May 10, 2002

MEMORANDUM TO: Gary Holahan, Director
Division of Systems Safety and Analysis

THRU: Ralph Caruso, Section Chief ***/RA/***
Section Chief, Reactor Systems Branch
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SUBJECT: SENSITIVITY STUDY OF PWR REACTOR VESSEL BREAKS

This memorandum documents a study using RELAP5 to determine the consequences of postulated breaks in the reactor vessel of PWRs. The study was performed in response to questions arising following recent occurrences of cracks in CRDM housings at several PWRs and corrosion in the upper head at Davis Besse. Breaks up to 8 inches in diameter (0.349 ft^2) were analyzed in the upper reactor vessel head. Most of the analyses were for a B&W design (Oconee) with discussions of the effect of plant differences between Oconee and Davis Besse. Plant designs by Westinghouse and CE were also analyzed. Full and partial plant safeguards, delayed reactor pump trip and failure to scram were analyzed.

For postulated breaks in the upper head, the consequences are similar to piping breaks analyzed in plant FSARs as part of the design basis. No new phenomena or unexpected results were obtained. Plants designed by all three PWR reactor vendors were analyzed.

The consequences of failure to scram were evaluated and found to be minimal. This is because of the negative reactivity produced by core voiding and later by boric acid addition.

Attachment:
As stated

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SENSITIVITY STUDY FOR BREAKS IN THE REACTOR VESSEL

Walton Jensen
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1. BACKGROUND

Leaks and cracks in the CRDM thimbles have been observed at Oconee and were subsequently observed at other operating PWRs. The extensive upper head pitting at Davis-Besse has added interest to the consequences of a LOCA in a reactor vessel upper head. This is a break location not routinely analyzed in plant FSARs except for the consequences of an ejected control rod assembly that might result from the rupture of a CRDM thimble. The LOCA that would result from the rupture of a single CRDM thimble is judged to be bounded by the larger break sizes assumed to occur in the coolant piping in FSAR analyses. As part of this study the reactor response to a postulated upper head break is compared to a break of equivalent size in the coolant piping.

Breaks up to an equivalent diameter of 8 inches (0.349 ft^2) are analyzed in the top head of PWRs. B&W defines small breaks as less than 0.5 ft^2. Westinghouse defines small breaks as less than 1.0 ft. For the purposes of LOCA analysis the break sizes analyzed in this study are all in the small break range.

2. ANALYSIS OF REACTOR VESSEL UPPER HEAD BREAKS

We utilized a RELAP5 input model prepared by INEL for Oconee (Ref. 1). Both loops of the reactor system and portions of the steam and feedwater systems are described in the RELAP5 model. Davis Besse is a Babcock and Wilcox plant similar to Oconee. The major difference is that Davis Besse has a raised loop design which means that the once through steam generators are above the reactor core. See Figure 1. The hot legs are extended upward and provide an additional source of water above the core in comparison to Oconee. The lowered loop arrangement for Oconee and the other B&W operating plants is shown in Figure 2.

Seven RELAP5 cases were analyzed for Oconee. Cases 1 through 5 are for breaks in the upper head of progressively larger size to explore the occurrence of unexpected phenomena. Case 4 assumes operators were late in tripping the reactor coolant pumps on loss of subcooling margin as required by procedures. Cases 6 and 7 explore the consequences of the control rods not being inserted. Cases 8 and 9 compare the consequences of a CRDM break to a break of equivalent size in a hot leg. In all these analyses the first 100 seconds is a null transient to achieve steady state. The break occurs at the end of 100 seconds in the attached plots.

<u>Oconee Cases with Partial Safeguards</u>

Case 1: The rupture of a single CRDM thimble with AFW and minimum HPI flow. See Figures 3 and 4.

Case 2: The rupture of two CRDM thimbles with AFW and minimum HPI flow. See Figures 5 and 6.

Case 3: The rupture of three CRDM thimbles with AFW and minimum HPI flow. See Figures 7 and 8.

Case 4: The rupture of three CRDM thimbles with AFW and minimum HPI flow and delayed reactor coolant pump trip. See Figure 10.

Case 5: A rupture in the reactor vessel head 8 inches in diameter with AFW and minimum HPI flow. See Figures 11 and 12.

Case 6: A rupture in the reactor vessel head 8 inches in diameter with AFW and minimum HPI flow for which the control rods did not insert. See Figures 13 and 14.

Case 7: The rupture of a single CRDM thimble with AFW and minimum HPI flow for which the control rods did not insert. See Figures 15 and 16.

Case 8: The rupture of a single CRDM thimble without AFW, HPI or steam dump capability to the condenser. See Figure 17.

Case 9: A hot leg leak equivalent in area to a CRDM thimble. AFW, HPI and steam dump capability to the condenser were assumed to be unavailable. See Figure 17.

In all analyses but case 4 the reactor coolant pumps were assumed to trip on loss of subcooling margin in accordance with emergency procedures. In all cases but 8 and 9 operator action was assumed to raise the steam generator level to the top of the operating range as required by procedures on loss of forced flow and subcooling.

Cases 1, 2. 3, and 5 approximate the assumptions made in the plant's design basis. No unusual phenomena were calculated to occur. Because of the coolant loop arrangement of Babcock and Wilcox designed reactors natural circulation can be lost during a small break LOCA so that the core can be isolated from the heat removal capabilities of the steam generators. Loss of natural circulation would cause an increase in reactor system pressure and an increase in water loss from the break. Operators are instructed to manually increase steam generator level following loss of subcooling margin to ensure that the core remains covered. This action was assumed to be taken at ten minutes after the break occurred. No significant loss of natural circulation was calculated to occur and no core uncovery was calculated.

As the break sizes became larger the reactor was depressurized by break flow without the need for steam generator heat removal. For smaller breaks operator action to depressurize the steam generators would be required to make low pressure safety injection effective for the establishment of long term cooling. Figure 5 shows the effect of operator action in enhancing cooldown so that low pressure safety injection could be successfully initiated.

Effect of Delayed Reactor Coolant Pump Trip (Case 4)

After TMI-2 the need to trip the reactor coolant pumps on loss of subcooling margin was recognized. For a range of small break sizes a window of time was identified for which if the reactor coolant pumps were tripped within the window, extended core uncovery might occur. The window calculated by B&W is shown in Figure 9. For the 3-CRDM break case operator action in tripping the coolant pumps was assumed to be delayed so that the trip occurred after 10 minutes which is in the middle of the window period. The effect of the delayed pump trip (Figure 10) was a spike in core voiding but core uncovery was not predicted by RELAP5.

Effect of Failure to Scram (Cases 6 and 7)

Normally control rod entry is assumed for SBLOCA but not for LBLOCA because of concern that hydraulic forces would interfere with control rod motion following a large break. The 8 inch break size was the largest break size examined and is analyzed here to determine the consequences of failure of the control rods to enter the core.

For the 8 inch break size the reactor system was predicted to depressurize rapidly and decrease below that of the steam generators since this break size is sufficient to remove all decay heat (Figure 11). LPI was automatically initiated and no core uncovery was predicted (Figure 12). The effect of failure of the control rods to enter the core was minimal. The core did not uncover for this case and the calculated core void fractions (Figure 13) are almost identical to those for the case for which the control rods were assumed to be inserted. The reason for this is that voiding in the reactor core shuts down the reactor with negative reactivity soon after the break occurs. See Figure 14. Further negative reactivity is added by the boric acid in the core flood tanks and in the HPI and LPI water. A degree of positive reactivity addition occurs from temperature reactivity feedback but the boric acid addition is sufficient to overcome this effect.

The failure of a single CRDM thimble was also analyzed with the assumed failure of the control rods to enter the core. This break size would be expected to depressurize slower that the 8 inch break size so that negative reactivity from void formation and boron addition would be expected to occur later. This case is the same as case 1 with the exception that the control rods did not enter the core. The failure of the control rods had little effect on the results. Void formation rapidly reduced the core power level. The core was predicted to remain covered with water. See Figure 15. Core reactivity with and without scram is shown in Figure 16. The reactivity is continuously negative. The sudden increase in reactivity after 2000 seconds is the result of cold water entering the core from the core flood tanks displacing some of the steam bubbles. The core cannot return to critical from addition of this water since it contains 2000 ppm of boric acid.

Comparison or an Upper Head Break to a Break in a Hot Leg (Cases 8 and 9)

Figure 17 shows the effect of break location on the consequences from a LOCA. No feedwater or ECCS was assumed in these analyses so that the core would dry and begin to heat up. This was so that the time to dryout could be compared as a figure of merit to judge the severity of the break location. The time to dryout also gages the time available for operator action to restore safeguards equipment if required. A much longer time was required for core uncovery for a break in the top of the reactor vessel than for an equivalent break in a hot leg. The hot leg break was located close to the reactor vessel in the horizontal section. Much of the liquid in the vertical hot leg sections was lost out the break before it could reach the core. The Oconee design with once through steam generators includes hot leg piping that extends 43 feet above the core. A break at the top of the reactor vessel permits this liquid to flow into the core and extends the time to core uncovery.

The hot legs at Davis-Besse extend even further above the core and should provide even more water for core cooling than Oconee. These results indicate that PRA conclusions for a hot leg break close to the reactor vessel would be conservative if applied to a potential CRDM break at Oconee or Davis-Besse.

Upper Head Break Cases for Westinghouse and CE Designs

So that the effect of break location could be assessed for other PWR designs, we preformed the following additional RELAP5 analyses. These analyses utilized an input deck prepared by INEL for Seabrook (Ref. 2) and for ANO-2 (Ref. 3). No feedwater or ECCS was assumed in these analyses so that the core would dry out and begin to heat up. This was so that the time to dryout could be compared as a figure of merit to judge the severity of the break location.

Case 10: The rupture of a single CRDM thimble without AFW, HPI or steam dump capability to the condenser for a four loop Westinghouse plant (Seabrook). See Figure 18.

Case 11: A break in a hot leg equivalent to the size to a single CRDM thimble without AFW, HPI or steam dump capability to the condenser for a four loop Westinghouse plant (Seabrook). See Figure 18.

Case 12: A break in a cold leg equivalent in size to a single CRDM thimble without AFW, HPI or steam dump capability to the condenser for a four loop Westinghouse plant (Seabrook). See Figure 18.

Case 13: The rupture of a single CRDM thimble without AFW, HPI or steam dump capability to the condenser at a Combustion Engineering plant (ANO-2). See Figure 19.

Case 14: A break in a hot leg equivalent to the size to a single CRDM thimble without AFW, HPI or steam dump capability to the condenser for a four loop Westinghouse plant (ANO-2). See Figure 19.

The conclusions for Seabrook in comparing a CRDM break and a hot leg break are the opposite of those for Oconee. For Seabrook a CRDM thimble break at the top of the vessel caused core dryout to begin earlier than for a hot leg break (Fig. 18). The reason for the difference in results lies in the design of the reactor vessel internals. In both designs the upper head is separated from the upper plenum by a plenum cover plate. In the Oconee design the plenum cover plate is porous providing an open path for coolant to flow up through the control rod guide tubes and down through the plenum cover plate into the upper plenum. The upper head at Oconee is heated to the temperature of the core outlet during operation. For the Seabrook design, flow within the upper head is restricted. During operation leakage flow is permitted from the reactor vessel downcomer into the upper head. Flow then passes downward through the control rod guide tubes to the top of the core. During operation the upper head at Seabrook is approximately at the core inlet temperature.

For a postulated CRDM break at Oconee flow from the core to the break is primarily though the upper plenum where the large flow area permits steam/water separation so that steam can flow out the break and water can remain above the core. For a CRDM thimble break at Seabrook, flow from the core to the break is primarily through the control rod guide tubes which have a small hydraulic diameter and permit little steam/water separation. Water from the core is sucked up to the break through the control rod guide tubes in a process similar to drinking through straw. Note in Figure 18 that for Seabrook that the consequences of a CRDM rupture are more advantageous than for a cold leg break. The results lie between the hot leg and cold leg break cases so the CRDM break is thus still bounded by the FSAR analysis.

The upper head flow design at ANO-2 is less restrictive than that for Seabrook but more restricted than Oconee. During operation the upper head temperature is between that of the core inlet and that of the core outlet. The time for the beginning of core dryout was found to be approximately the same whether the break was in the upper head or in a hot leg. See Figure 19.

In investigating the effect of break location, use of hot leg break analyses appear to be conservative in describing a CRDM thimble break for Oconee and slightly conservative for ANO2. Use of cold leg break analyses to describe a CRDM thimble break at Seabrook would be conservative.

3. CONCLUSIONS

For postulated breaks in the upper head of a PWR reactor vessel the consequence are similar to piping breaks analyzed in plant FSARs as part of the design basis. No new phenomena or unexpected results were obtained. Plants designed by all three reactor vendors were analyzed.

The consequences of failure to scram were evaluated and found to be minimal. This is because of the negative reactivity produced by core voiding and later by boric acid addition.

REFERENCES

1. K. S. Quick, “Oconee Unit 1 Pressurized Water Reactor RELAP5/Mod3 Input Model,” Idaho National Engineering Laboratory, August 1994.

2. J. R. Larson and J. D. Burtt, “Seabrook Pressurized Water Reactor RELAP5/Mod3 Model,” September 1994.

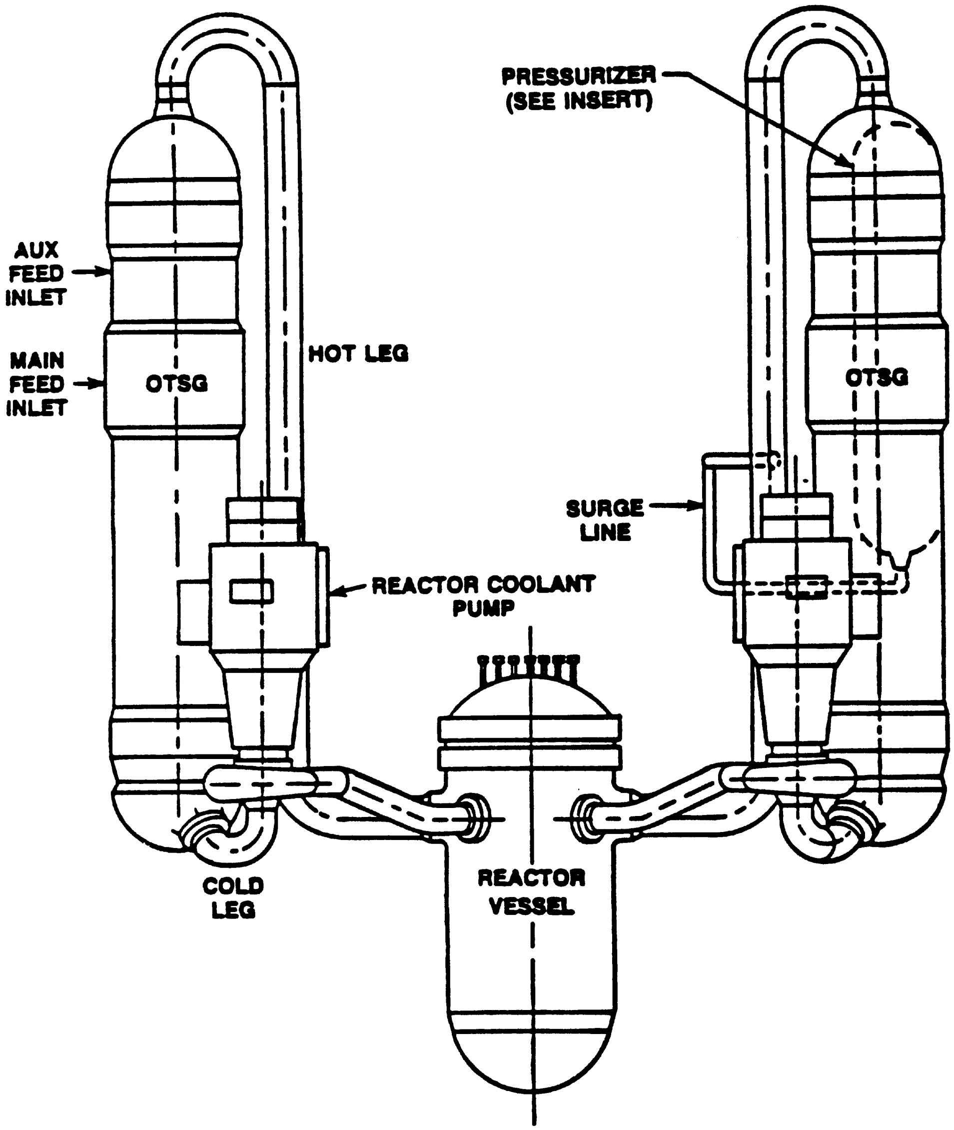


Figure 1 **Babcock & Wilcox 177 "Raised-Loop" PWR NSSS**

3. W. K. Terry, “Arkansas Nuclear One, Unit 2 RELAP5/Mod3.2 Input Model,” June 1996.

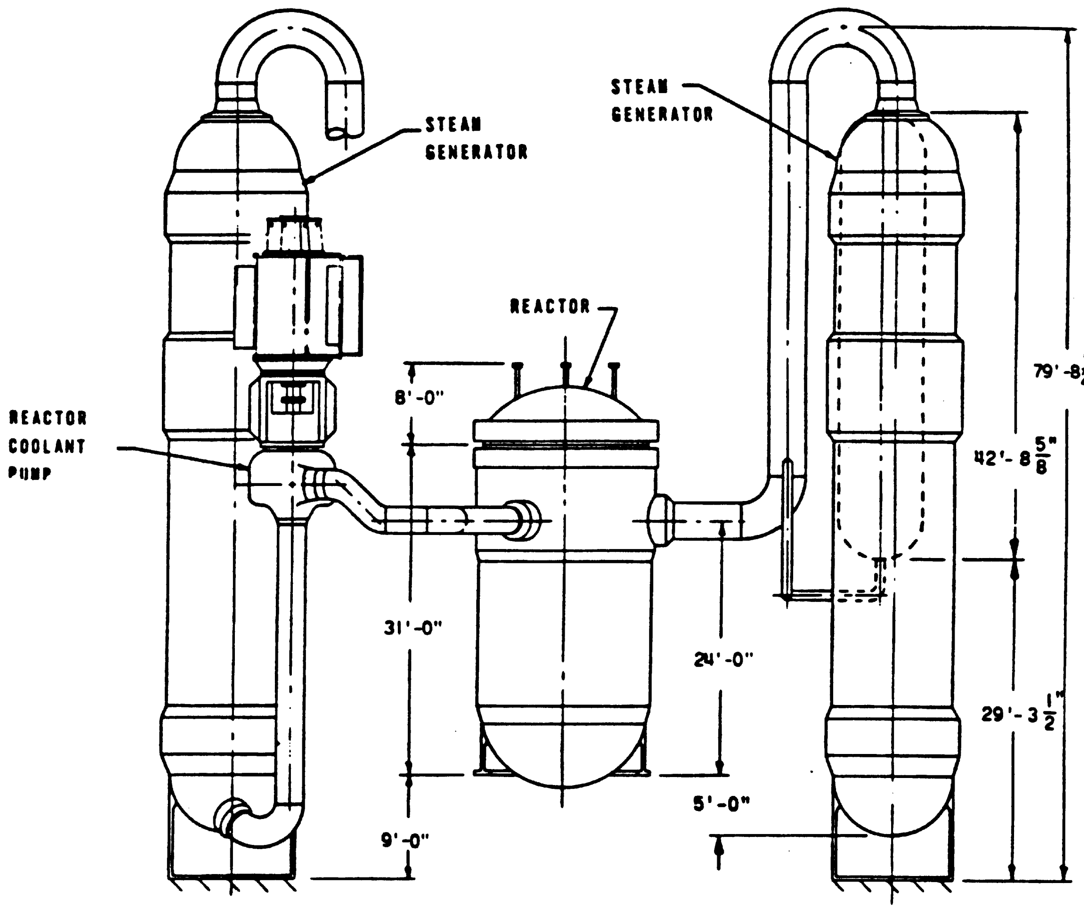


Figure 2 **Babcock & Wilcox 177 "Lowered-Loop" PWR NSSS**

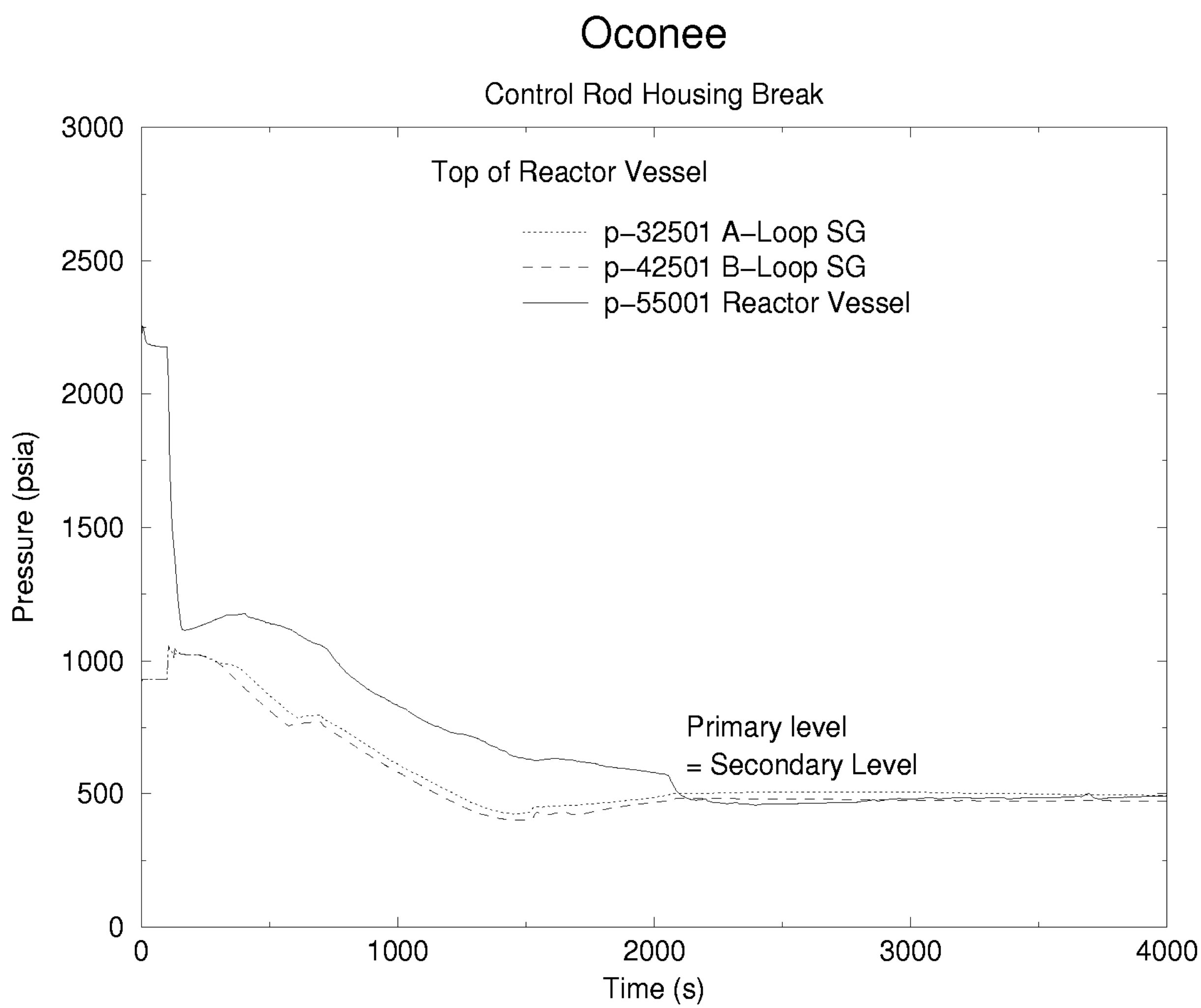

Oconee
Control Rod Housing Break
Top of Reactor Vessel
p–32501 A–Loop SG
p–42501 B–Loop SG
p–55001 Reactor Vessel
Primary level
= Secondary Level
Pressure (psia)
Time (s)
0
500
1000
1500
2000
2500
3000
0
1000
2000
3000
4000

Figure 3

Oconee

Control Rod Housing Break

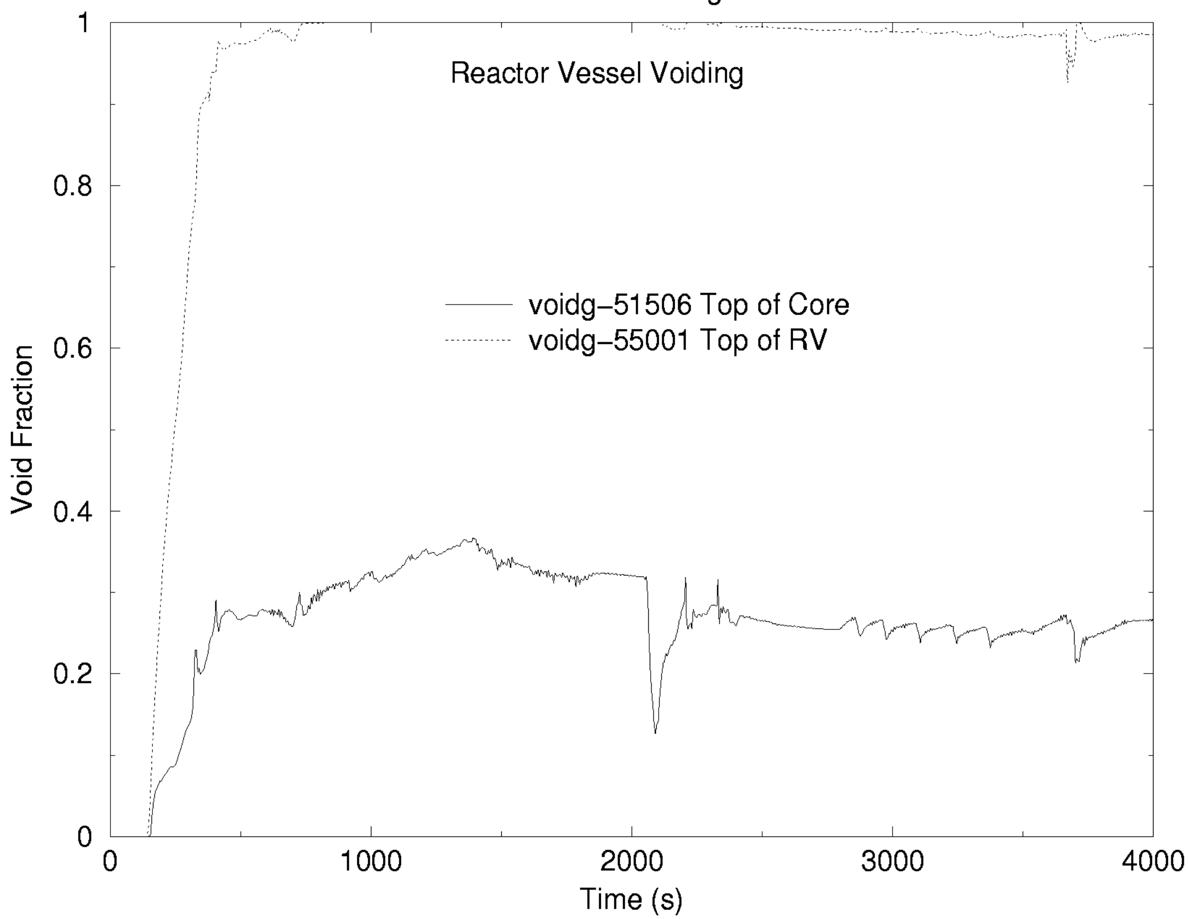


Figure 4

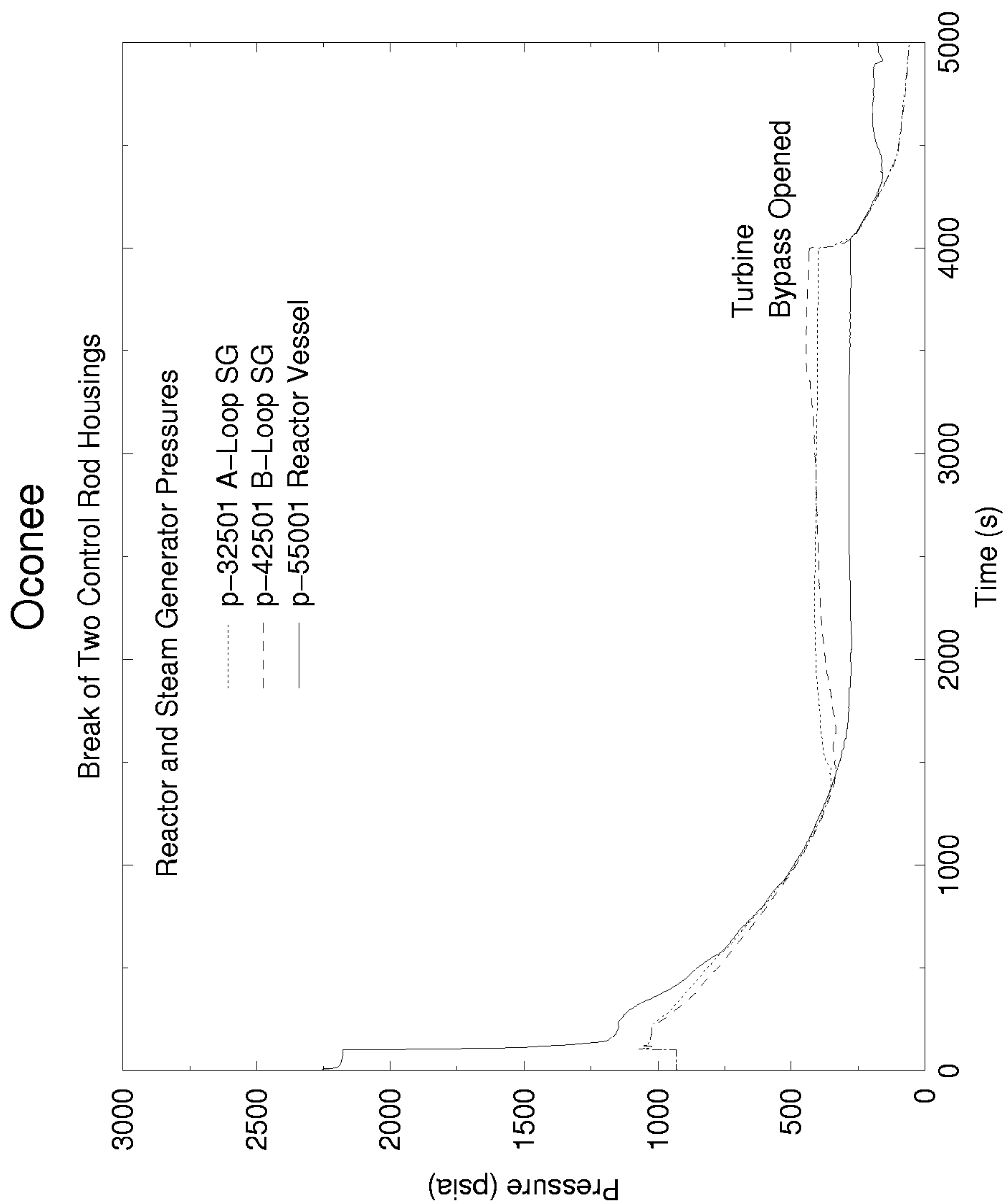

Oconee
Break of Two Control Rod Housings
Reactor and Steam Generator Pressures
p–32501 A–Loop SG
p–42501 B–Loop SG
p–55001 Reactor Vessel
Turbine
Bypass Opened
Pressure (psia)
Time (s)
0
500
1000
1500
2000
2500
3000
0
1000
2000
3000
4000
5000

Figure 5

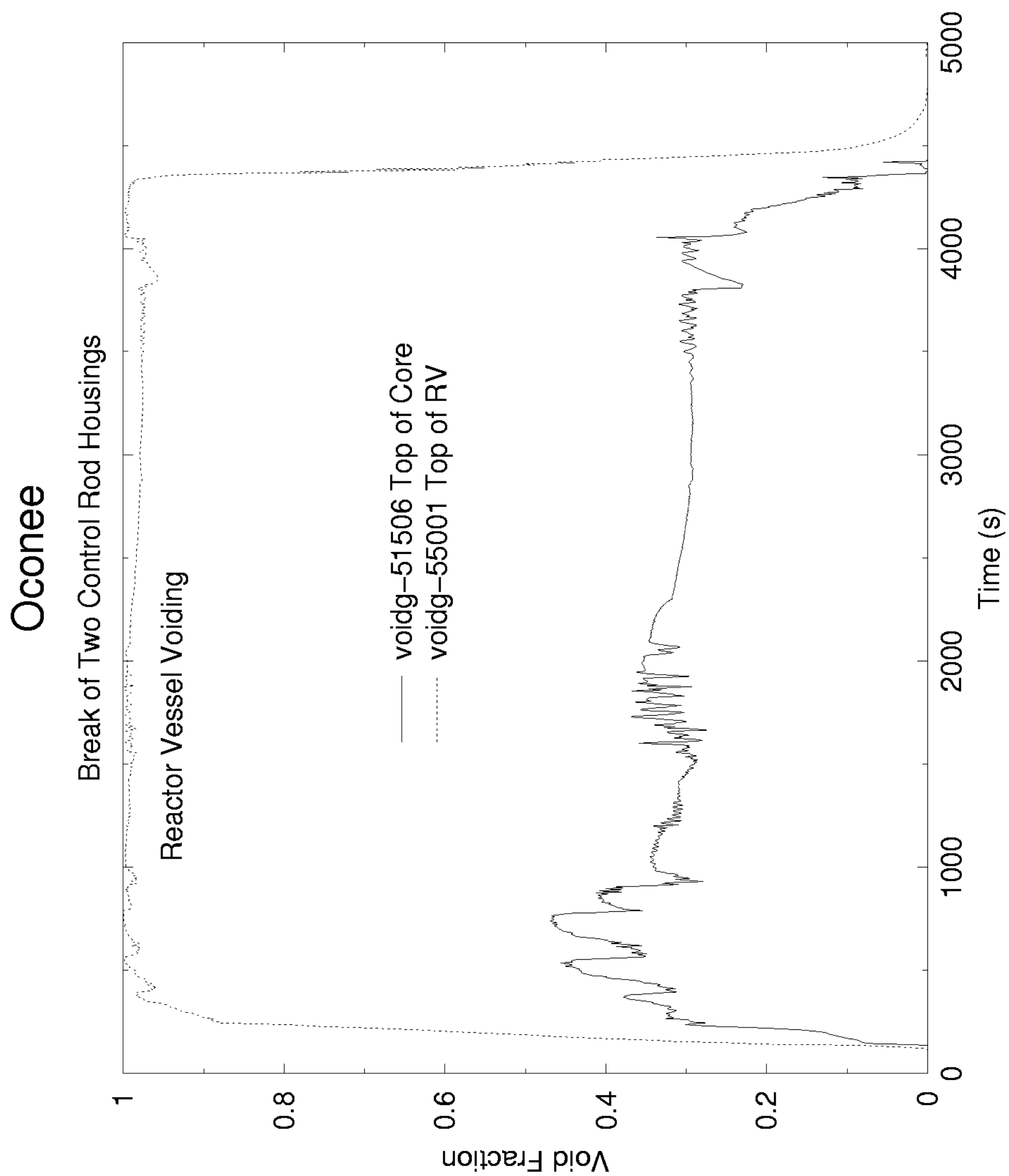

Oconee
Break of Two Control Rod Housings
Reactor Vessel Voiding
voidg–51506 Top of Core
voidg–55001 Top of RV
Void Fraction
0
0.2
0.4
0.6
0.8
1
Time (s)
0
1000
2000
3000
4000
5000

Figure 6

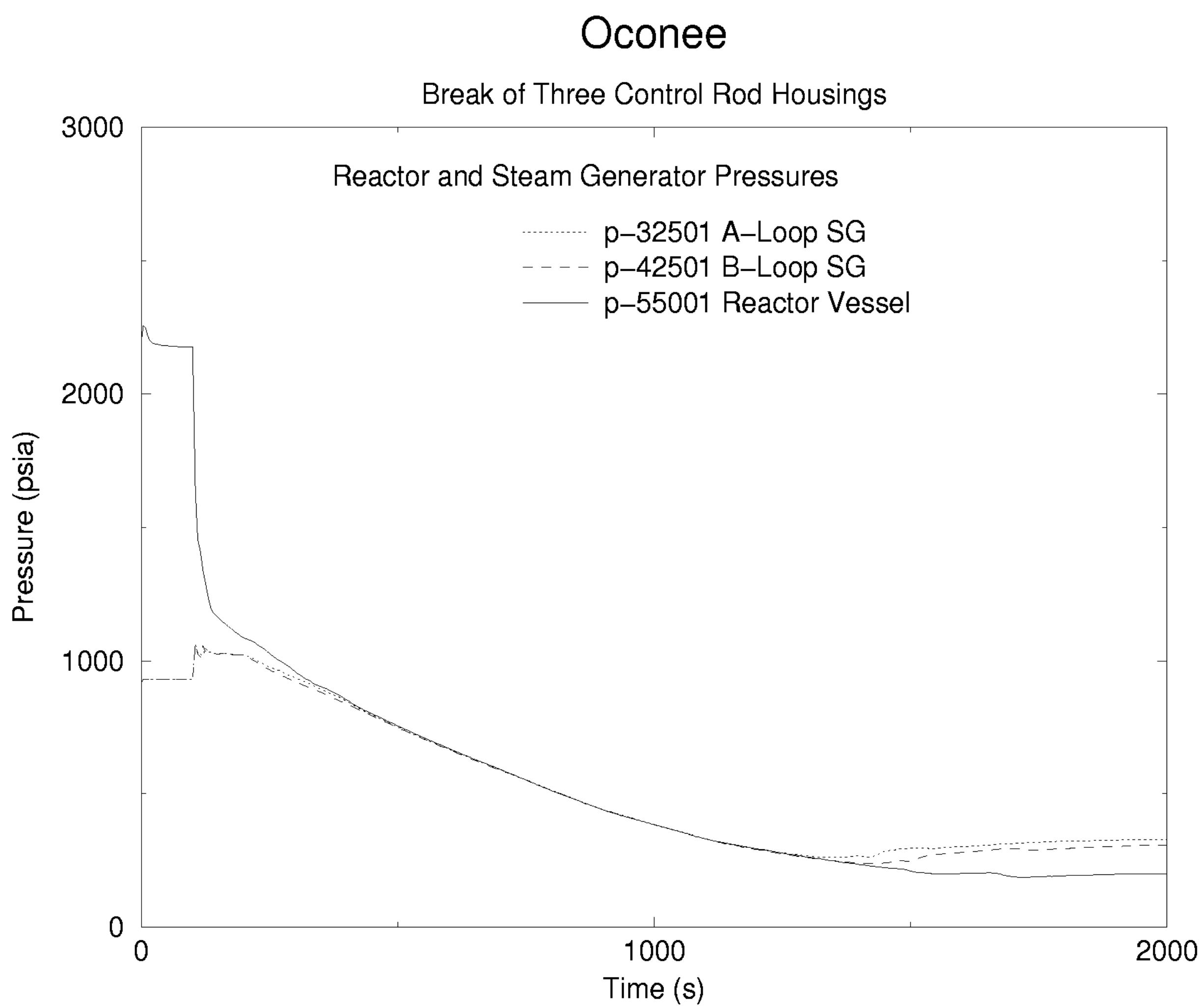

Oconee
Break of Three Control Rod Housings
Reactor and Steam Generator Pressures
p−32501 A−Loop SG
p−42501 B−Loop SG
p−55001 Reactor Vessel
Pressure (psia)
Time (s)
0
1000
2000
3000
0
1000
2000

Figure 7

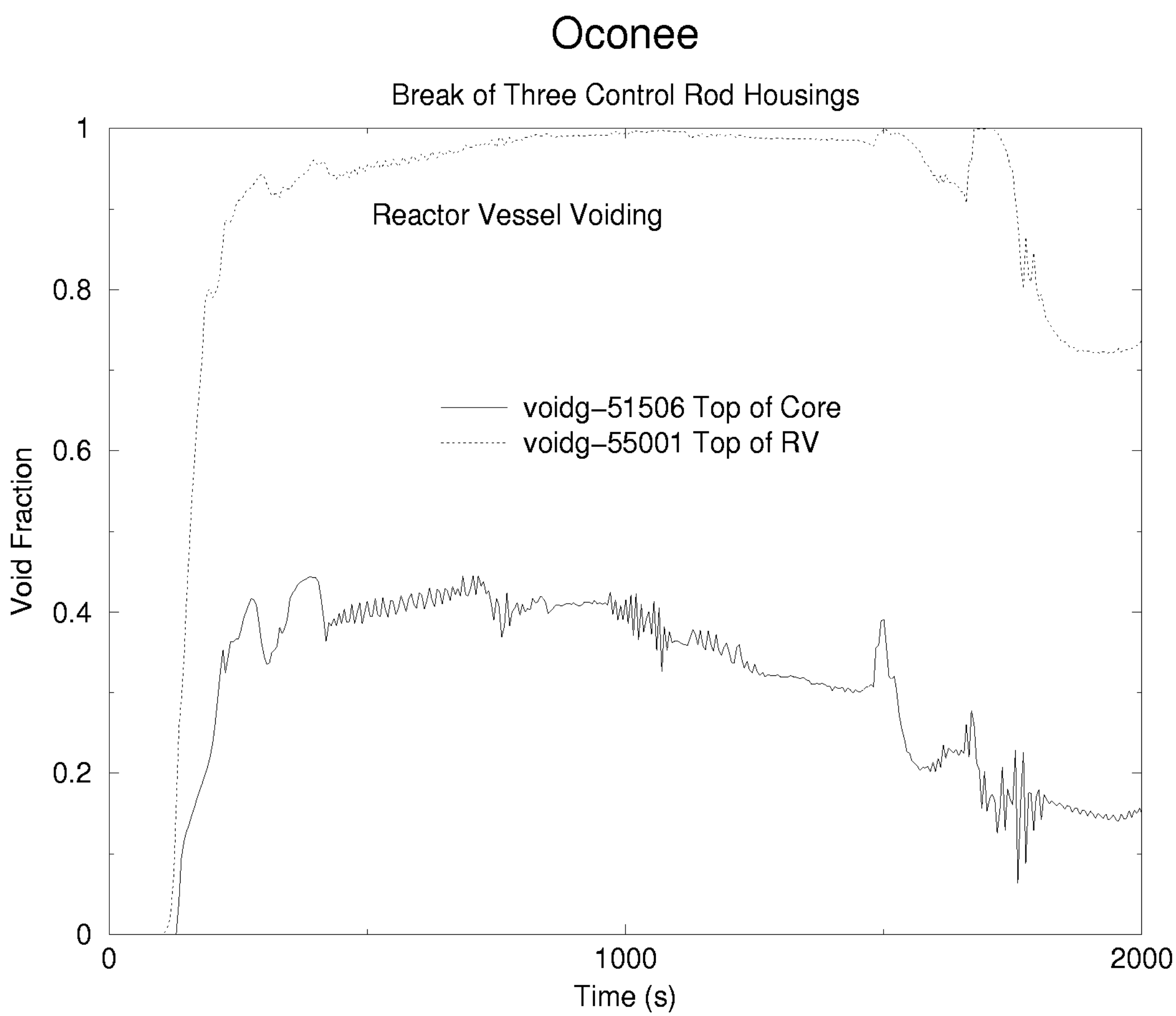

Oconee
Break of Three Control Rod Housings
Reactor Vessel Voiding
voidg–51506 Top of Core
voidg–55001 Top of RV
Void Fraction
0
0.2
0.4
0.6
0.8
1
0
1000
2000
Time (s)

Figure 8

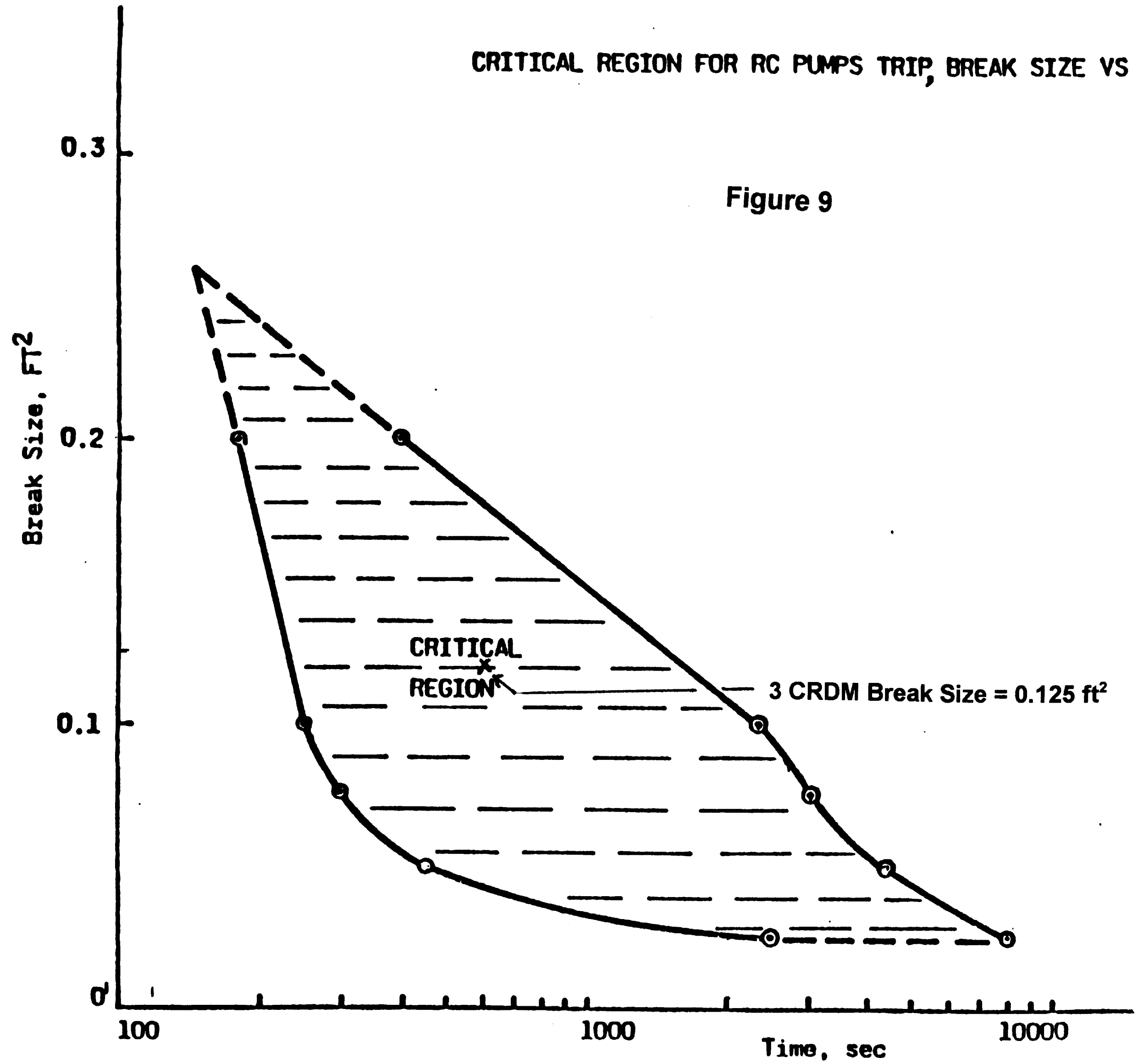


Figure 9

CRITICAL REGION FOR RC PUMPS TRIP, BREAK SIZE VS TIME

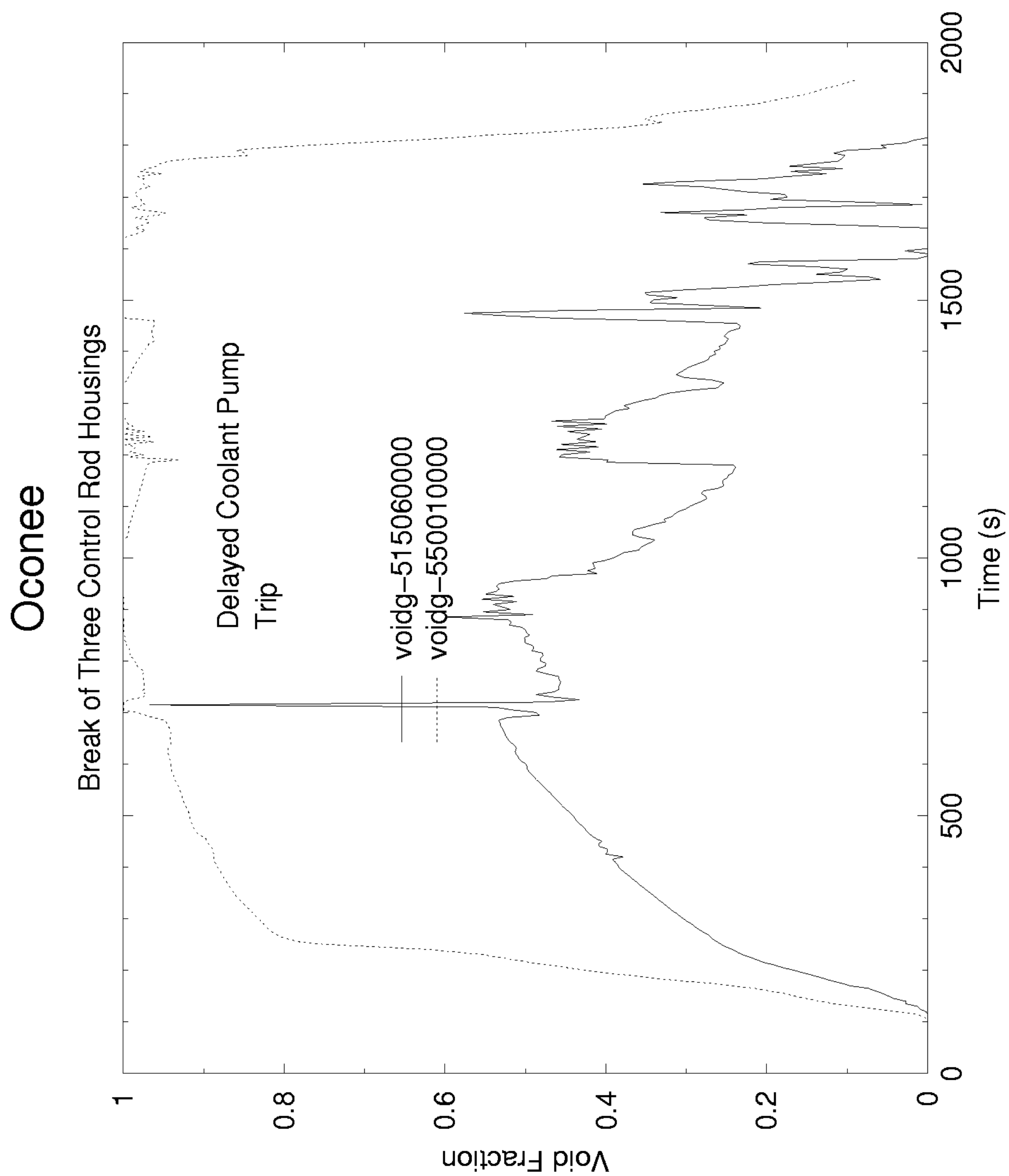

Oconee
Break of Three Control Rod Housings
Delayed Coolant Pump
Trip
voidg−515060000
voidg−550010000
Void Fraction
Time (s)
0
0.2
0.4
0.6
0.8
1
0
500
1000
1500
2000

Figure 10

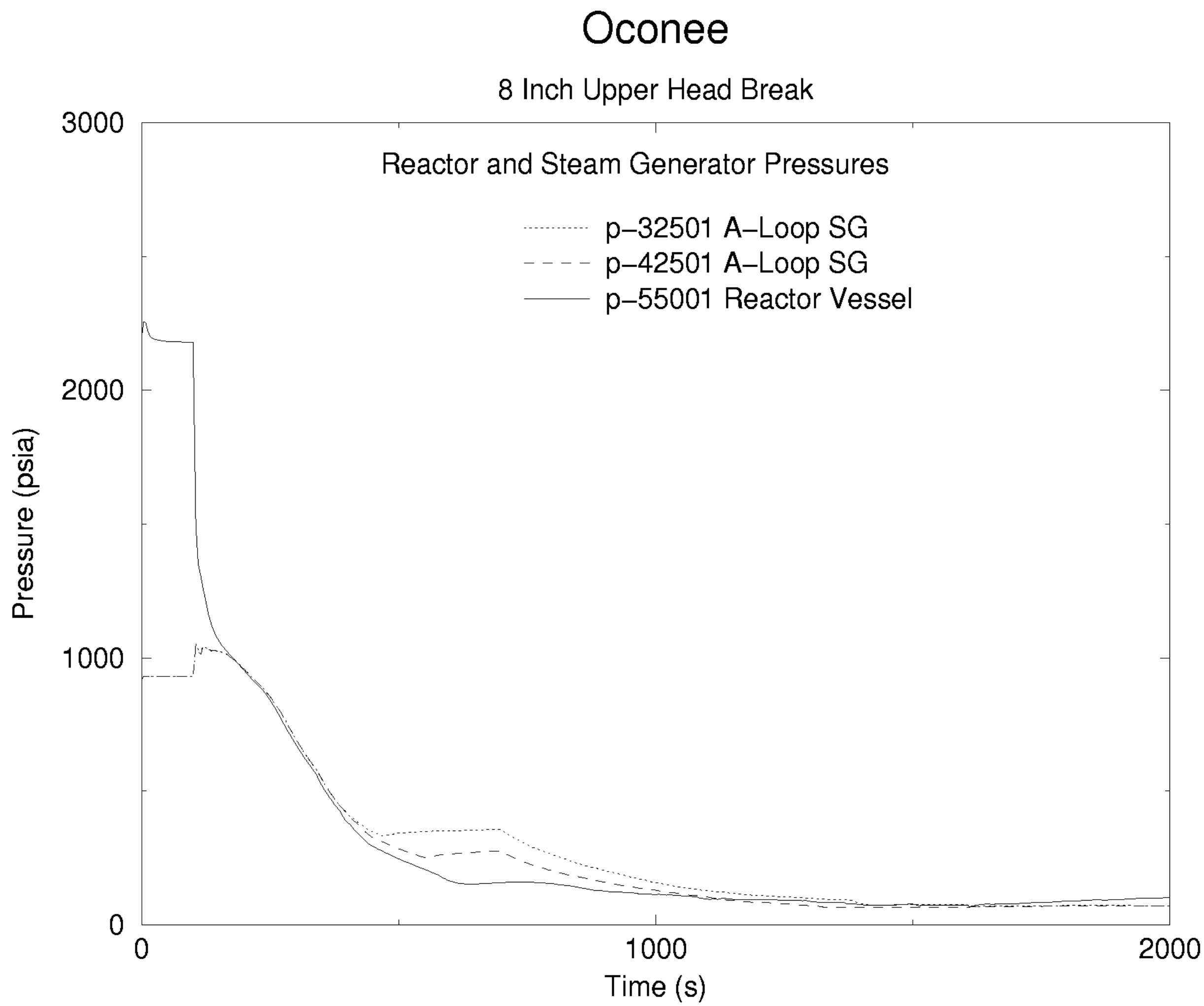

Oconee
8 Inch Upper Head Break
Reactor and Steam Generator Pressures
p–32501 A–Loop SG
p–42501 A–Loop SG
p–55001 Reactor Vessel
Pressure (psia)
0
1000
2000
3000
Time (s)
0
1000
2000

Figure 11

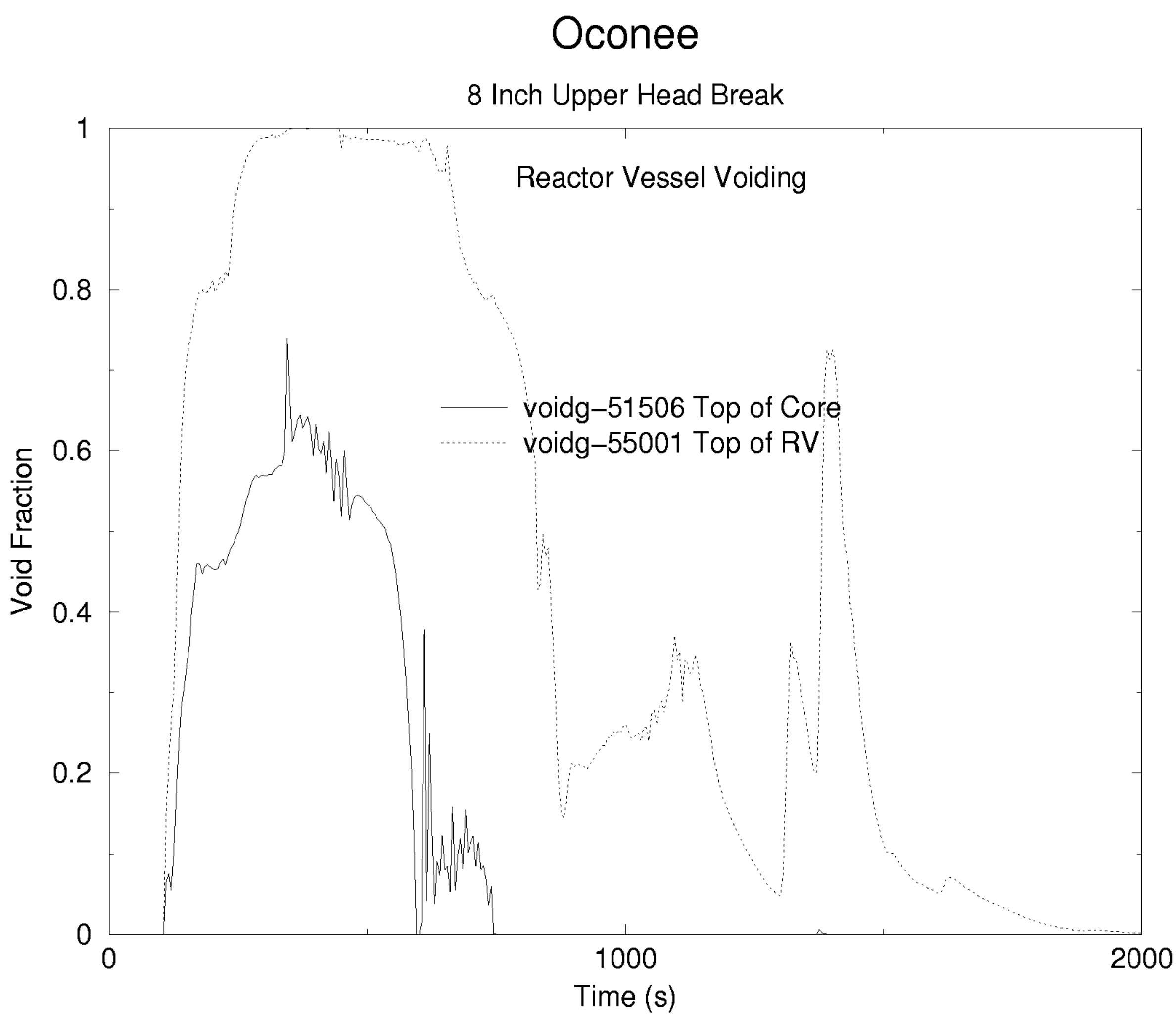

Oconee
8 Inch Upper Head Break
Reactor Vessel Voiding
voidg−51506 Top of Core
voidg−55001 Top of RV
Void Fraction
0
0.2
0.4
0.6
0.8
1
0
1000
2000
Time (s)

Figure 12

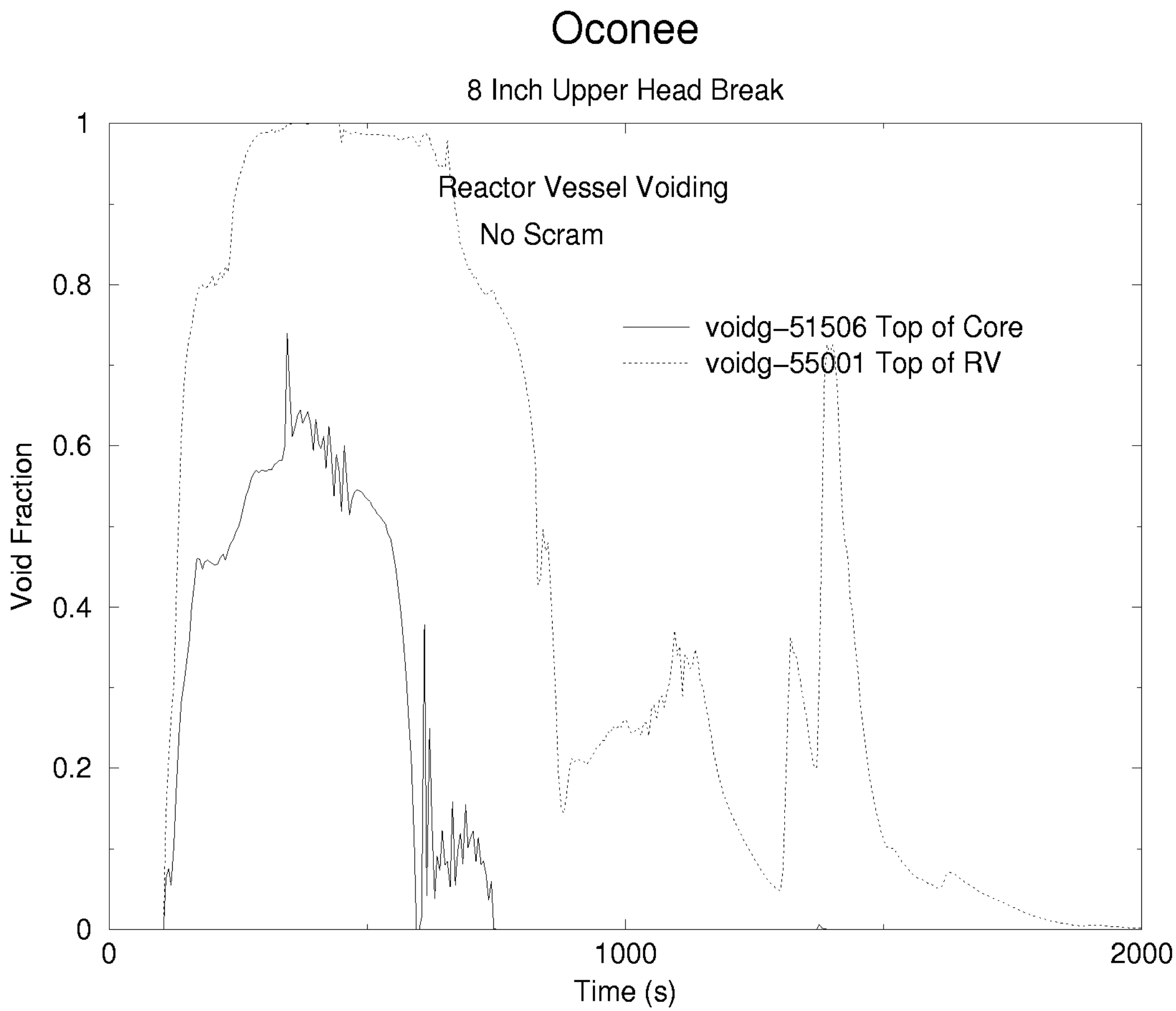

Oconee
8 Inch Upper Head Break
Reactor Vessel Voiding
No Scram
voidg−51506 Top of Core
voidg−55001 Top of RV
Void Fraction
Time (s)
0
0.2
0.4
0.6
0.8
1
0
1000
2000

Figure 13

Oconee

8 Inch Upper Head Break

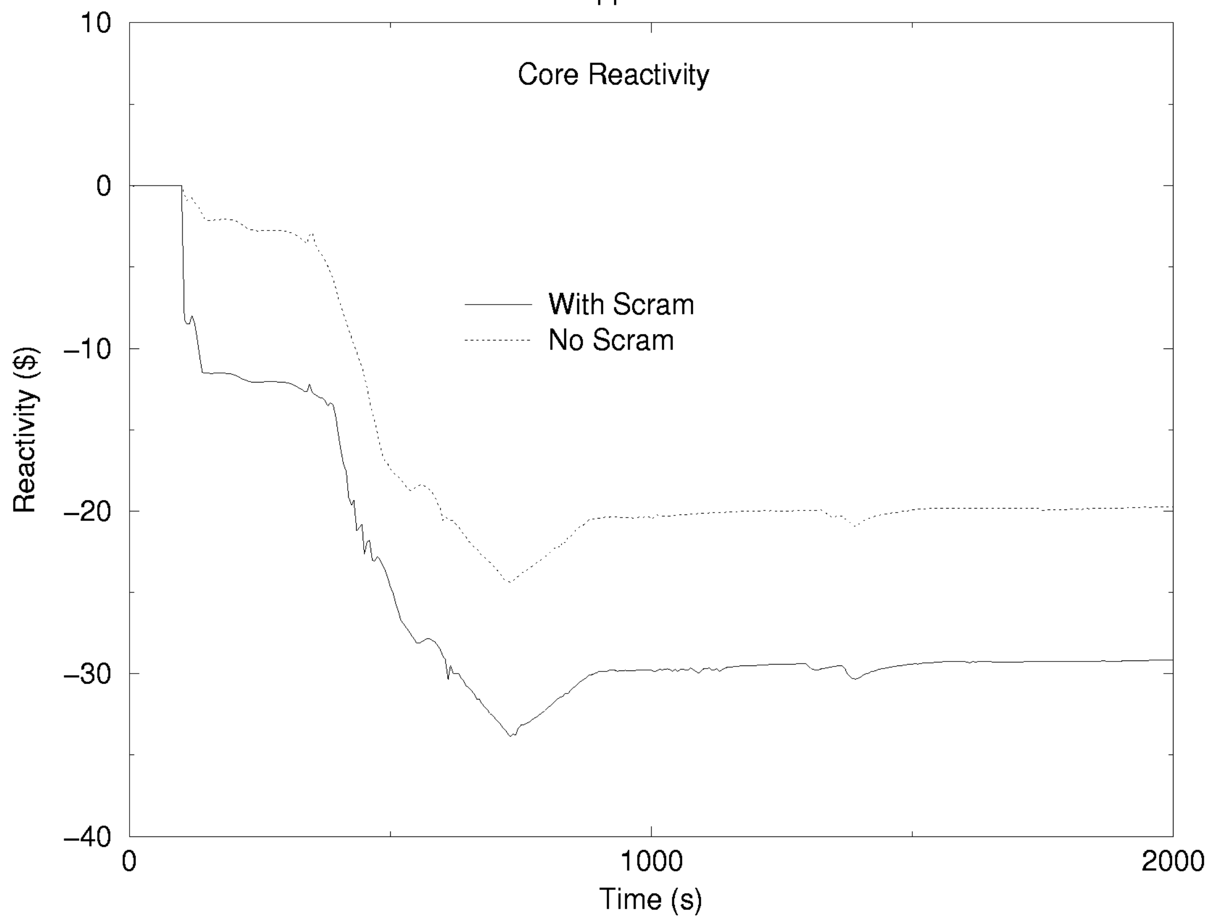


Figure 14

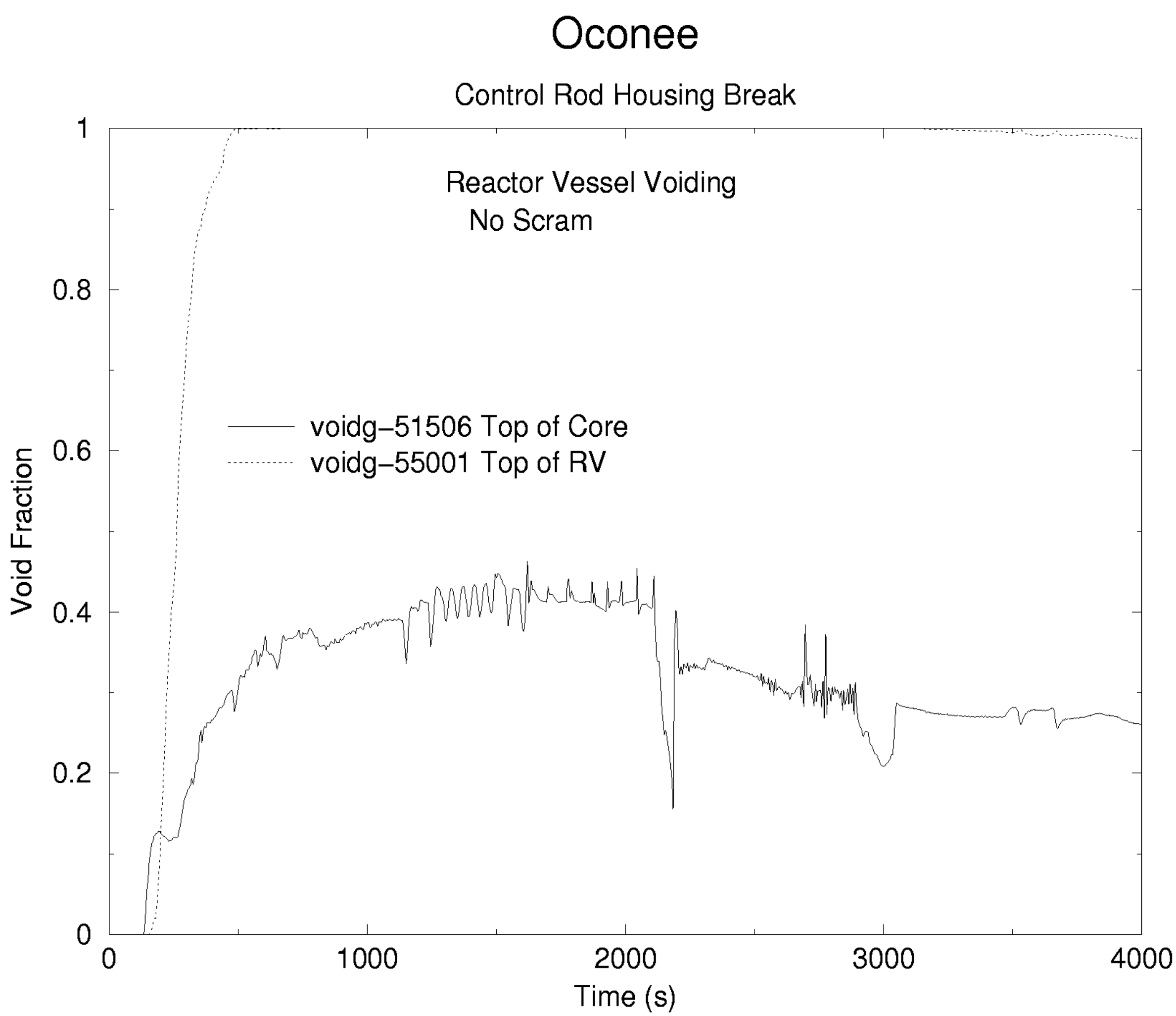
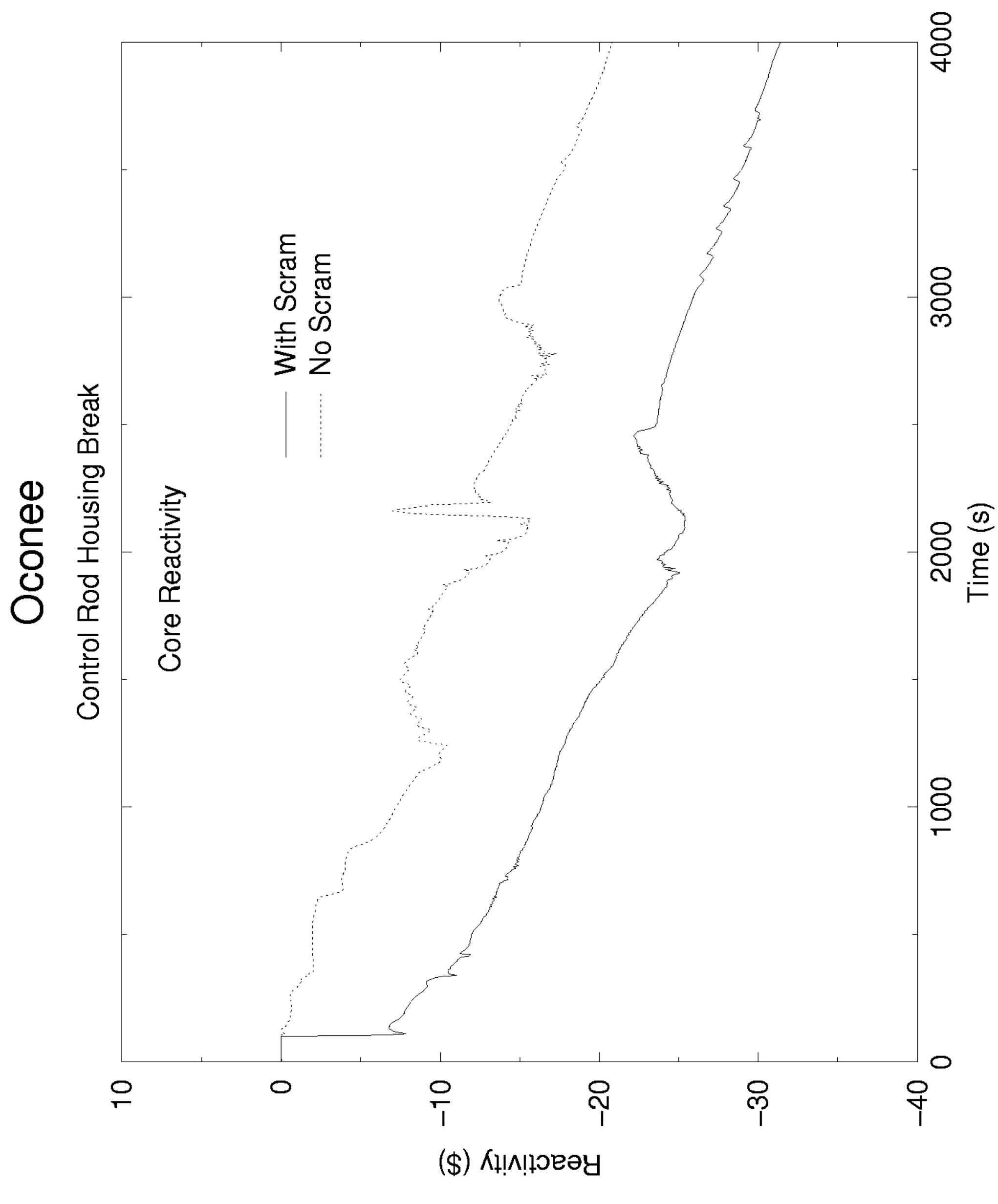

Oconee
Control Rod Housing Break
Reactor Vessel Voiding
No Scram
voidg–51506 Top of Core
voidg–55001 Top of RV
Void Fraction
0
0.2
0.4
0.6
0.8
1
0
1000
2000
3000
4000
Time (s)

Figure 15



Oconee
Control Rod Housing Break
Core Reactivity
With Scram
No Scram
Reactivity ($)
Time (s)
10
0
−10
−20
−30
−40
1000
2000
3000
4000

Figure 16

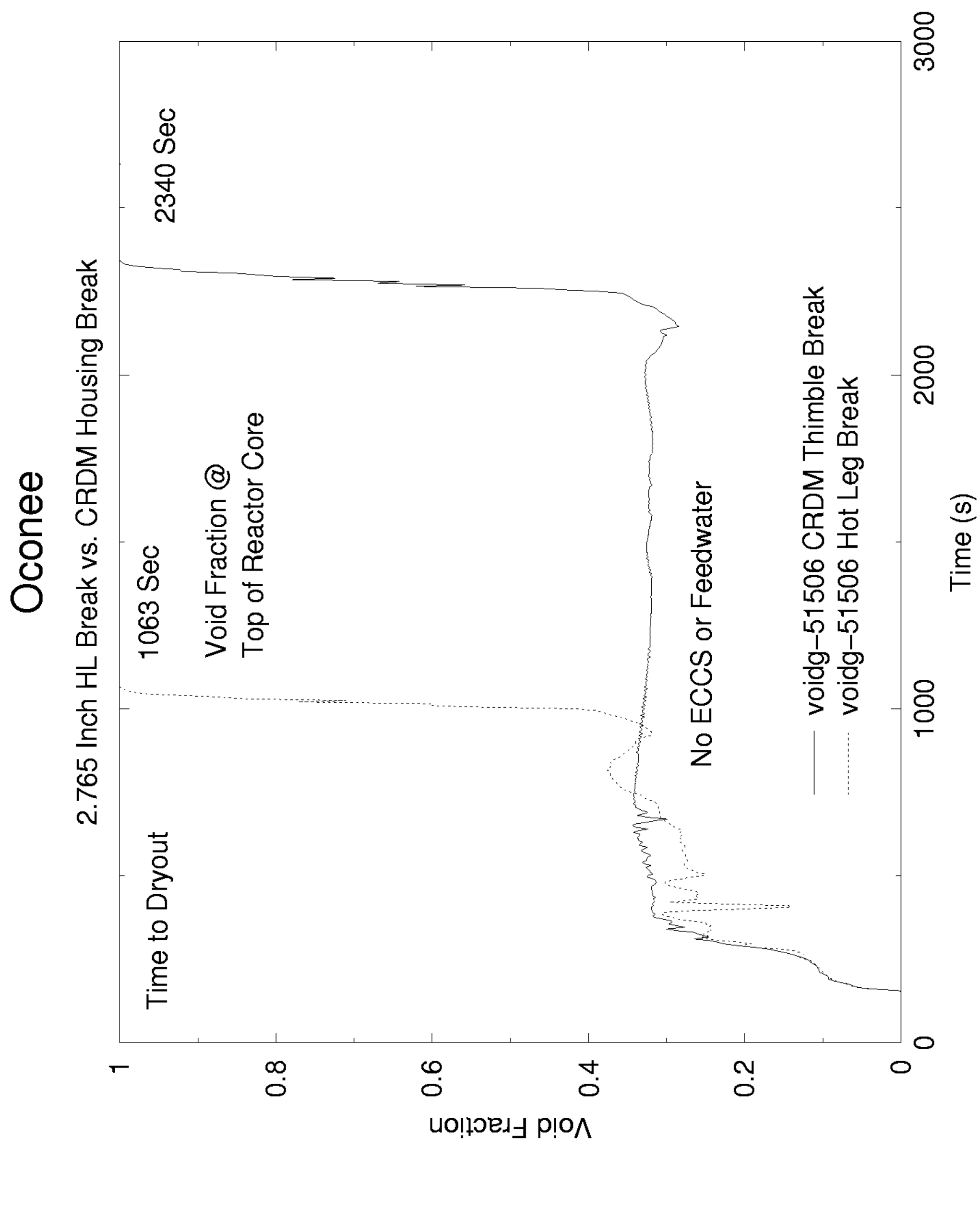

Oconee
2.765 Inch HL Break vs. CRDM Housing Break
Time to Dryout
1063 Sec
Void Fraction @
Top of Reactor Core
2340 Sec
No ECCS or Feedwater
voidg–51506 CRDM Thimble Break
voidg–51506 Hot Leg Break
Void Fraction
Time (s)
0
0.2
0.4
0.6
0.8
1
0
1000
2000
3000

Figure 17

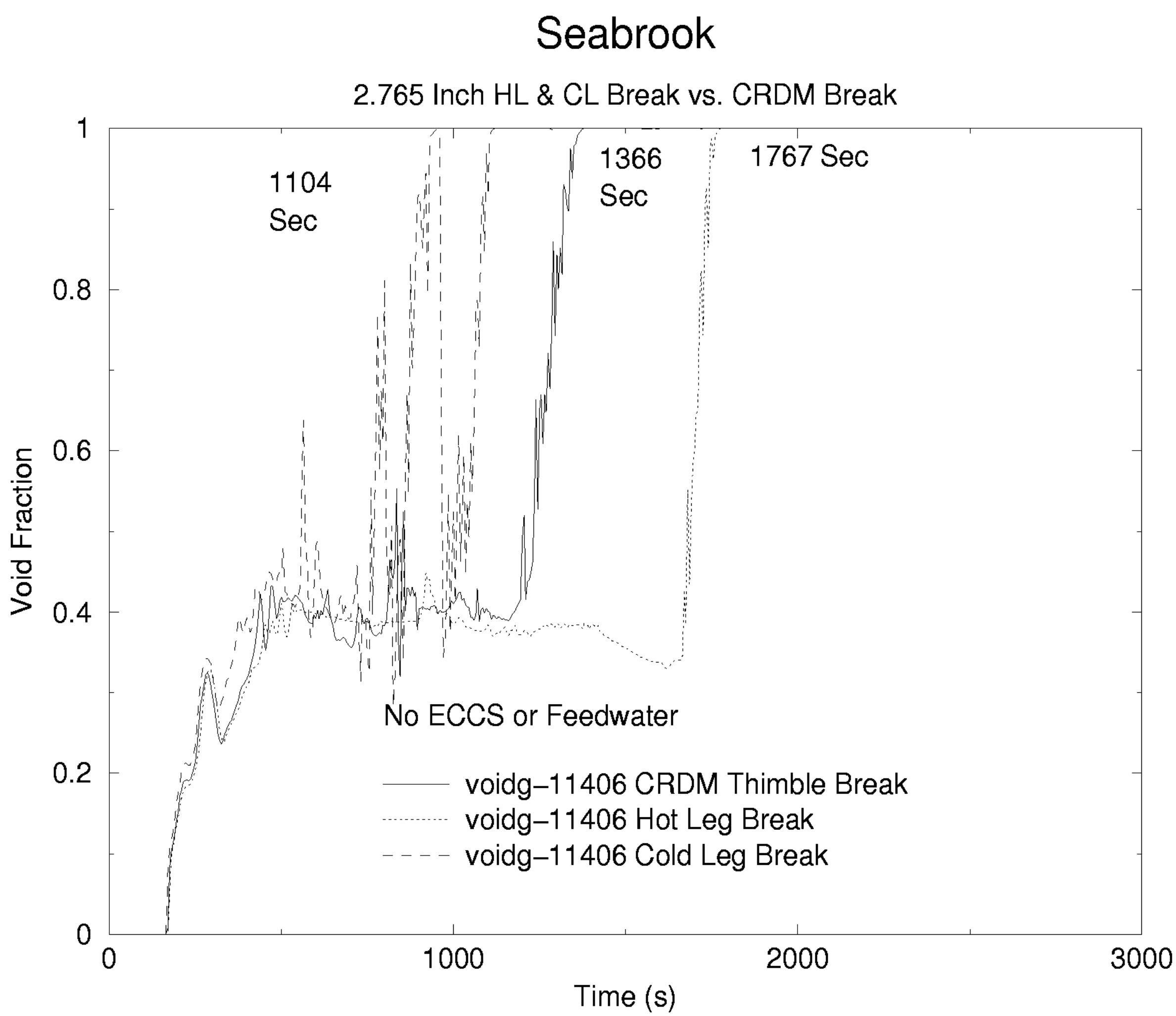

Seabrook
2.765 Inch HL & CL Break vs. CRDM Break
1104
Sec
1366
Sec
1767 Sec
No ECCS or Feedwater
voidg−11406 CRDM Thimble Break
voidg−11406 Hot Leg Break
voidg−11406 Cold Leg Break
Void Fraction
Time (s)
0
0.2
0.4
0.6
0.8
1
0
1000
2000
3000

Figure 18

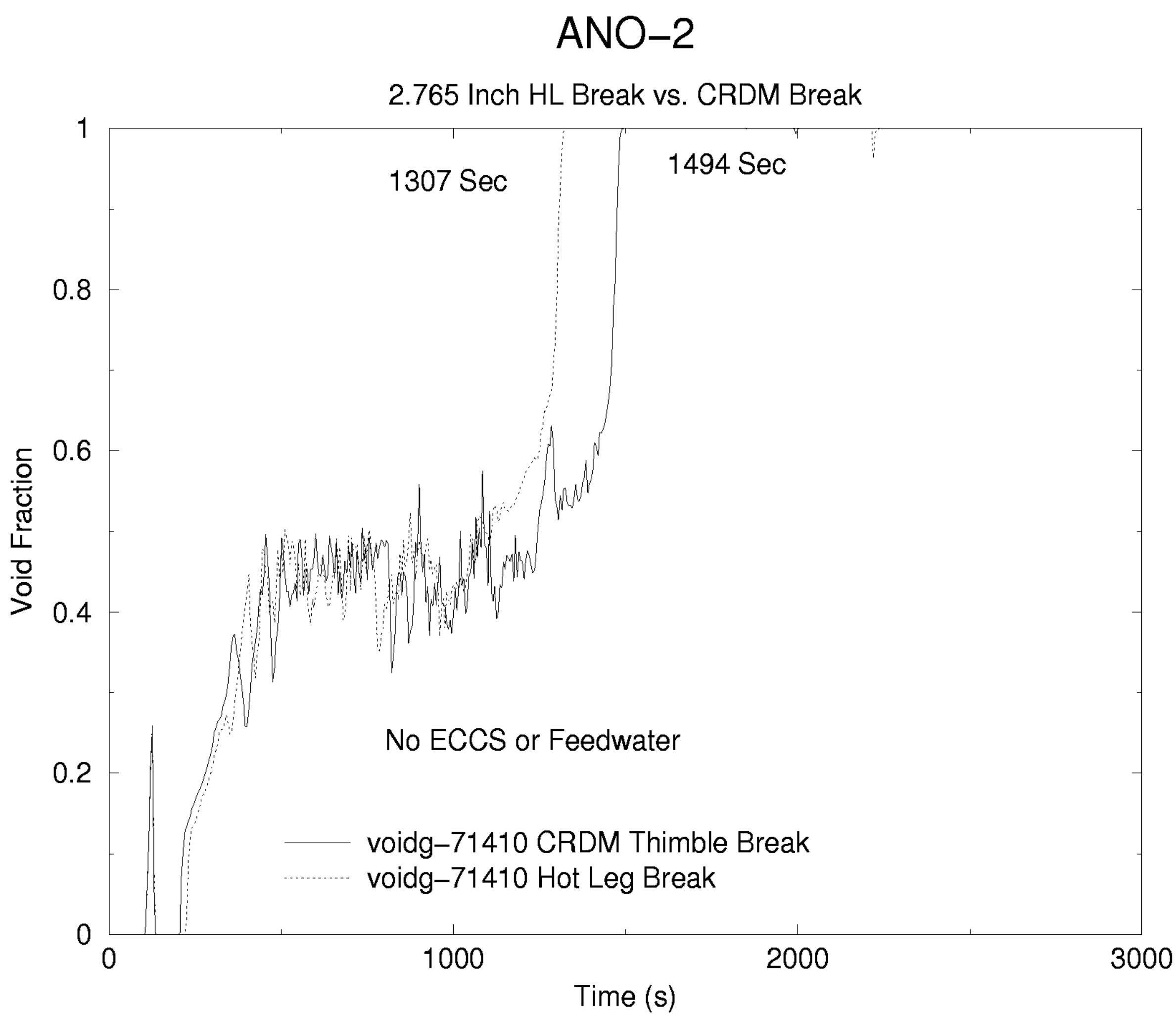

ANO−2
2.765 Inch HL Break vs. CRDM Break
1307 Sec
1494 Sec
No ECCS or Feedwater
voidg−71410 CRDM Thimble Break
voidg−71410 Hot Leg Break
Void Fraction
Time (s)
0
0.2
0.4
0.6
0.8
1
0
1000
2000
3000

Figure 19