed for ZIRLO™ clad fuel. The intent of this reanalysis was to demonstrate continued conformance with the emergency core cooling system (ECCS) Acceptance Criteria as set forth in 10CFR50.46 for Indian Point Unit 3, including the effects of ZIRLO™ cladding, as well as the effects of various temporary and permanent PCT assessments documented in previous 10CFR50.46 reporting letters[10][11].

The LOCA analysis performed with the Westinghouse NOTRUMP Small Break ECCS Evaluation Model, including previously approved modifications for the analysis of ZIRLO™ cladding, meets the requirements of the 10CFR50.46 ECCS Acceptance Criteria[11]. For the worst Small Break case (6 inch cold leg break), this analysis resulted in a Peak Clad Temperature (PCT) of 1470 °F at an F_Q of 2.42 for the Zircaloy-4 clad fuel at beginning of life (BOL) conditions. As a point of reference, the same analysis (6 inch break, F_Q=2.42, BOL) for ZIRLO™ clad fuel with and without IFBA resulted in Peak Clad Temperatures of 1466 °F and 1464 °F, respectively.

2.0 LOCA Background

A LOCA is the result of a pipe rupture of the Reactor Coolant System (RCS) pressure boundary. Ruptures of extremely small cross-sections will cause expulsion of the coolant at a rate which can be accommodated by the high head safety injection/charging pumps and which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant with its fission product inventory would be released to the containment.

A small break loss-of-coolant accident, as considered in this section, is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered an ANS Condition III event, in that it may be expected to occur infrequently during the lifetime of the plant.

The ECCS Acceptance Criteria for the LOCA results are described in 10CFR50.46 as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F,

2. The calculated total oxidation of the cladding shall nowhere 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 shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling, and

5. After any calculated successful operation of the ECCS, the calculated core temperature

shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in the performance of the ECCS following a LOCA.

The Small Break LOCA analysis was performed for Indian Point Unit 3 assuming a full core of 15x15 VANTAGE 5 fuel without IFMs. Separate cases were modeled assuming a full core of fuel utilizing Zircaloy-4 cladding, and a full core of fuel utilizing ZIRLO™ cladding. The Small Break analysis for Zircaloy-4 cladding utilized NRC approved NOTRUMP Small Break ECCS Evaluation Model. The Small Break analysis for ZIRLO™ cladding utilized a modified version of the NRC approved NOTRUMP Small Break ECCS Evaluation Model. Modifications were made to the Small Break Evaluation Model computer codes to represent the ZIRLO™ cladding as discussed in Reference 9. A complete spectrum of break sizes were analyzed with Zircaloy-4 clad and the limiting 6 inch break thermal-hydraulic transient was then used to provide boundary conditions for rod heatup calculations for the ZIRLO™ clad fuel.

It is noted that the ZIRLO™ specific metal-water reaction discussed in Reference 9 was not used in the Small Break LOCA analysis. Instead, the Baker-Just equation as discussed in Appendix K to 10CFR50 was utilized. The effect of fuel burnup on the results of the Small Break LOCA accident was specifically considered.

3.0 Description of Small Break LOCA Accident

A loss-of-coolant accident is defined as a rupture of the RCS piping. For Small Break LOCAs, the most limiting single active failure is the one that results in the minimum ECCS flow delivered to the RCS. This has been determined to be the loss of an emergency power train which results in the loss of one complete train of ECCS components. This means that credit was taken for two out of three high head safety injection pumps, and one RHR (low head) pump. During the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (two for each loop). Since the break is postulated to be larger than the diameter of two injection lines (about 2 inches each), two injection lines are assumed to spill to containment backpressure (conservatively assumed to be 0 psig).

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. Loss-Of-Offsite-Power (LOOP) is assumed to occur coincident with reactor trip. A safety injection signal is generated when the appropriate setpoint (pressurizer low pressure SI) is reached. After the safety injection signal is generated, an additional delay ensues. This delay (modeled as 45 seconds to conservatively bound 25 seconds) accounts for the instrumentation delay, the diesel generator start time, plus the time necessary to align the appropriate valves and increase the pumps to full speed. The safety features described will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, in the small break LOCA analysis, credit is taken for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position, and

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is assumed to be in normal plant operation at hot full power, i.e., the heat generated in the core is being removed via the secondary system. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as the pumps coast down following LOOP. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of heat transfer from the RCS to the secondary, heat addition to the secondary results in increased secondary system pressure which leads to steam relief via the safety valves. Makeup to the secondary is automatically provided by the auxiliary feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater control and bypass valves and also initiates motor driven auxiliary feedwater flow. In the Small Break LOCA analysis, flow from a single motor driven auxiliary feedwater pump is assumed to begin 90 seconds after the accident initiation. The secondary flow aids in the reduction of RCS pressure. Also due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analysis.

When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, the vessel mixture level starts to increase to cover the fuel with ECCS pumped injection before the accumulator injection for most breaks.

4.0 Method of Analysis - Small Break LOCA

For small breaks (less than 1.0 ft^2) the NOTRUMP digital computer code[(2)(3)] is employed to calculate the transient depressurization of the Reactor Coolant System as well as to determine the mass and energy of the coolant through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611[(4)].

The reactor coolant system model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the Evaluation Model are provided in References 2 and 3.

Peak clad temperature calculations are performed with the LOCTA-IV code[5] using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions (see Figure 1). Figure 2 depicts the hot rod axial power shape used to perform the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break LOCA analysis assumes the core continues to operate at full power until the control rods are completely inserted. However, for conservatism, it is assumed that the most reactive RCCA does not insert.

After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer pressure reactor trip signal (1815 psia). Soon after the reactor trip signal is generated, the safety injection actuation signal is generated due to a low pressurizer pressure (1689 psia). Considered in the analysis are both gas pressurized accumulator tanks and pumped injection systems. The small break LOCA analysis assumed nominal accumulator water volume with a cover gas pressure of 600 psia. Minimum emergency core cooling system availability is assumed for the analysis at the maximum RWST temperature. Assumed pumped safety injection characteristics as a function of RCS pressure used in the analysis are shown in Figure 3 and in Table 1. The safety injection flow rates presented are based on pump performance curves degraded 5 percent from the design head. The effect of flow from the RHR pumps is not considered in the small break LOCA analyses (except for the 8 inch break size) since their shutoff head is lower than the RCS pressure during the time portion of the transient considered here. For the 8 inch break case, the RHR flows were necessary to overcome the large break flow associated with the larger break size. Safety injection is delayed 45 seconds after the occurrence of the low pressure condition. This accounts for signal initiation, diesel generator startup and emergency power bus loading consistent with the assumed loss of offsite power coincident with reactor trip, as well as the delay involved in aligning the valves and bringing the pumps up to speed. The small break LOCA analysis also conservatively assumed that the rod drop time is 3.4 seconds which bounds the actual value of 2.4 seconds.

On the secondary side, a main feedwater isolation signal is assumed to be generated on safety injection actuation with a two second signal delay and a five second valve closure time. The auxiliary feedwater pumps are assumed to start and deliver full flow (conservatively modeled as the flow from one motor driven pump) at 90 seconds after LOOP. The auxiliary feedwater enthalpy is assumed to be that of the main feedwater until after an additional plant specific feedwater purge time (804.3 seconds for this application) has elapsed.

5.0 Results - Small Break LOCA

5.1 Limiting Case

This section presents results of the limiting small break LOCA analysis (as determined by the highest calculated peak clad temperature) from a range of break sizes, fuel types (ZIRC-4 cladding, ZIRLO™ cladding with and without IFBA), and times in life (fuel burnups). NUREG-0737[6], Section II.K.3.31, required a plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-35[7], generic analyses using NOTRUMP[2][3] were performed and are presented in WCAP-11145[8]. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting. The limiting break size for Indian Point 3 was found to be a 6 inch diameter cold leg break. A list of input assumptions used in the analyses is provided in Table 2. The results of a spectrum analysis (three break sizes) performed for the Zircaloy-4 fuel, as well as limiting break size analyses performed for the ZIRLO™ clad fuel and burnups are summarized in Table 3, while the key transient event times are listed in Table 4. The peak clad temperature in small break LOCA is largely a function of the depth of core uncovery which in turn is dependent on the overall mass inventory and ultimately the primary side pressure. Since the break size is the predominant determiner of the primary pressure transient and the geometry independent fuel characteristics have no direct impact on these parameters, the limiting break size is not expected to change with fuel type or fuel burnup.

Figures 4 through 11 show the following parameters, respectively, for the limiting 6 inch break transient:

- RCS pressure,
- Core mixture level,
- Hot rod clad average temperature,
- Core outlet steam flow rate,
- Hot assembly rod surface heat transfer coefficient,
- Hot spot fluid temperature,
- Cold leg break mass flow rate, and
- Safety injection mass flow rate.

In addition, the following parameters:

- RCS pressure,
- Core mixture level, and
- Hot rod clad average temperature,

are shown for the 4 inch break case in Figures 12 through 14, respectively, while Figures 15 and 16 show the first two parameters for the 8 inch break case. Since the 8 inch break case did not involve core uncovery, no rod heatup calculation was performed and therefore, no plot of hot rod clad average temperature is available.

During the initial period of the small break transient the effect of the break flow rate is not strong enough to overcome the flow rate maintained by the reactor coolant pumps as the pumps coast down following LOOP. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following reactor trip the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the clad temperature transients for the limiting break calculation shown in Figure 6, it is seen that the peak clad temperature occurs near the time (334.3 seconds) when the core is most deeply uncovered and the top of the core is steam cooled. This time is accompanied by the highest vapor superheating above the mixture level. The limiting peak clad temperature during the transient was 1470 °F and was for Zircaloy-4 fuel. At the time the transient was terminated, the safety injection flow rate that was delivered to the RCS exceeded the mass flow rate out the break. The decreasing RCS pressure results in greater safety injection flow as well as reduced break flow. As the RCS inventory continues to gradually increase, the reactor vessel mixture level will continue to increase and the fuel clad temperatures will continue to decline.

The maximum PCT calculated for the Small Break is 1470 °F, for Zircaloy-4 clad fuel, which is less than the 10CFR50.46 ECCS Acceptance Criteria limit of 2200 °F. The maximum local metal-water reaction is below the embrittlement limit of 17 percent as required by 10CFR50.46. The total metal-water reaction is less than 1 percent, as compared with the 1 percent criterion of 10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided. The burnup application of both the Zircaloy-4 and ZIRLO™ clad fuel rods demonstrates that the beginning of life remains the most limiting for the Small Break Peak Clad Temperature calculation.

5.2 Non-limiting Cases

Studies documented in Reference 3 determined that the limiting small-break size occurred for breaks less than 10 inches in diameter. To insure that the 6 inch diameter break was limiting, calculations were run with breaks of 4 inches and 8 inches for the Zircaloy-4 clad fuel. To determine the limiting fuel type and fuel burnup, rod heatup calculations were completed at the limiting break size for other burnups and fuel types.

The results of these calculations are shown in Table 3, and the Sequence of Events Table 4.

5.3 Burnup Studies

The concern over burnup in Small Break LOCA is due to the potential for fuel rod burst which may be calculated to occur as a result of increasing fuel rod internal pressures as a function of burnup. Westinghouse is currently evaluating fuel rod burnup for Small Break LOCA ECCS analyses as it relates to fuel rod swelling and rupture (burst and blockage). Typically, rupture of the fuel cladding is not calculated for beginning of life fuel rod conditions in a Small Break LOCA due to the relatively low fuel rod internal pressure and the smaller differential pressure across the cladding. However, the fuel rod internal pressure increases with burnup and the rupture of the cladding may be calculated for middle or end of life conditions. A rupture of the fuel rod in a Small Break LOCA could result in an increase in the PCT due to flow blockage effects and the effect of the metal water reaction on the inside of the cladding.

For this reanalysis, rod heatup calculations were done at both beginning and end of life conditions to determine if this concern would have any adverse impact on the Indian Point 3 Small Break LOCA analysis. Except for the case assuming end of life conditions (60,000 MWD/MTU) for ZIRLO™ fuel with IFBA, no cladding burst was calculated to occur. In the burst case, the PCT was significantly lower than the result for the beginning of life case. Therefore, fuel burnup is determined not to be an issue for the three fuel types (Zircaloy-4 cladding, ZIRLO™ cladding with and without IFBA) present in the Indian Point 3 Cycle 9 core.

5.4 Auxiliary Feedwater Studies

The Small Break LOCA analyses performed for Indian Point Unit 3 using the NOTRUMP Small Break Evaluation Model assumed the availability of one motor driven auxiliary feedwater pump. This pump was assumed to have a delivery capability of 300 GPM at the set pressure of the lowest steam generator safety valve. Due to asymmetric restrictions on the delivery of auxiliary feedwater, the analyses reported in Tables 3 and 4 were performed assuming delivery to only the broken loop steam generator at 150 GPM. Sensitivity studies to asymmetric auxiliary feedwater flow were performed for the limiting 6 inch break, using a modified NOTRUMP noding scheme which divided the three lumped intact loops into a single intact loop and two lumped intact loops. These studies were performed to confirm the results of previous studies after incorporating the changes described in Section 1.0, particularly the improved convergence criteria. The cases examined were 1) injection of 300 GPM total to two intact loops, and 2) injection of 150 GPM to one intact loop and 150 GPM to the broken loop. These studies calculated lower PCTs than the limiting break value of 1470 °F discussed above for the basic assumption of auxiliary feedwater injection to the broken loop steam generator only. Thus, the operation of one motor driven auxiliary feedwater pump with a capacity of 300 GPM has been shown to result in Small Break LOCA PCT within the criterion of 10CFR50.46.

6.0 Conclusions

Analyses presented in this section show that the high head and low head safety injection of the Emergency Core Cooling System, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below the required limit of 10CFR50.46. Hence adequate protection is provided by the Emergency Core Cooling System in the event of a small break loss-of-coolant accident.

The results of this analysis demonstrate that for Small Break LOCA, the Emergency Core Cooling System will meet the acceptance criteria as presented in 10CFR50.46.

7.0 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Water Cooled Nuclear Power Reactors" 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Meyer, P. E., "NOTRUMP - A nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary), August 1985.

3. Lee, N., et. al., "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," WCAP-10054-P-A (Proprietary), August 1985.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants," NUREG-0611, January 1980.

5. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974, WCAP-8301 (Proprietary), June 1974.

6. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

7. NRC Generic Letter 83-35 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31", November 2, 1983.

8. Rupprecht, S. D., et. al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code," WCAP-11145-P-A (Proprietary), October 1986.

9. Kachmar, M. P., et. al., "Appendix F LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications," WCAP-12610, December, 1990.

10. Letter from J. C. Brons (New York Power Authority) to NRC Document Control Desk, "Indian Point Unit 3 Nuclear Power Plant ECCS Evaluation Models," IPN-90-008, Docket Number 50-

286, February 14, 1990.

11. Letter from R. E. Beedle (New York Power Authority) to NRC Document Control Desk, "Indian Point Unit 3 Nuclear Power Plant Report of ECCS Evaluation Model Changes," IPN-91-028, Docket Number 50-286, July 26, 1991.

TABLE 1

Pumped Safety Injection Flows

RCS Pressure (psia)	Flow (lbm/sec)
14.7	488.7
51.7	486.7
71.7	454.1
91.7	418.5
111.7	379.0
131.7	333.2
151.7	276.2
171.7	185.9
191.7	118.2
214.7	116.9
314.7	111.0
414.7	104.8
514.7	98.2
614.7	91.1
714.7	83.5
814.7	75.2
914.7	66.0
1014.7	55.9
1114.7	44.3
1214.7	30.8
1314.7	13.8
1414.7	0.0

TABLE 2

Initial Input Parameters for Small Break LOCA Analysis

License Core Power[1] (MWt)	3025
Total Peaking Factor, F_Q	2.42
Axial Offset (%)	30
Hot Channel Enthalpy Rise Factor, $E_{\Delta H}$	1.62
Maximum Assembly Average Power, P_{HA}	1.44
Fuel Assembly Array	15x15 V5 w/o IFMs
Accumulator Water Volume (ft^3)	795
Accumulator Tank Volume (ft^3)	1100
Minimum Accumulator Gas Pressure, (psia)	600
Loop Flow (gpm)	80,900
Vessel Inlet Temperature (°F)	537.85
Vessel Outlet Temperature (°F)	602.95
RCS Pressure (psia)	2280
Steam Pressure (psia)	668.18
Steam Generator Tube Plugging Level (%)	30
Maximum Refueling Water Storage Tank Temperature (°F)	120
Maximum Condensate Storage Tank Temperature (°F)	120
Fuel Backfill Pressure (psig)	275/150
Reactor Trip Setpoint (psia)	1815
Safety Injection Signal Setpoint (psia)	1689
Safety Injection Delay Time (sec)	45
Signal Processing Delay and Rod Drop Time (sec)	5.4
Reactor Coolant Pump Delay Time (sec)	4.4
Main Feedwater Isolation Delay Time (sec)	2
Main Feedwater Valve Closure Time (sec)	5
Auxiliary Feedwater Enthalpy Delay Time (sec)	804.3

1 Two percent is added to power to account for calorimetric error.

TABLE 3

Summary of Results

	Zircaloy-4				6" ZIRLO™			
	4 inch BOL	6 inch BOL	8 inch BOL	6 inch EOL	IFBA BOL	IFBA EOL	w/o IFBA BOL	w/o IFBA EOL
Peak Clad Temperature (°F)	975	1470	No Uncovery	1389	1466	1379	1464	1389
Peak Clad Temperature Location (ft)	11.25	11.25	N/A	11.25	11.25	11.0	11.25	11.25
Peak Clad Temperature Time (sec)	869.3	334.3	N/A	308.6	334.3	324.2	334.3	324.5
Local Zr/H_2O Reaction Maximum[1] (%)	0.0361	0.2196	N/A	12.7505	0.2131	6.7613	0.2144	6.6442
Local Zr/H_2O Reaction Location (ft)	11.5	11.25	N/A	11.25	11.25	11.25	11.25	11.25
Total Zr/H_2O Reaction[2] (%)	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	N/A	N/A	N/A	N/A	N/A	303.6	N/A	N/A
Hot Rod Burst Location (ft)	N/A	N/A	N/A	N/A	N/A	11.25	N/A	N/A

Notes:

1. Includes initial pretransient ZrO_2 thickness.
2. Total during Small Break LOCA transient only.

TABLE 4

Time Sequence of Key Events

	Break Size 4"	6"	8"
Start (sec)	0	0	0
Reactor Trip Signal (sec)	11.21	6.28	5.09
Safety Injection Signal (sec)	14.92	8.22	6.94
Pump Injection Begins (sec)	59.92	53.22	51.94
Start of Auxiliary Feedwater Delivery (sec)	101.7	96.5	95.2
Loop Seal Clearing (sec)	290.4	141.0	70.9[1]
Core Uncovery (sec)	629.1	124.2	No Uncovery
Accumulator Injection Begins (sec)	826.5	274.2	183.3
Top of Core Recovery (sec)	971.1	409.1	No Uncovery

Note:

1. This time is for the loop seal clearing in the broken loop.

Figure 1

Code Interface Description for Small Break Model

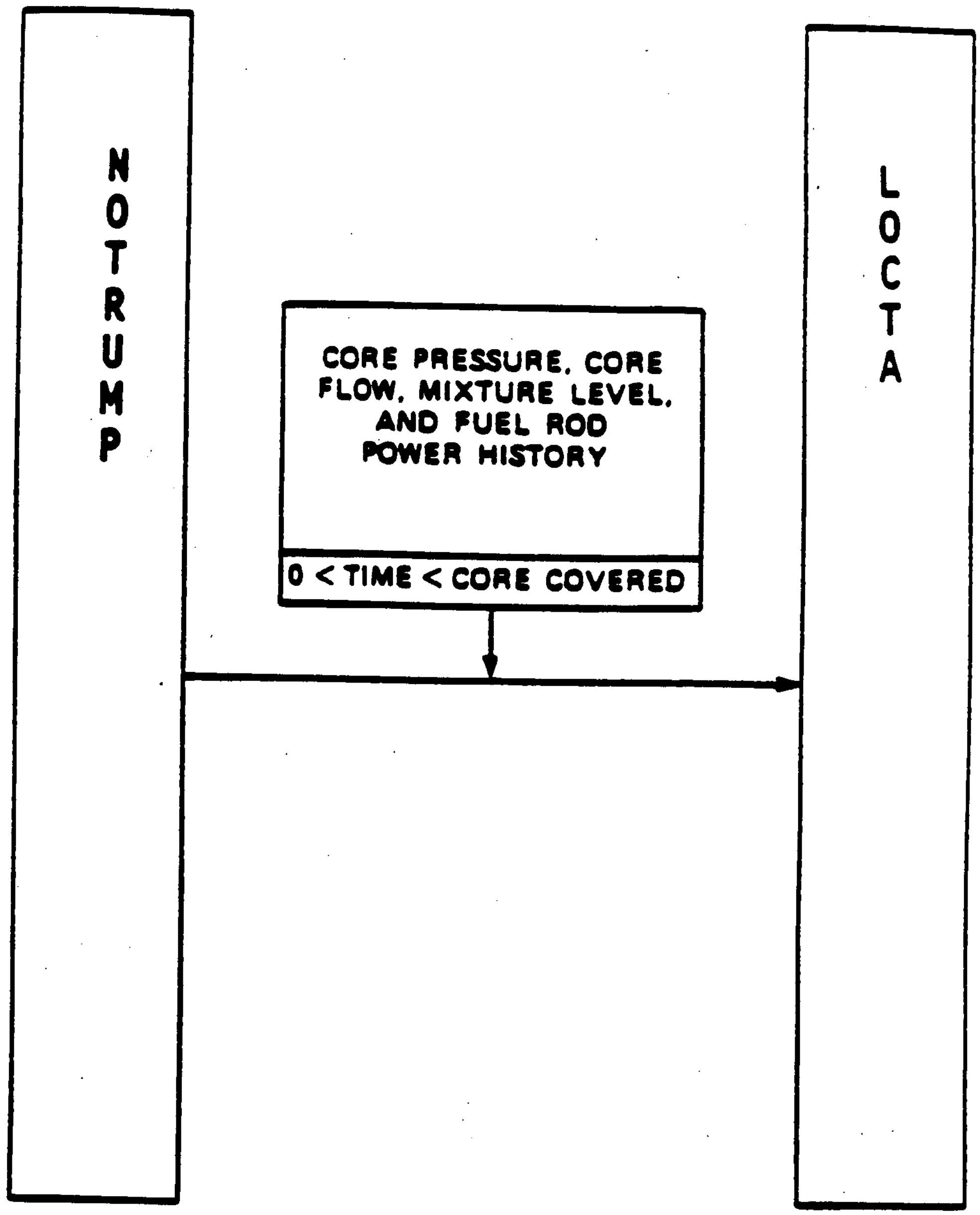

Figure 2

Small Break Hot Rod Power Shape

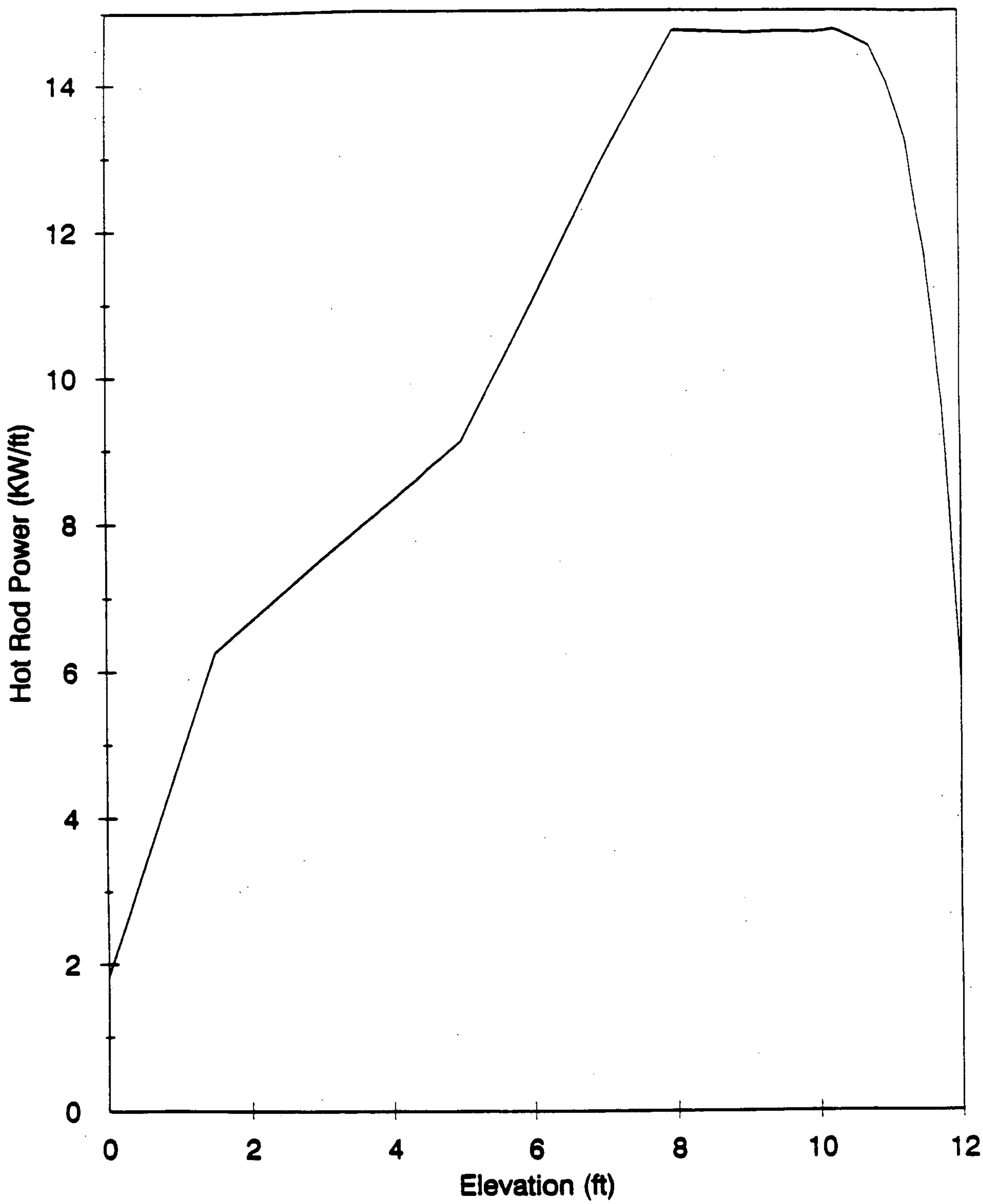

Figure 3

Small Break Safety Injection Flowrate

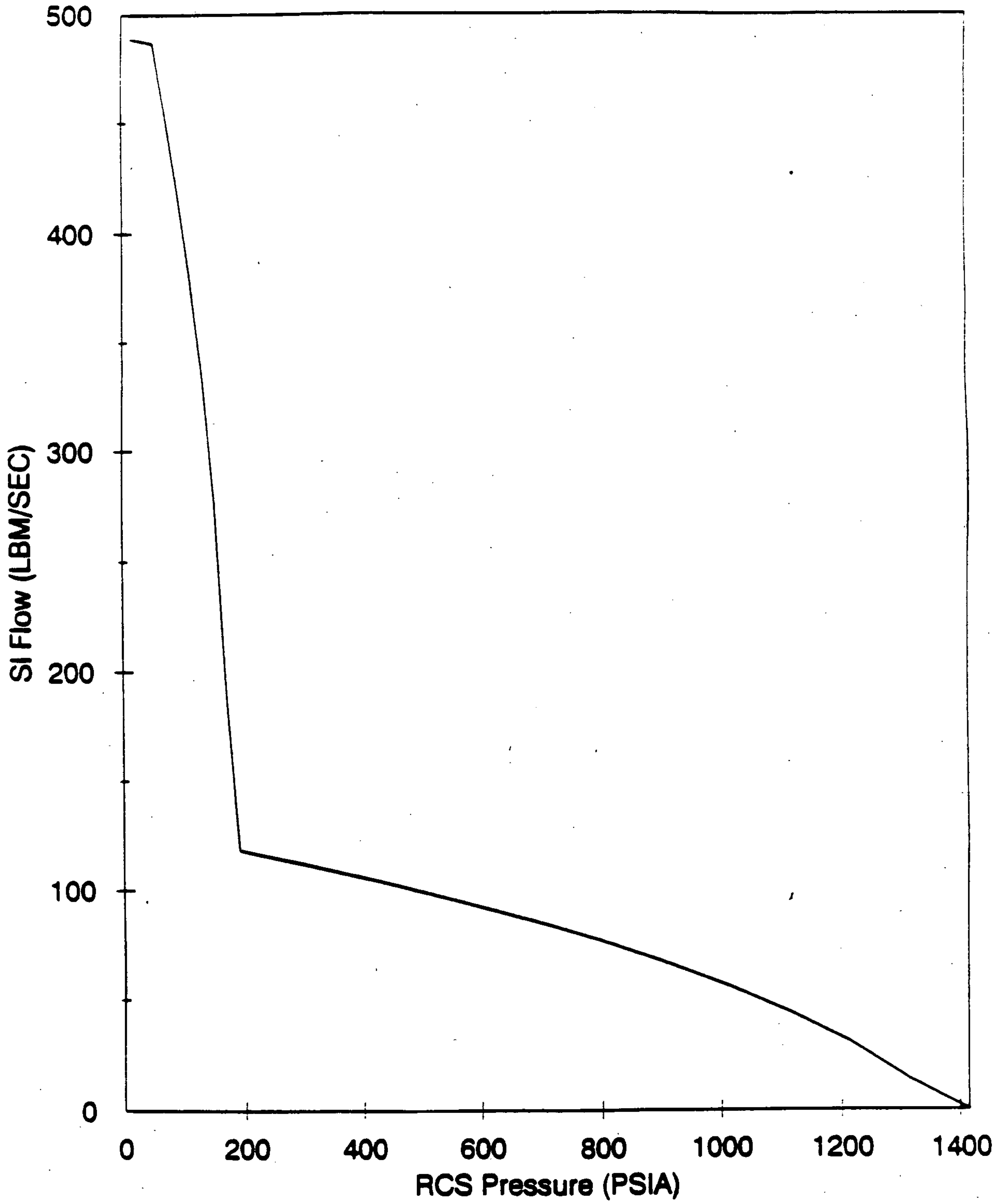

Figure 4

RCS Pressure

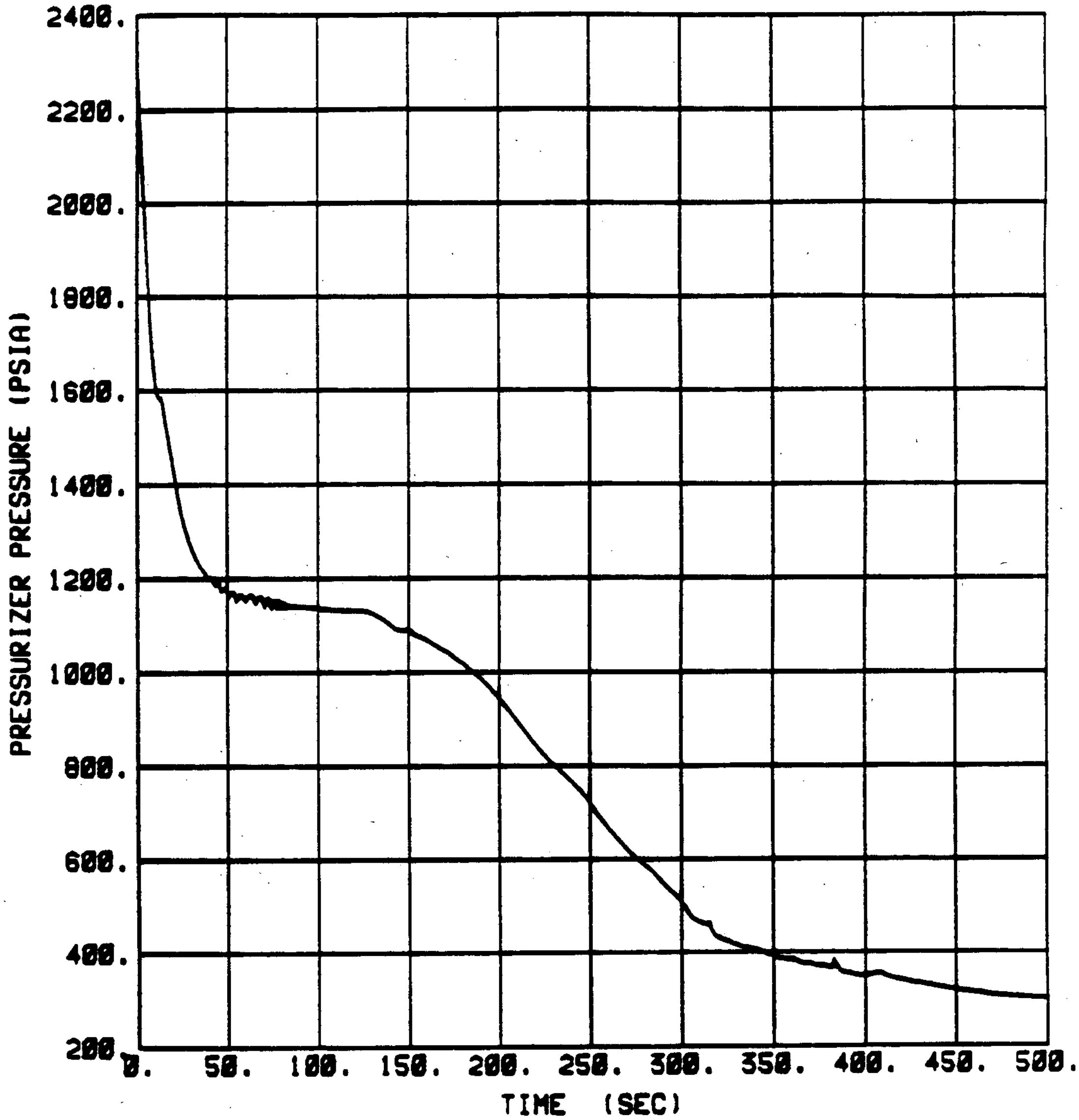

Figure 5

Core Mixture Level

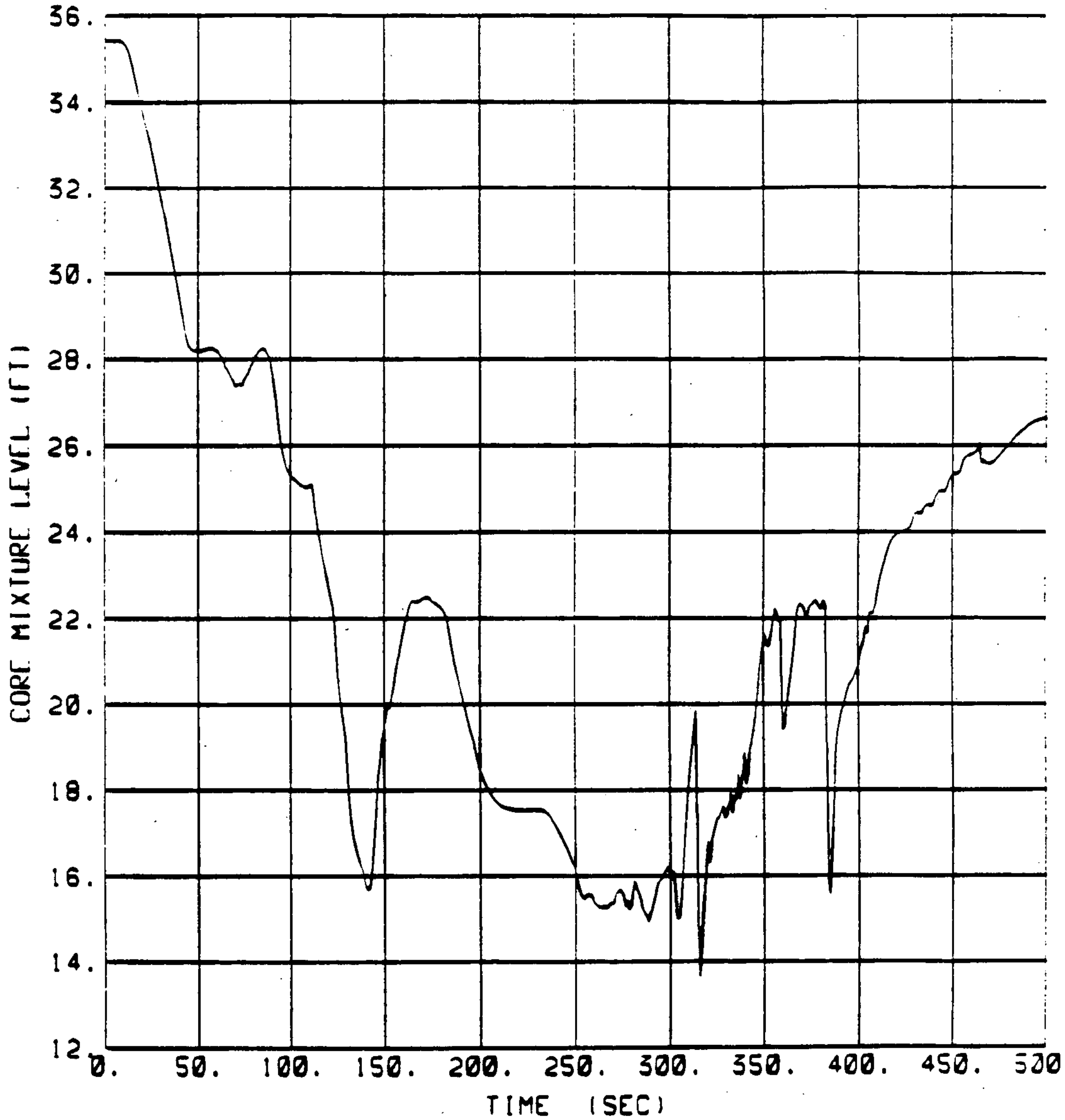

Figure 6

Hot Rod Clad Average Temperature

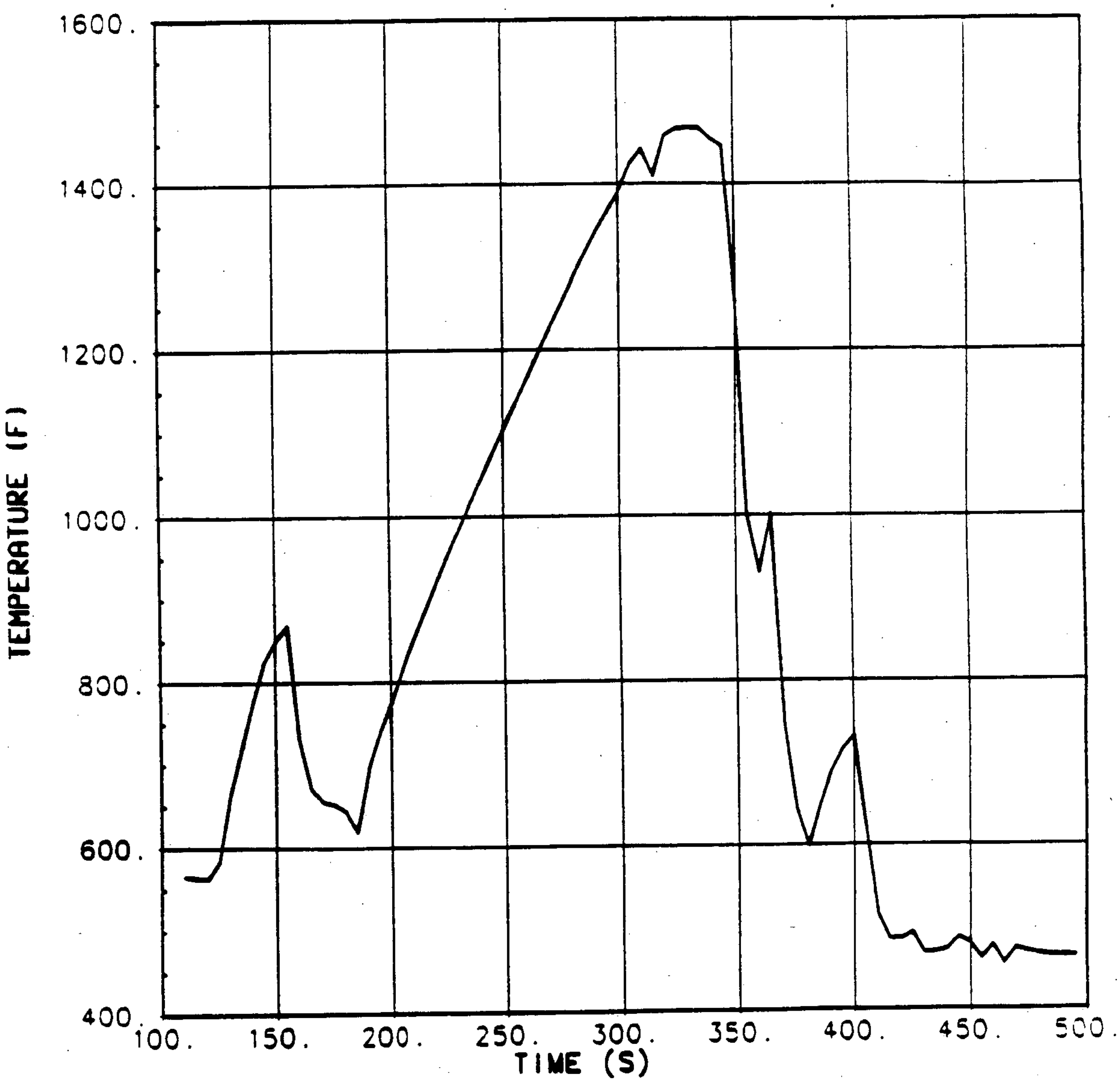

Figure 7

Core Outlet Steam Flowrate

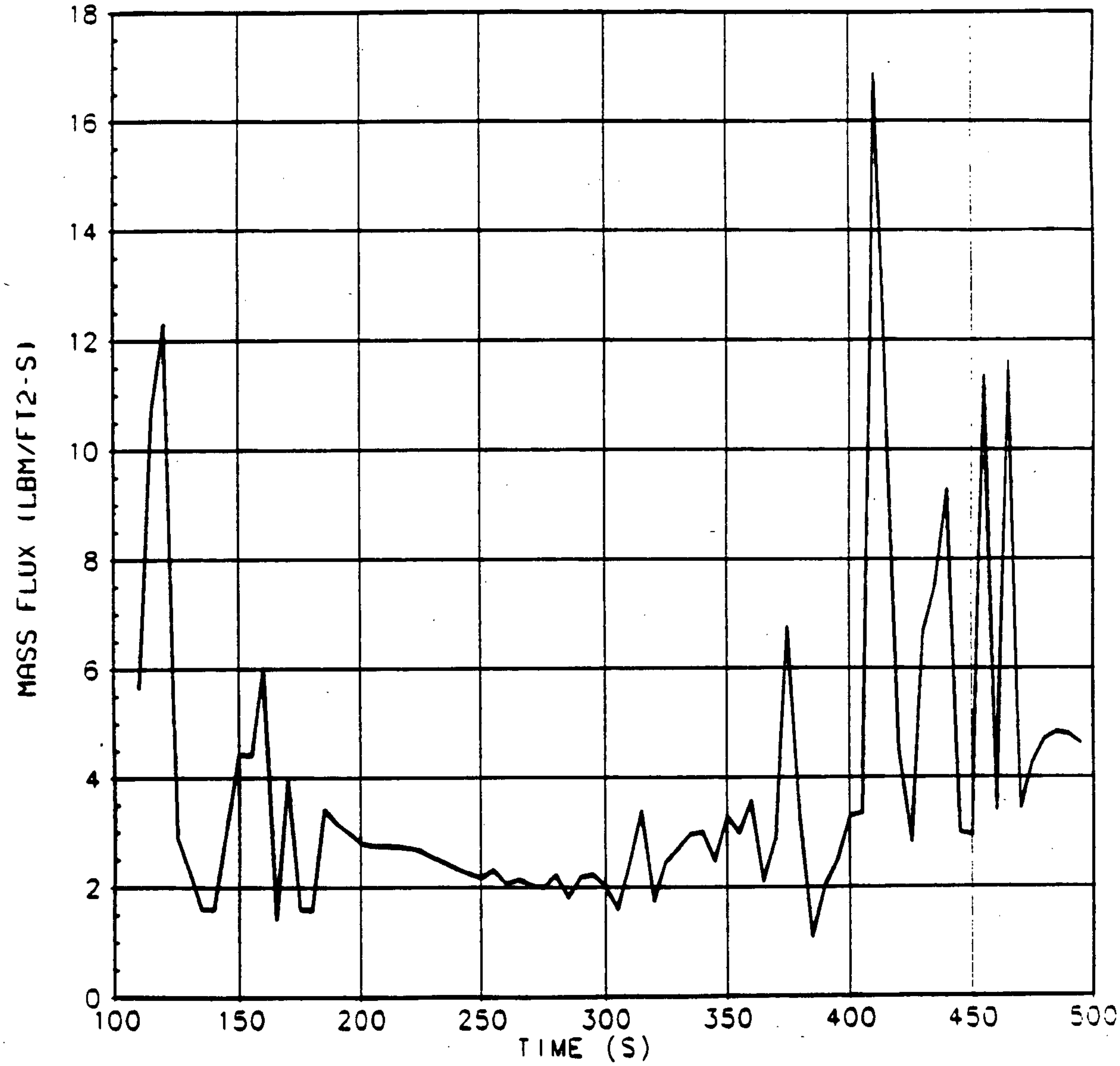

Figure 8

Hot Assembly Rod Surface Heat Transfer Coefficient

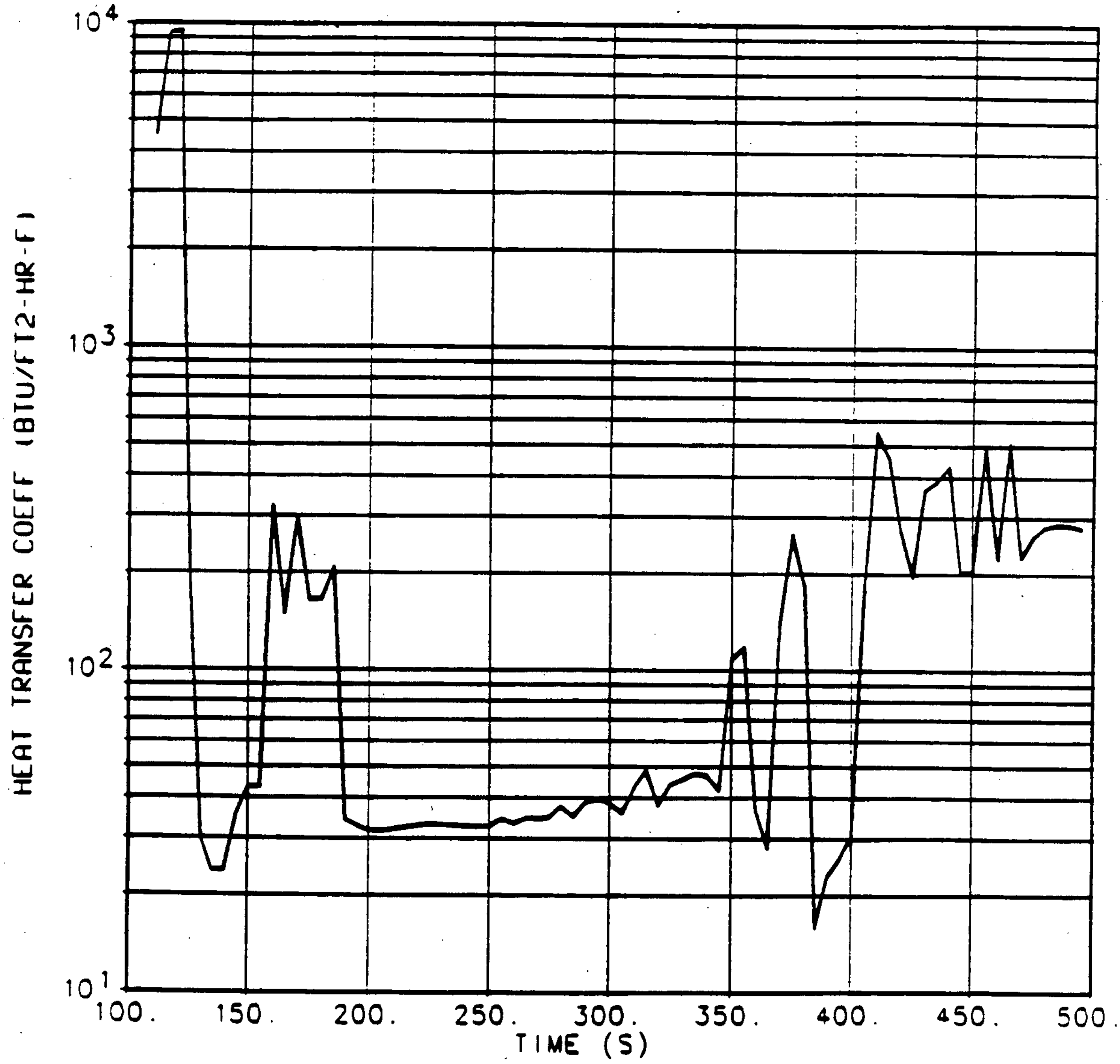

Figure 9

Hot Spot Fluid Temperature

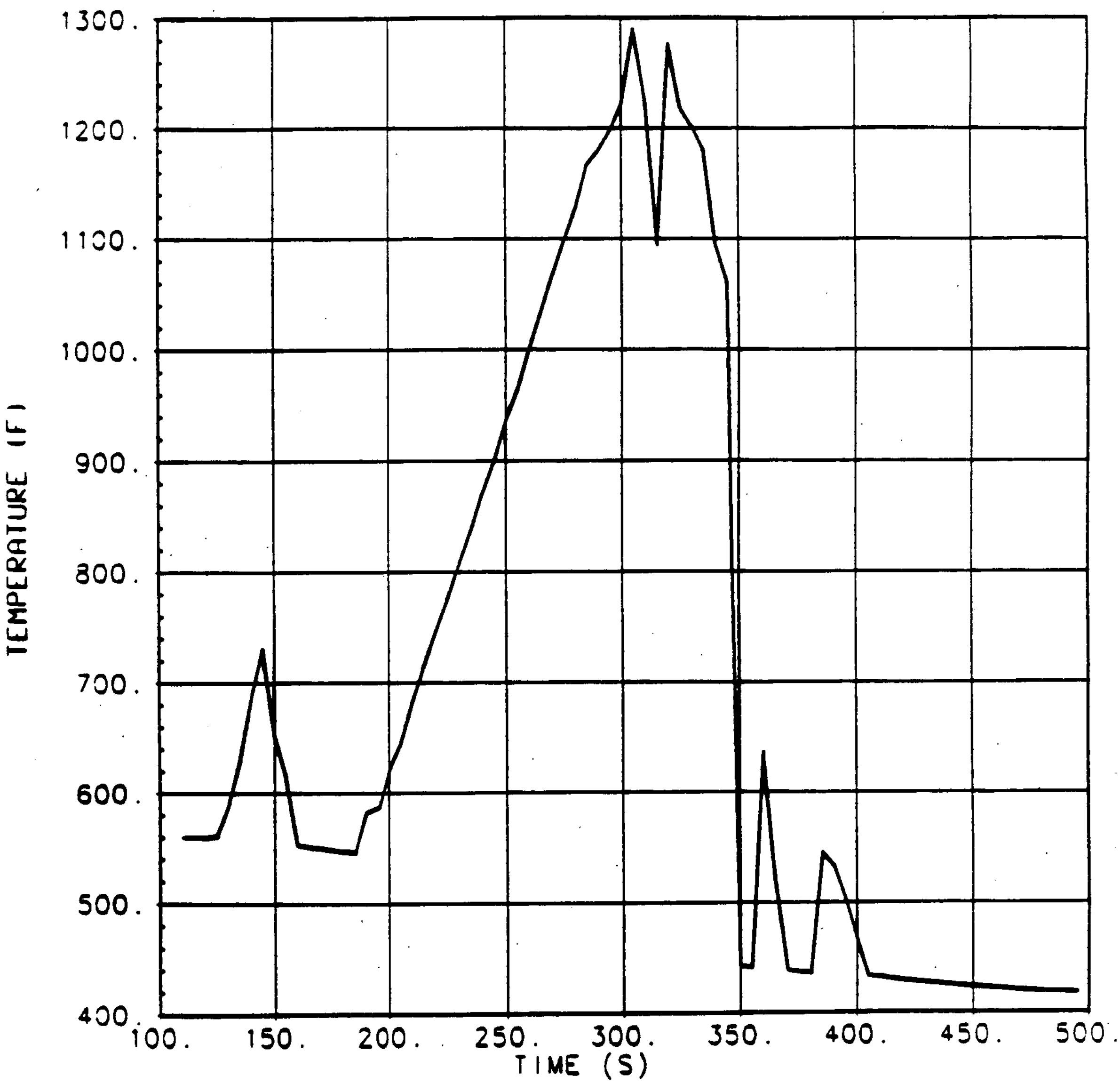

Figure 10

Cold Leg Break Mass Flow Rate

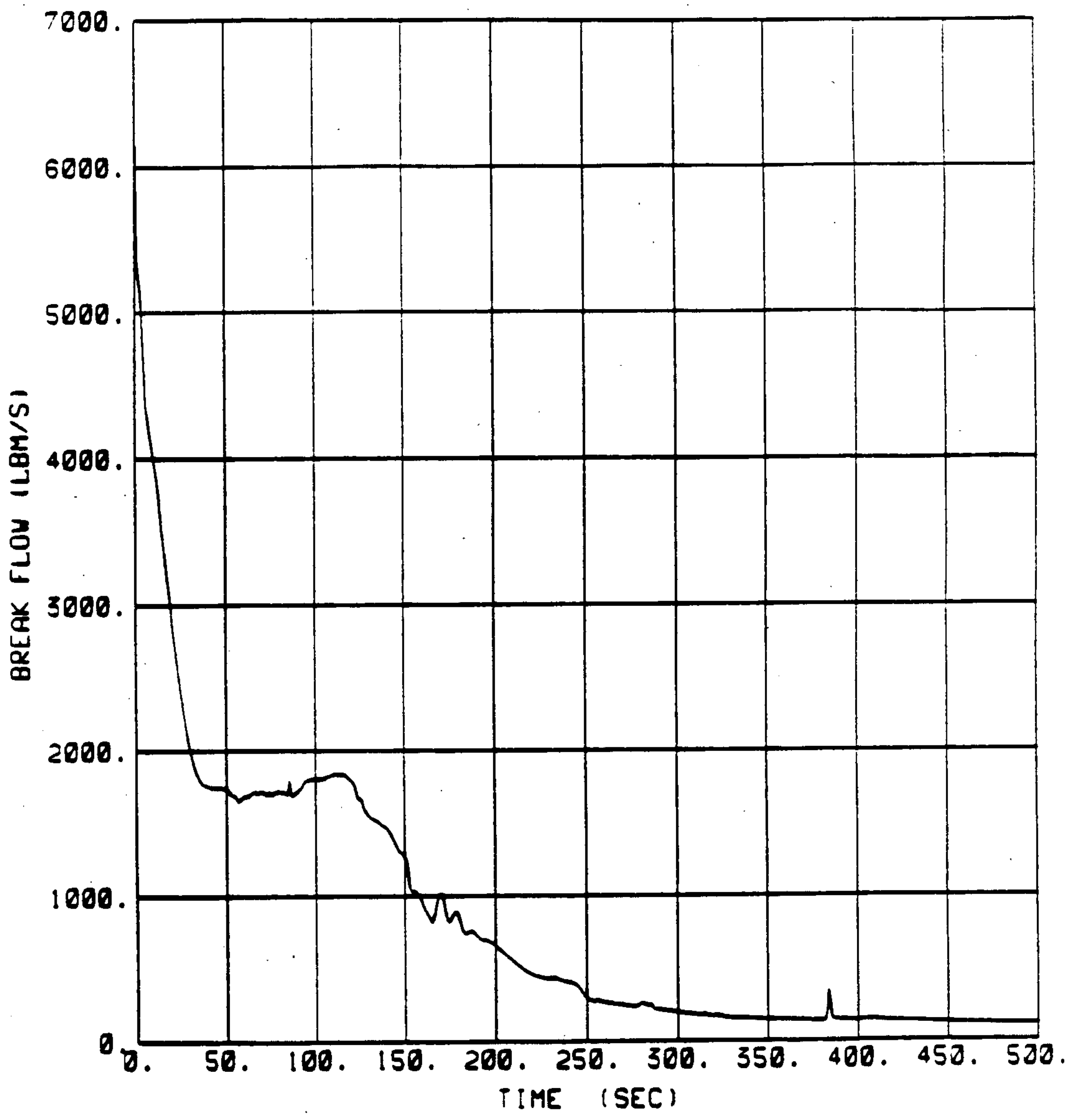

Figure 11

Safety Injection Mass Flow Rate

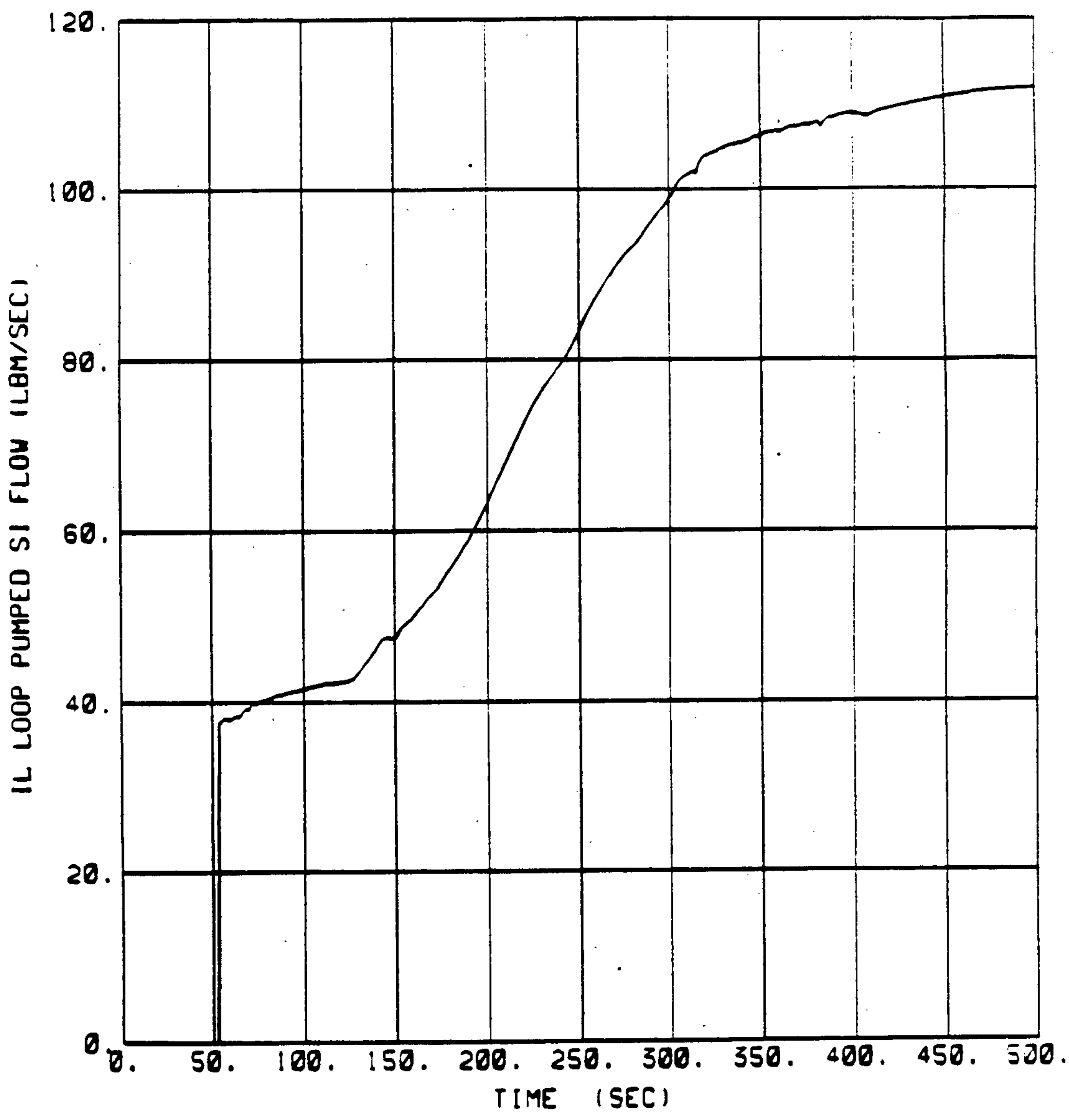

Figure 12

RCS Pressure

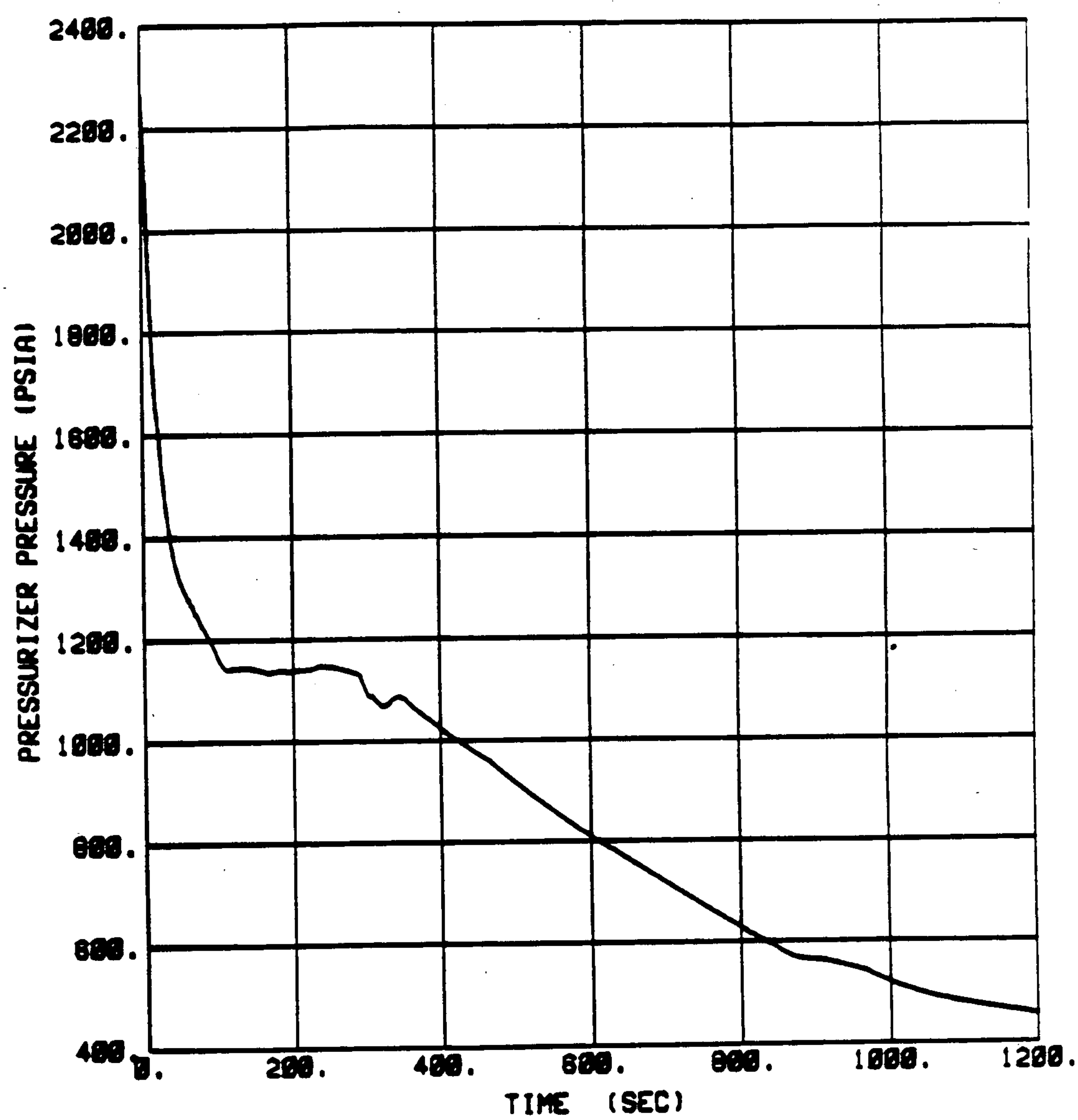

Figure 13

Core Mixture Level

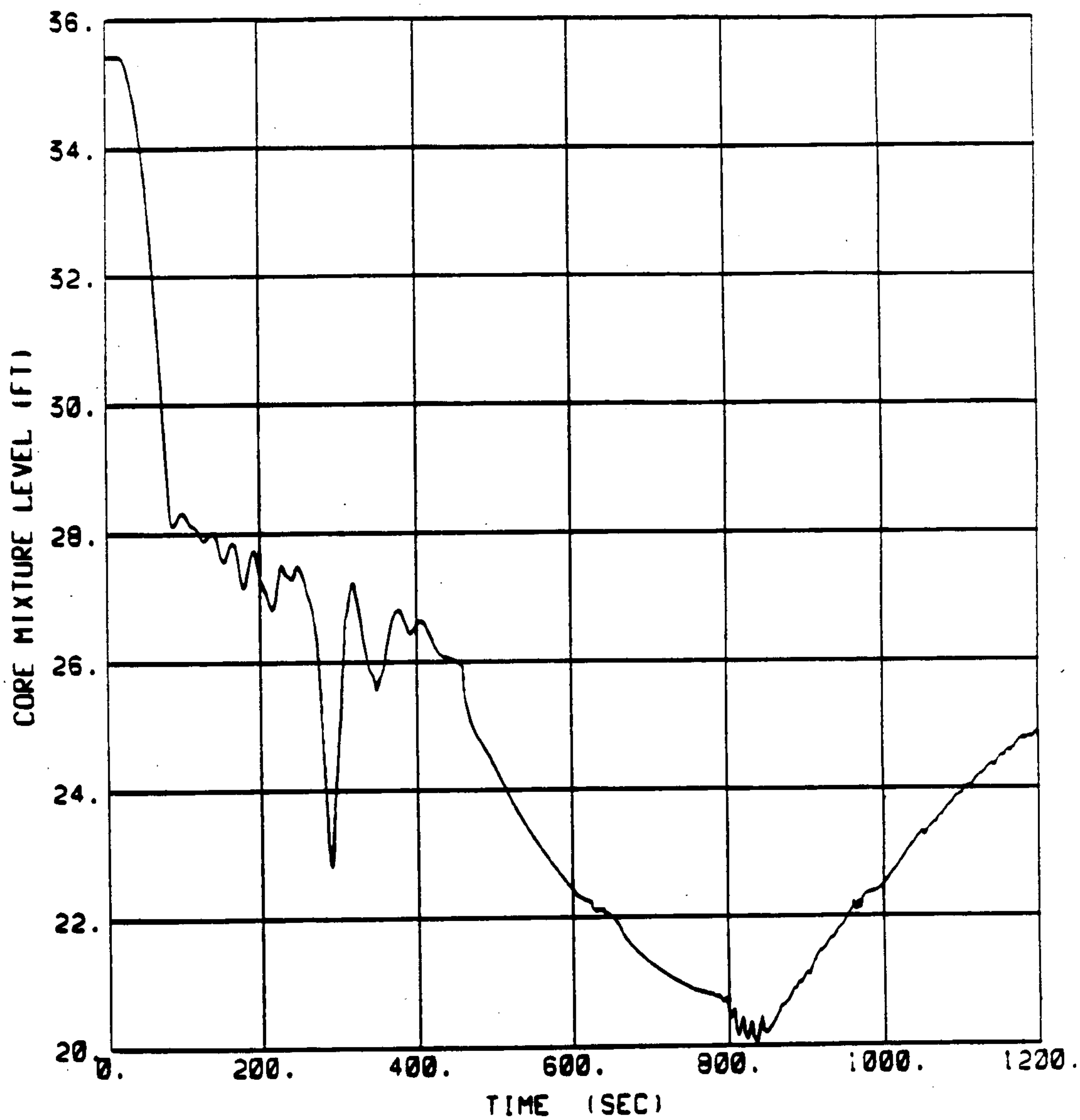

Figure 14

Hot Rod Clad Average Temperature

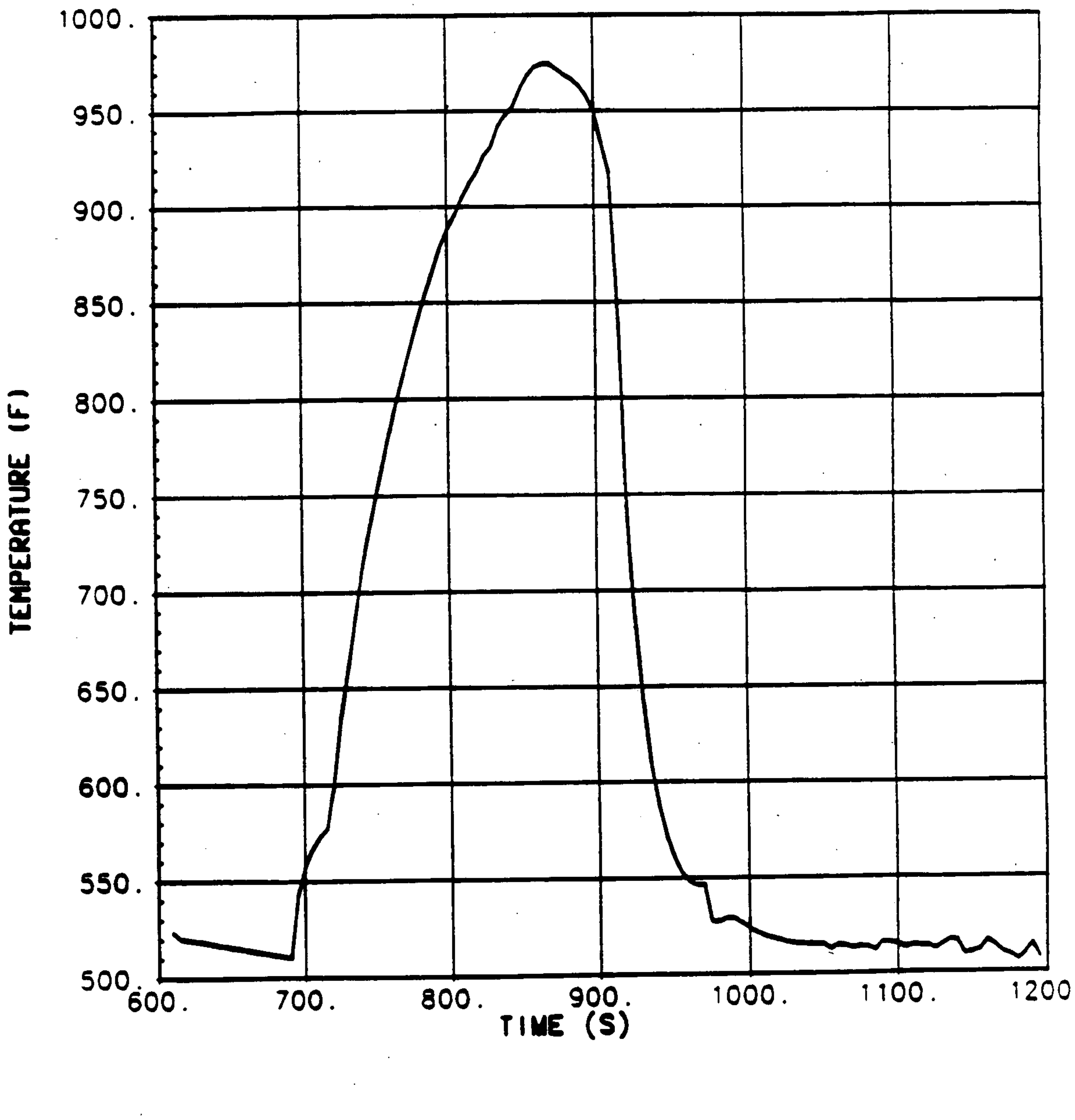

Figure 15

RCS Pressure

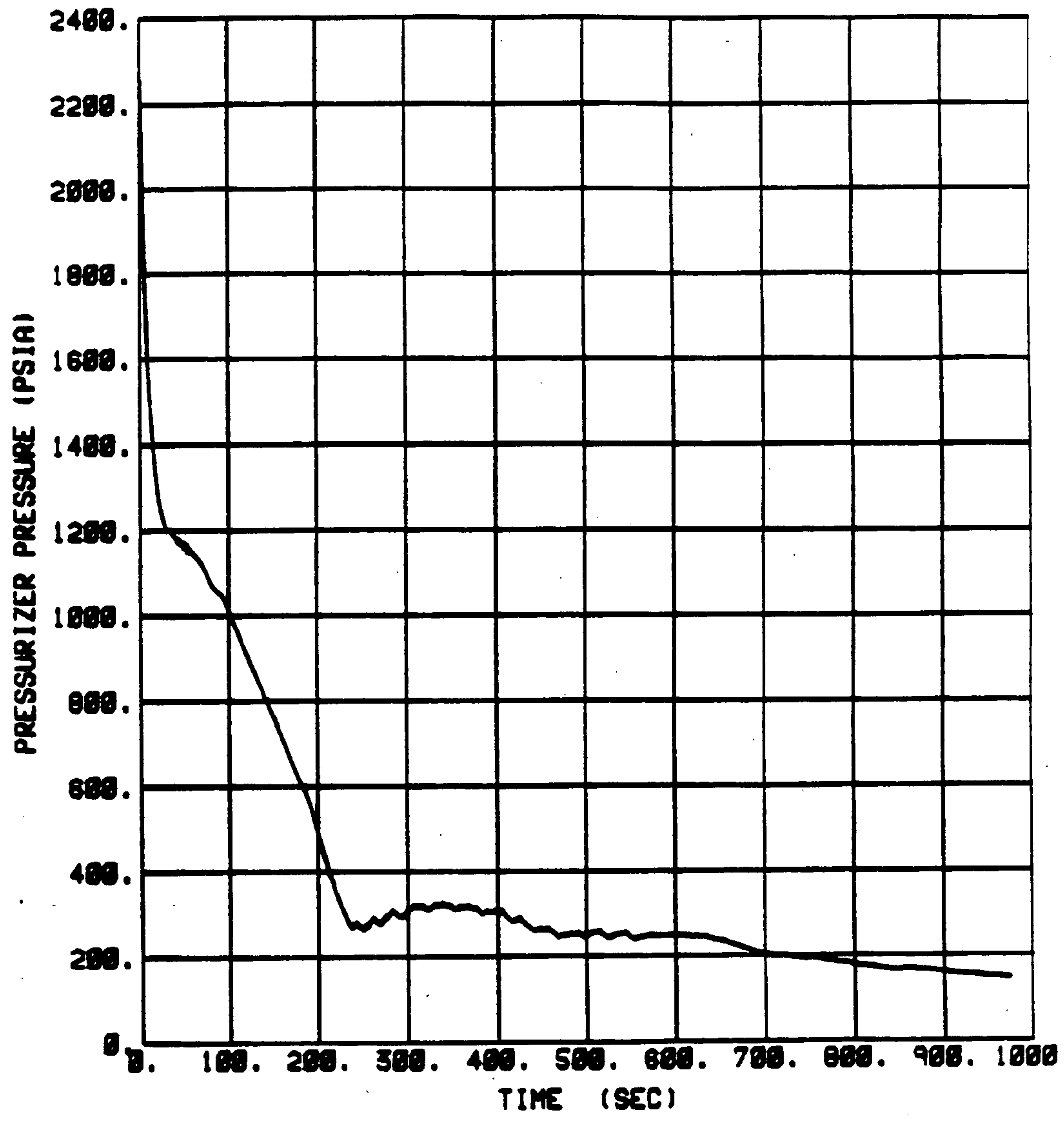

Figure 16

Core Mixture Level