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Atlantic

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The Northeast Utilities System

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NYN- 94035

April 4, 1994

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

References:

(a) Facility Operating License No. NPF-86, Docket No. 50-443

(b) North Atlantic letter NYN-93020, dated February 2, 1993, "Request for NRC Review and Approval of Analysis Methodologies to be Applied to Seabrook Station," T. C. Feigenbaum to USNRC

(c) USNRC Letter dated December 16, 1993, "Request for Additional Information (TAC M86957)"

(d) USNRC Letter dated December 16, 1993, "Request for Additional Information (TAC M86958)"

(e) North Atlantic letter NYN-94024, dated March 9, 1994, "Response to Request for Additional Information (TAC M86957 and TAC M86958)," T. C. Feigenbaum to USNRC

Subject: Response to Request for Additional Information (TAC M86957 and TAC M86958)

Gentlemen:

North Atlantic Energy Service Corporation (North Atlantic) has provided in the Enclosures non-proprietary versions of additional information regarding the application to Seabrook Station of Yankee Atomic Electric Company Topical Reports YAEC-1856P, "System Transient Analysis Methodology Using Retran for PWR Applications", and YAEC-1849P, "Thermal Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications". This information was requested by the NRC staff [References (c) and (d)]. The proprietary responses were previously submitted by Reference (e). The pages with proprietary information have been deleted from this submittal. North Atlantic believes that this information will support the completion of the review of YAEC-1856P and YAEC-1849P [Reference (b)].

Should you have any questions regarding this letter, please contact Mr. Terry L. Harpster, Director of Licensing Services, at (603) 474-9521, extension 2765.

Very truly yours,

Ted Feigenbaum

Ted C. Feigenbaum

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North Atlantic
April 4, 1994

ENCLOSURE 1 TO NYN-94035

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REVIEW OF YAEC-1849P

NON-PROPRIETARY VERSION

FIGURE 3

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c. Mixing Technology

The rate of heat exchange by mixing between flow channels is proportional to the difference in the local mean fluid enthalpy of the respective channels of the local fluid density and flow velocity. The proportionality is expressed by the dimensionless thermal diffusion coefficient (TDC) which is defined as:

$$TDC = \frac{w'}{\rho V a} \qquad (4.4\text{-}9)$$

where:

w' = flow exchange rate per unit length, (lb_m/ft-sec)
ρ = fluid density, lb_m/ft^3
V = fluid velocity, ft/sec

a = lateral flow area between channels per unit length, ft^2/ft

The application of the TDC in the THINC analysis for determining the overall mixing effect or heat exchange rate is presented in Reference 7.

As a part of an ongoing research and development program, Westinghouse has sponsored and directed mixing tests at Columbia University (Reference 12). These series of tests, using the "R" mixing vane grid design on 13, 26 and 32 inch grid spacing, were conducted in pressurized water loops at Reynolds numbers similar to that of PWR core under the following single and two phase (subcooled boiling) flow conditions:

Pressure	1500 to 2400 psia
Inlet temperature	332 to 642°F
Mass velocity	1.0 to 3.5×10^6 $lb_m/hr\text{-}ft^2$
Reynolds number	1.34 to 7.45×10^5
Bulk outlet quality	-52.1 to 13.5%

TDC is determined by comparing the THINC Code predictions with the measured subchannel exit temperatures. Data for 26 inch axial grid spacing are presented in Figure 4.4-4 where the thermal diffusion coefficient is plotted versus the Reynolds number. TDC is found to be independent of Reynolds number, mass velocity, pressure and quality over the ranges tested. The two phase data (local, subcooled boiling) fell within the scatter of the single phase data. The effect of two-phase flow on the value of TDC has been demonstrated by Cadek (Reference 12), Rowe and Angle (References 13 and 14), and Gonzalez-Santalo and Griffith (Reference 15). In the subcooled boiling region, the values of TDC were indistinguishable from the single phase values. In the quality region, Rowe and Angle show that in the case with rod spacing similar to that in PWR reactor core geometry, the value of TDC increased with quality to a point and then decreased, but never below the single phase value. Gonzalez-Santalo and Griffith showed that the mixing coefficient increased as the void fraction increased.

The data from these tests on the "R" grid showed that a design TDC value of 0.038 (for 26 inch grid spacing) can be used in determining the effect of coolant mixing in the THINC analysis.

A mixing test program similar to the one described above was conducted at Columbia University for the 17x17 geometry and mixing vane grids on 26 inch spacing (Reference 16). The mean value of TDC obtained from these tests was 0.059, and all data was well above the current design value of 0.038.

FIGURE 3 (Continued)

Since the actual reactor grid spacing is approximately 20 inches, additional margin is available for this design, as the value of TDC increases as grid spacing decreases (Reference 12).

4.4.7 References

7. Chelemer, H., Weisman, J. and Tong, L. S., "Subchannel Thermal Analysis of Rod Bundle Cores," WCAP-7015, Revision 1, January 1969.

12. Cadek, F. F., Motley, F. E. and Dominicis, D. P., "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-P-A (Proprietary), January 1975 and WCAP-7959-A, January 1975.

13. Rowe, D. S., Angle, C. W., "Crossflow Mixing Between Parallel Flow Channels During Boiling, Part II Measurements of Flow and Enthalpy in Two Parallel Channels," BNWL-371, part 2, December 1967.

14. Rowe, D. S., Angle, C. W., "Crossflow Mixing Between Parallel Flow Channels During Boiling, Part III Effect of Spacers on Mixing Between Two Channels," BNWL-371, part 3, January 1969.

15. Gonzalez-Santalo, J. M. and Griffith, P., "Two-Phase Flow Mixing in Rod Bundle Subchannels," ASME Paper 72-WA/NE-19.

16. Motley, F. E., Wenzel A. H., Cadek, F. F., "The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8298-P-A (Proprietary), January 1975 and WCAP-8299-A, January 1975.

North Atlantic
April 4, 1994

ENCLOSURE 2 TO NYN-94035

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REVIEW OF YAEC-1856P

NON-PROPRIETARY VERSION

RESPONSE TO
REQUEST FOR ADDITIONAL INFORMATION
REVIEW OF YAEC-1856P

1. *Identify clearly and accurately transient analyses performed with RETRAN for which YAEC received NRC approval. YAEC should identify: (1) which version of RETRAN was used, (2) the scope of application approved, such as reload and transient specificicity, (3) for which plant and (4) references to NRC approvals.*

RESPONSE TO QUESTION 1:

YAEC has performed transient analysis using RETRAN since the early 1980's. An overview of the history of YAEC's NRC approved applications is provided in the attached Table. As noted in the Table, YAEC has approved applications of RETRAN for Main Steam Line Break and Loss of load events for PWR's and the full range of transients for reload applications on BWR's. For the most part these applications have been in conjunction with reload analyses. The Loss of Load analysis application on Seabrook was directed at revising the safety valve setpoint tolerance from 1% to 3%. With the approval of this application this analysis is now the analysis of record for Seabrook Station. YAEC's reload analysis experience was summarized in Table 2.1 of YAEC-1856P. It should also be noted that the reference numbers given in the attached Table refer to references given in YAEC-1856P.

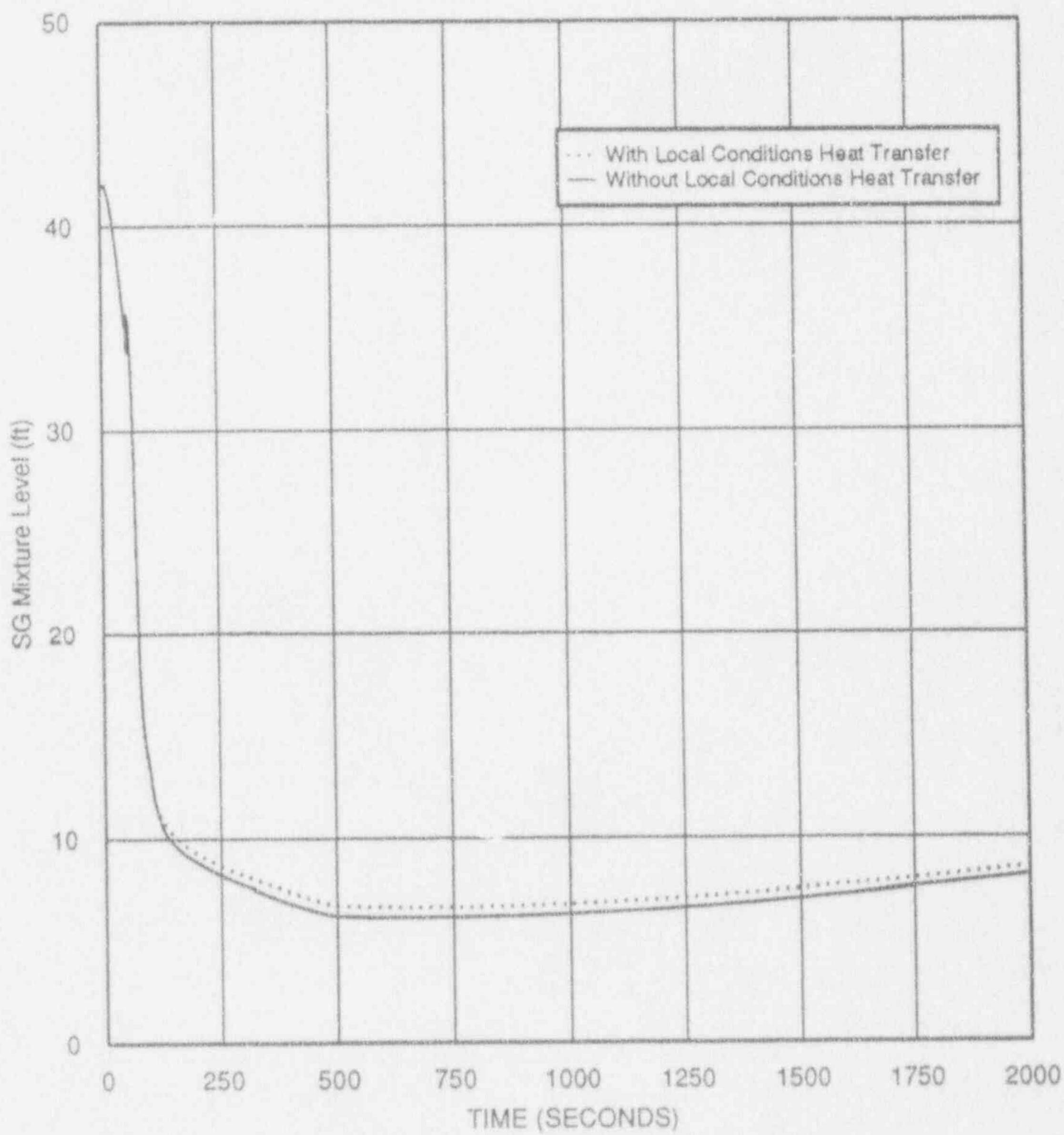
The Main Steam Line Break (MSLB) applications employed on Maine Yankee and Yankee Rowe in the 1981, 1983, and 1984-85 time frames used RETRAN and a criteria of no-return-to-power to establish shutdown margin requirements. This approved technology was extended in 1990-1991 in conjunction with the STAR code to treat cases with the potential for a return-to-power. STAR is a 3-D space-time treatment of the reactor kinetics with 3-D thermal-hydraulics (VIPRE). STAR is used in conjunction with RETRAN to predict the overall response to MSLB including the return to power following the extended cooldown. The thermal-hydraulic approach with RETRAN was consistent with previously approved YAEC applications approved in the early to mid-80's. The modeling of the reactor vessel was modified to capture the effects related to the faulted and intact loops as they impact the core neutronic feedback. The approved Star application also included rod ejection. Both the MSLB and rod ejection application was approved on a generic basis applicable to PWRs.

As noted in the attached Table, YAEC has applied a number of versions of RETRAN. These applications have coincided with the release/ NRC approval of the corresponding version of the code. YAEC is currently using RETRAN02 MOD05 in all of its applications. The options selected for each application are consistent with the original NRC approval. The upgraded versions have been tested through our quality assurance process to assure that results obtained with the later versions are consistent with the approved applications.

OVERVIEW OF YAEC LICENSING HISTORY

RETRAN

	PLANT	TRANSIENT	SCOPE	VERSION	SUBMITTAL	NRC APPROVAL
1981	Maine Yankee	MSLB	Reload	01 MOD 3	Ref. 9	Ref. 11
1981	Vermont Yankee	BWR reload transients	Reload	15F	YAEC-1233	NRC SER Nov. 1981
1983	Yankee Rowe	MSLB	Reload	02 MOD 2	Ref. 8	Ref. 10
1984-85	Maine Yankee	MSLB	Reload	02 MOD 2	YAEC-1447	NRC SER Oct. 1985
1989	Vermont Yankee	BWR transients 1-D kinetics	Generic	02 MOD 04	Ref. 12	Ref. 13
1990-91	Maine Yankee	MSLB/rod ejection	Generic	02 MOD 05 STAR/CHIC-KIN	Ref. 15	Ref. 14
1991	Seabrook	SGTR	Address licensing issue	02 MOD 02	Ref. 16	--
1992	Seabrook	Loss of Load	Safety valve	02 MOD 05	Ref. 17	Ref. 18



SEABROOK STATION
RETRAN SENSITIVITY STUDY

Loss of Feedwater -
Sensitivity of RETRAN's SG Mixture Level to the Use of Local Conditions Heat Transfer

FIGURE 3.1

4. *On page A.10, YAEC briefly described its special technique to preserve the pressure drop when specifying geometric data. Discuss how this method also preserves the volume so that the liquid inventory is accurately predicted.*

RESPONSE TO QUESTION 4:

The special technique is applied to nodes that are not simple pipes, so that the pressure drop due to wall friction within the node is correctly represented. Since the wall friction pressure drop is calculated from FLOWA, FLOWL, and DIAMV, the total volume, specified by V does not affect the pressure drop calculation. The actual value of the fluid volume is independently specified.

5. *Specify the gap conductance assumed for transient analysis. If values vary for different transients, discuss how they are determined.*

RESPONSE TO QUESTION 5:

The gap conductance assumed for transient analysis is determined from an approved fuel performance code. The gap conductance is developed to provide a consistent thermal resistance through the fuel pellet gap and clad to establish fuel average temperatures in RETRAN for calculating Doppler feedback during the transient. For the RETRAN plant response analyses, the gap conductance is selected to provide conservative Doppler reactivity feedback for events where Doppler reactivity plays an important role.

6. *Explain and justify the upper head circulation path modeling and demonstrate that predicted flows are realistic and/or conservative on a transient-by-transient.*

RESPONSE TO QUESTION 6:

The upper head bypass flow in the RETRAN model is conservatively low based on NSSS vendor data. The RETRAN upper head volume is, therefore, assumed to be relatively stagnant reducing its availability to attenuate the rate of heatup and cooldown during Chapter 15 type events. For Seabrook Station the Reactor Vessel (RV) internals have flow passages which by design allow cold coolant from the inlet nozzles to bypass the core and pass directly to the upper head. Therefore, initial coolant temperature in the upper head is equal to T_{cold}. The upper head modeling has an insignificant impact on Chapter 15 transients except for some primary side depressurization events, e.g., MSLB. Refer to Appendix A of these RAI responses for additional discussion.

7. *YAEC should provide FSAR benchmark analysis for the full spectrum of transients before its methodology can be reviewed for generic application. Comparison should be made for all 5 categories of transients with focus upon three categories (loss of heat sink, overcooling, reactivity anomalies) including loss of normal feedwater, loss of AC power and feedline break for Category 1, locked rotor and loss of flow for Category 2, rod ejection, rod withdrawal at different powers for Category 3. All such comparison must be thoroughly discussed and differences explained and justified.*

RESPONSE TO QUESTION 7:

The YAEC RETRAN analysis methodology described in YAEC-1856P has been applied to Seabrook Station and results are presented in YAEC-1871 which has been submitted to the NRC. In Appendix A to these responses to RAIs, comparisons of the YAEC RETRAN analysis methodology results to the Seabrook Station UFSAR results are presented. For some cases, results from YAEC-1871 are compared directly to the UFSAR. Differences are explained and justified. For other cases, YAEC has performed additional RETRAN analyses not shown in YAEC-1871 to facilitate the comparison by eliminating input parameter differences inherent in the YAEC-1871 analyses versus the UFSAR.

8. *Provide tables containing YAEC's approach to performance of Chapter 15 non-LOCA transient analysis for each plant. One table should contain ranges of key plant parameters and the other should contain (1) SG nodalization, (2) core nodalization, (3) reactivity coefficients, (4) key plant parameters and their status (high/low or min/max), (5) transient assumptions and (6) operable and inoperable components.*

RESPONSE TO QUESTION 8:

This information is provided in Appendix B to these responses to RAIs on a transient by transient basis. The illustration provided in Appendix B is specifically for Seabrook Station. In general, the key plant parameters and their status would be similar for any PWR, e.g., Maine Yankee. Specific design features or lack thereof may yield minor differences plant to plant, e.g., automatic rod control.

TABLE 5.0-3

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
High Neutron Flux, P-8	50%	0.5
Overtemperature ΔT	Variable	6.0*
Overpower ΔT	Variable	6.0*
High pressurizer pressure	2425 psia	2.0
Low pressurizer pressure	1935 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip	70% nominal	1.5
Turbine Trip	Not applicable	1.0
Low-low steam generator level	0% of narrow range level span**	2.0
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	94% of narrow range level span	2.0
Safety Injection Actuation	1665 psia	30.0

* Total time delay (including RTD time response and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

** Zero percent of the narrow range level span is the limiting trip point based on the feedwater system pipe break analysis. All other analyses assume a trip point corresponding to 10 percent of narrow range span.

TABLE 9.1

ENGINEERED SAFETY FEATURES AND CODE SAFETY VALVE SETPOINTS

Engineered Safety Feature	Actuation Setpoint Assumed in the Safety Analyses	Time Delay (seconds)
Safety Injection	Pressurizer Pressure = 1665 psia OR Main Steam Pressure = 430 psia	30
Emergency Feedwater	SG level = 0%*	75
Main Steam Line Isolation	Main Steam Pressure = 430 psia	2

* Zero percent of the narrow range level span is the setpoint assumed in the feedwater system pipe break analysis. All other analyses assume a setpoint corresponding to 10 percent of narrow range span.

Safety Valve	Opening Pressure (psia)	Fully Open Pressure (psia)	Fully Closed Pressure (psia)
Pressurizer Safety Valves	2575	2652.3	2475
MSSV #1	1236	1273.1	1161
MSSV #2	1254.5	1292.1	1179.5
MSSV #3	1272.1	1310.3	1197.1
MSSV #4	1290.6	1329.3	1215.6
MSSV #5	1308.1	1347.3	1233.1

11. *Explain why RTN initial conditions were not matched with the data for natural circulation test for Seabrook. Justify the statement that the difference in hot leg temperatures between 100 and 300 seconds while the cold leg temperature were matched with an offset in the initial conditions was caused by the modeling of steam dump control system. Discuss the source(s)of the hot leg temperatures. Does YAEC have similar analysis for other plants? In order to assure that the steam dump control system modeling is the source of mismatch (as YAEC postulates) in some benchmark analyses, provide the description of the simple steam dump model.*

RESPONSE TO QUESTION 11:

Our primary focus and concern in the benchmark of the Seabrook natural circulation test was the final conditions reached following the tripping of reactor coolant pumps (RCPs). The matching of final conditions is a strong indication that the hydraulic modeling of the plant is an accurate representation of specific plant conditions for transients where power to the RCPs is lost. The model used in the test benchmark was initialized based on data (primary to secondary) at power. The test were run at zero power. The difference of approximately 2 degrees in initial average temperature between the RETRAN simulation and the data is attributed to this difference.

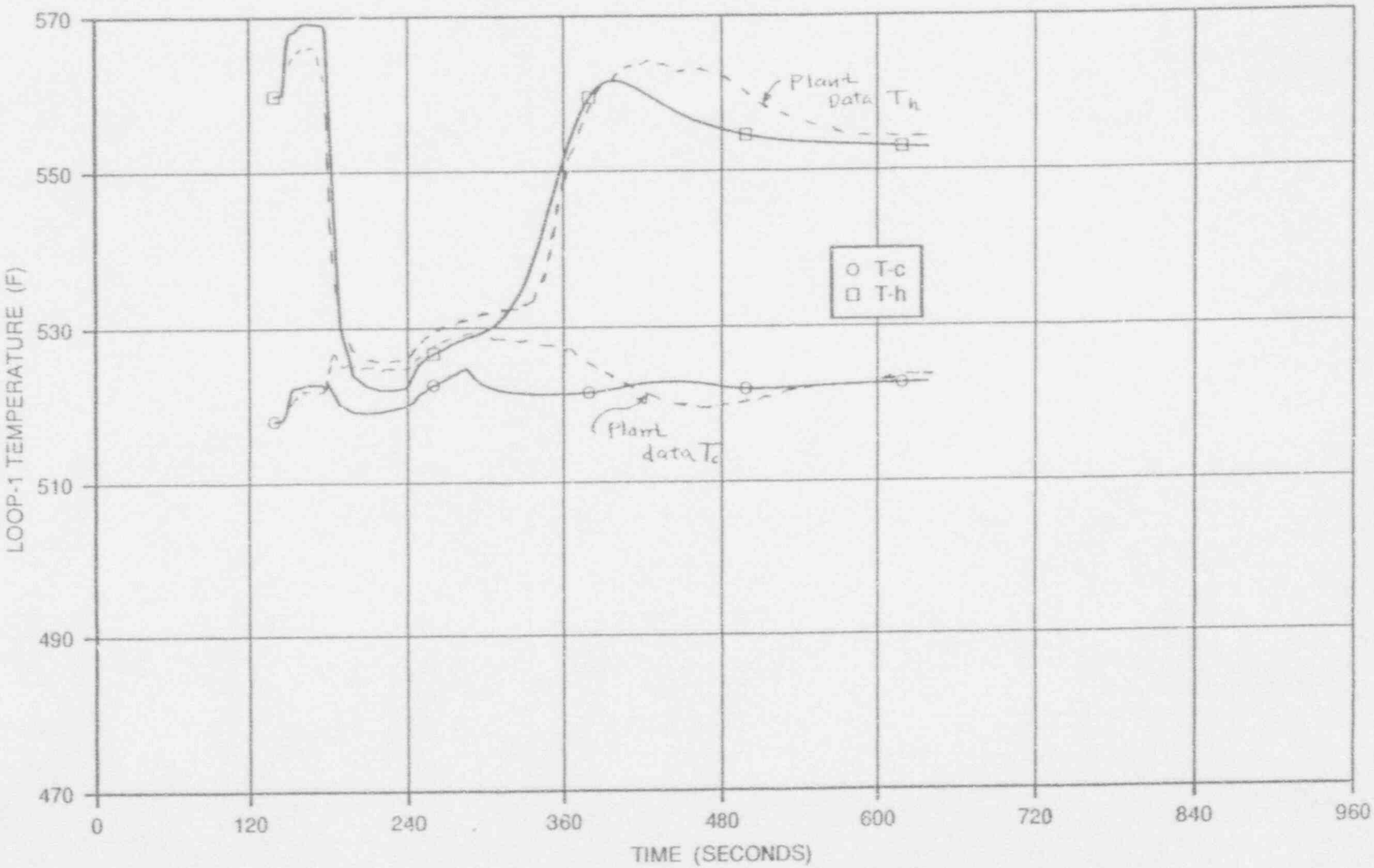
Your observation that the difference in hot leg response could not be attributed to the steam dump controller because the cold leg response agreed is correct. YAEC was misled in our postulation because of our focus on the final conditions and because of our understanding that the plant had some difficulty with the steam dump controller during the test. After evaluating the differences in hot leg responses further, and looking at similar analysis for Yankee Rowe, we have concluded that the differences are attributed to differences in RTD response times and the specific location of the hot leg RTD used in the Seabrook test.

The attached Figure shows the results of a similar analysis performed for Yankee Rowe. This analysis was of an actual event where one of two offsite power lines were lost and the plant was manually tripped by the operators. The operators also subsequently tripped the remaining RCPs placing the plant in natural circulation. The hot and cold leg responses, as shown in the attached Figure, are in good agreement with RETRAN predictions. The RTDs used to measure the responses at Yankee Rowe are mounted in thermo-wells in the loops and have relatively fast response times. At Seabrook the hot and cold leg RTDs consist of both narrow range and wide range. The narrow range RTDs, which feed into the reactor protection system, have a relatively fast response time similar to those at Yankee Rowe. The narrow range RTDs, at the time of the natural circulation test, were located in the loop bypass manifold and were unreliable during natural circulation conditions. The temperature measurements for the hot and cold leg responses for the natural circulation test were thus taken from the wide range channels. The wide range RTDs are located in the loops as opposed to the loop bypass manifold. Although the response times for the wide range channels have not been measured they are known to be slower than the response time of the narrow range channels (when the narrow range RTDs were located in the loop bypass manifold the response time was about 6 seconds with the RCPs operating). Under natural circulation conditions it is plausible for the wide range RTDs to have a response time on the order of 60 seconds. In addition, the location of the hot leg wide range RTD is closer to the steam generator than assumed in the RETRAN simulation accounting for part of the difference noted. This contributed approximately 40 seconds to the delay in the hot leg temperature response. Therefore, the difference noted in hot leg response in Figure 4.1.4 of YAEC-1856P is attributed in part to the slow response time of the wide range RTDs and in part to the assumed location of the hot leg RTD in the RETRAN simulation.

YR RETRAN SIMULATION

LOSS OF Z-126 HIGH LINE EVENT

(LOSS OF OFF-SITE POWER)



12. *What is the timing of the operator action during the reactor trip test? For the Maine Yankee reactor trip test benchmark analysis, the pressurizer pressure was predicted to be about 300 psi higher than the plant data by the end of simulation. Explain in depth the cause of this difference since the pressurizer level was well matched.*

RESPONSE TO QUESTION 12:

The reactor trip data used in the Maine Yankee benchmark was based on an event that occurred on April 29, 1991 involving a switchyard failure resulting in a turbine trip reactor trip sequence. There were no specific known operator actions taken in the time frame of interest. The higher pressurizer pressure predicted by RETRAN at the end of the simulation is attributed to the response of pressurizer heaters and interfacial heat transfer in the pressurizer. As indicated in Section 3.1.1 of YAEC-1856P YAEC assumes a conservatively low interfacial heat transfer coefficient for pressurizer insurge transients. Assuming a low heat transfer coefficient results in an overprediction of the pressure response. A conservatively low interfacial heat transfer coefficient was assumed in the reactor trip test comparison.

We believe that the sensitivity to interfacial heat transfer in this test was weak and not the major contribution to pressure differences noted. Based on sensitivity analysis the difference noted is primarily due to difference in assumptions regarding pressurizer heaters. For the RETRAN analysis, the pressurizer backup heaters remained on throughout the simulation. In the actual event, the backup heaters were de-energized when the pressurizer level dropped below 28%. This occurs within seconds of the reactor trip. Terminating the heaters on low level in the RETRAN model results in a consistent pressurizer pressure response.

13. *Provide a loss of load analysis for Maine Yankee. YAEC demonstrated that for Yankee Rowe, GEMINI predicted higher pressure than RETRAN at slightly earlier time. Explain the reasons for the divergent results after the pressure peaked.*

RESPONSE TO QUESTION 13:

The intent of the Yankee Rowe loss of load comparison presented in Figure 4.2.2 of YAEC-1856P was to provide an indirect comparison of YAEC's pressurizer modeling technique (treatment of interfacial heat transfer etc.) for overpressure transients against data. Figure 4.2.1 of YAEC 1856P presented results of a comparison of GEMINI predictions to data from Shippingport for a rapid insurge transient. GEMINI is YAEC's currently approved transient analysis model for these types of events. The comparison shows that the GEMINI prediction is conservative relative to the data. Comparison of the RETRAN response to GEMINI provides an indirect comparison of the RETRAN response to data. Peak pressures agree well but diverge following reactor trip. The divergence is attributed to differences in the post-trip heat input to the Reactor Coolant System (RCS) from the following sources:

1. Decay Heat - The decay heat in GEMINI is greater than that in RETRAN. RETRAN uses the 1973 ANS standard. The attached figure compares the GEMINI decay heat model to the 1971 ANS standard.

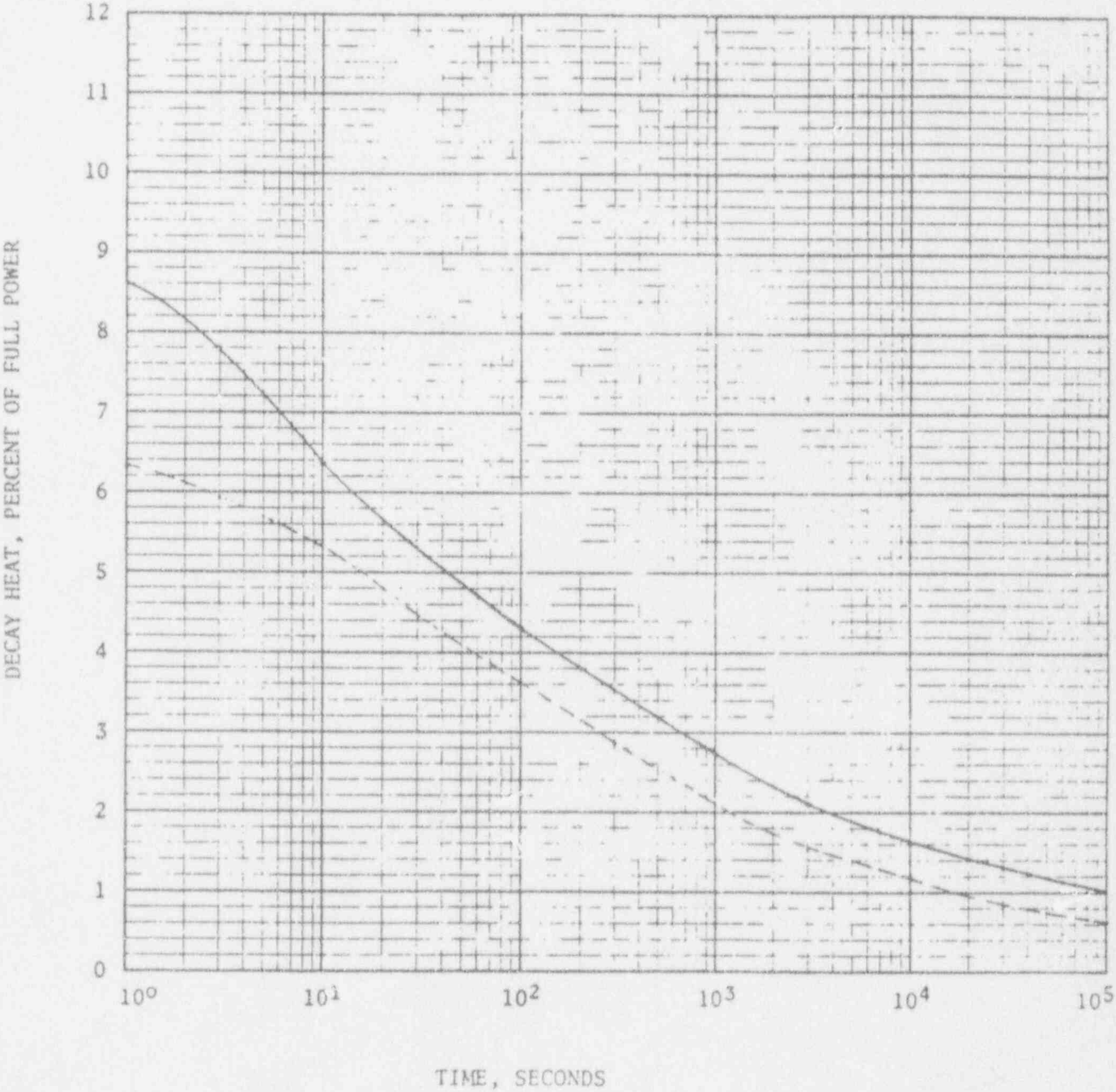
2. Delayed Fissions - The heat from delayed fissions following the trip is greater in GEMINI than in RETRAN due to the different delayed neutron fractions used.

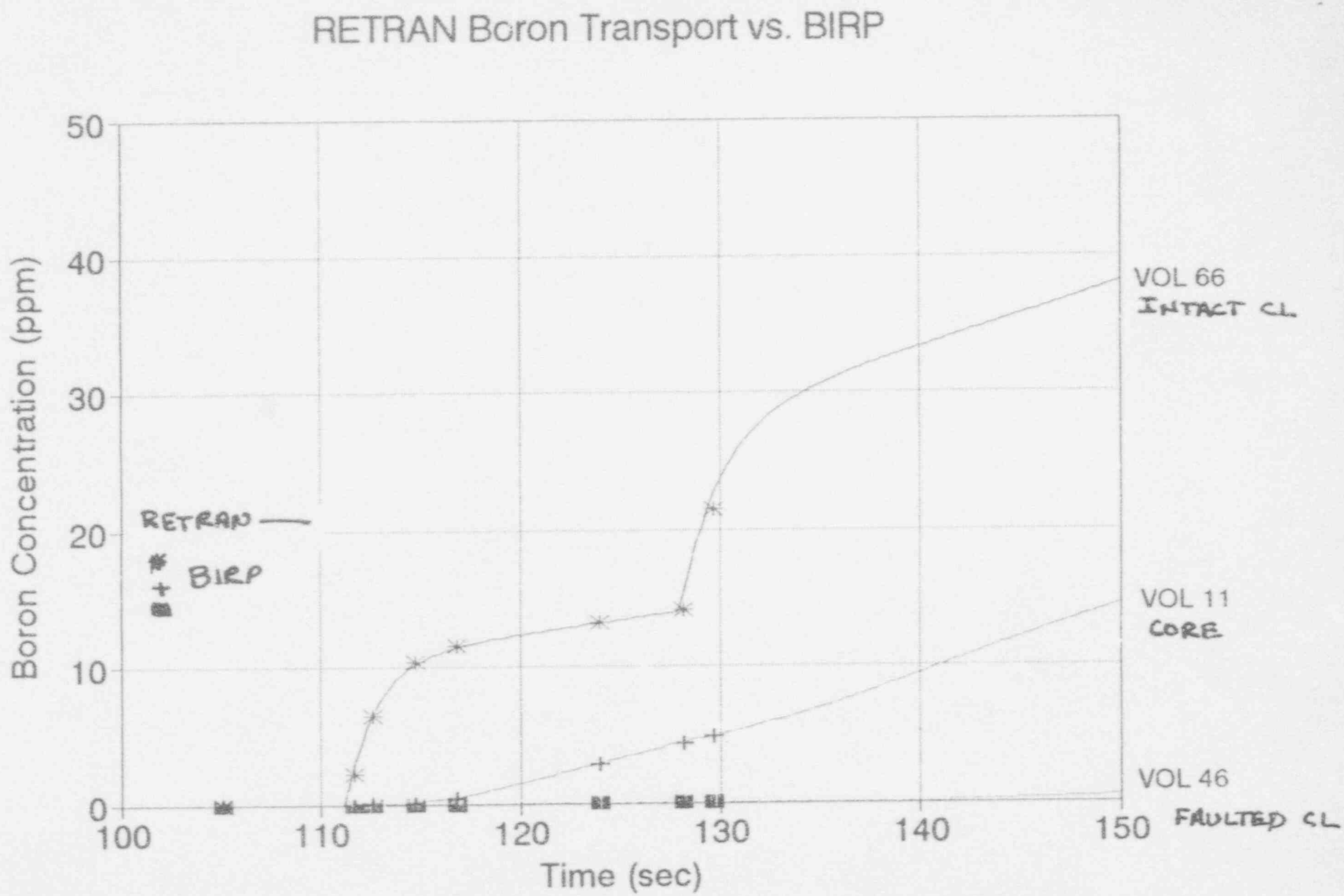
3. Stored Energy - The initial stored energy in the fuel used in GEMINI is conservatively larger than that in the RETRAN simulation.

GEMINI-II DECAY HEAT FUNCTION

______ GEMINI-II

------ ANS Standard





RETRAN Boron Transport vs. BIRP
Boron Concentration (ppm)
Time (sec)
50
40
30
20
10
0
100
110
120
130
140
150
RETRAN
BIRP
VOL 66
INTACT CL
VOL 11
CORE
VOL 46
FAULTED CL

APPENDIX A
COMPARISONS TO SEABROOK STATION UFSAR

INTRODUCTION

In this appendix, comparisons of the YAEC RETRAN analysis methodology results to the Seabrook Station UFSAR results are presented. For some cases, results from YAEC-1871 are compared directly to the UFSAR. Differences are explained and justified. For other cases, YAEC has performed additional RETRAN analyses not shown in YAEC-1871 to facilitate the comparison by eliminating input parameter differences inherent in the YAEC-1871 analyses versus the UFSAR. Table A-0.1 summarizes initial conditions and uncertainties used in the RETRAN and UFSAR analyses.

The transient analyses presented in the Seabrook Station UFSAR and those performed by Yankee and documented in YAEC-1871 used different scram reactivity curves. Figure A-0.1 shows the normalized reactivity worth versus rod position scram data used in both sets of analyses. The figure shows that the Yankee scram data is more conservative than the UFSAR data. The Yankee data was chosen to bound more adverse power distributions which may result from transients when initiated from within Yankee's Wide Band Operation Strategy. Figure A-0.2 shows the comparison of the normalized worth versus time data. The UFSAR analyses assumed a scram time (from release to dashpot) of 3.3 seconds for all transients except the LOFA which assumed a 2.2 second time. The Yankee analyses assumed a scram time of 2.4 seconds for all transients and is consistent with our proposed Technical Specifications.

APPENDIX A (Continued)

TABLE A-0.1

INITIAL CONDITIONS AND UNCERTAINTIES USED IN THE RETRAN AND UFSAR ANALYSES

Parameters	RETRAN	UFSAR
Core thermal power:		
- Rated, MWT	3411	3411
- Engineered Safety Features design rating, MWT	Not used	3565
- Uncertainty, %	2.0[(4)]	2.0
Nominal RCS loop average temperature @ 100% RTP, °F	588.5	588.5
- Uncertainty, °F	5.8[(1)]	5.8
Pressurizer Pressure, psia	2250	2250
- Uncertainty, psi	50[(4)]	30
Main Steam Pressure @ 100% RTP, psia	1000	1000
Main Steam Flow Rate @ 100% RTP, 10^6 lb_m/hr	15.14	15.14
RCS Thermal Design Flow, gpm	375,100[(2)]	382,800
RCS Minimum Measured Flow[(3)], gpm	392,800[(2)]	Not used

Notes:

(1) 1.8°F is assumed to be a bias on T_{avg} to allow for the effects of SG tube fouling and flow streaming. This bias is applied in all RETRAN analyses. The remaining 4°F is a measurement uncertainty and is not applied when RETRAN is used for the analysis of DNB related events.

(2) These values of RCS flow include a 2% penalty to allow for SG tube plugging.

(3) The minimum measured flow is used in the analysis of DNB related events.

(4) As discussed in the Introduction of Appendix B, these uncertainties are not applied when RETRAN is used for the analysis of DNB related events.

FIGURE A-0.1
SEABROOK STATION
SCRAM CURVE COMPARISON
Normalized Worth Versus Position

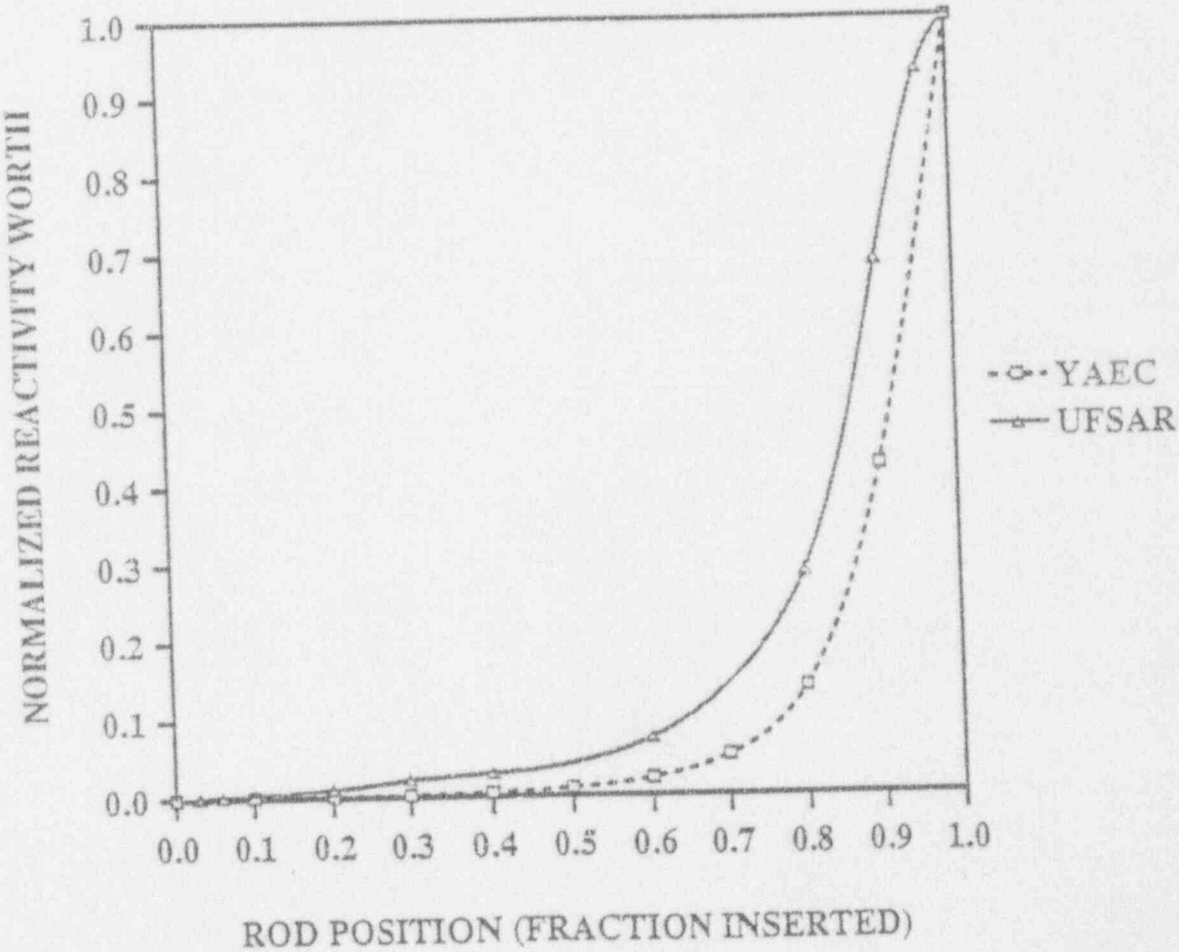
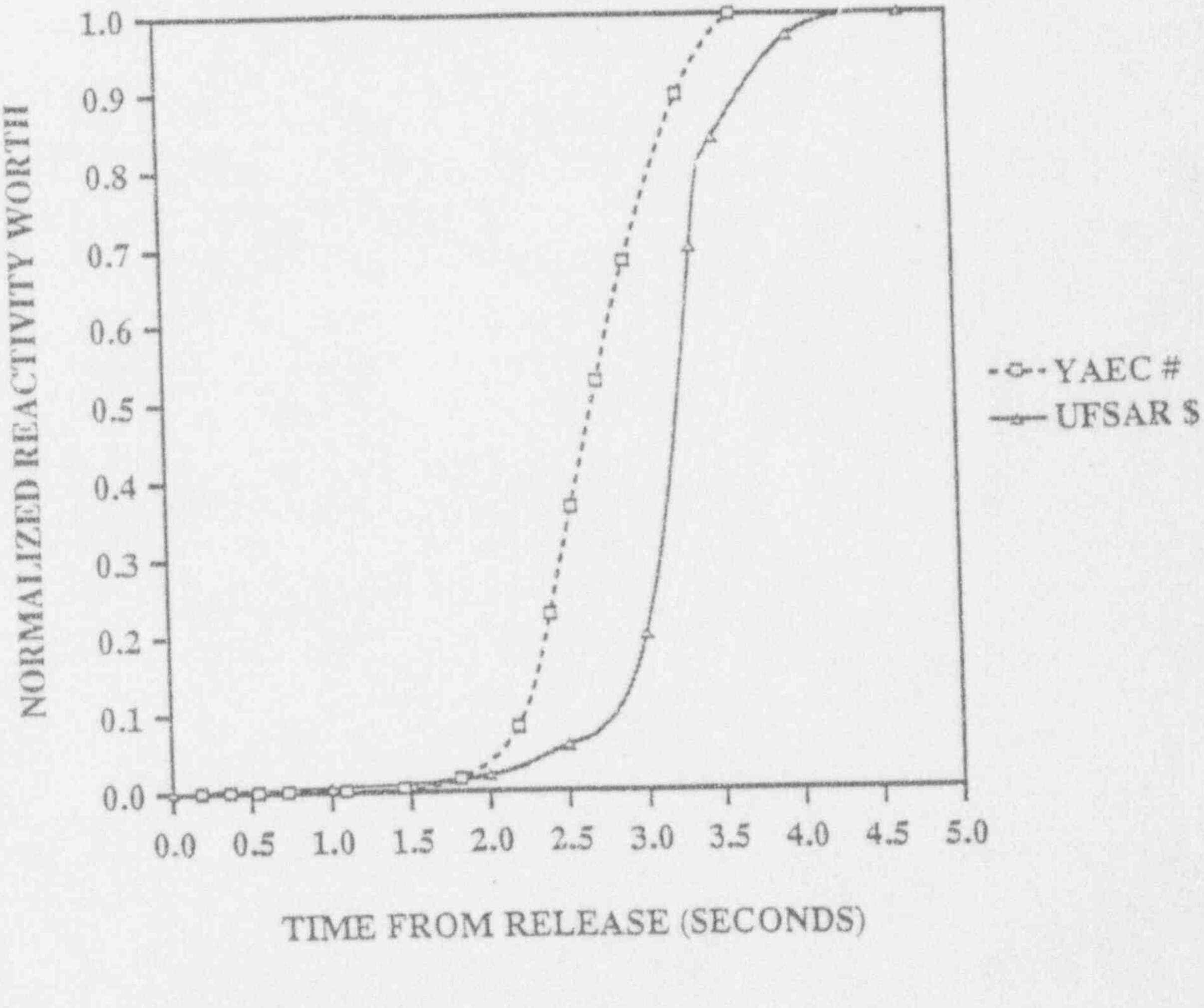


FIGURE A-0.2
SEABROOK STATION
SCRAM CURVE COMPARISON
Normalized Worth Versus Time



= 2.4 Seconds to Dashpot
$ = 3.3 Seconds to Dashpot

1. MAIN STEAM LINE BREAK (MSLB)

YAEC's approved methodology for MSLB analysis is documented in YAEC-1752A. A comparison of RETRAN results with the Seabrook Station UFSAR results was provided in that report. The comparison is also reproduced here in Figures A-1.1 through A-1.5. The sequence of events is shown in Table A-1. It should be noted that the RETRAN analysis documented in YAEC-1752A has differences in input parameters and assumptions from the analysis documented in YAEC-1871. For example, in YAEC-1871 the Chapter 15 RETRAN analysis of Seabrook Station is performed using a statistical combination of uncertainties while a deterministic treatment was used in YAEC-1752A.

Core Power

The RETRAN power response is more conservative than the UFSAR. The RETRAN results are based on the approved methodology using STAR. The reactivity weighting coefficients developed in YAEC-1752A and used in the benchmarks to the UFSAR were derived to produce a conservative power response relative to STAR. The overprediction of power as compared to the UFSAR is consistent with the conservative derivation of the reactivity weighting.

Figure A-1.5 shows the various components of reactivity. As shown in the figure, the total reactivity drops to zero when boron reaches the core, causing the core power to decrease. Dryout of the faulted SG occurs beyond this point.

As noted above, the MSLB response from the YAEC-1752A UFSAR comparison presented here differs from the response presented in YAEC-1871. In the YAEC-1752A UFSAR comparison the steam piping resistance from the intact SGs to the break was neglected to be consistent with UFSAR assumptions. In the YAEC-1871 analysis the steam piping is explicitly modelled consistent with YAEC's approved MSLB methods. This changes the relative timing of when boron reaches the core and SG dryout.

RV Inlet Temperatures

The RETRAN predictions for both the faulted and intact reactor vessel inlet temperatures are in good agreement with the UFSAR. Small differences are attributed to differences in the break flow.

RCS Pressure

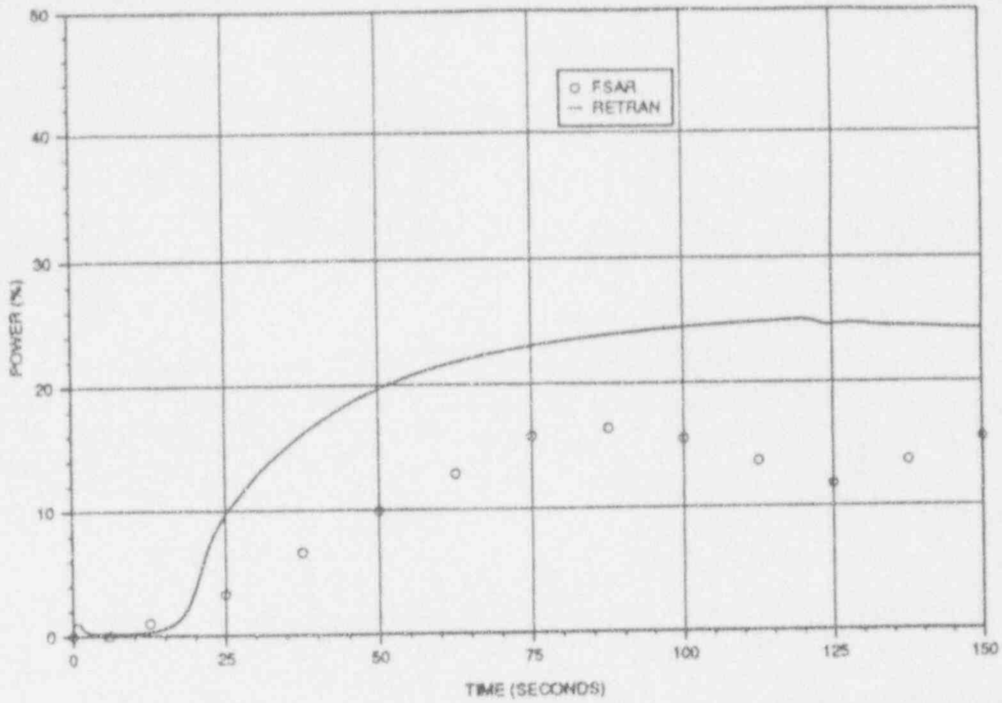
The RETRAN model includes a non-equilibrium volume in the upper head region. Non-equilibrium effects in the upper head will result in the upper head behaving as a second pressurizer yielding higher RCS pressure as RCS inventory recovers by the action of safety injection. The pressure response from the UFSAR appears to follow the saturation curve corresponding to the intact loop temperatures. It appears that the UFSAR analysis neglected the non-equilibrium effects in the upper head. The overall effect of higher RCS pressure is to cause a reduced injection of boron and a higher core power.

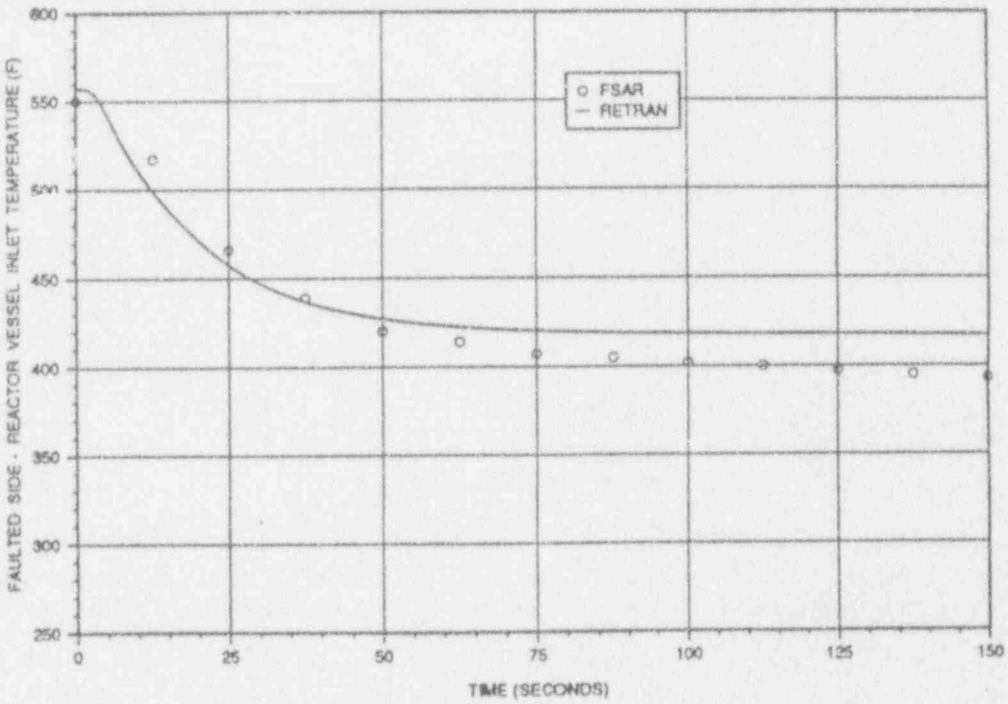
Break Flow

The RETRAN and UFSAR results for break flow show good agreement. In the RETRAN analysis, the choked flow model based on Moody was used to compute the break flow rate through the SG outlet nozzle flow restrictor (1.4 ft^2). In addition, the break flow was assumed to be dry steam.

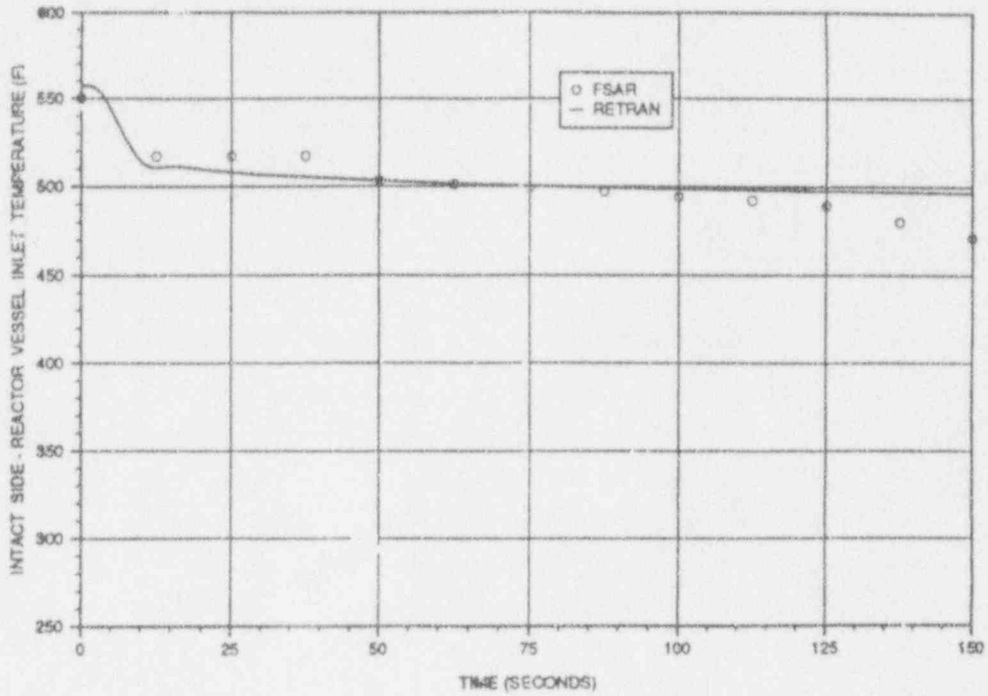
TABLE A-1
SEQUENCE OF EVENTS FOR THE
MAIN STEAM LINE BREAK (MSLB)

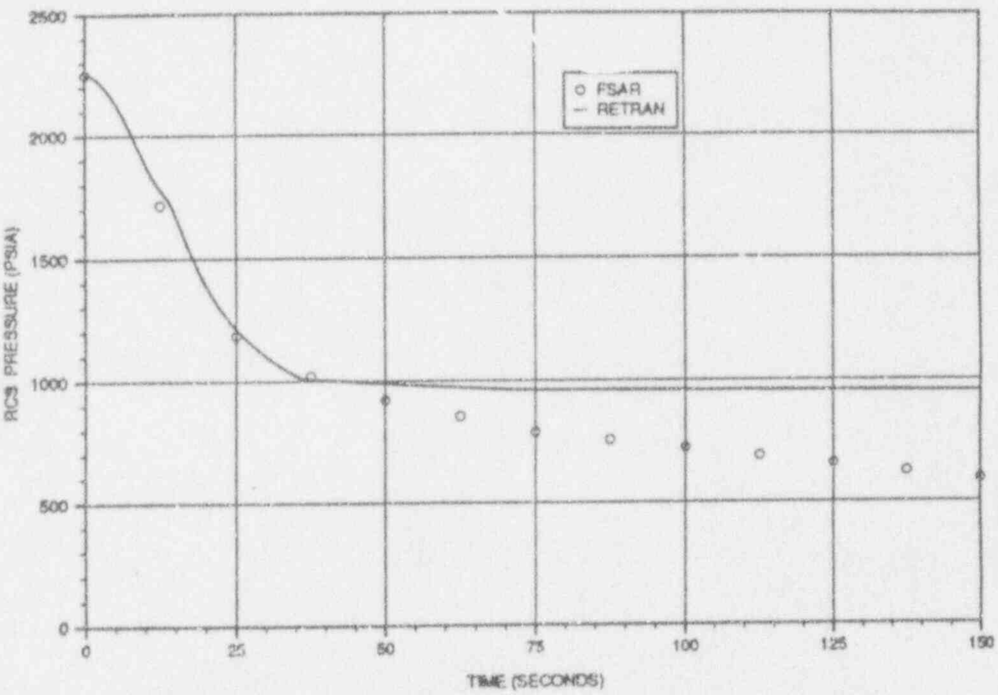
Event	Time (Seconds) RETRAN	UFSAR
Steam line ruptures	0.0	0.0
Pressurizer empty	13.5	13.6
Criticality attained	14.	12.4
Boron reaches core	~ 120.	~ 125.



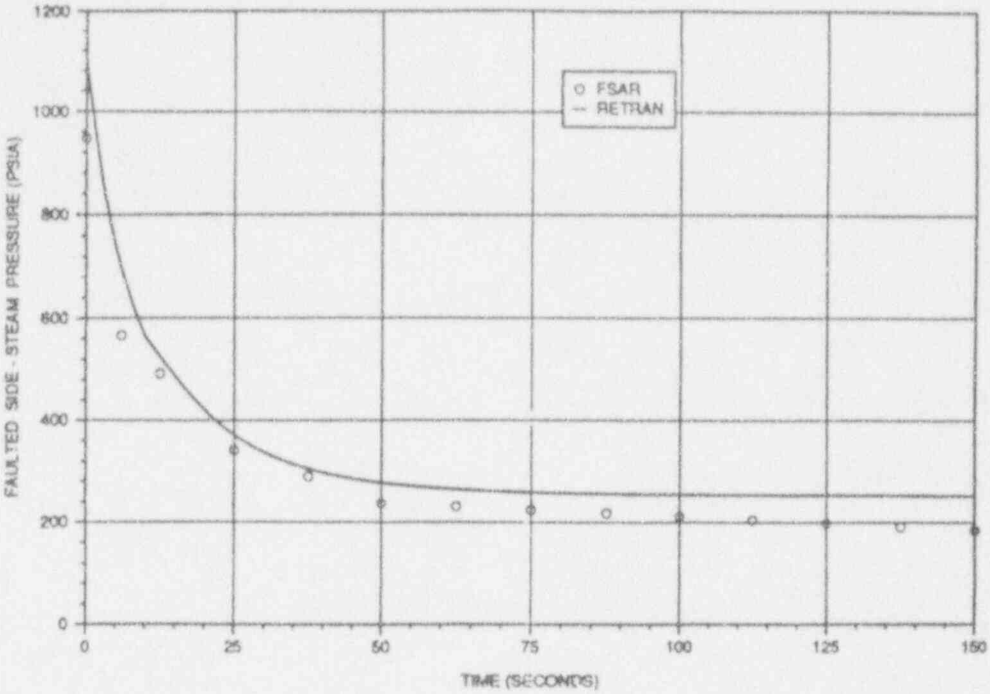


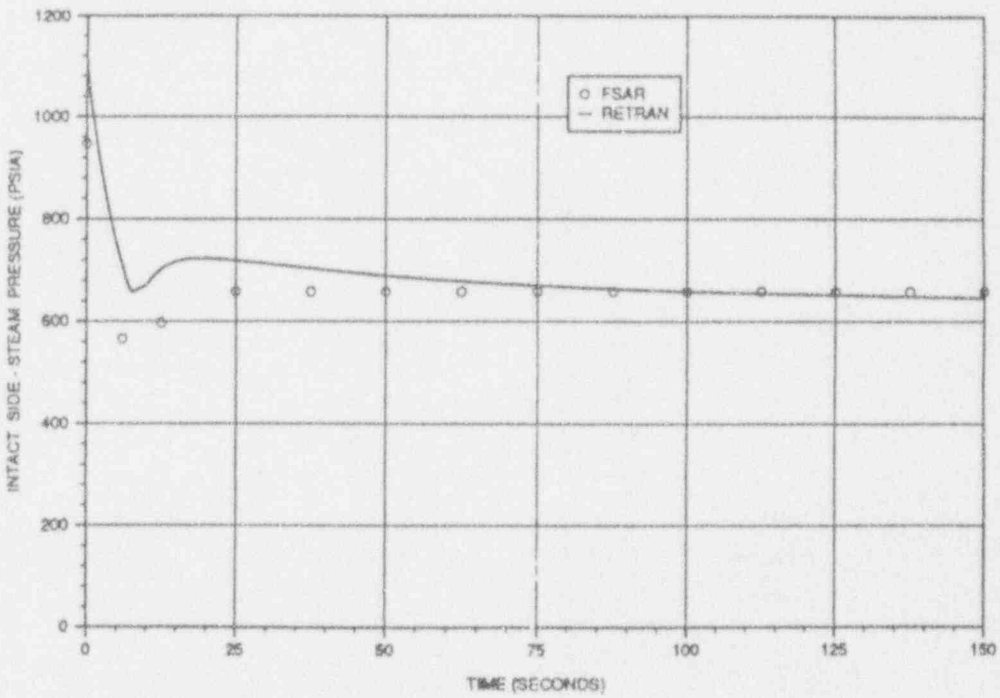
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Steam Line Rupture (Offsite Power Available) FIGURE A-1.1



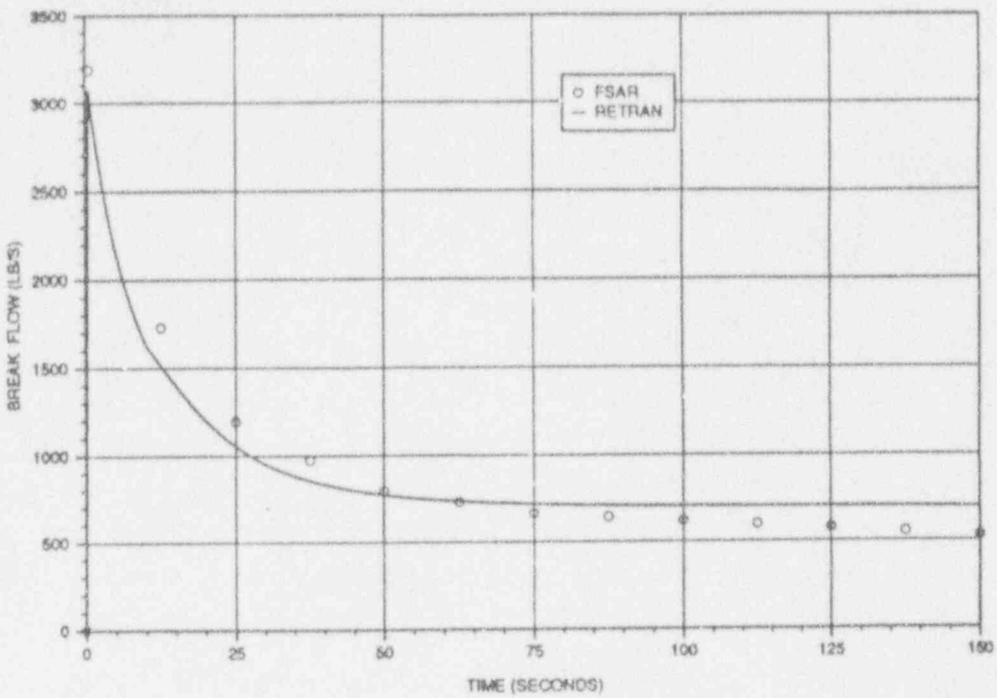


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Steam Line Rupture (Offsite Power Available) FIGURE A-1.2

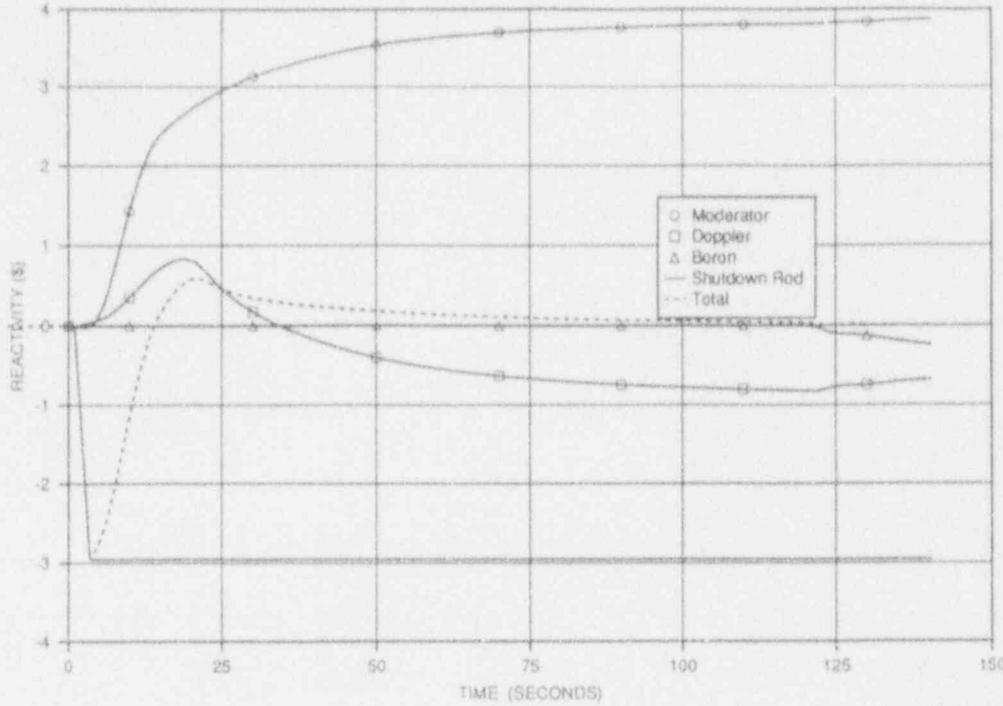


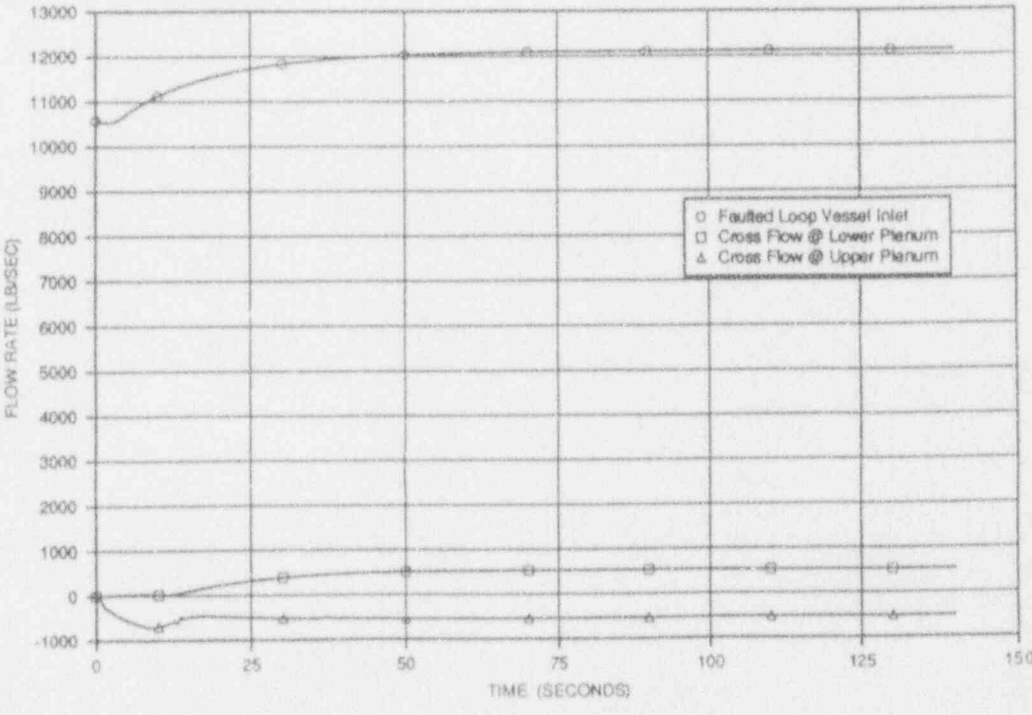


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Steam Line Rupture (Offsite Power Available) FIGURE A-1.3



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Steam Line Rupture (Offsite Power Available) FIGURE A-1.4





SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Steam Line Rupture (Offsite Power Available) FIGURE A-1.5

2. TURBINE TRIP (WITHOUT PRESSURIZER PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK)

A comparison of the RETRAN results versus the Seabrook Station UFSAR results is shown in Figures A-2.1 and A-2.2. The sequence of events is shown in Table A-2. The case presented (without pressurizer pressure control, minimum reactivity feedback) results in the greatest primary and secondary system pressures of the four cases presented in the UFSAR. The RETRAN results are taken from a UFSAR benchmark that was performed to verify the Seabrook Station RETRAN model (relative to the prediction of peak pressures) prior to evaluating an increase in the pressurizer and main steam safety valve setpoint tolerances from ±1% to ±3%.

Normalized Core Power

The core powers match very well. RETRAN results in a slightly later (about 1 second) reactor trip on high pressurizer pressure that is attributed to small differences in the pressurizer modelling between the RETRAN and UFSAR analyses and the initial pressure assumed.

Pressurizer Pressure

The peak pressurizer pressures match almost exactly between RETRAN and the UFSAR. Due to the slightly later trip in the RETRAN analysis, the time of the peak pressure is delayed about 1 second from the UFSAR analysis. The decrease in the RETRAN pressure below the UFSAR pressure is due to the blowdown that is included in the RETRAN MSSV model. The MSSV blowdown results in a lower SG pressure, which causes a lower RCS temperature and a lower pressurizer pressure.

Pressurizer Water Volume

The peak pressurizer water volume in the RETRAN analysis is greater than that in the UFSAR analysis due to the slight delay in the reactor trip and possibly to small differences in the pressurizer modelling. The decrease in the RETRAN water volume below that in the UFSAR is due to the blowdown that is included in the RETRAN MSSV model. The MSSV blowdown results in a lower SG pressure, which causes a lower RCS temperature and a lower pressurizer water volume.

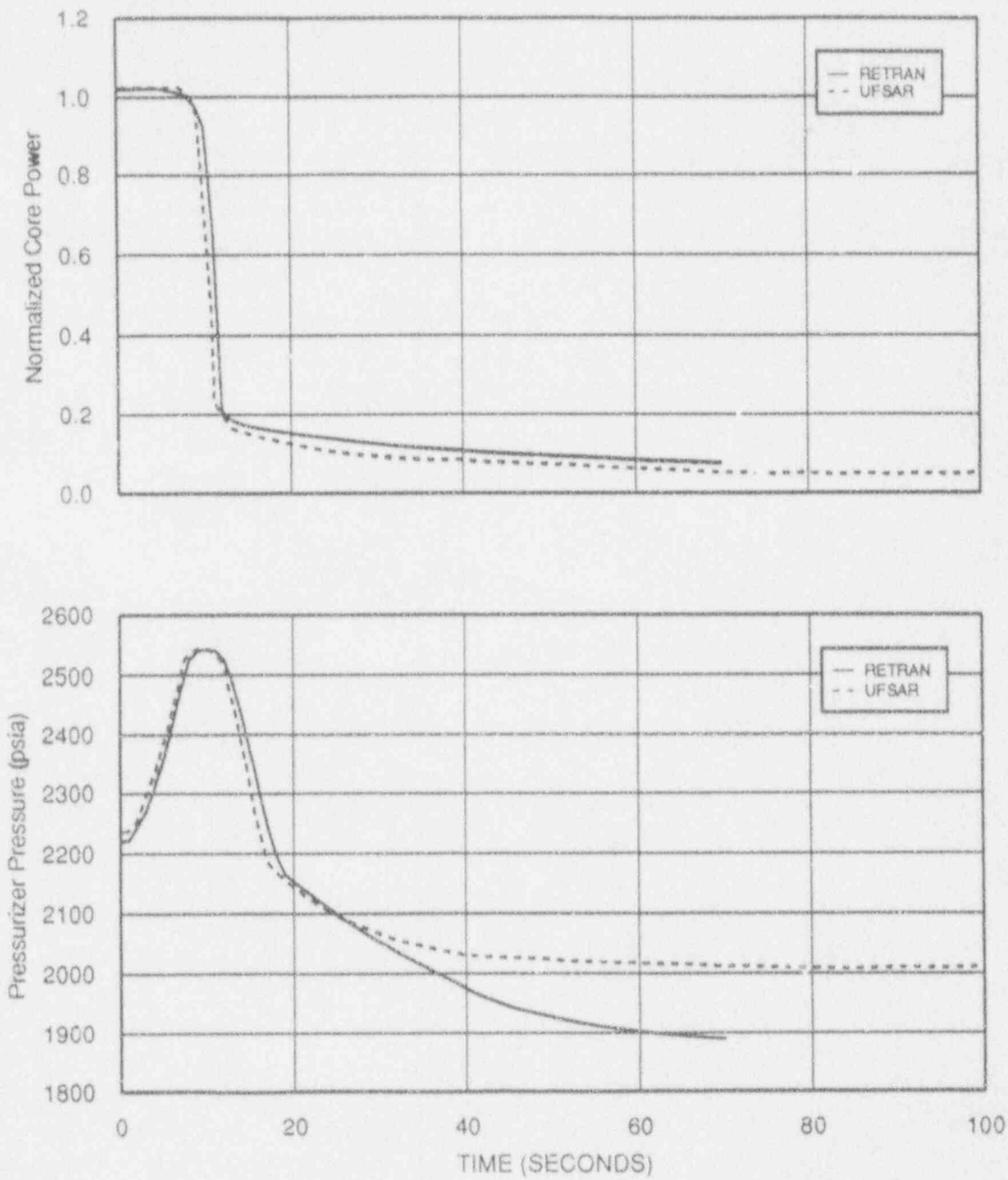
Core Average Temperature

Accounting for the slight delay in the RETRAN analysis reactor trip, the core average temperature matches well with the UFSAR analysis. The decrease in the RETRAN temperature below the UFSAR temperature is due to the blowdown that is included in the RETRAN MSSV model. The MSSV blowdown results in a lower SG pressure which causes a lower RCS temperature.

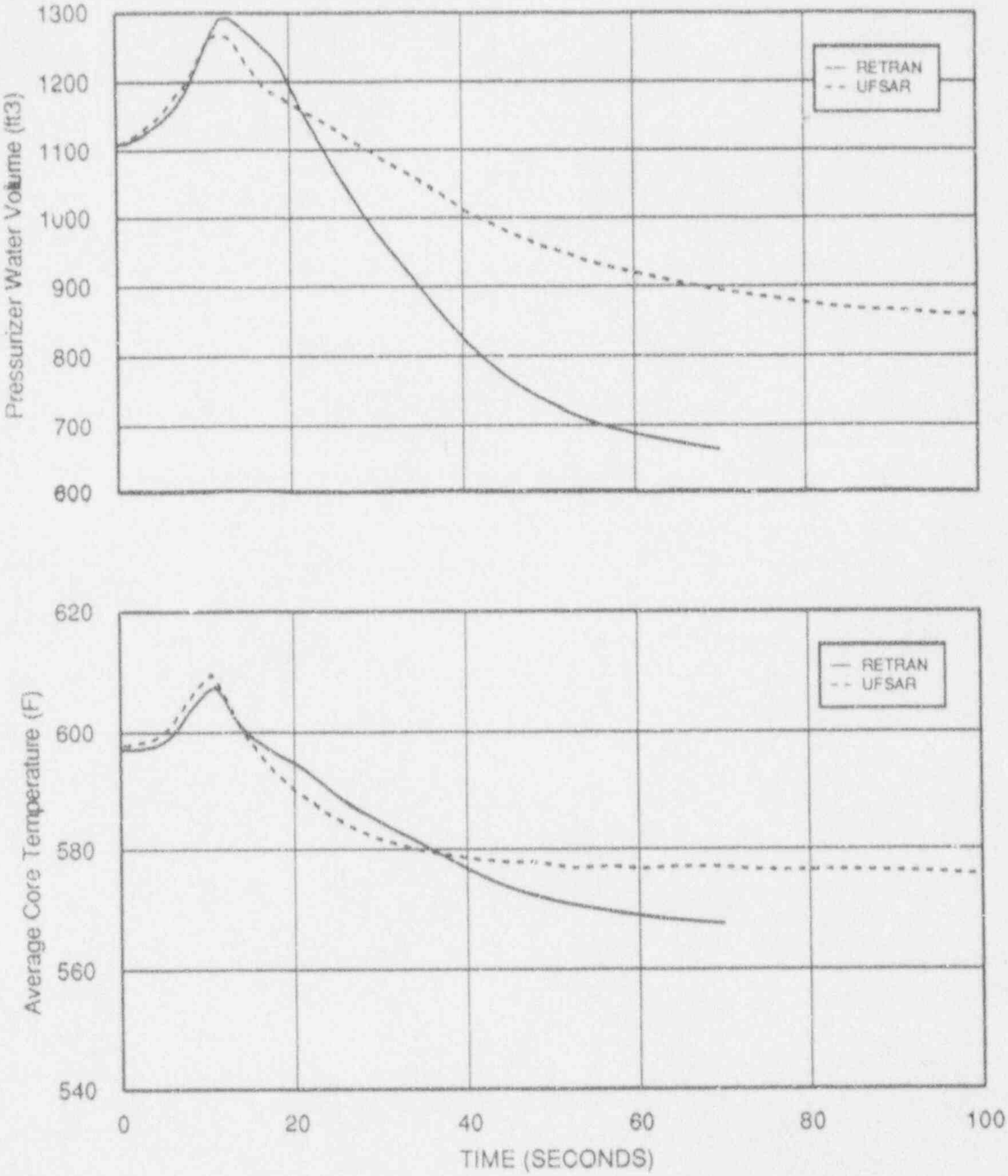
APPENDIX A (Continued)

TABLE A-2
SEQUENCE OF EVENTS FOR THE TURBINE TRIP

Event	Time (Seconds) RETRAN	UFSAR
Turbine trip, loss of main feedwater flow	0.0	0.0
Initiation of steam release from steam generator safety valves	7.5	5.5
High pressurizer pressure reactor trip point reached	6.2	5.1
Rods begin to drop	8.2	7.1
Peak pressurizer pressure occurs	10.0	9.0



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Turbine Trip Without Pressurizer Pressure Control Minimum Reactivity Feedback FIGURE A-2.1



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Turbine Trip Without Pressurizer Pressure Control Minimum Reactivity Feedback FIGURE A-2.2

Normalized Core Flow

Except for the earlier trip of the RCPs, the RETRAN core flow coastdown agrees very well with the UFSAR. The flow coastdown begins earlier due to the earlier reactor trip in the RETRAN analysis since the RCPs are assumed to trip at the time of the reactor trip.

Core Average Temperature

The peak temperatures match very well. The final temperature is lower in the RETRAN analysis due to the blowdown included in the RETRAN MSSV model. The MSSV blowdown results in a lower SG pressure which causes a lower RCS temperature.

SG Pressure

The initial SG pressure is lower in the RETRAN analysis. The lower initial pressure conservatively delays the opening of the MSSVs and the corresponding RCS heat removal, minimizing the DNBR and maximizing the peak RCS pressure. The peak SG pressure is greater in the RETRAN analysis due to the greater setpoint tolerance assumed (3% in the RETRAN analysis vs. 1% in the UFSAR analysis). The SG pressure in the RETRAN analysis falls below that in the UFSAR analysis due to the MSSV blowdown included in the RETRAN MSSV model.

SG Water Volume

The initial decrease in the SG water volume matches very well. The RETRAN analysis has a greater minimum volume and a faster increase in volume for several reasons:

- The initial RETRAN power level is lower than assumed in the UFSAR (3411 vs. 3565 MWT), with a correspondingly lower decay heat.
- The reactor trip occurs approximately 5 seconds sooner in the RETRAN analysis.
- The RETRAN analysis uses the EFW pump head-flow curve adjusted to give the minimum flow required by Technical Specifications, while the UFSAR analysis uses a constant 650 GPM, the minimum EFW flow at 1221 psig.
- The RETRAN analysis uses an EFW temperature of 100 °F, as opposed to the 120 °F EFW temperature assumed in the UFSAR analysis.

SG Mixture Level

Figure A-3.5 shows the RETRAN mixture level in the tube bundle region of the steam generators. The dotted line in the figure represents the top of the SG U-tubes. The SG tubes are uncovered at approximately 65 seconds. Direct comparison to the UFSAR mixture level is not possible since the UFSAR does not include a plot of SG level.

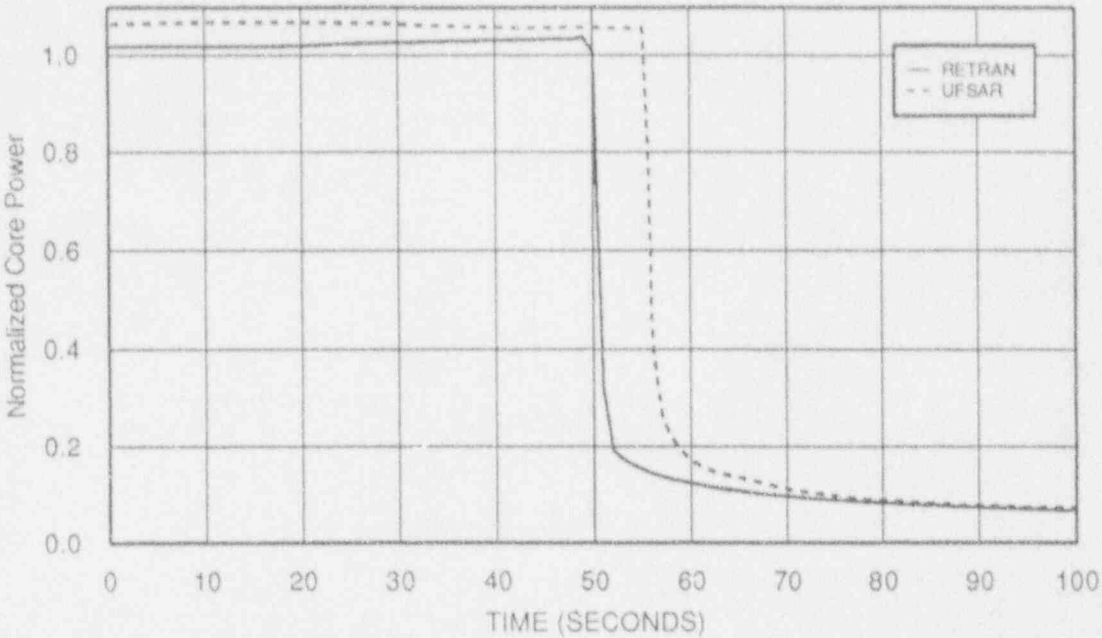
Total EFW Flow

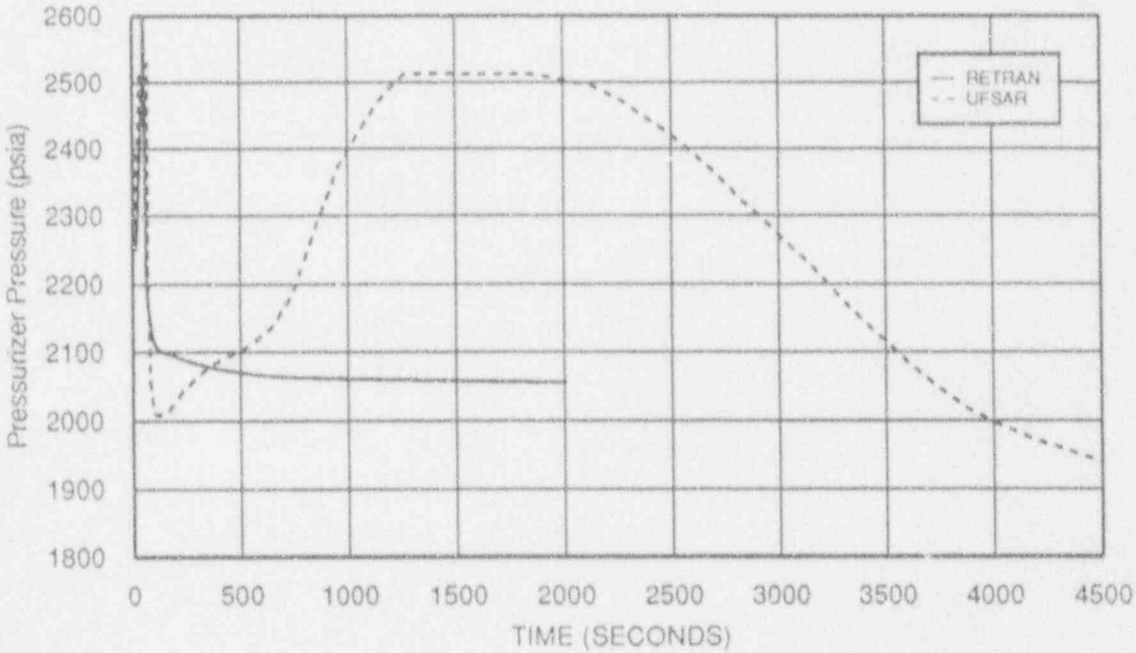
The RETRAN analysis has a greater Emergency Feedwater (EFW) flow to the SGs since the flow varies with SG pressure, while the UFSAR analysis assumes a constant flow of 650 GPM (the minimum flow at 1221 psig). EFW flow in the RETRAN analysis is based on the actual head-flow curve of the EFW pumps, which has been adjusted to give a flow of 650 GPM at 1221 psig. The difference in integrated EFW flow is approximately 300 ft^3 over the first 2000 seconds of the transient.

APPENDIX A (Continued)

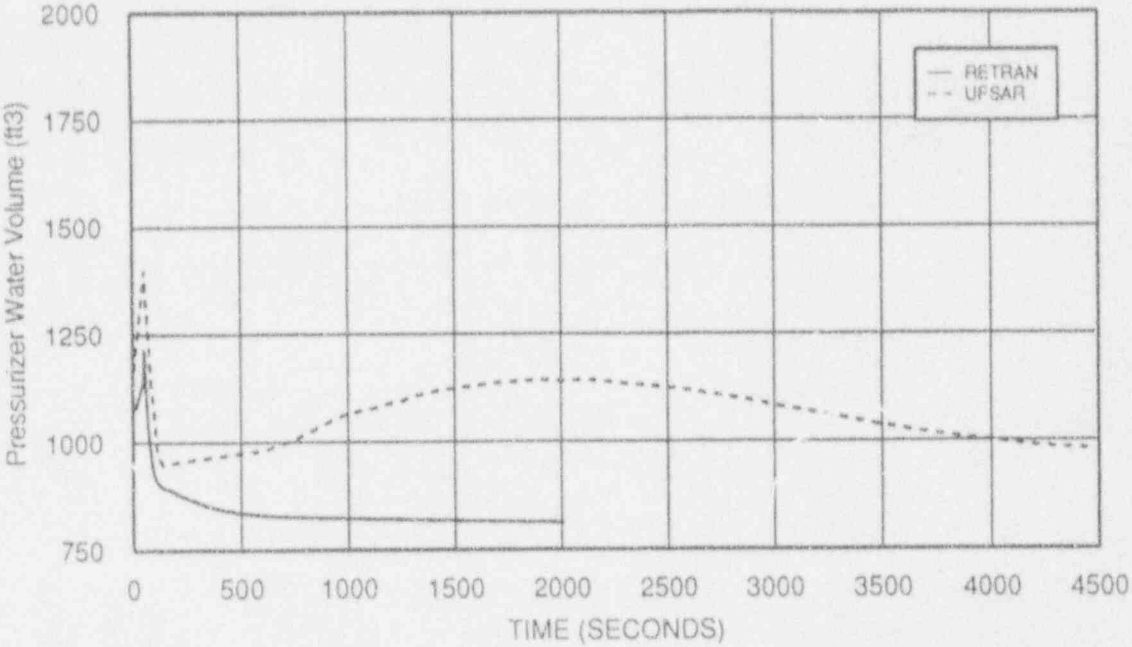
TABLE A-3
SEQUENCE OF EVENTS FOR THE
LOSS OF NON-EMERGENCY AC POWER

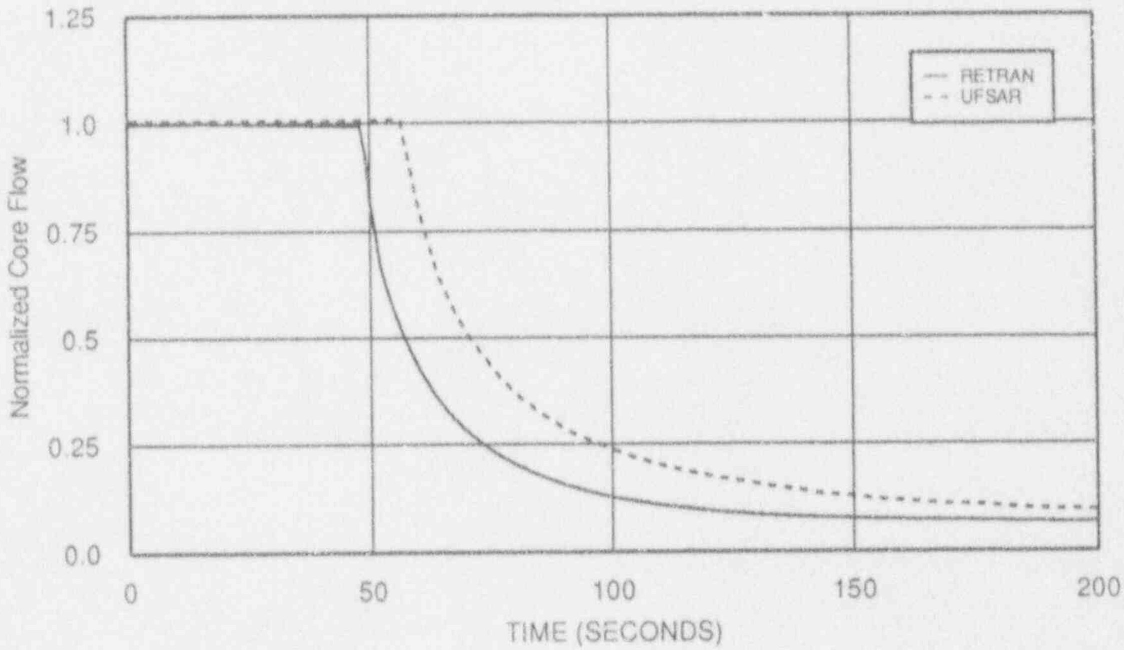
	Time (Seconds)	
Event	RETRAN	UFSAR
AC power is lost	0.0	0.0
Main feedwater flow stops	0.0	0.0
Low-low steam generator water level trip	36.3	41.1
Rods begin to drop	38.3	43.1
Reactor coolant pumps begin to coast down	38.3	43.1
Peak water level in pressurizer occurs	43	46.5
Four steam generators begin to receive emergency feed from one motor-driven emergency feed-water pump	120	112.0
Minimum steam generator inventory	490	1892



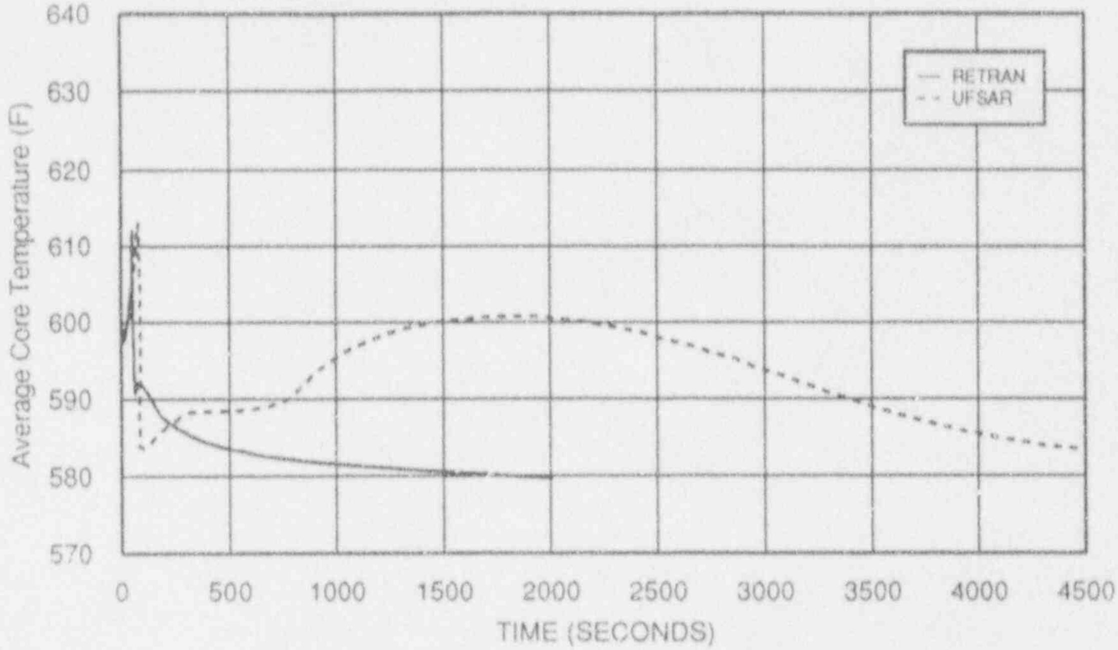


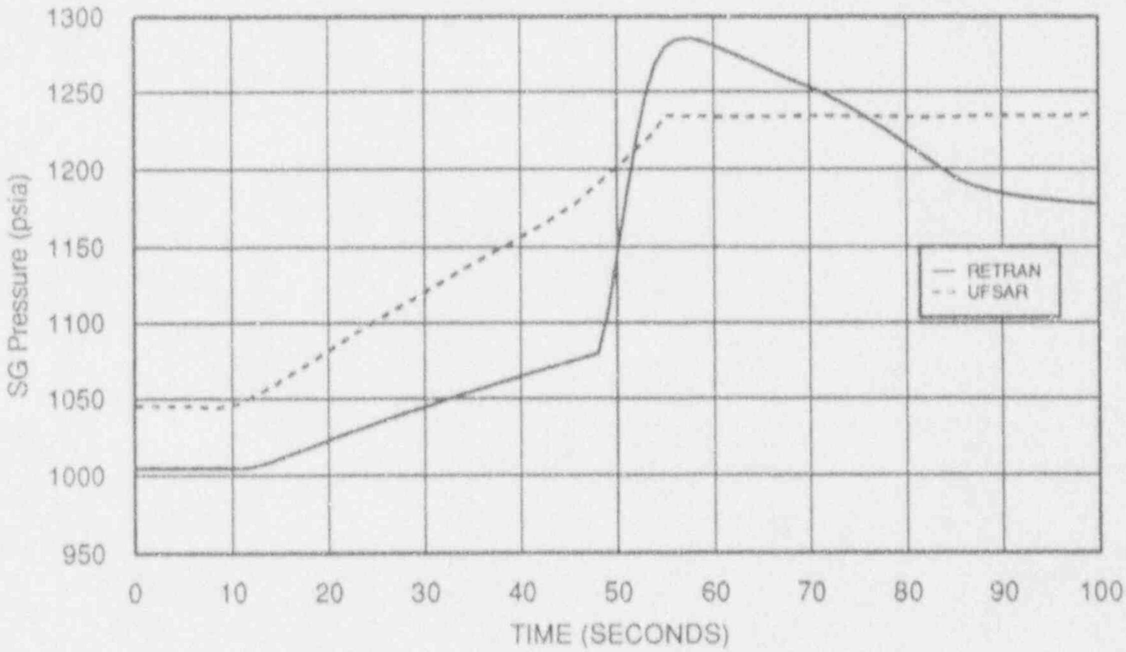
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Non-Emergency AC Power FIGURE A-3.1



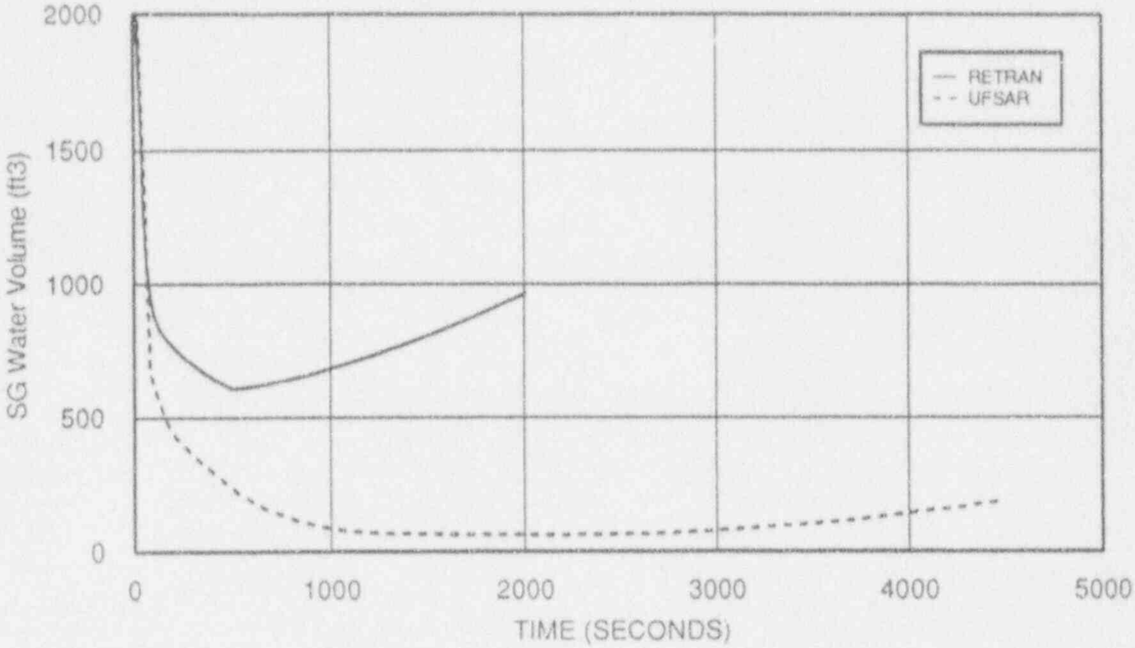


SEABROOK STATIO[illegible] COMPARISON OF R[illegible] vs. UFSAR	Loss of Non-Emergency AC Power FIGURE A-3.2

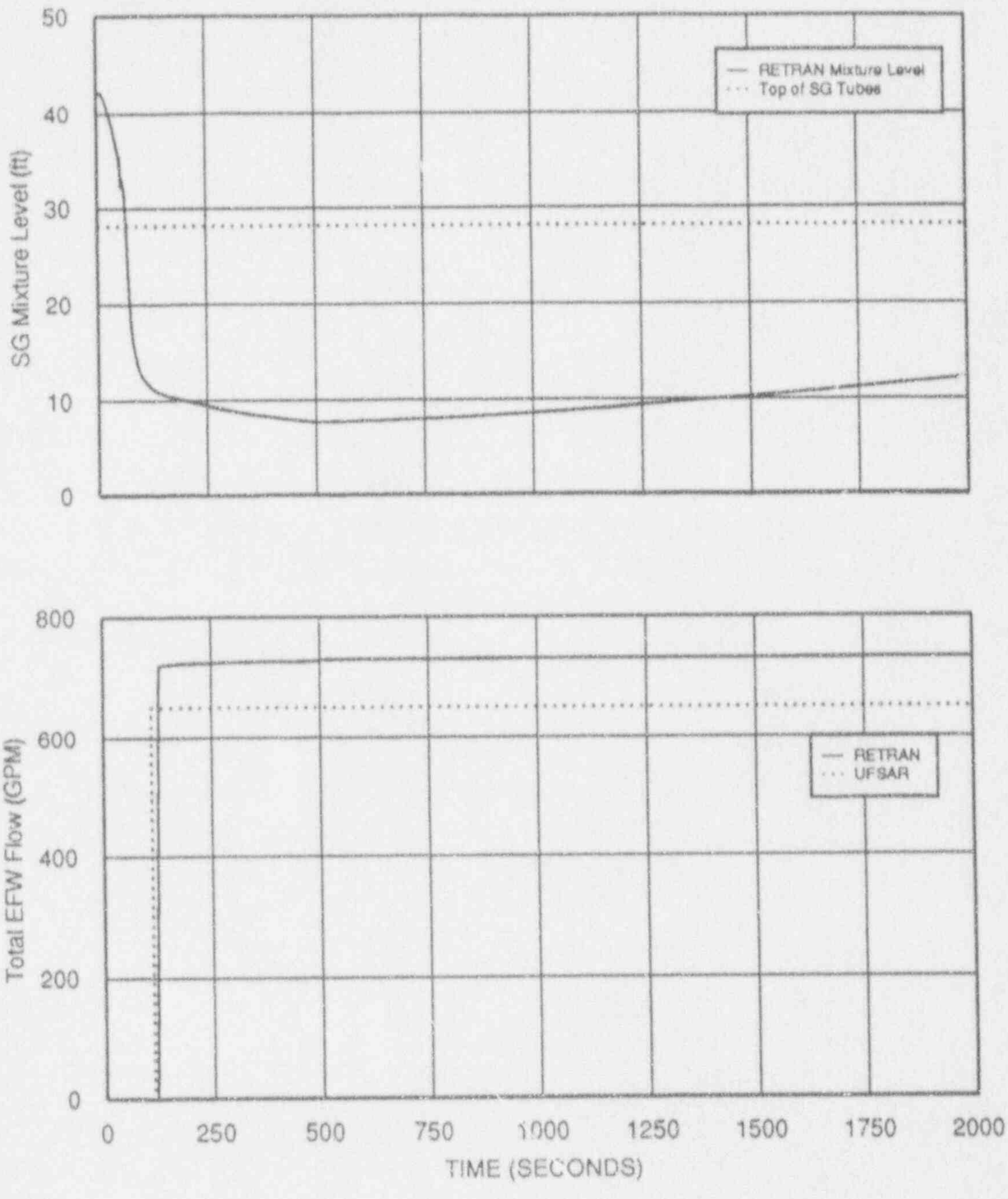




SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Non-Emergency AC Power FIGURE A-3.3

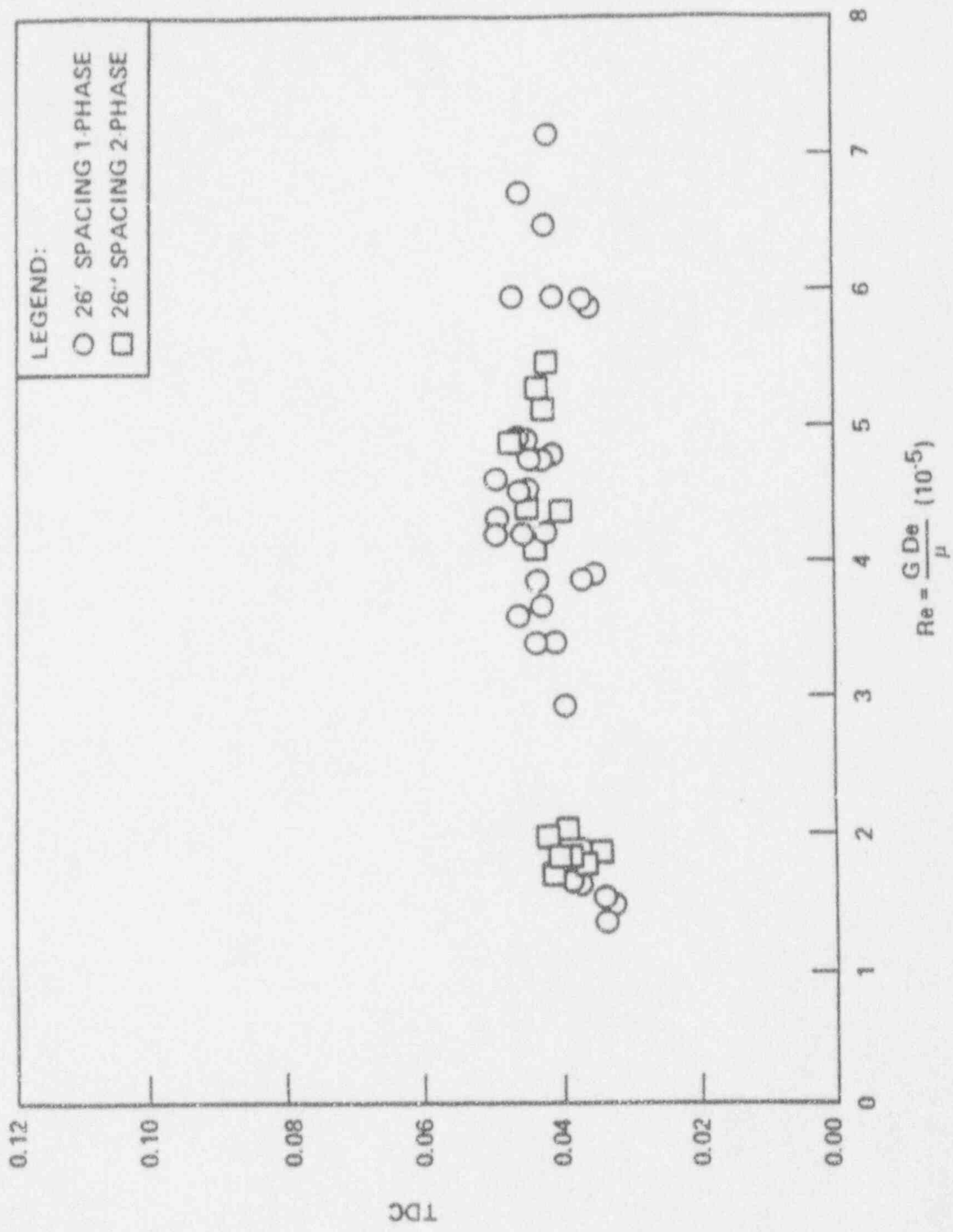


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Non-Emergency AC Power FIGURE A-3.4



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Non-Emergency AC Power FIGURE A-3.5

FIGURE 3 (Continued)



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	TDC vs. Reynolds Number for 26 Inch Grid Spacing	
		FIGURE 4.4-4

14 *Justify the use of the gap conductance indicated on Table B.1.3*

RESPONSE TO QUESTION 14:

The gap conductance values used with our approved and VIPRE based methodologies are determined by the approved FROSSTEY-2 fuel performance code. The FROSSTEY-2 code provides gap conductance values for a range of KW/FT and fuel burnup. The gap conductance chosen for a particular transient is selected to bound the FROSSTEY-2 results over the range of interest. In general, when the transient criteria is DNBR a high gap conductance is selected to maximize the surface heat flux. When the criteria is fuel temperature a low gap conductance is selected to retain the heat generated in the fuel pellet.

15 *Explain and justify how the RTDP is determined for Seabrook with the old methodology and how the proposed new limit differs. Is the methodology described in the topical report the same as that used to generate the current value and only the code is different?*

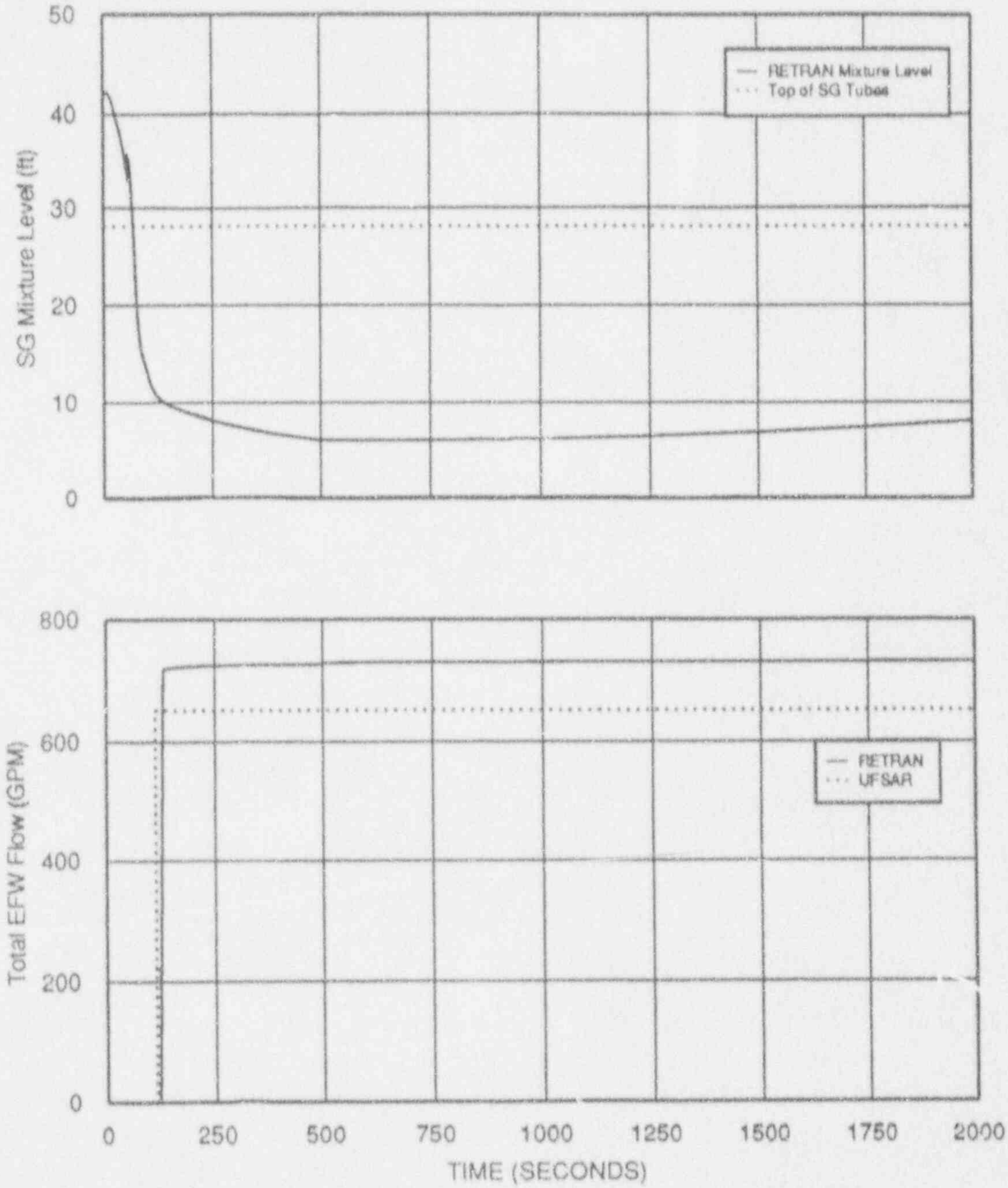
RESPONSE TO QUESTION 15:

The RTDP methodology has not been implemented for Seabrook's safety analysis prior to the YAEC-1849P submittal. The current DNBR methodology uses a deterministic treatment of uncertainties with the W-3 correlation. The RTDP methodology implementation for Seabrook as described in YAEC-1849P uses WRB-1. RTDP is applied consistent with the approved method used by Westinghouse.

16 *Explain why the uncertainty associated with the effective flow fraction is assumed to be uniformly distributed rather than in accordance with a normal distribution function.*

RESPONSE TO QUESTION 16:

Because the amount of measured data available for Seabrook was insufficient to characterize the distribution, it was difficult to justify the use of a normal distribution. As a result, the conservative assumption of a uniform distribution was made. This assumption is consistent with the older (more conservative) ITDP approach described in WCAP-8567 (ITDP).



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Feedwater FIGURE A-4.4

Core Average Temperature

The peak temperatures match very well. The final temperature is lower in the RETRAN analysis due to the blowdown included in the RETRAN MSSV model. The MSSV blowdown results in a lower SG pressure which causes a lower RCS temperature.

SG Pressure

The initial SG pressure is lower in the RETRAN analysis. The lower initial pressure conservatively delays the opening of the MSSVs and the corresponding RCS heat removal, minimizing the DNBR and maximizing the peak RCS pressure. The peak SG pressure is greater in the RETRAN analysis due to the greater setpoint tolerance assumed (3% in the RETRAN analysis vs. 1% in the UFSAR analysis). The SG pressure in the RETRAN analysis falls below that in the UFSAR analysis due to the MSSV blowdown included in the RETRAN MSSV model.

SG Water Volume

The initial decrease in the SG water volume matches very well. The RETRAN analysis has a greater minimum volume and a faster increase in volume for several reasons:

- The initial RETRAN power level is lower than assumed in the UFSAR (3411 vs. 3565 MWT), with a correspondingly lower decay heat.
- The reactor trip occurs approximately 5 seconds sooner in the RETRAN analysis.
- The RETRAN analysis uses the EFW pump head-flow curve adjusted to give the minimum flow required by Technical Specifications, while the UFSAR analysis uses a constant 650 GPM, the minimum EFW flow at 1221 psig.
- The RETRAN analysis uses an EFW temperature of 100 °F, as opposed to the 120 °F EFW temperature assumed in the UFSAR analysis.

SG Mixture Level

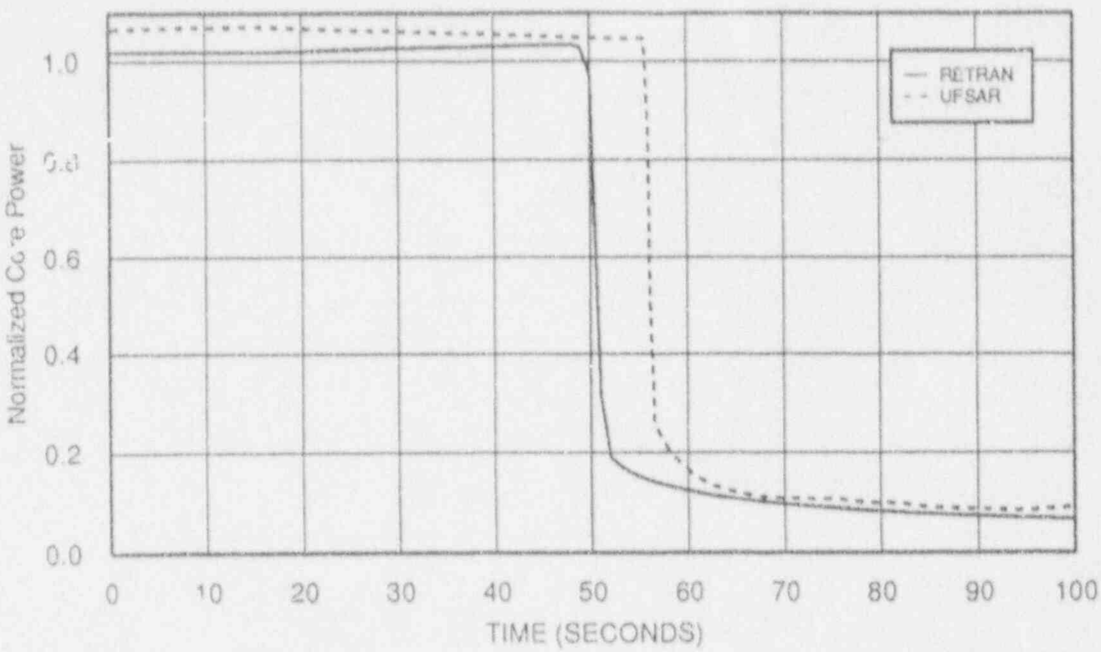
Figure A-4.4 shows the RETRAN mixture level in the tube bundle region of the steam generators. The dotted line in the figure represents the top of the SG U-tubes. The SG tubes are uncovered at approximately 65 seconds. Direct comparison to the UFSAR mixture level is not possible since the UFSAR does not include a plot of SG level.

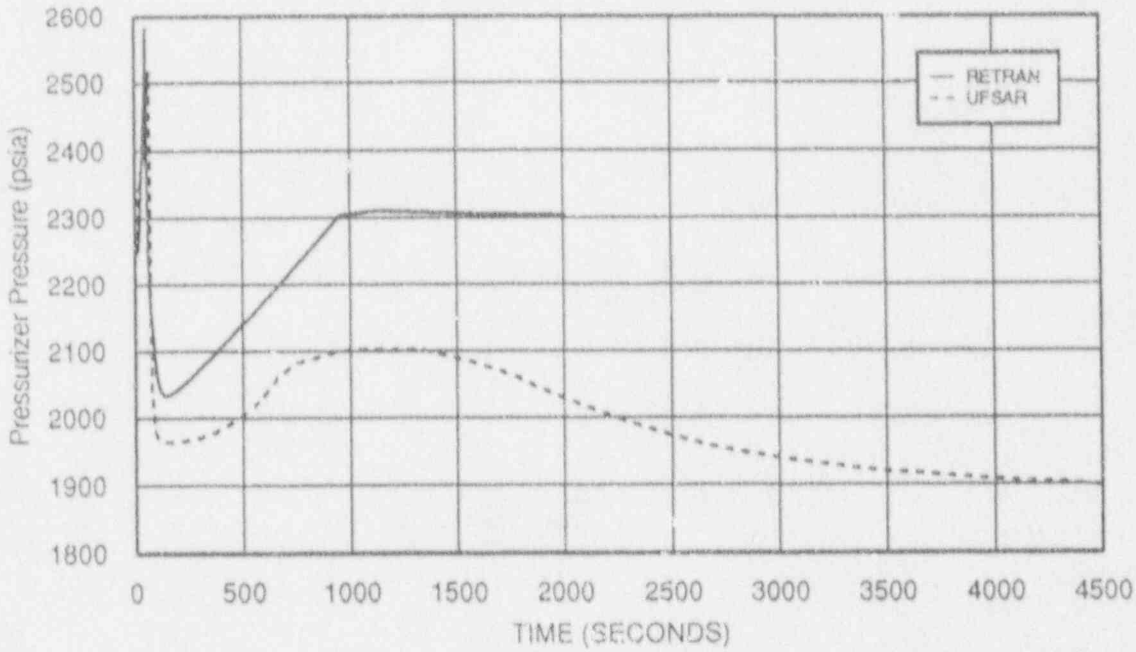
Total EFW Flow

The RETRAN analysis has a greater Emergency Feedwater (EFW) flow to the SGs since the flow varies with SG pressure, while the UFSAR analysis assumes a constant flow of 650 GPM (the minimum flow at 1221 psig). EFW flow in the RETRAN analysis is based on the actual head-flow curve of the EFW pumps, which has been adjusted to give a flow of 650 GPM at 1221 psig. The difference in integrated EFW flow is approximately 300 ft^3 over the first 2000 seconds of the transient.

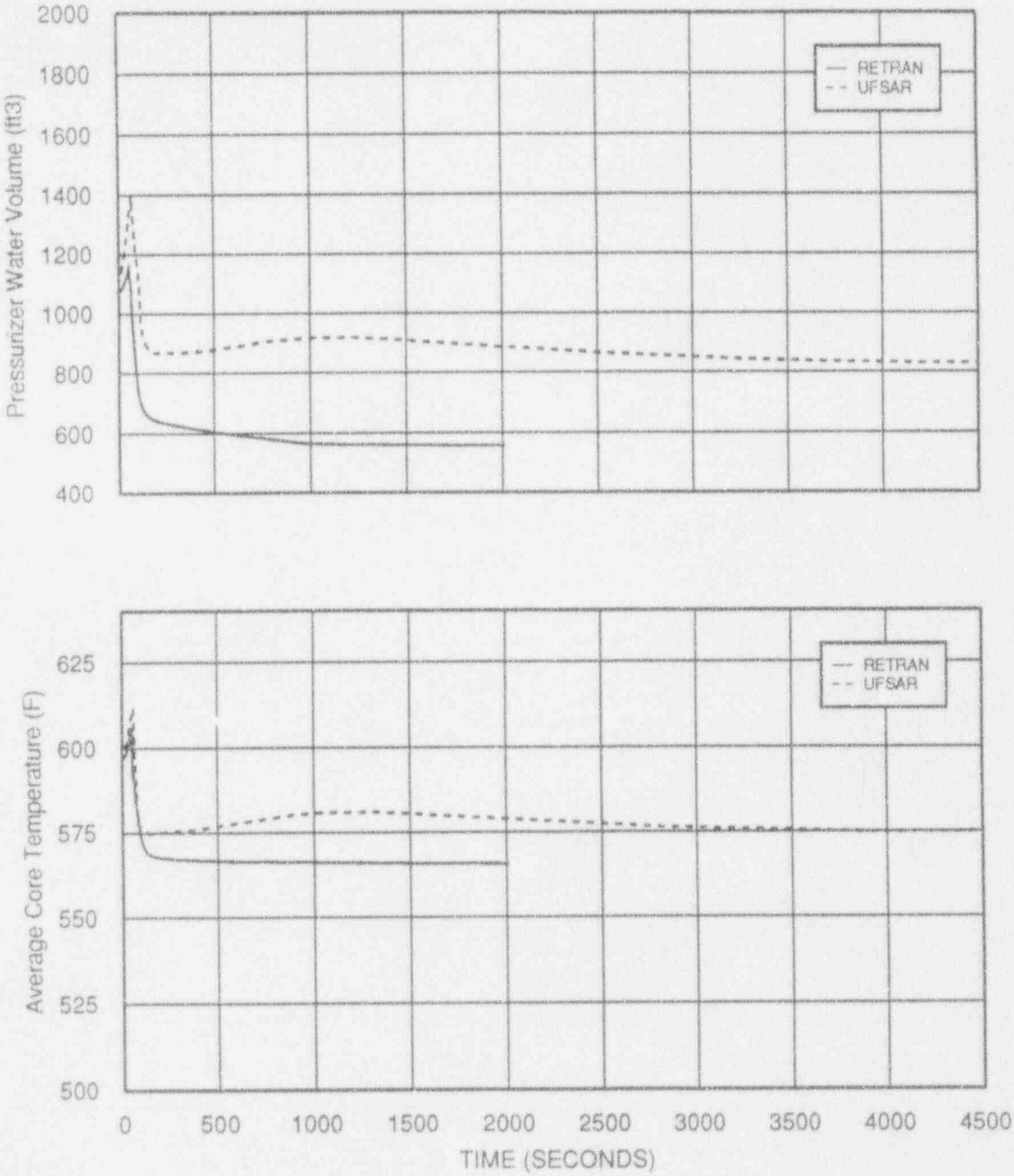
TABLE A-4
SEQUENCE OF EVENTS FOR THE
LOSS OF FEEDWATER FLOW

	Time (Seconds)	
Event	RETRAN	UFSAR
Main feedwater flow stops	0.0	0.0
Low-low steam generator water level trip	36.3	41.1
Rods begin to drop	38.3	43.1
Peak water level in pressurizer occurs	42	46.0

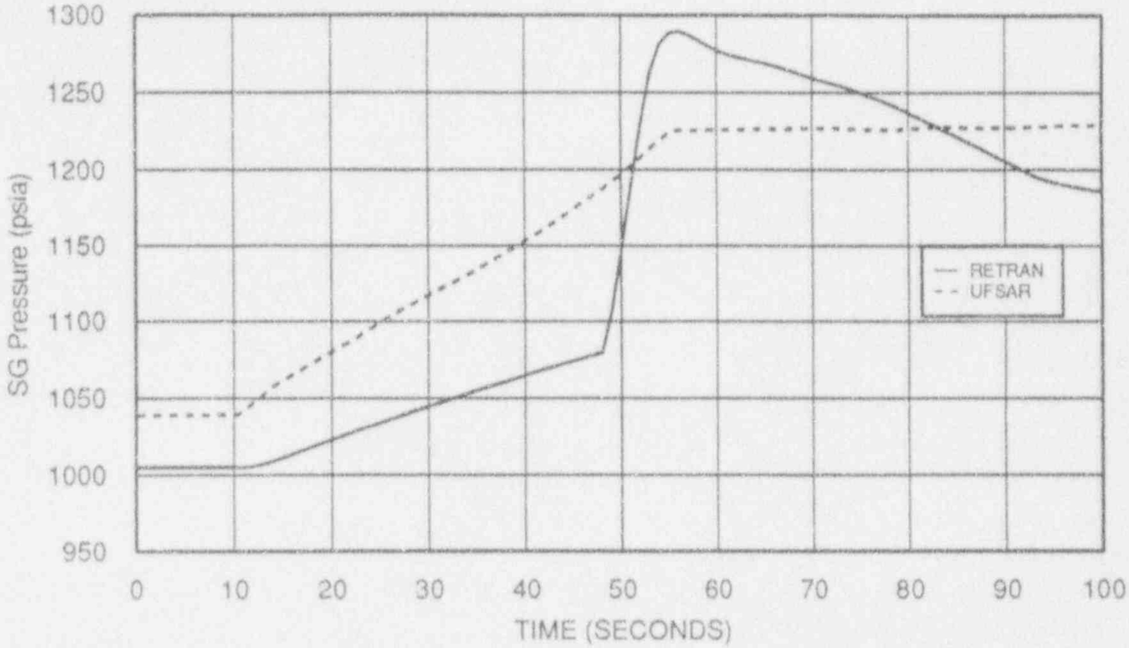


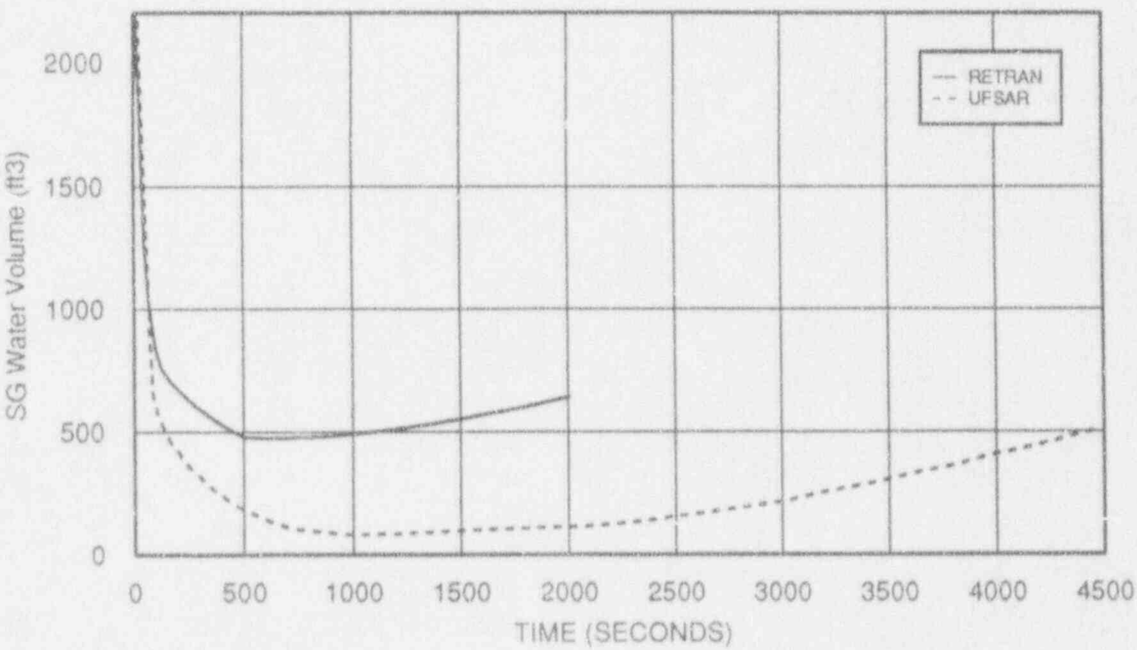


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Feedwater FIGURE A-4.1



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Feedwater FIGURE A-4.2



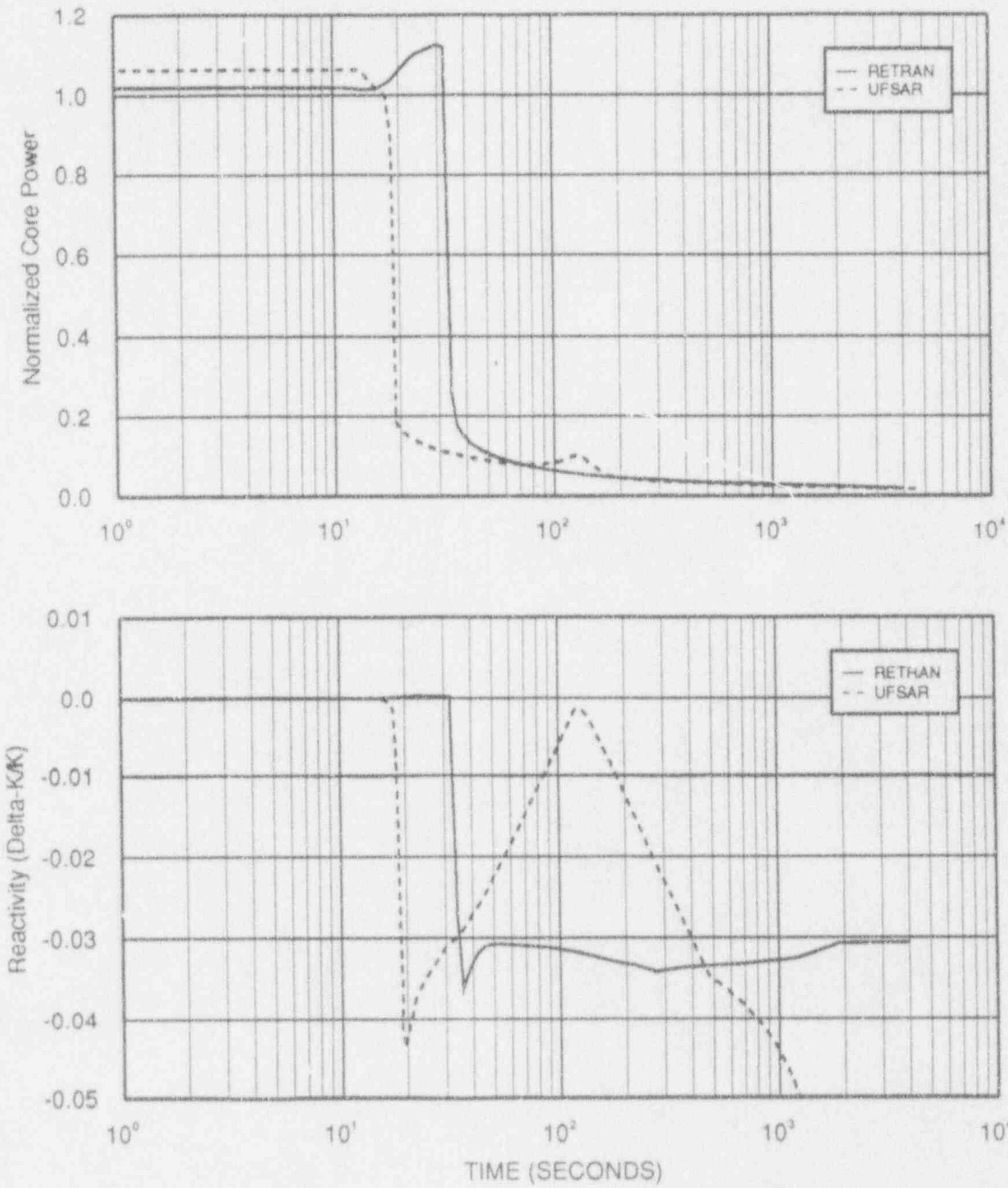


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Loss of Feedwater FIGURE A-4.3

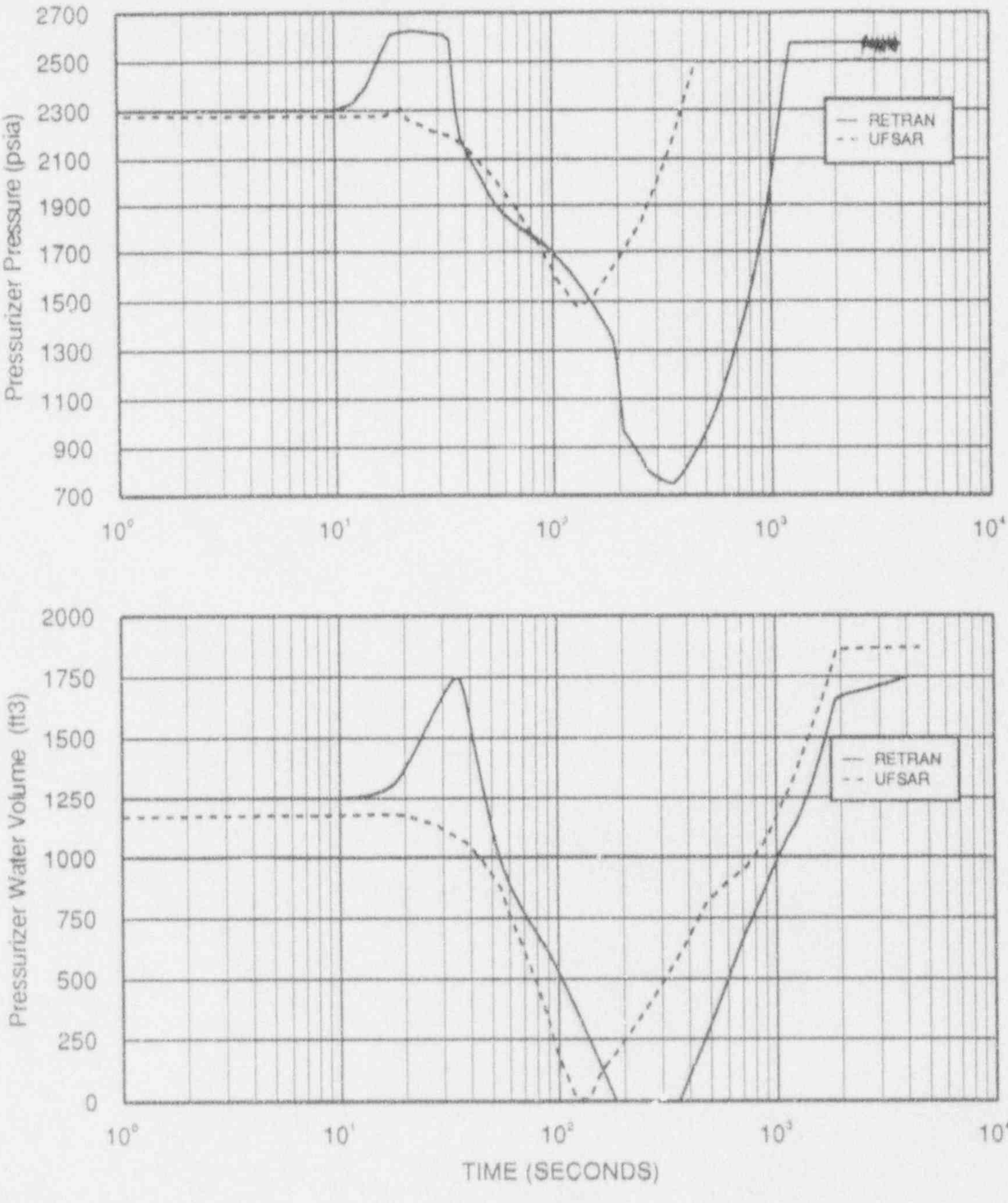
APPENDIX A (Continued)

TABLE A-5
SEQUENCE OF EVENTS FOR THE
FEEDWATER LINE BREAK (WITH OFFSITE POWER AVAILABLE)

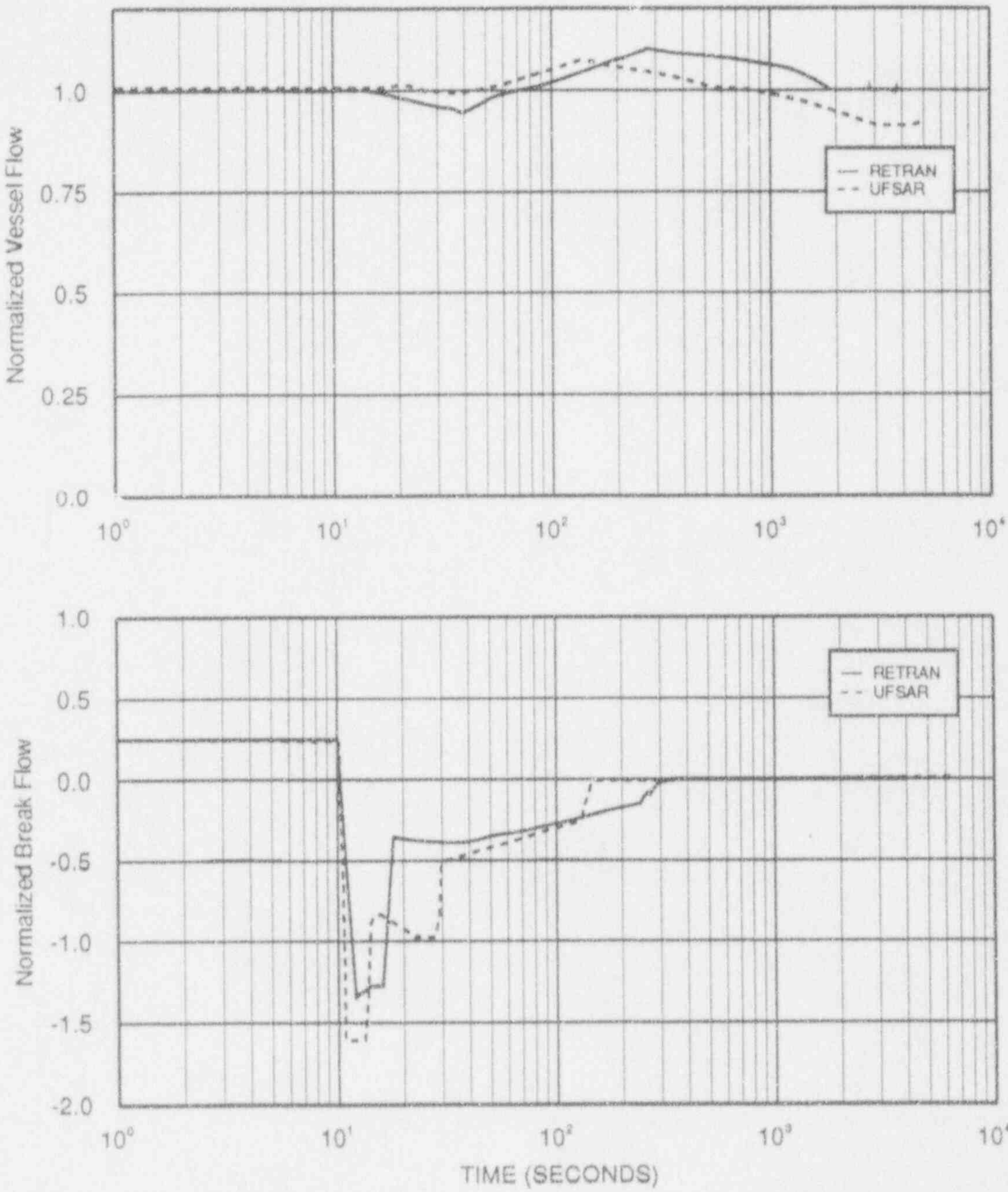
	Time (Seconds)	
Event	RETRAN	UFSAR
Main feedwater line rupture occurs	10.0	10.0
Pressurizer safety valve setpoint reached prior to reactor trip	16.0	--
MSSV setpoint reached prior to reactor trip	18.0	--
Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	29.0	13.6
Rods begin to drop	31.0	15.6
Emergency feedwater is started	106.0	73.6
Low steam line pressure setpoint reached in ruptured steam generator	238.3	118.0
All main steam line isolation valves close	245.3	125.0
Safety injection flow started	268.3	--
Ruptured steam generator inventory completely discharged through break	334.0	--
Feedwater lines are purged and emergency feedwater is delivered to intact steam generators	342.0	456.0
Safety injection flow terminated	1158.0	--
Pressurizer safety valve setpoint reached	1232.0	416.0
Steam generator safety valve setpoint reached in intact steam generators	1852.0	468.0
Core decay heat plus pump heat decrease to emergency feedwater heat removal capacity	4494.0	3820.0



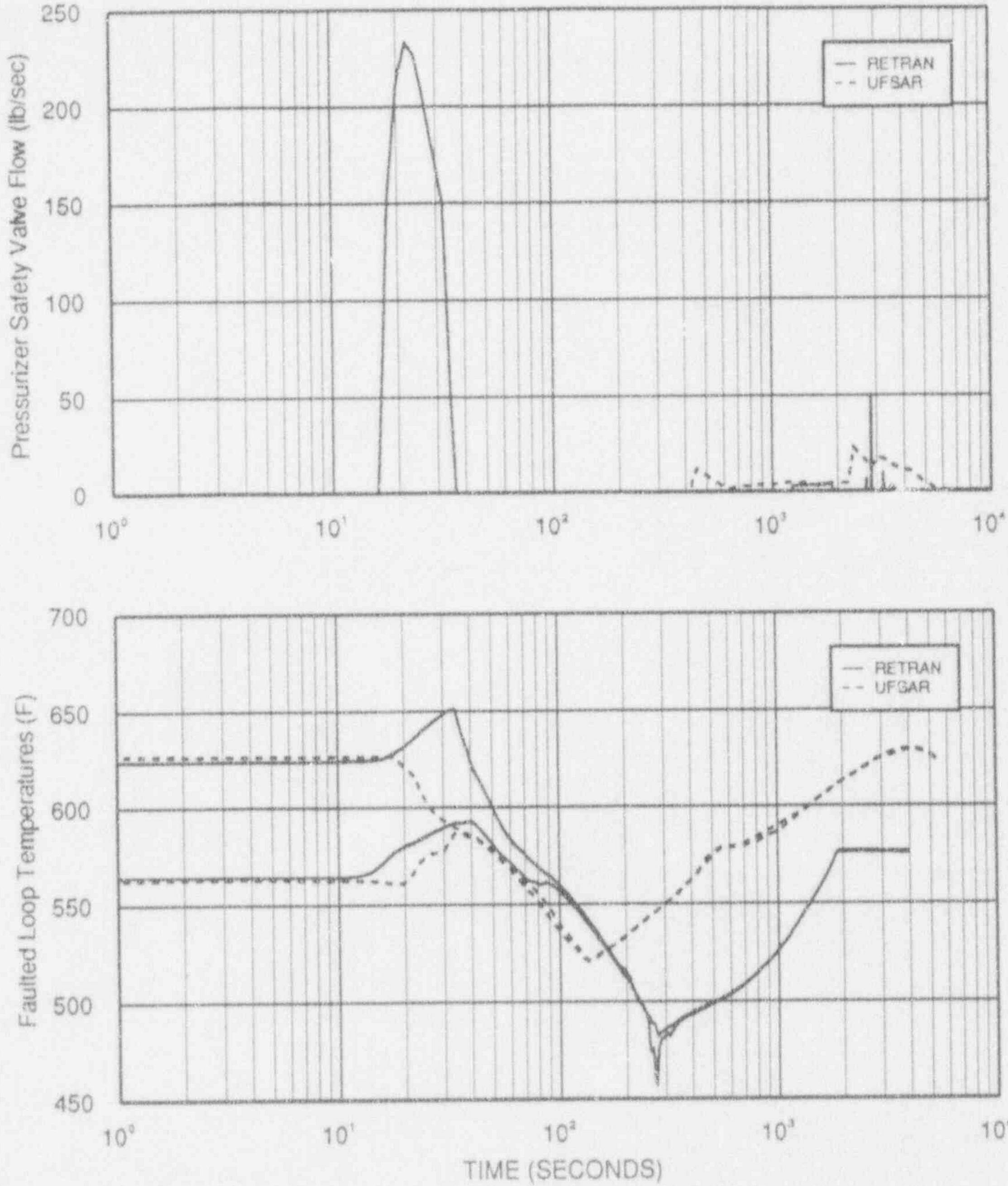
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.1



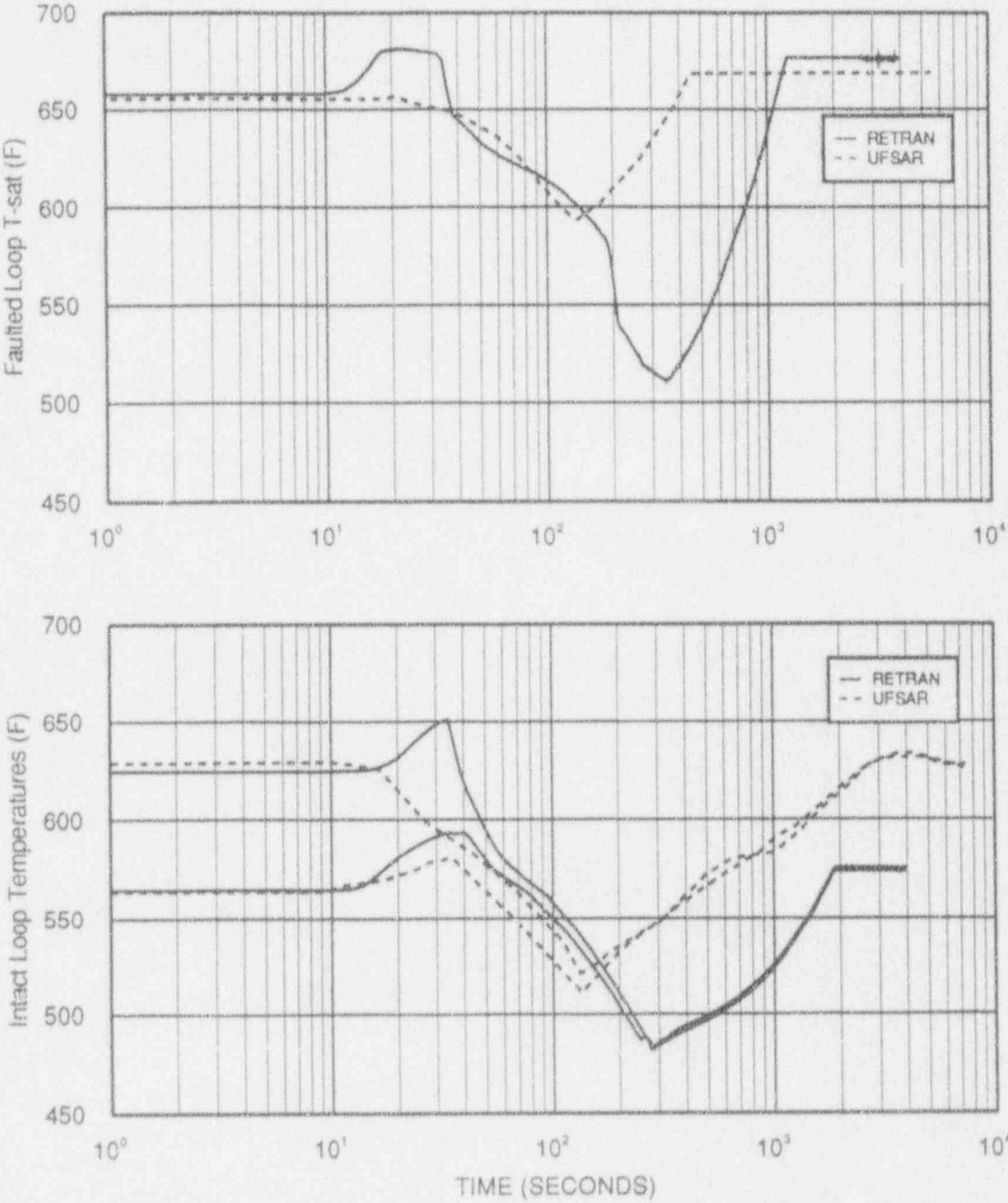
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.2



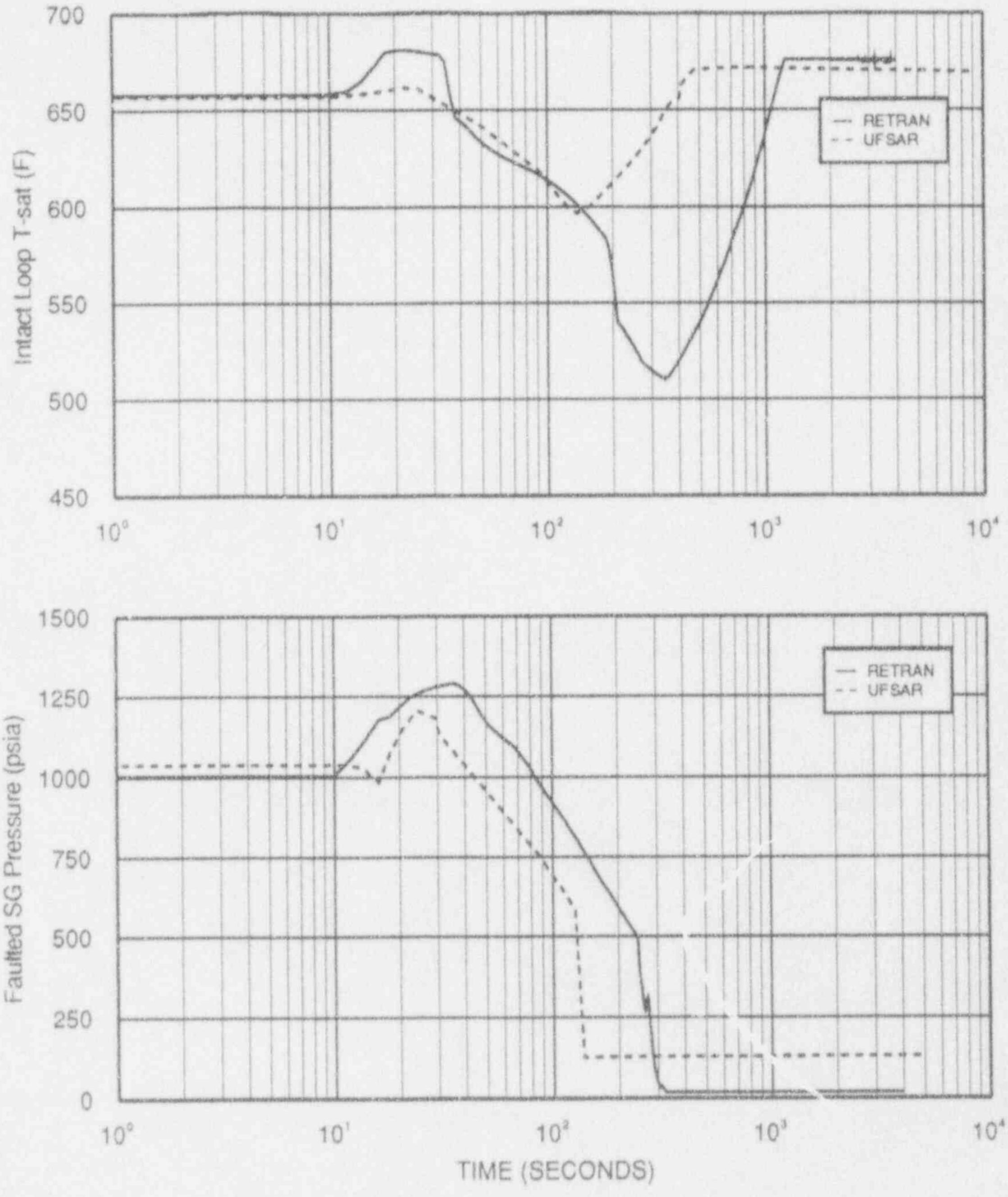
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.3



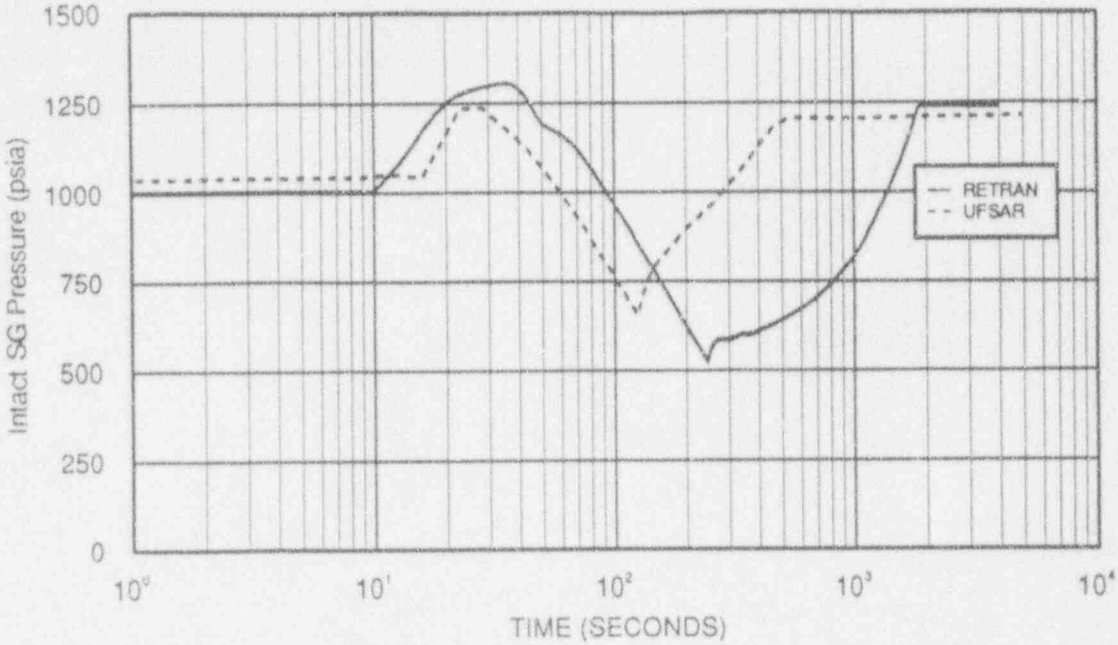
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.4



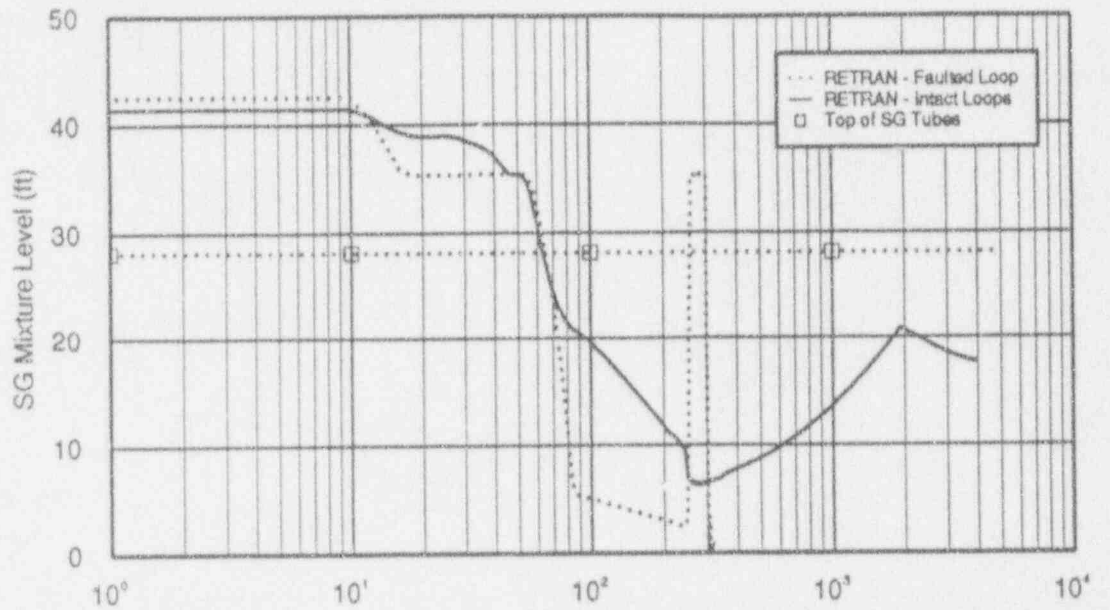
SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.5

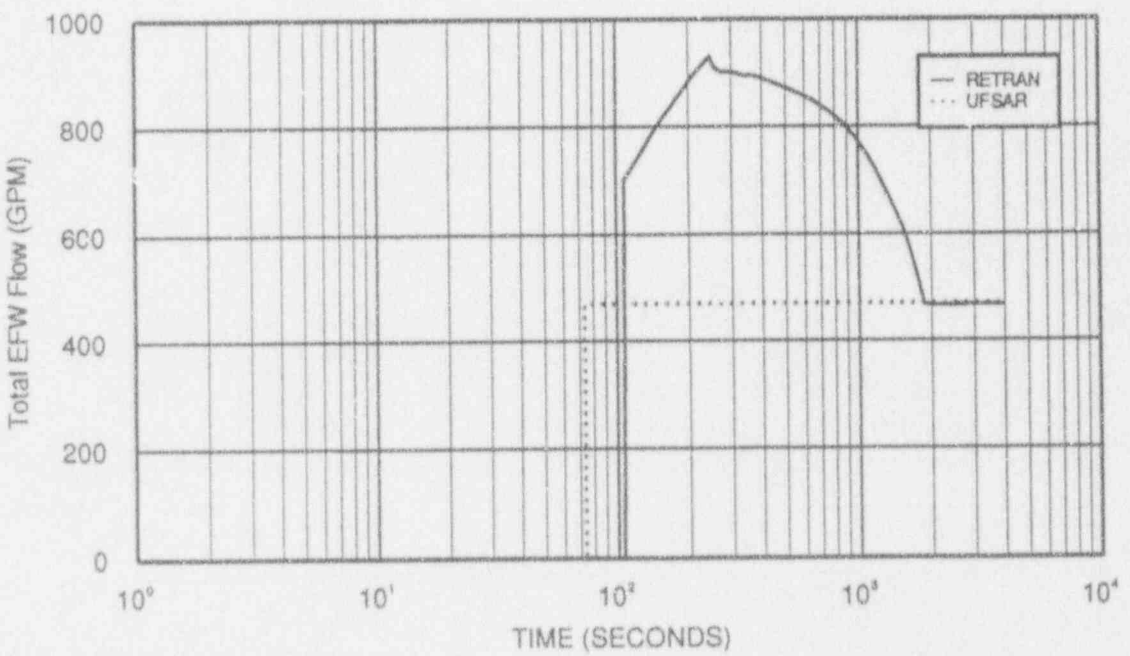


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.6



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.7

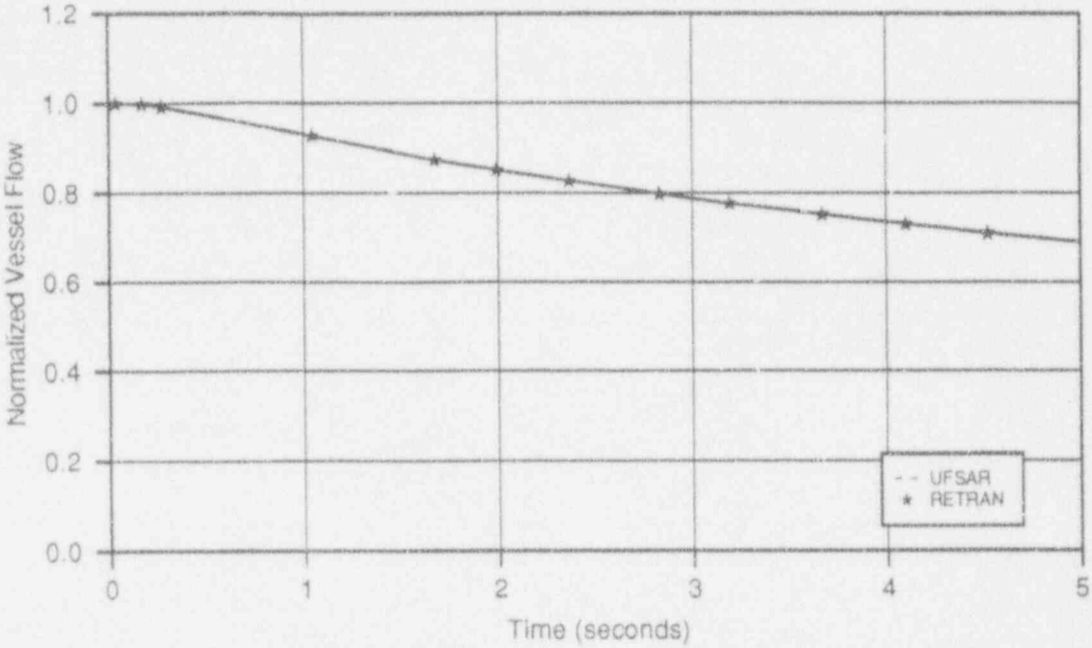


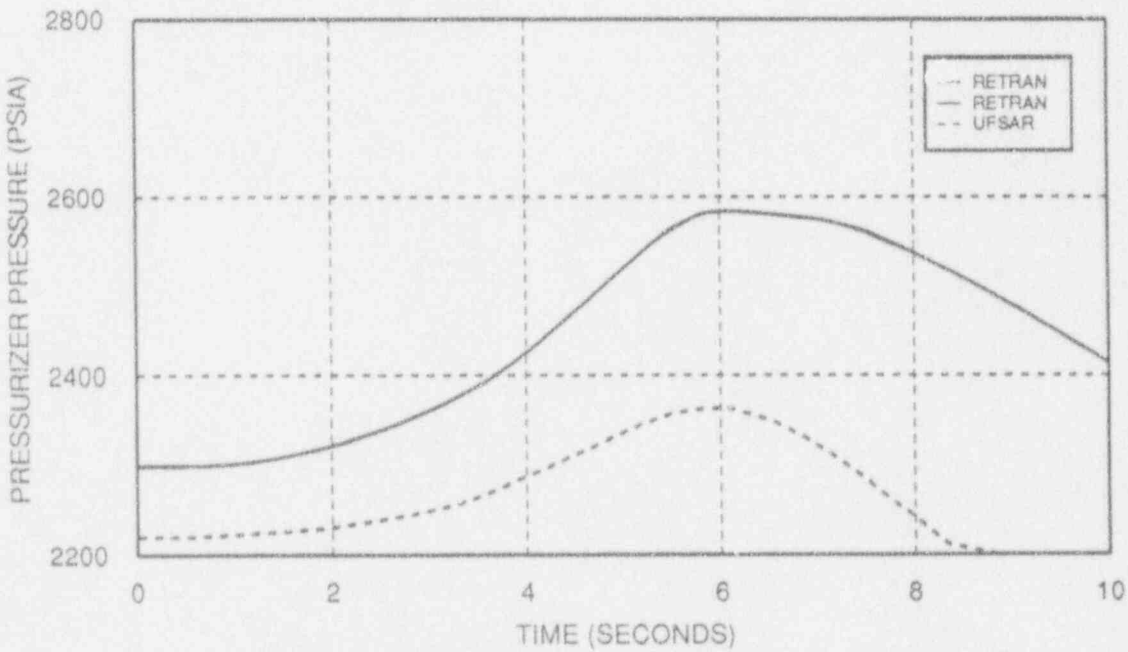


SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Feedwater Line Break (Offsite Power Available) FIGURE A-5.8

TABLE A-6
SEQUENCE OF EVENTS FOR THE
LOSS OF RCS FLOW

Event	Time (Seconds) RETRAN	UFSAR
All operating pumps lose power and begin coasting down	0	0
Reactor coolant pump undervoltage trip point reached	0.24	0.24
Rods begin to drop	1.5	1.5
Minimum DNBR occurs	3.1	3.1





SEABROOK STATION COMPARISON OF RETRAN Vs UFSAR	FOUR LOOPS IN OPERATION FOUR PUMPS COASTING DOWN FIGURE A-6.1

7. LOCKED ROTOR

Comparisons of the RETRAN results from YAEC-1871 versus the Seabrook Station UFSAR results for the cases with offsite power available and with a loss of offsite power are shown in Figures A-7.1 and A-7.2, respectively. The sequence of events are shown in Tables A-7.1 and A-7.2. For the locked rotor analysis, RETRAN is used only to determine the normalized core flow vs. time, the timing of the low RCS flow reactor trip, and the peak pressures.

Normalized Core Flow

For both cases (with offsite power available and with a loss of offsite power) the core flow in the RETRAN analysis initially decreases more slowly than in the UFSAR analysis. However, at the most limiting time in the transient (MDNBR and maximum fuel failures occur at 2.2 seconds) the RETRAN core flows agree well with the UFSAR. The RETRAN core flows following the limiting time in the transient fall slightly below those from the UFSAR analysis. The difference in core flow is attributed to minor differences in fluid inertia and loss coefficients between the RETRAN and UFSAR analyses.

Pressurizer Pressure

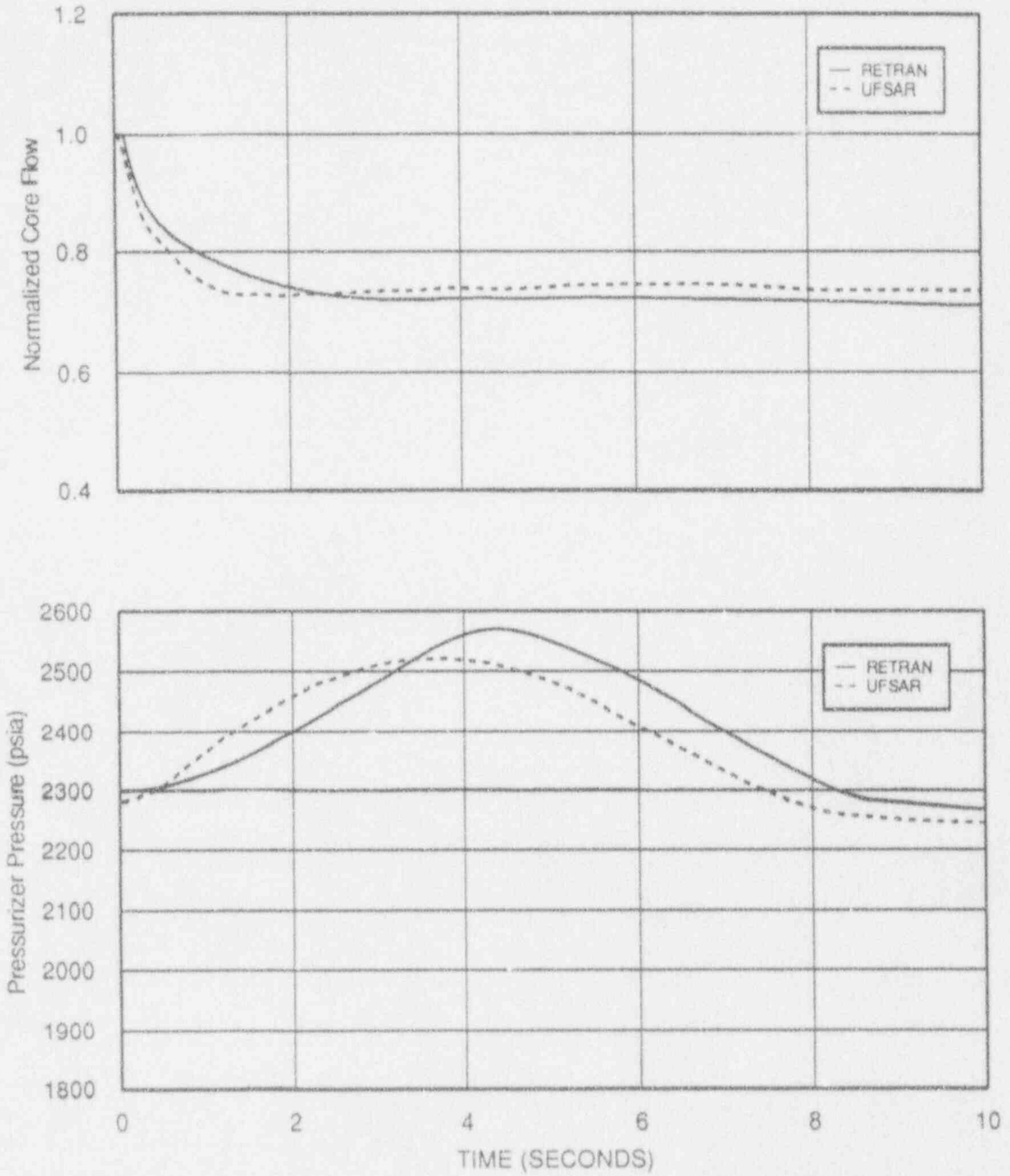
For both cases (with offsite power available and with a loss of offsite power) the initial pressurizer pressure is slightly higher (20 psi) in the RETRAN analysis due to a difference in the assumed pressure uncertainty and pressure control deadband (50 psi in RETRAN vs. 30 psi in the UFSAR). A higher initial pressure is used to conservatively increase the peak RCS pressure. Note that this RETRAN pressure response is not used in determining the MDNBR. For both cases, the peak pressure is greater than the UFSAR analysis due to the positive MTC assumed in the RETRAN analysis. This causes an increase in core power prior to the trip, depositing more heat in the RCS which increases the RCS temperature and pressure.

TABLE A-7.1
SEQUENCE OF EVENTS FOR THE
LOCKED ROTOR (WITH OFFSITE POWER AVAILABLE)

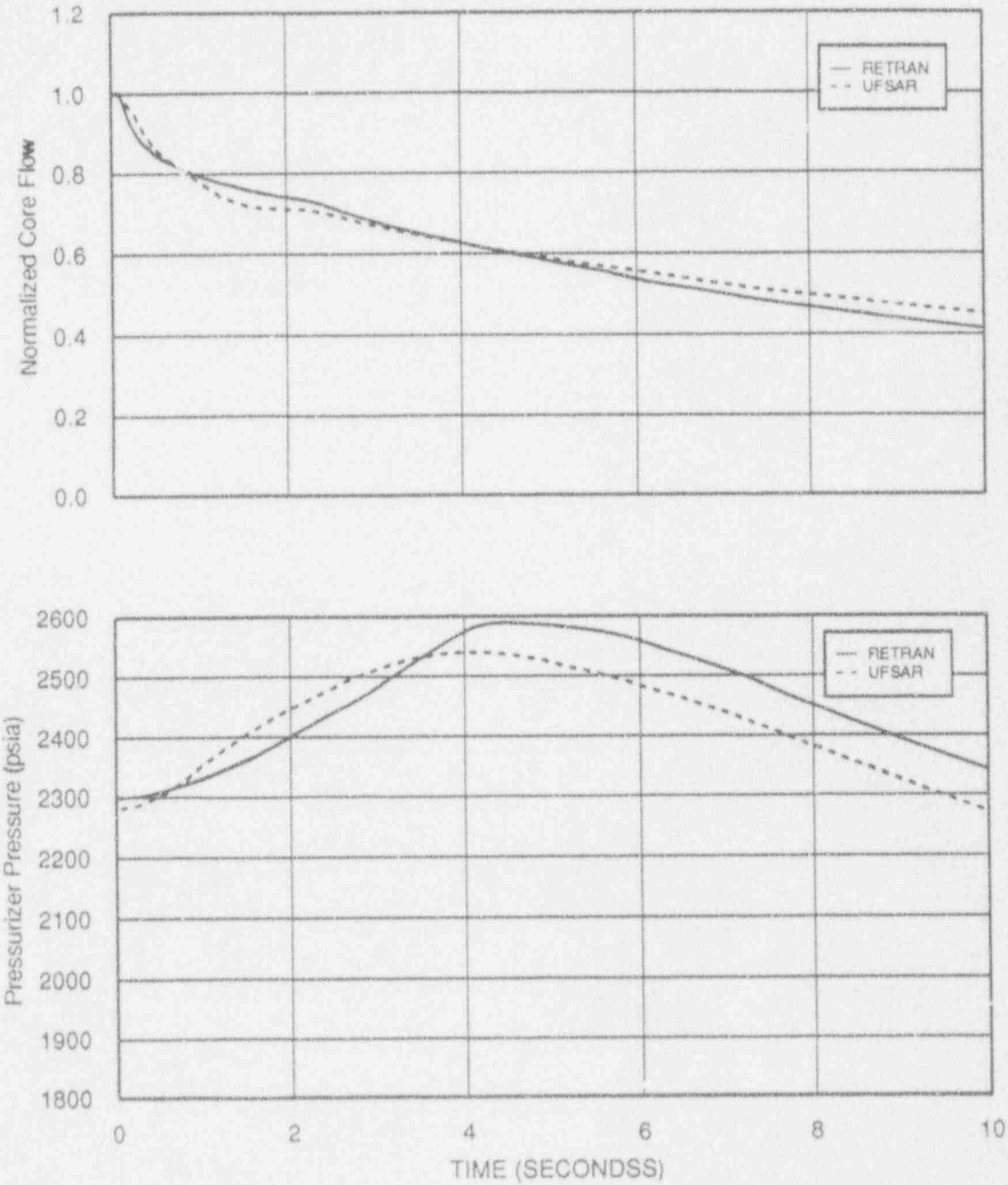
	Time (Seconds)	
Event	RETRAN	UFSAR
Rotor on one pump locks	0.0	0.0
Low flow trip point reached	0.15	0.04
Rods begin to drop	1.15	1.04
Maximum RCS pressure occurs	4.00	3.60

TABLE A-7.2
SEQUENCE OF EVENTS FOR THE
LOCKED ROTOR (WITH A LOSS OF OFFSITE POWER)

Event	Time (Seconds) RETRAN	UFSAR
Rotor on one pump locks	0.0	0.0
Low RCS flow trip setpoint reached	0.15	0.08
Rods begin to drop	1.15	1.08
Remaining reactor coolant pumps begin to coast down	2.15	2.08
Maximum RCS pressure occurs	4.65	4.0



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Locked Rotor (With Offsite Power Available) FIGURE A-7.1



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Locked Rotor (With a Loss of Offsite Power) FIGURE A-7.2

8. ROD WITHDRAWAL

Comparisons of RETRAN results are shown in Figures A-8.1 and A.8-2. The sequence of events is shown in Table A-8. The RETRAN case presented mimics the 80 pcm/sec withdrawal rate presented in the UFSAR. Only system response parameters are presented for comparison.

Core Power

The core power undergoes a similar increase as the UFSAR although the UFSAR includes a 2% calorimetric uncertainty consistent with the deterministic approach. After the high neutron flux trip is initiated there is a faster decrease in core power from RETRAN caused by the use of a less conservative scram curve than that which is used in the UFSAR. The scram curve in RETRAN is still very conservative when compared to the axial flux profiles being used in the UFSAR analysis.

Heat Flux

The core heat flux undergoes a similar increase as the UFSAR. The UFSAR includes a 2% calorimetric uncertainty consistent with the deterministic approach. After the high neutron flux trip, the core heat flux is conservative compared to the UFSAR results.

Pressurizer Pressure

The initial pressure in the UFSAR is based on the deterministic approach where pressure is the nominal value of 2250 psia minus the uncertainty value of 30 psia for a final pressure of 2220 psia. The pressure goes through a similar increase as the UFSAR until the high neutron flux trip is initiated. The RETRAN pressure decreases at a smaller rate after the trip consistent with a smaller rate of decrease in the core heat flux.

9. DROPPED ROD CLUSTER CONTROL ASSEMBLY

A "typical" case with rod control in automatic is presented in the UFSAR. Specifics of the case are not discussed. However, a reasonable comparison is possible by selecting a RETRAN case with 180 pcm dropped rod worth, and an early in life MTC of -3pcm/°F. Comparison of RETRAN results versus the Seabrook Station UFSAR results are shown in Figures A-9.1 through A-9.2.

Normalized Power

RETRAN results show good agreement with the UFSAR results.

Normalized Core Heat Flux

Results show good agreement. The lag in the RETRAN heat flux is explained by the smaller prompt increase in power following the control rod drop.

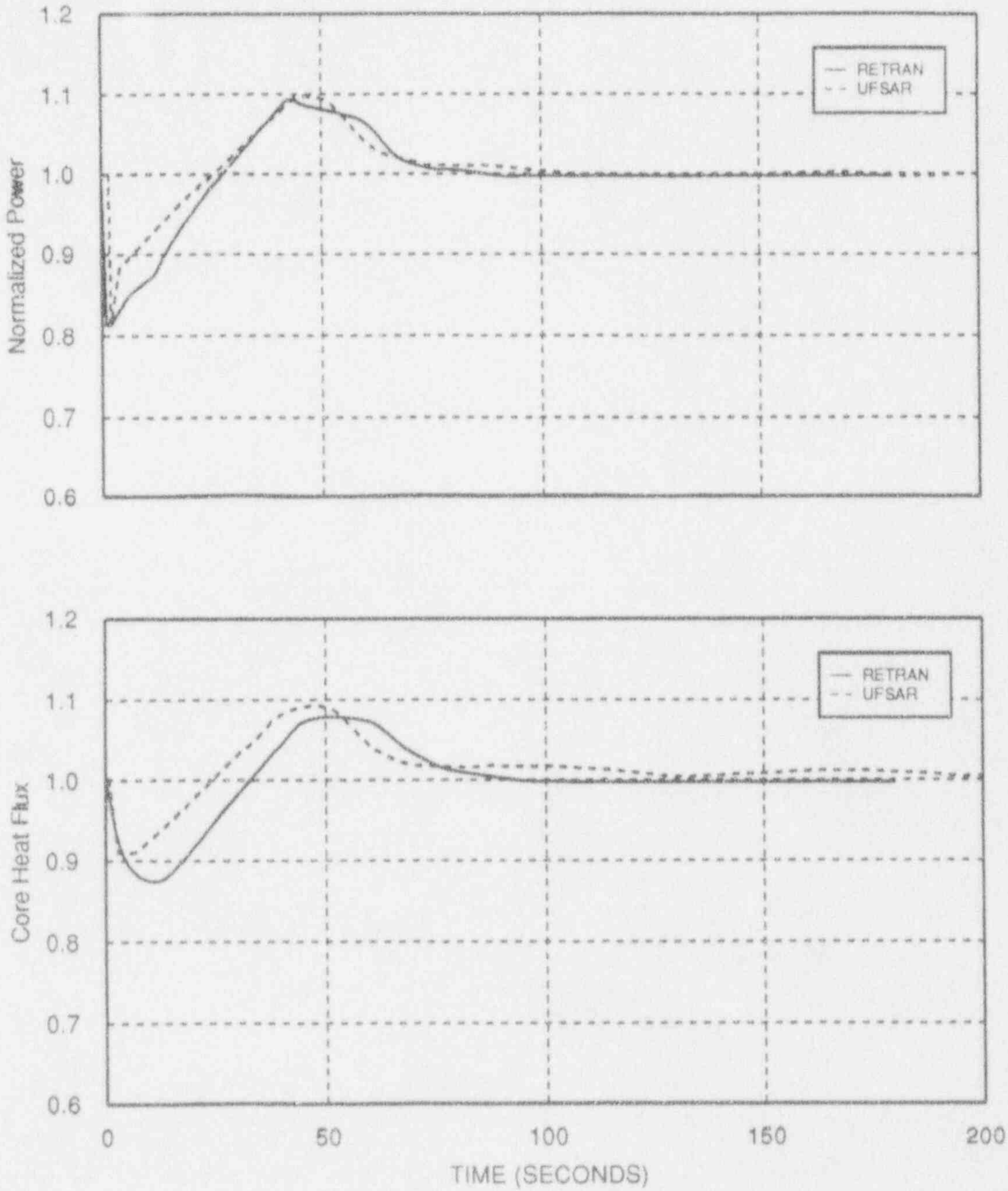
Average Coolant Temperature

The difference in temperatures is attributed to the difference in initial conditions.

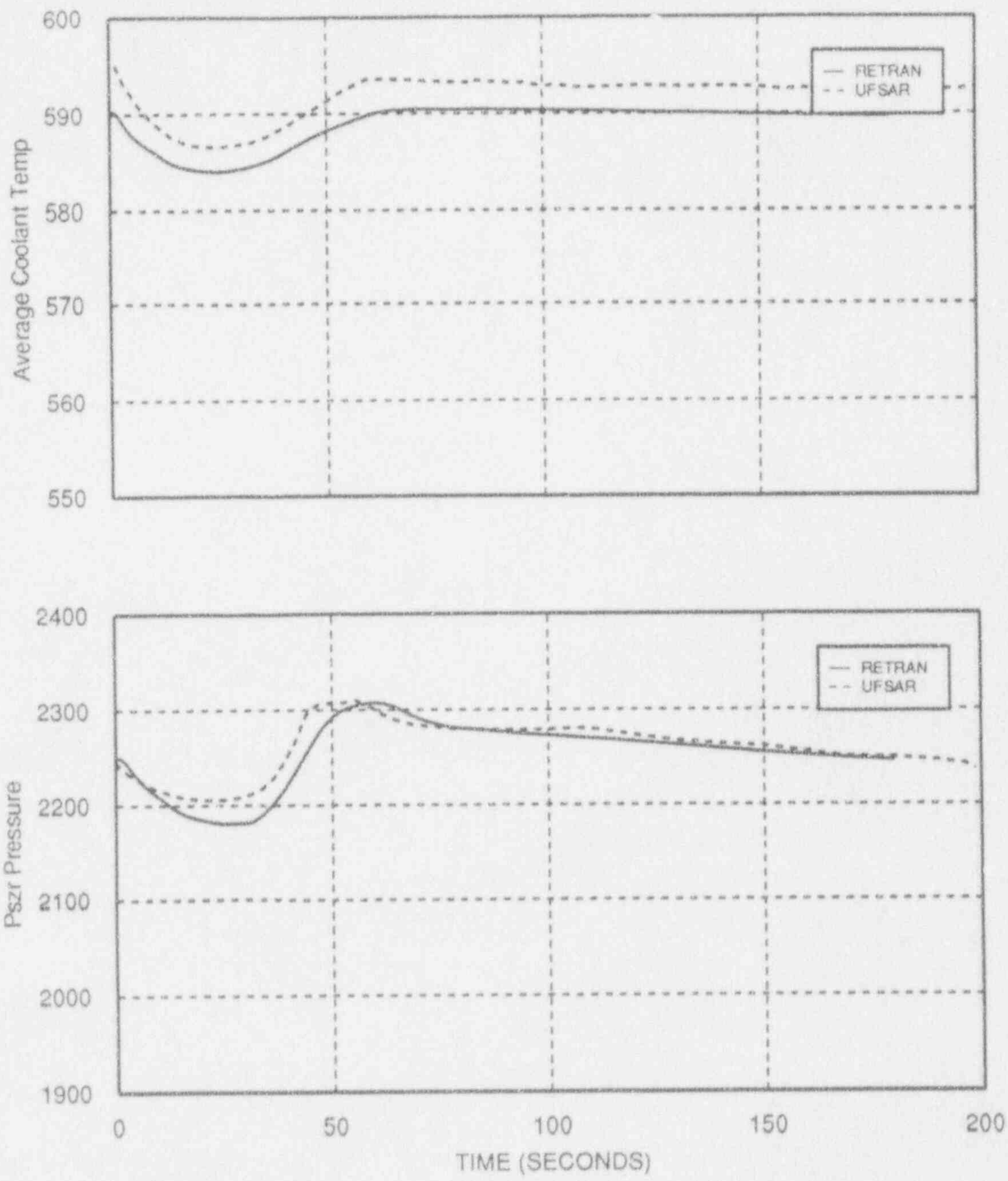
The UFSAR initial temperature = nominal + uncertainty. RETRAN initial temperature = nominal.

Pressurizer Pressure

The difference in pressures is attributed to the smaller prompt power increase following the control rod drop in RETRAN.



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Dropped Rod Cluster Control Assembly AUTOMATIC Rod Control 180 pcm Dropped Worth, -3 pcm/F MTC FIGURE A-9.1



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Dropped Rod Cluster Control Assembly AUTOMATIC Rod Control 180 pcm Dropped Worth, -3 pcm/F MTC FIGURE A-9.2

10. ACCIDENTAL RCS DEPRESSURIZATION

A comparison of the RETRAN results from YAEC-1871 versus the Seabrook Station UFSAR results is shown in Figures A-10.1 and A-10.2. The sequence of events is shown in Table A-10.

Normalized Core Power

The initial core power is lower in the RETRAN analysis since it uses nominal initial conditions, without the 2% uncertainty in power. The nominal power is used to provide the appropriate input to the MDNBR evaluation, since the uncertainties on power, pressure, temperature, and flow are all included in the MDNBR limit for the WRB-1 DNB correlation that is used. Due to the positive MTC assumed in the RETRAN analysis (implemented as a table of reactivity vs. coolant density), the core power rises slightly prior to the trip as the coolant density decreases with the pressure decrease.

The earlier reactor trip in the RETRAN analysis is due to an Over-Temperature ΔT (OTΔT) trip which occurs prior to the low pressurizer pressure trip reflected in the UFSAR analysis. The UFSAR indicates that auto rod control is assumed to avoid the OTΔT trip. RETRAN cases with auto and manual rod control were evaluated, both of which resulted in an OTΔT trip at the same point in time. The difference in reactor trip signals may be due to a disabled OTΔT trip in the UFSAR analysis. If the reactor were assumed to trip on low pressurizer pressure in the RETRAN analysis, the trip time would be almost identical to the UFSAR analysis due to the similarity in the pressurizer pressure responses.

Pressurizer Pressure

The initial pressurizer pressure is slightly lower (20 psi) in the RETRAN analysis due to a difference in the assumed pressure uncertainty and pressure control deadband (50 psi in RETRAN vs. 30 psi in the UFSAR). A lower initial pressure is used to conservatively minimize the MDNBR. Except for the difference in initial pressures and trip times, the pressurizer pressure response matches very well.

Average RCS Temperature

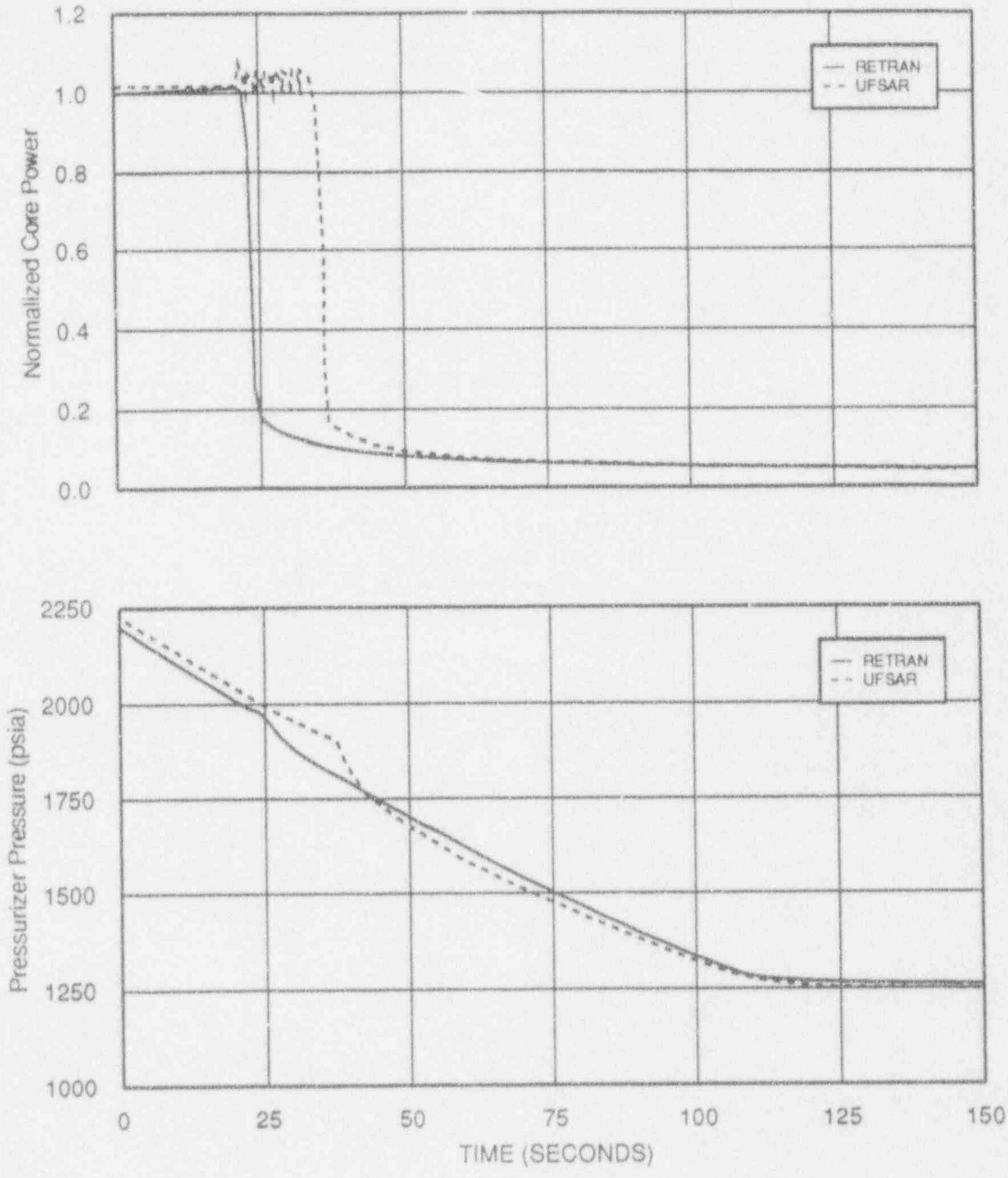
The initial average RCS temperature is slightly lower (4 °F) than in the UFSAR analysis since the RETRAN analysis uses nominal initial conditions to provide the appropriate input to the MDNBR evaluation, as discussed above. The difference in reactor trip times causes a difference in the timing of the post-trip decrease in the average temperature. The difference in temperature is attributed to differences in MSSV modelling between the RETRAN and UFSAR analyses. The SG pressure response in WCAP-7769 ("Overpressure Protection for Westinghouse Pressurized Water Reactors") indicates that Westinghouse may be modelling the MSSVs as open/closed valves. The RETRAN MSSV model ramps the MSSVs open between the setpoint pressure and the accumulation pressure (3% above the setpoint).

The slight increase in the average temperature at the end of the RETRAN analysis is due to the decrease in mass flow through the RCPs as the pressure drops and the steam quality increases in the loops upstream of the RCPs. This results in an increasing ΔT across the RCPs since the flow decreases more than the pump heat. The increasing RCP ΔT causes an increasing cold leg temperature which increases the average temperature.

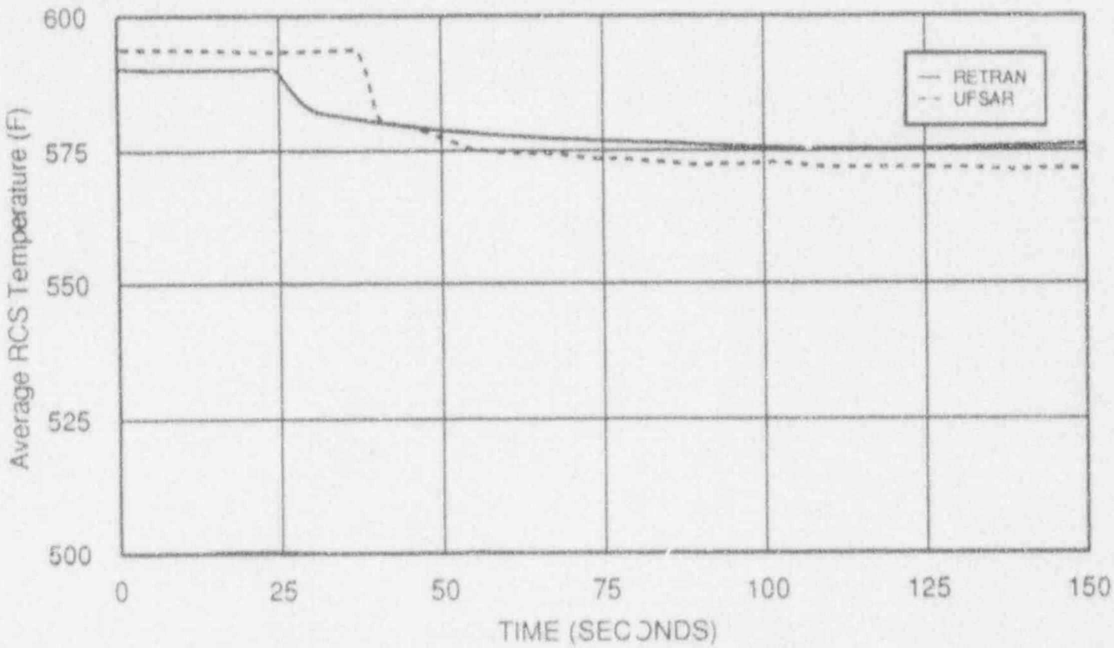
APPENDIX A (Continued)

TABLE A-10
SEQUENCE OF EVENTS FOR THE
ACCIDENTAL RCS DEPRESSURIZATION

Event	Time (Seconds) RETRAN	UFSAR
One pressurizer safety valve failed open	0.0	0.0
Overtemperature ΔT reactor trip setpoint reached	19.1	--
Low pressurizer pressure reactor trip setpoint reached	--	31.3
Rods begin to drop	21.1	33.3



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Accidental RCS Depressurization FIGURE A-10.1



SEABROOK STATION COMPARISON OF RETRAN vs. UFSAR	Accidental RCS Depressurization FIGURE A-10.2

APPENDIX B
ILLUSTRATION OF THE YAEC APPROACH TO CHAPTER 15 ACCIDENT ANALYSIS USING RETRAN APPLIED TO SEABROOK STATION
(YAEC-1871)

YANKEE METHODOLOGY FOR PWR TRANSIENT ANALYSIS USING RETRAN

- Key Features of the Base RETRAN Model in YAEC-1856-P

- Illustration of Application to Specific Non-LOCA Transients

  -- For each transient:

  a) Goal (output) of the RETRAN analysis

  b) Initial conditions

  c) Model changes (if any)

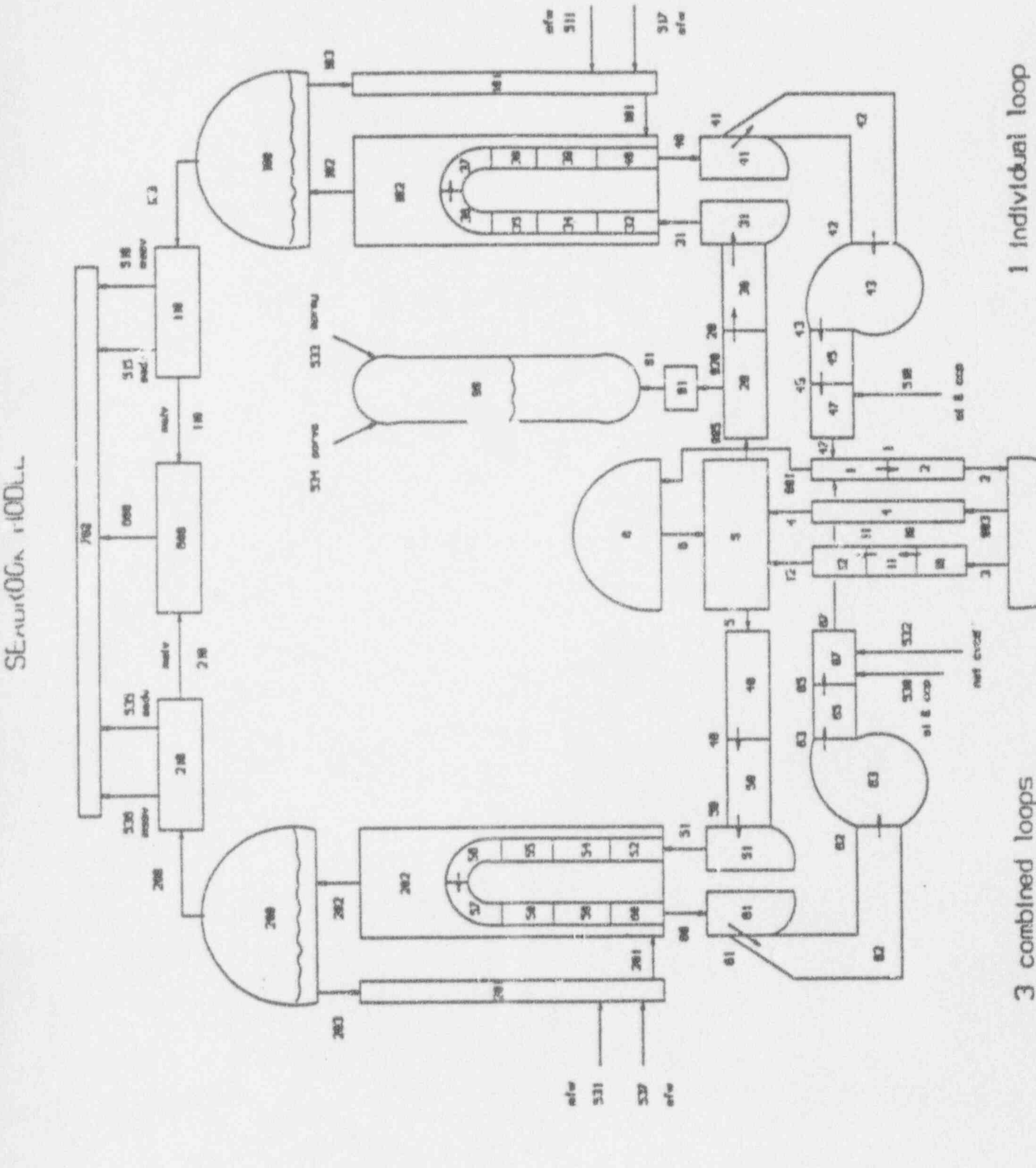
  d) Transient-specific input

  e) Sensitivity studies

- Transient-Specific Illustrations Available in YAEC-1871 (Submitted on the Seabrook Docket)

RETRAN Nodalization

FIGURE B-1



LIST OF NON-LOCA TRANSIENTS

- MSLB
- Loss of Feedwater
- Feedwater Line Break
- Turbine Trip

- Dropped Rod
- Bank & Single Rod Withdrawal
- Loss of Flow
- RCP Locked Rotor and Shaft Break
- 10% Step Load Increase
- Feedwater Flow Increase
- Accidental Depressurization

LOSS OF FEEDWATER (Continued)

- Transient-Specific Input:

  -- Maximum EFW delay and limiting single failure (1 out of 2 EFW pumps operate)

  -- Main FW flow is terminated at the start

  -- Pressurizer PORVs, spray, ASDVs, and Steam Dump are disabled

  -- Rods in manual (no response)

  -- Trip on high pressurizer pressure trip disabled (occurs prior to low SG level)

  -- Hot FW must be swept before cold EFW reaches the SGs

- Sensitivity Studies:

  -- Loss of offsite power (both reported in YAEC-1871)

  -- Passive heat conductors; RCS pipe walls, RV walls and internals, SG walls and internals (slightly more limiting results without)

  -- Sensitivity of primary to secondary heat transfer to tube uncovery (local fluid conditions heat transfer option):

    a) No effect on peak RCS and MSS pressures

    b) SG minimum inventory was lower without local conditions heat transfer (difference ~ 1.5% of initial mass)

FEEDWATER SYSTEM PIPE BREAK

- Goals:
  - -- Show that the EFW capacity is adequate to:
    - a) remove decay heat,
    - b) prevent RCS overpressurization
    - c) prevent fuel damage (DNB)
  - -- Maximum RCS heatup

- Initial Conditions:
  - -- RCS temperatures, pressure, core power = maximum + uncertainties
  - -- Minimum steam volume in pressurizer
  - -- Most positive MTC
  - -- Faulted SG level = nominal + uncertainty (delays trip & EFW actuation)
  - -- Intact SG levels = nominal - uncertainty (maximizes heatup)

- Model Changes:
  - -- DEG break of FW line at SG. Choked flow (extended Henry & Moody)
  - -- SG low level trip setpoint conservatively calculated using minimum SG mass at the low level setpoint (see Loss of FW Flow)

- Transient-Specific Input:
  - -- Offsite power available (RCP heat addition maximizes heatup)
  - -- Pressurizer pressure and level control disabled
  - -- Turbine trip at time of break
  - -- ASDV's and Steam Dump disabled

FEEDWATER SYSTEM PIPE BREAK (Continued)

- Sensitivity Studies:

  -- Local conditions heat transfer option versus standard heat transfer model in SG boiler region. Peak RCS and MSS pressures not sensitive. Timing of peak RCS pressure affected. (Local conditions model = more limiting)

  -- Limiting single failure in the EFW system (1 of 2 pumps versus a failed-closed branch line control valve to one of the intact steam generators) (1 EFW train is limiting)

  -- Containment backpressure effect on break flow (no sensitivity)

LOSS OF LOAD/TURBINE TRIP

- Goal:
  - -- Determine peak RCS and MSS pressures
- Initial Conditions:
  - -- RCS temps and core power = maximum + uncertainties
  - -- RCS pressure = nominal - uncertainty (delays high pressure trip and maximizes core heat input)
  - -- RCS flow = TDF
  - -- Most positive MTC
  - -- Least negative doppler
- Transient-Specific Input:
  - -- EFW, ASDV's, Steam Dump disabled
  - -- Main FW flow terminated at time of trip
  - -- Rod control system disabled to maximize heat input prior to trip
- Sensitivity Studies:
  - -- Comparison to UFSAR show in YAEC-1856P
  - -- Initial SG pressure:
    - a) Low is conservative for RCS peak pressure
    - b) High is conservative for MSS peak pressure
  - -- RCS pressure control:
    - a) off = maximum RCS pressure
    - b) on = maximum MSS pressure
  - -- MTC (most positive limiting)
  - -- Pressurizer level (no sensitivity from YAEC-1847)

DROPPED RCCA

- Goal:
  - -- Calculate Core T&H conditions for DNB analysis during power overshoot

- Initial Conditions:
  - -- Nominal RCS temperatures, pressures, and flow (uncertainties in DNBR limit value)
  - -- Core power = variable
  - -- Doppler = least negative
  - -- MTC = variable
  - -- Pressurizer pressure control

- Model Changes:
  - -- Turbine inlet junction modeled as negative fill with constant flow (i.e., constant turbine load)

- Transient-Specific Input:
  - -- Dropped rod worth (up to full bank)
  - -- Limiting single failure in rod control system causes power overshoot
  - -- Rods in automatic

- Sensitivity Studies:
  - -- Dropped rod worth
  - -- MTC
  - -- Doppler (least negative)
  - -- Initial power level (limiting at high powers but affected by ΔI bandwidth)

DROPPED RCCA (Continued)

- Sensitivity Studies (Continued):
  - -- Rods in manual (rods in automatic)
  - -- Control bank worth (high worth)
  - -- Excore nuclear power "tilt" factor (smaller factor conservative)

BANK WITHDRAWAL ANALYSIS

- Goals:
  - Plant T&H response for DNB analysis
  - Demonstrate RTS coverage/effectiveness
- Initial Conditions:
  - 10%, 70%, 100%
  - Nominal RCS temperature, pressures, & power level
- Model Changes:
  - None
- Transient-Specific Input:
  - Pressurizer pressure + level controls systems functional (to minimize pressure)
  - Rod control disabled
- Sensitivity Studies:
  - Reactivity Insertion Rate (power level dependent)
  - Minimum and maximum reactivity feedback (power level dependent)
  - B_{eff} (weak sensitivity, conservative = large)

SINGLE RCCA WITHDRAWAL

- Goal:
  - Plant T&H response for DNB analysis
- Initial Conditions:
  - RCS temperatures, pressure, and core power = nominal
  - MTC = most positive
  - Doppler = least negative
- Model Changes:
  - None
- Transient-Specific Input:
  - Reactivity addition for withdrawal of a single rod at maximum rate
  - Control banks frozen
- Sensitivity Studies:
  - Core power = variable
  - Minimum/Maximum feedback (least negative feedback)
  - Minimum/Maximum B_{eff} (no sensitivity)
  - Pressure + level control operable/disabled (operable more limiting)
  - Withdrawn RCCA location (limiting = bank D diagonal)

LOSS OF FLOW

- **Goals:**
  - -- Calculate Core T&H conditions for DNB analysis (need flow coastdown only)
  - -- Flow coastdown shown in YAEC-1856P
  - -- Calculate peak RCS pressure response
- **Initial Conditions:**
  - -- RCS pressure, temperature, core power = maximum + uncertainty
  - -- Most positive MTC
  - -- Least negative Doppler
- **Model Changes:**
  - -- None
- **Transient-Specific Input:**
  - -- RCP's tripped
  - -- Pressure and level control disabled
  - -- ASDVs and Steam Dump disabled
  - -- Main FW flow terminated at time of trip
- **Sensitivity Studies:**
  - -- None

RCP LOCKED ROTOR AND SHAFT BREAK

- Goals:
  - -- Plant T&H response
  - -- Peak RCS and MSS pressures
  - -- Normalized core flow to DNB (fuel failure) analysis
  - -- Reactor trip time

- Initial Conditions:
  - -- RCS temps, pressure, and core power = maximum + uncertainties
  - -- Most positive MTC
  - -- Least negative Doppler

- Model Changes:
  - -- None

- Transient-Specific Input:
  - -- Pressurizer PORVs, spray, ASDVs, and Steam Dump disabled
  - -- Main feed flow terminated at time of trip
  - -- Shaft break simulated by forcing pump torque to zero, allowing reserve rotation, and simulating pump impeller inertia only
  - -- Locked rotor simulated by forcing pump speed to zero

RCP LOCKED ROTOR AND SHAFT BREAK (Continued)

- Sensitivity Studies:
  - Looked at Locked Rotor/Shaft Break/LOOP (most conservative = Locked Rotor with LOOP)
  - Looked at normalized coastdown from TDF and Minimum Measure Flow (no difference)
  - Looked at most negative Doppler (least negative Doppler is most limiting)

10% STEP LOAD INCREASE

- Goal:
  - -- Calculate core T&H conditions for DNB analysis
- Initial Conditions:
  - -- RCS temperatures, core power = maximum + uncertainty (conservative)
  - -- RCS pressure = nominal - uncertainty (conservative)
- Model Change:
  - -- Turbine throttle valve junction is changed to a negative fill junction to model 10% step increase in steam flow
- Transient-Specific Input:
  - -- Pressurizer heaters and charging/letdown are disabled (yields minimum pressure)
- Sensitivity Studies:
  - -- Most positive & most negative MTC
  - -- Rods in automatic and manual

FEEDWATER FLOW INCREASE

- **Goals:**
  - -- Calculate plant T&H response
  - -- Verify that reactivity insertion rates are bounded by the RCCA bank withdrawal analysis
  - -- Perform DNBR check

- **Initial Conditions:**
  - -- RCS temperature, core power = maximum + uncertainties (conservative)
  - -- RCS pressure = nominal - uncertainties (conservative)
  - -- MTC = most negative
  - -- Doppler = least negative

- **Model Changes:**
  - -- Split RV model from MSLB analysis (no mixing between affected loop and other loops)
  - -- Split core model from MSLB analysis (with reactivity weighting)

- **Transient-Specific Input:**
  - -- Step increase in FW flow to one SG disabled
  - -- Pressurizer heaters and charging/letdown

- **Sensitivity Studies:**
  - -- HZP & HFP
  - -- With and without reactor trip on turbine trip (turbine trip is from high SG level)

ACCIDENTAL RCS DEPRESSURIZATION

- Goal:
  - -- Calculate Core T&H conditions for DNB analysis
- Initial Conditions:
  - -- RCS temperatures and core power = maximum + uncertainties
  - -- RCS pressure = nominal - uncertainty
  - -- Most positive MTC
  - -- Least negative Doppler
- Model Changes:
  - -- Pressurizer safety valve:
    - a) Single valve stuck open
    - b) Critical flow (extended Henry & Moody)
    - c) Area adjusted to provide rated flow
- Transient-Specific Input:
  - -- Main FW flow terminated at time of trip
  - -- ASDVs, Steam Dump and pressurizer heaters disabled
  - -- Level control disabled
  - -- Rod control system in manual
- Sensitivity Studies:
  - -- None