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Mr. Harold R. Denton
Director, Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: Waterford 3 SES
Docket No. 50-382
Clarification of Transient Analyses with Potential for Fuel Damage and Feedwater Line Break Confirmatory Items

ENCLOSURES: (1) Evaluation of the Effect of Break Size Upon the Consequences of Steam Line Breaks with Concurrent Loss of Offsite Power - Waterford 3

(2) Analysis of Feedwater Line Breaks

Dear Mr. Denton,

Supplement Number 1 to the Waterford 3 SER contains a confirmatory issue concerning clarification of transient analyses with potential for fuel damage. (Section 15.3.1). The concern of this item was that we had not addressed the consequences of a small steam line break concurrent with a loss of offsite power. Enclosure (1) describes this analysis and demonstrates that no steam line break with concurrent loss of offsite power yields consequences more adverse than the limiting consequences reported in the FSAR.

Supplement 1 to the SER also contains a confirmatory issue requesting confirmation that small feedwater line breaks combined with the limiting single failure and with offsite power available do not exceed the 110% design pressure criteria. This analysis is described in enclosure (2). The NRC staff reviewer also requested a probability analysis for feedwater line breaks which is included in Section I of enclosure (2).

It is hoped that this evaluation will alleviate the staff's concern on these confirmatory issues. If you have any questions or require further information, please feel free to contact either myself or R. W. Prados.

Very truly yours,

L. V. Maurin

B001

LVM/DEB:keh

cc: J. Wilson, W. M. Stevenson, E. L. Blake, J. Guttman

Enclosure (1)

Evaluation of the Effect of Break Size Upon the Consequences of Steam Line Breaks with Concurrent Loss of Offsite Power Waterford Unit 3

Introduction

Supplement No. 1 to the Safety Evaluation Report for Waterford Unit 3 requires that the consequences of small steam line breaks (SLBs) with concurrent loss of offsite power (LOP) be addressed in order to demonstrate that the SLB analyses presented in the FSAR are the limiting cases (p. 15-1 of Ref. 1). The effect of break size on the consequences of SLBs with concurrent LOP has been evaluated. No SLB, of any break size, with concurrent LOP yields consequences more adverse than the limiting consequences for SLBs reported in the FSAR.

The consequence of concern for SLB's is offsite dose. The contribution to offsite dose that can be affected by break size is fuel failure. The possible break sizes of concern range from zero up to the maximum flow area for outside containment breaks, the area of the flow venturis. Inside containment breaks result in much less offsite dose. Degradation in fuel performance (fuel failure) can occur during SLB initiated events either during the portion of the transient prior to and during reactor trip (henceforth referred to as the pre-trip portion) or during the post-trip return-to-criticality, or approach-to-criticality, portion of the transient (henceforth referred to as the post-trip portion).

Pre-trip fuel failure

For cases initiated from a power operating limit, and where loss of offsite power is assumed to occur concurrent with the SLB, there will be a CPC trip on projected DNBR within the first 0.6 seconds of the initiation of the event. The power operating limit is determined such that the CPC trip will prevent the transient minimum DNBR due to a loss of flow (LOP) from being less than 1.19. The only significant additional effect of the SLB, over that of the LOF, up to the time of minimum transient "pre-trip" DNBR will be a reduction in RCS pressure. Therefore for SLBs with concurrent loss of offsite power the transient minimum "pre-trip" DNBR will be only incrementally lower than 1.19. Further, the rate of reduction of RCS pressure due to the SLB will be maximum for the maximum break area.

For the Waterford Unit 3 NSSS a conservative evaluation of the decrease in DNBR due to the maximum area, outside containment SLB yields a minimum transient DNBR greater than 1.17. The resultant calculated fuel failure would be less than 0.05%. The consequent 2 hour exclusion area boundary thyroid dose for the event (including secondary releases) would not be greater than 10 rem. This is less than the design basis radiological consequences of the limiting main steam line break of the FSAR. Thus no SLB, of any break size, with concurrent LOP yields consequences due to pre-trip degradation in fuel performance which are more adverse than the limiting consequences for SLBs reported in the FSAR.

Post-trip fuel failure

Degradation in fuel performance during the post-trip portion of SLB initiated transients can only occur if there is a return-to-power (R-t-P). Therefore the primary consideration for maximizing post-trip degradation in fuel performance is to select those parameters and conditions which will maximize R-t-P. The magnitude of R-t-P is primarily determined by the value of the maximum post-

trip reactivity, the timing of this reactivity, and the duration of the reactivity peak. The timing of the maximum post-trip reactivity has an important effect on the post-trip R-t-P, the same reactivity will produce less R-t-P later in a transient.

The maximum R-t-P results from the maximum break area. A smaller break delays the time of maximum post-trip reactivity and therefore decreases the magnitude of the R-t-P generated. Additionally, for the slower transients due to smaller breaks, more time is available for the transfer of heat from the metal of the RCS walls and structure and of decay heat from the fuel to the coolant. This reduces the moderator cooldown and consequently the maximum post-trip reactivity.

Therefore the minimum "post-trip" DNBR occurs for the maximum break area. The SLB with maximum break area and with concurrent LOP has been presented in the FSAR. Thus no SLB, of any break size, yields consequences due to post-trip degradation in fuel performance which are more adverse than those reported in the FSAR.

Conclusion

SLBs with concurrent LOP have been evaluated for the effect of break size upon event consequences. The conclusion of this evaluation is that the events reported in the FSAR are the most limiting events.

Reference

1. NUREG-0787, Supplement No. 1, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit No. 3", USNRC, October, 1981.

ANALYSIS OF FEEDWATER LINE BREAKS

I. PROBABILITY OF FEEDWATER LINE BREAKS

The feedwater line break analyzed in FSAR Section 15.2.3.4 is postulated to occur in the piping between either of the two steam generator nozzles and the containment wall. The methods and data contained in WASH-1400 can be used to estimate the recurrence frequency of such a break. WASH-1400 (Appendix III, Table III 6-9) provides a summary of pipe rupture rates per plant year for various sized "LOCA sensitive" piping. The median LOCA recurrence frequency for large piping (6" diameter) is given as 1×10^{-4} per plant year. For the Waterford design, the total length of "LOCA sensitive" piping with respect to the event analyzed in Section 15.2.3.1 is 232 feet. (See Appendix A). Therefore, using the methodology discussed in WASH-1400, Appendix III, Section 6.4, the estimated recurrence frequency for the above postulated Main Feedwater Line Break is 3.2×10^{-5} PER PLANT YEAR.

Thus it is shown that the initiating event which is analyzed in Section 15.2.3.4 is in fact a very low probability event that is highly unlikely to occur in a plant's lifetime.

Additionally, the high primary system pressures reported in FSAR Section 15.2.3.1 are due to the conservatively assumed coincident occurrence of a loss of normal a/c power. Using WASH-1400 value of 1×10^{-3} for conditional loss of normal a/c power, the estimated joint recurrence frequency for the above postulated main feedwater line break with concurrent loss of normal a/c power is less than 1×10^{-7} per plant year.

II. REANALYSIS OF SMALL FEEDWATER LINE BREAKS

Introduction

The following responds to NRC's request for demonstration that small feedwater line breaks (FWLB) meet 110% design pressure criterion when combined with the limiting single failure, and with offsite power available. The existing FWLB evaluation in FSAR Section 15.2.3.1 demonstrates that all FWLBs, even when combined with a loss of offsite power, are well below 120% design pressure. As indicated above the probability of a FWLB in pipes with diameters greater than 6 inches is sufficiently low to allow application of the 120% design pressure criterion. Therefore, the following FWLB reanalysis addresses break sizes which are less than 0.2 ft^2 (i.e., 6 inch diameter).

Methodology

The original FWLB evaluation methods (1) which are conservative for application to the full spectrum of break sizes have been modified for the small FWLB reanalysis. The modifications include the treatment of reactor trip on low water level in the "ruptured" steam generator, and the enthalpy of the fluid discharged from the break.

As discussed in Reference 5, the original FWLB method credited low water level trip in the ruptured steam generator only after its liquid inventory decreased to approximately 9000 lbm. This assured conservative treatment of low level trip even if the FWLB (i.e., large breaks) caused rapid steam generator depressurization and consequently swelling of the downcomer level due to flashing of the downcomer liquid. However, for breaks less than 0.2 ft^2 the average steam generator pressure remains constant or increases prior to reactor trip and no downcomer level swell will occur due to flashing. Therefore, in the reanalysis of samll FWLBs steam generator low water level trip is credited with a larger liquid inventory remaining.

The low level trip setpoint corresponds to a downcomer liquid level of approximately 27 feet above the tube sheet and a liquid inventory of over 70,000 lbm. However, the reanalysis of small FWLBs conservatively delays low level trip until 7.5 feet above the tube sheet (approximately 22,000 lbm of liquid).

To expedite the reanalysis of small FWLBs the enthalpy of the break discharge fluid is assumed to be saturated liquid until all liquid is depleted from the ruptured steam generator. This differs from the original FWLB steam generator method employed for Waterford 3 FSAR (see Reference 5).

Analysis Input

A spectrum of small (0.2 ft^2) FWLBs with the limiting single failure and offsite power available were analyzed with the new method. As for the FWLB analysis presented in the Waterford FSAR, the CESEC-ATWS code was used to model the FWLB transient.

As a result of the evaluation method applied to the FWLB analysis, the only mechanisms for mitigation of the reactor coolant system (RCS) pressurization are the pressurizer safety valves, the reactor coolant flow and the main steam safety valves. The last two mechanisms influence the RCS to steam generator heat transfer rate.

There are no credible failures which can degrade pressurizer safety valve or main steam safety valve capacity. A decrease in RCS to steam generator heat transfer due to reactor coolant flow coastdown can only be caused by a failure to fast transfer one half the electrical loads to offsite power or a loss of offsite power following turbine trip (i.e., two or four pump coastdown respectively). The FWLB analysis of Subsection 15.2.3.1 considers the worst of the two, the loss of offsite power.

The WSES-SER requires that small feed line breaks with the limiting single failure and offsite power available should meet 110% of design pressure, in accordance with the SRP requirements. Therefore, the limiting single failure assumed in the reanalysis of feed line breaks is the failure to fast transfer.

In order to determine the limiting small feedwater line break, the initial pressurizer pressure and steam generator inventories were adjusted within their operational ranges to obtain a coincident trip on high pressurizer pressure and low steam generator water level. This coincidence ensures that the RCS pressure at the time of trip is at its maximum value, maximizing the RCS pressurization potential of the FWLB.

Results

Four different break sizes were analyzed including 0.01, 0.05, 0.10, and 0.20 ft^2. A plot of the results of these analyses is provided in Figure 1. In all cases, the maximum pressure remained below 2700 psia which is less than 110% of design.

The worst case small FWLB in terms of primary pressurization is the 0.2 sq ft break. Tables 1 and 2 provide the initial conditions assumed and the sequence of events for the transient. Figure 2 provides a plot of RCS pressure vs. time for the event.

Conclusion

Small feedwater line breaks with the limiting single failure and offsite power available result in maximum primary pressures of less than 110% of design.

References

1. CENPD-107 Supplement 1, "ATWS model modifications to CESEC", September 1974. (Section 3.0).

2. CENPD-107 Supplement 1, Amendment 1-P, "ATWS model modifications to CESEC", November 1975. (Section 3.3).

3. CENPD-107 Supplent 3, "ATWS model modifications to CESEC", August 1975. (Sections 240.8, 240.11 and 240.9).

4. CENPD-107 Supplement 4, "ATWS model modifications to CESEC", December 1975. (Sections 1.6, 1.8 and 4.2).

5. FSAR, Appendix 15A, Section 15.A.2 Amendment 22, September, 1981

TABLE 1

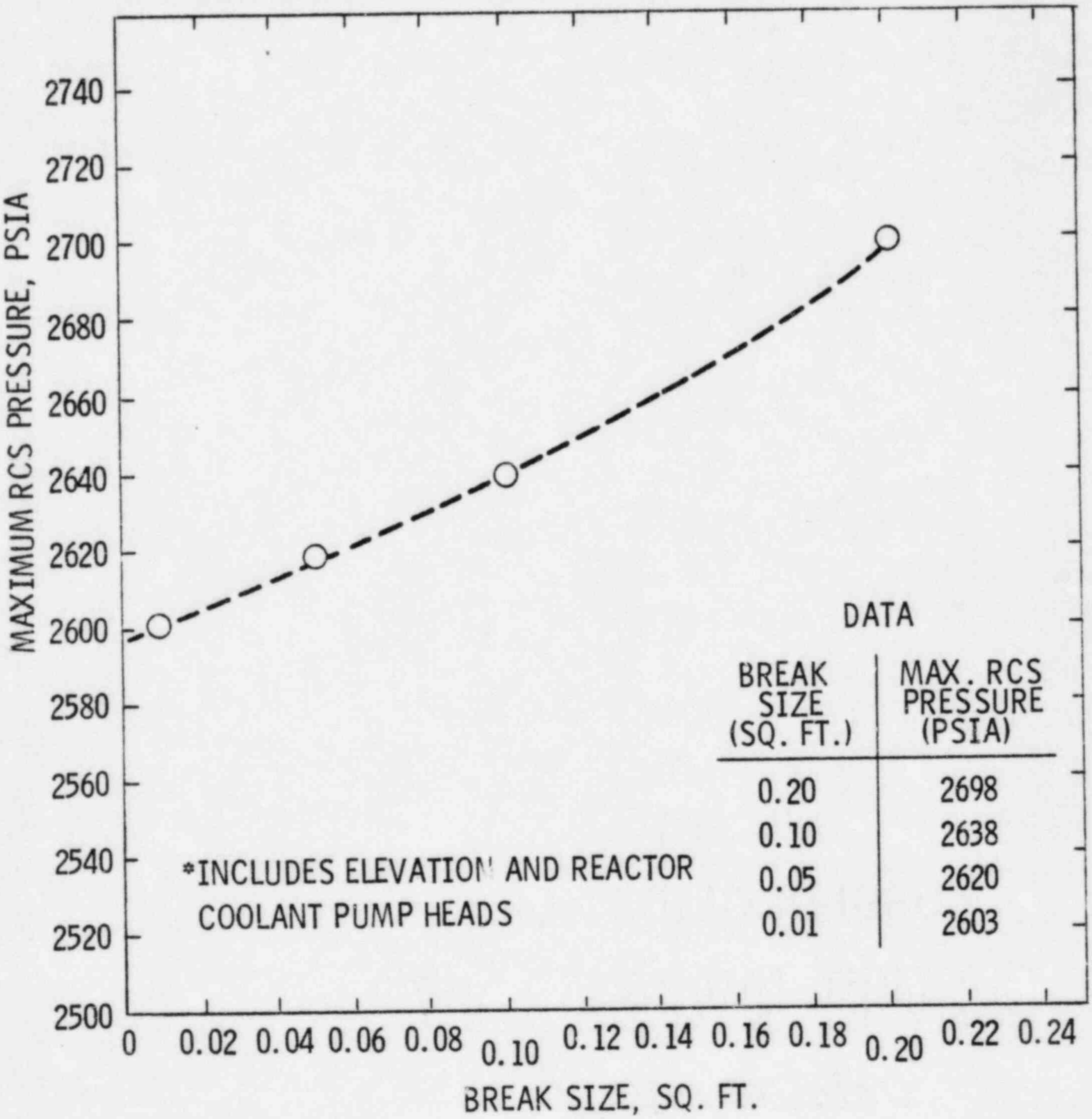
ASSUMPTIONS FOR THE FEEDWATER SYSTEM PIPE BREAK

Parameter	Assumption
Initial core power, MWt	3,478
Core inlet coolant temperature	560
Core mass flowrate, 10^6 lbm/hr	132
Reactor coolant system pressure, psia	2,250
Steam generator pressure, psia	964
Moderator temperature coefficient, 10^{-4}	0.0
Doppler coefficient multiplier	0.85
CEA worth for trip, 10^{-2}	-8.55
Steam bypass control system	Inoperative
Pressurizer pressure control system	Inoperative
Pessurizer level control system	Inoperative
Feedwater line break area, ft^2	0.2
Initial steam generator total inventory, lbm	132,000

Table 2

SEQUENCE OF EVENTS FOR THE FEEDWATER SYSTEM PIPE BREAK

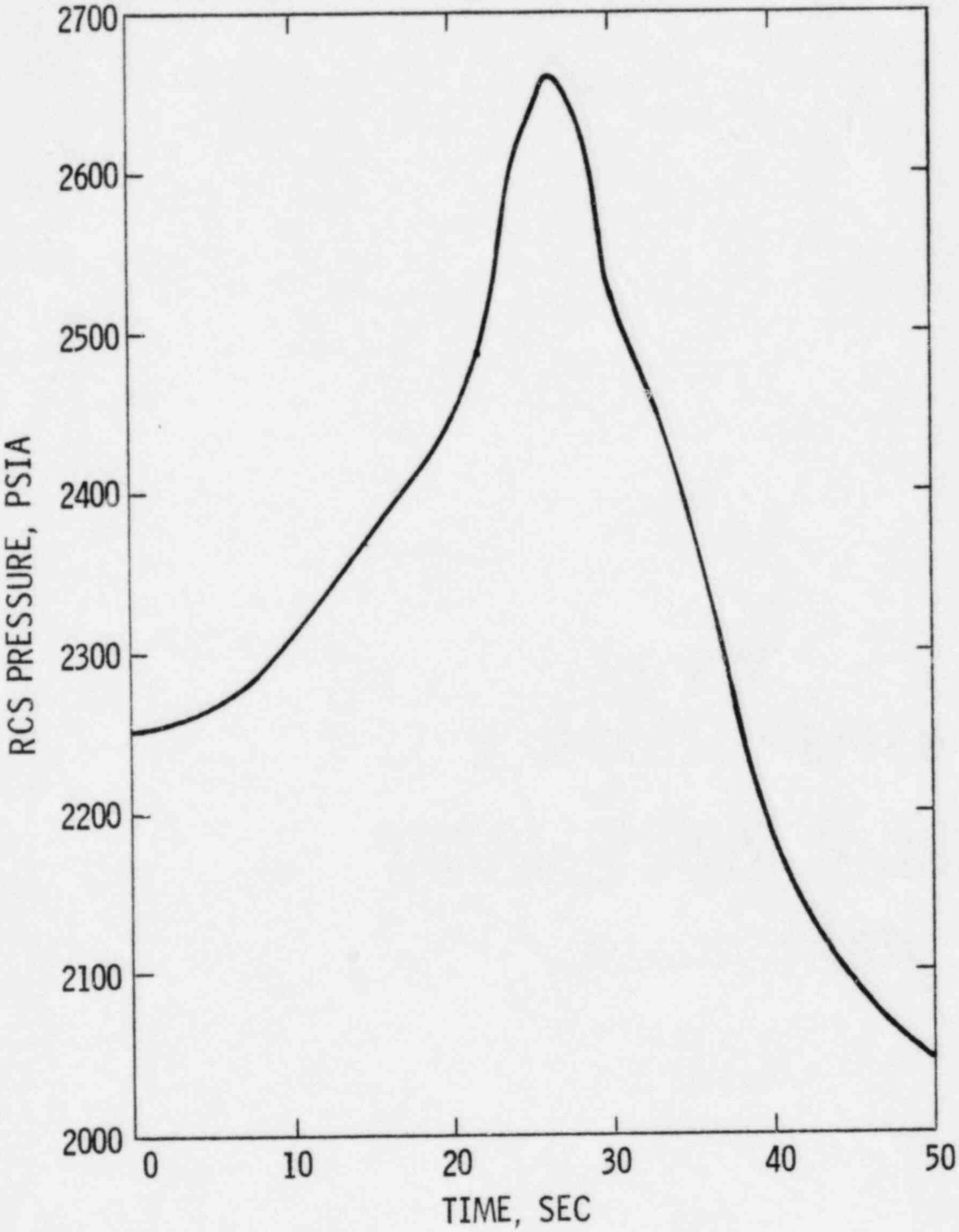
TIME	EVENT	SETPOINT OR VALUE
0.0	Double-ended rupture of the main feedwater line	----
0.0	Complete loss of feedwater to both S.G.s	----
21.9	High pressurizer pressure trip condition (psia)	2474
23.1	High pressurizer pressure trip signal occurs (psia)	2474
23.4	Steam generator safety valves open (psia)	1085
23.8	Pressurizer safety valves open (psia)	2525
24.0	CEAs begin to drop into core	____
25.8	Maximum pressurizer surge line flow (lbm/sec)	1374
26.0	Maximum RCS pressure (psia)	2698
28.6	Maximum steam generator pressure (psia)	1156
30.6	Pressurizer safety valves close (psia)	2525



DATA

BREAK SIZE (SQ. FT.)	MAX. RCS PRESSURE (PSIA)
0.20	2698
0.10	2638
0.05	2620
0.01	2603

LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station	MAXIMUM RCS PRESSURE vs BREAK SIZE	Figure 1



LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station	RCS PRESSURE vs TIME FOR THE LIMITING SMALL FWLB (DOES NOT INCLUDE REACTOR COOLANT PUMP AND ELEVATION HEAD)	Figure 2