Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057
Tel 504 739 6660
Fax 504 739 6678
kwalsh1@entergy.com

Kevin T. Walsh
Vice President, Operations
Waterford 3

W3F1-2007-0038

August 9, 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Emergency Core Cooling System Performance Analysis
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES: Entergy letter to the NRC "License Amendment Request NPF-38-271 to Support Next Generation Fuel" dated August 2, 2007 (W3F1-2007-0037)

Dear Sir or Madam:

Pursuant to 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems for light water nuclear power reactors*, and the draft Nuclear Regulatory Commission (NRC) Safety Evaluation (SE) for Westinghouse topical report (TR) WCAP-16500, *CE* [Combustion Engineering] *16 x 16 Next Generation Fuel Core Reference Report*, Entergy Operations, Inc. (Entergy) hereby requests an NRC review of the Waterford Steam Electric Station, Unit 3 (Waterford 3) revised Emergency Core Cooling System (ECCS) Performance Analysis that supports the implementation of CE 16x16 Next Generation Fuel (NGF) described in WCAP-16500. A license amendment request was submitted (Reference 1) to address the Waterford 3 Technical Specification changes for NGF.

Waterford 3 has committed by letter (Reference 1) to provide an addendum to the ECCS Performance analysis to address a limitation and condition in the final NRC SE for the Westinghouse topical report (TR) CENPD-132, Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood." The addendum will include the comparison graphical results needed to confirm the acceptability of the use of the optional steam cooling method described in the TR.

Entergy requests approval of the revised analysis by March 14, 2008 in order to support the spring 2008 refueling outage. Once approved and following startup from the spring 2008 refueling outage, the analysis shall become the analysis of record. Although this request is neither exigent nor emergency, your prompt review is requested.

A002

LIRR

This letter contains no commitments. If you have any questions or require additional information, please contact Ron Williams at 504-739-6255.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 9, 2007.

Sincerely,

KTW/DM

Attachment:
1. ECCS Performance Analysis

cc: Dr. Bruce S. Mallett
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Senior Resident Inspector
Waterford 3
P.O. Box 822
Killona, LA 70066-0751

U.S. Nuclear Regulatory Commission
Attn: Mr. Kaly Kalyanam
MS O-7E1
Washington, DC 20555-0001

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
P. O. Box 4312
Baton Rouge, LA 70821-4312

Attachment 1

W3F1-2007-0038

ECCS Performance Analysis

ECCS Performance Analysis

1.0 Introduction

This report summarizes the Emergency Core Cooling System (ECCS) performance analyses performed for the full core implementation of Combustion Engineering (CE) 16x16 Next Generation Fuel (NGF) assemblies into Waterford Steam Electric Station, Unit 3 (Waterford 3). CE 16x16 NGF as defined in WCAP-16500-P (Reference 1-15) will be implemented at Waterford 3 beginning in Cycle 16 commencing after the spring 2008 refueling outage.

Limitations and Conditions number 7 of the Safety Evaluation (SE) for WCAP-16500-P states: *"Implementation of CE 16 x 16 NGF assemblies necessitate re-analysis of the plant specific LOCA [Loss of Coolant Accident] analyses. Licensees are required to submit a license amendment containing the revised LOCA analyses for NRC review. Upon approval, the revised LOCA analyses constitute the analysis-of-record and baseline for which future changes will be measured against in accordance with 10 CFR 50.46 (a)(3)."* Entergy committed to provide the results of these re-analyses as part of the Waterford 3 license amendment request NPF-38-271 submitted on August 2, 2007.

These ECCS performance analyses were performed to demonstrate conformance to the acceptance criteria for ECCS for light water nuclear power reactors, 10 CFR 50.46 (Reference 1-1). Analyses were performed for a spectrum of Large Break (LB) and Small Break (SB) Loss-of-Coolant Accidents (LOCAs).

The fuel design changes for NGF which are important for ECCS performance analyses are compared to standard fuel assembly characteristics as follows:

- The NGF design contains Optimized ZIRLO™ clad fuel rods. In contrast, the standard fuel assemblies are comprised of ZIRLO™ clad fuel rods.

- The NGF rod cladding and UO_2 fuel pellet radial dimensions are reduced compared to the standard fuel rod design. This produces an increase in the fuel rod pitch-to-diameter ratio compared to the standard 16x16 fuel assembly design and an increase in the core cross-sectional area for coolant flow. Also, the NGF rod cladding diameter-to-thickness ratio is increased relative to the standard 16x16 fuel rod design. This ratio is used in calculating the engineering hoop stress across the fuel rod cladding for analyzing any mechanical deformation of the cladding.

- The NGF assembly hydraulic resistance is increased relative to the standard fuel assembly due to the addition of mixing grids. As a result, a transition mixed core assessment for NGF was performed in order to address the impact of co-resident hydraulically dissimilar fuel assemblies (i.e., NGF and standard fuel assemblies) on ECCS performance.

2.0 Objective

The objective of the ECCS performance analysis is to demonstrate conformance to the ECCS acceptance criteria of 10 CFR 50.46(b):

Criterion 1: Peak Cladding Temperature: The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

Criterion 2: Maximum Cladding Oxidation: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

Criterion 3: Maximum Hydrogen Generation: 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 cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Criterion 4: Coolable Geometry: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Criterion 5: Long-Term Cooling: After any calculated successful initial 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.

3.0 Regulatory Basis

As required by 10 CFR 50.46(a)(1)(i), the ECCS performance analysis must conform to the ECCS acceptance criteria identified in Section 2.0. Additionally, the ECCS performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. The evaluation model may either be a realistic evaluation model as described in 10 CFR 50.46(a)(1)(i) or must conform to the required and acceptable features of Appendix K ECCS Evaluation Models (Reference 1-2). The evaluation models used to perform the ECCS performance analyses documented herein are Appendix K evaluation models.

As previously stated Optimized ZIRLO™ fuel rod cladding material will be used in the design of NGF assemblies. The acceptance criteria and requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K currently are limited in applicability to the use of fuel rods clad with Zircaloy or ZIRLO™. 10 CFR 50.46 and 10 CFR Part 50, Appendix K cannot apply to the proposed use of NGF assemblies since Optimized ZIRLO™ has a slightly different composition than Zircaloy or ZIRLO™. Therefore an exemption request has been submitted (Reference 1-20) to apply these regulations to Optimized ZIRLO™.

4.0 Method(s) of Analysis

WCAP-16500 (Reference 1-15) is the Core Reference Report for CE 16x16 Next Generation Fuel, pending NRC approval. Section 5.2 of Reference 1-15 documents the ECCS performance methods suitable for use to analyze the implementation of NGF. The methods used for the ECCS performance analyses of Waterford 3 are summarized in the following sections.

The CE 16x16 NGF design utilizes Optimized ZIRLO™, an advanced cladding alloy. The implementation of Optimized ZIRLO™ in CE plants is documented in Reference 1-16 and approved by the NRC in Reference 1-17. As required by the SER limitations in Reference 1-17, the ECCS performance analysis computer codes have been updated to include the Optimized ZIRLO™ cladding property changes detailed in the topical report.

4.1 Large Break LOCA (LBLOCA)

The Westinghouse ECCS Performance Appendix K Evaluation Model for CE plants is the 1999 Evaluation Model (1999 EM) for LBLOCA (Reference 1-3). The 1999 EM for LBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding (Reference 1-18), and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding (Reference 1-16). Also, the 1999 EM is supplemented by WCAP-16072-P-A (Reference 1-19) for implementation of ZrB_2 IFBA fuel assembly designs.

The 1999 EM for LBLOCA includes the following computer codes: The CEFLASH-4A computer code (Reference 1-5) is used to perform the blowdown hydraulic analysis of the reactor coolant system (RCS) and the COMPERC-II computer code (Reference 1-6) is used to perform the RCS refill/reflood hydraulic analysis and to calculate the containment minimum pressure. It is also used in conjunction with the methodology described in Reference 1-7 to calculate the FLECHT-based reflood heat transfer coefficients used in the hot rod heatup analysis. The HCROSS (Reference 1-8) and PARCH (Reference 1-9) computer codes are used to calculate steam cooling heat transfer coefficients. The hot rod heatup analysis, which calculates the peak cladding temperature and maximum cladding oxidation, is performed with the STRIKIN-II computer code (Reference 1-10). Core-wide cladding oxidation is calculated using the COMZIRC computer code (Appendix C of Supplement 1 of Reference 1-6). The initial steady state fuel rod conditions used in the analysis are determined using the FATES3B computer code (Reference 1-11). Computer code process improvements have been made to facilitate the implementation of NGF assemblies in the LBLOCA analysis. These improvements will be reported to the NRC in the Westinghouse generic yearly letter of 2007 in compliance with 10 CFR 50.46(a)(3)(ii) (Reference 1-1).

The Appendix K steam cooling heat transfer component model for less than 1 in/sec core reflood in the 1999 EM has been modified to include spacer grid heat transfer effects. The details of this improvement to the 1999 EM are documented in Reference 1-4. For Waterford 3, the LBLOCA analysis credits the use of the modified model including spacer grid heat transfer effects.

In performing the LBLOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. It is assumed that offsite power is lost and all pumps must await diesel startup before they can begin to deliver flow. (It is assumed, however, that offsite power is available for the Containment Spray System and containment fan coolers). Also, it is assumed that all safety injection flow delivered to the broken cold leg is lost directly to the containment.

The limiting initial fuel rod conditions used in the LBLOCA analysis (i.e., the conditions that result in the highest calculated peak cladding temperature) were determined by performing burnup dependent calculations with the 1999 EM using initial fuel rod conditions calculated by FATES3B. The LBLOCA analysis included both UO_2 and ZrB_2 burnable absorber fuel rods in both the NGF and standard fuel rod designs.

A study was performed to determine the most limiting single failure of ECCS equipment. The study analyzed no failure, failure of an emergency diesel generator, failure of a high pressure safety injection (HPSI) pump, and a failure of a low pressure safety injection (LPSI) pump consistent with approved topical reports. Maximum safety injection pump flow rates were used in the no failure case; minimum safety injection pump flow rates were used in the emergency diesel generator, HPSI or LPSI pump failure cases. The pumps were actuated on a safety injection actuation signal (SIAS) generated by low pressurizer pressure with appropriate startup delay. Minimum refueling water storage pool temperature was used in all four cases as a result of a sensitivity study of the refueling water storage pool water temperature. The study also investigated the impact of variation in safety injection tank (SIT) pressure, water temperature and water volume on peak cladding temperature and peak local cladding oxidation.

A spectrum of guillotine breaks in the reactor coolant pump discharge leg was analyzed. As described in Section 3.4 of Reference 1-3 Supplement 4-P-A, the discharge leg is the most limiting break location and a guillotine break is more limiting than a slot break. In particular, the 0.4, 0.6, 0.8, and 1.0 Double-Ended Guillotine breaks in the reactor coolant Pump Discharge leg (DEG/PD) were analyzed for Waterford 3.

Since the CE 16x16 NGF assembly has a higher pressure drop, a transition mixed core assessment was performed to address the effect of flow redistribution on the CE 16x16 NGF assemblies during the transition cycles consisting of co-resident hydraulically dissimilar fuel assemblies.

4.2 Small Break LOCA (SBLOCA)

The small break LOCA ECCS performance analysis used the Supplement 2 version (referred to as the S2M or Supplement 2 Model) of the Westinghouse small break LOCA evaluation model for Combustion Engineering PWRs (Reference 1-12). The S2M for SBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding (Reference 1-18), and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™

cladding (Reference 1-16). Also, the S2M is supplemented by WCAP-16072-P-A for implementation of ZrB_2 IFBA fuel assembly designs (Reference 1-19).

The S2M for SBLOCA uses the following computer codes: The CEFLASH-4AS computer program (Reference 1-13) is used to perform the hydraulic analysis of the RCS until the time the safety injection tanks (SITs) begin to inject. After injection from the SITs begins, the COMPERC-II computer program (Reference 1-6) is used to perform the hydraulic analysis. COMPERC-II is only used in the SBLOCA evaluation model for larger break sizes that exhibit prolonged periods of SIT flow and significant core voiding. The hot rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer program (Reference 1-10) during the initial period of forced convection heat transfer and by the PARCH computer program (Reference 1-9) during the subsequent period of pool boiling heat transfer. Core-wide cladding oxidation is conservatively represented as the rod-average cladding oxidation of the hot rod. The initial steady state fuel rod conditions used in the analysis are determined using the FATES3B computer program (Reference 1-11).

The small break LOCA analysis was performed for the fuel rod conditions that result in the maximum initial stored energy in the fuel. The calculations included the analysis of both UO_2 and ZrB_2 burnable absorber fuel rods in both the NGF and standard fuel rod designs.

For Waterford 3, the analysis was performed using the failure of a direct current (DC) bus as the most limiting single failure of the ECCS. A DC bus failure would prevent startup of an emergency diesel generator that would cause the loss of a high pressure safety injection (HPSI) pump and a low pressure safety injection LPSI pump, and results in a minimum of safety injection water being available to cool the core. The LPSI pumps are not explicitly credited in the small break LOCA analysis since the RCS pressure never decreases below the LPSI pump shutoff head during the portion of the transient that is analyzed.

For Waterford 3, the analysis credits operation of the steam generator atmospheric dump valves (ADVs). The ADVs are safety grade equipment. They are modeled in automatic mode with an opening pressure of 1040 psia. The most limiting single failure of a DC bus, which prevents start up of a diesel generator, results in loss of DC power to an ADV controller. Thus, only one of the two ADVs (one ADV per SG) is available for control of secondary side pressure.

A spectrum of three break sizes in the reactor coolant pump discharge (PD) leg was analyzed to bracket the limiting break size, which for Waterford 3 was the 0.055 ft^2/PD break. The reactor coolant pump discharge leg is the limiting break location because it maximizes the amount of spillage from the ECCS. The limiting small break LOCA is the largest small break for which the hot rod cladding heatup transient is terminated solely by injection from a HPSI pump.

No SBLOCA mixed-core analysis is necessary during transition core cycles due to the negligible effect of variations in core hydraulic losses on SBLOCA analysis results.

4.3 Post-LOCA Long Term Cooling

As documented in Reference 1-15, the analyses performed with the Westinghouse post-LOCA long-term cooling evaluation model for CE plants (CENPD-254-P-A, Reference 1-14) are not sensitive to the fuel assembly changes being introduced for the CE 16x16 NGF design. As a result, no plant-specific post-LOCA long-term cooling analyses were required to support the introduction of the CE 16x16 NGF assembly.

5.0 Results for Waterford 3

5.1 Plant Design Data

Important core, RCS, ECCS, and containment design data used in the LBLOCA analysis are listed in Tables 5-1 and 5-2. The listed fuel rod conditions are for rod average burnup of the hot rod that produced the highest calculated peak cladding temperature. In particular, the results of this ECCS Performance analysis support a peak linear heat generation rate of 12.9 kW/ft. Plant design data for the containment (e.g., data for the containment initial conditions, containment volume, containment heat removal systems, and containment passive heat sinks) were selected to minimize the transient containment pressure. The core inlet temperature was the minimum RCS cold leg temperature at the full power including uncertainty.

For Waterford 3, the assumed minimum containment temperature is 95°F, which is a 5°F increase from the current Technical Specification. A license amendment request has been submitted to change the containment minimum temperature (Reference 1-21). The containment temperature change will be applicable above 70% of the rated core power and if temperature falls below the minimum Technical Specification limit and remains above 90°F, then, as demonstrated by an ECCS Performance analysis, a peak linear heat generation rate reduction to 12.7 kW/ft will be required.

For Waterford 3, the assumed maximum SIT water volume is 1586 ft^3 which is a 100 ft^3 reduction from the current Technical Specification. A license amendment request has been submitted to address maximum SIT water volume (Reference 1-21).

Important core, RCS, and ECCS design data used in the SBLOCA analysis are listed in Tables 5-7 and 5-8. The listed fuel rod conditions are for the hot rod burnup that produces the maximum initial stored energy.

5.2 Large Break LOCA

Table 5-3 lists the peak cladding temperature and oxidation percentages for the spectrum of large break LOCAs. Times of interest are listed in Table 5-4. The break spectrum results for peak cladding temperature of the hot rod were most limiting for the

UO_2 fuel type at a burnup of 32 GWD/MTU. The most limiting case for maximum local cladding oxidation of the hot rod was for the UO_2 fuel type at a burnup of 0.5 GWD/MTU. The variables listed in Tables 5-5 and 5-6 are plotted as functions of time in Figures 5-1 through 5-22 for the 1.0 DEG/PD break, the limiting large break LOCA. The variables listed in Table 5-5 are plotted as functions of time for the 0.8 DEG/PD break in Figures 5-23 through 5-30. The variables listed in Tables 5-5 are plotted for the 0.6 DEG/PD in Figures 5-31 through 5-38. The variables listed in Tables 5-5 are plotted for the 0.4 DEG/PD in Figures 5-39 through 5-46. The results for the full core implementation of NGF demonstrate conformance to the ECCS acceptance criteria as summarized below. The results for the current AOR with 20% SGTP are provided for comparison.

Parameter	Criterion	NGF Results	Current AOR Results
Peak Cladding Temperature	≤2200°F	2166°F	2132°F
Maximum Cladding Oxidation	≤17%	16.9%	15.32%
Maximum Core-Wide Oxidation	≤1%	<1%	<0.99%
Coolable Geometry	Yes	Yes	Yes

The results are applicable to Waterford 3 for a rated core power of 3716 MWt (analyses are performed at 3735 MWt to account for a 0.5% power measurement uncertainty) for the implementation of CE 16x16 NGF. These results support a peak linear heat generation rate (PLHGR) of 12.9 kW/ft.

5.3 Small Break LOCA

Table 5-9 lists the peak cladding temperature and oxidation percentages for the spectrum of small break LOCAs. Times of interest are listed in Table 5-10. The variables listed in Table 5-11 are plotted as a function of time for each break in Figures 5-47 through 5-70. The results for the 0.055 ft^2/PD break, the limiting small break LOCA, demonstrate conformance to the ECCS acceptance criteria as summarized below.

Parameter	Criterion	NGF Results	Current AOR Results
Peak Cladding Temperature	≤2200°F	1973°F	1972°F
Maximum Cladding Oxidation	≤17%	14.3%	12.8%
Maximum Core-Wide Oxidation	≤1%	<0.80%	<0.71%
Coolable Geometry	Yes	Yes	Yes

The results are applicable to Waterford 3 for a PLHGR of 13.2 kW/ft and a rated core power of 3716 MWt (analyses are performed at 3735 MWt to account for a 0.5% power measurement uncertainty) for the implementation of CE 16x16 NGF.

5.4 Post-LOCA Long Term Cooling

There is no significant impact of NGF implementation on the post-LOCA LTC analysis results. The results of the AOR for post-LOCA LTC continue to apply.

5.5 Inadvertent Opening of a Pressurizer Safety Valve

There is no significant impact of NGF implementation on the inadvertent opening of a pressurizer safety valve (IOPSV) analysis results. The results of the AOR for IOPSV continue to apply.

5.6 Transition Mixed Core

A transition mixed core assessment was performed for NGF in order to address the impact of co-resident hydraulically dissimilar fuel assemblies (i.e., NGF and standard fuel assemblies) on ECCS performance. The NGF core hydraulic resistance is greater than the standard fuel assembly due to the addition of mixing grids. Therefore, adjacent NGF and standard assemblies will experience a net redistribution of flow from the higher resistant NGF assembly to the lower resistant standard assembly.

This flow redistribution in the NGF mixed transition cores produces a slight penalty on the NGF assembly ECCS performance during the LBLOCA. However, a smaller cross-sectional core area for coolant flow (relative to a full core of NGF assemblies) is credited in the transition core assessment to improve the core hydraulics behavior during the blowdown period. Also, the smaller cross-sectional core area increases the core reflooding rates during the reflood period relative to the bounding full core NGF analysis. The net impact on ECCS performance is a slight reduction in the peak cladding temperature, peak cladding oxidation, and core-wide cladding oxidation percentages.

For Waterford 3, two mixed core configurations were examined to address core loading differences that are expected in the coming cycles of operation. The transition mixed core ECCS performance assessment determined that the results were bounded by the results of the full core NGF implementation analysis.

6.0 Conclusions

An ECCS performance analysis was completed for Waterford 3 at the power uprate rated core power of 3716 MWt (analyses performed at 3735 MWt to account for a 0.5% power measurement uncertainty) for the implementation of CE 16x16 NGF. The calculations included the analysis of both UO_2 and ZrB_2 IFBA rods in both the NGF and standard fuel rod designs, including a mixed core assessment. The analysis included consideration of large break LOCA, small break LOCA, and post-LOCA long term cooling. The limiting break size, i.e., the break size that resulted in the highest peak cladding temperature, was determined to be the 1.0 DEG/PD break.

The results of the analysis demonstrate conformance to the ECCS acceptance criteria at a PLHGR of 12.9 kW/ft as follows:

Criterion 1: Peak Cladding Temperature: The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

Result: The ECCS performance analysis calculated a peak cladding temperature of 2166°F for the 1.0 DEG/PD break.

Criterion 2: Maximum Cladding Oxidation: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

Result: The ECCS performance analysis calculated a maximum cladding oxidation of 0.169 times the total cladding thickness before oxidation for the 1.0 DEG/PD break.

Criterion 3: Maximum Hydrogen Generation: 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 cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Result: The ECCS performance analysis calculated a maximum hydrogen generation of less than 0.01 times the hypothetical amount for the 1.0 DEG/PD break.

Criterion 4: Coolable Geometry: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Result: The cladding swelling and rupture models used in the ECCS performance analysis account for the effects of changes in core geometry that would occur if cladding rupture is calculated to occur. Adequate core cooling was demonstrated for the changes in core geometry that were calculated to occur as a result of cladding rupture. In addition, the transient analysis was performed to a time when cladding temperatures were decreasing and the RCS was depressurized, thereby precluding any further cladding deformation. Therefore, a coolable geometry was demonstrated.

Criterion 5: Long-Term Cooling: After any calculated successful initial 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.

Result: The large break and small break LOCA ECCS performance analyses demonstrated that the Waterford 3 ECCS successfully maintains the fuel cladding temperature at an acceptably low value in the short term. Subsequently, for the extended period of time required by the long-lived radioactivity remaining in the core, the ECCS continues to supply sufficient cooling water from the refueling water tank and then from the sump to remove decay heat and maintain the core temperature at an acceptably low value. In addition, at the appropriate time, the operator realigns a HPSI pump for simultaneous hot and cold leg injection in order to maintain the core boric acid concentration below the solubility limit.

Table 5-1
Large Break LOCA ECCS Performance Analysis
Core and Plant Design Data

Quantity	Value	Units
Reactor power level (100.5% of rated power)	3735	MWt
Peak linear heat generation rate (PLHGR) of the hot rod	12.9	kW/ft
Average linear heat generation rate (100.5% of rated)	5.846	kW/ft
Gap conductance at the PLHGR**	2275	BTU/hr-ft^2-°F
Fuel centerline temperature at the PLHGR**	3016	°F
Fuel average temperature at the PLHGR**	1888	°F
Hot rod gas pressure**	1467	psia
Moderator temperature coefficient at 583°F, 2250 psia	$+0.0 \times 10^{-4}$	$\Delta\rho$/°F
RCS flow rate	148.0×10^6	lbm/hr
Core flow rate	144.15×10^6	lbm/hr
RCS pressure	2250	psia
Cold leg temperature	533.0	°F
Hot leg temperature	598.7	°F
Plugged tubes per steam generator	1870	---
Low pressurizer pressure SIAS setpoint	1560	psia
Safety injection tank pressure (min/max)	584.7/714.7	psia
Safety injection tank water volume (min/max)	926/1586	ft^3
LPSI pump flow rate (min, 1 pump/max, 2 pump)	4084/11300	gpm
HPSI pump flow rate (min, 1 pump/max, 2 pump)	787/1970	gpm
Containment pressure	14.025	psia
Containment temperature	95	°F
Containment humidity	100	%
Containment net free volume	2.684×10^6	ft^3
Containment spray pump flow rate	2250	gpm/pump
Refueling water tank temperature (min/max)	50/100	°F
Containment passive heat sinks	Table 5-2	---

** These quantities correspond to the rod average burnup of the hot rod (32 GWD/MTU) that yields the highest peak cladding temperature.

Table 5-2
Large Break LOCA ECCS Performance Analysis
Containment Passive Heat Sink Data

Wall No.	Description	Material	Thickness (ft)	Surface Area (ft^2)
1	Containment Primary Cylinder and Dome	Carbon Steel	0.118879	92819.00
2	Concrete Underwater (one side faces ground)	Concrete Concrete Concrete	0.25 0.25 10.963	15427.75
3	Concrete Underwater (all remaining)	Concrete Concrete Concrete	0.25 0.25 1.549	8553.69
4	Concrete in Air – less than 6 feet thick(1)	Concrete Concrete Concrete	0.25 0.25 0.6025	47663.92
5	Concrete in Air – greater than or equal to 6 feet thick (1)	Concrete Concrete Concrete	0.25 0.25 2.865	9913.15
6	Stainless Steel (1)	Stainless Steel	0.003734	59114.40
7	Galvanized Steel (Zinc Coating on Carbon Steel) (1)	Zinc Carbon Steel	0.000122 0.005628	192827.75
8	Structural and Miscellaneous Exposed Steel – less than 0.2 inch thick (1)	Carbon Steel	0.008134	194549.18
9	Structural and Miscellaneous Exposed Steel – greater than or equal to 0.2 inch thick but less than 0.5 inch thick (1)	Carbon Steel	0.03154	215234.76
10	Structural and Miscellaneous Exposed Steel – greater than 0.5 inch thick (1)	Carbon Steel	0.065582	71308.76

(1) Thickness is effective thickness as a result of combining similar thickness walls.

Table 5-3
Large Break LOCA ECCS Performance Analysis Results

Break Size	Peak Cladding Temperature (°F)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Cladding Oxidation (%)
Spectrum Results for Peak Cladding Temperature**			
1.0 DEG/PD*	2166	16.8	<1
0.8 DEG/PD*	2159	16.5	<1
0.6 DEG/PD*	2101	14.4	<1
0.4 DEG/PD*	2015	11.6	<1
Case Results for Maximum Local Cladding Oxidation***			
1.0 DEG/PD*	2155	16.9	<1

* DEG/PD: Double Ended Guillotine Break at Pump Discharge Leg
** Results are for UO_2 fuel type at Burnup of 32 GWD/MTU
*** Results are for UO_2 fuel type at Burnup of 0.5 GWD/MTU

Table 5-4
Large Break LOCA ECCS Performance Analysis
Times of Interest (seconds after break)

Break Size	SITs On	End of Blowdown	Start of Reflood	SITs Empty	SI Pumps on	Hot Rod Rupture
Spectrum Results for Peak Cladding Temperature**						
1.0 DEG/PD*	9.8	24.0	41.0	99.9	34.1	39.0
0.8 DEG/PD*	11.1	25.5	42.5	101.3	34.2	40.8
0.6 DEG/PD*	13.1	27.7	44.6	103.6	34.3	45.7
0.4 DEG/PD*	16.9	32.1	48.7	108.0	34.6	51.5
Case Results for Maximum Local Cladding Oxidation***						
1.0 DEG/PD*	9.8	24.0	41.0	99.9	34.1	47.4

* DEG/PD: Double Ended Guillotine Break at Pump Discharge Leg
** Results are for UO_2 fuel type at Burnup of 32 GWD/MTU
*** Results are for UO_2 fuel type at Burnup of 0.5 GWD/MTU

Table 5-5
Large Break LOCA ECCS Performance Analysis
Each Break
Variables Plotted as a Function of Time

Variable
Core Power
Pressure in Center Hot Assembly Node
Leak Flow Rate
Hot Assembly Flow Rate (Below and Above Hot Spot)
Hot Assembly Quality
Containment Pressure
Mass Added to Core During Reflood
Peak Cladding Temperature

Table 5-6
Large Break LOCA ECCS Performance Analysis
Limiting Break
Variables Plotted as a Function of Time

Variable
Mid Annulus Flow Rate
Quality Above and Below the Core
Core Pressure Drop
Safety Injection Flow Rate into Intact Discharge Legs
Water Level in Downcomer During Reflood
Hot Spot Gap Conductance
Maximum Local Cladding Oxidation Percentage
Fuel Centerline, Fuel Average, Cladding, and Coolant Temperature at the Hot Spot
Hot Spot Heat Transfer Coefficient
Hot Pin Pressure
Core Bulk Channel Flow Rate
Effective Spray and Spillage to Containment
Containment (steam) Temperature
Containment (water) Temperature

Table 5-7
Small Break LOCA ECCS Performance Analysis
Core and Plant Design Data

Quantity	Value	Units
Reactor power level (including uncertainty)	3735	MWt
Peak linear heat generation rate (PLHGR)	13.2	kW/ft
Axial shape index	-0.25	—
Gap conductance at PLHGR(1)	1768	BTU/hr-ft^2-°F
Fuel centerline temperature at PLHGR(1)	3205	°F
Fuel average temperature at PLHGR(1)	2027	°F
Hot rod gas pressure(1)	705	psia
Moderator temperature coefficient at initial density	0.0×10^{-4}	$\Delta\rho$/°F
RCS flow rate	148.0×10^6	lbm/hr
Core flow rate	144.15×10^6	lbm/hr
RCS pressure	2250	psia
Cold leg temperature	552.0	°F
Hot leg temperature	615.5	°F
Plugged tubes per steam generator	1870	-----
MSSV first bank opening pressure	1117.2	psia
Low pressurizer pressure reactor trip setpoint	1560	psia
Low pressurizer pressure SIAS setpoint	1560	psia
HPSI Flow Rate	Table 5-8	gpm
Safety injection tank pressure	584.7	psia
Atmospheric Dump Valve Opening Pressure	1040	psia

Note:
(1) These quantities correspond to the rod average burnup of the hot rod (500 MWD/MTU) that yields the maximum initial stored energy.

Table 5-8
High Pressure Safety Injection Pump
Minimum Delivered Flow to RCS
(Assuming Failure of an Emergency Diesel Generator)

RCS Pressure, psig	Flow Rate, gpm
0.0	800.0
200	735.9
400	666.3
600	589.2
800	501.4
1000	396.0
1200	254.0
1300	143.3
1355.1	0.0

Notes:
1. The flow is split equally to each of the four discharge legs.
2. The flow to the broken discharge leg is spilled out the break.

Table 5-9
Small Break LOCA ECCS Performance Analysis Results

Break Size	Peak Cladding Temperature (°F)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Cladding Oxidation (%)
0.05 ft^2/PD	1933	12.4	<0.71
0.055 ft^2/PD	1973	14.3	<0.80
0.06 ft^2/PD	1969	6.4	<0.41

Table 5-10
Small Break LOCA ECCS Performance Analysis
Times of Interest

Break Size	HPSI Flow Delivered to RCS (seconds after break)	LPSI Flow Delivered to RCS (seconds after break)	SIT Flow Delivered to RCS (seconds after break)	Peak Cladding Temperature Occurs (seconds after break)
0.05 ft^2/PD	158	(a)	1904 (b)	1887
0.055 ft^2/PD	146	(a)	1656 (b)	1622
0.06 ft^2/PD	135	(a)	1462	1463

(a) Calculation completed before LPSI flow delivery to RCS begins.
(b) Injection from the SITs is not credited. This value is the time injection would have begun had it been credited.

Table 5-11
Small Break LOCA ECCS Performance Analysis
Variables Plotted as a Function of Time for Each Break

Variable
Core Power
Inner Vessel Pressure
Break Flow Rate
Inner Vessel Inlet Flow Rate
Inner Vessel Two-Phase Mixture Level
Heat Transfer Coefficient at Hot Spot
Coolant Temperature at Hot Spot
Cladding Temperature at Hot Spot

Figure 5-1
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Core Power

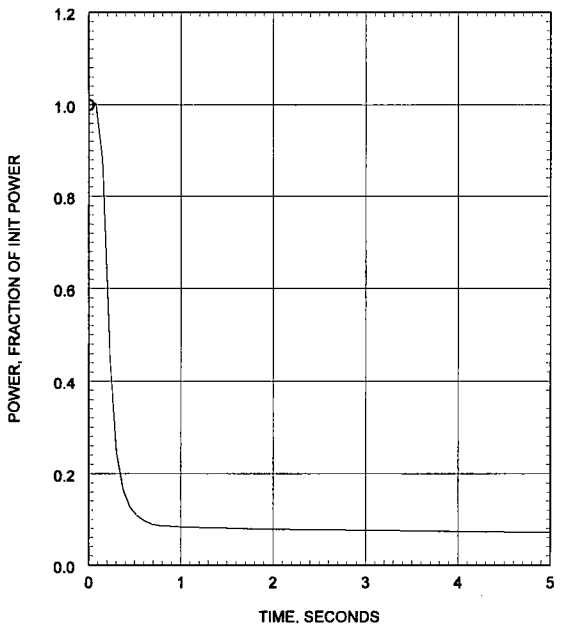


Figure 5-2
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Pressure in Center Hot Assembly Node

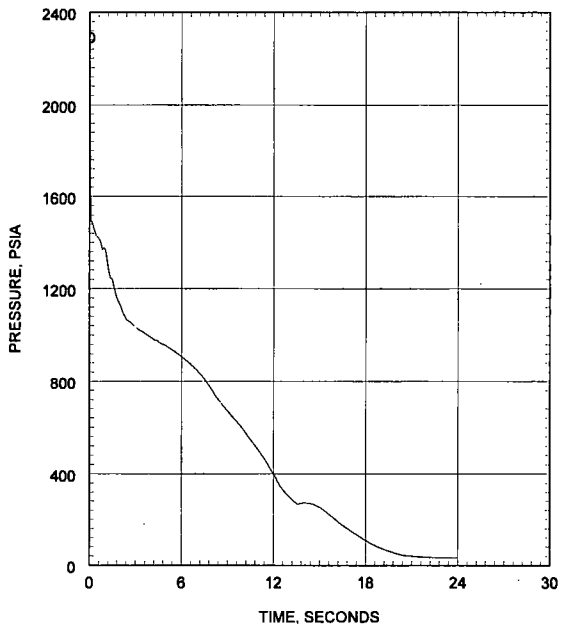


Figure 5-3
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Leak Flow Rate

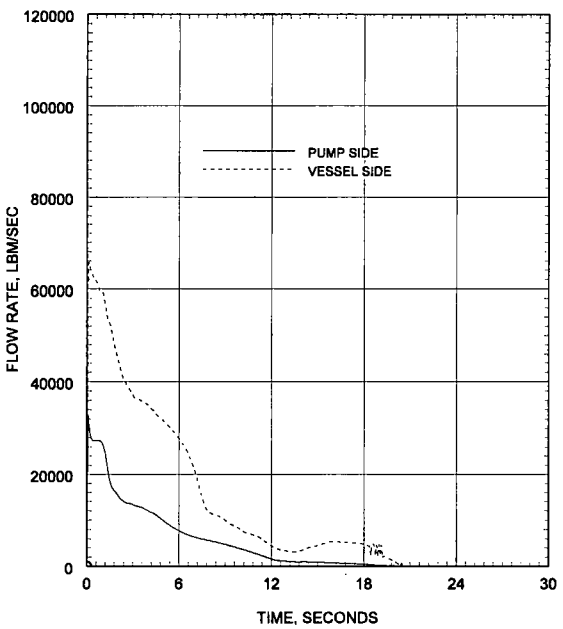


Figure 5-4
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Hot Assembly Flow Rate (Below and Above Hot Spot)

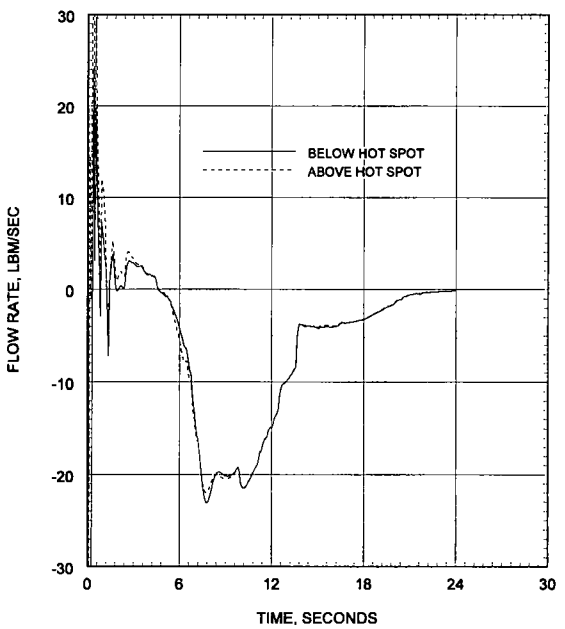


Figure 5-5
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Hot Assembly Quality

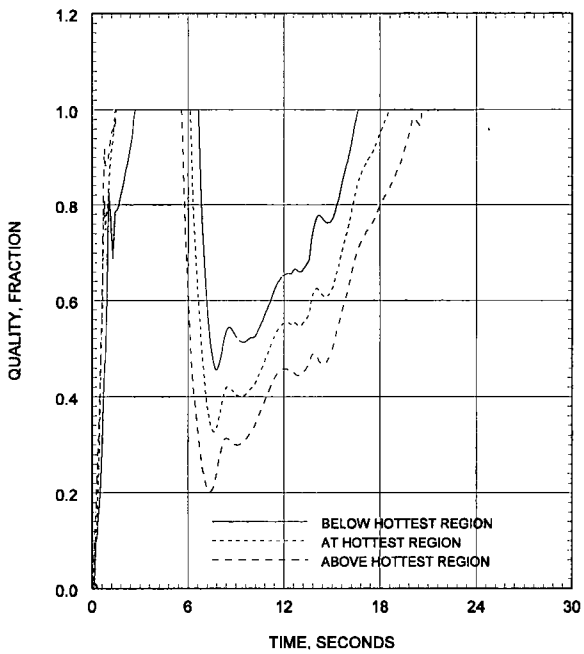


Figure 5-6
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Containment Pressure

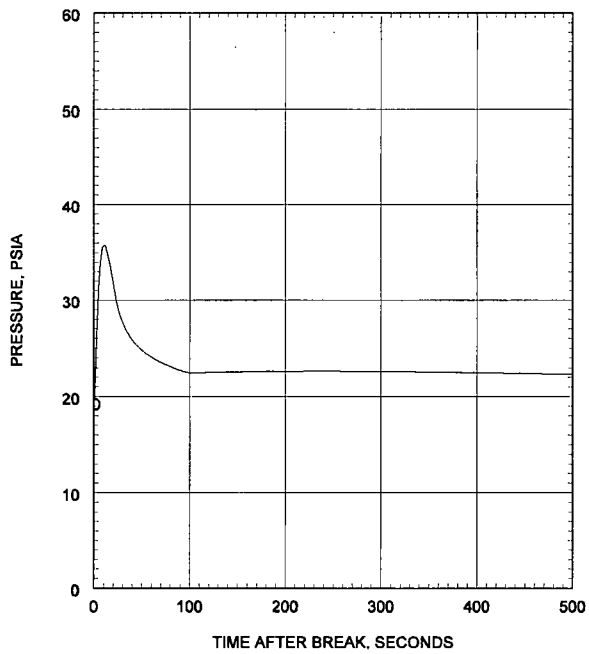


Figure 5-7
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Mass Added to Core During Reflood

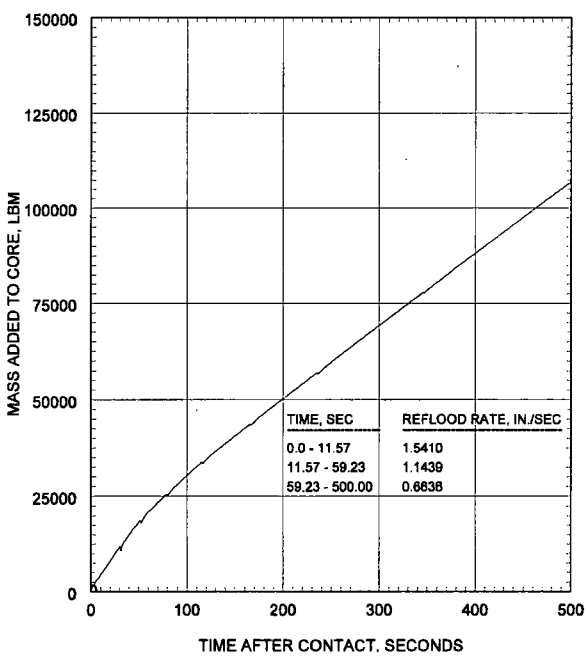


Figure 5-8
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Peak Cladding Temperature

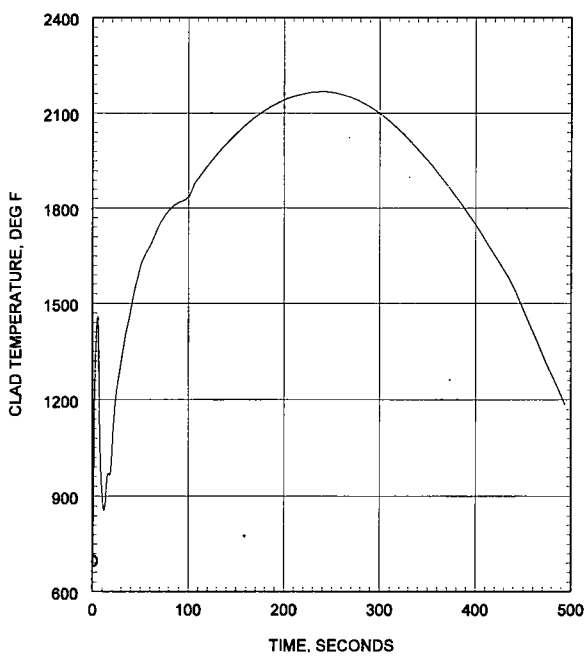


Figure 5-9
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Mid Annulus Flow Rate

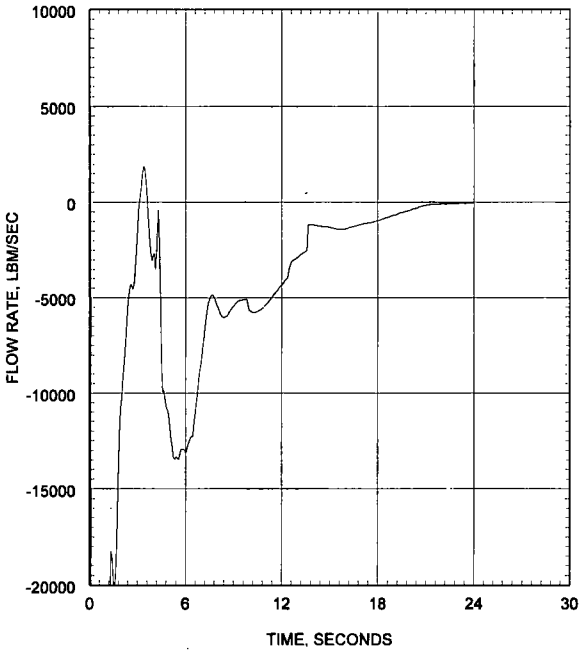


Figure 5-10
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Quality Above and Below the Core

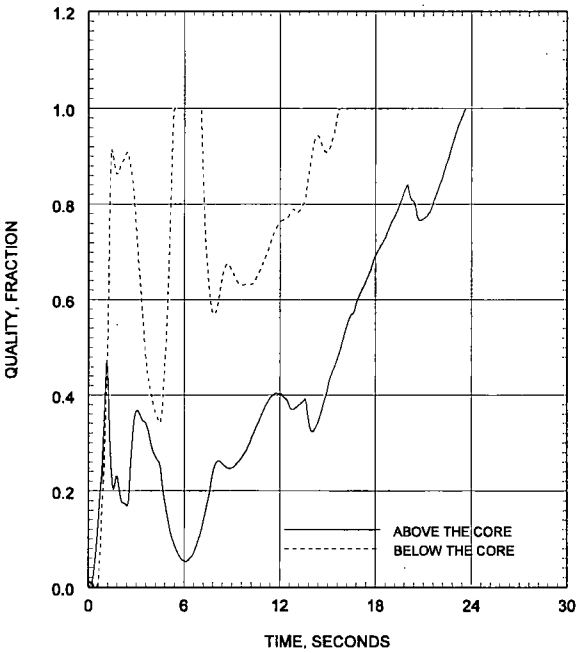


Figure 5-11
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Core Pressure Drop

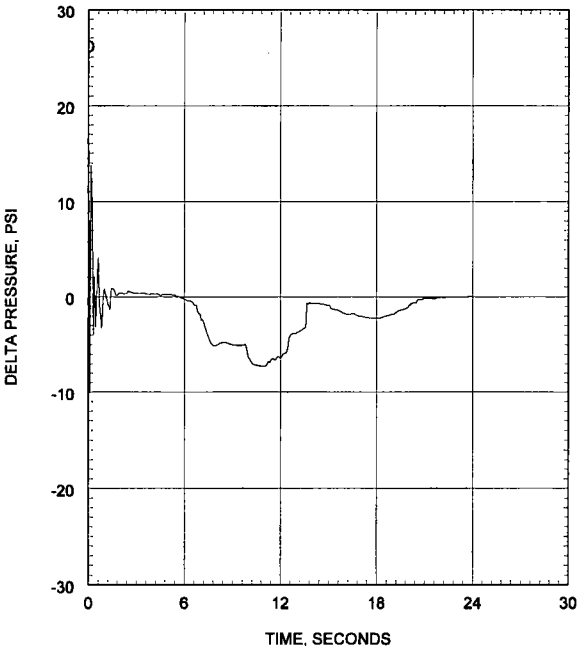


Figure 5-12
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Safety Injection Flow Rate into Intact Discharge Legs

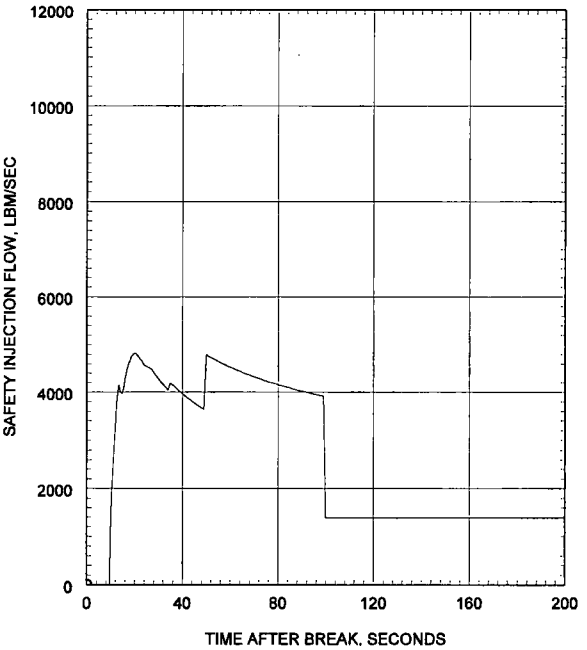


Figure 5-13
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Water Level in Downcomer During Reflood

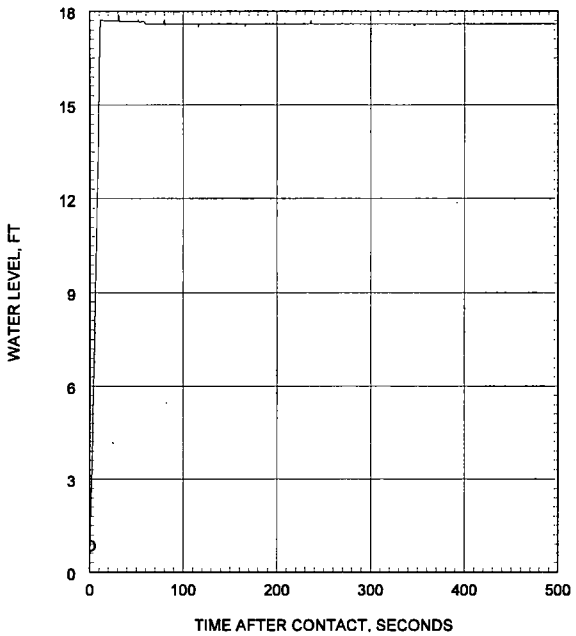


Figure 5-14
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Hot Spot Gap Conductance

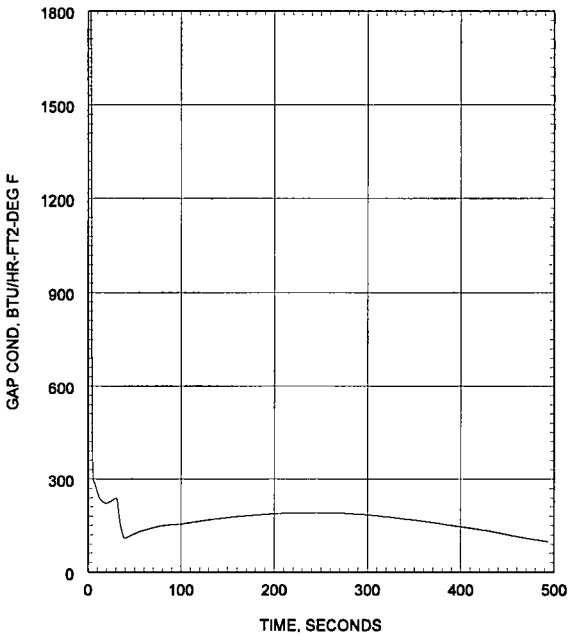


Figure 5-15
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Maximum Local Cladding Oxidation Percentage

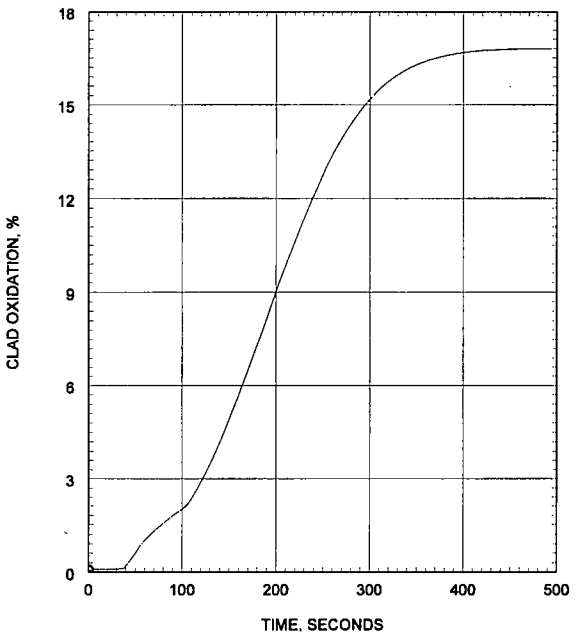


Figure 5-16
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Fuel Centerline, Fuel Average, Cladding, and Coolant Temperature at the Hot Spot

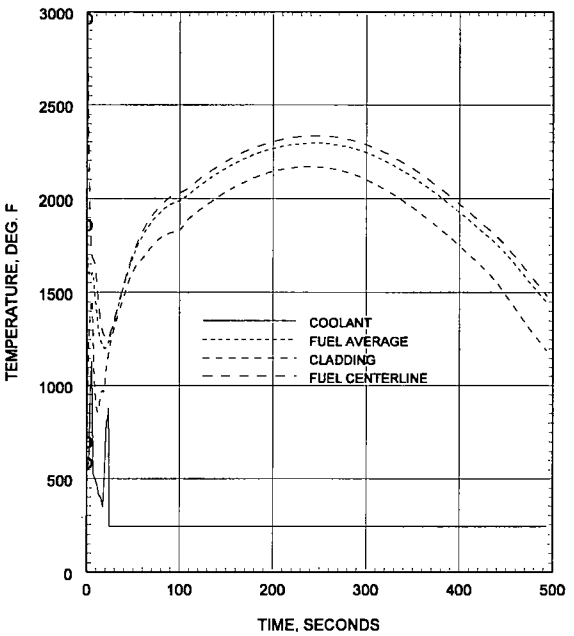


Figure 5-17
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Hot Spot Heat Transfer Coefficient

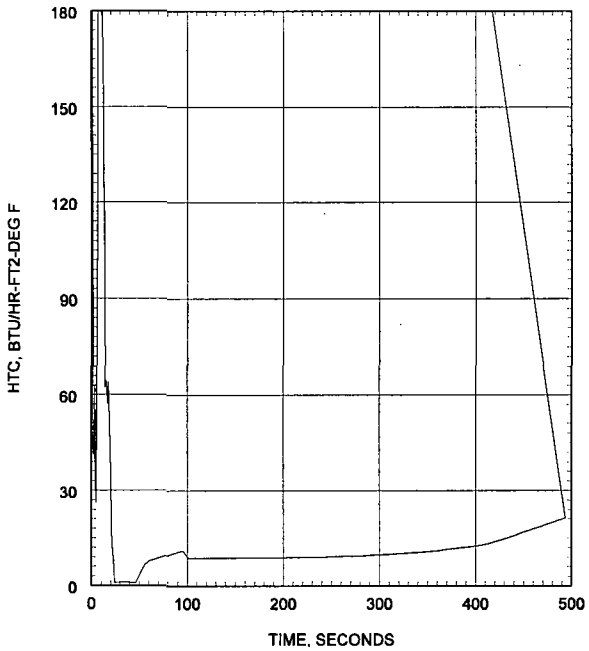


Figure 5-18
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Hot Pin Pressure

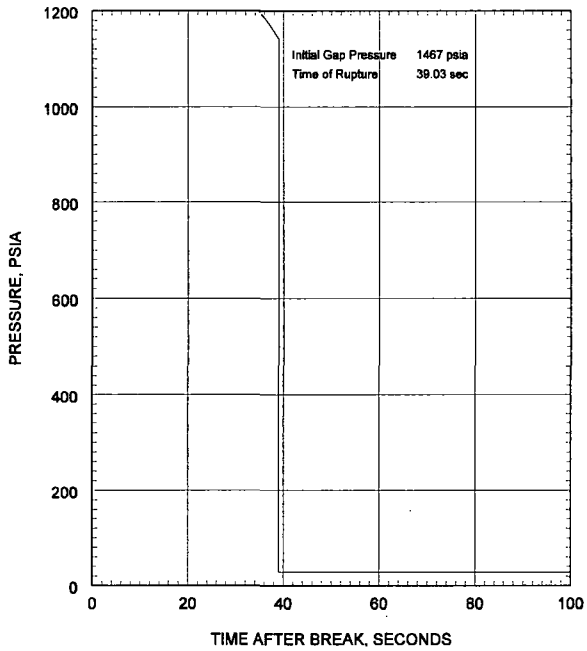


Figure 5-19
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Core Bulk Channel Flow Rate

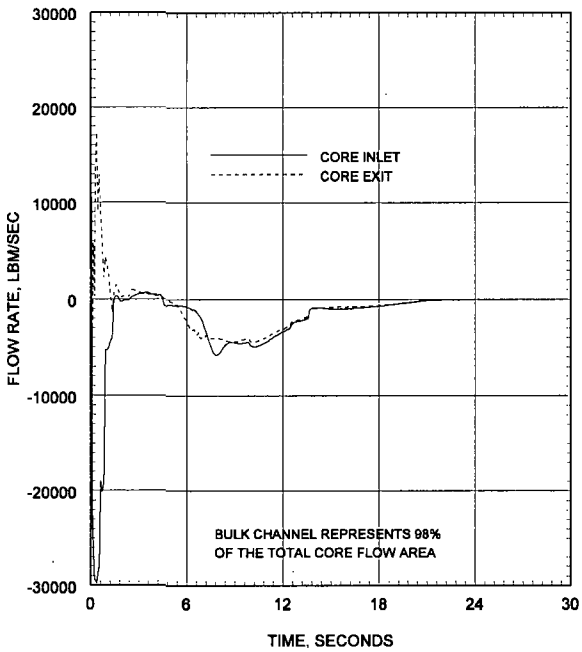


Figure 5-20
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Effective Spray and Spillage to Containment

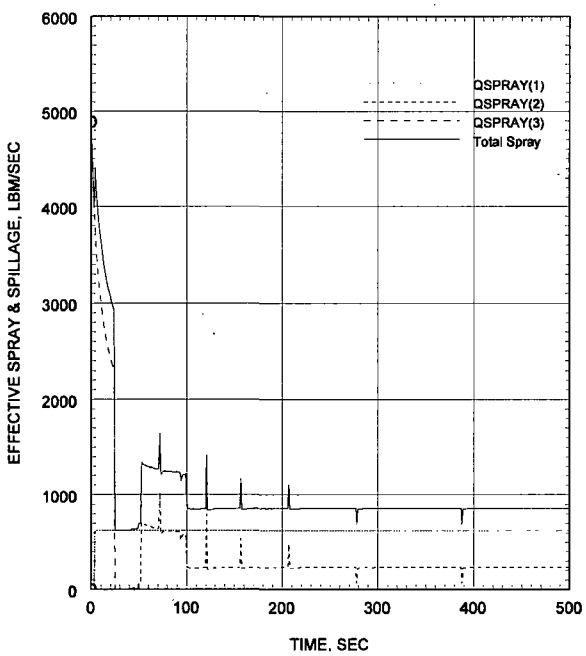


Figure 5-21
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Containment (steam) Temperature

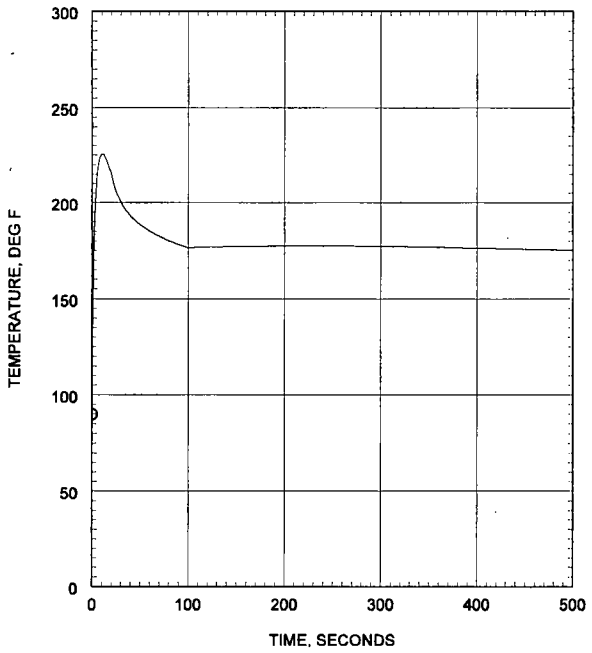


Figure 5-22
Waterford 3 NGF LBLOCA ECCS Performance Analysis
1.0 DEG/PD Break
Containment (water) Temperature

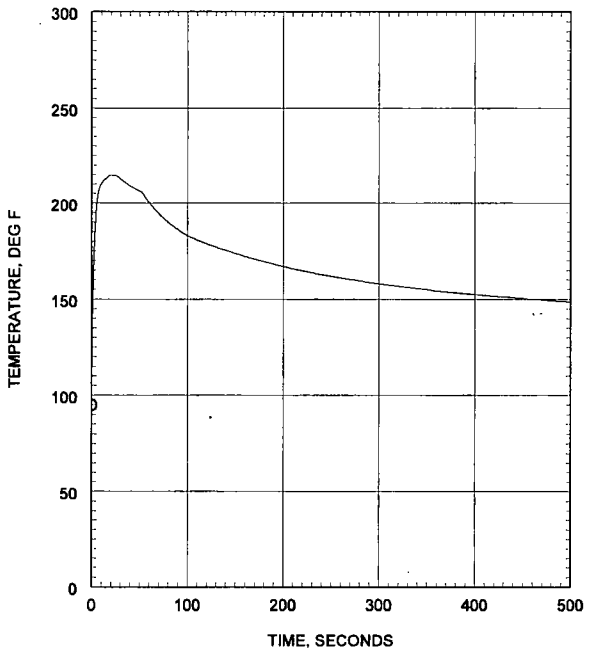


Figure 5-23
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Core Power

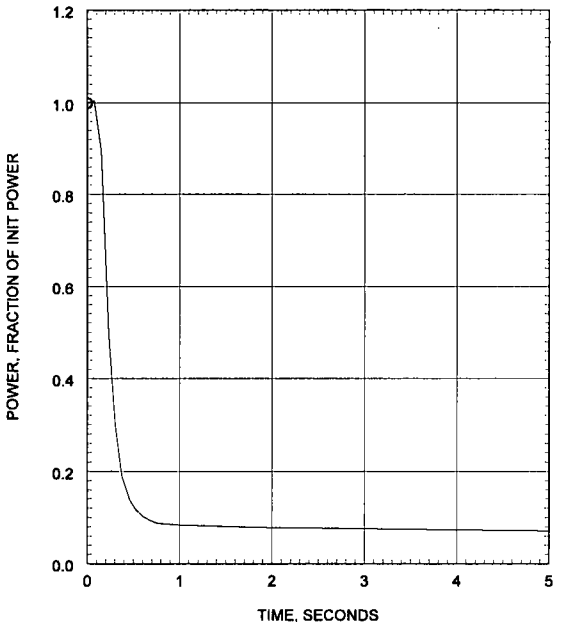


Figure 5-24
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Pressure in Center Hot Assembly Node

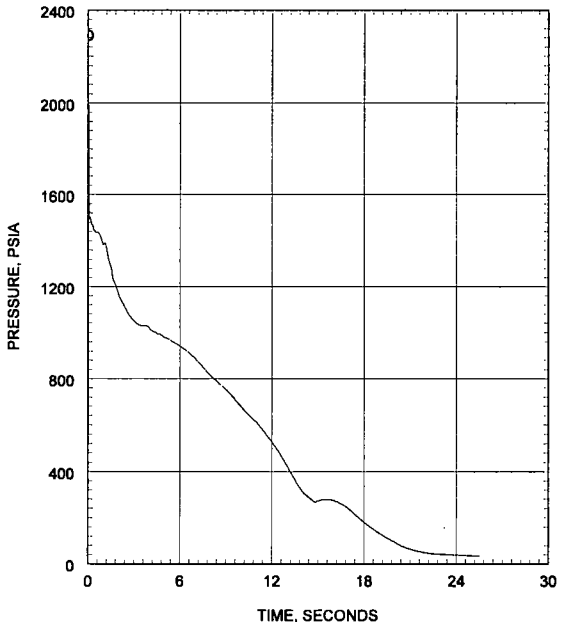


Figure 5-25
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Leak Flow Rate

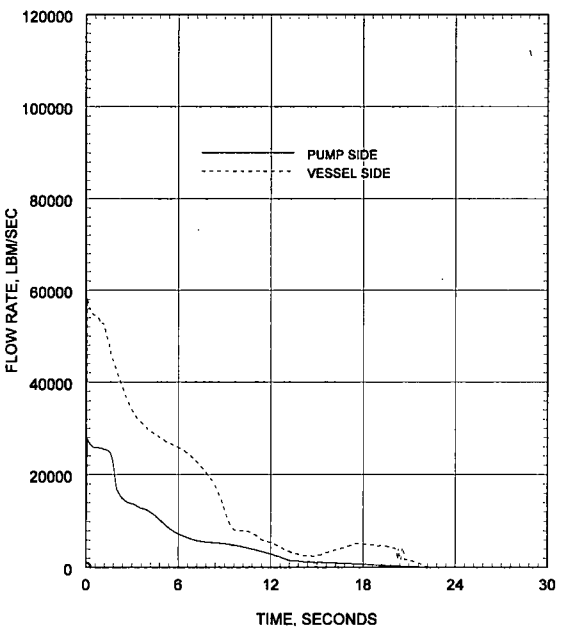


Figure 5-26
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Hot Assembly Flow Rate (Below and Above Hot Spot)

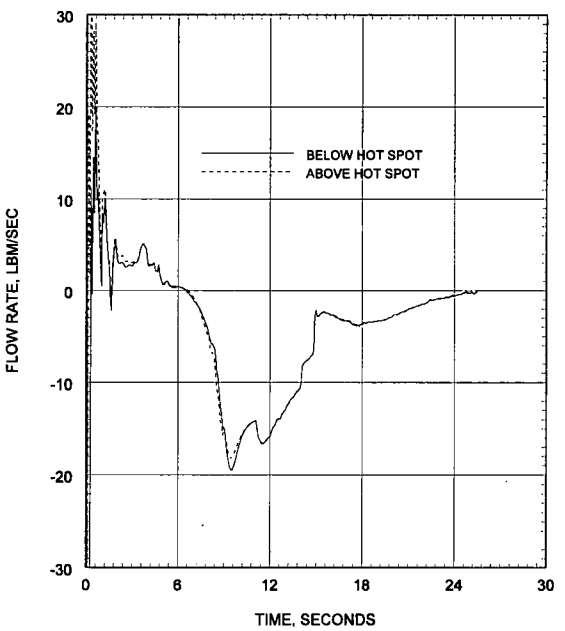


Figure 5-27
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Hot Assembly Quality

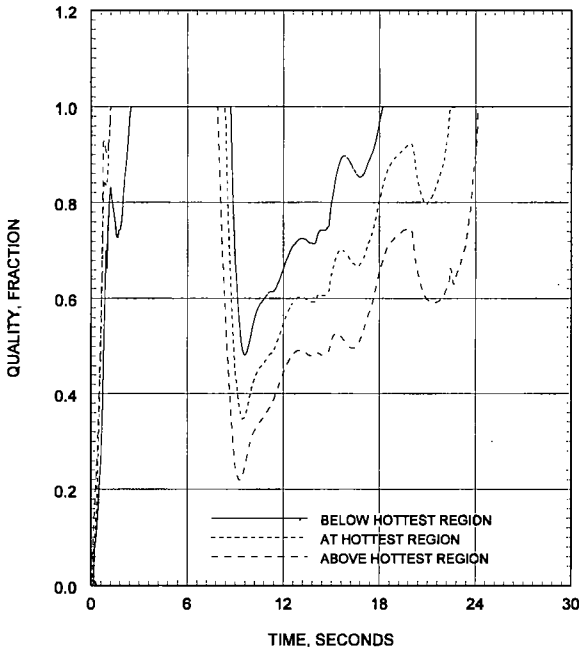


Figure 5-28
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Containment Pressure

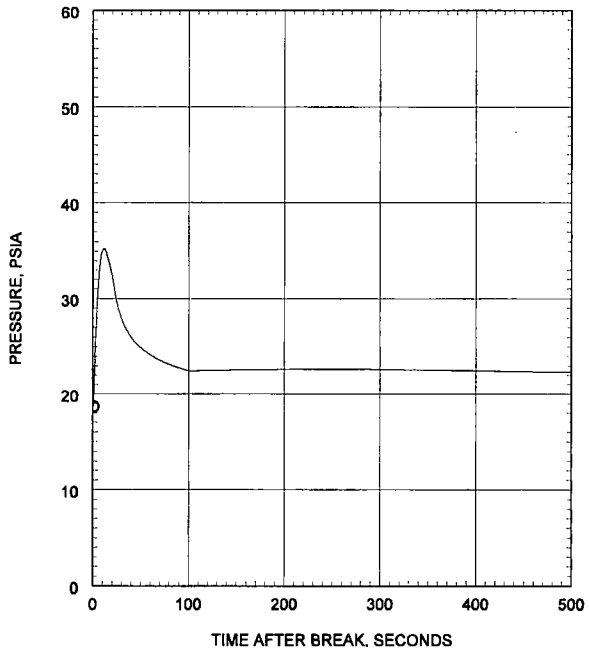


Figure 5-29
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Mass Added to Core During Reflood

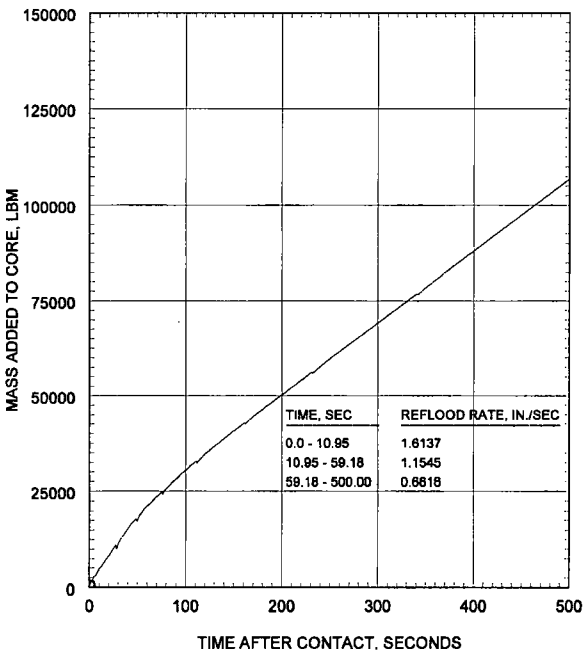


Figure 5-30
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.8 DEG/PD Break
Peak Cladding Temperature

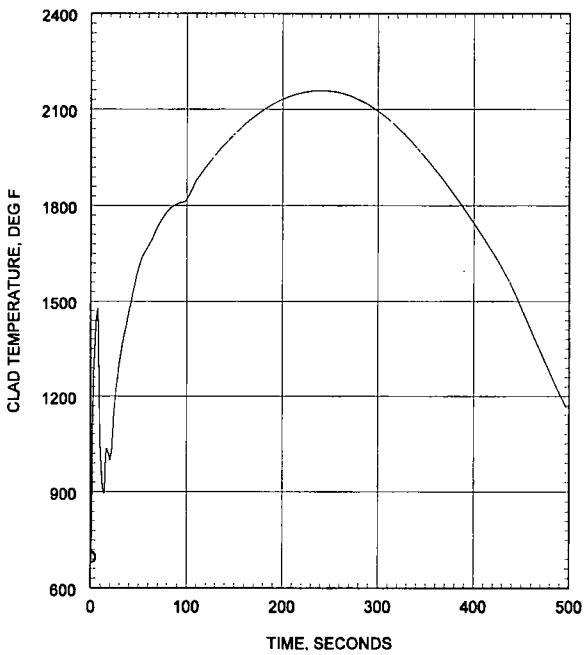


Figure 5-31
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Core Power

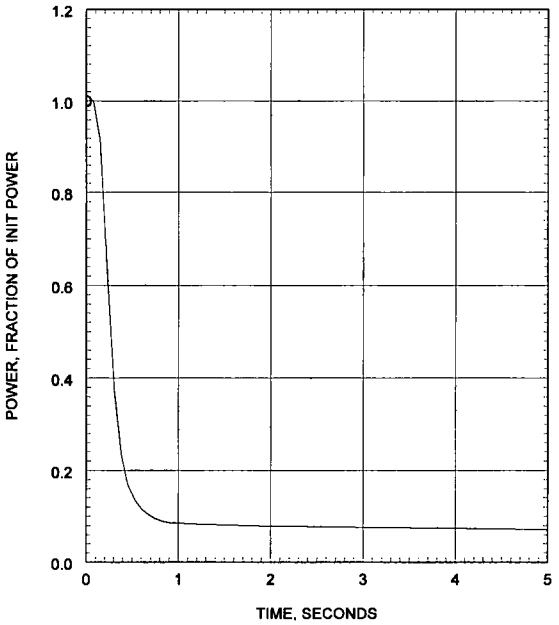


Figure 5-32
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Pressure in Center Hot Assembly Node

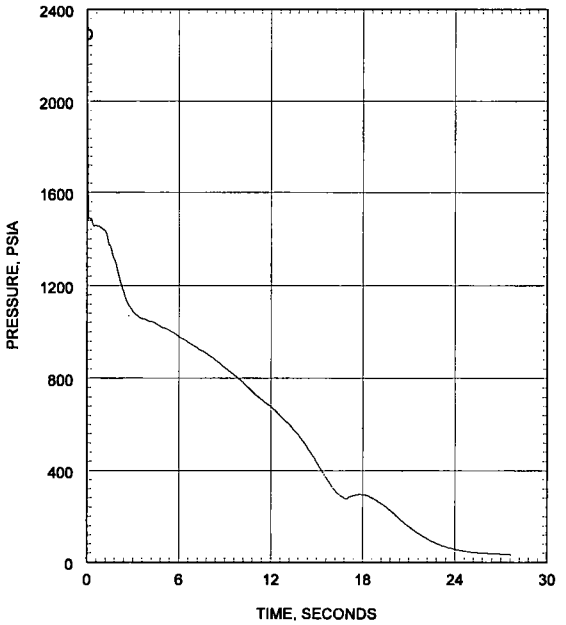


Figure 5-33
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Leak Flow Rate

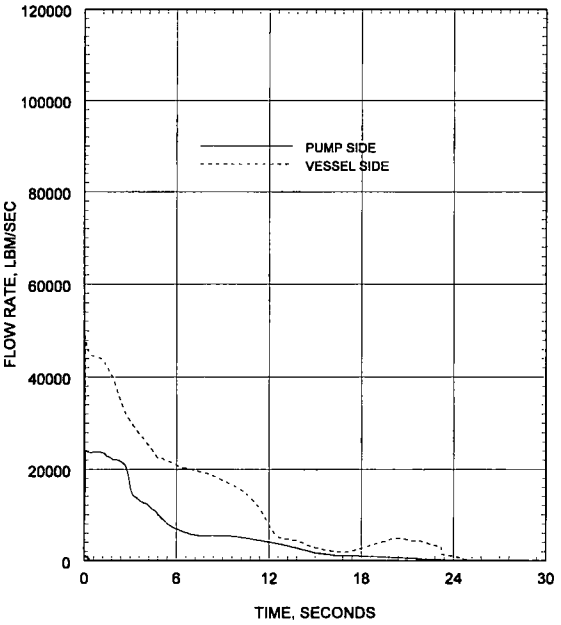


Figure 5-34
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Hot Assembly Flow Rate (Below and Above Hot Spot)

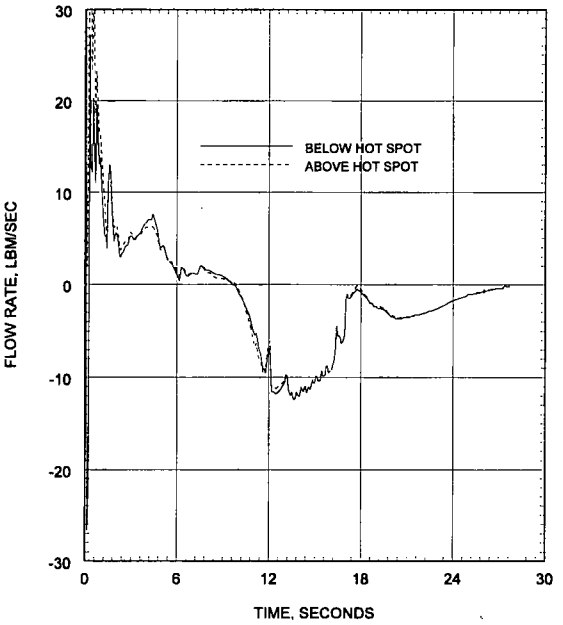


Figure 5-35
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Hot Assembly Quality

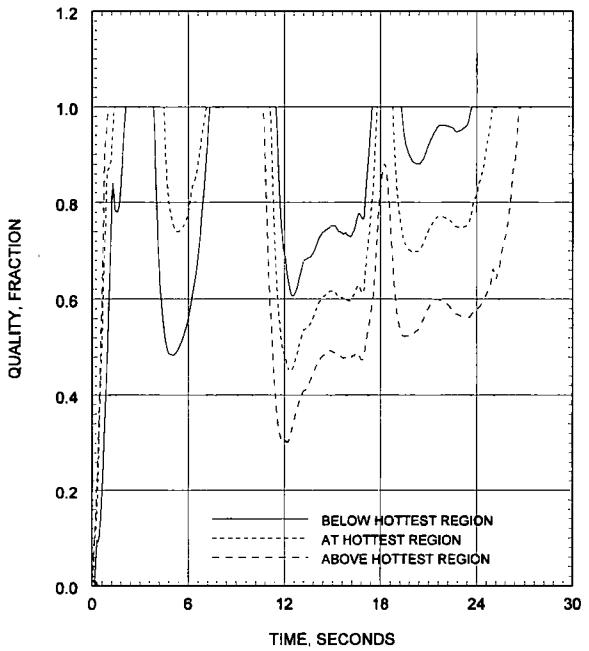


Figure 5-36
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Containment Pressure

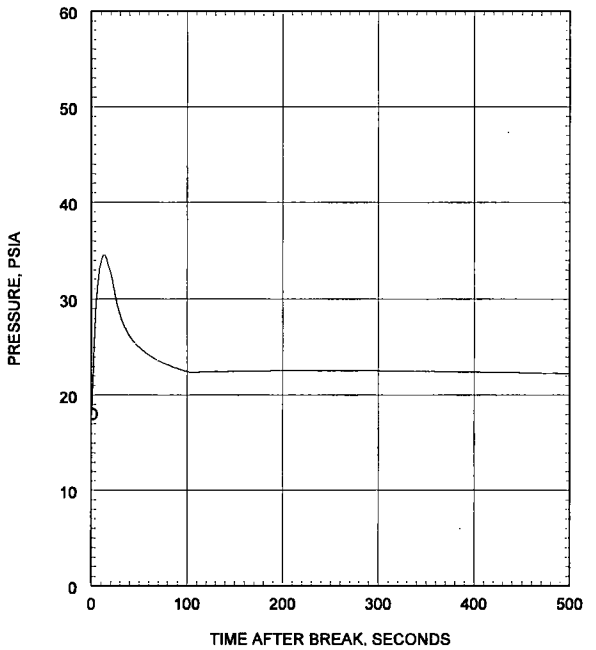


Figure 5-37
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Mass Added to Core During Reflood

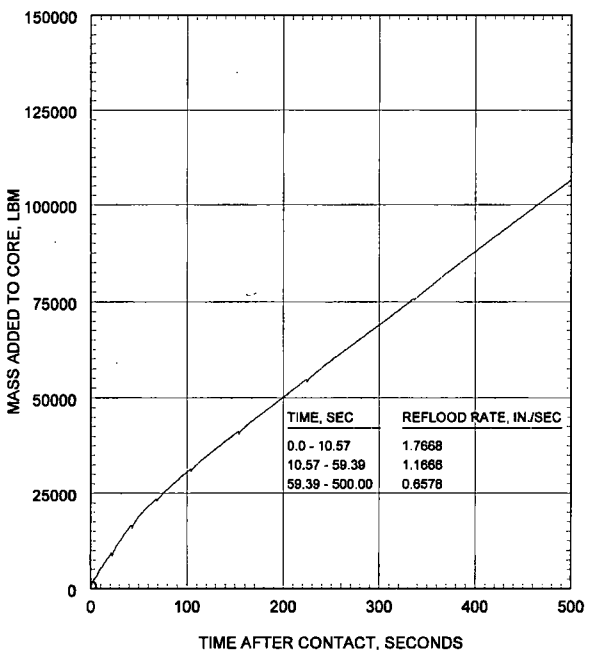


Figure 5-38
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.6 DEG/PD Break
Peak Cladding Temperature

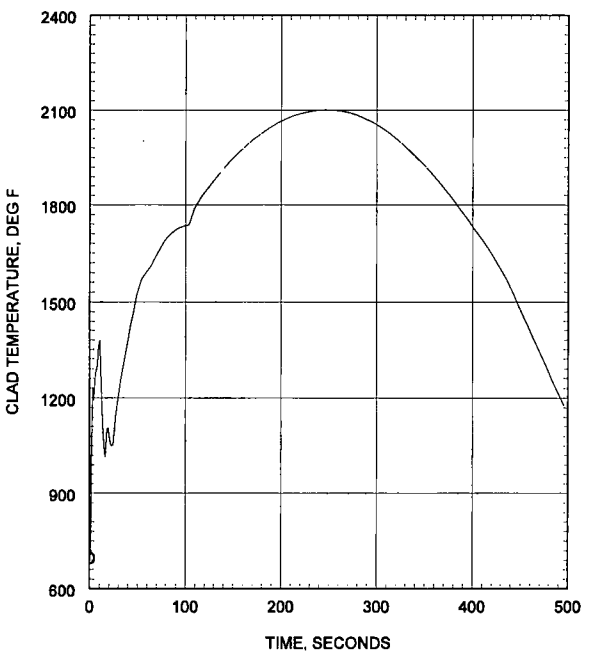


Figure 5-39
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Core Power

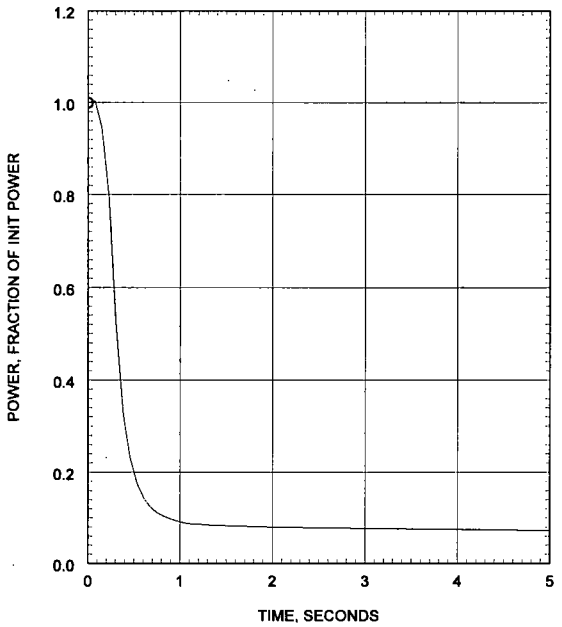


Figure 5-40
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Pressure in Center Hot Assembly Node

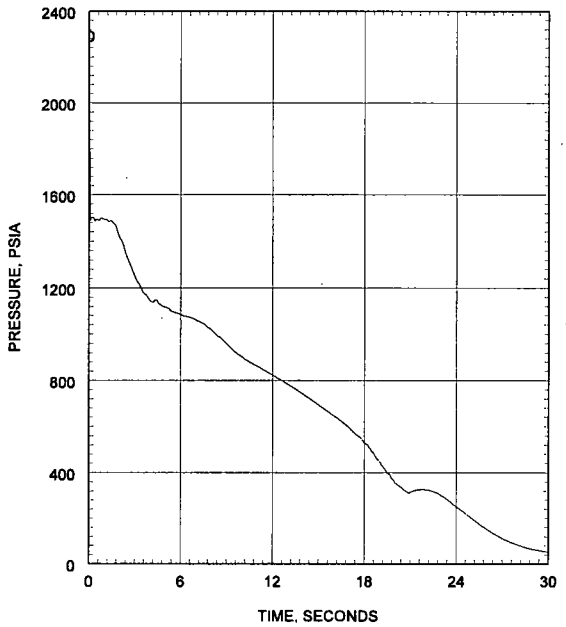


Figure 5-41
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Leak Flow Rate

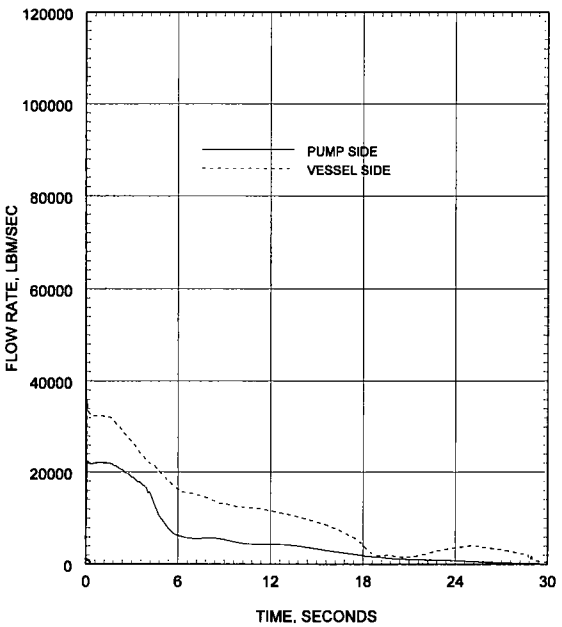


Figure 5-42
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Hot Assembly Flow Rate (Below and Above Hot Spot)

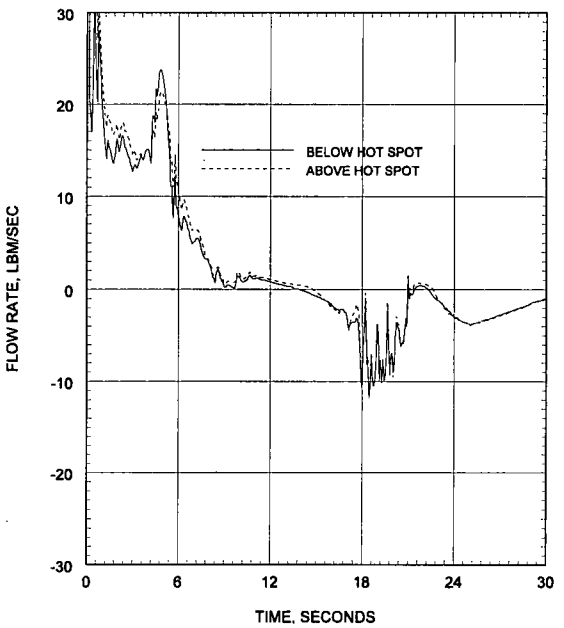


Figure 5-43
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Hot Assembly Quality

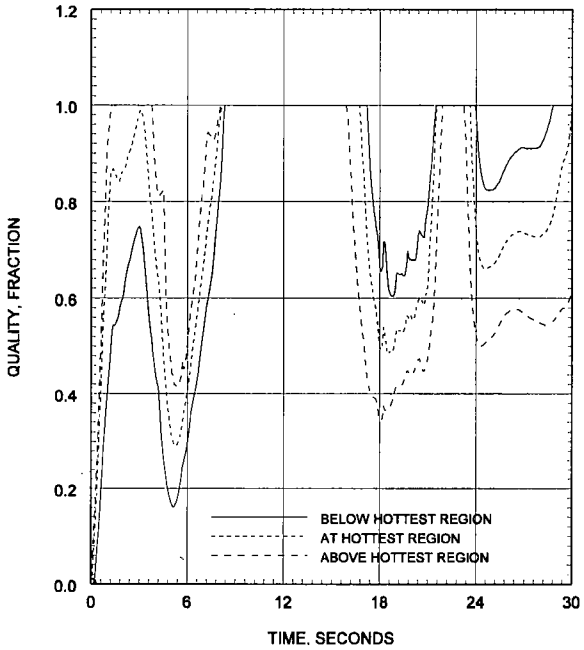


Figure 5-44
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Containment Pressure

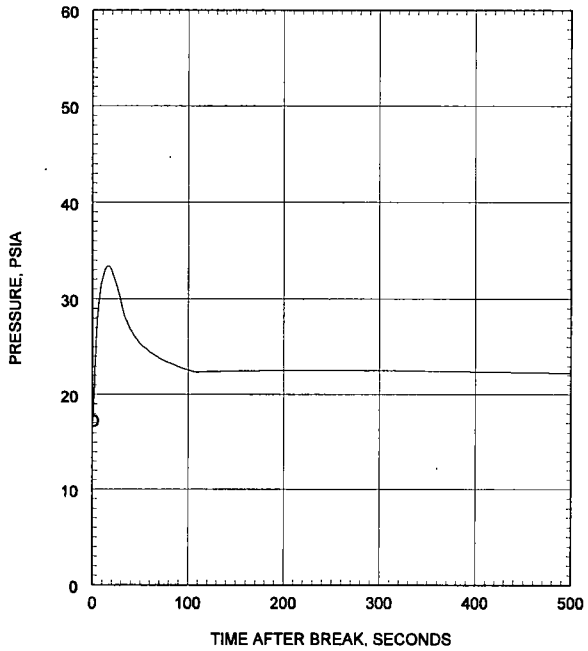


Figure 5-45
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Mass Added to Core During Reflood

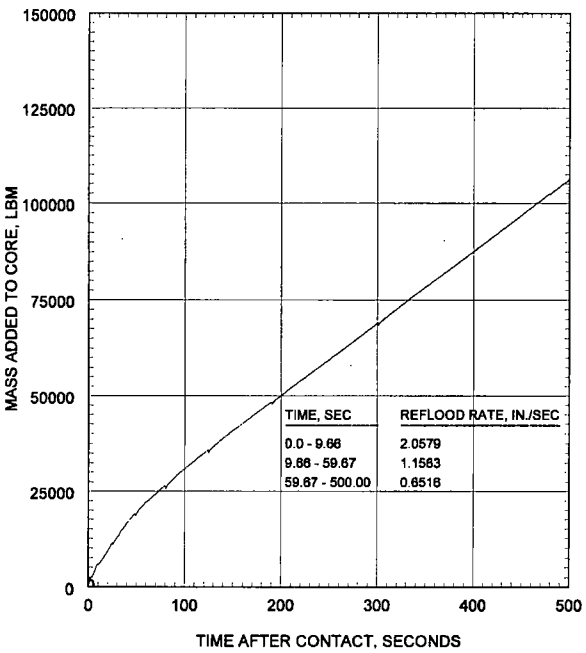


Figure 5-46
Waterford 3 NGF LBLOCA ECCS Performance Analysis
0.4 DEG/PD Break
Peak Cladding Temperature

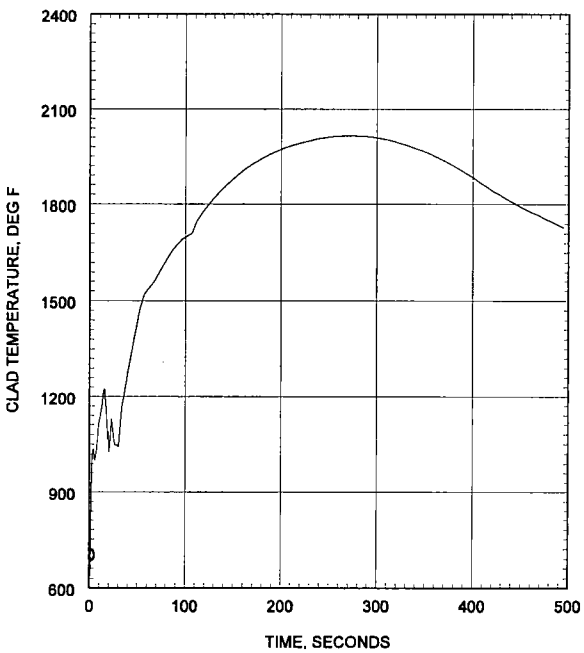


Figure 5-47
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Core Power

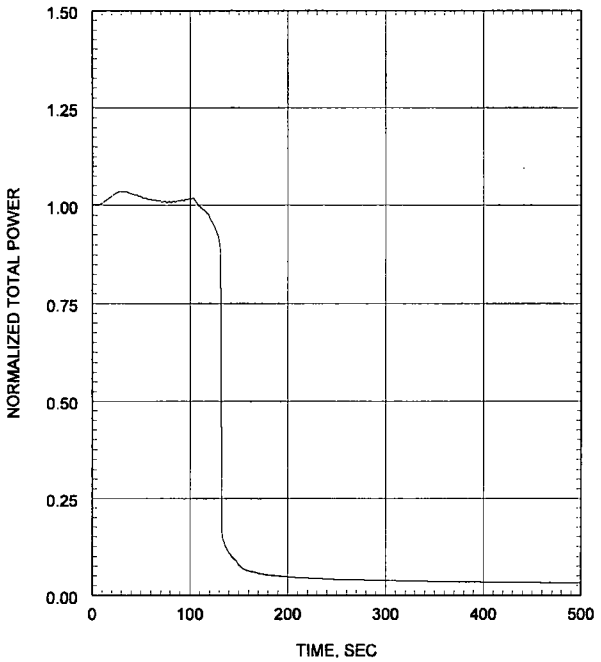


Figure 5-48
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Inner Vessel Pressure

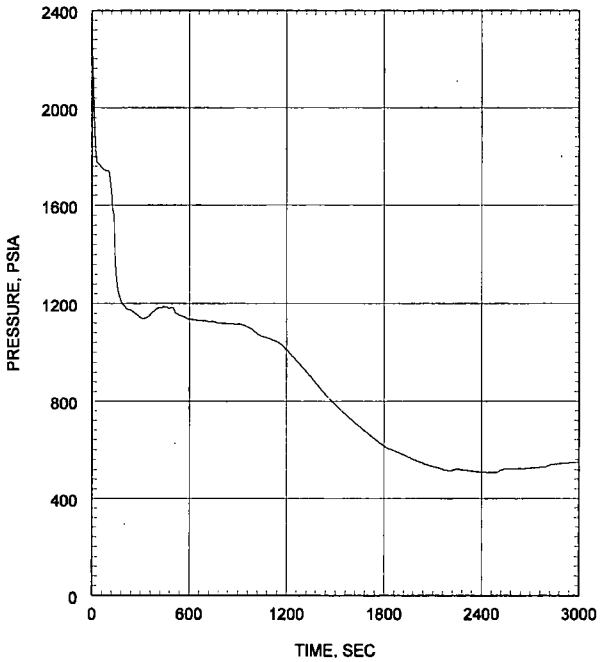


Figure 5-49
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Break Flow Rate

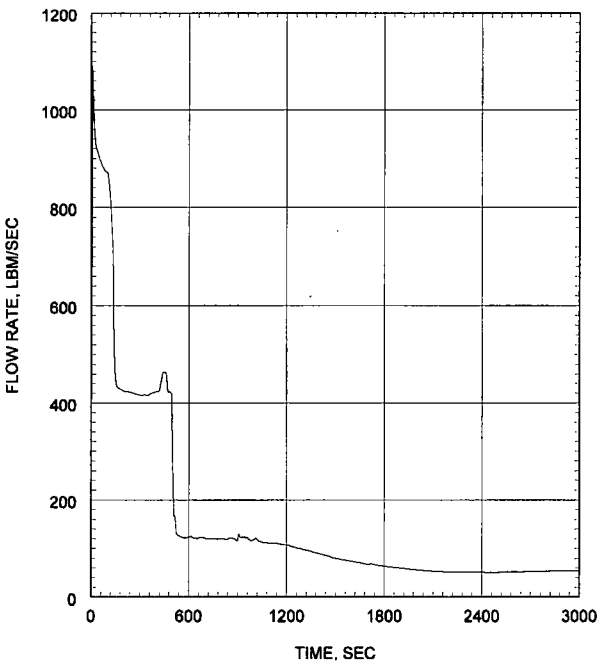


Figure 5-50
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Inner Vessel Inlet Flow Rate

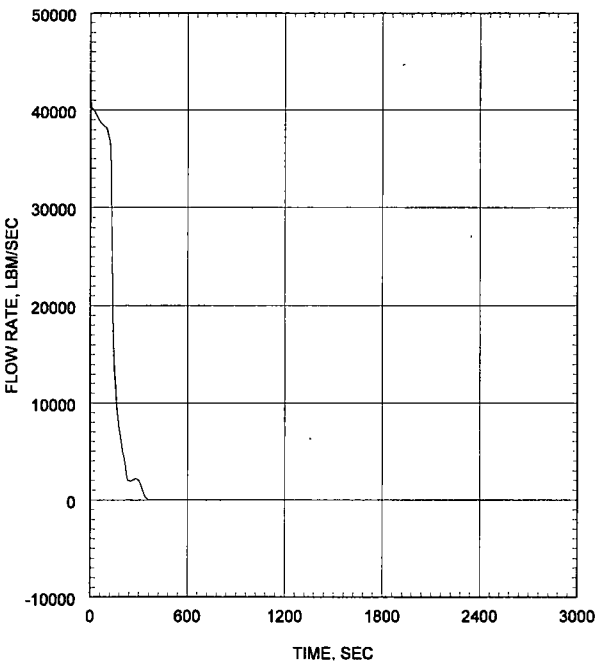


Figure 5-51
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Inner Vessel Two-Phase Mixture Level

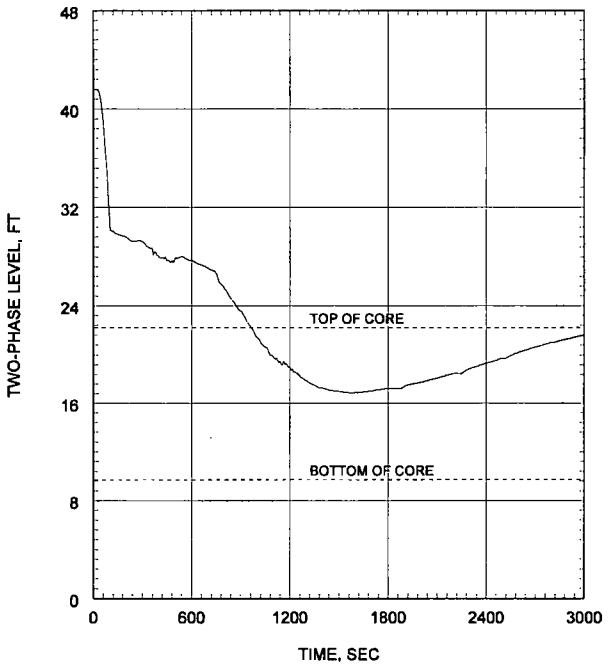


Figure 5-52
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Heat Transfer Coefficient at Hot Spot

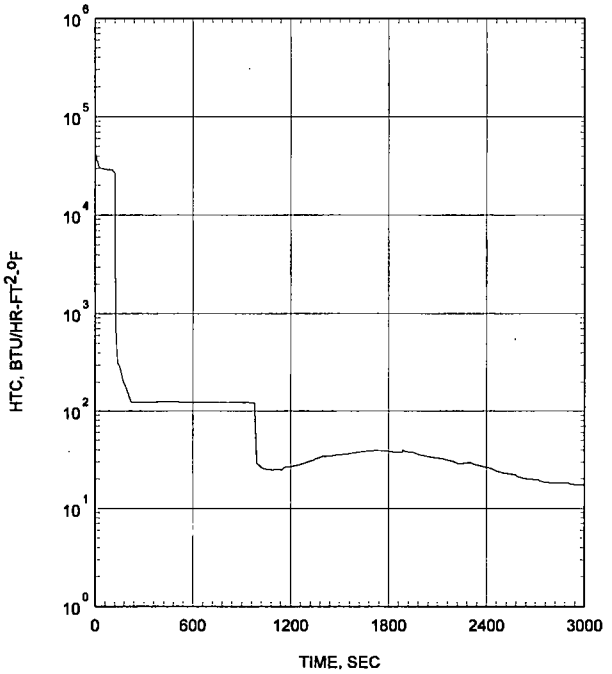


Figure 5-53
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Coolant Temperature at Hot Spot

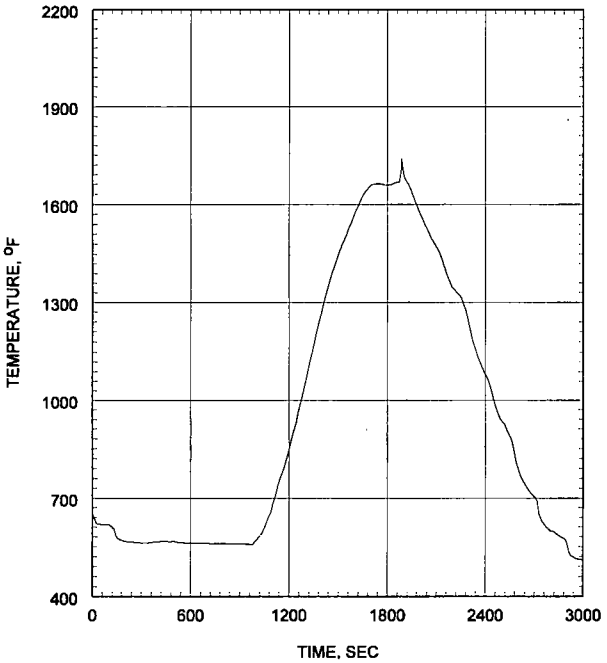


Figure 5-54
Waterford 3 NGF SBLOCA 0.05 ft²/PD Break
Cladding Temperature at Hot Spot

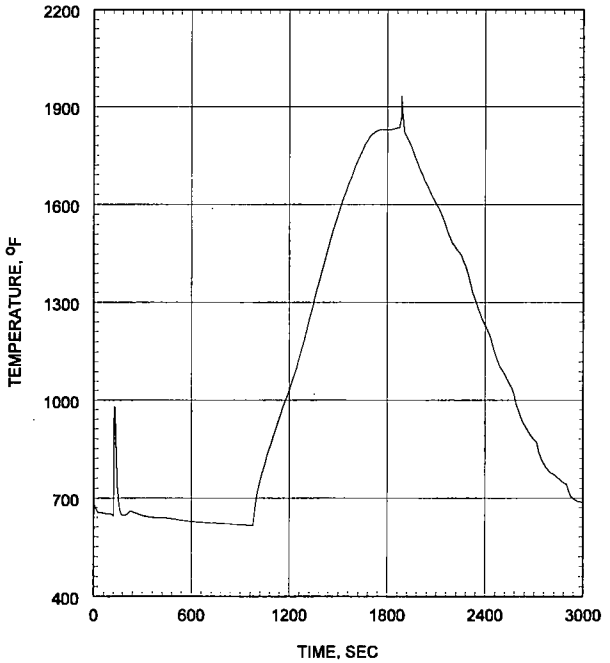


Figure 5-55
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Core Power

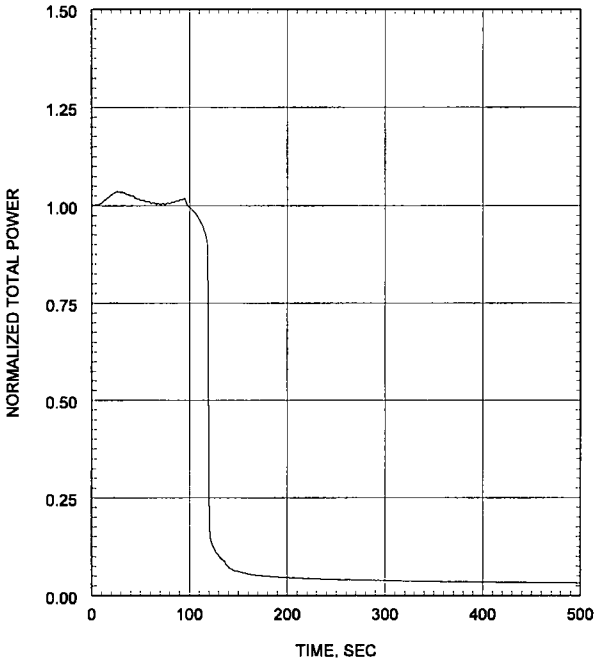


Figure 5-56
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Inner Vessel Pressure

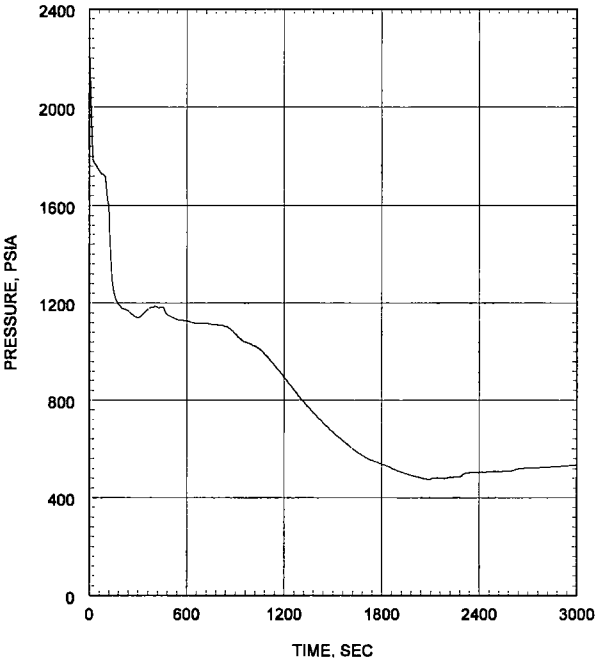


Figure 5-57
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Break Flow Rate

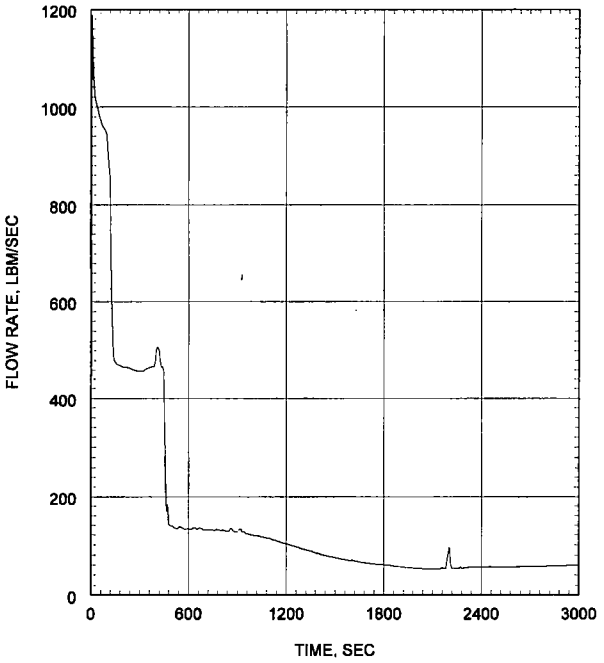


Figure 5-58
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Inner Vessel Inlet Flow Rate

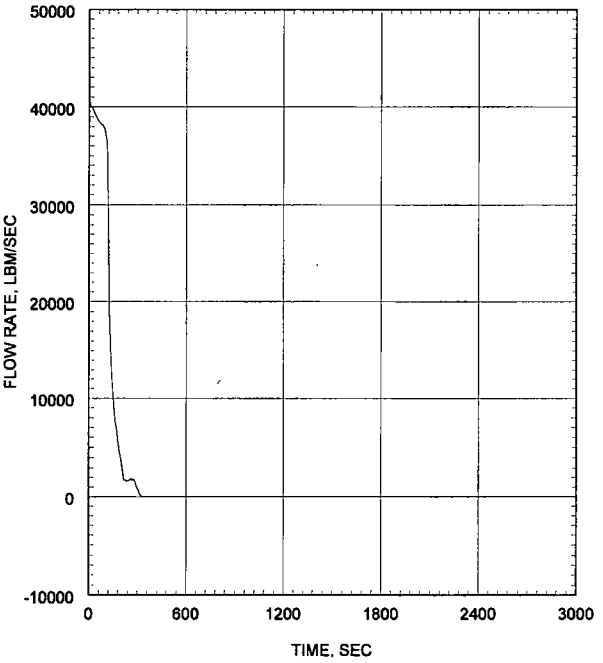


Figure 5-59
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Inner Vessel Two-Phase Mixture Level

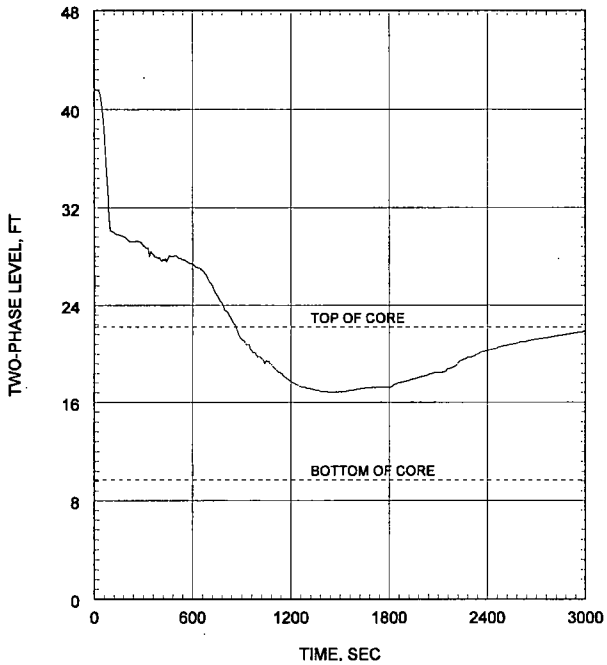


Figure 5-60
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Heat Transfer Coefficient at Hot Spot

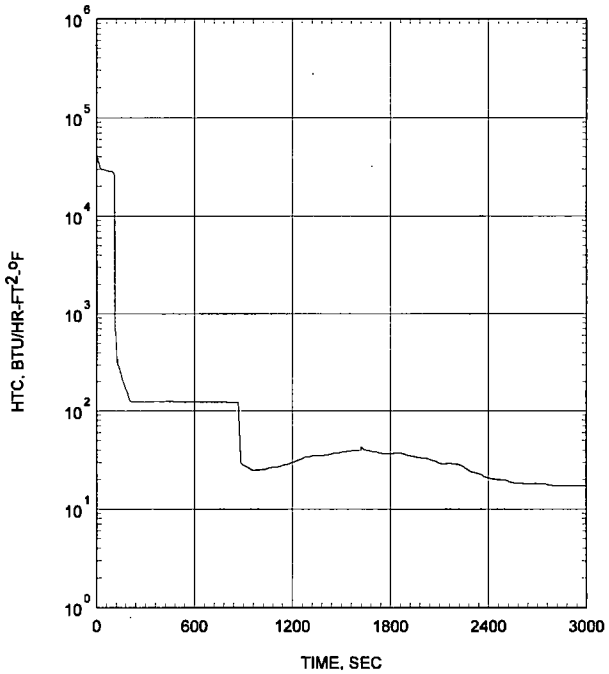


Figure 5-61
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Coolant Temperature at Hot Spot

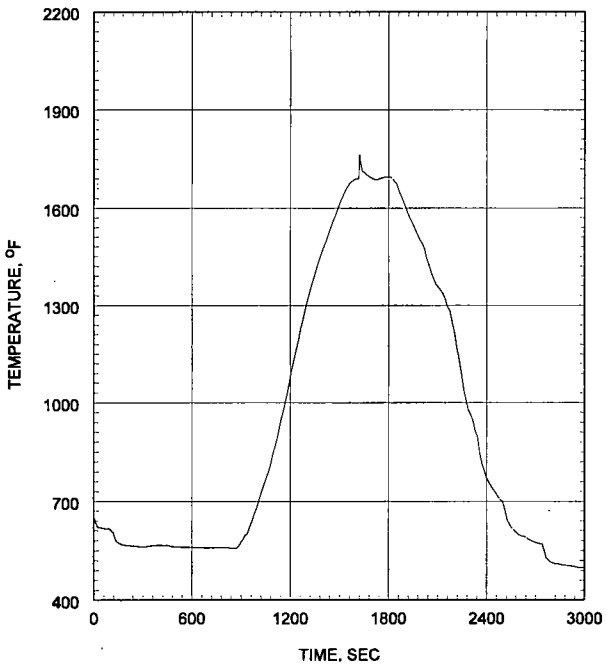


Figure 5-62
Waterford 3 NGF SBLOCA 0.055 ft²/PD Break
Cladding Temperature at Hot Spot

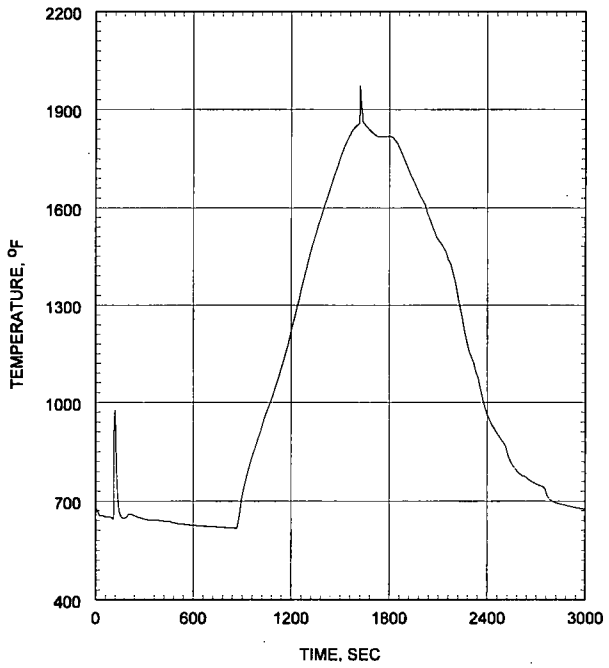


Figure 5-63
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Core Power

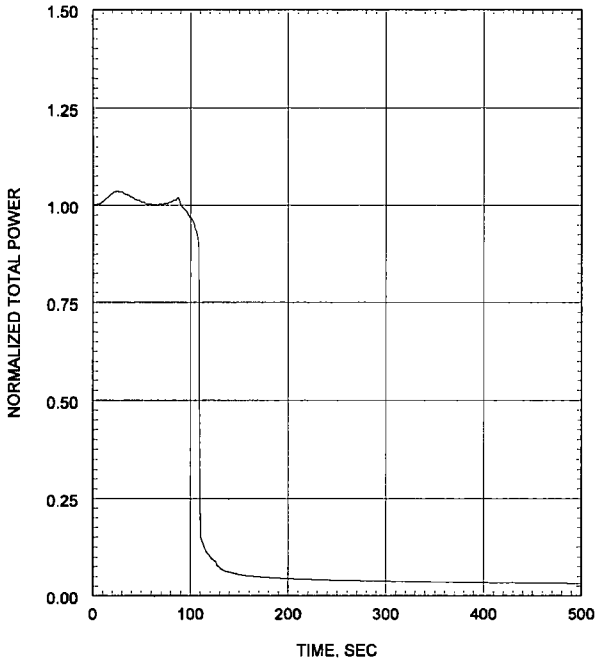


Figure 5-64
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Inner Vessel Pressure

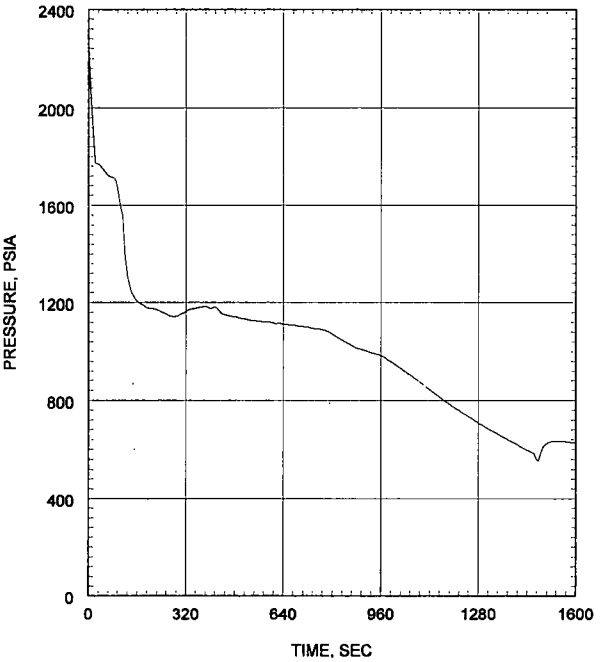


Figure 5-65
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Break Flow Rate

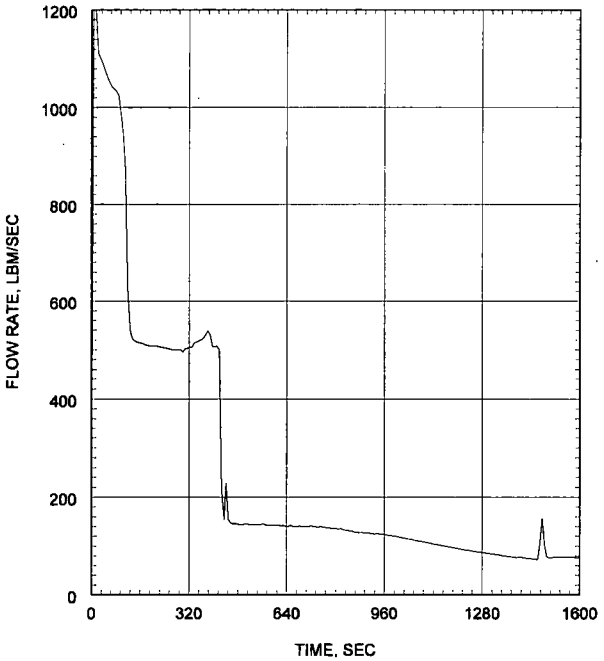


Figure 5-66
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Inner Vessel Inlet Flow Rate

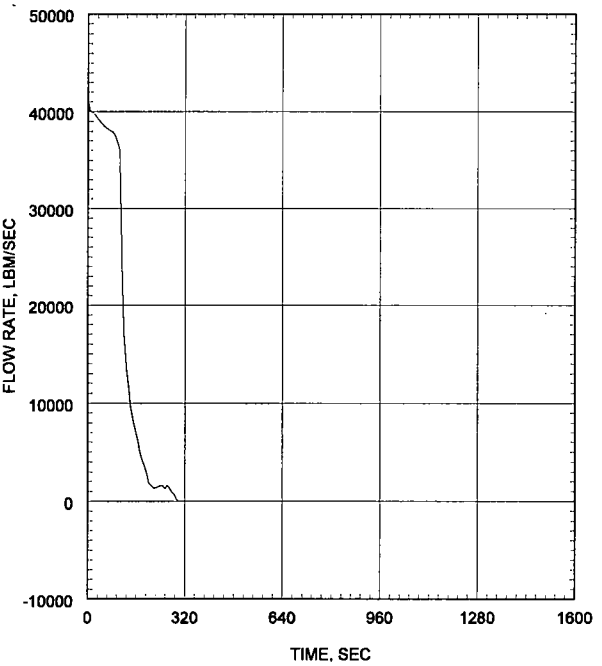


Figure 5-67
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Inner Vessel Two-Phase Mixture Level

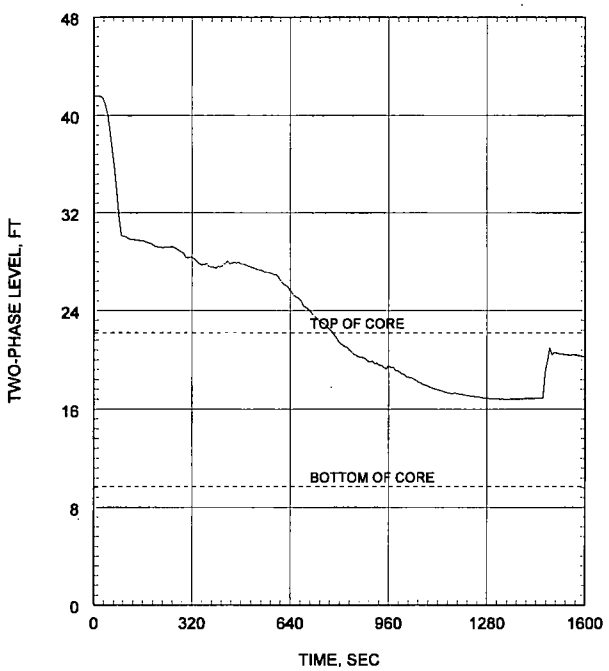


Figure 5-68
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Heat Transfer Coefficient at Hot Spot

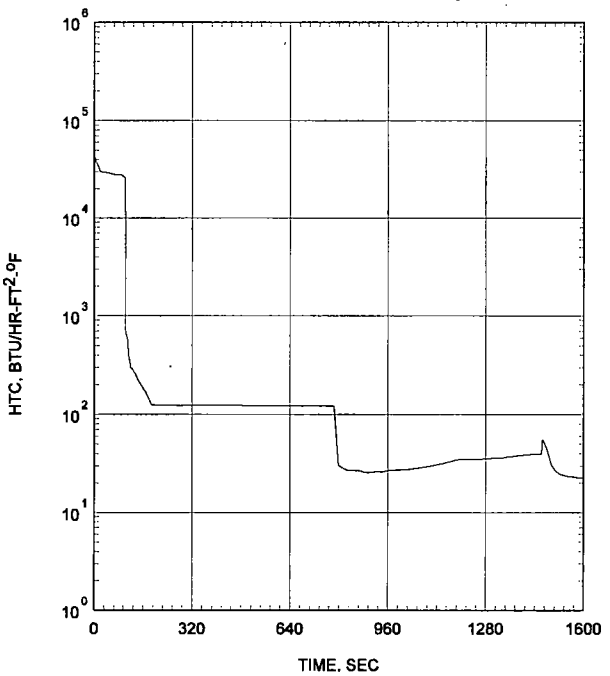


Figure 5-69
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Coolant Temperature at Hot Spot

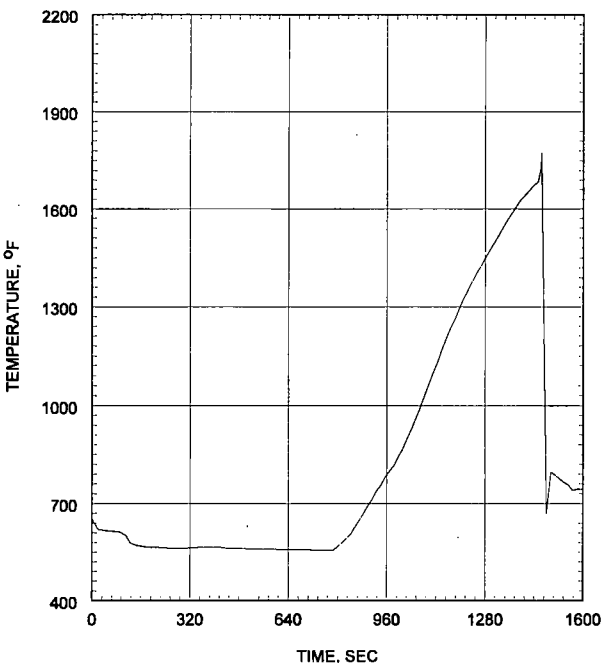
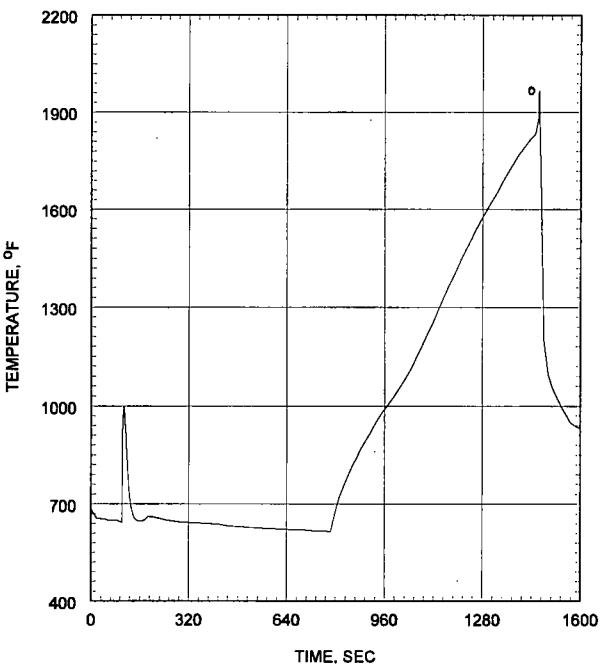


Figure 5-70
Waterford 3 NGF SBLOCA 0.06 ft²/PD Break
Cladding Temperature at Hot Spot



7.0 References

1-1 Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors."

1-2 Code of Federal Regulations, Title 10, Part 50, Appendix K, "ECCS Evaluation Models."

1-3 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

CENPD-132, Supplement 4-P-A, "Calculative Methods for the C-E Nuclear Power Large Break LOCA Evaluation Model," March 2001.

1-4 CENPD-132, Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model, Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," May 2006 and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," (TAC No. MD2161) dated June 27, 2007.

1-5 CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

1-6 CENPD-134 P, "COMPERC-II, A Program for Emergency-Refill-Reflood of the Core," August 1974.

CENPD-134 P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

1-7 CENPD-213-P, "Application of FLECHT Reflood Heat Transfer Coefficients to C-E's 16x16 Fuel Bundles," January 1976.

1-8 LD-81-095, Enclosure 1-P-A, "C-E ECCS Evaluation Model, Flow Blockage Analysis," December 1981.

1-9 CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.

CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

1-10 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April, 1977.

1-11 CENPD-139-P-A, "C-E Fuel Evaluation Model," July 1974.

CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

1-12 CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.

CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.

CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

1-13 CENPD-133P, Supplement 1, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," August 1974.

CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977.

1-14 CENPD-254-P-A, "Post-LOCA Long Term Cooling Evaluation Model," June 1980.

1-15 WCAP-16500-P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16500-P, Revision 0, "CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report," July 30, 2007.

1-16 WCAP-12610-P-A and CENPD-404-P-A Addendum 1, "Addendum 1 to WCAP-12610-P-A and CENPD-404-P-A Optimized ZIRLO™," February 2003.

1-17 Letter from H. N. Berkow (NRC) to J. A. Gresham (Westinghouse), "Final Safety Evaluation for Addendum 1 to Topical Report WCAP-12610-P-A and CENPD-404-P-A, 'Optimized ZIRLO™' (TAC No. MB8041)," June 10, 2005.

1-18 CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

1-19 WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," August 2004.

1-20 Entergy letter to the NRC dated April 24, 2007, "License Amendment Request to Allow the Use of Optimized ZIRLO™ Fuel Rod Cladding" (W3F1-2007-0020).

1-21 Entergy letter to the NRC dated August 2, 2007, "License Amendment Request NPF-38-271 to Support Next Generation Fuel" (W3F1-2007-0037).