Surry Power Station
Updated Final Safety Analysis Report

Chapter 14

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CHAPTER 14 SAFETY ANALYSIS

14.1 GENERAL

This chapter evaluates the safety aspects of the station and demonstrates that the station can be operated safely and that exposures from credible accidents are less than or equal to the limits of 10 CFR 50.67 or Regulatory Guide 1.183, as applicable.

This chapter is divided into sections, each dealing with a different behavior category. The sections are as follows:

1. Core and Coolant Boundary Protection Analysis, Section 14.2.

   The incidents presented in Section 14.2 are associated with an individual unit within the station.

2. Standby Safeguards Analyses, Section 14.3.

   The accidents presented in Section 14.3 are steam generator tube rupture, steam-line break, and control-rod ejection. High-energy line breaks outside containment are discussed in Appendix 14B (Reference 1).

3. General Station Accident Analysis, Section 14.4.

   The accidents presented in Section 14.4 are associated with shared systems and facilities that may cause the release of radioactive material to the environment.

4. Loss-of-Coolant Accident (including the design-basis accident), Section 14.5.

   The loss-of-coolant accident, or the rupture of a reactor coolant pipe, is the worst accident case and is the primary basis for the unit design requirements. It is shown that even this accident meets the limits of 10 CFR 50.67, assuming that the core has been operating at 2605 MWt. This core power level is conservative compared to 100.38% of the rated power level of 2587 MWt (i.e., 2596.9 MWt).

All accident analyses were originally performed assuming the use of Zircaloy fuel rod cladding. The impact of the use of ZIRLO as an alternate cladding material was evaluated by Westinghouse (Reference 2). The properties of these two zirconium-based alloys are essentially identical except for the temperature at which the alpha to beta phase change occurs, and its related effect on the thermophysical properties (particularly the specific heat over the phase transformation temperature range). Therefore, the use of ZIRLO does not affect the analyses of non-LOCA accidents for which the clad temperature remains below the ZIRLO phase change temperature (1380°F). This includes all Condition I and Condition II events. The only non-LOCA accident analyses in which the clad temperatures are predicted to reach 1380°F or greater are the locked rotor analysis (Section 14.2.9.2) and the rupture of a control rod mechanism housing (Section 14.3.3). The effect of the use of ZIRLO cladding is discussed in the applicable sections for these accident analyses. The impact of the use of the ZIRLO alloy on the large break LOCA (Section 14.5.1) and the small break LOCA (Section 14.5.2) analyses was also assessed.

Topical Report VEP-NE-2-A (Reference 3) describes the calculation of “retained DNBR margin” as the difference between the DNBR design limit (i.e., Safety Analysis Limit) and the Statistical Design Limit (SDL). The available retained DNBR margin is evaluated for each reload core, considering DNBR penalties for generic fuel design issues (e.g., fuel rod bow), cycle-specific violations of limits (e.g., fuel rod power census), and plant operating conditions. Surry UFSAR Section 3.4.3.5 also summarizes the applicable uses of retained DNBR margin.

The SDL is 1.27 for the 15 x 15 Surry Improved Fuel (SIF) product with the COBRA/WRB-1 code/correlation pair. The Safety Analysis Limit (SAL) for the 15 x 15 SIF product is 1.46. These limits are described in Surry UFSAR Sections 3.2.3.3 and 3.4.3.5. The difference between these DNBR limits is 13.0% retained DNBR margin that is used in the core reload thermal-hydraulic evaluation in accordance with Reference 3. The approach for the MUR power uprate is to apply a penalty against retained DNBR margin for the 15 x 15 SIF product. With the use of retained DNBR margin to accommodate the power uprate, the UFSAR Chapter 14 DNBR analyses are not affected, with the exception of Uncontrolled Control-Rod Assembly Withdrawal at Power (RWAP) (UFSAR Section 14.2.2) which was reanalyzed for the MUR uprate.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 4). This DNBR penalty is conservative and applicable to all statistically treated DNBR events except RWAP and will be deducted from the retained margin for the 15 x 15 SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 3).

For the implementation of the Westinghouse 15 x 1 5 Upgrade fuel design at Surry, all accident analyses were reviewed for potential impact upon the NSSS predictions. The 15 x 15 Upgrade fuel assembly is described in Section 3.3 of the UFSAR. The change in cladding from ZIRLO to Optimized ZIRLO (Reference 5) as approved by the NRC (Reference 8) does not significantly change the cladding material properties. Furthermore, there are no significant changes in fuel properties between the 15 x 15 SIF product and the 15 x 15 Upgrade fuel design. Since the changes in material properties are negligible, no transient reanalysis for the non-LOCA events (UFSAR Sections 14.2 and 14.3) was performed for the 15 x 15 Upgrade fuel design.

The implementation of the 15 x 15 Upgrade fuel design has an impact on the calculated DNBR results for the Chapter 14 accident analyses. The DNBR analyses were conducted using the VIPRE-D thermal-hydraulic code (Reference 6) and the WRB-1 and W-3 CHF correlations. In Reference 7, the NRC approved the use of the VIPRE-D/WRB-1/W-3 code/correlation pairs and the supporting DNB statepoint calculations for the 15 x 15 Upgrade fuel design. VIPRE-D/WRB-1 together with the Virginia Power Statistical DNBR Evaluation Methodology (Reference 3) has been applied to the accidents listed in Section 3.2.3. Statepoints for applicable DNB events were analyzed with VIPRE-D/WRB-1 at 2589.3 MWt for full-power, statistically-treated events. All statistically-treated DNB events show acceptable DNB performance for the Westinghouse 15 x 15 Upgrade fuel design at 2589.3 MWt core thermal power. The analyzed core power is bounding for the measurement uncertainty recapture (MUR) uprated core power of 2587 MWt. The deterministic events (see Section 3.2.3) analyzed with VIPRE-D/W-3 code/correlation pair are not impacted by the MUR (Reference 4).

Subsequent to the implementation of the Westinghouse 15 x 15 Upgrade fuel design, the ABB-NV and WLOP DNB correlations were approved for use as a replacement for the W-3 DNB correlation. Consistent with the implementation of the 15 x 15 Upgrade fuel design, a DNBR analysis for the affected events (Rod Withdrawal from Subcritical and Main Steamline Break) was performed using the ABB-NV and WLOP DNB correlations. In Reference 9, the NRC approved the use of the VIPRE-D/ABB-NV and VIPRE-D/WLOP code correlation pairs and the supporting DNB statepoint calculations for the Westinghouse 15 x 15 Upgrade fuel design at Surry.

The safety analyses described in Chapter 14 have been updated to support the following DNB analysis configurations:

- Westinghouse Surry Improved Fuel with COBRA/WRB-1 correlation pair with a full power $F\Delta H$ limit of 1.56 for statistical DNB calculations and COBRA/W-3 code correlation pair with a full power $F\Delta H$ limit of 1.62 for deterministic DNB calculations

- Westinghouse 15x15 Upgrade fuel with VIPRE-D/WRB-1 correlation pair with a full power FΔH limit of 1.56 for statistical DNB calculations and VIPRE-D/W-3 code correlation pair with a full power FΔH limit of 1.62 FΔH for deterministic DNB calculations
- Westinghouse 15x15 Upgrade fuel with VIPRE-D/WRB-1 correlation pair with a full power FΔH limit of 1.635 for statistical DNB calculations and VIPRE-D/ABB-NV or VIPRE-D/WLOP code correlation pairs with a full power FΔH limit of 1.70 FΔH for deterministic DNB calculations

The ability of the automatic rod control system to withdraw rods from the core has been eliminated (Chapter 7). This prevents addition of positive reactivity by rod withdrawal in response to transient event conditions thereby resulting in lower power during the event. Analyses with automatic rod control have been found to be limiting for the following events: Control Rod Assembly Drop/Misalignment (Section 14.2.4), Chemical and Volume Control System Malfunction (Section 14.2.5), Excessive Heat Removal Due to Feedwater System Malfunctions (Section 14.2.7), and Excessive Load Increase Incident (Section 14.2.8). The automatic rod control cases are retained herein, even though they are no longer credible, as they continue to bound the current plant design.

14.1 REFERENCES

1. Letter from Vepco to AEC, Subject: *Submittal of FSAR Appendix D (Effects of Piping System Breaks Outside Containment),* dated June 22, 1973.
2. S. L. Davidson (Ed.), et al, *VANTAGE+ Fuel Assembly Reference Core Report,* WCAP-12610-P-A (Proprietary), April 1995.
3. VEP-NE-2-A, *Statistical DNBR Evaluation Methodology*, June 1987.
4. Letter from K. Cotton (NRC) to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME2393 and ME3294)*, ML101750002, Dominion Serial Number 10-580, September 24, 2010.
5. H. H. Shah and P. Schueren, *Optimized ZIRLO™,* WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, July 2006.
6. DOM-NAF-2-A, *Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code*.
7. Letter from K. Cotton (NRC), to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Request for Technical Specification Revisions Related to the Core Operating Limits Report (TAC Nos. ME2591 and ME2592)*, October 19, 2010, Serial No. 10-645.
8. Letter from K. Cotton (NRC) to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding the Use of Optimized ZIRLO™ Fuel Rod Cladding (TAC Nos. ME3343 and ME3344)*, Serial No. 10-669, December 22, 2010; as amended by Letter from K. Cotton (NRC) to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Corrections to Amendments Regarding the Use of Optimized ZIRLO™ Fuel Rod Cladding (TAC Nos. ME3343 and ME3344)*, Serial No. 10-755, December 23, 2010.
9. Letter from V. Sreenivas (USNRC) to D. A. Heacock (Dominion), *North Anna and Surry Power Stations Units 1 and 2, Issuance of Amendments Regarding Addition of an Analytical Methodology to the North Anna and Surry Core Operating Limits Reports and an Increase to the Surry Minimum Temperature for Criticality (TAC Nos. MF2364, MF2365, MF2366, and MF2367),* Dominion Serial No. 14-410, dated August 12, 2014.

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14.2 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.2.1 Uncontrolled Control-Rod Assembly Withdrawal From a Subcritical Condition

A control-rod assembly withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by the withdrawal of control-rod assemblies, resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in Section 14.2.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low-power level during start-up by control-rod withdrawal. Although the initial start-up procedure used the method of boron dilution, the normal start-up is with control-rod assembly withdrawal. Control-rod assembly motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The control-rod drive mechanisms are wired into preselected banks, and these bank configurations are not altered during core life. The assemblies are therefore physically prevented from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The control-rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable-speed rod travel. The maximum reactivity insertion rate is postulated in a detailed analysis assuming the simultaneous withdrawal of the combination of the two rod banks of the maximum combined worth at maximum speed.

Should a continuous control-rod assembly withdrawal be initiated from subcritical or low power conditions, the transient will be terminated by the following automatic safety features:

1. Source range flux level trip—actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either intermediate-range flux channel indicates a flux level above the source range cutoff power level. It is automatically reinstated when both intermediate-range channels indicate a flux level below the source range cutoff power level.
2. Intermediate-range control-rod stop—actuated when either of two independent intermediate-range channels indicates a flux level above a preselected, manually adjustable value. This control-rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10% of full power. It is automatically reinstated when three of the four power range channels are below this value.
3. Intermediate-range flux level trip—actuated when either of two independent intermediate-range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when two of the four power range channels are reading above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

4. Power range flux level trip (low setting)—actuated when two out of the four power range channels indicate a power level above approximately 25% of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

5. Power range control-rod stop—actuated when one out of the four power range channels indicates a power level above a preset setpoint. This function is always active.

6. Power range flux level trip (high setting)—actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

Reactor protection for subcritical and low power rod withdrawal events has traditionally been assumed to be provided by the Power Range high flux trip (low setpoint) for events initiated both above and below permissive P-6. Source Range protection was assumed to not be available, since the Source Range channel lacked the redundancy required to assume trip availability in UFSAR accident analyses. Technical Specifications require two available Source Range Channels below permissive P-6. In conjunction with Source Range trip bistable operability testing to verify Source Range channel response characteristics, this validates the assumption of Source Range trip availability in accident analyses. Technical Specifications also impose an allowable Source Range channel outage time for power levels below P-6. This ensures start-up protection by providing confirmation of the availability of the Source Range channel.

Rod withdrawal from subcritical events may be initiated from above or below permissive P-6. Below P-6, one or two reactor coolant pumps may be operating, or reactor coolant system (RCS) cooling may be provided by the Residual Heat Removal (RHR) system. For any operating condition below P-6, Source Range protection or open trip breakers provide reactor protection against a rod withdrawal from subcritical event. Additional protection is provided by the other operable reactor protection system circuitry, including the Intermediate Range and Power Range (low setpoint) reactor trips. As was demonstrated in the North Anna Core Uprating submittal (Reference 27) and subsequent responses to NRC questions (References 28 & 29) events initiated from allowable operating conditions below P-6 will not result in significant power generation of core heat flux when a reactor trip is actuated on the Source Range channel. This conclusion is also applicable to Surry. Therefore, reactor protection is provided for all operating conditions below P-6, including one-RCP, two-RCP, and RHR operation.

The nuclear power response to a continuous reactivity insertion originating above P-6 is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a start-up incident since it limits the power to a tolerable level before external control action. After the initial power burst, the nuclear power is momentarily reduced and then if the incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

The termination of the start-up incident by the above protection channels prevents core damage. In addition, the reactor trip from high reactor pressure serves as a backup to terminate the incident before an overpressure condition could occur.

14.2.1.1 **Method of Analysis**

The rod withdrawal from subcritical event was analyzed using the RETRAN computer code and the associated Virginia Power reactor system transient methodology (Reference 12). The analysis includes the simulation of the plant neutron kinetics, and the core thermal and hydraulic feedback equations. The RETRAN code calculates nuclear power, core heat flux, average fuel, clad and coolant temperatures. The detailed core thermal-hydraulics analysis was performed for the SIF product using the COBRA computer code (Reference 13) to generate the MDNBR (Minimum Departure from Nucleate Boiling Ratio) at the statepoint for the DNB-limiting case of the transient. The MDNBR for the 15 x 15 Upgrade fuel design was determined at the statepoint for the DNB limiting case of the transient using the VIPRE-D computer code (Reference 36).

The analysis assumes the operation of all three reactor coolant pumps. Technical Specification 3.1.A.1.a prohibits achieving criticality with less than three reactor coolant pumps operating. The following additional assumptions were made to provide conservative results for this analysis:

1. Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, a conservative fuel-temperature-dependent Doppler coefficient was used.

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux response constant. However, after the initial nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value of +6 pcm/°F was used in the analysis since the positive value yields the maximum peak core heat flux (1 pcm = 10^{-5} $\Delta k/k$).

3. The reactor is assumed to be at hot zero power with a T_{avg} of 547°F. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-to-water thermal conductivity, a larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the nuclear flux peak. The high nuclear flux peak combined with a high fuel thermal capacity and large thermal conductivity yields a larger peak heat flux. Initial multiplication (k_o) is assumed to be 1.0 since this results in the maximum nuclear power peak.

4. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and control-rod assembly release, are taken into account. A 10% increase has been assumed for the power range flux trip setpoint, raising it from the nominal value of 25% to 35%. The rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible.
5. The rate of negative reactivity insertion corresponding to the trip action is based on the assumption that the highest worth control rod assembly is stuck in its fully withdrawn position. A conservatively low value was assumed for the total trip reactivity from zero power.
6. The maximum positive reactivity insertion rate assumed (112.5 pcm/sec) is greater than that for the simultaneous withdrawal of the combination of the two control banks having the greatest combined worth at maximum speed (45 in/min).
7. The initial power level was assumed to be below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.
8. The delayed neutron fraction (β_{eff}) was assumed to be at its maximum value, as that would maximize the thermal energy released into the coolant.
9. On the secondary side, the condenser dump valves are assumed closed, thus causing a pressure buildup that would contribute to the heatup of the primary system.
10. Since this event is evaluated at hot zero power conditions (0% rated core power), the UFSAR analysis of record is unaffected by the MUR power uprate to 2587 MWt.

For the pressure-limiting case, to conservatively overestimate the pressurization in the RCS, the following additional assumptions are made:

1. Initial pressurizer pressure is 2280 psia (30 psi above the nominal).
2. Initial pressurizer level is 5% above the nominal.
3. PORVs and pressurizer sprays that would mitigate the pressurization are not credited.
4. The PSV loop seals are filled with water. Displacing the liquid in the loop seal causes a delay in the opening of the PSVs, thus driving the primary system pressures higher.

In the DNB-limiting case, the following specific assumptions are made to decrease the primary pressurization and increase the energy released into the coolant, thus minimizing the calculated margin to DNB:

1. Initial pressurizer pressure and level are held at their nominal values.
2. Pressurizer sprays and PORVs are credited, thereby mitigating system pressurization.

3. For the MDNBR calculation, conservative values for the pressurizer pressure, RCS flow and the bypass flow fraction are used.

14.2.1.2 **Results**

Figure 14.2-2 shows the effect of the initial power level on peak heat flux for various reactivity insertion rates from 20 to 60 pcm/sec. It shows that peak heat flux initially decreases with increasing initial power level and then, depending on the rate, it increases again and approaches 35% of full power (reactor trip is assumed to be initiated at this value). It can also be seen that for the faster insertion rates, which result in the greatest energy addition, the flux peak is greatest for the lowest initial power level.

Figures 14.2-1, 14.2-3 and 14.2-4 show the transient behavior for a DNB-limiting case, with the incident terminated by reactor trip at 35% power. As seen in Figure 14.2-1, the nuclear power increases to the trip setpoint in 6.8 seconds. The power then overshoots to approximately 966%, but only momentarily. Therefore, the energy release and the fuel temperature increase are moderate. The thermal flux response, of interest for DNB considerations, is shown in Figure 14.2-3. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux of only 53% of 2546 MWt (52% of 2587 MWt). There is an adequate margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 14.2-4 shows the response of the average fuel, cladding and coolant temperatures. The average fuel temperature peaks at 956°F which is much lower than the nominal full power value of 1311°F. The average coolant temperature rises to only 566.4°F while the clad temperature peaks at 597°F. A COBRA calculation for the SIF product calculation at the statepoint with very conservative adjustments to the thermal-hydraulic input variables gives a minimum DNBR above the applicable SAL listed in Section 3.2.3. A VIPRE-D calculation for the 15 x 15 Upgrade fuel design using the same statepoint used in the COBRA model for the 15 x 15 SIF product gives a MDNBR above the applicable SAL listed in Section 3.2.3.

The pressure-limiting case results in a pressurizer pressure peak of 2656 psia at 11.6 seconds, while the overall primary system peaks at 2720 psia in the cold leg at 11.8 seconds.

14.2.1.3 **Conclusion**

It is concluded that, in the unlikely event of a control rod assembly withdrawal incident from subcritical conditions, the core and reactor coolant system are not adversely affected, as the peak thermal power and the peak coolant temperature in the DNB-limiting case are well below their nominal full power values. An explicit statepoint calculation using very conservative assumptions results in a minimum DNBR above the design limit.

In the case that examines primary system pressure, it can be shown that the peak RCS pressure will be less than 110% of design pressure.

14.2.2 Uncontrolled Control-Rod Assembly Withdrawal at Power

The Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power (RWAP) event is characterized by a reactivity increase resulting from the withdrawal of one or more RCCA banks from the core during power operation. The initiating event is a postulated single failure in a control system such as the rod control system or the reactor control system or faulty action by a reactor operator. The addition of reactivity to the core tends to be distributed uniformly, due to the RCCA bank arrangement. The energy removal capabilities of the secondary system tend to lag behind the core power increase resulting from the rod bank withdrawal. This energy mismatch causes the Reactor Coolant System (RCS) pressure and temperature to increase. The possibility exists that the core heat flux could exceed the ability of the RCS fluid to conduct the heat from the fuel, potentially leading to a Departure from Nucleate Boiling (DNB) and subsequent cladding failure. The RCS temperature and pressure transients can be limited by the operation of RCS and main steam (MS) pressure relief valves; however, the power excursion generally continues until terminated by the addition of negative reactivity from the safety control rod banks due to a reactor trip. The limiting event conditions occur shortly after safety control bank insertion, when the minimum DNB ratio (MDNBR) occurs. The Reactor Coolant Pumps (RCPs) remain operational throughout the event so that, in the absence of DNB, sufficient RCS flow exists to adequately handle the transfer of energy from the fuel to the reactor coolant.

As stated above, maintaining the fuel cladding integrity is the primary concern for the RWAP event. However, maintaining the RCS as a fission product barrier is also a concern. Specifically, the heating of the RCS fluid during a RWAP event causes the fluid density to decrease, resulting in a volumetric expansion of the fluid. Operation of the pressurizer sprays and Power Operated Relief Valves (PORVs) can mitigate the effects of the subsequent pressure increase, but do not counteract the volumetric expansion. Should the expansion of the RCS fluid continue uncontested, the potential exists for discharge of liquid through the PORVs or Pressurizer Safety Valves (PSVs). For the rod withdrawal at power event, the reactor protection system terminates the heatup of the reactor coolant system before any liquid relief occurs.

Provided the integrity of the fission product barriers is not compromised, sensible and decay heat can be removed by steaming to the condenser through the steam bypass system, to the atmosphere through the MS PORV or the Main Steam Safety Valves (MSSVs), or any combination of the three methods. Feedwater remains available to the steam generators (SGs) from either the Main Feedwater (MFW) system or the Auxiliary Feedwater (AFW) system to replenish the secondary coolant. Shortly after reactor trip, the energy removal capability of the SGs will exceed the RCS sensible and decay heat levels, and the reactor operators/automatic control systems will function to maintain the plant at the new equilibrium condition.

The automatic features of the reactor protection system that prevent core damage in a control-rod assembly withdrawal incident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip if two out of the four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of three delta T channels exceed an overtemperature delta T setpoint. This setpoint is automatically varied with axial power distribution and coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of three delta T channels exceed an overpower delta T setpoint. This setpoint is automatically varied with axial power distribution and coolant temperature to ensure that the allowable heat generation rate (kw/ft) is not exceeded.

4. A high-pressure reactor trip, actuated from any two out of three pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is actuated at a setpoint. This affords additional protection for control-rod assembly withdrawal incidents. The Technical Specifications require that the reactor be maintained subcritical by some minimum amount until normal water level is established in the pressurizer.

6. In addition to the above-listed reactor trips, there are the following control-rod assembly withdrawal blocks:

   a. High nuclear power (one out of four).

   b. High overpower delta T (two out of three).

   c. High overtemperature delta T (two out of three).

The manner in which the combination of overpower and overtemperature delta-T trips provides protection over the full range of reactor coolant system conditions is described in Chapter 7 and in Figure 14.2-5. Figure 14.2-5 illustrates reactor coolant loop average temperature and delta-T for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower delta-T trip and the overtemperature delta-T trip are represented as “protection lines” on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a trip would occur well within the area bounded by these lines. This diagram is useful in that the limit imposed by any given DNBR can be represented as Reactor Core Safety Limit lines. The DNB lines represent the locus of conditions for which the DNBR equals the limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The region of permissible operation (power, pressure, and temperature) is completely bounded by the following reactor trips: nuclear overpower (fixed setpoint), high pressure (fixed setpoint), low pressure (anticipatory rate dependent setpoint), and overpower and overtemperature delta T (variable setpoints). These trips are designed to prevent a DNBR less than the applicable SAL (Section 3.2.3).

The analysis presented below shows that no fuel damage occurs by demonstrating that the DNBR limit is met for the rod withdrawal event. Also shown is that the RCS and MS system pressure relieving devices have sufficient capacities to ensure the safety of the unit without relying on the mitigating capabilities of the pressurizer pressure control or MS bypass systems.

14.2.2.1 **Method of Analysis**

The RWAP transient is analyzed with the RETRAN computer code (Reference 12) and the detailed core thermal/hydraulic analysis is performed with either the COBRA computer code (Reference 13) for the 15 x 15 SIF product or the VIPRE-D computer code (Reference 36) for the 15 x 15 Upgrade fuel design. The RETRAN system code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level. The COBRA code is used to calculate the DNBR for the transient using the WRB-1 DNB correlation (Reference 30) for the SIF product. The VIPRE-D code is used to calculate the DNBR for the transient using the WRB-1 CHF correlation.

For the DNBR evaluation cases, the initial power level, pressurizer pressure, and RCS average temperature are assumed to be at values consistent with the conditions at 2589.3 MWt, which bounds the MUR uprated nominal power of 2587 MWt. The effects of normal control system variations and measurement uncertainties associated with these parameters are treated statistically and incorporated into the statistical design limit (SDL) (Section 3.2.3) in accordance with Virginia Power's Statistical DNBR Methodology (Reference 17). The calculation of the DNBR is consistent with the current Technical Specifications Core Operating Limit Report limit on $F\Delta H$ as modified by a 0.3 part power multiplier.

For cases where reactor coolant system pressures are of primary interest, the initial reactor power, pressurizer pressure and RCS average temperature are assumed to be at the maximum values consistent with steady state full power operation, including allowances for calorimetric and other instrument errors. In addition these cases are performed with the pressurizer pressure relieving devices (pressurizer spray and PORVs) disabled.

All cases incorporate the assumption of 15% steam generator tube plugging. To obtain conservative results the following assumptions are made:

1. Reactivity coefficients - two cases are analyzed:
   a. Minimum reactivity feedback. A positive moderator temperature coefficient of +6.0 pcm/°F in conjunction with a least negative Doppler temperature coefficient is used in the analysis.
   b. Maximum reactivity feedback. A conservatively large negative moderator coefficient -45.0 pcm/°F and a large (in absolute magnitude) negative Doppler temperature coefficient are assumed.

2. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of conditions at 2589.3 MWt, which bounds the MUR uprated nominal power of 2587 MWt. The delta T trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.
3. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.
4. A spectrum of reactivity insertion rates is analyzed. The maximum positive reactivity insertion rate is greater than the maximum rate of two sequential control rod banks moving at the maximum speed with normal overlap.

The effect of rod cluster assembly movement on the axial core power distribution is accounted for by causing a decrease in overtemperature and overpower delta-T trip setpoints proportional to a decrease in margin to DNB.

14.2.2.2 **Results**

Figure 14.2-6 shows the minimum DNBR for the 15 x 15 SIF product as a function of reactivity insertion rate from initial full power operation of 2589.3 MWt (which bounds the MUR uprated nominal power of 2587 MWt) for the minimum and maximum reactivity feedback. It can be seen that the high-neutron flux and overtemperature delta-T trip setpoints provide protection over the whole range of reactivity insertion rates since the minimum DNBR for all insertion rates is greater than the applicable SAL listed in Section 3.2.3.

Figures 14.2-7 and 14.2-9 show the response of nuclear power, pressurizer pressure, and average coolant temperature to the limiting DNBR case initiated from 2589.3 MWt (0.4 pcm/sec insertion rate). The slow rod withdrawal allows for a sufficient rise in temperature and pressure to cause a trip on overtemperature delta-T. The minimum DNBR for the 15 x 15 SIF product for this case remains well above the limit as indicated by Figure 14.2-6.

Figures 14.2-10 and 14.2-11 show the minimum DNBR for the 15 x 15 SIF product as a function of reactivity insertion rate for the rod withdrawal event starting at 60% and 10% of 2546 MWt, respectively. The results are similar to the 100% of 2589.3 MWt (100.1% of 2587 MWt) power case, except that as the initial power is decreased, the range over which the overtemperature delta-T trip is effective is increased. In all cases, the DNBR is greater than the applicable SAL (Section 3.2.3).

Figures 14.2-12 and 14.2-14 show the nuclear power, RCS average temperature, and cold leg pressure response to the limiting overpressure rod withdrawal incident. Sensitivity cases performed to maximize RCS pressure indicate that limiting results occur for an assumed initial power of 12% of 2546 MWt with a reactivity insertion rate of 55 pcm/sec and minimum reactivity feedback. The reactor trips on high flux at 12.63 seconds. The cold leg pressure reaches a peak value of 2699.00 psia at 13.8 seconds into the transient.

Cases performed to maximize the main steam pressure (maximize the RCS average temperature prior to trip) show that the maximum main steam pressure occurs for rod withdrawal events initiated at 60% of 2546 MWt. The cases providing the maximum main steam pressure are those which allow a gradual but large rise in the RCS average temperature. These are cases with low insertion rates and minimum reactivity feedback or relatively high insertion rates with maximum reactivity feedback. These cases trip on overtemperature delta-T and achieve approximately the same RCS temperature.

Therefore, the maximum main steam pressure is fairly constant (1190 psia) over a range of insertion rates. The analyses support up to 15% steam generator tube plugging.

The MDNBR for the 15 x 15 Upgrade fuel design was determined for the RWAP event at the statepoints for the DNB-limiting cases of the transient using the VIPRE-D computer code (Reference 36). It was determined that the MDNBR for all RWAP cases was above the applicable SAL (Section 3.2.3).

14.2.2.3 **Conclusions**

This analysis indicates that for an uncontrolled rod withdrawal at power event, the following criteria are met:

1. The minimum DNBR remains above the applicable SAL (UFSAR Section 3.2.3).
2. Pressure at the most limiting RCS location is less than 110% of RCS design pressure, or 2750 psia (the Emergency Condition Stress Limit specified in Section III of the ASME Code).
3. Pressure at the most limiting Main Steam System (MSS) location is less than 110% of MSS design pressure, or 1210 psia (the Emergency Condition Stress Limit specified in Section III of the ASME Code).

14.2.3 Malpositioning of the Part Length Control Rod Assemblies

The part length control rod assemblies have been removed from the core.

14.2.4 Control-Rod Assembly Drop/Misalignment

Control-rod misalignment accidents include (1) dropped full length assemblies, (2) dropped full-length assembly groups, and (3) statically misaligned assemblies.

Each control-rod assembly has a rod position indicator channel that displays the position of the assembly. The displays of assembly position are grouped for the operator's convenience. Fully inserted assemblies are further indicated by rod bottom indicators on the redundant rod position flat panel displays. Bank (demand) position is also indicated. Except during start-up physics testing and control-rod exercise testing, the assemblies are moved in preselected banks and the banks are moved in the same preselected sequence.

The dropping of a control-rod assembly could occur only when the drive mechanism is de-energized. This would result in a power reduction and an increase in the hot-channel factor. If no protective action occurred, the reactor control system would restore the power to the level that existed before the incident. This would lead to a reduced safety margin or possibly DNB, depending on the magnitude of the hot-channel factor.

Dropped assemblies or banks are detected by:

1. A sudden drop in the core power level.
2. Asymmetric power distribution as seen on ex-core neutron detectors (Reference 1) or core exit thermocouples.
3. Rod bottom indicators on the redundant rod position flat panel displays.
4. A rod deviation alarm.

The rod bottom condition signal from the rod position indication system is provided for each control-rod assembly. The initiation of this signal is independent of lattice location, reactivity worth, or power distribution changes inherent with the dropped control-rod assembly. The other independent indication of a control-rod assembly drop is obtained by using the ex-core power range channel signals. This rod drop detection circuit is actuated upon the sensing of a rapid decrease in local flux such as could occur from the depression of flux in one region by a dropped control-rod assembly. This detection circuit is designed such that normal load variations do not cause it to be actuated.

A rod drop signal from any control-rod assembly position indication channel, or from one or more of the four power range channels, initiates alarms in the main control room.

Misaligned assemblies are detected by:

1. Asymmetric power distribution as seen on ex-core neutron detectors or core exit thermocouples.
2. A rod deviation alarm.

The resolution of the rod position indicator channel is ±5% of span (±12 steps) under steady state conditions. The deviation of any assembly from its bank by twice this distance (10% of span or 24 steps) will not cause power distributions worse than the design limits. The rod deviation alarm alerts the operator to rod deviation in excess of 10 steps or 4.3% of span.

If one or more of the rod position indicator channels should be out of service, detailed operating instructions in accordance with Technical Specification requirements are followed to ensure the alignment of the nonindicating assemblies.

14.2.4.1 **Methodology of Analysis**

The dropped RCCA(s) event is conservatively evaluated. This evaluation consists of three analyses, transient, nuclear, and thermal/ hydraulic. These analyses provide (l) statepoints, i.e., the reactor power, pressure, and temperature at the most limiting time in the transient, (2) the radial peaking factor at the most limiting conditions in the transient, and (3) the DNB analysis at the conditions determined by 1 and 2.

These analyses are performed using a parametric approach so that cycle specific conditions may be evaluated using the data generated from the three analyses above. On a reload basis an analysis is made using two key cycle specific parameters (the rod worth available for withdrawal and the moderator temperature coefficient) to determine the radial peaking factor prior to the dropped RCCA(s) event necessary to produce the applicable SAL (Section 3.2.3) during the transient for a range of dropped RCCA(s) worths. This range covers those which could be expected for a three loop plant like Surry. These predrop radial peaking factors are compared to the reload design predictions to confirm that the limiting predrop conditions for DNB do not occur during the cycle.

The transient response is calculated using either the LOFTRAN (Reference 4) or the RETRAN (Reference 12) code. These codes simulate the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. Nuclear models are used to obtain hot channel factors consistent with the primary system conditions at the statepoints generated by the transient simulation. The DNB design basis is shown to be met for the SIF product using the COBRA code (Reference 13) or the VIPRE-D code (Reference 36) for the 15 x 15 Upgrade fuel design by combining the primary conditions from the transient analysis with the hot channel factor from the nuclear analysis. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 18.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 35). This DNBR penalty is conservative and applicable to all statistically treated DNBR events and will be deducted from the retained margin for the SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 17).

The ability of the automatic rod control system to withdraw rods from the core has been eliminated. This prevents addition of positive reactivity by rod withdrawal in response to transient event conditions thereby resulting in lower power during the event. The event cases with automatic rod control are retained herein, even though they are not credible, as they continue to bound the current plant design.

14.2.4.2 **Results**

For the dropped RCCA(s) event, power may be reestablished either by reactivity feedback or control bank withdrawal.

Following a dropped RCCA(s) in manual rod control, the plant will establish a new equilibrium condition. The drop will insert negative reactivity which causes the core power level to fall. The mismatch in power between that demanded by the turbine and that generated by the reactor core causes the reactor temperature to fall. The falling temperature in turn causes the reactor coolant pressure to fall. The plant will be tripped on low pressurizer pressure before the DNBR falls to the applicable SAL (Section 3.2.3). This process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA(s) event in the automatic control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 14.2-15 and 14.2-16 show a typical transient response to a dropped RCCA(s) while in the automatic control mode. Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 17. On a reload basis, it is shown that the minimum DNBR remains greater than the applicable SAL (Section 3.2.3)

14.2.4.3 **Conclusions**

For all cases the DNB design basis is met by demonstrating that the DNBR is greater than the applicable SAL (Section 3.2.3).

14.2.5 Chemical and Volume Control System Malfunction

14.2.5.1 **Identification of Causes and Accident Description**

Reactivity can be added to the core by feeding primary grade water into the reactor coolant system via the reactor makeup portion of the chemical and volume control system. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the reactor coolant system. The chemical and volume control system is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve creates a dilution flow path to the reactor coolant system. Inadvertent dilution from this source can be readily terminated by closing the control valve. For makeup water to be added to the reactor coolant system at pressure, at least one charging pump must be running in addition to a primary grade water transfer pump.

The boric acid from the boric acid tank is blended with primary grade water in the blender; the composition is determined by the preset flow rates of boric acid and primary grade water on the control board. Two separate operations are required to dilute: (l) The operator must-switch from the automatic makeup mode to the dilute mode; (2) The blender switch must be turned to the "on" position. Omitting either step prevents dilution, making the possibility of an inadvertent dilution very remote.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the chemical and volume control system. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

14.2.5.2 **Method of Analysis**

To cover all phases of plant operation, boron dilution during Refueling Shutdown, Cold Shutdown, Intermediate Shutdown, Hot Shutdown, Reactor Critical, and Power Operation are considered. Explicit analyses are performed for Reactor Critical and Power Operation. Analyses are not performed for the shutdown operating modes as discussed in Section 14.2.5.3. The case of an inadvertent dilution during a planned dilution or makeup activity is not considered here as an accident analysis, since evaluation of such dilutions is not required by the Standard Review Plan. Boron dilution during start-up of an inactive loop is discussed in UFSAR Section 14.2.6.

The following parameter value ranges were considered in the boron dilution analyses for Power Operation and Reactor Critical operating modes:

1. Steam Generator Tube Plugging Fraction (SGTPF): 0% to 15% SGTP. The effective RCS Volume excludes the pressurizer, reactor vessel upper head, and plugged steam generator tube volumes.
2. Maximum Dilution Flow Rate. When the reactor coolant system is pressurized and the plant is in Power Operation or Reactor Critical mode, the rate of addition of unborated water is limited by the capacity of the charging pumps. The maximum dilution flow rate of 165 gpm, corresponding to the maximum charging flow rate of a single charging pump to a pressurized RCS, is assumed in the analysis when in Reactor Critical or Power Operation modes when Technical specifications do not require Primary Grade water isolation.
3. Bounding values of dilution flow density and RCS Water Density were assumed (maximum dilution density and minimum RCS density).
4. Minimum Shutdown Margin at Power: 1.77% $\Delta K/K$.

The ability of the automatic rod control system to withdraw rods from the core has been eliminated. This prevents addition of positive reactivity by rod withdrawal in response to transient event conditions thereby resulting in lower power during the event. The event cases with automatic rod control are retained herein, even though they are not credible, as they continue to bound the current plant design.

14.2.5.3 **Boron Dilution in Shutdown Operating Modes**

Technical Specifications require that the PG makeup water flow path be closed during Refueling Shutdown, Cold Shutdown, Intermediate Shutdown and Hot Shutdown, thereby preventing a high flow rate boron dilution event from occurring during these operating conditions. To satisfy this requirement, manual valve 1-CH-233 (2-CH-233 for Unit 2), the PG makeup water isolation valve, is secured in the closed position. As an alternative, Technical Specifications allow that manual valves 1-CH-212, 1-CH-215 and 1-CH-218 (2-CH-212, 2-CH-215, and 2-CH-218 for Unit 2) may be secured in the closed position if for any reason it is desired that 1-CH-223 (2-CH-223) be maintained open. This alternative combination of valve closures has the same effect as closing valve 1-CH-223 (2-CH-223). The PG makeup water flow path isolation is also required to be closed within 15 minutes following a planned dilution in all shutdown operating modes. (Allowances are provided in the Technical Specifications for relaxing PG water isolation requirements during approach to critical and immediately after shutdown.) Compliance with Technical Specification requirements for closing the PG water flow path ensures that the highest capacity source of PG water is isolated from the reactor coolant system. Therefore, a high flow rate boron dilution event is not credible in the shutdown operating modes.

It is recognized that there are many paths for dilution of the moderator. The rationale behind isolating the main primary grade water flow path is to preclude dilutions that would cause a rapid, uncontrolled decrease in shutdown margin. Low dilution flow rates have a high probability of being identified and corrected before a significant loss of shutdown margin occurs.

There are a number of plant features that provide diagnostic indication of an inadvertent low flow rate boron dilution in the shutdown modes. These indications include audible count rate and high flux at shutdown alarm from the source range nuclear instruments. Reload core design checks have been implemented to ensure that the source range nuclear instruments provide effective indication of changes in core reactivity in subcritical conditions. In addition, RCS letdown divert valve position, VCT level, PG tank levels and PG header flow rate all provide indication in the Main Control Room of a potential mismatch between charging and letdown and unexpected usage of PG water.

14.2.5.4 **Boron Dilution at Reactor Critical and Power Operation**

The analysis of the boron dilution event at reactor critical conditions indicates that at least 15 minutes are available from positive indication of a dilution in progress (alarm or reactor trip) to loss of shutdown margin for corrective operator action. The analysis conservatively assumed a minimum of 1.77% shutdown margin at the beginning of the dilution.

The boron dilution at power event has been analyzed for the rods in automatic and manual control cases. The results of the analysis indicate that 15 minutes are available after positive indication of a dilution in progress (reactor trip) for corrective operator response before a return to criticality.

The “rods in automatic control” case was shown to be bounded by the “rods in manual control” case. To illustrate, if an initial boron concentration, a dilution flow rate, and a boron worth are assumed, the “rods in manual” case will result in a reduction of shutdown margin potentially beyond that of the minimum shutdown margin required by Technical Specifications. If rods are in automatic, rod insertion due to T_{avg} -T_{ref} deviation will result in a rod insertion limit (indicating dilution is in progress) before the rod bank reaches rod insertion limit, the point at which minimum shutdown margin is defined. Therefore, the “rods in manual” case is assumed to consume a portion of minimum shutdown margin resulting in an operator response time which is always less than that of the corresponding “rods in automatic control” case. The automatic control case is therefore bounded by the manual control case. With either automatic or manual rod control, boron dilution events initiated at or below 100% power will result in an RCS temperature increase and, ultimately, in a high RCS temperature alarm. Positive indication of a dilution in progress in the analyzed boron dilution at power case (100% power; manual rod control), is assumed to be provided by the OTΔT reactor trip. In this analysis case, the high RCS temperature alarm is conservatively assumed to not actuate.

The reactivity transient resulting from an inadvertent boron dilution is essentially identical to that of a control rod assembly withdrawal accident. The reactivity insertion rates used in the analysis are well within the range of reactivity insertion rates considered in UFSAR Section 14.2.2, *Uncontrolled Control-Rod Assembly Withdrawal at Power*. If the reactor is in manual control and the operator takes no action to correct an inadvertent boron dilution, the power and temperature will rise to the overtemperature delta-T trip setpoint. Before the overtemperature delta-T trip, an overtemperature delta-T alarm and turbine runback would be actuated. The time to trip varies with the reactivity insertion rate (which is a function of boron concentration and boron worth) and with the temperature and power reactivity feedback of the core (which are largely functions of burnup). It was shown that 15 minutes are available after a reactor trip before the reactor can return to critical, conservatively assuming a minimum of 1.77% shutdown margin at the beginning of the dilution.

The results of the reactor critical and both the automatic and manual control cases of the boron dilution at power analyses indicate that at least 15 minutes are available, from positive indication of a dilution in progress (alarm or reactor trip) to loss of shutdown margin, for corrective operator response to an unplanned boron dilution.

14.2.5.5 **Conclusions**

Because of the procedures involved in the dilution process and the Technical Specification PG water isolation requirement, a high flow rate boron dilution is not considered credible in the shutdown operating modes. Numerous alarms and indications are available to alert the operator to any unintended low flow rate boron dilution in sufficient time for detection and corrective action prior to loss of shutdown margin.

For Reactor Critical and Power Operation modes, reload cores are designed and analyzed to ensure at least 15 minutes are available between positive indication (alarm or reactor trip) and loss of shutdown margin for corrective operator action in response to a boron dilution.

14.2.6 Start-Up of an Inactive Loop (SUIL) Accident Analysis Design Basis

14.2.6.1 **Event Description**

The SUIL accident analysis considers reactivity additions due to inadvertent introduction of cold and/or unborated water from an isolated (or previously isolated) loop. Because loop stop valve operations are prohibited at conditions other than COLD SHUTDOWN and REFUELING SHUTDOWN, inadvertent reactivity additions due to introduction of cold or unborated water at INTERMEDIATE SHUTDOWN, HOT SHUTDOWN, REACTOR CRITICAL, or POWER OPERATION are not considered.

An SUIL event is defined as an uncontrolled reduction in coolant temperature and/or boron concentration in the core region resulting from either the start-up of a reactor coolant pump (RCP) on an idle loop (the loop stop valves open case), or recirculation through a loop stop valve bypass line on an isolated loop (the loop stop valves closed case) when a reduced coolant temperature or boron concentration exists in the idle (or isolated) loop. A loop is considered idle when its hot and cold leg loop stop valves are open, but the reactor coolant pump on the loop is not operating. When no coolant temperature or boron concentration differential exists between the idle (or isolated) loop and the active portion of the reactor coolant system (RCS), the start-up of an RCP does not result in reactivity insertion, erosion of shutdown margin (SDM), power excursion, or reduction in margin to a departure from nucleate boiling (DNB) condition. Under these conditions, start-up of an RCP is simply a start-up procedure, and does not represent an SUIL event.

Because starting an RCP is a deliberate action under operator control, the initiator for an SUIL event is postulated to be multiple administrative errors. If a significant coolant temperature or boron concentration differential existed between the idle (or isolated) loop and the active portion of the RCS, starting the RCP on the idle (or isolated) loop could result in a reactivity insertion, erosion of SDM, and a power excursion. If the core heat flux exceeded the ability of the RCS fluid to conduct the heat from the fuel, the power excursion could lead to DNB and subsequent cladding failure at localized hot spots. Further, coolant expansion in the core region could lead to overpressurization of the RCS. Administrative controls governed by Technical Specifications ensure that, prior to starting an RCP, the differential coolant temperature and boron concentration between the idle (or isolated) loop and the active portion of the RCS are less than those which could result in complete loss of shutdown margin.

If the administrative controls governed by Technical Specifications are circumvented, and a differential coolant temperature and boron concentration beyond that ensured by the administrative controls is achieved, the start-up of an RCP could result in fuel cladding failure due to the onset of DNB, and potential overpressurization of the RCS. The DNB response of the fuel

would be governed primarily by the core power and RCS temperature transient responses. The major contributors to the core power response are the change to the RCS boron concentration and the change to the RCS fluid temperature. Power changes induced by changing the RCS temperature are driven by the magnitude and direction of the moderator reactivity feedback. For example, the moderator temperature coefficient (MTC) is negative throughout most of the fuel cycle and thus acts to increase the total reactivity of the reactor core, i.e. reactor power, as the RCS temperature decreases. Finally, the fuel Doppler coefficient also plays a minor role in the determination of the transient core power response because the reactivity feedback caused by the heating of the fuel inherently limits the power peaking.

The consequences of an SUIL event would be mitigated by operating RCPs or residual heat removal (RHR) pumps, which would remain operational throughout the event to transfer energy from the fuel to the reactor coolant. Although the RCS temperature and pressure transients would be limited by the operation of RCS and main steam (MS) pressure relief valves, the power excursion would ultimately be terminated by either (a) the addition of negative reactivity from the safety control rod banks due to a reactor trip, (b) aborting the RCP start-up, or (c) manual initiation of safety injection. A reactor protection signal and reactor trip would be generated by one of the following reactor trip system (RTS) functions: source range neutron flux, power range neutron flux (low setpoint), power range neutron flux (high setpoint), overtemperature delta-T, overpower delta-T, loss of flow, or manual reactor trip.

After operator or automatic action to stabilize the RCS conditions, sensible and decay heat would be removed by steaming to the condenser through the steam bypass system, to the atmosphere through the MS power operated relief valves (PORVs) or the main steam safety valves (MSSVs), or any combination of the three methods. However, the desirability of a given method is based on system availability and the extent to which the fission product barriers have been compromised. In all scenarios, feedwater would remain available to the steam generators from the auxiliary feedwater (AFW) system to replenish the secondary coolant. At this point in the transient, the reactor operators or automatic control systems would function to maintain the plant at shutdown conditions.

14.2.6.2 **Accident Evaluation**

An SUIL event is defined as an uncontrolled reduction in coolant temperature or boron concentration in the region of the core resulting from either the start-up of an RCP on an idle loop (loop stop valves open case), or recirculation through a loop stop valve bypass line on an isolated (loop stop valves closed case) loop, when a reduced coolant temperature or boron concentration exists in the idle (or isolated) loop. A loop is considered idle when its hot and cold leg loop stop valves are open, but the RCP on the loop is not operating. The ultimate goal of the accident analysis is to demonstrate that a DNB condition is not reached during the accident and, hence, fuel failure is not predicted to occur.

A high level of confidence that a DNB condition will not be reached is demonstrated by consideration of the Technical Specification requirements for loop stop valve operation, and for filling drained and isolated loops. Technical Specifications and associated procedures ensure that the preconditions necessary for significant reactivity insertion during an SUIL event (i.e., reduced temperature and boron concentration in an isolated or idle loop) cannot be achieved under credible circumstances.

A calculation has been performed to verify that an SUIL event with the maximum credible temperature differential between an idle loop and the active portion of the RCS at COLD SHUTDOWN or REFUELING SHUTDOWN will not result in complete loss of shutdown margin. This calculation is described in Section 14.2.6.2.4. In addition, a calculation was performed to determine the reactivity insertion rate and time to loss of shutdown margin assuming isolated loop recirculation is being performed with 0 ppm boron in the isolated loop. This calculation is described in Section 14.2.6.2.5.

14.2.6.2.1 Loop Configurations Permitted by Technical Specifications

The Technical Specifications permit the following RCS and RHR loop configurations to be achieved:

1. Two RCS Loops Operating, and One RCS Loop Idle (Unisolated)
2. Two RCS Loops Operating, and One RCS Loop Isolated
3. One RCS Loop Operating, and Two RCS Loops Idle (Unisolated)
4. One RCS Loop Operating, One RCS Loop Idle (Unisolated), and One RCS Loop Isolated
5. One or Two RHR Loops Operating, Three RCS Loops Idle (Unisolated)
6. One or Two RHR Loops Operating, Two RCS Loops Idle (Unisolated), and One RCS Loop Isolated
7. One or Two RHR Loops Operating, One RCS Loop Idle (Unisolated), and Two RCS Loop Isolated
8. Two RHR Loops Operating, Three RCS Loops Isolated

In cases 1 through 4, RHR may or may not be in operation. Of the above configurations, only those with an idle, unisolated loop are possible loop configurations for the loop stop valve open case (i.e., configurations 1, 3, 4, 5, 6, and 7). Similarly, only configurations 2, 4, 6, 7, and 8 are possible loop configurations for the loop stop valves closed case. As described below, achievement of these configurations in combination with a reduced idle (or isolated) loop temperature and reduced boron concentration would involve a non-credible combination of operator errors.

14.2.6.2.2 Procedural Requirements for Returning Isolated and Filled Loops to Service

To preclude the possibility of inadvertent reactivity insertion due to boron concentration or temperature mismatch between isolated and active portions of the RCS, Technical Specification 3.17 establishes requirements for loop stop valve operations:

1. Loop stop valves must remain open except during COLD SHUTDOWN or REFUELING SHUTDOWN. (An exception is made for short-term maintenance activities.)
2. When a reactor coolant loop is isolated, the loop stop valves must be de-energized, and their circuit breakers must be locked open.
3. An operable source range nuclear instrumentation channel with audible indication must be continuously monitored when returning an isolated loop to service. The loop stop valves must be closed if the source range count rate doubles.
4. Before opening the hot leg loop stop valve, the boron concentration in the isolated loop must be verified to be greater than or equal to the boron concentration corresponding to the shutdown margin requirements for the active volume of the Reactor Coolant System.
5. Before opening a cold leg loop stop valve, the hot leg loop stop valve must be open, and a relief line flow rate of at least 125 gpm must be established for at least 90 minutes. This time period and flow rate is sufficient to equilibrate the boron concentration and temperature of the isolated and active portions of the RCS. Further, the cold leg temperature of the isolated loop must be verified to be at least 70°F, and within 20°F of the highest cold leg temperature of the active loops. Verification of this condition must be completed within 30 minutes prior to opening the cold leg loop stop valve in the isolated loop. Finally, the boron concentration of the isolated loop must be greater than or equal to the boron concentration corresponding to the shutdown margin requirements for the active volume of the Reactor Coolant System.

Concerning the above requirements for loop stop valve operation, the Basis for Technical Specification 3.17 states: "The return to service of an isolated and filled loop is done in a controlled manner that virtually precludes the possibility of an uncontrolled positive reactivity addition from cold water or boron dilution." The recirculation activity described above is performed under strict administrative controls. Therefore, this activity itself does not constitute a boron dilution event.

14.2.6.2.3 Procedural Requirements for Filling Isolated and Drained Loops

In order to return an isolated and drained loop to service, Technical Specification 3.17 requires that the following conditions be met:

1. The isolated loop must be verified to be drained. Verification must be completed within 2 hours prior to partially opening the hot or cold leg loop stop valve in the isolated loop.

2. The RCS level must be at least 18 feet during the opening of the loop stop valves and during filling of the isolated loop. This requirement is established to ensure that the RCS water level does not drop below mid-nozzle level, thereby ensuring adequate suction conditions for the RHR pumps.
3. A source range nuclear instrument channel is required to be monitored to detect any unexpected positive reactivity addition.

Concerning the return of isolated and drained loops to service, the Basis for Technical Specifications 3.17 states: "An initially isolated and drained loop may be returned to service by partially opening the cold leg loop stop valves and filling the loop in a controlled manner from the Reactor Coolant System. To eliminate numerous reactor coolant pump jogs to completely fill a drained loop, a partial vacuum may be established in the isolated loop prior to commencing filling from the active volume of the Reactor Coolant System. The vacuum-assist loop fill evolution requires initiating seal injection to the reactor coolant pump to permit establishing an adequate vacuum in the isolated loop. A portion of the reactor coolant pump seal injection enters the isolated loop. To eliminate the reactivity concerns associated with the water injected into the isolated and drained loop from the seal injection, a water source of known boron concentration is used."

By returning isolated and drained loops to service in the manner described above, achievement of reduced idle loop temperature and reduced boron concentration would involve a non-credible combination of operator errors.

14.2.6.2.4 Inactive Loop Start-Up with Temperature Mismatch

A bounding calculation has demonstrated that an SUIL event with the maximum credible temperature differential between an idle loop and the active portion of the RCS during COLD SHUTDOWN or REFUELING SHUTDOWN will not result in complete loss of shutdown margin. This calculation assumes that the boron concentration in the idle loop is equal to the concentration in the active portion of the RCS, but that the idle loop temperature is 150°F lower than the active portion of the RCS. Based on this calculation, it is concluded that the reactivity insertion driven only by temperature differential will not result in erosion of SDM, power excursion, or reduction in margin to a DNB condition. Development of a significant boron concentration differential between an idle loop (i.e., loop stop valves open) and the active portion of the RCS is not considered credible.

14.2.6.2.5 Isolated Loop Recirculation with Boron Mismatch

The start-up of an inactive reactor coolant loop with the loop stop valves initially closed has been analyzed. The analysis assumes the inactive loop is at a boron concentration of 0 ppm, while the active portion of the system is at 1500 ppm, a conservatively high value for the beginning of core life. The flow through the relief line is assumed to be at its maximum value of 400 gpm. Even with the assumption that administrative procedures are violated to the extent that an attempt is made to open the loop stop valves with 0 ppm in the inactive loop while the remaining portion

of the system is at 1500 ppm, the dilution of the boron in the core region is slow. The initial reactivity insertion rate is calculated to be 3.2×10^{-5} Δk/sec, considerably less than the reactivity insertion rates considered in the Rod Withdrawal at Power and Rod Withdrawal at Subcritical accident analyses. The operator will recognize a high source range count rate signal, and will terminate the dilution by turning off the pump in the inactive loop or by borating to counteract the dilution.

14.2.6.3 **Conclusion**

The Technical Specifications and associated procedural requirements for unisolation of an isolated loop ensure with a high degree of confidence that the RCS and RHR loop configurations presented above cannot achieve the preconditions (i.e., boron concentration and temperature in the isolated loop) necessary for a significant reactivity insertion due to unisolated loop start-up. The recirculation activity which constitutes the loop stop valves closed case is an operating procedure performed under strict administrative control, and does not by itself constitute a reactivity insertion accident. An SUIL event with the maximum credible temperature differential (and no boron concentration differential) between an idle loop and the active portion of the RCS will not result in complete loss of shutdown margin.

14.2.7 Excessive Heat Removal Due to Feedwater System Malfunctions

14.2.7.1 **Identification of Causes and Accident Description**

Reductions in feedwater temperature or additions of excessive feedwater can result in an increase of core power above full power. Such transients are attenuated by the thermal capacity in the secondary plant and in the reactor coolant system. The overpower overtemperature protection (nuclear overpower and delta T trips) prevents any power increase that could lead to a DNBR of less than the applicable SAL (Section 3.2.3).

A feedwater temperature reduction and subsequent reactor coolant system load increase can be initiated by any of the following events: the inadvertent opening of a high-pressure feedwater heater bypass valve which diverts flow around a first-point feedwater heater, the inadvertent opening of a low-pressure feedwater heater bypass valve which diverts flow around the second-, third-, and fourth-point feedwater heaters, or the isolation of extraction steam to the first-point feedwater heaters. Inadvertent bypass valve opening or extraction steam isolation results in a sudden reduction in feedwater inlet temperature to the steam generators. The increased subcooling creates a greater load demand on the RCS. The feedwater heater bypass valves can only be opened manually.

A second example of excessive heat removal is a transient associated with the accidental full opening of feedwater regulating and bypass valves in one or more steam generator loops due to control system malfunction or operator error. The sudden increase in feedwater flow would increase the subcooling of the primary system resulting in a higher core power due to reactivity feedback.

14.2.7.2 **Method of Analysis**

The feedwater temperature reduction event is evaluated by determining a conservative feedwater temperature reduction for the initiating events described in Section 14.2.7.1. The resulting feedwater temperature reduction from each initiating event is shown to be less than the temperature reduction required to generate a primary system load increase of 10% of full power. The event was explicitly analyzed at 2546 MWt for a bounding, 60°F feedwater temperature reduction with the transient analysis code RETRAN (Reference 12), which simulates the reactor coolant system, core kinetics, and the feedwater and steam systems. DNBR analysis for the SIF product was performed with the thermal-hydraulic code COBRA (Reference 13) or the VIPRE-D code (Reference 36) for the 15 x 15 Upgrade fuel design. The analysis incorporates the Statistical DNBR Evaluation Methodology (Reference 17).

The feedwater temperature reduction transient analysis was performed at nominal values consistent with steady-state full power operation: initial pressurizer pressure of 2250 psia, RCS average temperature of 573°F, and full power at 2546 MWt. The use of nominal conditions is consistent with the Virginia Power Statistical DNBR Methodology (Reference 17). The limiting case had a Doppler temperature coefficient of -1.2 pcm/°F, a moderator temperature coefficient of -45 pcm/°F, and automatic rod control enabled.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 35). This DNBR penalty is conservative and applicable to all statistically treated DNBR events and will be deducted from the retained margin for the SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 17).

Excessive feedwater addition due to a feedwater control system malfunction or operator error, which allows a feedwater control valve to open fully, was also explicitly analyzed. The analyses were performed using the transient simulation code RETRAN. The excessive feedwater flow transient was analyzed for the 15 x 15 SIF product using deterministic conditions consistent with steady-state full power operation to allow for calibration and instrument errors. DNBR analysis for the SIF product used the thermal hydraulic code COBRA. Initial pressurizer pressure (2220 psia), reactor coolant average temperature (577°F), and power (102% of 2546 MWt = 2596.9 MWt or 100.38% of 2587 MWt) were assumed at deterministic values consistent with steady-state full power operation to allow for calibration and instrument errors.

In the 15x15 Upgrade statistical DNBR submittal using VIPRE-D to the NRC (Reference 37), it was stated that the Feedwater malfunction event would be analyzed using statistical methods. With NRC approval (Reference 38) of the LAR in Reference 37, the excessive feedwater flow transient will be analyzed using statistical conditions for the 15 x 15 Upgrade fuel design.

The maximum capacity of the feedwater pumps at Surry is no more than 125% of nominal full power flow. However, the excess feedwater transient was analyzed at 125%, 150%, and 200%

of nominal flow. The multi-loop cases assumed equal flows in all three secondary loops. The analyses show that the multi-loop transients experience a lower DNBR than the corresponding single-loop cases. Transients with automatic rod control were shown to have a slightly lower DNBR than the manual rod control cases. The limiting case is the multi-loop analysis with 150% feedwater flow and automatic rod control.

The ability of the automatic rod control system to withdraw rods from the core has been eliminated. This prevents addition of positive reactivity by rod withdrawal in response to transient event conditions thereby resulting in lower power during the event. The event cases with automatic rod control are retained herein, even though they are not credible, as they continue to bound the current plant design.

14.2.7.3 **Results**

14.2.7.3.1 Excessive Feedwater Flow Transient

Figures 14.2-17 through 14.2-21 show the multiple loop 150% feedwater transient with reactor control. The positive reactivity feedback from the sudden increase of feedwater at 0.001 second results in an increase of core power which levels off at 106% of 2546 MWt at about 22 seconds into the transient. The excessive feedwater addition to the steam generators causes an overcooling of the reactor coolant system, resulting in a decrease in pressurizer pressure and RCS average temperature. The analysis results demonstrate that no rods have a calculated MDNBR less than the applicable SAL listed in Section 3.2.3. The mismatch between feedwater flow and steam flow causes the steam generator level to rise until the SG high-high level setpoint is reached, actuating feedwater isolation at 115 seconds. With the feedwater flow reduced to zero, the primary system heats up causing a rise in RCS temperature and pressurizer pressure and a decrease in core power due to negative moderator temperature feedback. The steam generator inventories continue to boil off to dissipate the core power that is still being generated. Eventually, the SG inventory drops to the low-low level setpoint, tripping the reactor at 207 seconds, followed by a turbine trip 2 seconds later.

14.2.7.3.2 Feedwater Temperature Reduction Event

The feedwater temperature reduction event was analyzed with RETRAN and COBRA for the 15 x 15 SIF product or the VIPRE-D computer code (Reference 36) for the 15 x 15 Upgrade fuel design for a 200-second duration, which was adequate to demonstrate a new steady-state condition well beyond the point of the event minimum DNBR. Feedwater temperature reduction, normalized nuclear power, change in pressurizer pressure, change in RCS loop ΔT and change in RCS average temperature as a function of time are illustrated in Figures 14.2-22 through 14.2-26. The reduction in feedwater temperature causes a cooldown of the reactor coolant system resulting in a decrease in coolant average temperature and pressurizer pressure. The reduction in coolant average temperature results in an increase in core power from the large negative moderator temperature coefficient present at end of cycle. This increase in core power balances the RCS cooldown so that the system reaches a new steady-state condition at 109% of 2546 MWt, with

T_{avg} 2.3°F below nominal and RCS loop ΔT 5.7°F above nominal. The reactor does not trip under these conditions. Pressurizer pressure decreases to 25.6 psi below nominal before recovering due to pressurizer heater actuation. The analysis results demonstrate that no rods have a calculated MDNBR less than the applicable SAL listed in Section 3.2.3. Analysis results confirm that the excessive load increase event evaluated in Section 14.2.8 is more limiting with respect to DNBR.

14.2.7.3.3 Excessive Feedwater Flow Hot Zero Power

Multiple loop excessive flow malfunction is not considered credible at no load conditions. The Feedwater Control System (FWCS) would be in manual mode at start-up and low power. Thus, a series of operator actions inadvertently opening the main and bypass control valves in more than one loop simultaneously would be extremely improbable. At full power, the FWCS is in automatic and a multiple loop control system malfunction becomes more credible. However, the possibility of single loop malfunction at hot zero power has been considered and is discussed below.

The reactivity insertion rate at no load following an excessive feedwater accident has also been calculated, with the following assumptions:

1. A step increase in feedwater flow to one steam generator from 0 to the nominal full-load flow.
2. The most negative reactivity moderator coefficient at the end of life. The value used in the calculation was for a rodded core. The value when just critical at no load will be less negative.
3. A constant feedwater temperature of 70°F.
4. Neglect of the heat capacity of the reactor coolant system and the thick metal of the steam generator shell.
5. Neglect of the energy stored in the fluid of the unaffected steam generators.

The maximum reactivity insertion rate was calculated to be 3.9×10^{-4} delta k/sec, which is less than the maximum reactivity insertion rate analyzed in Section 14.2.1, *Uncontrolled Control-Rod Assembly Withdrawal From a Subcritical Condition*. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range flux level trip at a low setting (approximately 25%). As shown in Section 14.2.1 there is a large DNB margin with the above-calculated reactivity insertion rate.

The continuous addition of cold feedwater after a reactor trip is prevented since the reduction of the reactor coolant system temperature, pressure, and pressurizer level leads to the actuation of safety injection on low-low pressurizer pressure. The safety injection signal trips the main feedwater pumps and closes the feedwater pump discharge valves as well as the main feedwater control valves.

14.2.7.4 **Conclusions**

Primary system load increase due to the inadvertent opening of a feedwater heater manual bypass valve or the isolation of extraction steam to both first-point feedwater heaters is bounded by that assumed for the excessive load increase event presented in Section 14.2.8. The excessive load increase event evaluates the consequences of a 10% step load increase from full power. The feedwater temperature reduction event is shown to be bounded by the excessive load increase event.

Representative transient results for excessive load increases due to reduced feedwater temperature and excessive feedwater flow indicate that a core power increase is accompanied by a reactor coolant system average temperature decrease. This has the effect of maintaining an adequate margin to the applicable SAL (Section 3.2.3). It has been shown that the maximum reactivity insertion rate that occurs at no load following excessive feedwater addition is less than the maximum value considered in the analysis of a control rod assembly withdrawal incident from a subcritical condition. It has further been shown that automatic action occurs to prevent the continuous addition of cold feedwater after a unit trip. The event acceptance criteria (DNBR greater than the applicable SAL, reactor coolant system and main steam system pressures less than 110% of design limits, no event propagation) are satisfied for the feedwater malfunction that results in either an increase in feedwater flow or a decrease in feedwater temperature.

14.2.8 Excessive Load Increase Incident

An excessive load increase (ELI) incident is defined as a rapid increase in steam generator flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp-load increase, without a reactor-trip, in the range of 15% to 100% of full power. Any loading rate in excess of these values may cause a reactor trip to be actuated by the reactor protection system. If the load increase exceeds the capability of the reactor control system, the transient is terminated in sufficient time to prevent the DNBR from being reduced below the SAL (Section 3.2.3), since the core is protected by the combination of the nuclear overpower and the overpower-overtemperature trips discussed in Chapter 7, although the analysis conservatively does not credit the latter trip. An excessive load increase incident could result from either an administrative violation, such as excessive loading by the operator, or an equipment malfunction in the steam bypass control or turbine speed control.

For excessive loading by the operator or by system demand, the turbine load limiter keeps the maximum turbine load from exceeding 100% rated load.

During power operation, steam bypass to the condenser is controlled by reactor coolant condition signals; high reactor coolant temperature indicates a need for steam bypass. A single controller malfunction does not cause steam bypass; an interlock blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

14.2.8.1 **Method of Analysis**

Three cases were analyzed to demonstrate the unit behavior for a 10% step increase from 2546 MWt. The first two cases were at end-of-life (EOL) conditions, when the moderator temperature coefficient (MTC) for the core is assumed to be at its most negative limit of -45 pcm/°F, with and without automatic rod control. The third case was at beginning-of-life (BOL), with the MTC at its least negative limit of 0.0 pcm/°F and automatic rod control. Previous analyses indicate that a BOL case without automatic rod control is bounded by the other cases. The analyses were performed using the RETRAN code to provide a detailed simulation of the RCS, core kinetics, and the feedwater and steam systems. Following the RETRAN calculation of the RCS transient initiated from nominal conditions, the core thermal hydraulic code COBRA was used to compute the statistical minimum DNBR as a function of time for the SIF product. The VIPRE-D computer code was used to compute the MDNBR at the limiting statepoints for the 15 x 15 Upgrade fuel design from the RETRAN forcing functions used in the COBRA analysis for the 15 x 15 SIF product.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 35). This DNBR penalty is conservative and applicable to all statistically treated DNBR events and will be deducted from the retained margin for the SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 17).

The ability of the automatic rod control system to withdraw rods from the core has been eliminated. This prevents addition of positive reactivity by rod withdrawal in response to transient event conditions thereby resulting in lower power during the event. The event cases with automatic rod control are retained herein, even though they are not credible, as they continue to bound the current plant design.

14.2.8.2 **Results**

Figures 14.2-27, 14.2-28, 14.2-29, and 14.2-30 illustrate the results of the ELI transient with the reactor in manual control at EOL conditions, while Figures 14.2-31, 14.2-32, 14.2-33, and 14.2-34 represent the same event under automatic control. As expected, in the manual control case the decrease in RCS pressure and temperature is much more pronounced due to the high moderator temperature feedback. Under automatic control, rod movement will significantly retard the decrease in pressure and temperature. The nuclear power levels off at approximately 110% of 2546 MWt in both cases to balance the steam flow; but it does so sooner under automatic control. The transient DNBR decreases initially and flattens out as the power equilibrium is reached. Rod control has only a minimal effect on the magnitude of the minimum DNBR.

The third case, at BOL under automatic control with enhanced rod worth, is represented in Figures 14.2-35, 14.2-36, 14.2-37, and 14.2-38. The behavior is similar to that of the second case above. As the moderator feedback is assumed to be negligible at BOL, the reactivity to counteract the overcooling effect comes entirely from the control rods. Although RCS pressure and

temperature drop initially, they recover and rise to a comparable level later in the transient, and are expected to reach an equilibrium, if the transient is followed long enough.

14.2.8.3 Conclusions

The three cases analyzed have considerable margin to the applicable SAL (Section 3.2.3). It is concluded that unit integrity is maintained throughout lifetime for the excessive load increase incident.

14.2.9 Loss of Reactor Coolant Flow

14.2.9.1 Flow Coastdown Incidents

A loss-of-coolant-flow incident can result from a mechanical or electrical failure in a reactor coolant pump or from an interruption in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect is a rapid increase in coolant temperature.

This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provides the necessary protection against any loss-of-coolant-flow incident:

1. Low reactor coolant flow.
2. Reactor Coolant Pump motor circuit breaker opening,
3. Low voltage on pump power supply busses, and
4. Low frequency on pump power supply busses (opens RCP supply breakers).

Of these, only the low reactor coolant flow reactor trip is assumed in the analysis. The low frequency and low voltage signals are not credited for reactor protection, but are assumed to trip the RCPs at their appropriate setpoints. They provide diverse backup protection for loss of flow accidents. Even though these reactor protection system inputs do not meet IEEE-279 requirements, no credible failure mechanism has been identified which would impact the operability of the reactor protection system.

The reactor trip setpoints and their redundancy are further described in UFSAR Section 7.2, *Reactor Protection System*.

The simultaneous loss of electric power to all reactor coolant pumps at full power is the most severe credible loss-of-coolant-flow condition. For this condition, reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent reactor coolant system overpressure and the DNBR from being reduced below the applicable SAL (Section 3.2.3).

The following discussion presents the loss-of-flow analysis performed for operation at 2546 MWt. This analysis does not include cases for two loop operation.

14.2.9.1.1 Method of Analysis

The two limiting cases that were analyzed are as follows:

1. Loss of three out of three RCPs from a power level of 2546 MWt, due to an undervoltage condition.
2. Loss of three out of three RCPs from a power level of 2546 MWt, due to a frequency decay condition (-5 Hz/sec).

Partial losses of flow from the loss of fewer than three reactor coolant pumps are protected by the same low flow reactor trip. Because of the identical protection setpoint, and correspondingly higher coolant flow rates throughout the transient, the partial loss of flow events are less limiting than the complete loss of flow events. Therefore, the partial loss of flow events are bounded by the complete loss of flow analyses and no specific partial loss of flow analyses are run.

The above analyses assume core characteristics associated with the 15 x 15 SIF fuel product. The analysis incorporates the *Statistical DNBR Evaluation Methodology* (Reference 17).

The normal power supplies for the pumps are three buses supplied by the generator. Each bus supplies power to one pump. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps continue to supply coolant flow to the core. The simultaneous loss of power to all reactor coolant pumps is a highly unlikely event. Following any turbine trip, where there are no electrical faults that require tripping the generator from the pump supply network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made. Since each pump is on a separate bus, a single-bus fault would not result in the loss of more than one pump.

A full unit simulation with RETRAN (Reference 12) is used in the analysis to compute the core average and hot-spot heat flux transient responses, including flow coastdown, temperature, reactivity, and control-rod assembly insertion effects.

These data are then used in a detailed thermal-hydraulic computation using the Virginia Power COBRA code (Reference 13) to compute the DNB margin for the SIF product. This computation solves the continuity, momentum, and energy equations of fluid flow, together with the WRB-1 DNB correlation discussed in Section 3.4.2. The assumptions made in the calculations are discussed below. The VIPRE-D computer code was used to compute the MDNBR and DNBR margin for the LOFA statepoint for the 15 x 15 Upgrade fuel design using the initial conditions specified in 14.2.9.1.2.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 35). This DNBR penalty is conservative and applicable to all statistically

treated DNBR events and will be deducted from the retained margin for the SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 17).

14.2.9.1.2 Initial Operating Conditions

The initial conditions which are assumed in the analysis are presented below. They are consistent with the statistical treatment of key analysis parameters for the 15 x 15 SIF analysis. (See Section 3.4.3.2).

1. Key thermal/hydraulic parameters used in analysis - 3 loops operating - 15 x 15 SIF:

Power	2546 MWt
Pressure	2249.7 psia
Inlet Temperature	541.9°F
Minimum Measured Flow	273,000 gpm

2. Thermal/Hydraulic Conditions used for the analysis of 15 x 15 Upgrade Fuel Design

Power	2589.3 MWt
Pressure	2250.0 psia
Inlet Temperature	540.7°F
RCS Flow Rate	273,000 gpm

14.2.9.1.3 Reactivity Coefficients

A least negative Doppler Temperature Coefficient (-1.0 pcm/°F) and most positive Moderator Temperature Coefficient (+6 pcm/°F) were assumed since these result in higher heat flux at the time of minimum DNBR. The sensitivity to the effective delayed neutron fraction was evaluated. A minimum delayed neutron fraction was used because it produced the most limiting DNBR.

14.2.9.1.4 Reactor Trip

Following the loss of flow induced by underfrequency or undervoltage, the reactor is assumed to trip on low flow in any loop. This trip meets the IEEE-279 criterion and therefore cannot be negated by a single failure. Neither the low voltage nor low frequency trip circuits meet the IEEE-279 criterion from sensor to trip and are therefore considered backup trips. The low flow trip setting is 90% of full loop flow; the trip signal is assumed to be initiated at 87% of minimum measured flow, allowing 3% for instrumentation errors. It is also assumed that, upon reactor trip, the most reactive control rod assembly is stuck in its fully withdrawn position, resulting in a minimum insertion of negative reactivity. The assumed trip reactivity was 4.0% $\Delta k/k$, which is confirmed to be bounding for each reload cycle.

14.2.9.1.5 Flow Coastdown

Reactor coolant flow coastdown curves for the limiting undervoltage and underfrequency induced loss of flow accidents are shown in Figures 14.2-39 and 14.2-40, respectively. The flow profile for the undervoltage transient includes an initial 2% flow penalty to account for the potential of a “back EMF” phenomenon prior to the trip of the RCP. The RCP will maintain flow at or above 98% for undervoltage conditions less severe than the undervoltage trip setpoint. This is modeled by a prompt drop in flow from 100% to 98% of minimum measured flow followed by a five second delay prior to the RCP trip on (undervoltage). The reactor is not assumed to trip until the low flow setpoint has been reached.

14.2.9.1.6 Results

Both the underfrequency and the undervoltage trip events were analyzed. The two events were found to have nearly identical values of minimum DNBR. The minimum DNBRs for the two accidents showed a considerable margin to the applicable SAL (Section 3.2.3).

The transient responses of power, inlet temperature, average temperature, and pressurizer pressure are plotted in Figures 14.2-41 through 14.2-44 for the undervoltage case and 14.2-45 through 14.2-48 for the underfrequency case.

14.2.9.1.7 Conclusions

The analyses performed have demonstrated that for the above loss of flow incidents, the DNBR does not decrease below the applicable SAL (Section 3.2.3) at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

14.2.9.2 **Locked Rotor Incident**

14.2.9.2.1 Identification of Causes and Accident Description

The Locked Rotor/Sheared Shaft events are characterized by the rapid loss of forced circulation in one Reactor Coolant System (RCS) loop. A Locked Rotor event is defined as the seizure of a Reactor Coolant Pump (RCP) motor due to a mechanical failure. The Sheared Shaft event is defined as the separation of the RCP impeller from the motor due to the severance of the impeller shaft. For both the Locked Rotor and the Sheared Shaft events, the postulate RCP failure causes the reactor coolant flow rate to decrease more rapidly than a normal RCP coastdown.

During power operation the reduction in RCS flow caused by a Locked Rotor or Sheared Shaft event results in degradation of the heat transfer between the fuel and the reactor coolant, and between the reactor coolant and the secondary coolant in the steam generator (SG). As a result of the reduced fluid velocity, the core differential (ΔT) and average temperatures (T_{avg}) increase. The reduced heat transfer to the secondary fluid also contributes to the reactor coolant temperature increase. The expansion of the RCS fluid that accompanies the temperature increase causes an insurge of coolant into the pressurizer, and thus an increase in the reactor coolant system pressure. The reduced fluid velocity and subsequent temperature rise also act to reduce the

heat transfer from the fuel, causing the fuel temperature to increase. Fuel damage could then result if specified acceptable fuel damage limits are exceeded during the transient, i.e., if the fuel experiences a Departure from Nucleate Boiling (DNB). Due to the severe nature of these postulated failures, the likelihood that a limited number of fuel rods will experience DNB is significant. Thus, timely actuation of the Reactor Protection System (RPS) is required to help limit the number of potential fuel failures.

The immediate core power response during a Locked Rotor or Sheared Shaft event will change in accordance with the RCS temperature and pressure based on the magnitude and direction of the moderator reactivity feedback. As such, a Locked Rotor or Sheared Shaft event occurring in the presence of a positive Moderator Temperature Coefficient (MTC) will see an increase in core power at the RCS temperature increases. Conversely, the presence of a negative MTC will cause the core power to decrease as the RCS temperature increases. If the Rod Control System is in automatic, movement of the control rods will generally be in a direction such that a power reduction occurs.

The core power response is also influenced by the magnitude of the fuel Doppler coefficient. The reduced capability of the reactor coolant to remove energy from the reactor core causes the fuel temperature to increase. In the presence of a negative fuel Doppler coefficient, a fuel temperature increase contributes negative reactivity to the core, which acts to diminish the core power increase.

The potential for a Locked Rotor or Sheared Shaft event is present during all modes of operation where at least one RCP is functioning to provide forced circulation. However, the consequences of a Locked Rotor or Sheared Shaft event are reduced dramatically when the reactor is not at power. During subcritical or zero power operation, natural circulation is more than adequate to remove decay heat following the loss of forced circulation. Thus, the potential for exceeding the specified fuel design limits is nearly zero when the reactor is not at power.

Maintaining the fuel cladding integrity is a primary concern for the Locked Rotor/Sheared Shaft event, although integrity may not be maintained for all fuel rods. Therefore, maintaining the RCS as a fission product barrier becomes more significant. Specifically, RCS integrity may be challenged as a result of the volumetric expansion of the fluid caused by the heating of the RCS fluid. Operation of the pressurizer sprays and Power Operated Relief Valves (PORVs) can help limit the impact of the subsequent pressure increase, but cannot counteract the volumetric expansion of the RCS fluid. In general, the short duration of the Locked Rotor event acts in concert with the functioning of the Pressurizer Safety Valves (PSVs), to prevent excessive RCS pressurization. Thus, timely actuation of the RPS is also required to help limit the RCS pressure response.

Sensible and decay heat can be removed by steaming to the condenser through the steam bypass system, to the atmosphere through the Main Steam (MS) PORV or the Main Steam Safety Valves (MSSVs), or any combination of the three methods. However, the desirability of a given

method is based on system availability and the extent to which the fission product barriers have been compromised. In all scenarios, feedwater remains available to the Steam Generators (SGs) from either the Main Feedwater (MFW) System or the Auxiliary Feedwater (AFW) System to replenish the secondary coolant. Shortly after the reactor is shut down, the energy removal capability of the SGs will exceed the RCS sensible and decay heat levels, and the reactor operators/automatic control systems will function to maintain the plant at the new equilibrium condition.

The use of the ZIRLO and Optimized ZIRLO (References 31 and 39) alloy in Surry fuel assemblies has a negligible effect on the number of rods in DNB or the peak RCS pressure, and has a negligible effect on the total Zirconium/water reaction compared to Zircaloy. Therefore, the analysis for the 15 x 15 SIF product or the 15 x 15 Upgrade fuel design remains applicable, and reanalysis of the locked rotor event was not required for the implementation of this cladding material.

14.2.9.2.2 Method of Analysis

14.2.9.2.2.1 General. To cover all applicable phases of plant operation, Locked Rotor and Sheared Shaft events during Cold Shutdown, Intermediate Shutdown, Hot Shutdown, Reactor Critical (manual rod control), and Power Operation (automatic and manual rod control modes) are considered. A transient analysis is only required for the Locked Rotor and Sheared Shaft events at full power with manual rod control. The results for a Locked Rotor or Sheared Shaft event at any of the remaining operating conditions are bounded by those of the full power manual rod control case.

Except where otherwise noted, the following assumptions are made in the Locked Rotor/Sheared Shaft transient analysis:

1. The DNB analysis employs a statistical treatment of key analysis uncertainties; the transient cases are initiated from the condition listed in Section 14.2.9.1.2.
2. The main steam and RCS overpressurization analyses employ a deterministic treatment of key analysis uncertainties (102% of 2546 MWt or 100.38% of 2587 MWt; nominal T_{avg} +4°F; nominal pressurizer pressure +30 psi; and Thermal Design Flow).
3. Reactor protection is assumed to be provided by the low coolant loop flow rate reactor trip at 87% of the applicable analysis flow rate. A 1.0-second trip delay is assumed.
4. The analysis supports a moderator temperature coefficient (MTC) core design limit of +6.0 pcm/°F from 0% to 50% power and a linearly decreasing limit to 0.0 pcm/°F at 100% power. The analysis is non-limiting at EOC.
5. Unaffected reactor coolant pumps were assumed to trip 2.0 seconds after reactor trip on low loop coolant flow. The inertia of the unaffected pumps was conservatively reduced by 10% from the design value.

6. In the DNB transient analyses, the turbine trip following reactor trip was conservatively assumed to not function. In the main steam and RCS overpressurization transient analyses, the turbine trip following reactor trip was conservatively assumed to actuate.

7. Manual rod control was assumed.

8. In the DNB transient analyses, the pressurizer sprays and PORVs are conservatively assumed to be operable. In the main steam and RCS overpressurization transient analyses, the pressurizer sprays and PORVs are conservatively assumed to not actuate.

9. The RCS overpressurization analysis assumes 50% bypass flow. The high degree of bypass flow in the overpressurization cases compensates for the uncertainty associated with the thermal/hydraulic behavior of the core due to coolant voiding during a locked rotor event.

14.2.9.2.2.2 Transient Analysis for DNB. The transient analysis for DNB considerations utilizes the RETRAN transient analysis code (Reference 12) and the COBRA IIIC/MIT detailed core thermal/hydraulics code (Reference 13) to analyze the SIF product. The 15 x 15 Upgrade fuel design was analyzed using the VIPRE-D detailed core thermal/hydraulics code (Reference 36). The WRB-1 critical heat flux correlation (Reference 30) is used in the analysis.

The transient analysis for DNB is performed to determine the number of fuel pins that experience DNB as a result of a Locked Rotor or Sheared Shaft event. A fuel pin is assumed to fail if the predicted MDNBR is less than the applicable SAL (Section 3.2.3). The Locked Rotor DNB event scenario is therefore designed to produce the most limiting DNB response. From an analytical perspective, this goal is achieved by choosing initial conditions and analysis assumptions that will maximize coolant temperature and the power-to-flow ratio and minimize pressure during the event.

The analysis results demonstrate that no rods have a calculated MDNBR less than the applicable SAL (Section 3.2.3). The radiological dose consequences analysis for the locked rotor event assumes 1.4% of the rods fail. Figures 14.2-49 through 14.2-51 provide transient results for core inlet mass flow rate, core heat flux and core inlet temperature from the limiting DNBR analysis case.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 35). This DNBR penalty is conservative and applicable to all statistically treated DNBR events and will be deducted from the retained margin for the 15 x 15 SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 17).

14.2.9.2.2.3 Transient Analysis for RCS and Main Steam Overpressurization. The transient analysis for RCS and main steam overpressurization considerations also utilizes the RETRAN transient analysis code. The transient analysis for overpressurization considerations verifies that the peak RCS pressure (intact cold leg pump exit pressure) and peak main steam pressure (intact loop steam generator pressure) remain below 110% of RCS and main steam design pressure

(2750 psia and 1210 psia, respectively). The Locked Rotor overpressurization event scenario is designed to produce the most limiting overpressurization response. From an analytical perspective, this goal is achieved by choosing initial conditions and analysis assumptions that will minimize RCS energy removal and maximize core coolant expansion during the transient.

Figures 14.2-52 and 14.2-53 provide transient results for RCS pressure and steam generator pressure from the limiting pressurization analysis cases.

14.2.9.2.3 Conclusions

For the scenarios for which a transient analysis was performed, the following conclusions are applicable:

1. Acceptable offsite dose consequences are ensured, since the analysis demonstrates that the fraction of fuel rods predicted to experience Departure from Nucleate Boiling (DNB) is less than that which provides acceptable offsite dose analysis results.
2. Reactor Coolant System (RCS) integrity is maintained throughout the transient as demonstrated by analysis of transient RCS pressure. Specifically, the maximum RCS pressure, which occurred in the intact cold leg pump exit, remained below 2750 psia throughout the transient.
3. Main Steam System (MSS) integrity is maintained throughout the transient as demonstrated by analysis of transient MSS pressure. Specifically, the maximum main steam pressure, which occurred in the intact loop steam generator, remained below 1210 psia throughout the transient.

14.2.9.2.4 Environmental Consequences of Locked Rotor Accident (LRA)

The Locked Rotor Accident (LRA) evaluates the consequences of the sudden seizure of the rotor of one of the reactor coolant pumps. Similar results would be expected for a shear failure of a shaft in the reactor coolant pump. In these types of accidents, flow through the affected loop reduces rapidly while the core is still at power, and some degree of reverse flow would be expected through the affected loop. The low flow in the affected loop leads to a reactor and turbine trip, but the partial loss of flow while the core is at power results in a degradation in heat transfer which could in turn result in fuel damage.

Although there is no increase in the leakage of primary coolant to the secondary side in the LRA, activity (from the failed fuel) may be transported to the secondary side via any preexisting leaks in the steam generators. If there is a loss of offsite power, activity is released to the atmosphere through the steam generator safety valves and/or the power operated relief valves (PORVs) until the plant cools down and the reactor is secured in a safe condition.

14.2.9.2.4.1 LRA Analysis Assumptions. For this analysis the reactor is initially assumed to be operating at 2605 MWt, which is a higher value than the 100.38% of rated power required for the analysis. A turbine trip and coincident loss of offsite power are incorporated into the analysis,

which is consistent with Regulatory Guide 1.183 (Reference 32). With the assumed loss of offsite power, releases are through the steam generator PORVs and safety valves.

Coolant activities are based on 1.4% failed fuel, in accordance with the Alternative Source Term (AST) described in Regulatory Guide 1.183.

The possibility of uncovery of the upper portion of the steam generator tube bundle (based on collapsed liquid levels) during a LRA was not considered in previous LRA dose calculations. For the current evaluation, the approach taken was that developed by the Westinghouse Owners Group (Reference 19). This approach considers that the probability of coincidental occurrence of a LRA, a preexisting steam generator tube leak above the collapsed liquid level and condenser unavailability due to a loss of offsite power is sufficiently small that it is not necessary to evaluate this combination of conditions. Therefore, any leaks in the steam generator tube bundle were assumed to remain covered throughout the accident.

When the tubes are covered, the secondary side water provides a scrubbing action, trapping some of the activity from iodine in the primary fluid in the secondary liquid. The noble gases are unaffected by this process. Whenever the steam generator tubes are covered, this analysis uses an iodine partitioning factor of 0.01 to account for this effect. The value of this partitioning factor is given in Regulatory Guide 1.183, Appendix E for the Main Steam Line Break Accident, which has a release mechanism similar to that seen in the LRA. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators. It is assumed that there is 99% retention of particulates in the SG liquid. Moisture carryover is significantly less than 1% under post accident conditions; hence, assuming 99% retention in the SG liquid is conservative.

For a LRA where the PORVs cycle open and closed as designed, conditions which would generate an SI signal are not created. Without an SI signal, the Surry control room is not automatically isolated. Therefore, the analysis assumes that the control room is not isolated during the duration of the accident and control room air is drawn in through the normal intakes. The normal ventilation intakes are closer to the release point than the emergency ventilation intakes, do not have iodine filtration and supply air to the control room at a greater flow rate than the emergency ventilation system. Therefore, this scenario is more conservative for control room dose calculations.

A stuck open PORV produces higher steam releases during the first two hours after a LRA which gives slightly higher exclusion area boundary (EAB) doses. Therefore, the LRA was analyzed based on the steam releases expected for the first two hours with a stuck open PORV for conservatism in the calculation of EAB doses, but assuming no SI signal for conservatism in the calculation of control room dose.

Briefly, the assumed sequence of events used in this current dose evaluation of a LRA at Surry is as follows. The accident is initiated when one reactor coolant pump rotor locks. Power to the other two reactor coolant pumps is assumed to be lost shortly thereafter, after the reactor trips.

Assuming the steam condensers are unavailable due to loss of offsite power, the PORVs on the two unaffected steam generators open within seconds of the accident; the PORV on the steam generator in the affected loop also opens within one minute. Most of the releases in the first minute are through the unaffected steam generators; after the third PORV opens, the steam release through all three steam generators is assumed to be essentially identical. Releases are conservatively modeled as starting immediately.

These releases are assumed to occur for 9 hours, by which time the Reactor Coolant System (RCS) temperature has been decreased to 350°F. At this point the Residual Heat Removal (RHR) System is activated, and releases to the atmosphere through the steam generator PORVs cease.

14.2.9.2.4.2 Initial Radioisotope Concentrations. In accordance with Regulatory Guide 1.183 the amount of activity released is dependent on the amount of activity released due to fuel failures.

The dose consequence of the Locked Rotor Accident analysis is based on the assumption of 1.4% failed fuel—that is 1.4% of the fuel in the core (briefly) enters DNB during the accident and is therefore assumed to fail. These fuel failures are assumed to occur instantaneously at the start of the accident. The total amount of activity in the primary coolant at the start of the LRA is then transported to the steam generators by primary-to-secondary leakage at 1 gpm. Table 14.2-2 gives the initial radionuclide inventories for the LRA. The calculation of the radionuclide inventories in the gaps of the 1.4% failed fuel rods is based on an assembly radial peaking factor of 1.70. It should be noted that the thermal/hydraulic analysis predicts no fuel failure as a result of a locked rotor accident.

14.2.9.2.4.3 Locked Rotor Accident LOCADOSE Model. The LOCADOSE computer code system (References 20 through 22), is used to calculate the doses for the LRA. The primary and secondary system volumes used in this analysis are given in Table 14.3-12. The leakage from the primary coolant to the secondary system through the steam generators was conservatively set at 1 gpm through all three steam generators. The LRA was modeled assuming that any leaks in the steam generator tube bundle remain covered throughout the accident. The primary coolant was therefore modeled as leaking to the steam generator liquid volume. The flow from the secondary liquid to the secondary steam assumes a partition factor of 0.01 for iodine and particulates.

In the LRA, most of the releases in the first minute are through steam releases from the two unaffected steam generators. After the third PORV opens (within one minute), the steam release through all three steam generators is assumed to be essentially identical. To simplify the modeling of this accident, the releases were treated as being identical through all three steam generators for the entire release period. The releases are also modeled as starting immediately, rather than a few seconds after initiation of the accident (when the PORVs open), and continue for 9 hours. The steam releases for the LRA are given in Table 14.2-3.

As noted above, if the PORVs cycle normally during a LRA, no SI signal is generated to isolate the control room. The analysis therefore models normal control room ventilation (at a

3000 cfm flow rate) and does not credit isolation. Also, no emergency ventilation supply is assumed to be used for the remainder of the 30-day period for which control room doses are calculated (Reference 26).

The EAB and low population zone (LPZ) atmospheric dispersion factors (χ/Q) were determined based on the PAVAN (NUREG/CR 2858) methodology using meteorological data for 1994 to 1998. The EAB χ/Q value is 1.76E-03 sec/m^3. The LPZ χ/Q values are shown in Table 14.5-7. The control room χ/Q value was calculated using the Reference 24 methodology. A control room χ/Q value 7.71×10^{-3} sec/m^3 was used, which reflects the distance between the release point and the normal control room ventilation intake. Control room occupancy factors were also incorporated into the dose calculations to reflect that personnel would not be exposed to the released activity 100% of the time over the entire 30-day period. The factors which were used were determined based on the Regulatory Guide 1.183 methodology and are given in Table 14.5-9. The breathing rate used for the control room dose calculations was 3.5×10^{-4} m^3/sec.

14.2.9.2.4.4 Results of the Dose Calculations for LRA. The dose calculations for a LRA with the model and assumptions described above are summarized in Table 14.2-4. The control room, EAB, and LPZ doses given in Table 14.2-4 are in Rem TEDE. The calculated control room, EAB, and LPZ doses for a LRA are less than the criteria specified by Regulatory Guide 1.183, and the offsite doses are below the limits specified in 10 CFR 50.67.

14.2.10 Loss of External Electrical Load

14.2.10.1 Identification of Causes and Accident Description

The loss of external electrical load may result from an abnormal variation in network frequency or other adverse network operating conditions. It may also result from a trip of the turbine generator or the opening of the main breaker from the generator that fails to cause a turbine trip but causes a large, rapid nuclear steam supply system load reduction by the action of the turbine control.

The unit is designed to accept a step loss of load from 100% to 50% without actuating a reactor trip. The automatic steam bypass system, with 40% steam dump capacity to the condenser, is able to accommodate this load rejection by reducing the severity of the transient imposed on the reactor coolant system. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift for the 50% step loss of load with condenser steam dumps.

In the event the steam bypass (condenser dump) valves fail to open following a large load loss or in the event of a complete loss of load with the steam dump operating the steam generator safety valves may lift and the reactor may be tripped on a high pressurizer pressure, high pressurizer level, or overtemperature delta-T signal. The steam generator shell-side pressure and

reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the reactor coolant system and main steam systems, respectively, against all load losses, including a complete loss of steam load without the bypass system (condenser dumps) or atmospheric dumps (main steam PORVs) available. The steam dump valves will not be opened for load reductions of 10% or less. For larger load reductions they may open.

The most likely source of a complete loss of load on the nuclear steam supply system is a trip of the turbine generator. In this case, there is a direct reactor trip signal (unless below approximately 10% power) derived from either the turbine autostop oil pressure or a closure of the turbine stop valves. Reactor coolant temperatures and pressure do not significantly increase if the steam bypass system and pressurizer pressure control system are functioning properly. However, in this analysis, the behavior of the unit is evaluated for a complete loss of load from full power without direct reactor trip. The analysis, presented below, shows the adequacy of the pressure relieving devices to prevent Main Steam System and Reactor Coolant System overpressurization and to show that no fuel damage occurs. The latter is demonstrated by conservatively requiring that the applicable SAL (Section 3.2.3) is met for the hottest rod in the core.

As will be shown, the reactor coolant system and Main Steam System pressure relieving devices have sufficient capacities to ensure the safety of the unit without relying on the mitigating capabilities of the Automatic Rod Control, Pressurizer Pressure Control or Main Steam Bypass Systems.

14.2.10.2 **Method of Analysis**

The complete loss of load transients are analyzed with the Virginia Power RETRAN (Reference 12) and COBRA (Reference 13) for the SIF product. The 15 x 15 Upgrade fuel design was analyzed for the event using the VIPRE-D detailed core thermal/hydraulics code (Reference 36) at the limiting statepoint.

The RETRAN model is used to perform the overall Reactor System transient analysis. The model describes the neutron kinetics, Reactor Coolant System including the pressurizer and pressurizer safety and relief valves and spray, and the Main Steam System including the steam generators and main steam safety valves. Outputs of the RETRAN analysis include reactor power level, temperatures and pressures at various points in the Reactor Coolant System, pressurizer water volume and Main Steam System pressure.

The COBRA or VIPRE-D models are used to calculate the detailed subchannel thermal conditions, including a time and position dependent Departure from Nucleate Boiling Ratio (DNBR) for the SIF product or 15 x 15 Upgrade fuel design.

A DNBR penalty of 3.3% will be applied to account for the uprate from 2546 MWt to 2587 MWt (Reference 35). This DNBR penalty is conservative and applicable to all statistically

treated DNBR events and will be deducted from the retained margin for the SIF product during the core reload thermal-hydraulic evaluation in accordance with NRC-approved methodology in VEP-NE-2-A (Reference 17).

14.2.10.3 **Initial Operating Conditions**

The following assumptions are made in the DNBR cases:

1. The behavior of the unit is evaluated for a complete loss of steam load from power at 2546 MWt for the 15 x 15 SIF product (2589.3 MWt for the 15 x 15 Upgrade fuel design) without a direct reactor trip to demonstrate core protection margins. A statistical treatment of key DNBR analysis parameter uncertainties is employed. Therefore, nominal initial RCS conditions are assumed, and allowances for calibration and instrument errors are incorporated into the limiting DNBR value as described in Statistical DNBR topical report (Reference 17).
2. A positive moderator temperature coefficient conservative for BOC conditions and a least negative Doppler temperature coefficient are assumed.
3. Credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure.
4. Main feedwater flow is isolated at the time of the turbine trip for the DNB case only.

The following assumptions are made in the Non-DNBR (RCS Pressure) case:

1. The behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip to demonstrate the adequacy of the pressure-relieving devices. A deterministic treatment of uncertainties in initial RCS operating conditions [e.g. pressure, temperature, flow, and core power (102% of 2546 MWt or 2597 MWt (i.e., 100.38% of 2587 MWt))] is used in the analysis.
2. A zero moderator temperature coefficient and a most negative Doppler temperature coefficient are assumed.
3. The reactor is assumed to be in manual control, which is conservative from the standpoint of maximum pressure attained.
4. Main feedwater flow is isolated at the time of the reactor trip.
5. The pressurizer safety valve tolerance is modeled with +3% PSV tolerance and 0.1 second delay. (Only the results of the overpressure transients are sensitive to the safety valve tolerance. The DNBR results are not sensitive to these parameters.)
6. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure.

The following assumptions are made in both the DNBR case and non-DNBR case:

1. No credit is taken for the operation of the steam dump system, steam generator power operated relief valves, or direct reactor trip on turbine trip. The reactor is tripped on high pressurizer pressure. The steam generator pressure rises to the safety valve setpoint, where steam release through safety valves limits secondary steam pressure to less than design limit.
2. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. The auxiliary feedwater flow would remove core decay heat following plant stabilization.
3. All cases examined assumed reactor is in manual rod control mode. This provides the limiting initial reactor power response to the event. In addition, all cases incorporate the assumption of 15% steam generator tube plugging.

14.2.10.4 **Results**

Only the BOC cases are presented here, since they provide the limiting results with respect to the analysis acceptance criteria of interest.

14.2.10.4.1 DNBR Case

Transient results for the RETRAN DNBR case are presented in Figures 14.2-54 to 14.2-58. These are discussed as follows:

Figure 14.2-54 - Nuclear power initially increases in the presence of the RCS heatup and the assumed positive moderator coefficient. Peak power reaches about 114% of 2546 MWt (112% of 2587 MWt) before the effects of reactor trip on high pressurizer pressure dominate.

Figure 14.2-55 - RCS inlet temperature increases by about 39°F prior to the excursion being terminated by reactor trip.

Figure 14.2-56 - Pressurizer liquid volume responds to the RCS heatup by increasing from 804 cubic feet to a maximum of about 1153 cubic feet leaving about 147 cubic feet of minimum steam space.

Figure 14.2-57 - Cold leg pressure follows a similar trend, reaching a peak value of 2674 psia at 15 seconds.

Figure 14.2-58 - Main steam pressure reaches a maximum value of 1174 psia (36 psia margin to the design limit) at 20 seconds. This case is expected to be limiting for main steam pressure.

The Hot Channel DNBR is shown to have considerable margin to the SAL (Section 3.2.3).

14.2.10.4.2 Non-DNBR (RCS Pressure) Case

This is the limiting RCS overpressure case. The results are presented in Figures 14.2-59 to 14.2-63. These are discussed as follows:

Figure 14.2-59 - Nuclear power does not exceed the initial value of 102% of 2546 MWt (100.38% of 2587 MWt) before decreasing in response to the reactor trip on high pressurizer pressure.

Figure 14.2-60 - RCS inlet temperature increases by about 30°F. Again the temperature increase is less than for the pressure control case because of the earlier trip.

Figure 14.2-61 - Pressurizer liquid volume responds to the RCS heatup by increasing from 854 cubic feet to a maximum of about 991 cubic feet leaving about 309 cubic feet of minimum steam space.

Figure 14.2-62 - Cold leg pressure follows a similar trend, reaching a peak value of 2669 psia (about 81 psi margin to the analysis limit) at 9 seconds.

Figure 14.2-63 - Main steam pressure reaches a maximum value of 1161 psia (49 psi margin to the design limit) at 17 seconds or slightly less than the primary pressure control case, as expected.

14.2.10.5 **Conclusions**

The analysis indicates that for a complete loss of external electrical load without a direct or immediate reactor trip the following criteria are met:

1. The minimum transient DNBR remains above the applicable SAL (Section 3.2.3).
2. Pressure at the most limiting RCS location is less than 110% of RCS design pressure, or 2750 psia (the Emergency Condition Stress Limit Specified in Section III of the ASME Code).
3. Pressure at the most limiting Main Steam System (MSS) location is less than 110% of MSS design pressure, or 1210 psia (the Emergency Condition Stress Limit specified in Section III of the ASME Code).

14.2.11 Loss of Normal Feedwater

A loss of normal feedwater (from a pipe break, pump failures, valve malfunctions, or loss of offsite ac power) results in a loss in the capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this incident, reactor core damage could possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater were not available for the unit, residual and sensible heat following reactor trip would heat the reactor coolant system water to the point at which water relief from the pressurizer relief valves occurs. A loss of significant water from the reactor coolant system could conceivably lead to core

damage. A special case of this event is a main feedwater line break in the main steam valve house (outside containment). The transient is described in Section 14B.6.

The following provides the necessary protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in any steam generator, unless the RCS loop stop valves are closed, or on water level below the AMSAC (ATWS Mitigation System Actuation Circuitry) setpoint in two steam generators after a time delay, providing the C-20 permissive is satisfied.
2. Reactor trip on a main steam flow-feedwater flow mismatch coincidental with a low water level in any steam generator.
3. The operation of two motor driven auxiliary feedwater pumps (350 gpm design flow for each), which can be started either automatically or manually. They are started automatically on:
   a. A low-low water level in one of three steam generators as sensed by two of three channels on that steam generator, unless the RCS loop stop valves for that steam generator are closed.
   b. The opening of one of two feedwater pump breakers on two of two main feedwater pumps.
   c. Any safety injection signal.
   d. The loss of all ac power, as indicated by an undervoltage on the two transfer buses corresponding to that unit's emergency buses.
   e. AMSAC initiation.
4. The operation of one turbine-driven pump (700 gpm), which can be started automatically or manually. It is started automatically on:
   a. A low-low level in two of three steam generators as sensed by two of three channels for each steam generator unless the loop stop valves for those steam generators are closed.
   b. Undervoltage on two of three 4160V ac station service buses for greater than 5 seconds.
   c. AMSAC Initiation.

The motor-driven auxiliary feedwater pumps are supplied by the diesel generators if a loss of outside power occurs, and the turbine-driven pump uses steam from the steam generators. The turbine exhausts the steam to the atmosphere. The auxiliary feedwater pumps take suction directly from the 110,000-gallon emergency condensate storage tank for delivery to the steam generators.

The above provides functional diversity in equipment and control logic to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater, including that caused by a loss of ac power.

14.2.11.1 **Method of Analysis**

A detailed analysis using the RETRAN Code (Reference 12) was performed to obtain the plant transient following a loss of normal feedwater (LONF). The LONF analysis includes sensitivities on the operation of pressurizer heaters, sprays, and power operated relief valves for the effect on the pressurizer fill and RCS overpressure criteria.

The following assumptions were made:

1. Reactor trip occurs when the steam generator water level reaches the narrow range low-level tap in the steam generator.
2. The plant is operating at 102% of 2546 MWt (100.38% of 2587 MWt).
3. The core residual heat generation is based upon long-term operation at the initial power level preceding the trip.
4. The loss of alternating current power case assumes offsite power becomes unavailable at the time reactor trip occurs. The reactor coolant pumps are tripped off coincident with reactor trip.
5. For offsite power available - two motor-driven auxiliary feedwater pumps are available 1 minute after the accident. The pumps are capable of providing 250 gpm of auxiliary feedwater per pump. The turbine driven auxiliary feedwater pump is assumed inoperable.

   For loss of offsite power - one motor-driven auxiliary pump is available 1 minute after the accident. The pump is capable of providing 300 gpm of auxiliary feedwater. The turbine driven auxiliary feedwater pump is assumed inoperable and the second motor-driven auxiliary feedwater pump is not readily available, within the 1 minute timeframe.
6. Auxiliary feedwater is distributed to the steam generators through a common header.
7. Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief is typically through the power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for conservatism, these components are assumed unavailable.
8. The initial reactor coolant average temperature is 4°F higher than the nominal value, since this results in a greater expansion of reactor coolant system water during the transient and a higher water level in the pressurizer.
9. An uncertainty of 8.5% in the full-power programmed pressurizer level is assumed. It should be noted with regard to this incident that even if the pressurizer does fill, the low surge rate would not cause an excessive pressure rise.
10. Initial pressurizer pressure is 30 psi above its nominal value.

11. The analysis is performed with and without alternating current power to the station auxiliaries.

14.2.11.1.1 Case 1 - Offsite Power Unavailable

Figures 14.2-64 through 14.2-67 show the unit parameters following a loss of normal feedwater incident according to the assumptions listed above. Following the reactor and turbine trip, the water level in the steam generators will fall because of a reduction of the steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. For the limiting case, one minute following the initiation of the low-low level trip, one auxiliary feedwater pump is automatically started, reducing the rate of water level decrease. The capacity of the auxiliary feedwater pump is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valves.

The loss of alternating current power (LOAC) is a special case of the LONF event from an analysis standpoint. The LONF event followed by a reactor coolant pump trip on low-low steam generator water level conservatively bounds the LOAC event. Figures 14.2-64 through 14.2-67 present pressurizer pressure, pressurizer water volume, RCS loop temperature, and core inlet flow rate, respectively, for a case assuming the pressurizer heaters are operational. The analysis of the loss of normal feedwater event demonstrates that the auxiliary feedwater system will remove the stored and residual heat, thus preventing overpressurization and liquid relief of RCS inventory through the pressurizer safety valves or pressurizer power operated relief valves.

14.2.11.1.2 Case 2 - Offsite Power Remains Available

The offsite power available case assumes continuous operation of the reactor coolant pumps. All other assumptions are consistent with those cited earlier. Figures 14.2-68 through 14.2-71 present pressurizer pressure, pressurizer water volume, RCS loop temperature, and core inlet flow rate, respectively, for a case assuming the pressurizer heaters are operational. This case demonstrates the adequacy of the long-term heat removal capability of the AFW System.

14.2.11.2 **Conclusions**

The loss of normal feedwater does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer relief or safety valves, nor does it result in an uncovering of the tube sheets of the steam generators being supplied with water. A long term decrease in the pressurizer water volume is shown, peak RCS pressure does not exceed 2750 psia, main steam pressure is less than 1210 psia, and the total secondary liquid inventory of the three steam generators does not decrease below 15,000 lbm.

14.2.12 Loss of All Alternating Current Power to the Station Auxiliaries

In the event of a complete loss of offsite power and a turbine trip, there would be a loss of power to the unit auxiliaries (i.e., the reactor coolant pumps, main feedwater pumps, etc.). The events following a loss of ac power with turbine trip are as follows:

1. Unit vital instrument loads are supplied by the emergency power sources.
2. As the steam system pressure increases, the steam system power-operated relief valves are automatically opened to the atmosphere. (Steam bypass to the condenser is assumed to be unavailable, since the steam bypass is not required for reactor protection.)
3. If the steam flow rate through the power-operated relief valves is not sufficient (or if the power relief valves are not available), the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.
4. As the no-load temperature is approached, the steam power-operated relief valves (or self-actuated safety valves if the power-operated relief valves are not available for any reason) are used to dissipate the residual heat and to maintain the unit in the hot-shutdown condition.
5. The emergency diesel generators will start on a loss of voltage on the emergency 4160V buses to supply unit vital loads.

The auxiliary feedwater system is started automatically as discussed in Section 14.2.11. The steam-driven auxiliary feedwater pump uses main steam and exhausts to the atmosphere. The motor-driven auxiliary feedwater pumps are supplied by power from the diesel generators. The pumps take suction directly from the 110,000-gallon emergency condensate storage tank for delivery to the steam generators. The auxiliary feedwater system ensures a feedwater supply of at least the 300 gpm value assumed in the analysis upon loss of power to the station auxiliaries.

The auxiliary steam turbine-driven feedwater pump has a nominal capacity of 700 gpm and the motor driven auxiliary feedwater pumps have a nominal capacity of 350 gpm each.

The steam-driven pump can be tested at any time by admitting steam to the turbine driver. The motor-driven pumps also can be tested at any time. The valves in the system can be operationally tested at any time.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and for the removal of residual heat is maintained by natural circulation in the reactor coolant loops. The natural circulation flow was calculated for the conditions of equilibrium flow and maximum loop flow impedance. The results given by the model are within 15% of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant and confirmed at San Onofre and Connecticut Yankee. The natural circulation flow ratio as a function of reactor power is given in Table 14.2-1.

It is shown in Section 14.2.11 that a loss of normal feedwater from any cause, including a loss of offsite ac power, does not result in water relief from the pressurizer relief or safety valves.

The loss of ac power to the station auxiliaries does not cause any adverse condition in the core since it does not result in water relief from the pressurizer relief or safety valves.

14.2.13 Likelihood of Turbine-Generator Unit Overspeed

A turbine missile can be generated by a rotor fracture releasing fragments capable of causing significant damage. A large rotor fragment is interpreted to be a sector of a rotor disk forging, of between 90° and 180° included angle, separated by fracture along several radial-axial planes and rupture of the welds.

The material used for the six disk forgings of each LP rotor is a 2%Cr-Ni-Mo steel. The absence of defects of any significant size from the disk forgings, as purchased, is ensured by stringent ultrasonic inspection. The rotors are of welded construction designed to ensure long-term integrity. Each rotor consists of six separate forgings, joined at their outside diameters by submerged-arc welding. In each of the opposed flows, a center disk carries the first six stages of moving blades, the intermediate disk carries the penultimate stage blades and the last disk carries the last stage blades. The improved welded rotor design characteristics include low yield strength material, no shrunk-on disks, homogeneous properties due to small volume of each disk, and verification of absence of material defects by high resolution ultrasonic inspection performed on small size of each forging. These design features contribute to the elimination of the risk of rotor fracture.

There are two quite different circumstances in which the risk of rotor fracture may arise, which are categorized as high-speed burst and low-speed burst events.

The high-speed burst could occur if there is an accidental loss of electrical load concurrent with the failure of turbine protection system components. If the HP turbine governor valves were not automatically closed then the rotor would accelerate to a speed approaching twice the normal running speed. At this speed there is a high probability that LP rotors would fracture, releasing fragments. The low probability of such an event is determined by the low probability of this overspeed ever occurring. The probability of a high-speed burst is unaffected by the turbine retrofit due to the high reliability of the control system components.

A low-speed burst could occur due to a mechanism of deterioration leading to the progressive weakeninq of the rotor which may fail at normal speed, or at a low overspeed. This includes 10 percent above normal speed during periodic overspeed trip tests at no load and 20 percent above normal speed following loss of electrical load and an overspeed trip. The low probability of such an event is determined by design of the rotors to ensure long-term integrity and by periodic inspection to detect any sign of deterioration.

The Alstom Power methodology for the turbine missile generation probability calculation is included in Alstom Standard STD0010572, Reference 40.

The methodology used in this report is the same methodology used by ABB (now Alstom) in the missile analysis report for the Maine Yankee Unit and several others for US utilities. This missile generation probability methodology was the basis for the change in the turbine rotor inspection frequency requested in Maine Yankee Technical Report Amendment 134 to the NRC. The NRC approved Maine Yankee Amendment 134 and stated in their approval that the ABB's probability analysis (turbine missile generation probability calculation) is consistent with NRC approved methodology.

Alstom Standard STD0011103, Reference 44, confirmed that the missile generation probability methodology in Alstom Standard ST0010572 that is used for Surry Power Station is the same missile generation probability methodology as that approved for Maine Yankee.

Therefore, use of the Alstom Power methodology for turbine missile generation probability calculations included in Alstom Standard ST00010572 is consistent with the NRC requirements included in NUREG 0800 for turbine missile generation probability calculations.

This methodology used by Alstom is for estimating the probability of low-speed rotor fracture and/or missile generation due to stress corrosion cracking. The following two failure modes were evaluated:

1. Growth of an Initial Defect by Fatigue

   The ultimate inspection standards ensure that any initial defect of significant size is detected and rejected. A conservative assumption is made that, despite this, a large embedded defect of 0.4 inch diameter remains in the disk in a location subject to the highest tangential stress. The evaluation indicates that the margin between a large extended defect size and the minimum critical crack size is a factor of more than 10, and there is no credible failure by this mechanism.

2. Initiation and Growth of Cracks due to Stress Corrosion Cracking (SCC)

   Design procedures developed using the Alstom Power Threshold Stress Approach (TSA), as described in Reference 41, indicate that any risk of stress corrosion cracking Surry retrofit LP rotors is eliminated by design and materials selection. LP rotors of welded construction type indicate no SCC in the relevant radial-axial plane which could extend to release large rotor fragments. Despite the fact that a rotor fracture has never occurred on an Alstom welded rotor, the residual risk of missile generation has been evaluated by probabilistic methods.

   The probability of rotor fracture has been determined by assigning probability distributions to the values of SCC growth rate, stress, and crack geometry. A Monte Carlo analysis was performed to determine the probability of failure. The calculation method and results of estimating the probability of missile generation resulting from the initiation and growth to a critical size of a SCC is described in the following sections:

a. Crack Initiation

The initiation probability is taken as a constant value calculated on the basis of the statistics of SCC cracks found in Alstom Power welded rotors during inspections after periods of service. The statistics are based on long-term service but no credit is assumed for design procedures introduced to eliminate susceptibility to SCC initiation as described in Reference 41.

Using the methodology of Reference 43, the best estimate of the probability of SCC initiation, for 50 percent confidence, is 0.00125. There have been a small number of SCC cracks which have initiated at the internal corners of the blade root slots. In the limited number of cases where this cracking has been observed, the rotors were produced before the application of the Alstom Power TSA, and the calculated stresses at the locations of the cracking have been found to exceed those permitted by TSA. No SCC cracking has been observed at a location where the calculated stress satisfies the requirements of the Alstom Power TSA, to which the Surry retrofit LP rotors have been designed.

b. Crack Growth Rate

Stress corrosion cracks appear after an initial period of exposure to stress in the presence of wet steam. It is conservatively assumed that any crack that initiates does so immediately on entering service and begins to grow immediately.

Based on the available data for stress corrosion cracks found in the power-industry service and determined from laboratory tests, the rate of crack growth under steady load can be described (Reference 43) as a log-normally distributed function of temperature and material yield stress.

The probability distribution of stress corrosion crack growth is illustrated in Figure 14.2-72. The rate of crack growth is independent of the crack stress intensity, and therefore independent of applied stress. The decrease in rate at low stress intensities is neglected in this analysis. The increase in rate at high stress intensities is dealt by assuming, very conservatively, that if the crack stress intensity reaches 100 ksi. $\sqrt{\mathrm{in}}$ (ksi square root inches is a unit in the category of fracture toughness, ksi. $\sqrt{\mathrm{in}}$ has a dimension of $\sigma\sqrt{L}$ where σ is applied stress in ksi and L is the crack length in inches) then the acceleration is so great that the fracture follows almost immediately. This is equivalent to reducing the assumed fracture toughness to 100 ksi. $\sqrt{\mathrm{in}}$ in calculating critical crack size at normal speed, as discussed below.

c. Critical Crack Size

The critical crack size at which rotor fracture will occur is dependant on crack geometry factor, rotor disk fracture toughness, and applied stress.

The minimum rotor disk fracture toughness guaranteed by the property specification of the most vulnerable rotor disks, the center disks, is 154 ksi. $\sqrt{\text{in}}$, and the actual values achieved are likely to be significantly higher. However, the calculation is performed by substituting lower value of 100 ksi. $\sqrt{\text{in}}$.

The crack geometry factor has limiting values of 1.99 for a parallel sided crack and 1.26 for a semi-circular crack. These values are appropriate to planar cracks and, conservatively, neglect any increase in critical crack size that might result from crack branching. Between these limits the geometry factor is assumed to be uniformly distributed, so that any value in this range is equally probable. The probability distribution is shown in Figure 14.2-73.

The applied stress is taken to be the mean tangential stress acting over the critical crack depth, determined from the tangential stress contours of Figure 14.2-74. For a given value of crack geometry factor, the mutually compatible values of crack depth and mean tangential stress over that crack depth which satisfy the condition that the crack tip stress intensity equals the limit of 100 ksi. $\sqrt{\text{in}}$ for stable crack propagation are calculated. The variation of this critical crack depth with the crack geometry factor is shown in Figure 14.2-75.

The distribution of tangential stress shown in Figure 14.2-74, which was determined by finite element analysis, is conservatively assumed to be subject to a calculational error of ±10% in line with Reference 43, and is assumed to be distributed linearly around the calculated value. The probability distribution and the corresponding cumulative probability distribution are shown in Figure 14.2-76.

d. Failure Probability

The probability of failure, i.e., that a stress corrosion crack grows to a size that could cause fast fracture of the rotor after any specified period of service, assuming immediate initiation, was determined using a Monte Carlo analysis. For each period of service, ranging from 60,000 hours to 150,000 hours, a large number of trial calculations were carried out in which each of the three variable parameters (SCC growth rate, crack geometry factor, and rotor tangential stress) were randomly selected from its associated probability distribution using randomly generated numbers between zero and one.

The probability of a rotor disk fracture due to SCC initiation and growth is determined as a function of time (Reference 42). Alstom recommends major rotor inspection intervals of 100,000 operating hours. The calculated probability of rotor fracture per unit year during the final year of operation prior to reaching 100,000 hours is 6.96×10^{-8}.

Alstom missile analysis (Reference 40) has concluded that the probability of lowspeed fracture for Surry Power Station LP turbine retrofit rotors is 6.96×10^{-8} per unit year which is less than one hundredth of the acceptable limit of 1×10^{-5} permitted for unfavorably orientated plants by the NRC guidelines. The LP turbine rotor will be inspected per Dominion's inspection program

based on the manufacturer's recommendations (Reference 42). This inspection verifies that the rotor will continue to meet the required design safety limit.

Dominion inspection requirements for LP rotors during major overhaul ensure that any indications of SCC which could develop to cause rotor fracture will be detected. The inspection includes a thorough visual inspection for erosion and corrosion damage and magnetic particle examination of selected areas to detect any cracking at the rotor surfaces. In the very unlikely event of surface indications being detected, additional ultrasonic examinations would be performed.

Alstom Power evaluated the probability of missile generation for Surry turbine using the methodology which was previously used for Maine Yankee unit. The methodology was approved by the NRC in Maine Yankee Amendment 134 Safety Evaluation Report. This methodology complies with the SRP Acceptance criteria of NUREG 0800 Section 3.5.1.3, Turbine Missiles.

The probability of missile generation from either unit is so low that it can be discounted and there is no missile strike envelope.

Because of the redundant means of overspeed protection and reliability of the turbine control protection system and of the main steam system, the possibility of unit speeds above the design value (120%) is very remote.

A description of the electro-hydraulic governing system and its operation is given in Section 10.3.3.

In addition to design provisions associated with the turbine control and protection system, the governor and main stop valves are exercised on a periodic basis during unit operation to further reduce the possibility of valve stem sticking. Analyses of oil samples are performed regularly.

The turbine is periodically oversped to check the tripping speed. The remaining tripping devices are routinely checked.

14.2 REFERENCES

1. Westinghouse Electric Corporation, *Power Distribution Control in Westinghouse Pressurized Water Reactor*, WCAP-7208, 1968.
2. Letter from C. M. Stallings, Vepco, to K. R. Goller, NRC, Subject: Unit 2 Cycle 2 Reload Evaluation, dated March 12, 1975 (Serial No. 458).
3. Westinghouse Electric Corporation, *Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)*, WCAP-7551, August 1970.

4. Westinghouse Electric Corporation, *LOFTRAN Code Description*, WCAP-7907-A, April 1984.

5. Westinghouse Electric Corporation, *Fuel Densification - Surry Power Station*, WCAP-8013, December 1972.

6. Westinghouse Electric Corporation, *FACTRAN - A Fortran IV Code for Thermal Transients in a* UO^2 *Fuel Rod*, WCAP-7337, 1972.

7. Westinghouse Electric Corporation, *Fuel Densification, Surry Units 1 and 2, Low Pressure Analysis*, WCAP-8117, 1973.

8. Westinghouse Electric Corporation, *Fuel Densification Experimental Results and Model for Reactor Application*, WCAP-8219, 1973.

9. Letter from C. M. Stallings, Vepco, to K. R. Goeller, NRC, dated March 12, 1975 (Serial No. 458).

10. Letter from C. M. Stallings, Vepco, to K. R. Goeller, NRC, dated June 5, 1975 (Serial No. 553).

11. Letter from C. M. Stallings, Vepco, to K. R. Goeller, NRC, dated September 9, 1977 (Serial No. 403).

12. *Vepco Reactor System Transient Analyses Using the RETRAN Computer Code*, VEP-FRD-41, Rev. 0.2-A, March 2015.

13. F. W. Sliz and K. L. Basehore, *Vepco Reactor Core Thermal-Hydraulic Analysis Using the COBRA III C/MIT Computer Code*, VEP-FRD-33-A, October 1983.

14. Letter from W. L. Stewart, Vepco, to H. R. Denton, NRC, dated April 1, 1985 (Serial No. 457).

15. Letter from C. P. Patel, USNRC, to W. L. Stewart, Vepco, Surry Units 1 and 2 - Issuance of Amendments 116/116 Re: Control Rod Assemblies and Surry Improved Fuel, dated January 6, 1988.

16. Letter from B. C. Buckley (NRC) to W. L. Stewart (Virginia Electric and Power Company), Surry Units 1 and 2 - *Issuance of Amendments Re: F Delta H Limit and Statistical DNBR Methodology*, Serial 92-405, June 1, 1992.

17. R. C. Anderson, *Statistical DNBR Evaluation Methodology*, VEP-NE-2-A, June 1987.

18. Westinghouse Electric Corporation, Methodology for the Analysis of the Dropped Rod Event, WCAP-11394-P-A, January 1990.

19. Letter from L. A. Walsh (Westinghouse Owner's Group Steam Generator Tube Uncovery Task Team) to R. C. Jones, NRC, *Westinghouse Owner's Group Steam Generator Tube Uncovery Issue*, OG-92-25, March 31, 1992.

20. *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, Theoretical Manual, Revision 10, August 2004, Bechtel Power Corporation, San Francisco, CA.

21. *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, User's Manual, Revision 10, August 2004, Bechtel Power Corporation, San Francisco, CA.

22. *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, Validation Manual, Revision 12, August 2004, Bechtel Power Corporation, San Francisco, CA.

23. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, *Standard Review Plan*, NUREG-0800, Revision 2, July 1981.

24. K. G. Murphy and K. M. Campe, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19*, 13th AEC Air Cleaning Conference, August 1974.

25. U.S. Nuclear Regulatory Commission, Office of Standards Development, *Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants*, Regulatory Guide 1.52, Revision 2, March 1978.

26. Letter from W. L. Stewart (Virginia Power) to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Control Room Habitability, Operator Dose Assessment*, Serial Number 89-381, June 1989.

27. Letter from W. L. Stewart to USNRC, *Amendment to Operating Licenses NPF-4 and NPF-7; North Anna Power Station Units 1 and 2; Proposed Technical Specifications Changes*, NRC Letter Serial No. 85-077, dated May 2, 1985 (North Anna Core Uprating Project).

28. Letter from W. L. Stewart to H. R. Denton, *Virginia Electric and Power Company; Response to NRC Request to Additional Information; Core Uprate Program; North Anna Power Station Units 1 and 2*, Serial No. 85-772A, dated February 6, 1986.

29. Letter from L. B. Engle (NRC) to W. L. Stewart, *Amendments 84 and 71 to Facility Operating Licenses NPF-4 and NPF-7*, NRC Letter Serial No. 86-575, dated August 25, 1986 (North Anna Core Uprating Project).

30. R. C. Anderson, *Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code*, VEP-NE-3-A, July 1990.

31. S. L. Davidson (Ed.), et al, *VANTAGE+ Fuel Assembly Reference Core Report*, WCAP-12610-P-A (Proprietary), April 1995.

32. NRC - Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,* July 2000.

33. Letter - NRC to Virginia Power, *Surry Units 1 and 2 - Issuance of Amendments RE: Alternative Source Term (TAC no. MA8649 and MA8650),* March 8, 2002.

34. NUREG/CR-2858, PAVAN: *Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations,* USNRC, 1982.

35. Letter from K. Cotton (NRC) to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME2393 and ME3294)*, ML101750002, Dominion Serial Number 10-580, September 24, 2010.

36. DOM-NAF-2-A, *Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code*.

37. Letter from J. A. Price (Dominion) to USNRC Document Control Desk, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed License Amendment Request, Relocation of Core Operating Limits to the Core Operating Limits Report (COLR) and Addition of COLR References*, Serial No. 09-581, October 16, 2009.

38. Letter from K. Cotton (NRC), to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Request for Technical Specification Revisions Related to the Core Operating Limits Report (TAC Nos. ME2591 and ME2592)*, October 19, 2010, Serial No. 10-645.

39. H. H. Shah and P. Schueren, *Optimized ZIRLO™,* WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, July 2006.

40. ALSTOM Power, Inc., *North Anna 1 & 2 and Surry 1 & 2 LP Retrofit Missile Analysis*, STD0010572, 2009.

41. ALSTOM Power, Inc., *Finite Element Investigation and Optimization of Stages 3 to 6 to Eliminate SCC Risk,* STD0001300, 2003.

42. ALSTOM Power, Inc., *Recommendations for the Inspection of LP Retrofit Internals on Nuclear Turbines*, HTGD672084, Revision A, 2004.

43. W. G. Clark, Jr., B. B. Seth &D. H. Shaffer, A*SME 81-JPGC-Pwr-31, Procedures for Estimating the Probability of Steam Turbine Disc Rupture from Stress Corrosion Cracking*, 1981.

44. ALSTOM Power, Inc., *Missile Analysis Methodology Comparison*, STD0011103, 2009.

Table 14.2-1
NATURAL CIRCULATION REACTOR COOLANT FLOW VERSUS REACTOR POWER (ORIGINAL)

Reactor Power (% full power)	Reactor Coolant Flow (% nominal flow)
3.5	5.0
3.0	4.7
2.5	4.4
2.0	4.1
1.5	3.8
1.0	3.3

Table 14.2-2
INITIAL RADIONUCLIDE INVENTORY
FOR THE LRA FUEL ROD GAP RELEASE WITH 1.4% FAILED FUEL

Isotope	Activitiy (Ci)	Isotope	Activitiy (Ci)
I-130	2.9107E+03	Kr-85	1.9073E+03
I-131	1.2641E+05	Kr-87	3.8961E+04
I-132	1.1353E+05	Kr-88	5.4835E+04
I-133	1.6041E+05	Kr-89	6.6735E+04
I-134	1.7648E+05	Kr-83m	9.6616E+03
I-135	1.5042E+05	Kr-85m	2.0337E+04
Xe-133	1.6053E+05	Cs-134	3.9498E+04
Xe-135	3.9330E+04	Cs-136	9.0478E+03
Xe-137	1.4042E+05	Cs-137	2.5353E+04
Xe-138	1.3245E+05	Cs-138	3.5243E+05
Xe-131m	8.7917E+02	Cs-139	3.3387E+05
Xe-133m	5.0099E+03	Cs-134m	9.8218E+03
Xe-135m	3.1499E+04	Br-82	4.2662E+02
Rb-86	3.9327E+02	Br-83	9.6426E+03
Rb-88	1.3375E+05	Br-84	1.6684E+04
Rb-89	1.7127E+05	Br-85	2.0063E+04
Rb-90	1.6616E+05	Br-87	3.2808E+04
		Br-88	3.4974E+04

Table 14.2-3
STEAM GENERATOR VOLUMES RELEASED DURING A LOCKED ROTOR ACCIDENT

	Time Averaged Flow Rate	
Time Period (hours)	Liquid (CFM)	Steam (CFM)
0.00 - 0.25	172	4809
0.25 - 0.33	60	1685
0.33 - 1.00	58	1621
1.00 - 2.00	48	1337
2.00 - 8.00	37	1034
8.00 - 9.00	30	826

Table 14.2-4
LRA DOSE ANALYSIS RESULTS

Control Room 30-day Dose (Rem TEDE)	10 CFR 50.67 Dose Limit[a] (Rem TEDE)	EAB 2-hour Dose (Rem TEDE)	LPZ 30-day Dose (Rem TEDE)	RG 1.183 Dose Limit[a] (Rem TEDE)
1.2	5.0	0.2	0.1	2.5

a. 10 CFR Part 50.67 establishes TEDE dose limits for the EAB, the outer boundary of the LPZ, and for the control room for use with the alternate source term. The specified offsite dose limits are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. For events with a higher probability of occurrence, e.g., LRA postulated EAB and LPZ doses should not exceed the limits established in RG 1.183. The 10 CFR 50.67 control room criterion applies to all accidents.

Figure 14.2-1
ROD WITHDRAWAL FROM SUBCRITICAL - NEUTRON POWER

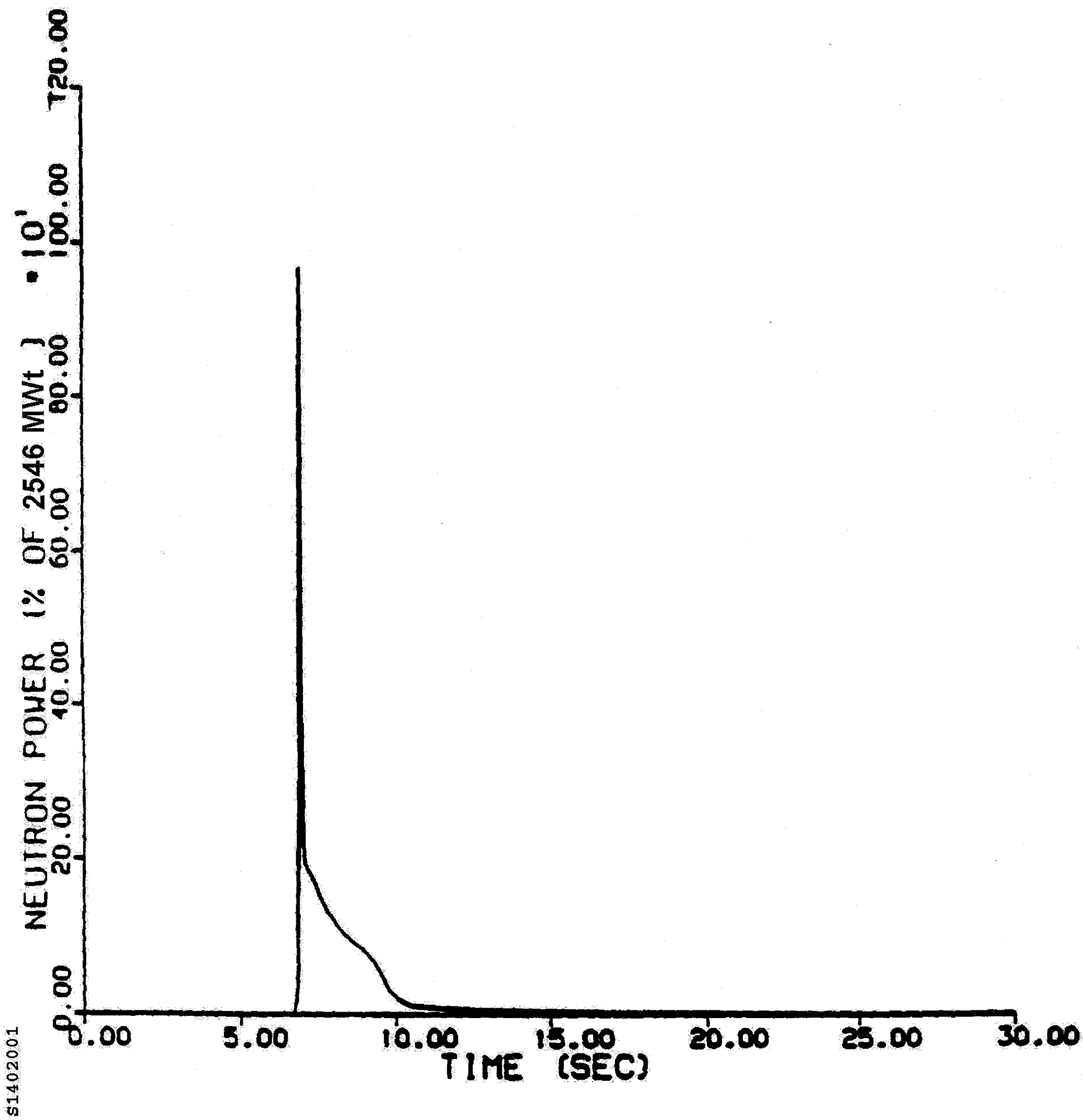


Figure 14.2-2
UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION,
PEAK HEAT FLUX

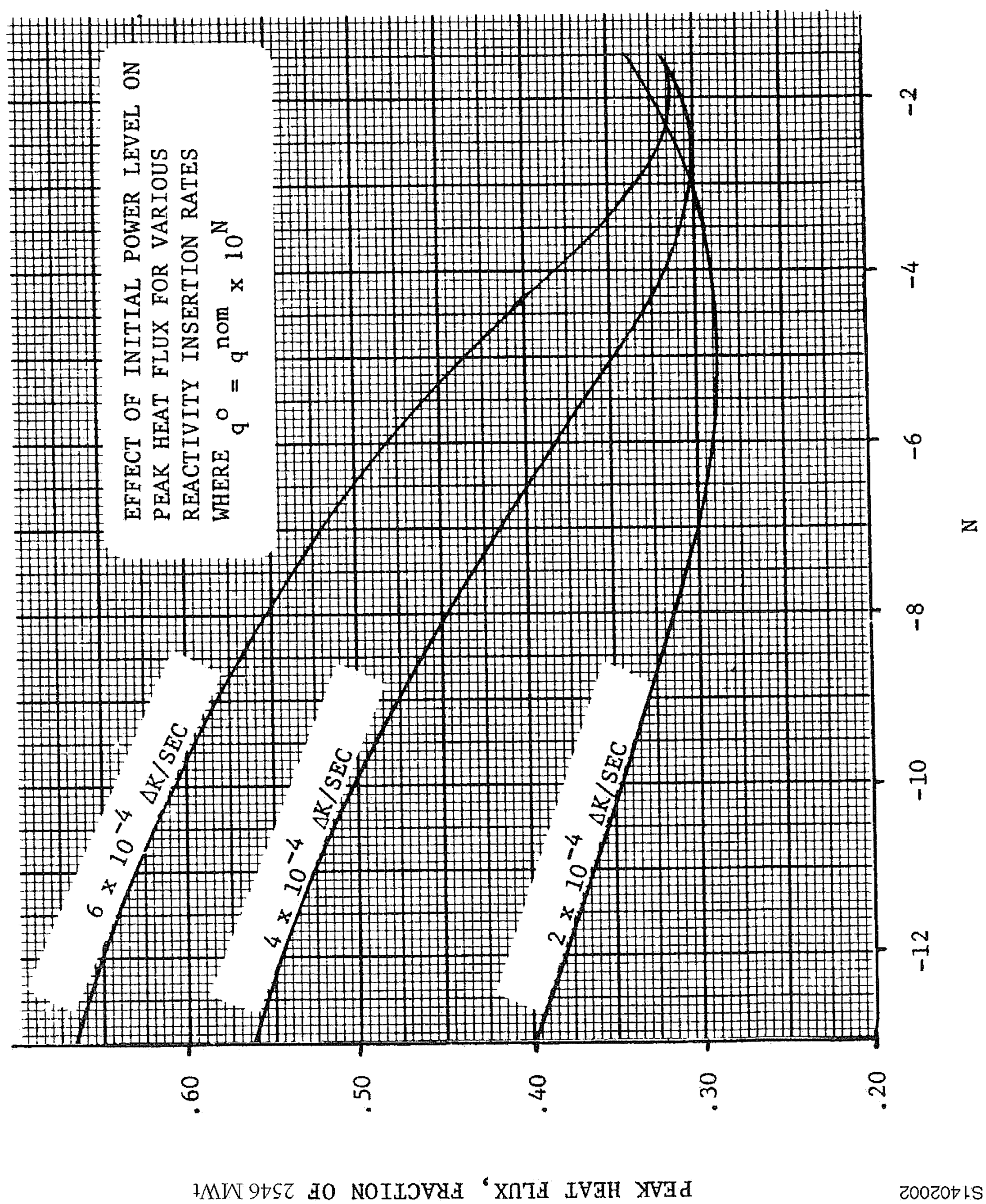


Figure 14.2-3
ROD WITHDRAWAL FROM SUBCRITICAL - CORE HEAT FLUX

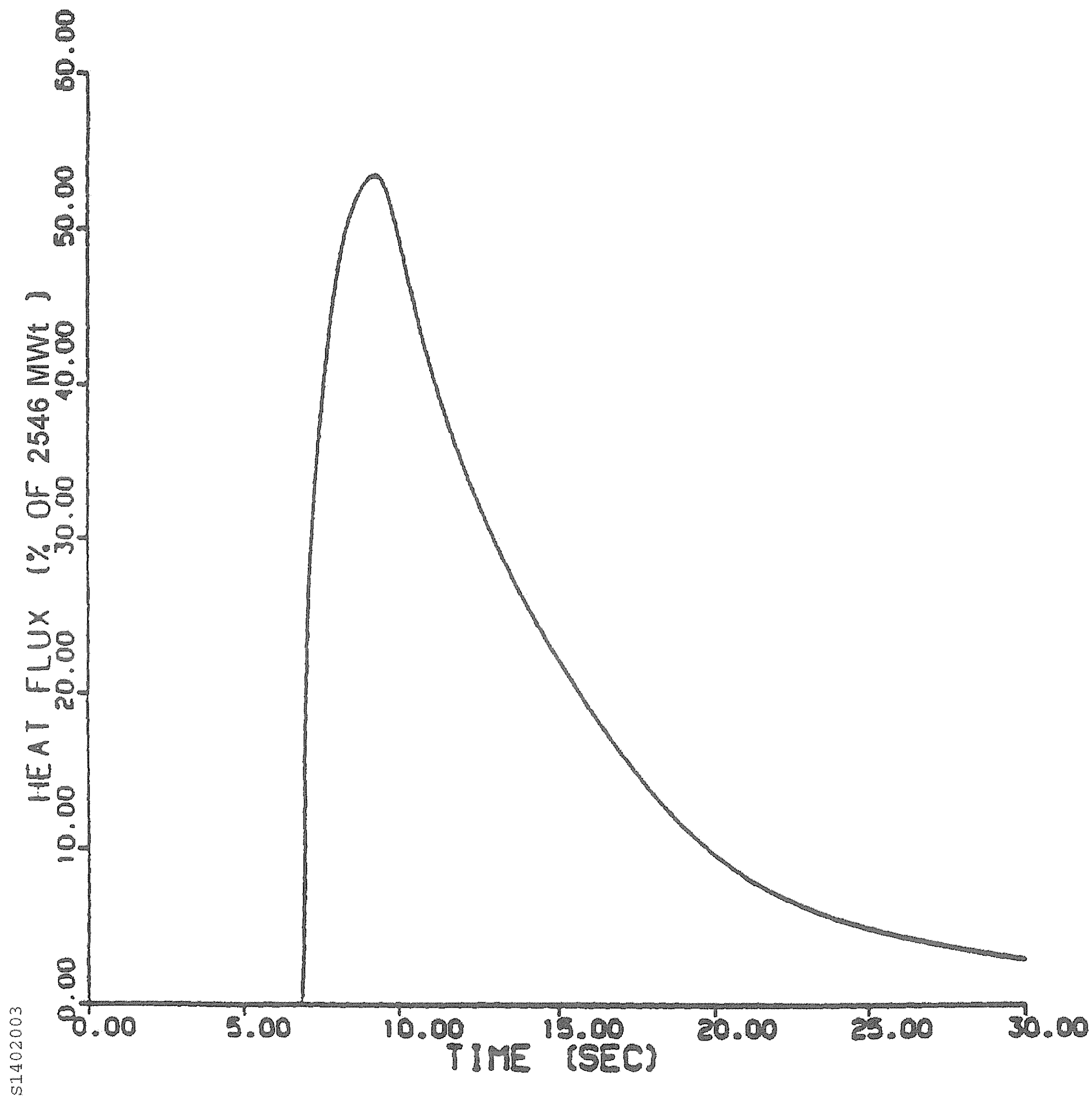


Figure 14.2-4
ROD WITHDRAWAL FROM SUBCRITICAL - TEMPERATURES
(FUEL, CLAD, MODERATOR)

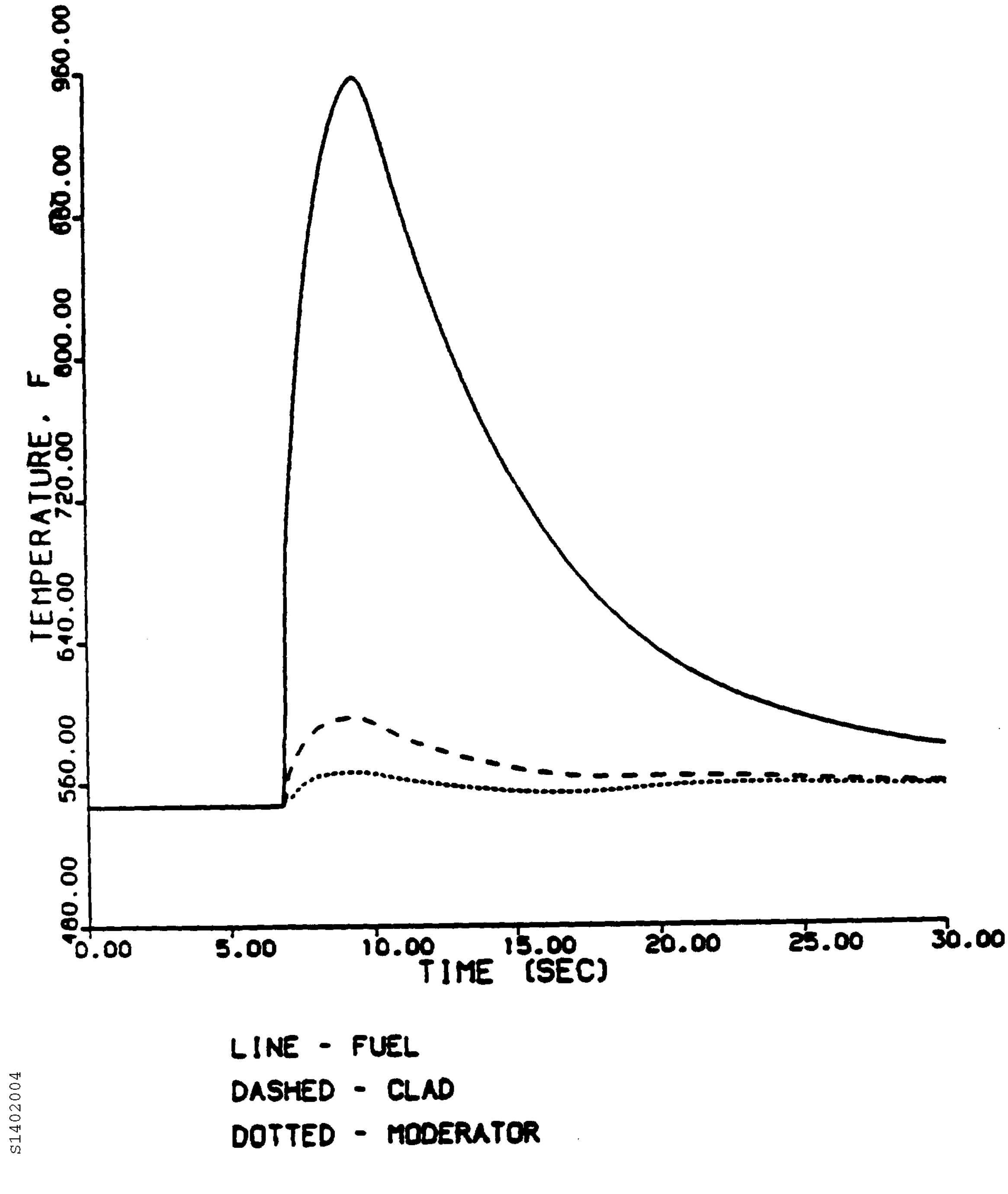


Figure 14.2-5
ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER DELTA-T PROTECTION

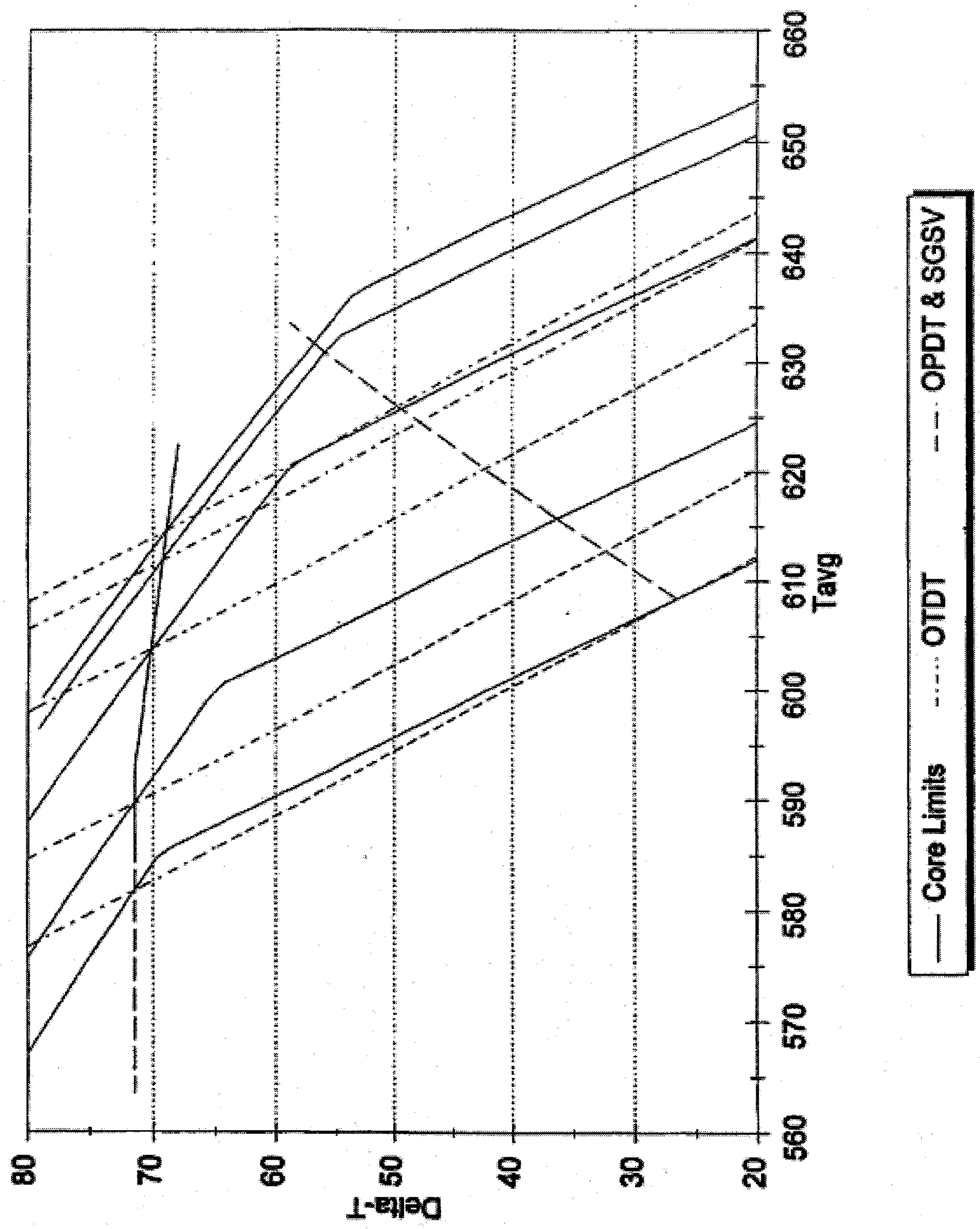


Figure 14.2-6
ROD WITHDRAWAL AT POWER
MINIMUM DNBR VS. INSERTION RATE AT 2589.3 MWt

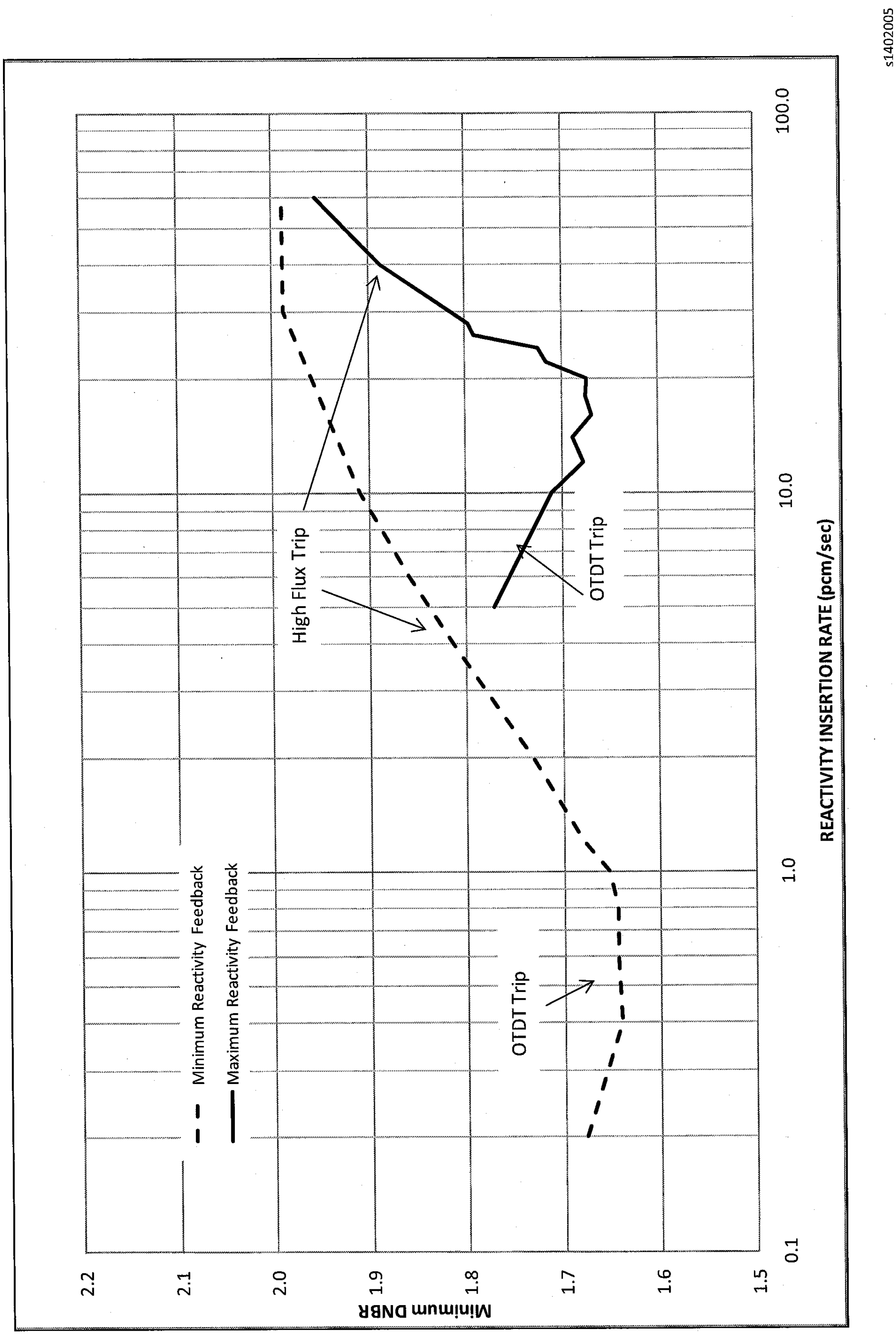


Figure 14.2-7
ROD WITHDRAWAL AT POWER
NUCLEAR POWER - LIMITING DNBR CASE AT 2589.3 MWt

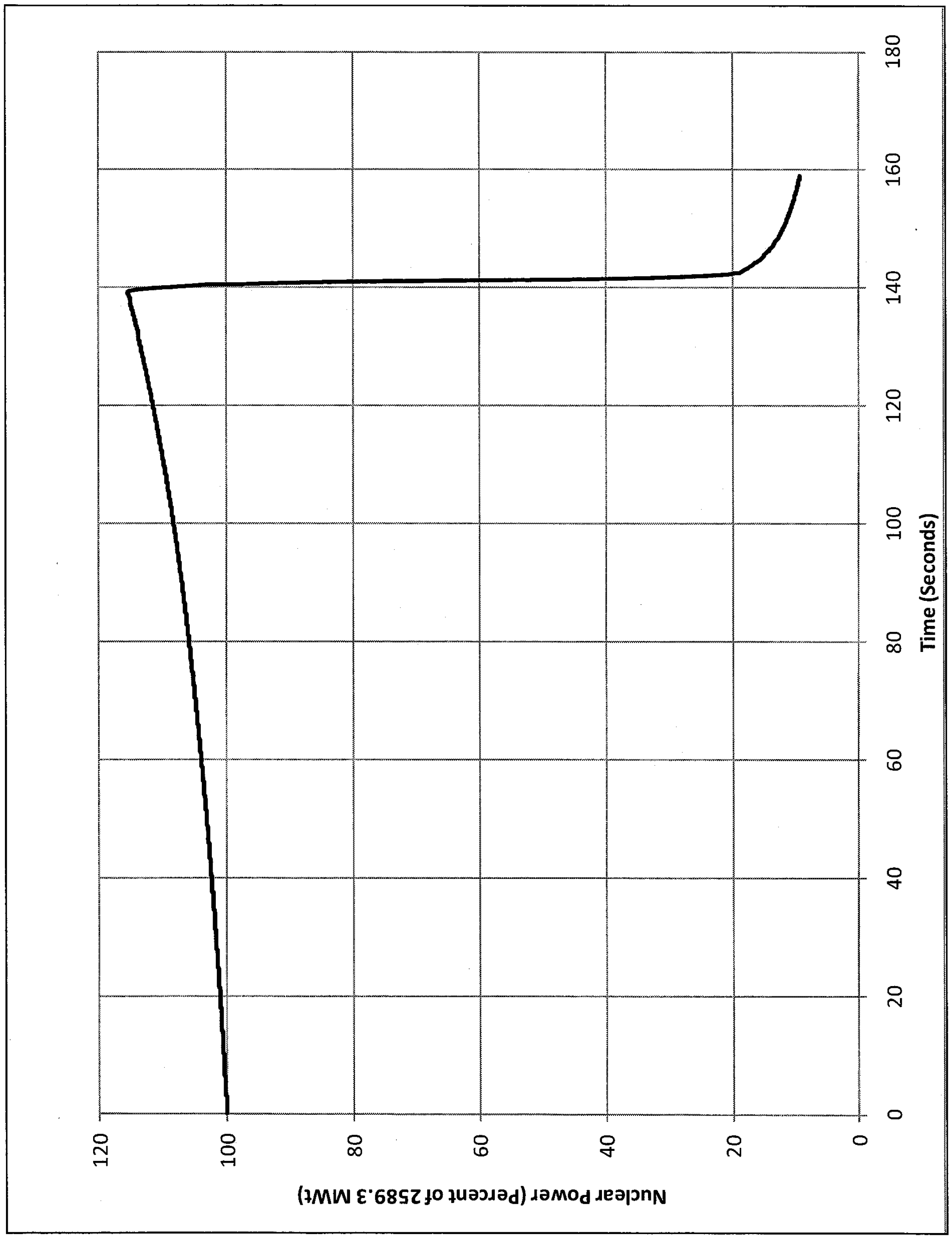
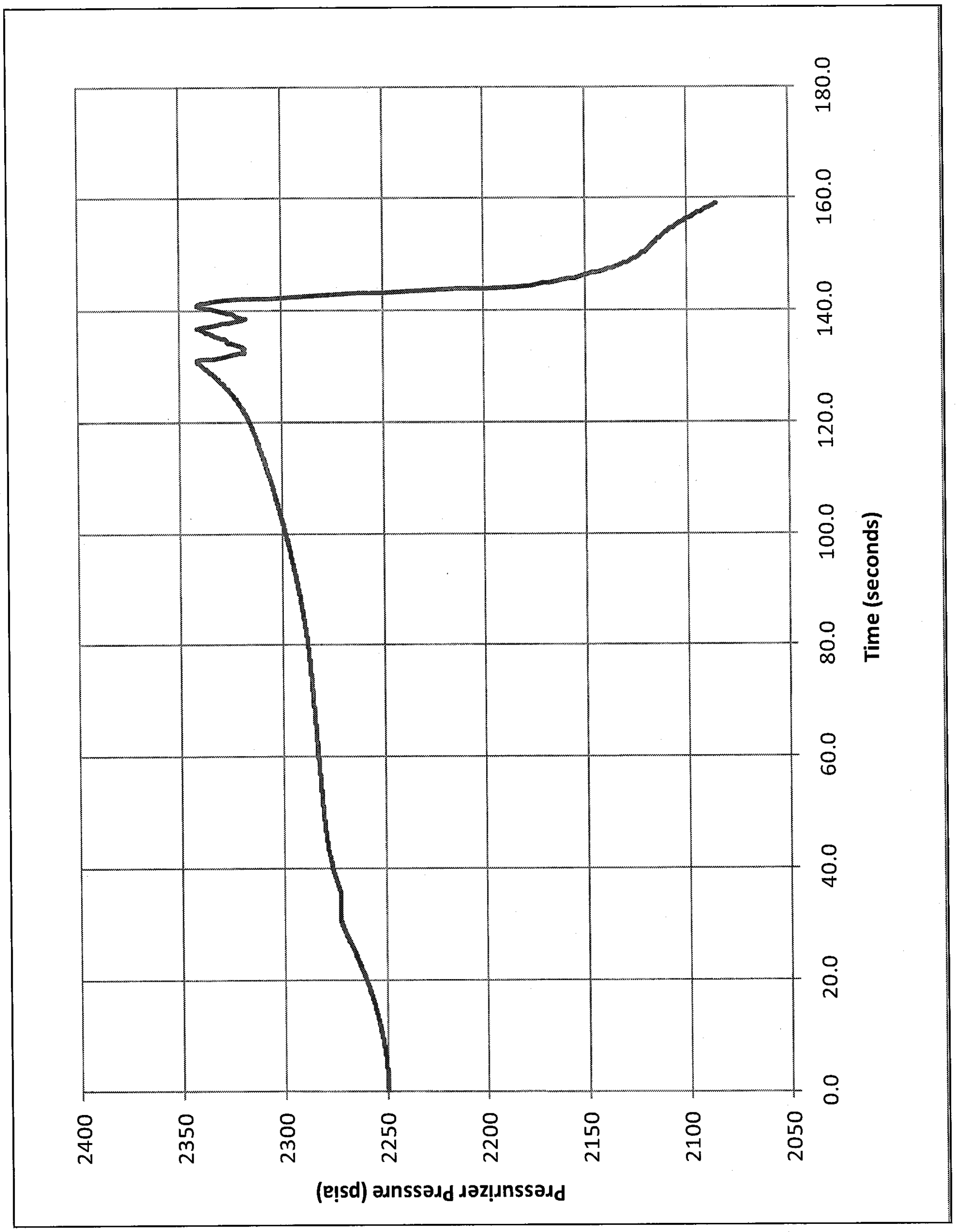


Figure 14.2-8
ROD WITHDRAWAL AT POWER
PRESSURIZER PRESSURE - LIMITING DNBR CASE AT 2589.3 MWt



s1402007

s1402008

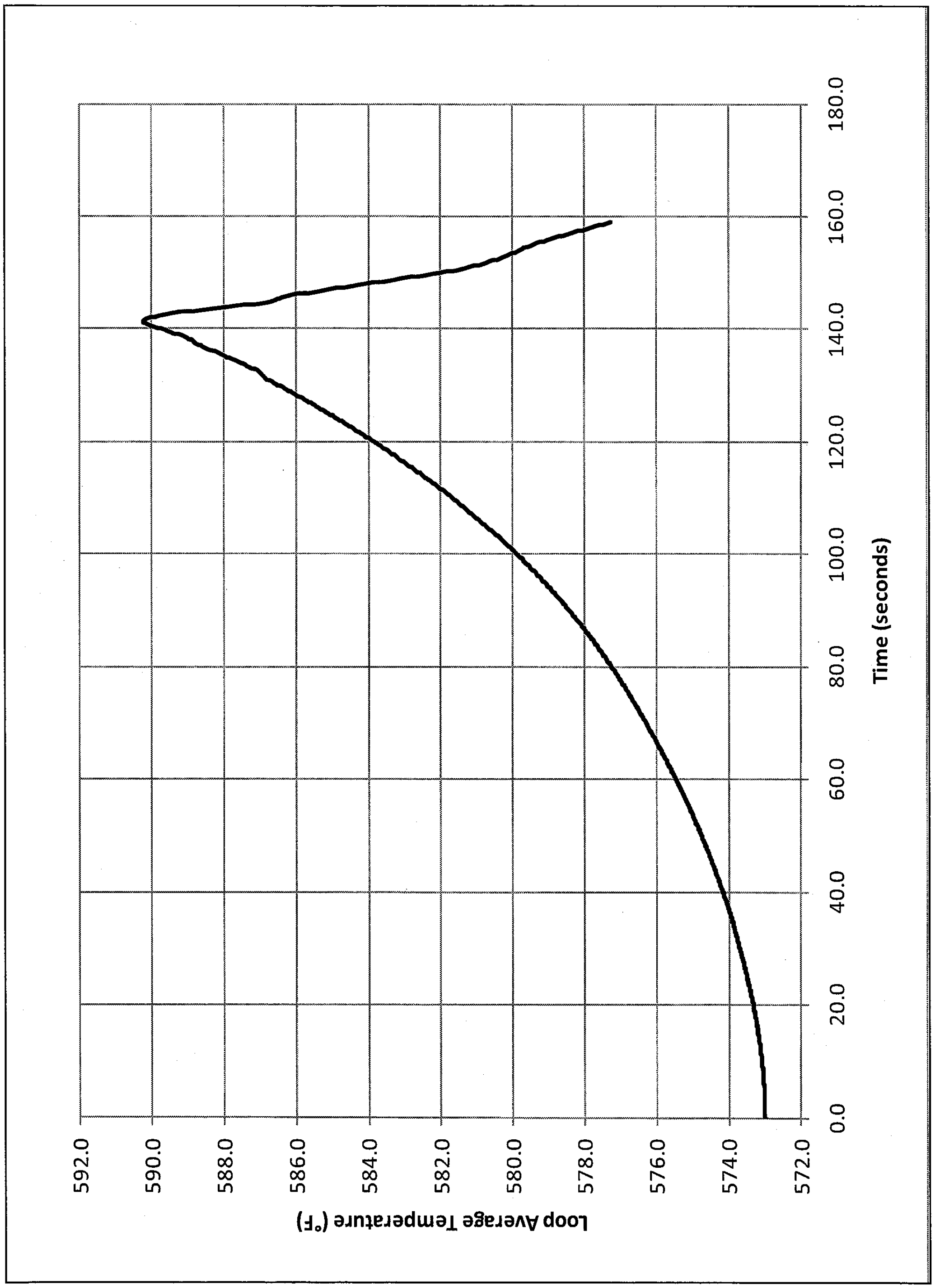
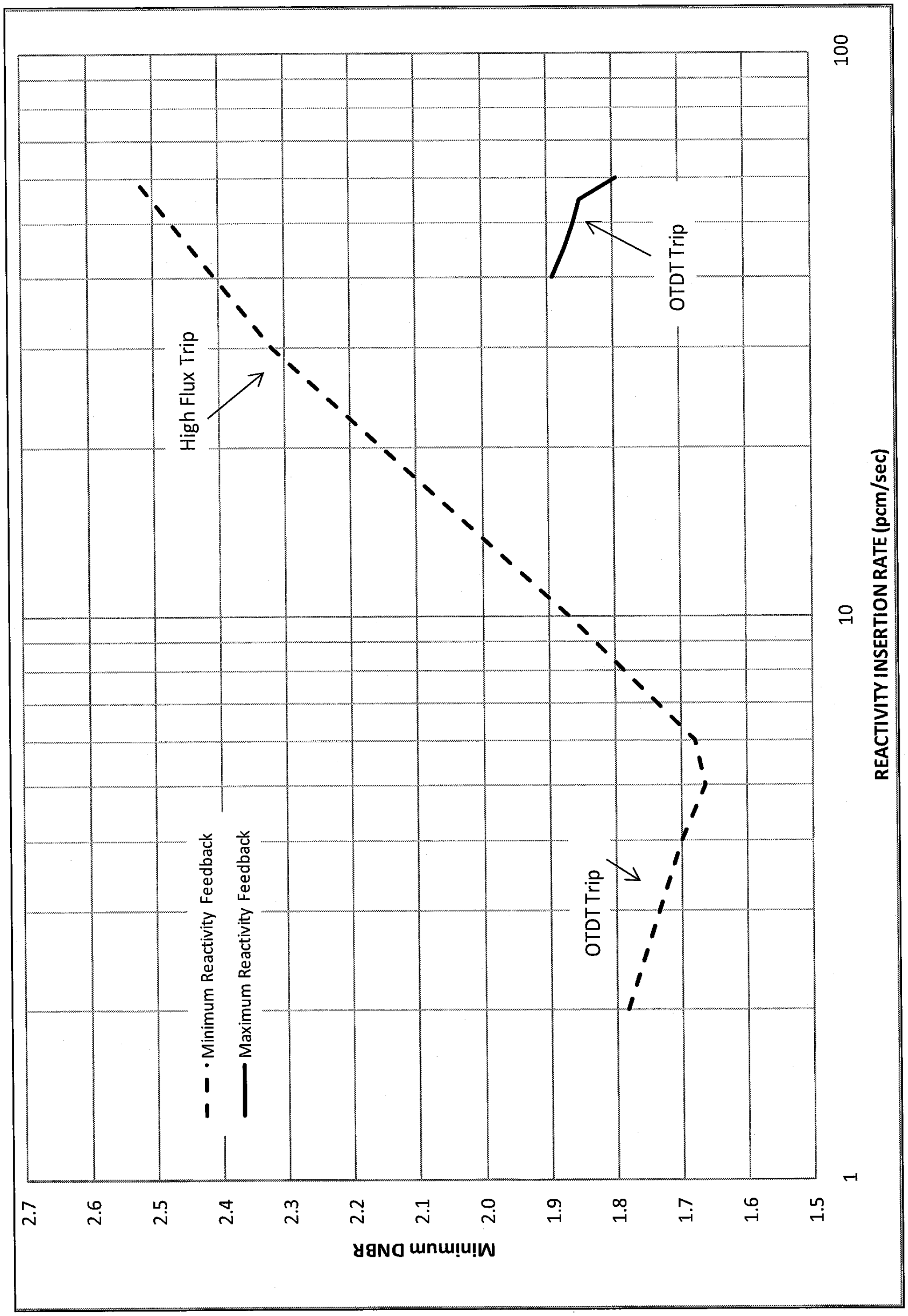


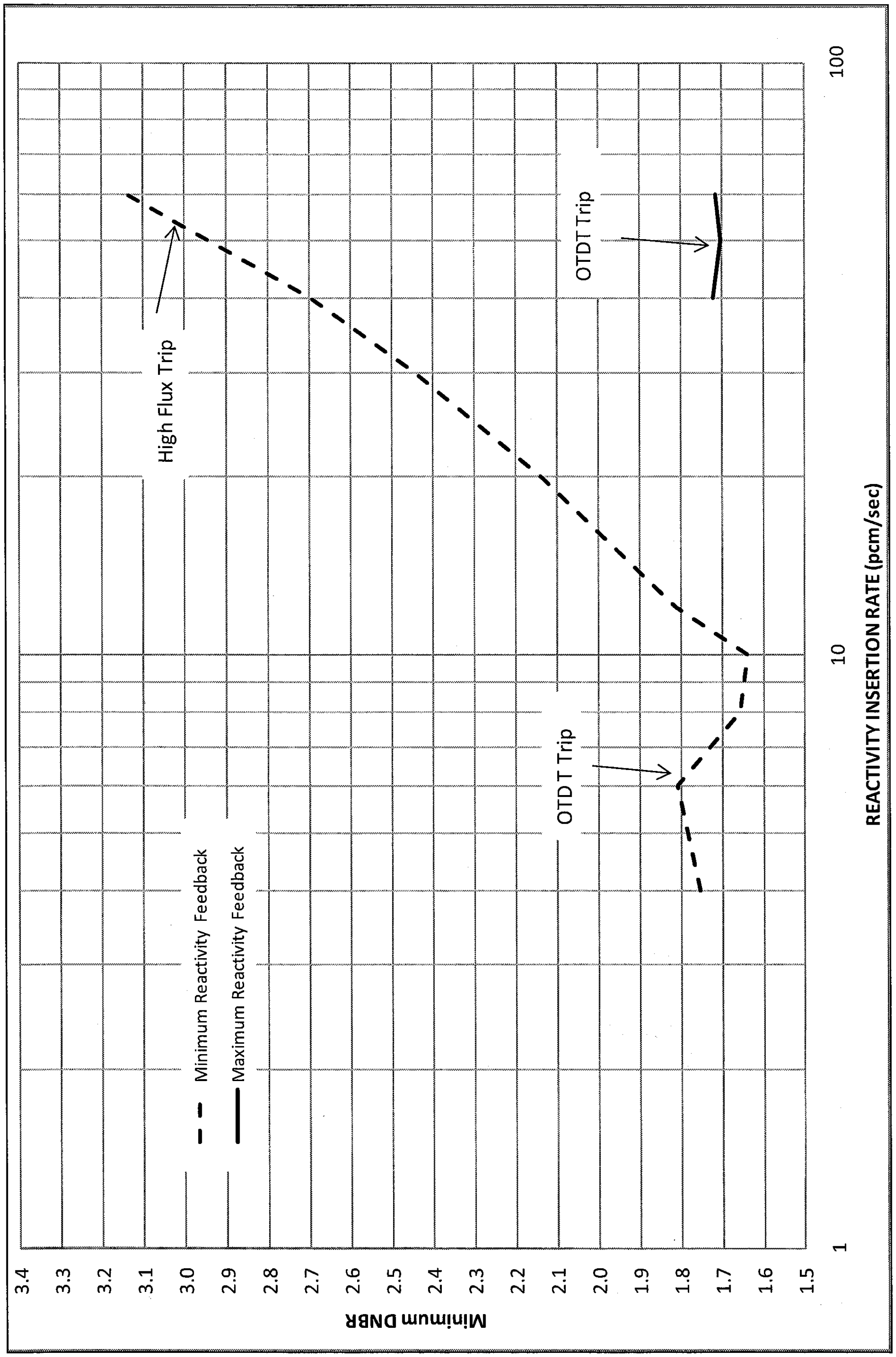
Figure 14.2-9
ROD WITHDRAWAL AT POWER
RCS AVERAGE TEMPERATURE - LIMITING DNBR CASE AT 2589.3 MWt

Figure 14.2-10
ROD WITHDRAWAL AT POWER
MINIMUM DNBR VS. INSERTION RATE AT 60% OF 2546 MWt



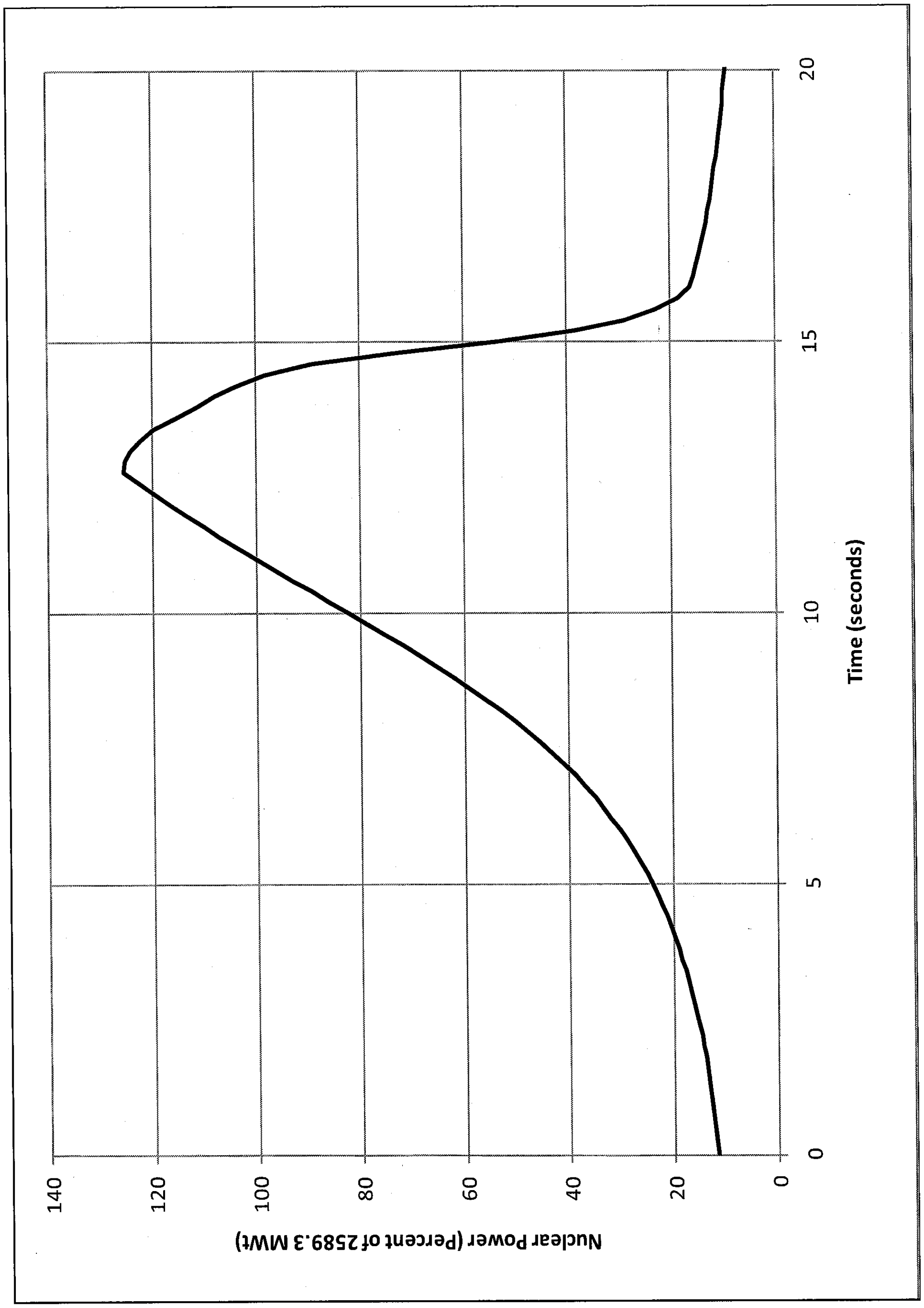
s1402009

Figure 14.2-11
ROD WITHDRAWAL AT POWER
MINIMUM DNBR VS. INSERTION RATE AT 10% OF 2546 MWt



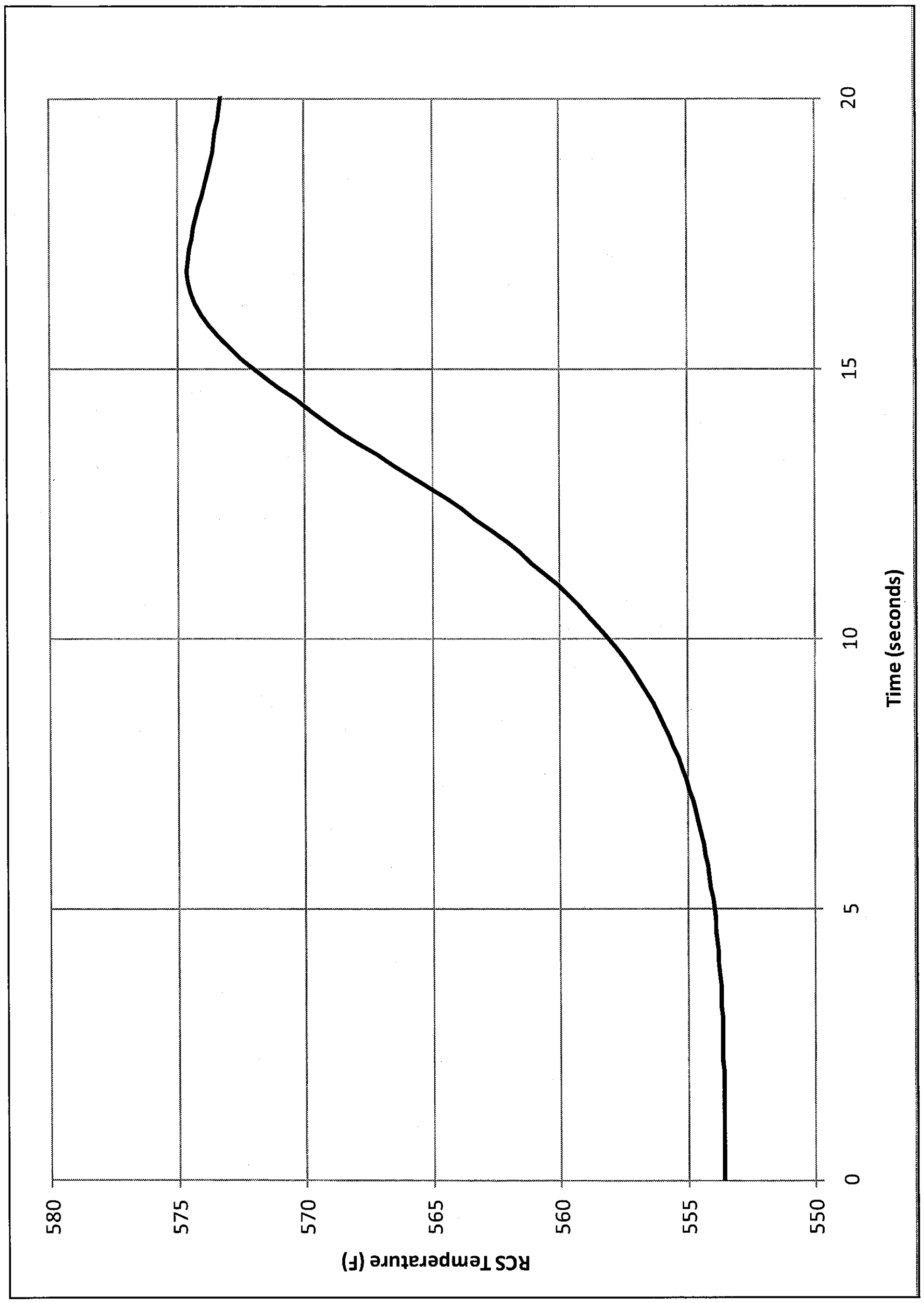
s1402010

Figure 14.2-12
ROD WITHDRAWAL AT POWER
NUCLEAR POWER - RCS OVERPRESSURE CASE



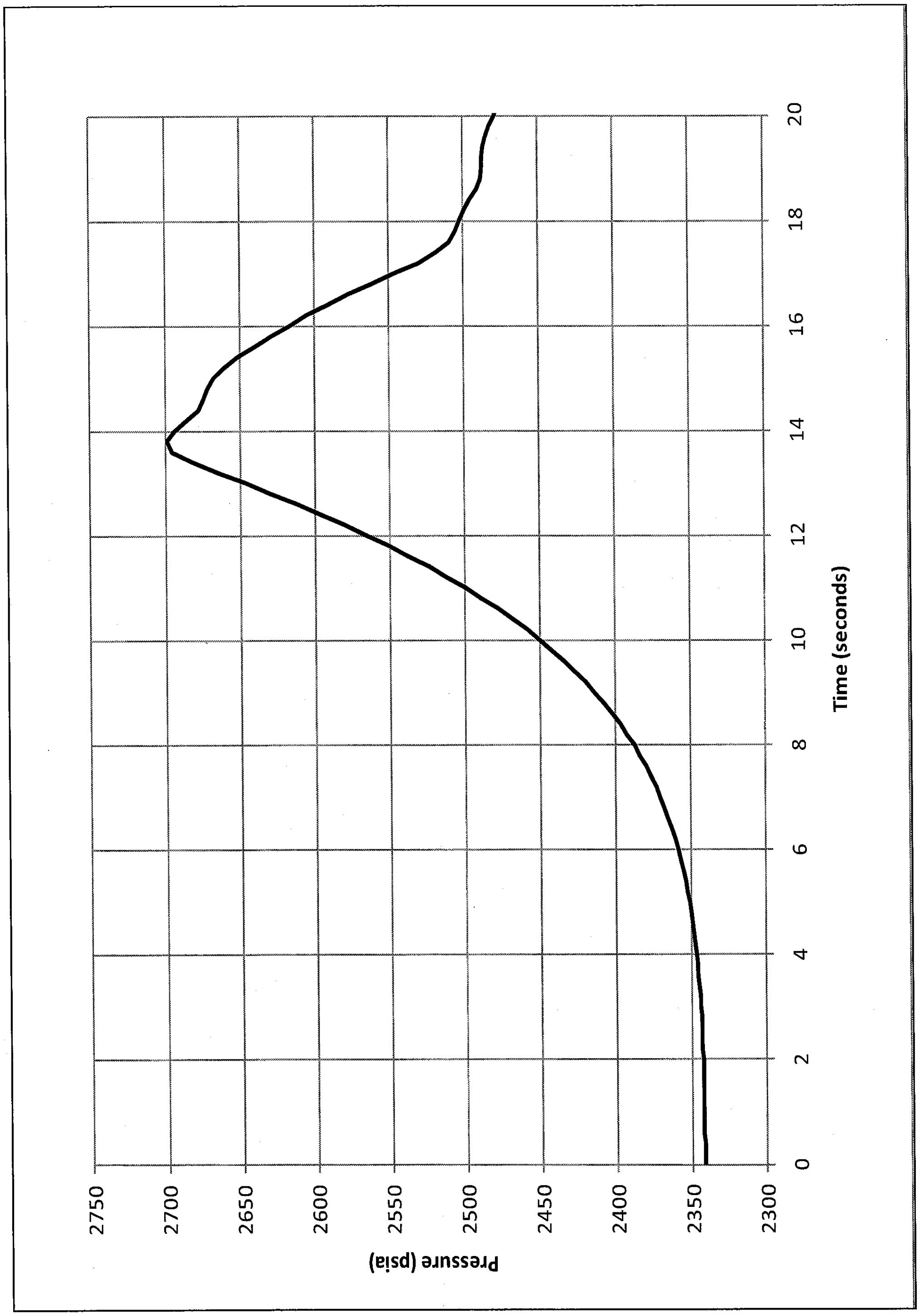
s1402011

Figure 14.2-13
ROD WITHDRAWAL AT POWER
RCS AVERAGE TEMPERATURE - RCS OVERPRESSURE CASE



s1402012

Figure 14.2-14
ROD WITHDRAWAL AT POWER COLD LEG PRESSURE - RCS OVERPRESSURE CASE



s1402013

Figure 14.2-15
NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX
TRANSIENT FOR DROPPED RCCA

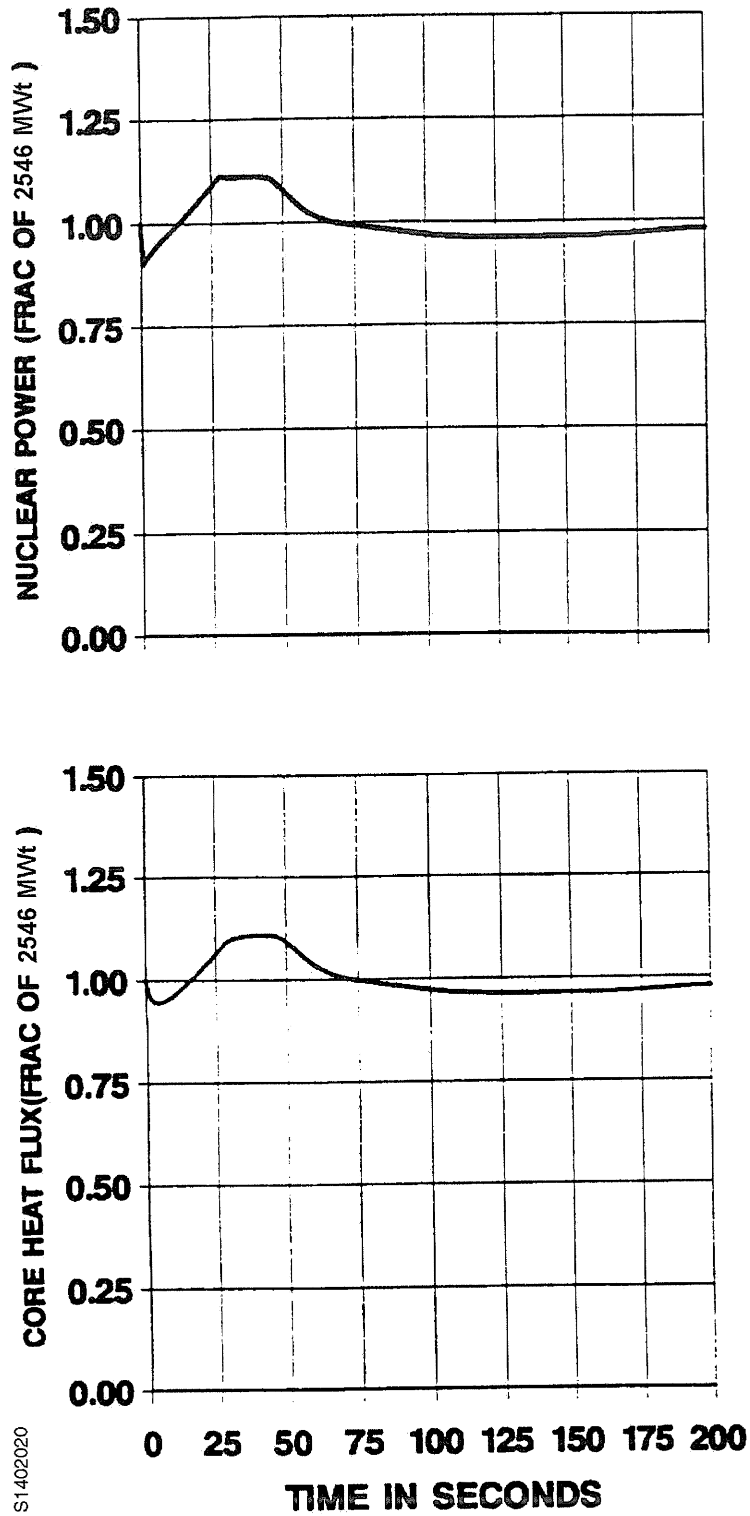


Figure 14.2-16
PRESSURIZER PRESSURE TRANSIENT AND COOLANT AVERAGE TEMPERATURE TRANSIENT FOR DROPPED RCCA

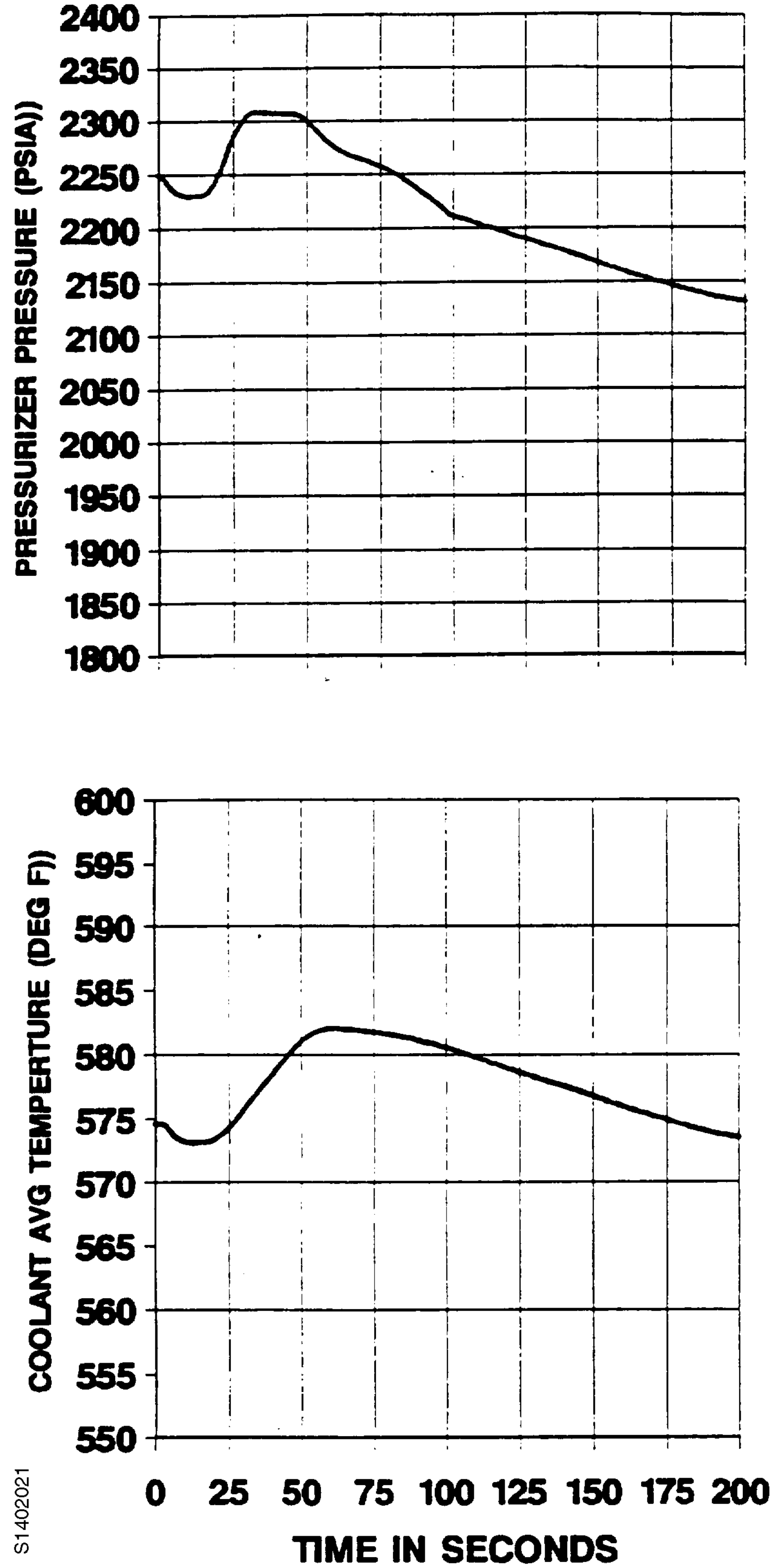


Figure 14.2-17
SURRY MLT-LOOP EXCESS FW TRANSIENT (150% FLOW W/ROD CONTROL) - NUCLEAR POWER, FRACTION OF 2546 MWt

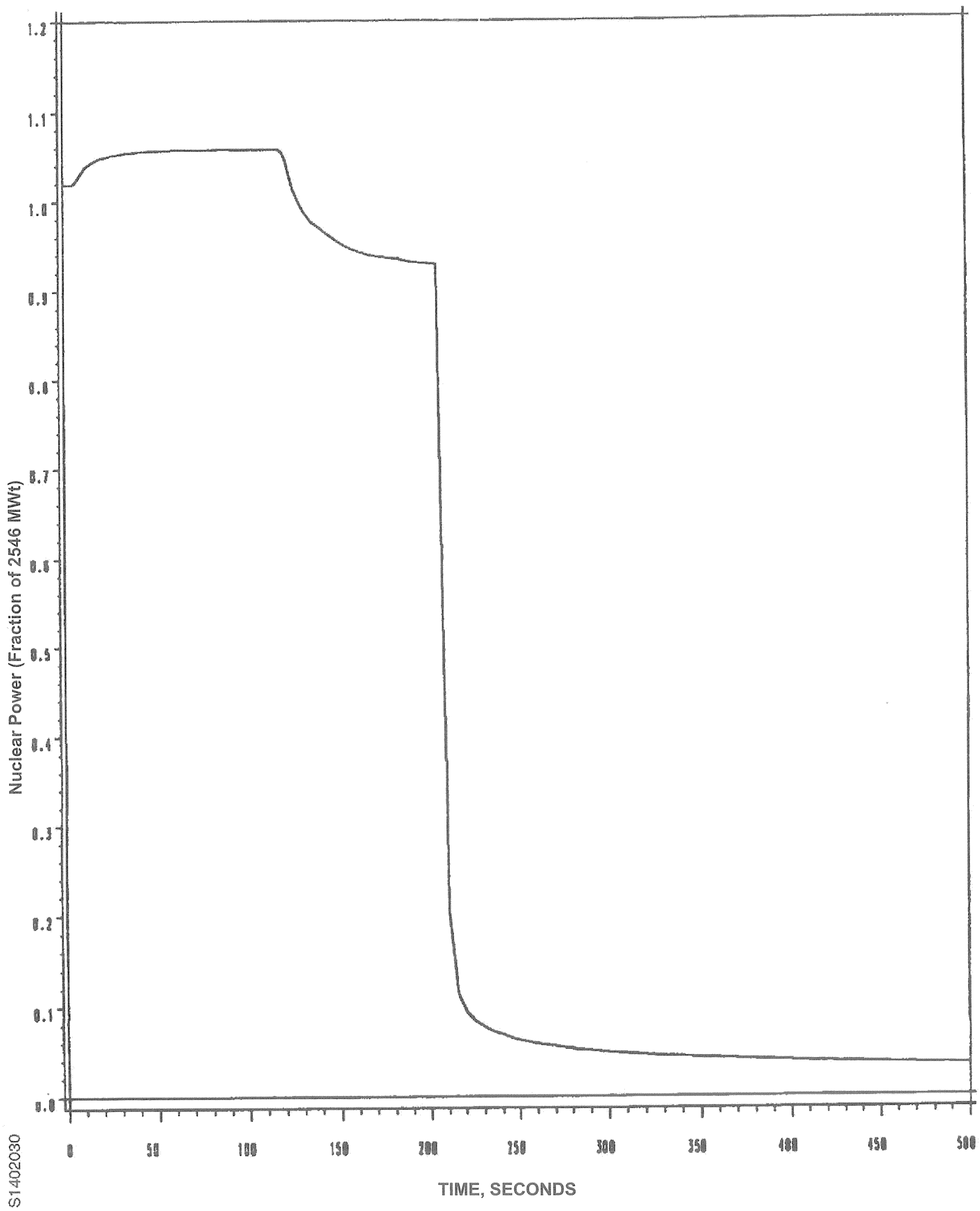


Figure 14.2-18
SURRY MLT-LOOP EXCESS FW TRANSIENT (150% FLOW W/ROD CONTROL) - LOOP ΔT, DEG F

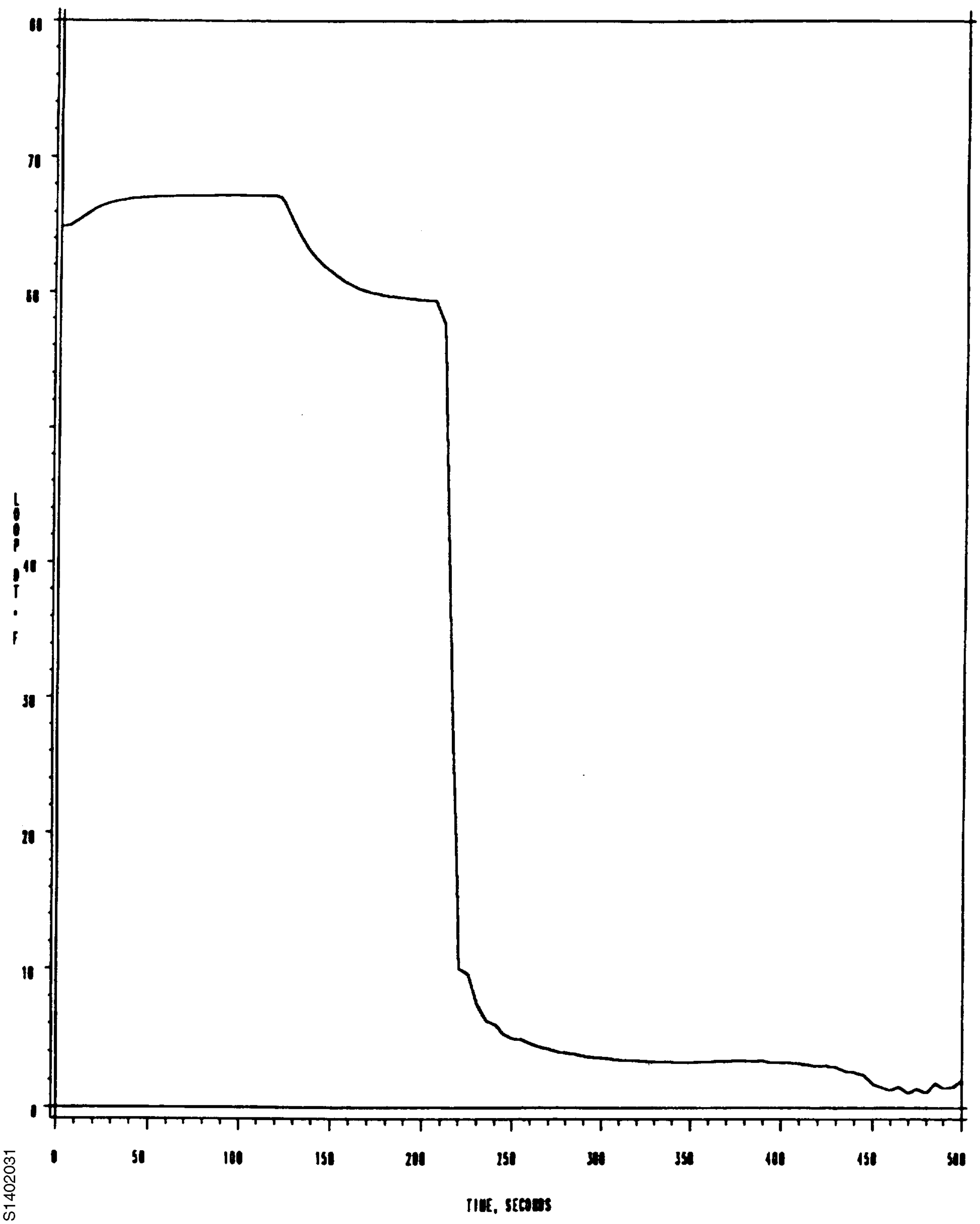


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Figure 14.2-19
SURRY MLT-LOOP EXCESS FW TRANSIENT (150% FLOW W/ROD CONTROL) - PRESSURIZER PRESSURE, PSIA

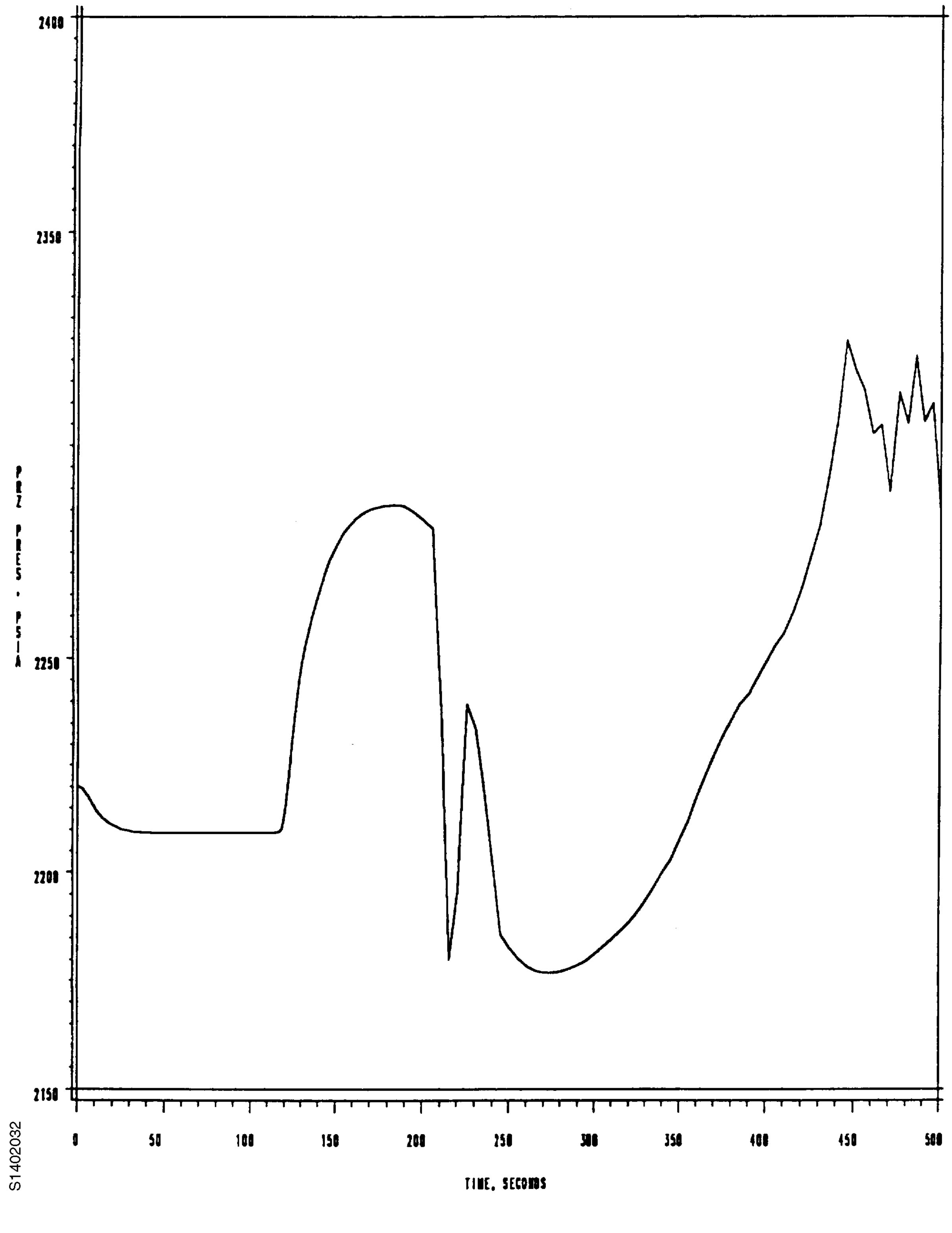

Figure 14.2-20
SURRY MLT-LOOP EXCESS FW TRANSIENT (150% FLOW W/ROD CONTROL) - CORE AVG TEMPERATURE, °F

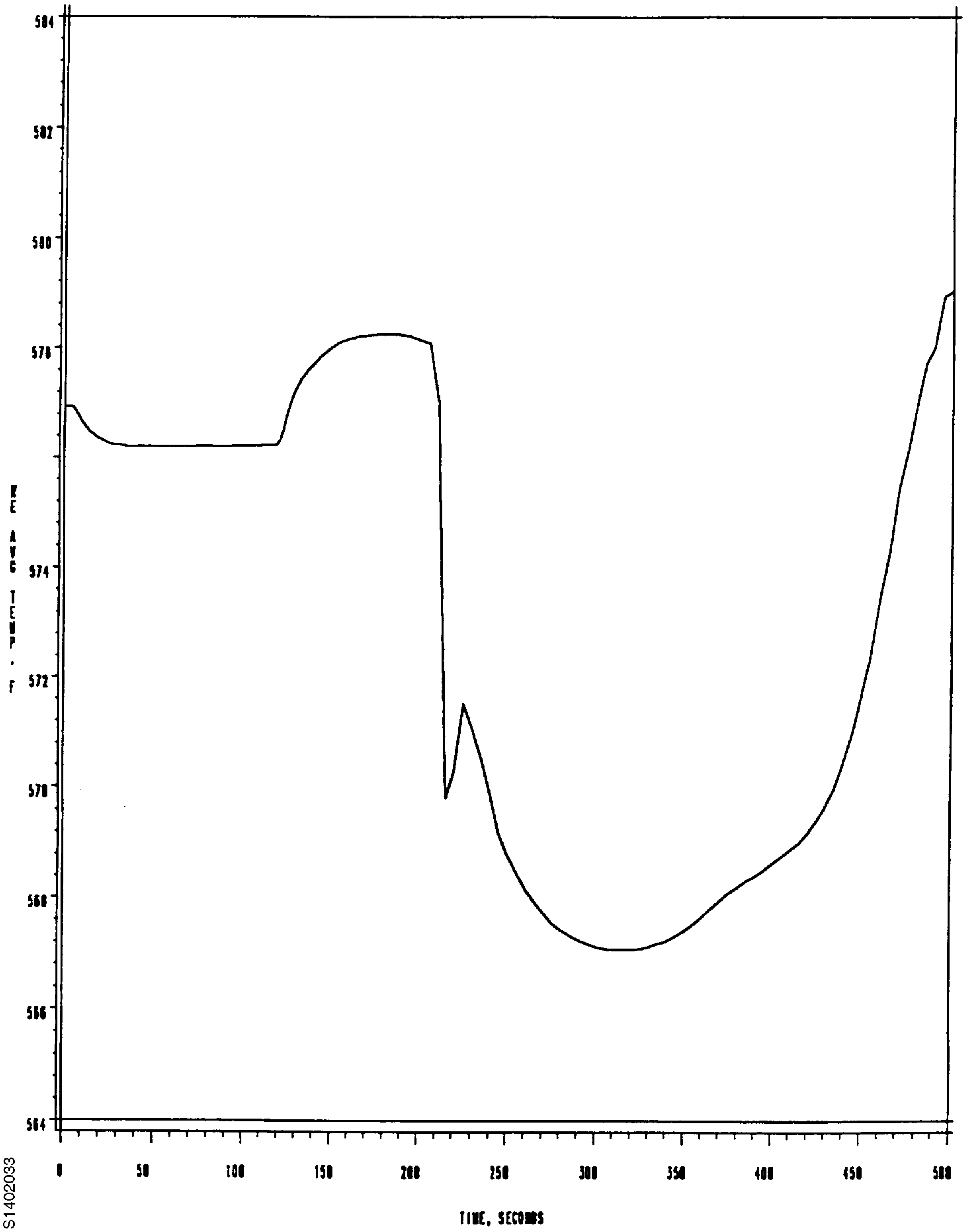

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Figure 14.2-21
SURRY MLT-LOOP EXCESS FW TRANSIENT (150% FLOW W/ROD CONTROL) -
STEAM GENERATOR MASS, LBM

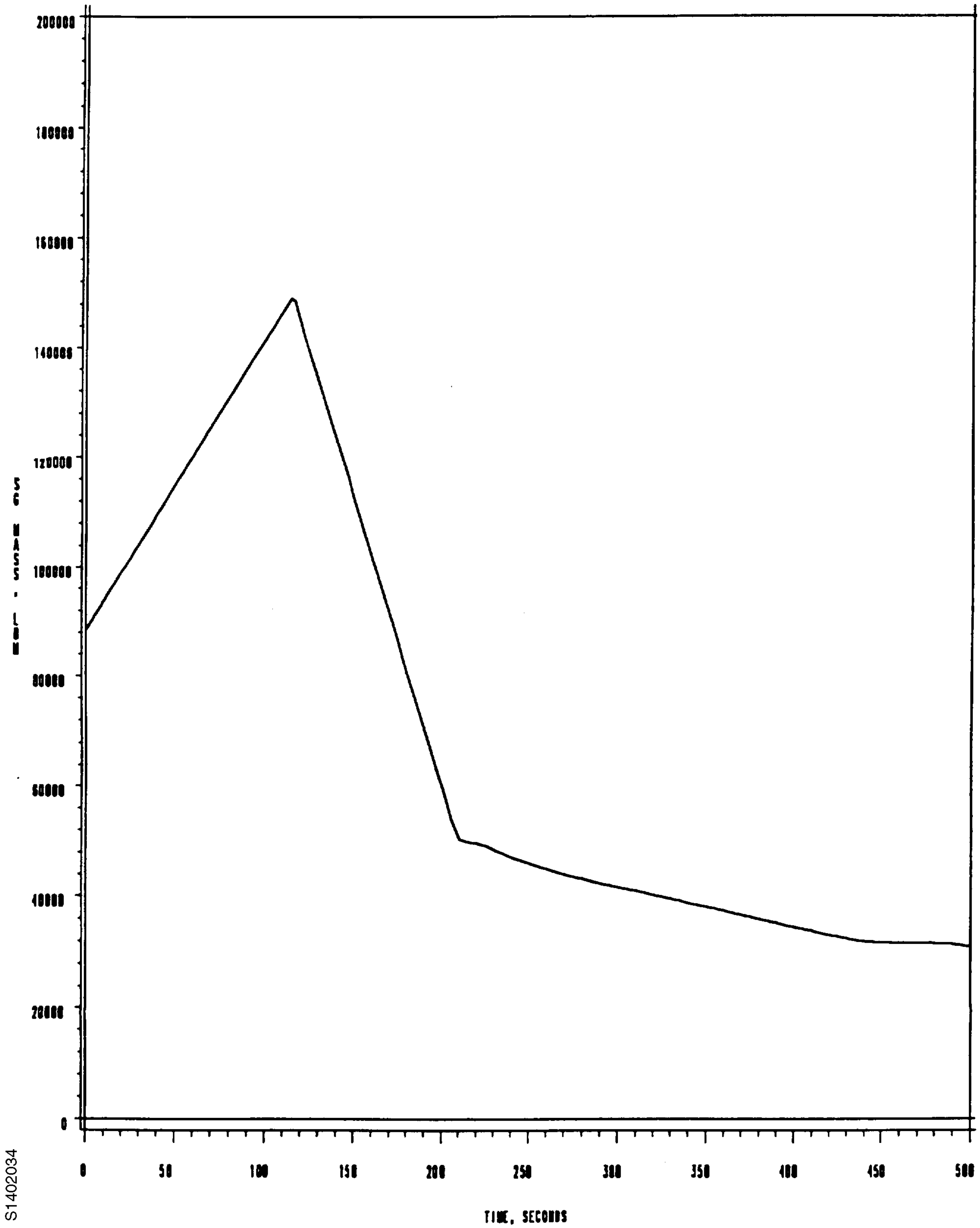

Figure 14.2-22
MAIN FEEDWATER TEMPERATURE REDUCTION EVENT
CHANGE IN FEEDWATER TEMPERATURE VS. TIME

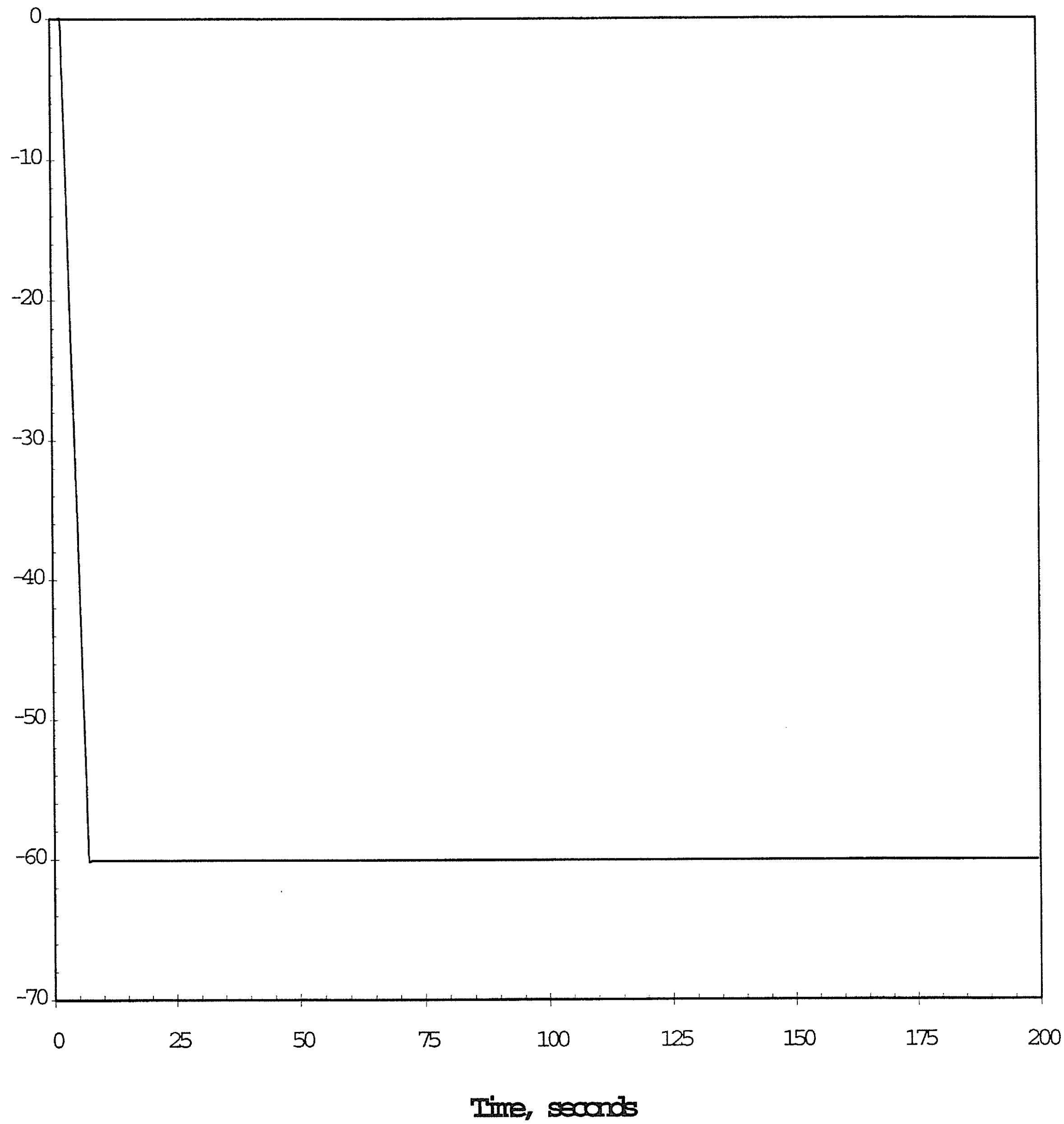

Figure 14.2-23
MAIN FEEDWATER TEMPERATURE REDUCTION EVENT
NORMALIZED POWER (FRACTION OF 2546 MWt) VS. TIME

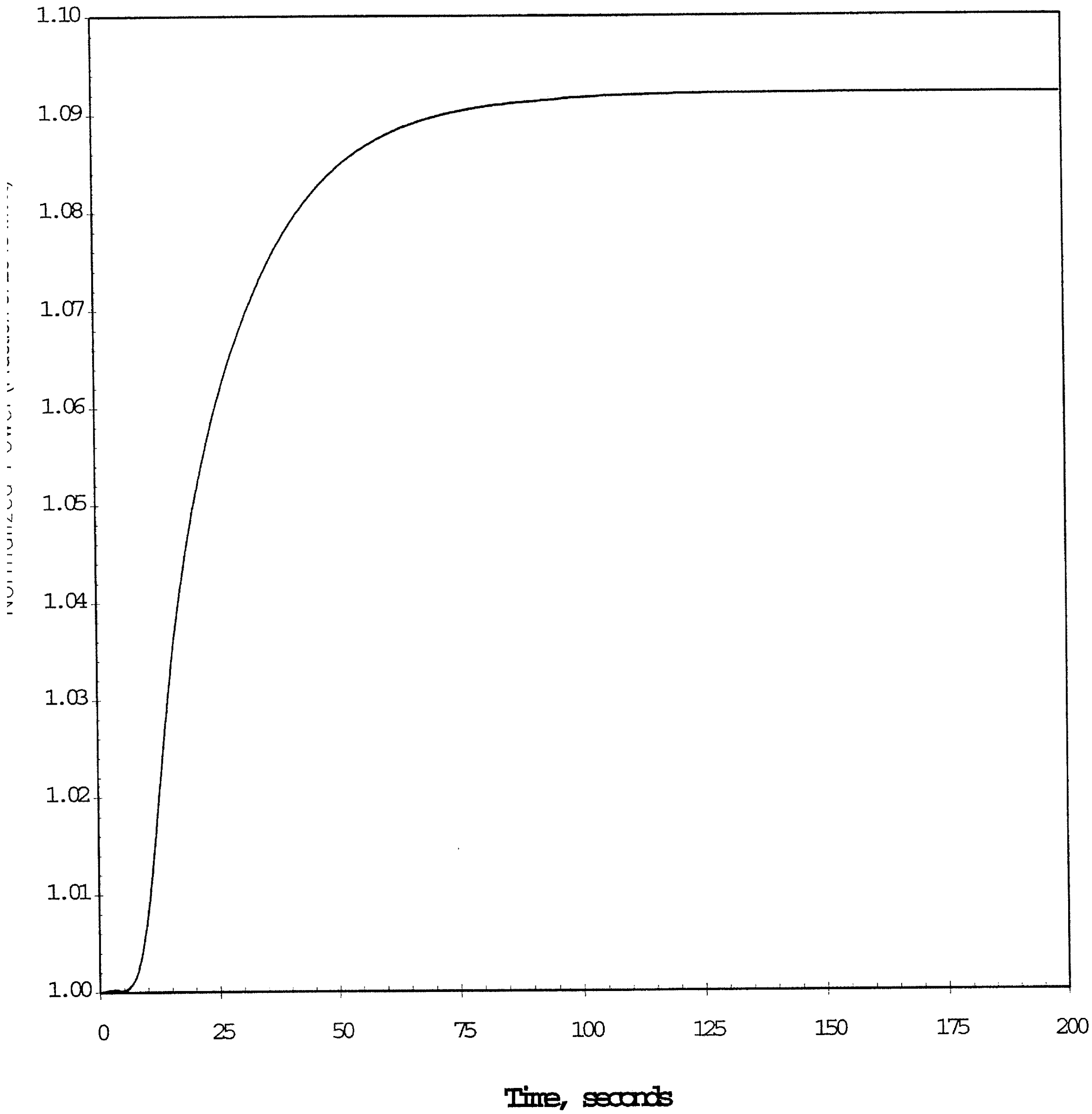

Figure 14.2-24
MAIN FEEDWATER TEMPERATURE REDUCTION EVENT
CHANGE IN PRESSURIZER PRESSURE VS. TIME

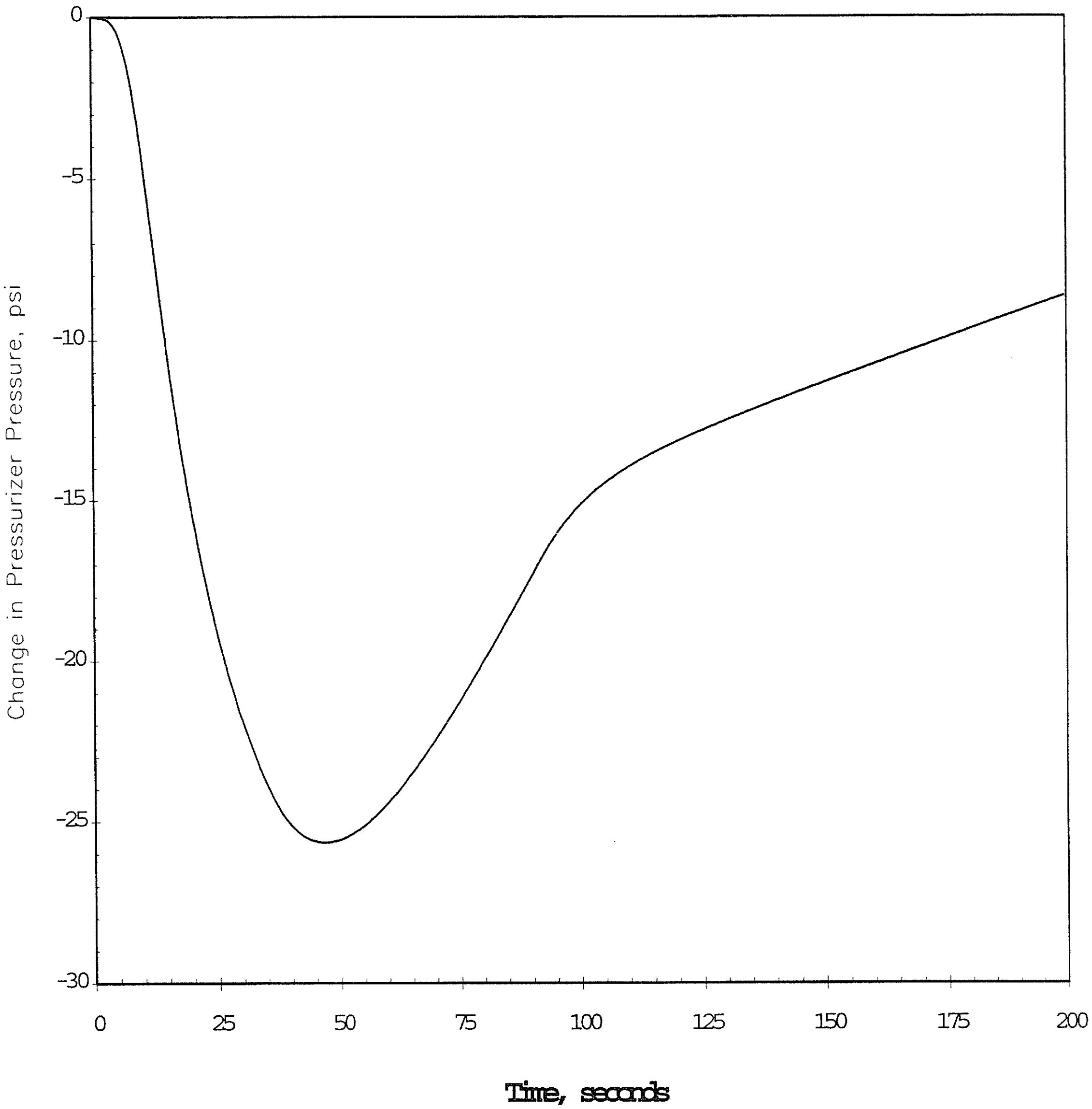

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Figure 14.2-25
MAIN FEEDWATER TEMPERATURE REDUCTION EVENT
CHANGE IN RCS AVERAGE TEMPERATURE VS. TIME

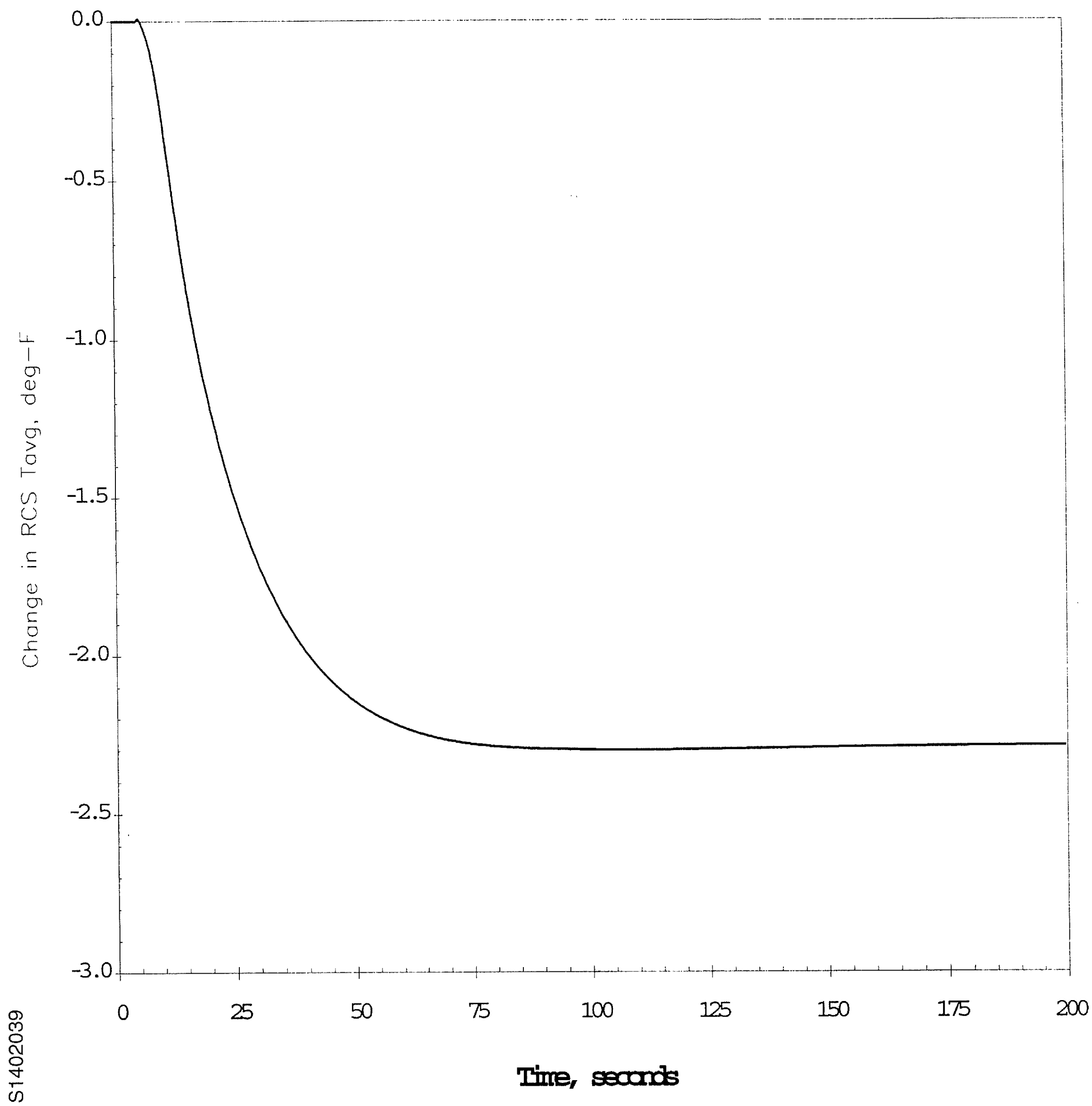

Figure 14.2-26
MAIN FEEDWATER TEMPERATURE REDUCTION EVENT
CHANGE IN RCS LOOP DELTA-T VS. TIME

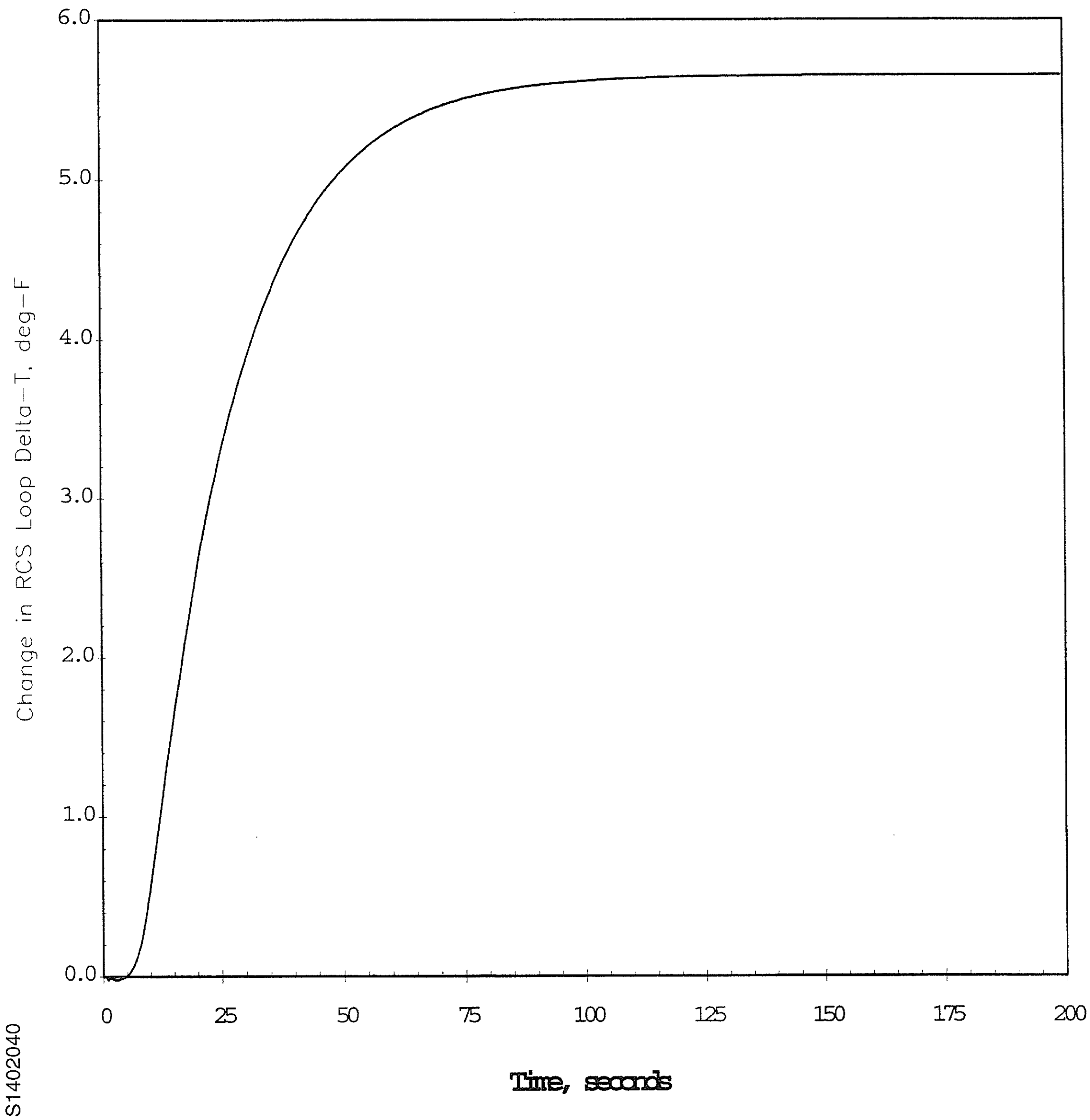

Figure 14.2-27
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC OFF (SELITURB) NUCLEAR POWER (% OF 2546 MWt)

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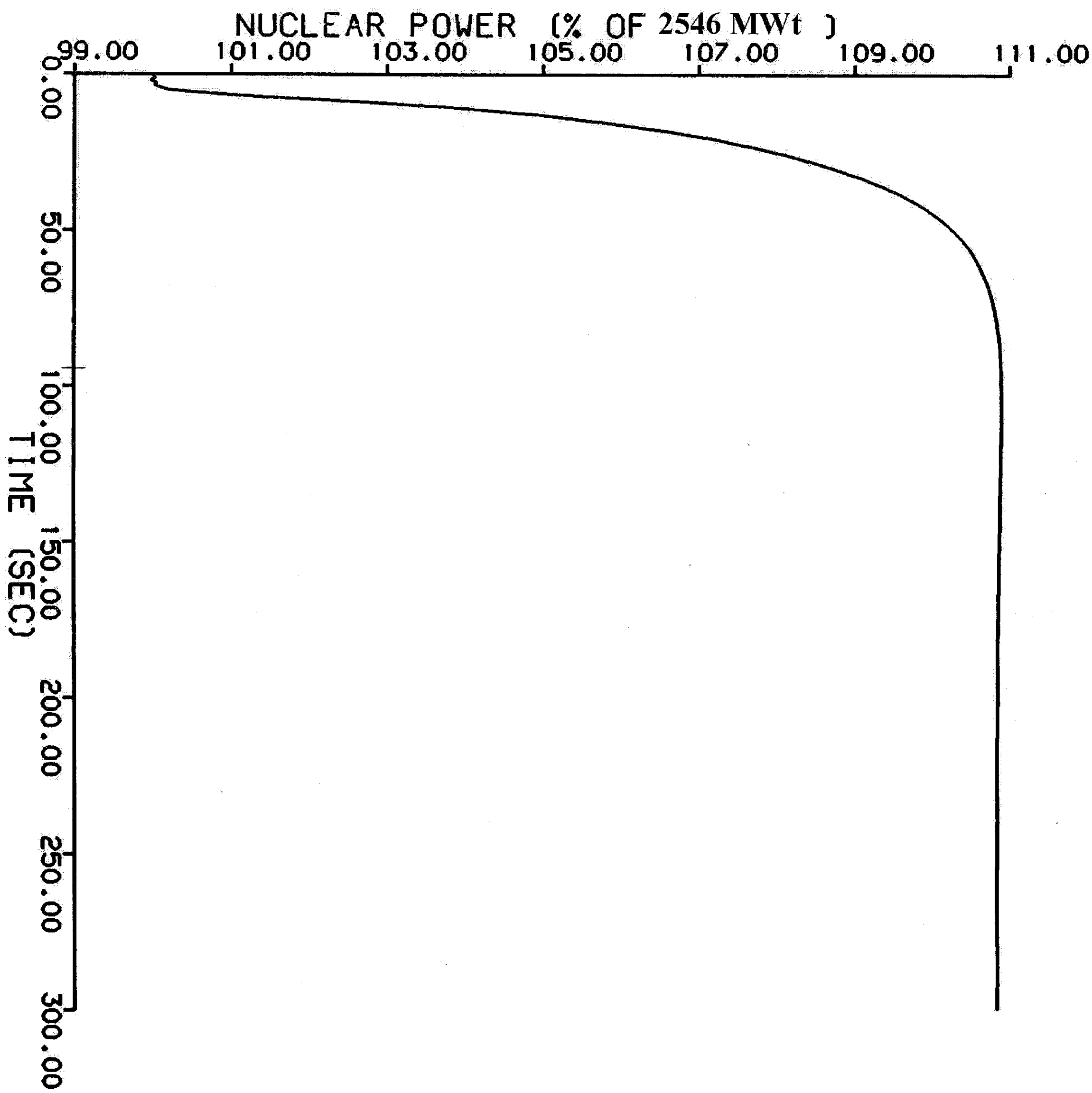

Figure 14.2-28
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC OFF (SELITURB) CHANGE IN PRESSURIZER PRESSURE

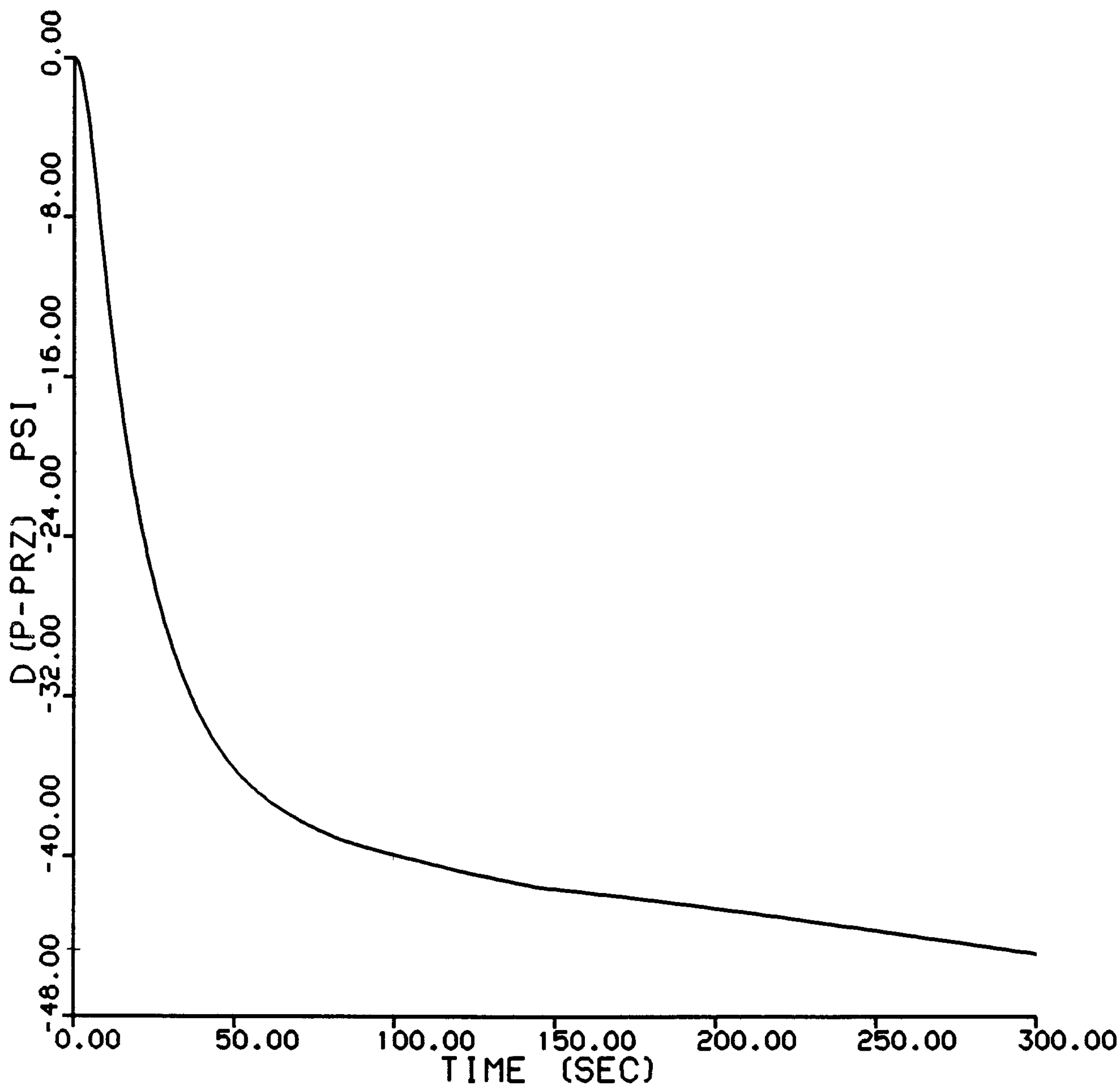

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Figure 14.2-29
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC OFF (SELITURB) CHANGE IN T_{avg}

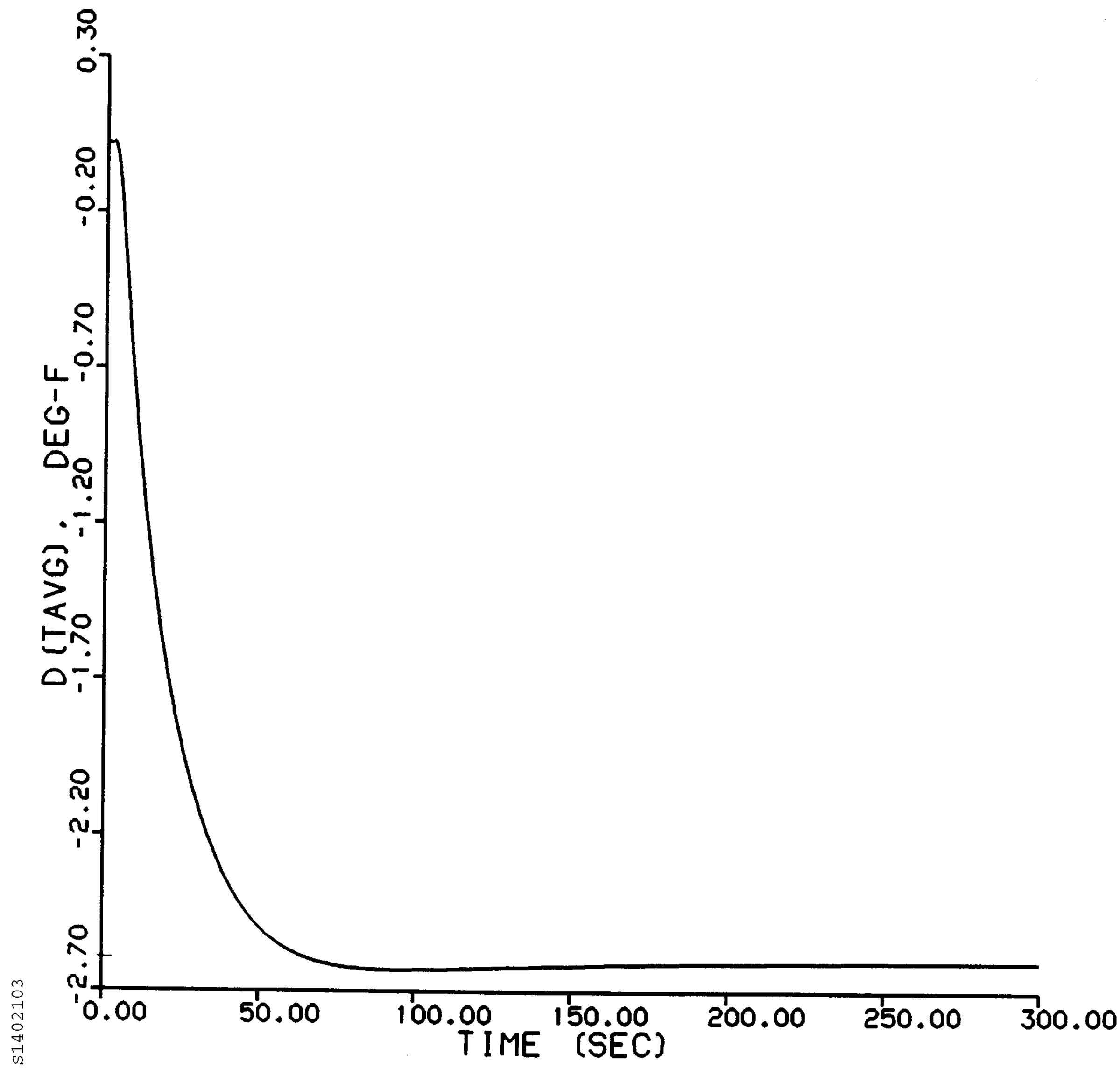

Figure 14.2-30
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC OFF (SELITURB) CHANGE IN ΔT

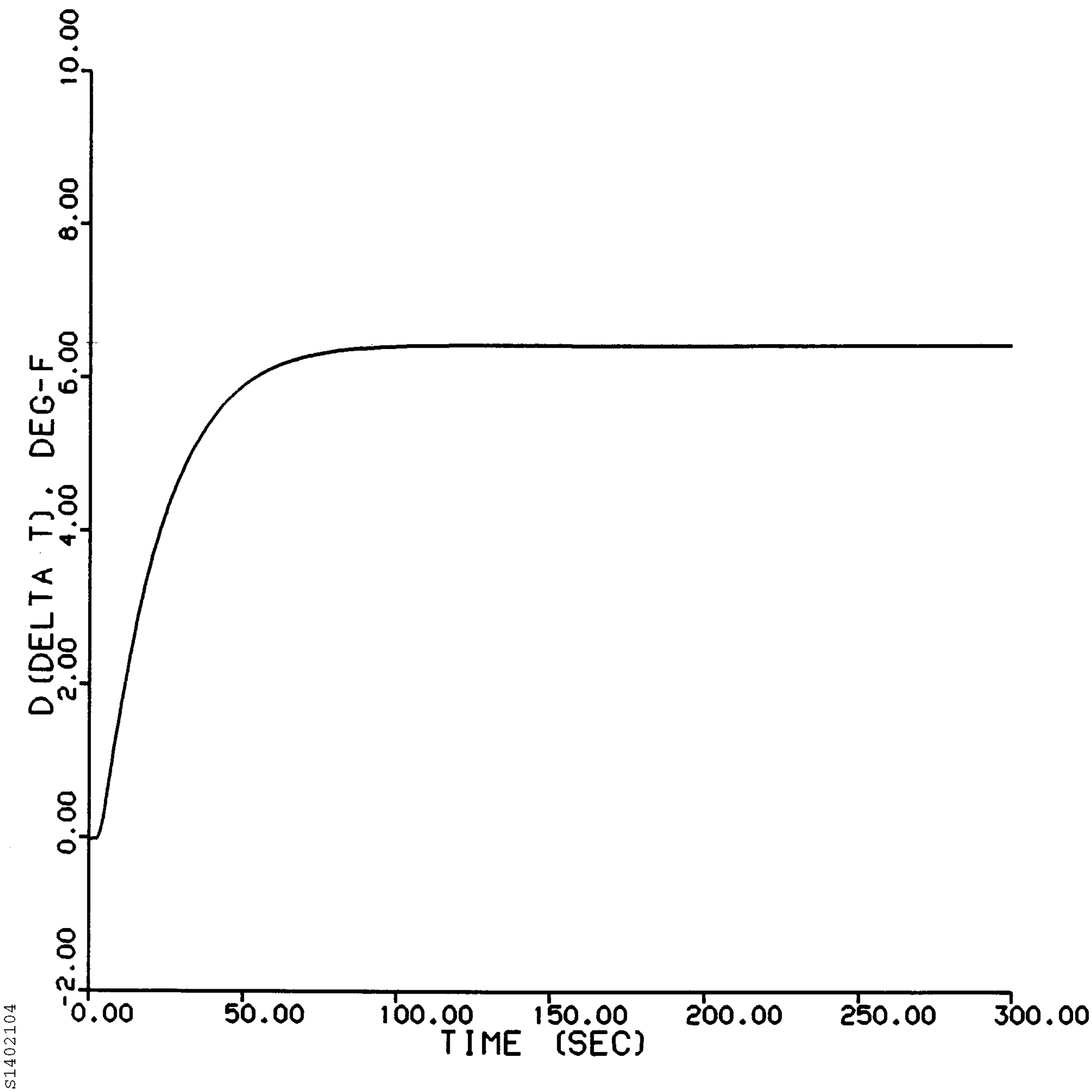

Figure 14.2-31
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC ON (SELITRBR) NUCLEAR POWER (% OF 2546 MWt)

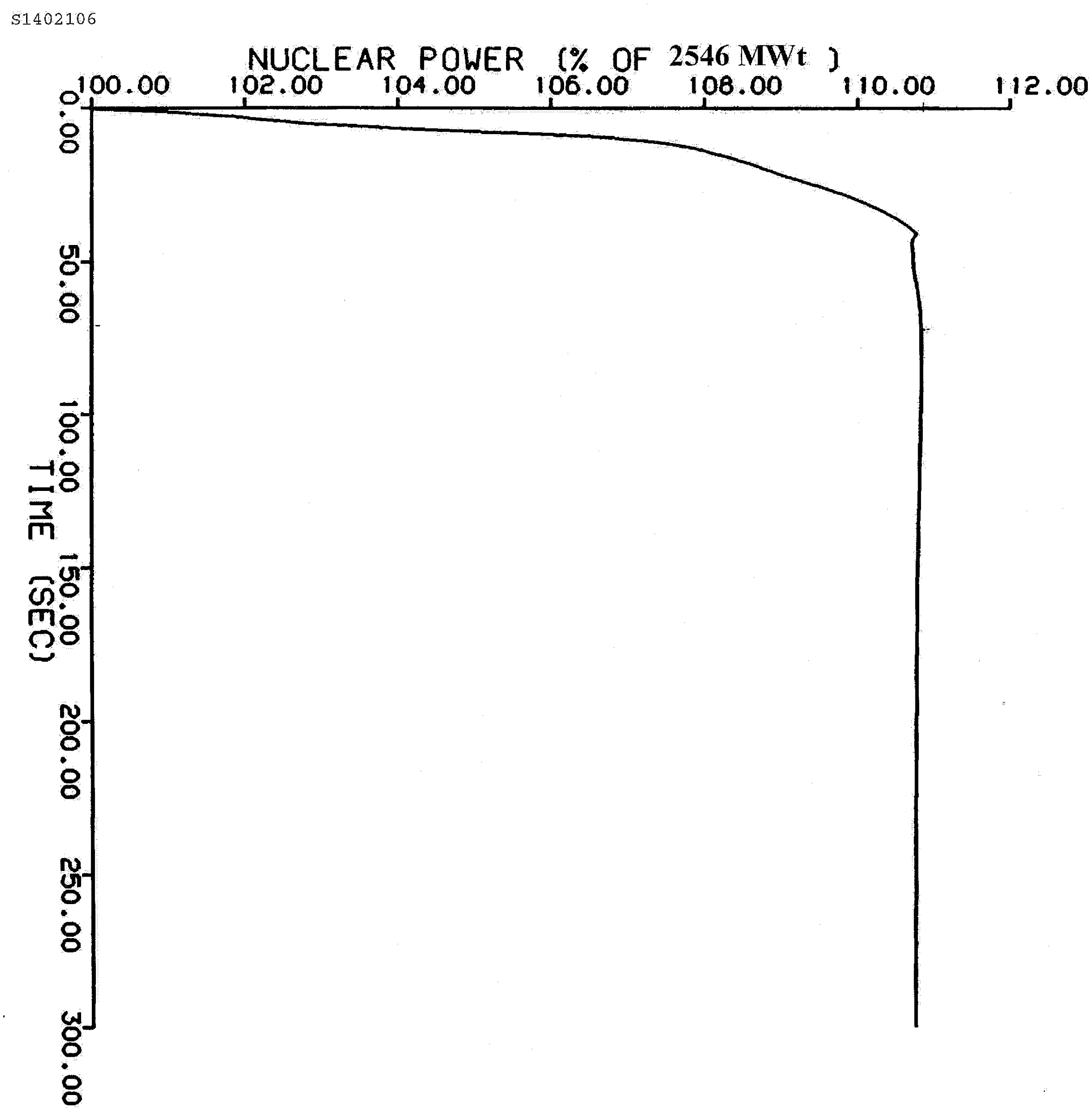

Figure 14.2-32
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC ON (SELITRBR) CHANGE IN PRESSURIZER PRESSURE

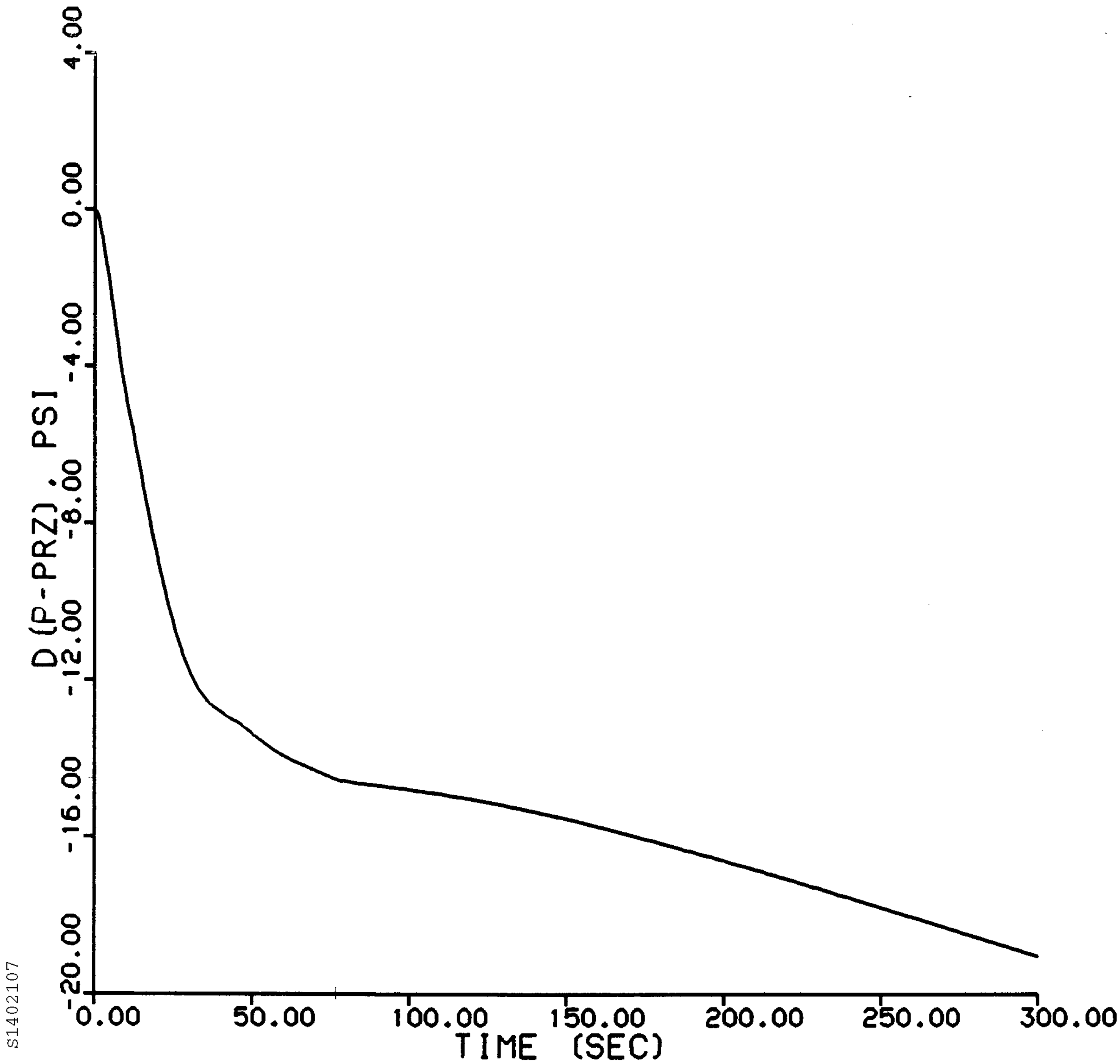

Figure 14.2-33
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC ON (SELITRBR) CHANGE IN T_{avg}

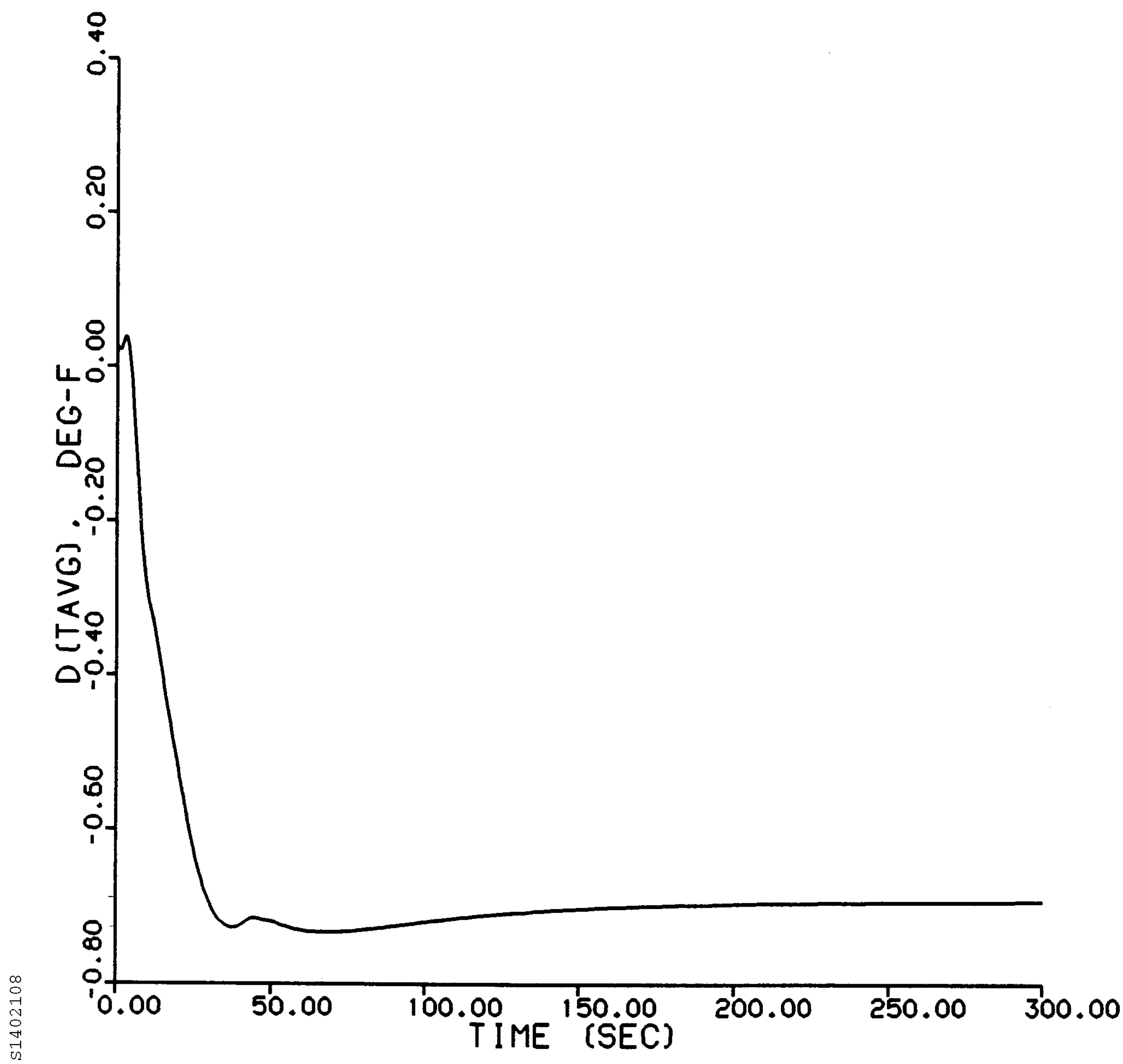

Figure 14.2-34
SURRY EXCESSIVE LOAD INCREASE HFP EOC 110% TURB FLOW
(AT 2546 MWt) RC ON (SELITRBR) CHANGE IN ΔT

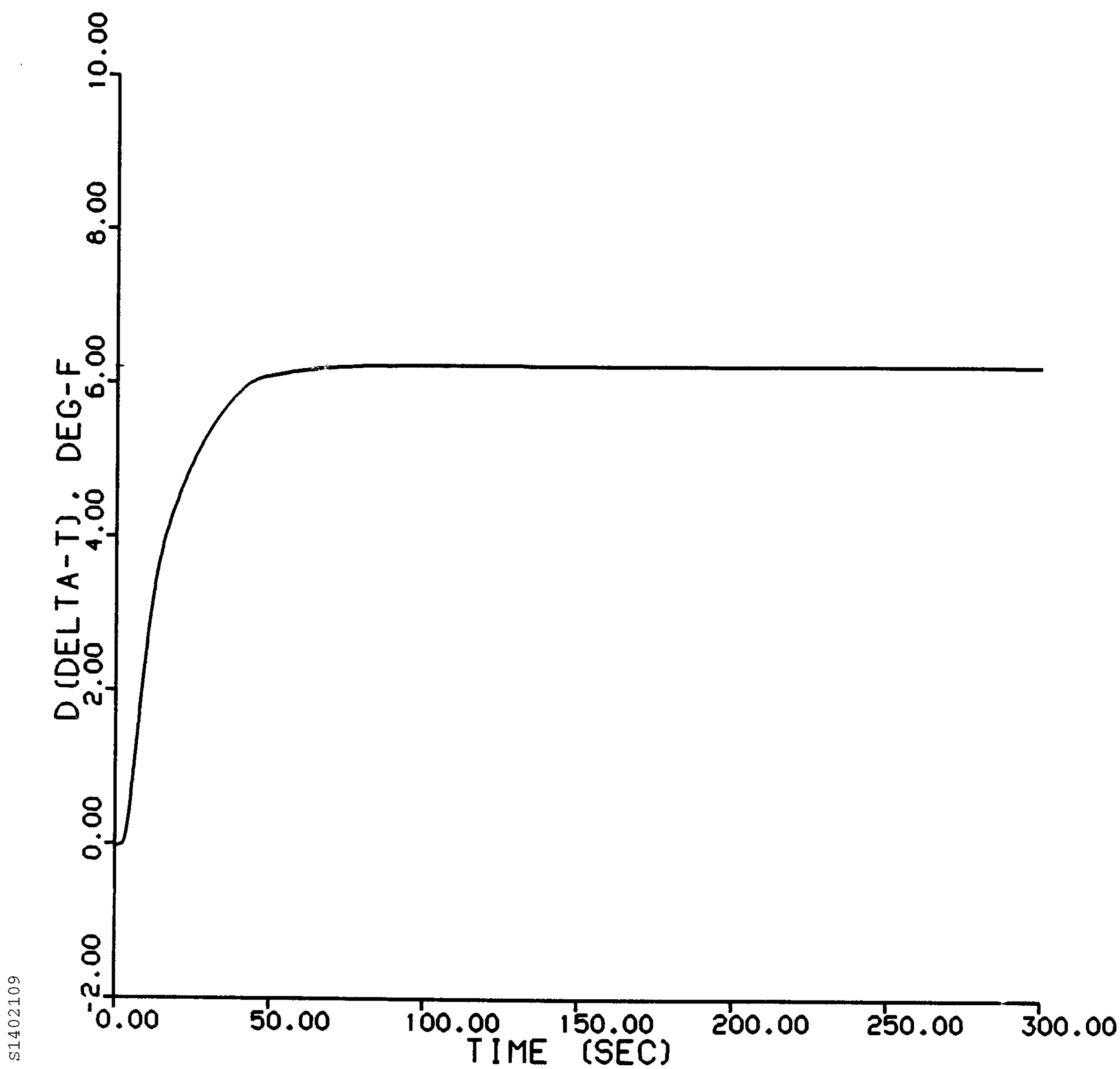

Figure 14.2-35
SURRY EXCESSIVE LOAD INCREASE HFP BOC 110% TURB FLOW
(AT 2546 MWt) (SELIBOCR) NUCLEAR POWER (% OF 2546 MWt)

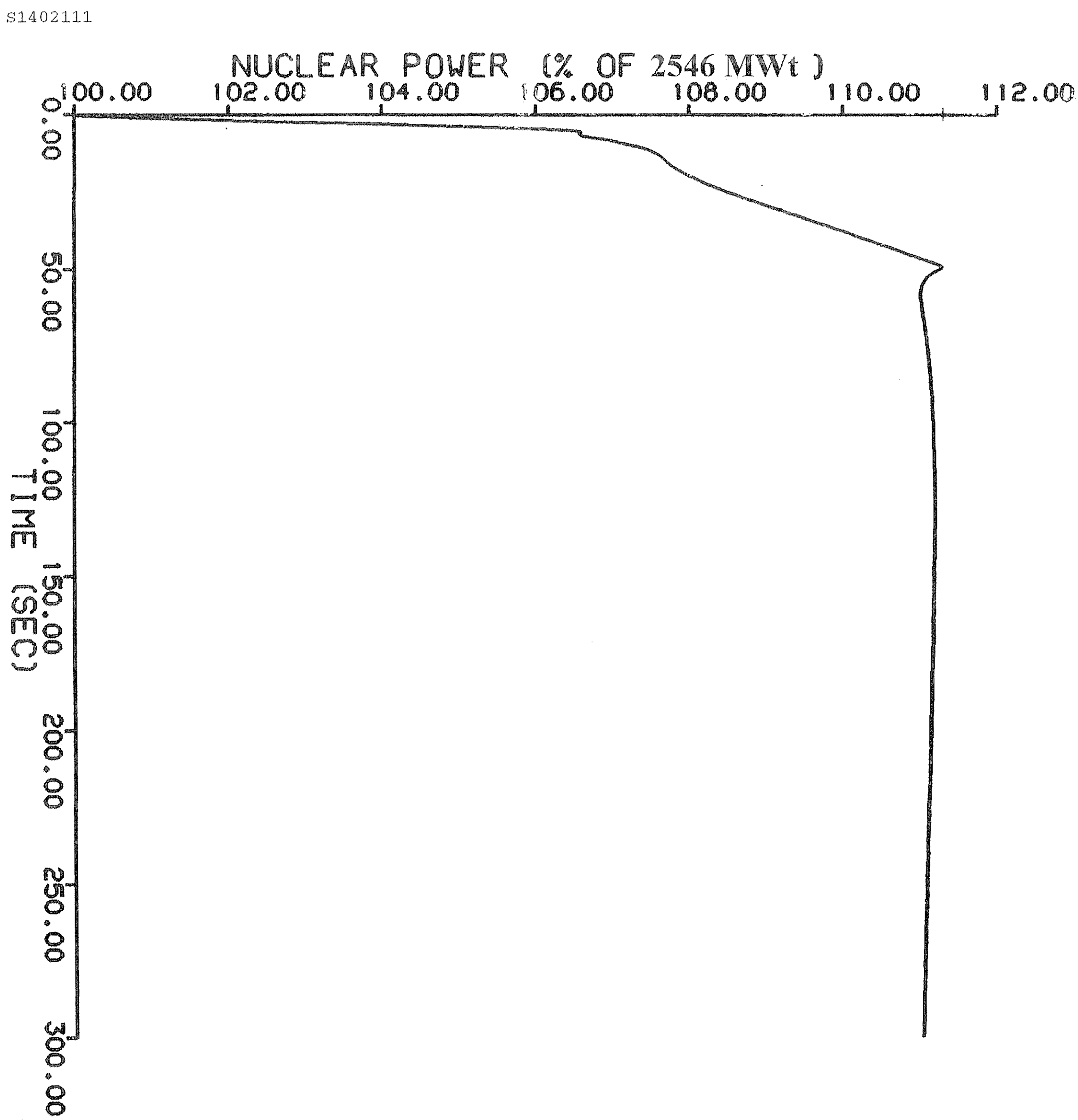

Figure 14.2-36
SURRY EXCESSIVE LOAD INCREASE HFP BOC 110% TURB FLOW
(AT 2546 MWt) (SELIBOCR) CHANGE IN PRESSURIZER PRESSURE

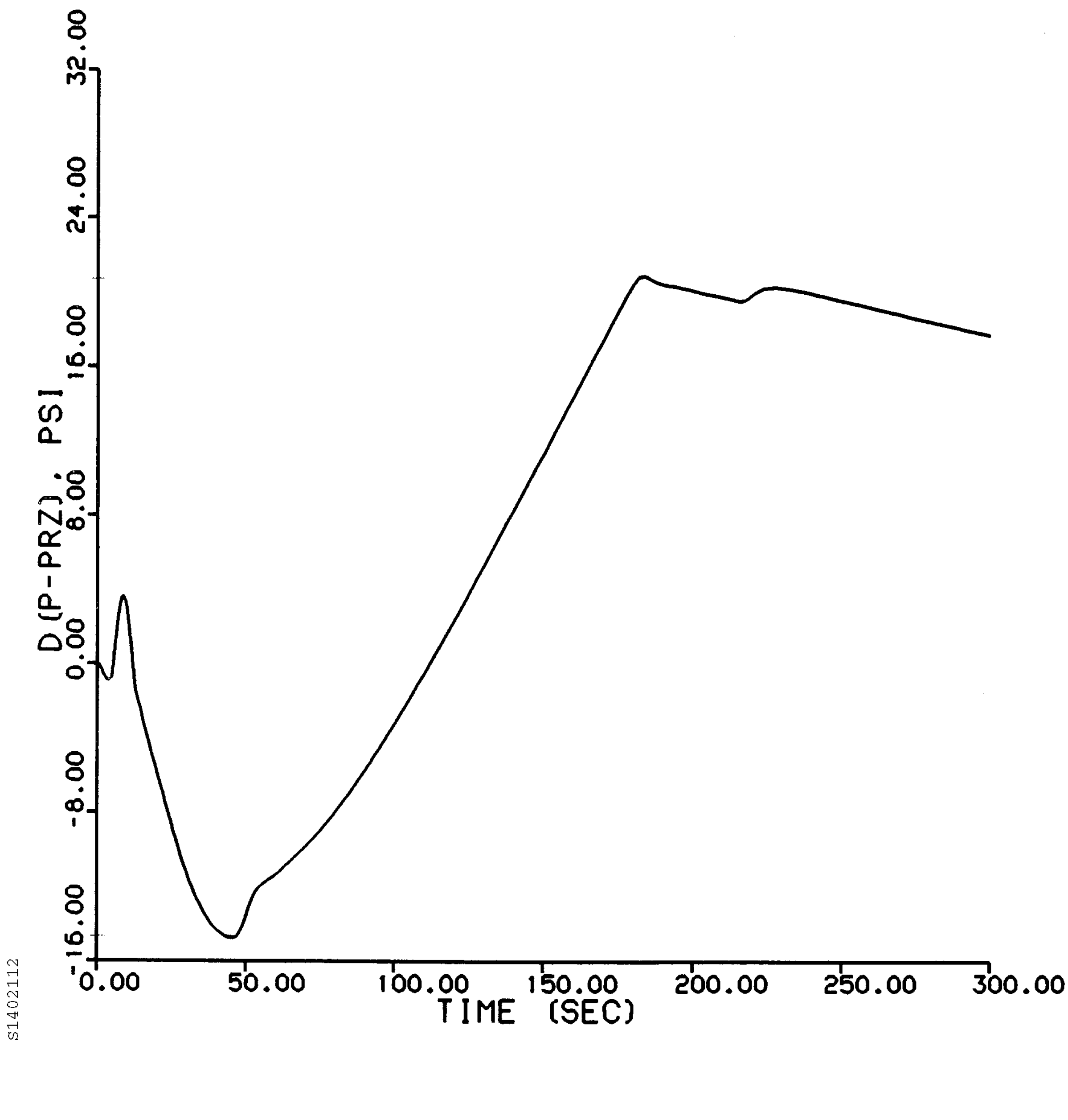

Figure 14.2-37
SURRY EXCESSIVE LOAD INCREASE HFP BOC 110% TURB FLOW
(AT 2546 MWt) (SELIBOCR) CHANGE IN T_{avg}

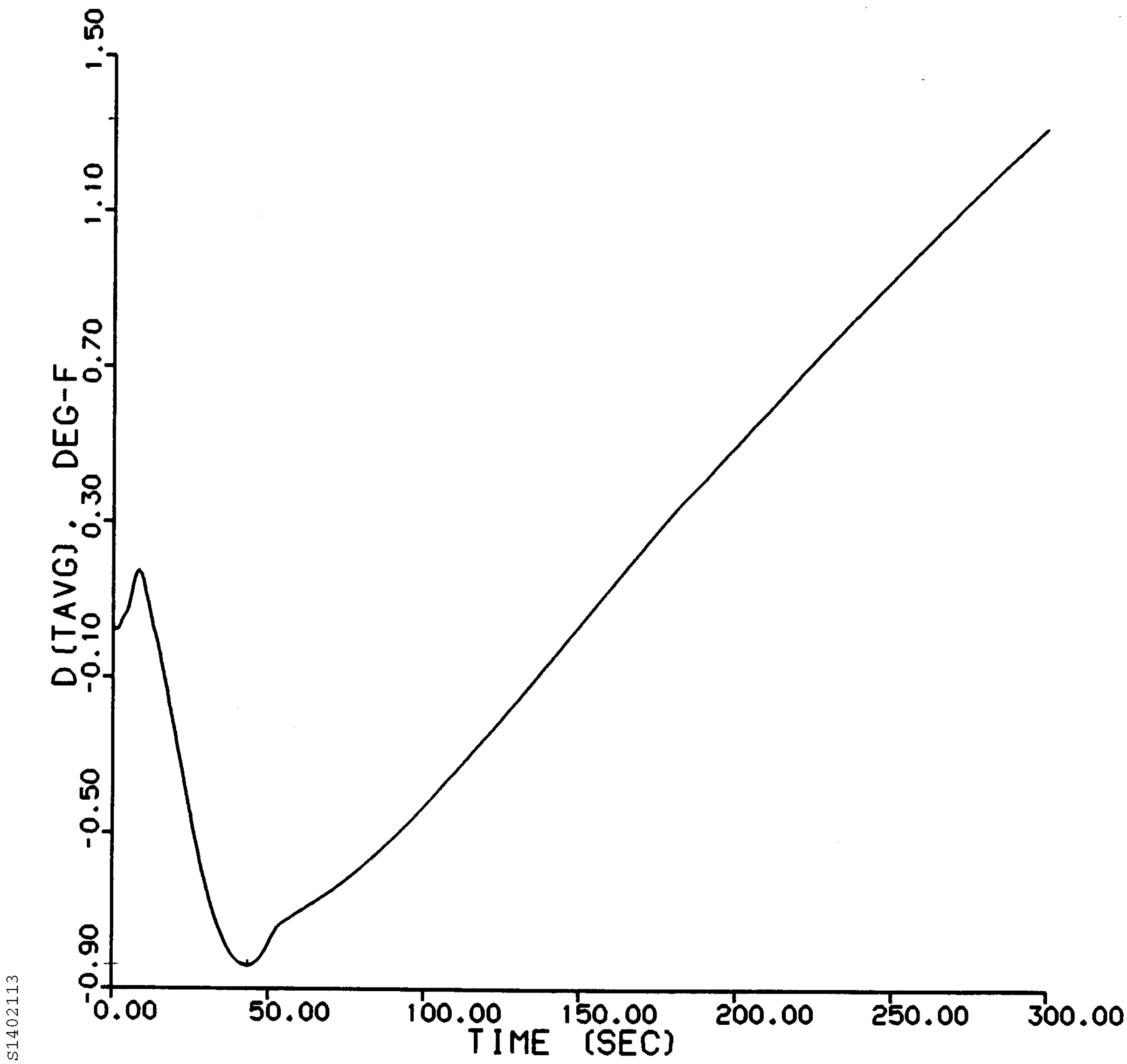

Figure 14.2-38
SURRY EXCESSIVE LOAD INCREASE HFP BOC 110% TURB FLOW
(AT 2546 MWt) (SELIBOCR) CHANGE IN ΔT

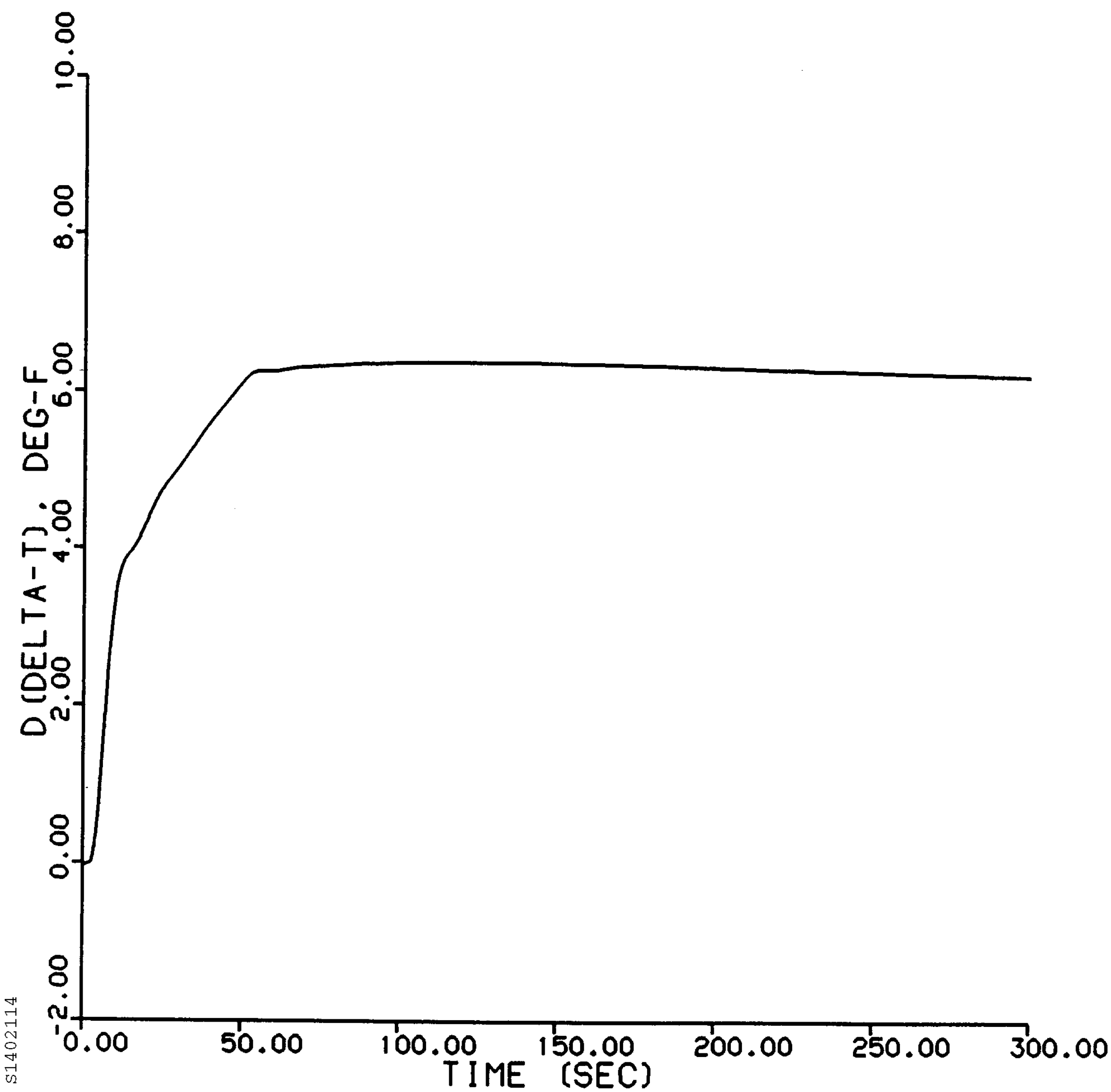

Figure 14.2-39
COMPLETE LOSS OF FLOW - UNDERVOLTAGE CASE RCS MASS FLOW RATE

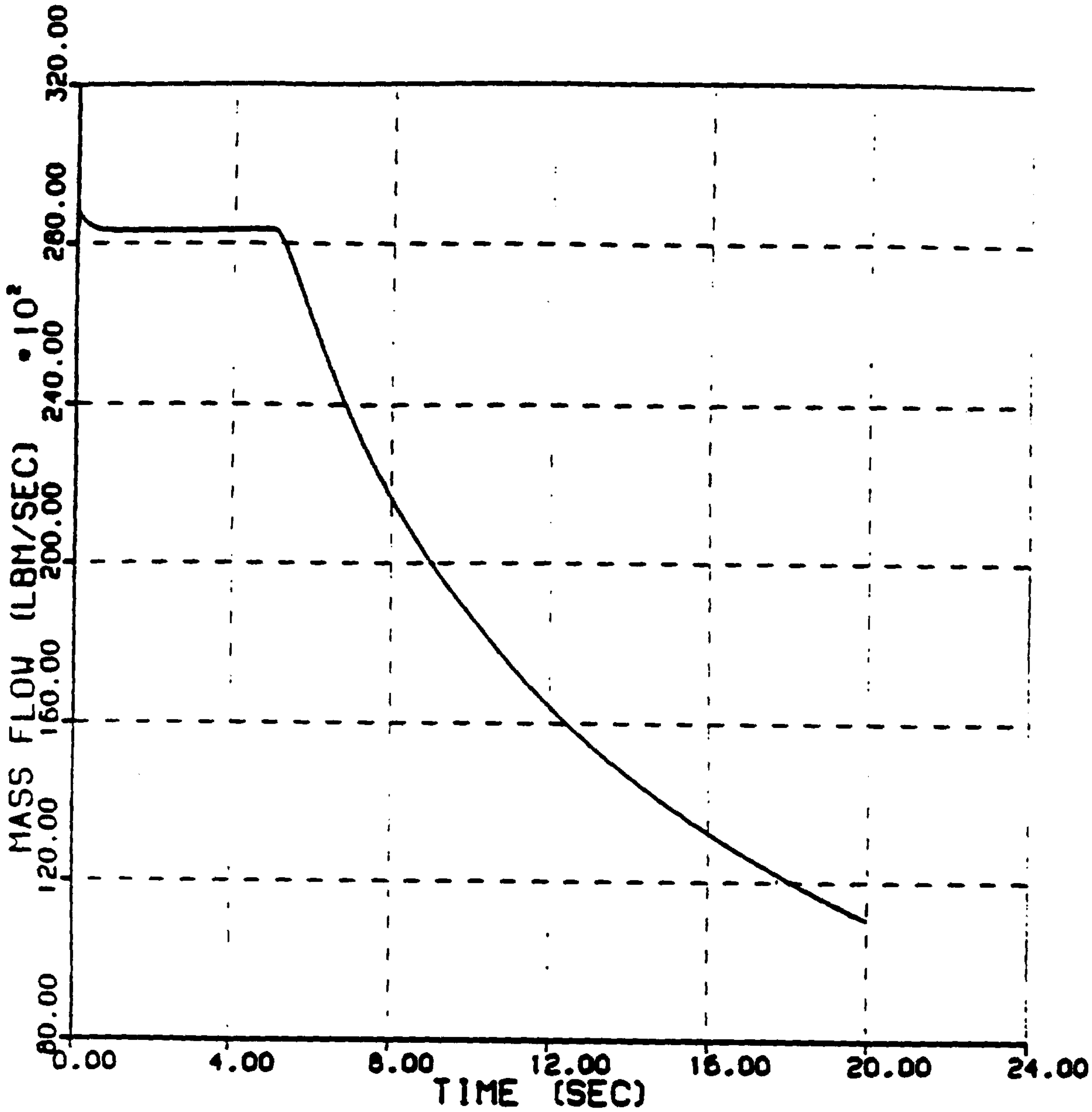

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Figure 14.2-40
COMPLETE LOSS OF FLOW - UNDERFREQUENCY CASE RCS MASS FLOW RATE

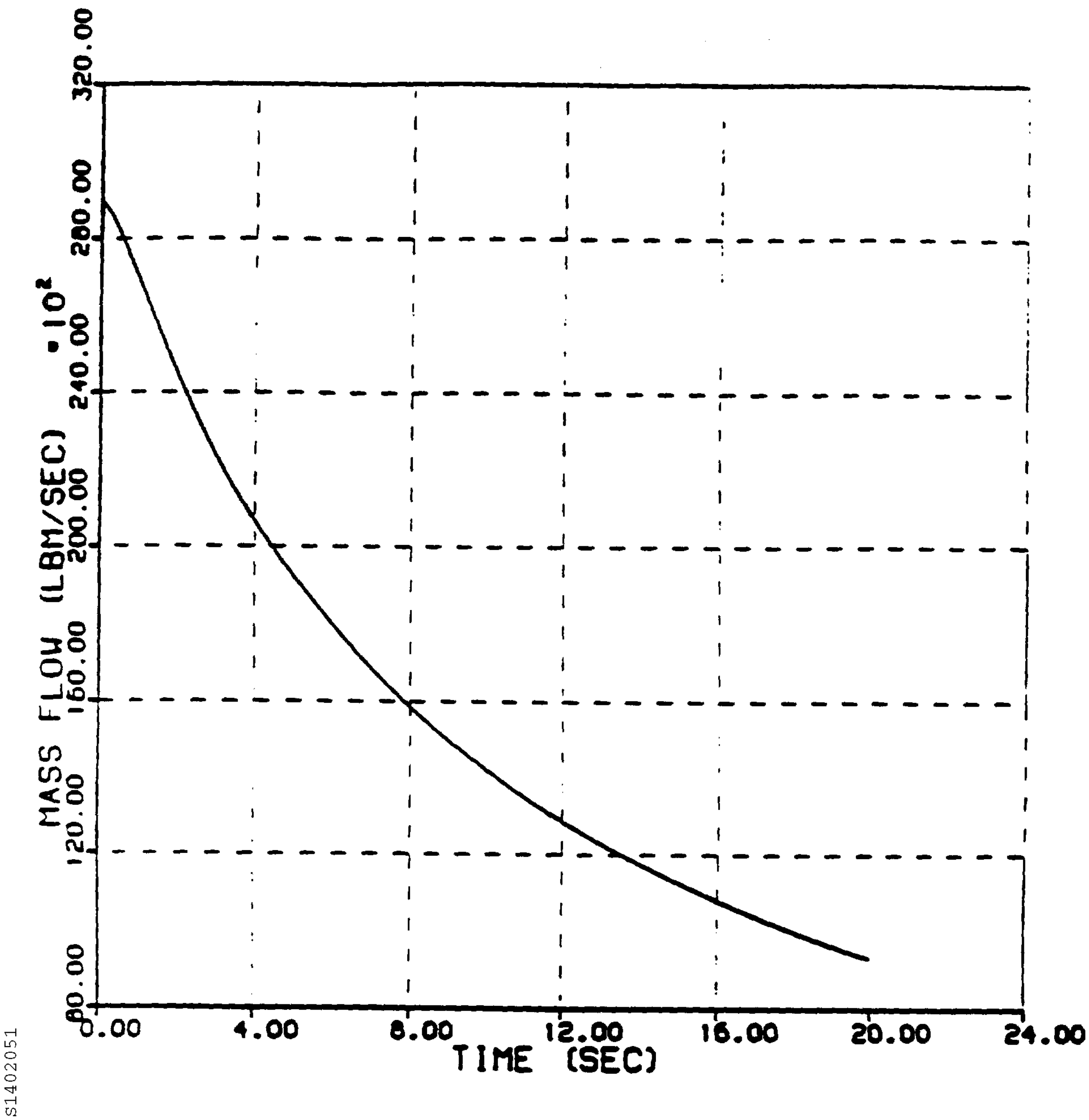

Figure 14.2-41
COMPLETE LOSS OF FLOW - UNDERVOLTAGE CASE
NUCLEAR POWER (% OF 2546 MWt)

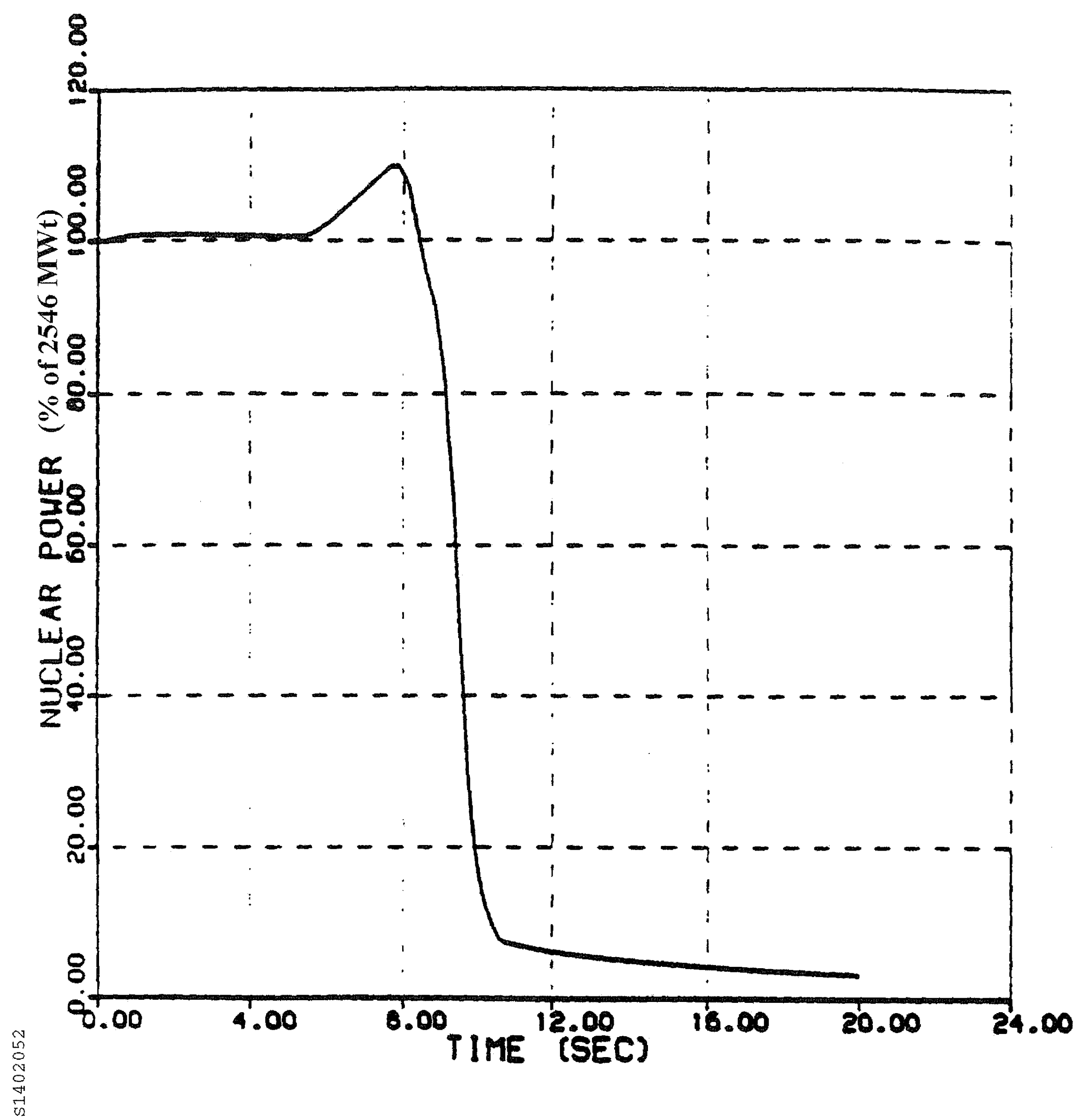

Figure 14.2-42
COMPLETE LOSS OF FLOW - UNDERVOLTAGE CASE CORE INLET TEMPERATURE

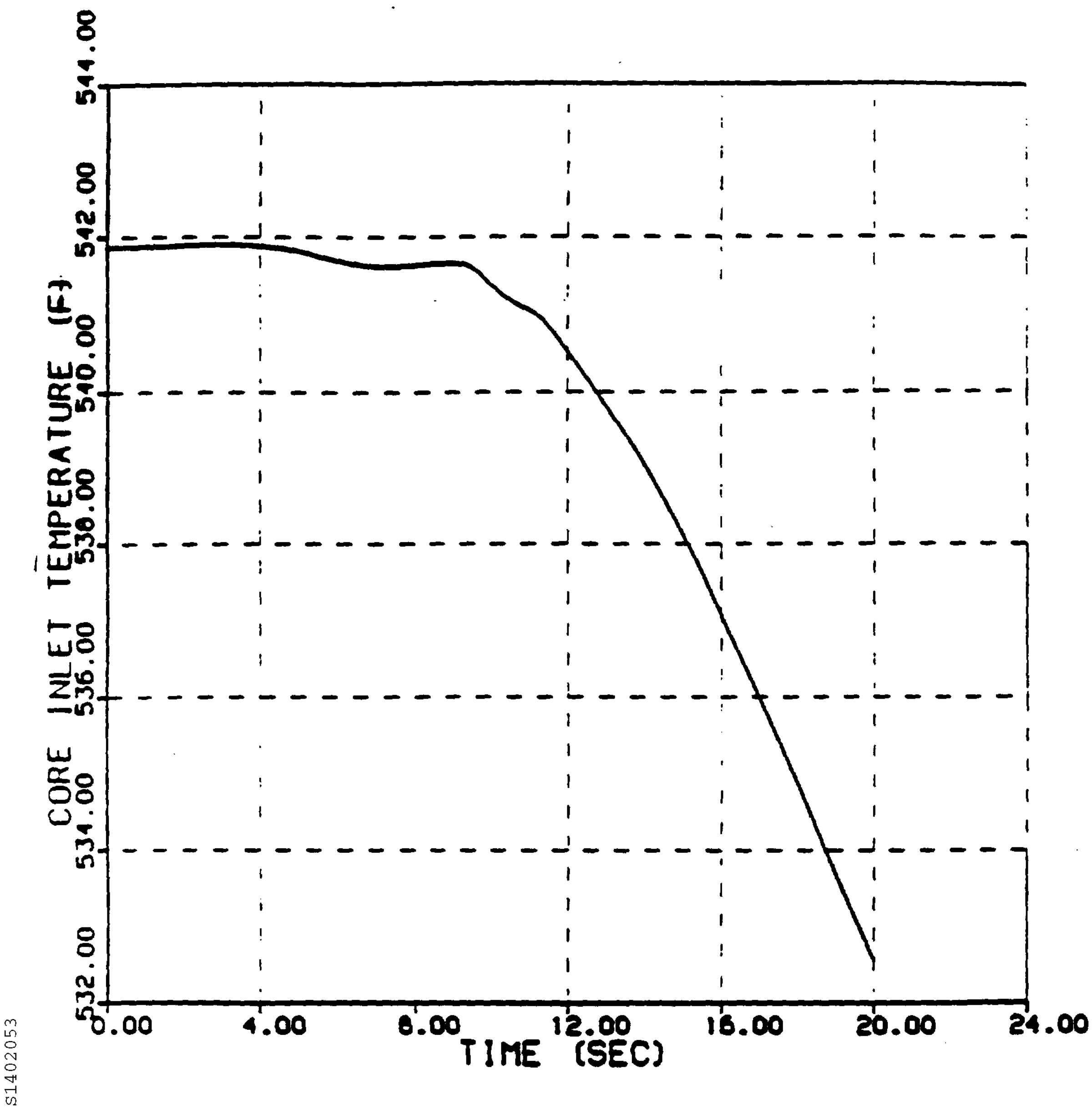

Figure 14.2-43
COMPLETE LOSS OF FLOW - UNDERVOLTAGE CASE
RCS AVERAGE TEMPERATURE

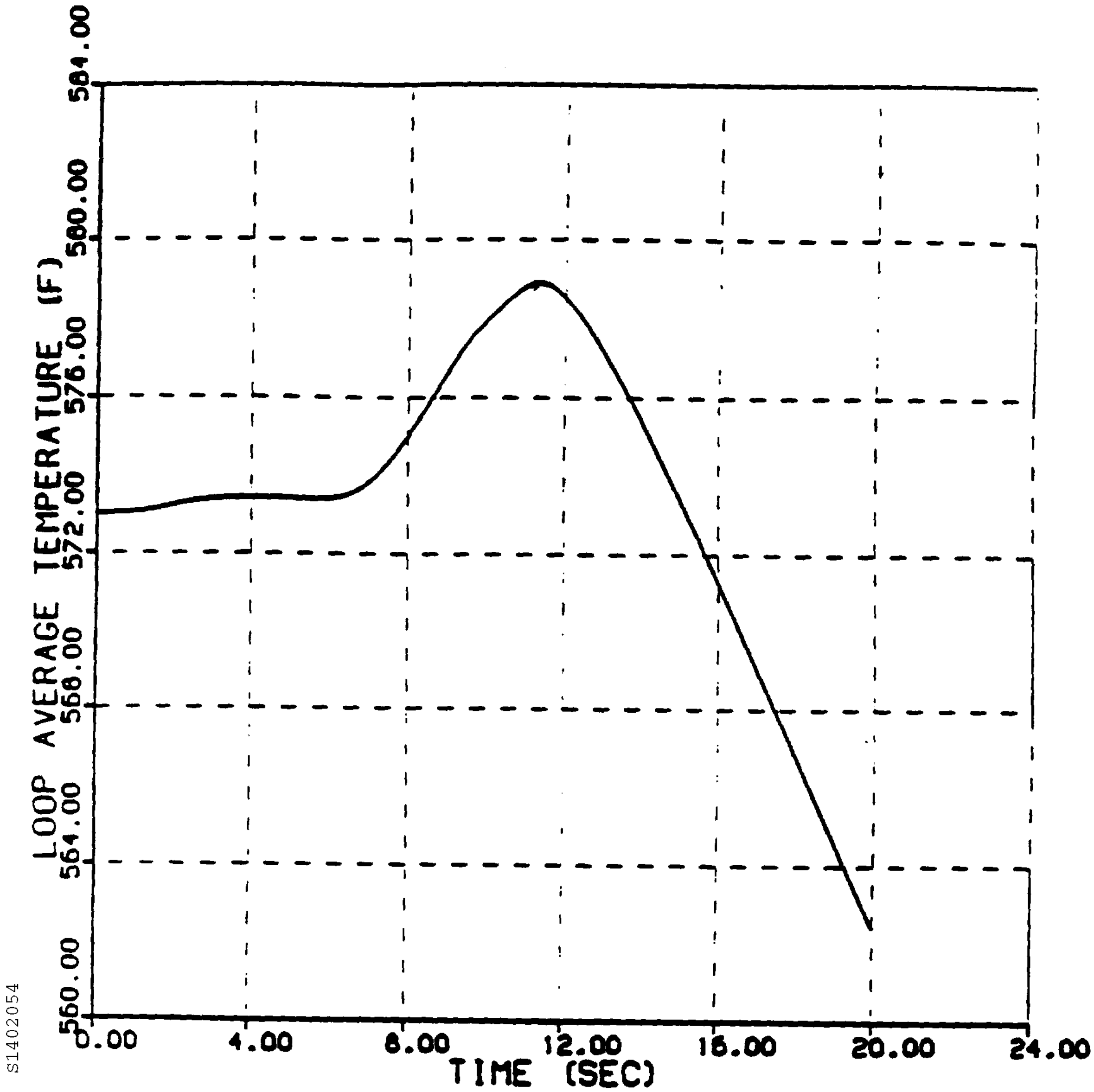

Figure 14.2-44
COMPLETE LOSS OF FLOW - UNDERVOLTAGE CASE PRESSURIZER PRESSURE

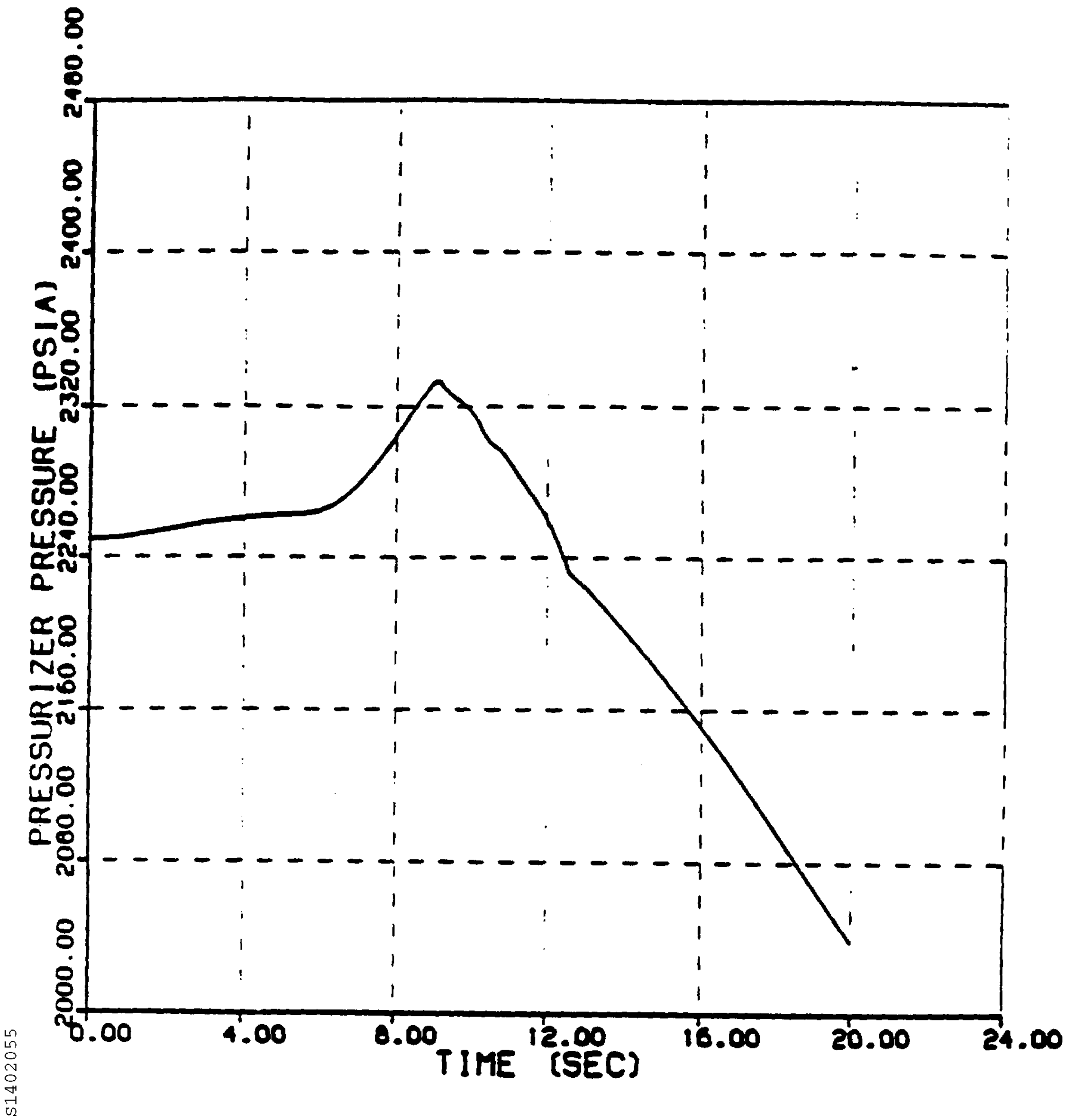

Figure 14.2-45
COMPLETE LOSS OF FLOW - UNDERFREQUENCY CASE
NUCLEAR POWER(% OF 2546 MWt)

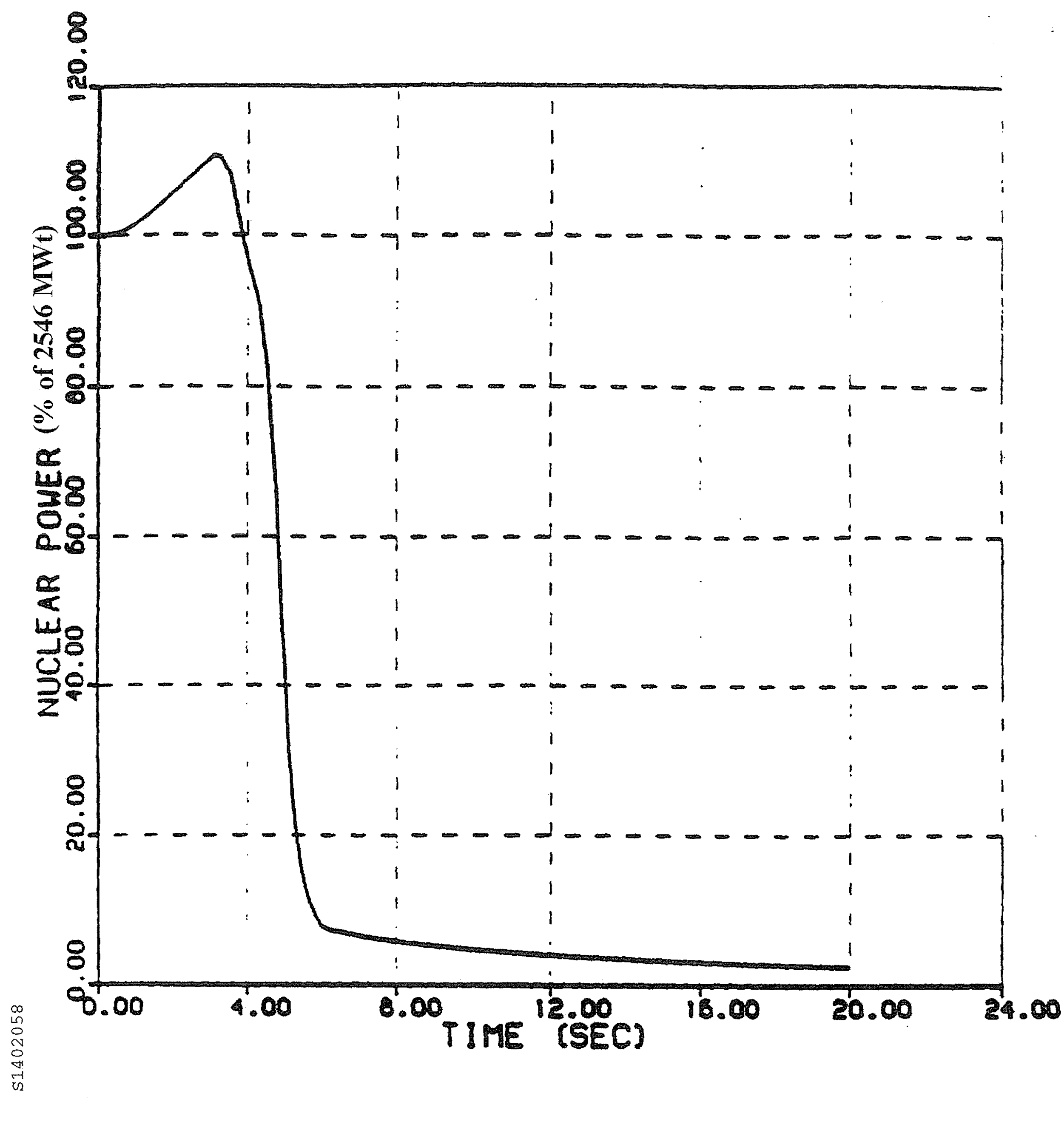

Figure 14.2-46
COMPLETE LOSS OF FLOW - UNDERFREQUENCY CASE
CORE INLET TEMPERATURE

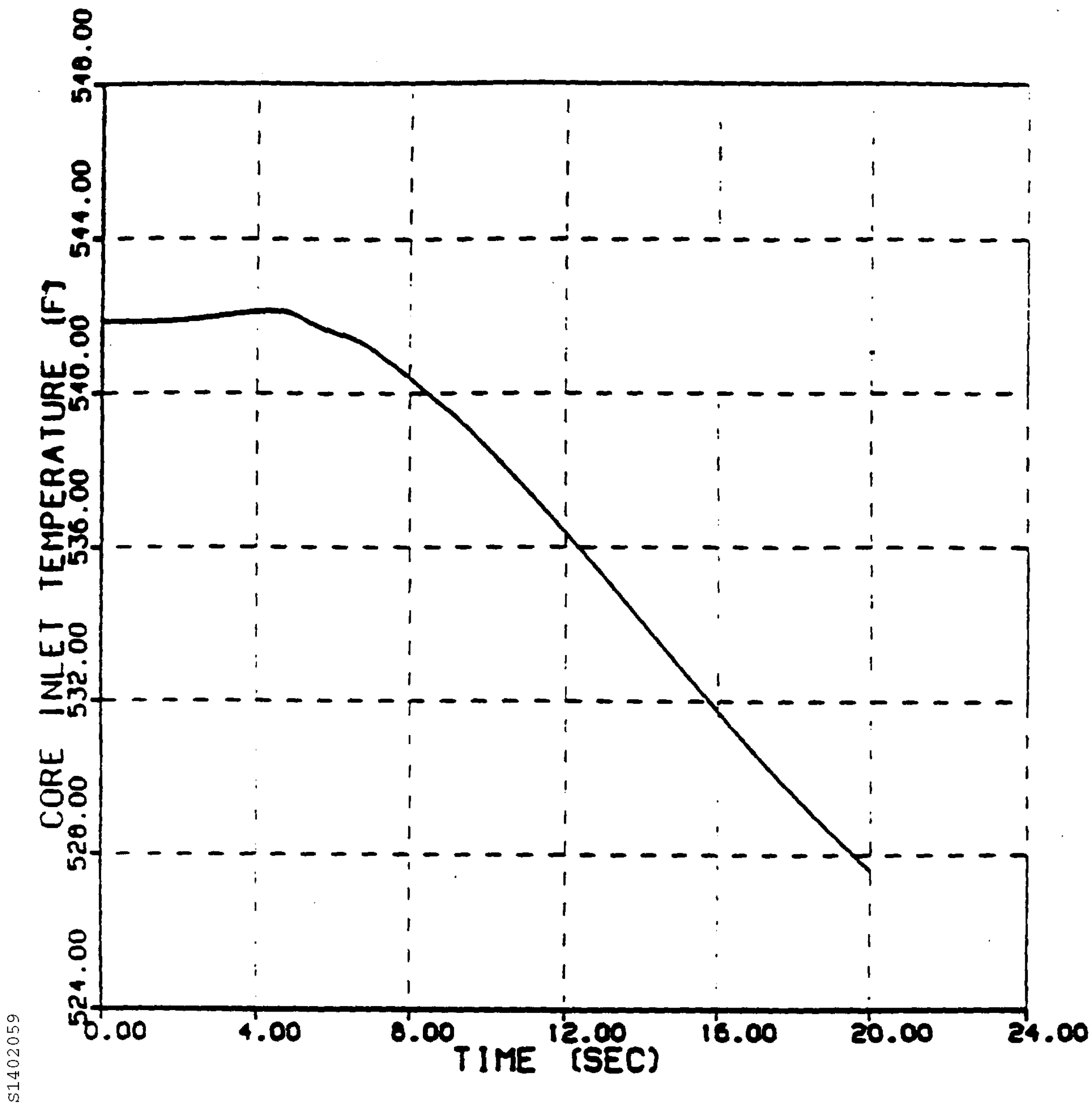

Figure 14.2-47
COMPLETE LOSS OF FLOW - UNDERFREQUENCY CASE
RCS AVERAGE TEMPERATURE

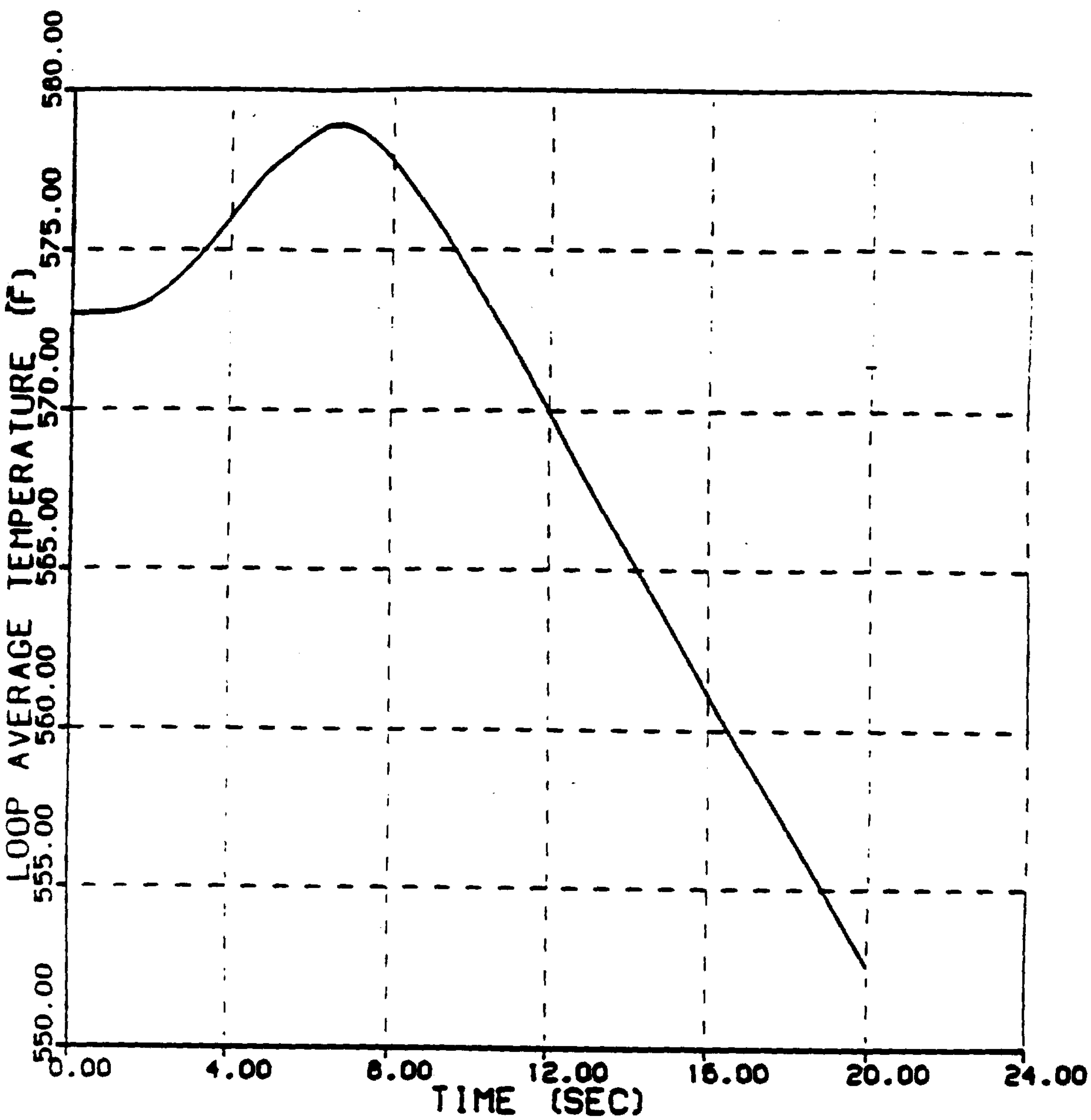

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Figure 14.2-48
COMPLETE LOSS OF FLOW - UNDERFREQUENCY CASE PRESSURIZER PRESSURE

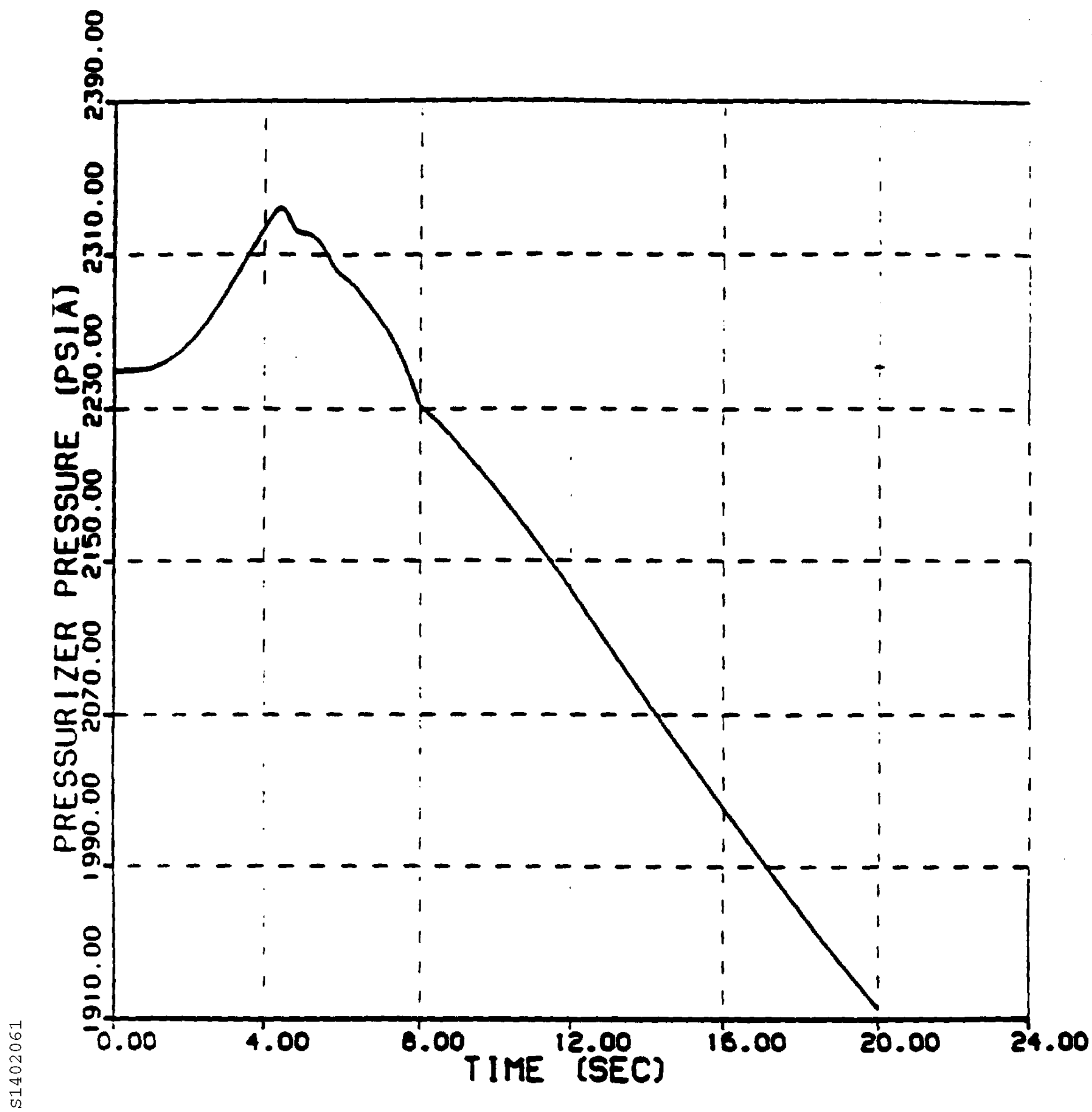

Figure 14.2-49
LOCKED ROTOR - DNBR ANALYSIS CASE CORE INLET MASS FLOW RATE

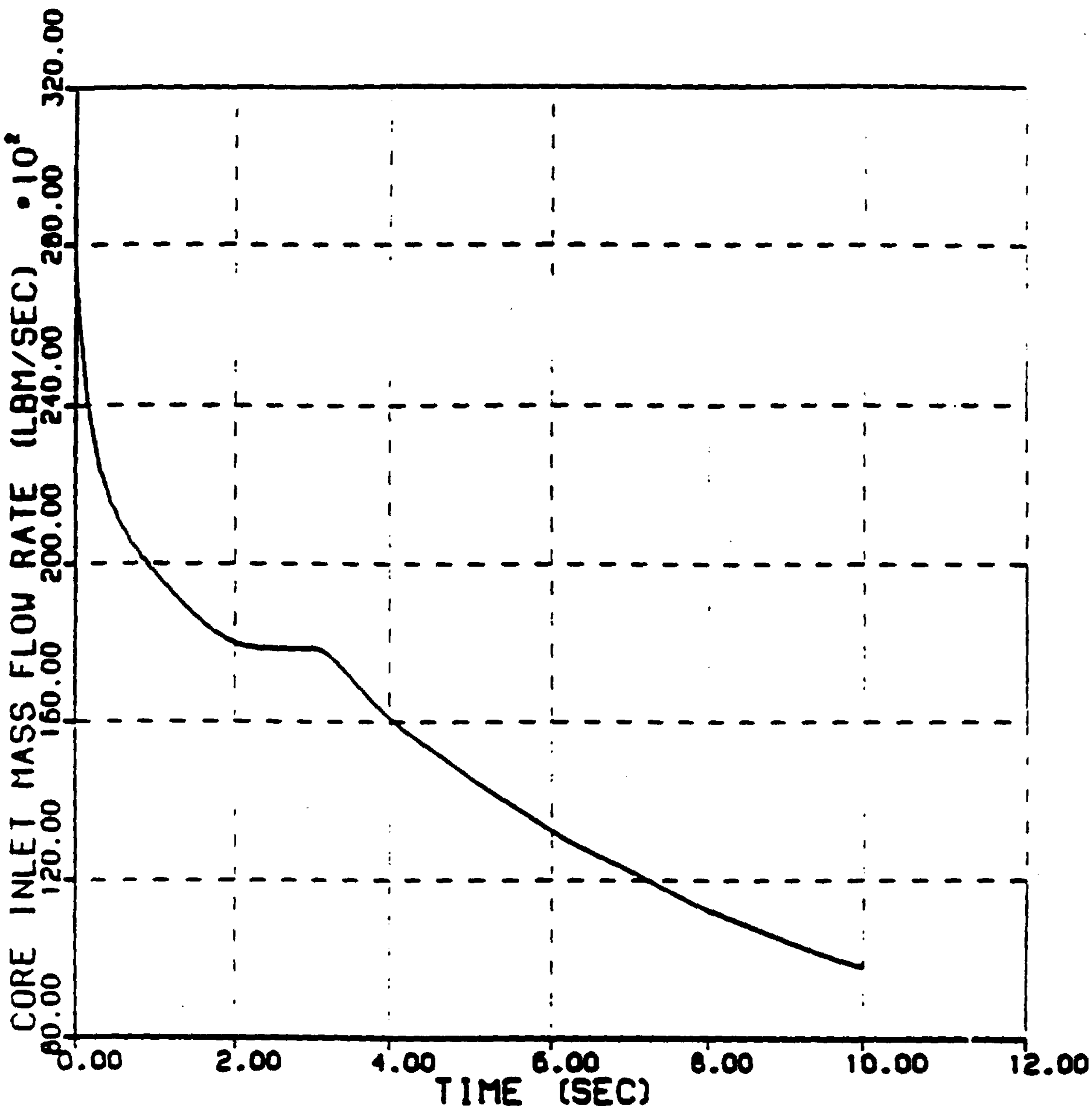

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Figure 14.2-50
LOCKED ROTOR - DNBR ANALYSIS CASE CORE HEAT FLUX

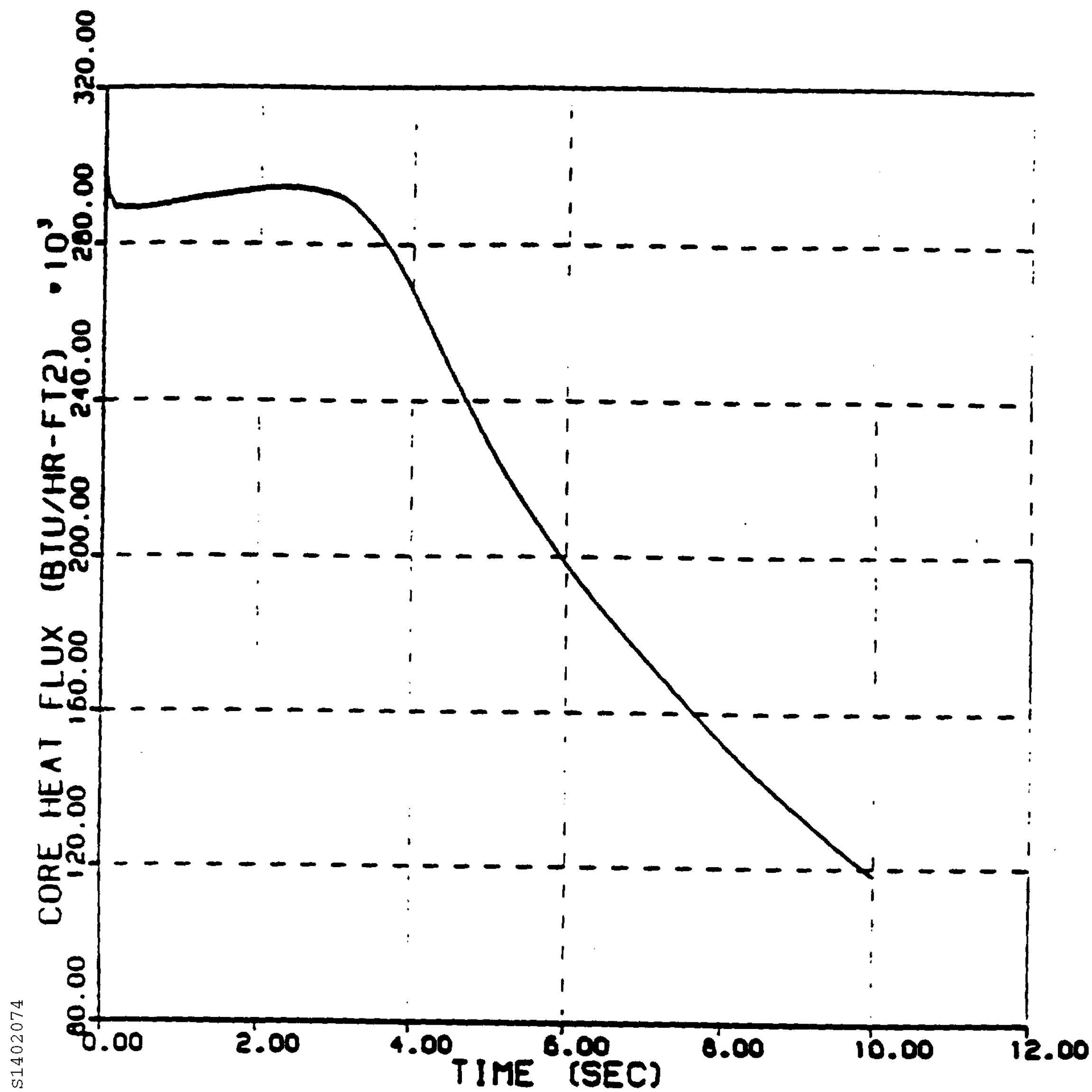

Figure 14.2-51
LOCKED ROTOR - DNBR ANALYSIS CASE CORE INLET TEMPERATURE

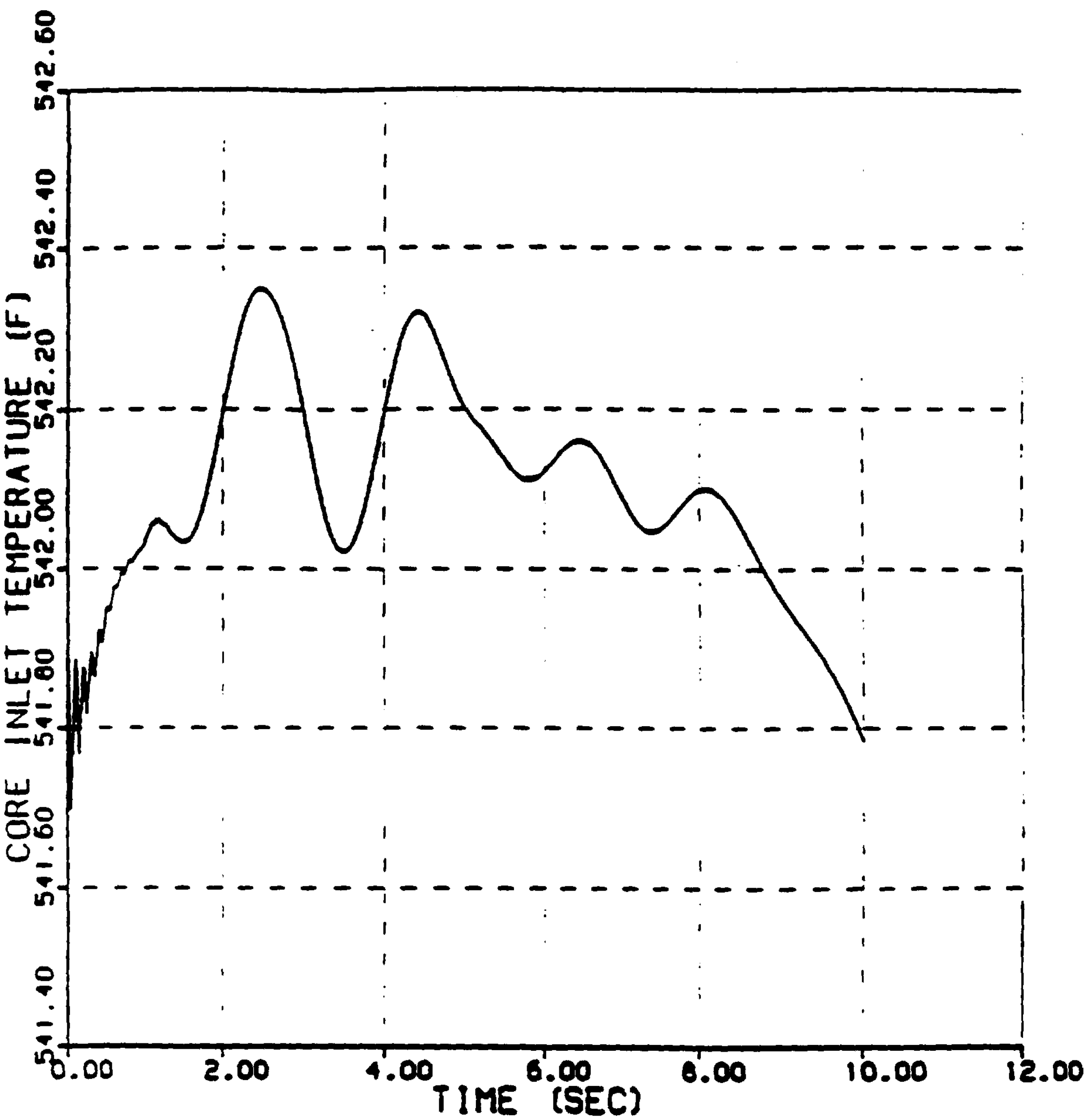

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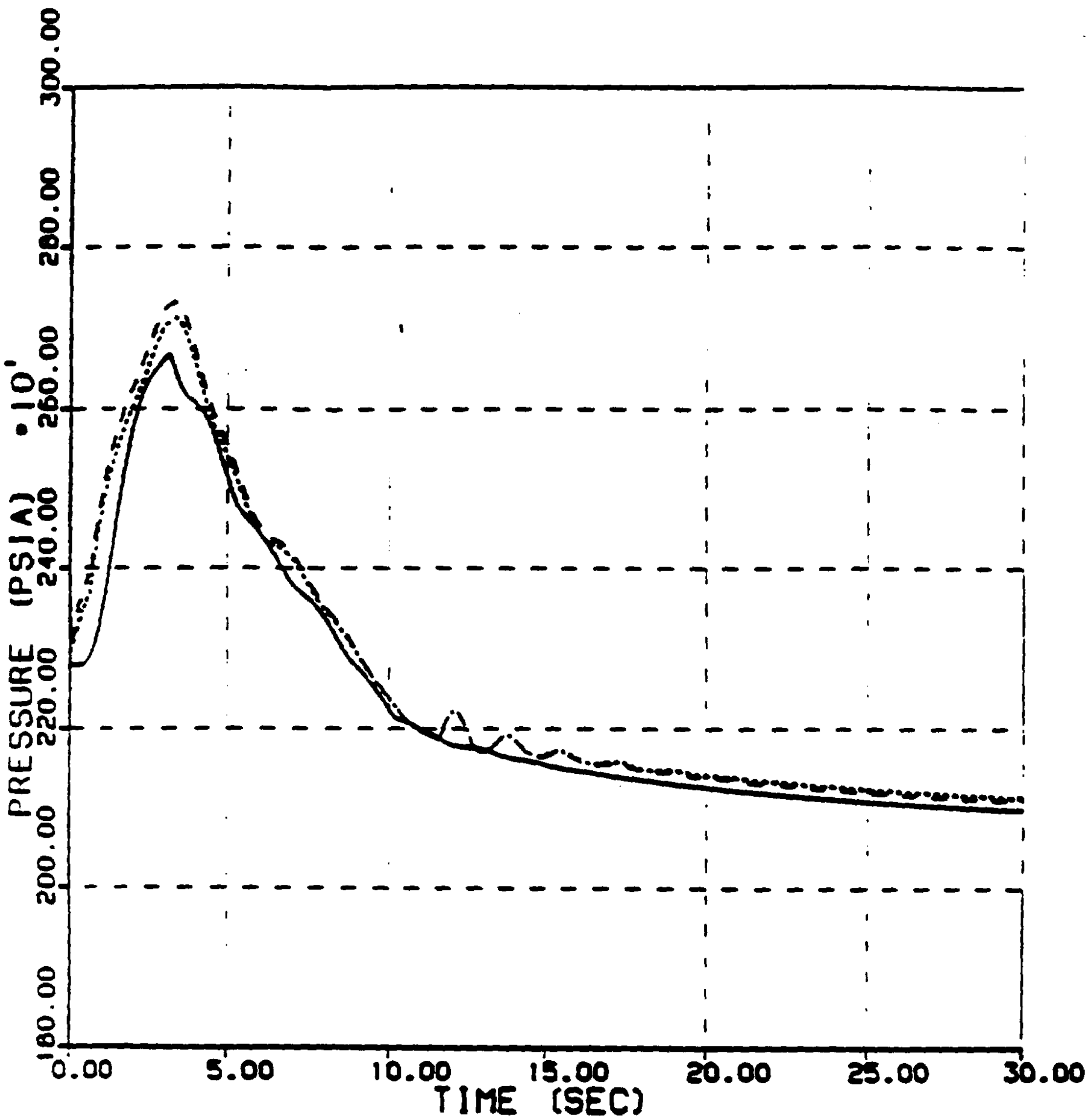
Figure 14.2-52
LOCKED ROTOR - RCS OVERPRESSURE CASE
RCS PRESSURE - PRESSURIZER, RCP EXIT, LOWER PLENUM



LINE - PRESSURIZER PRESSURE
DASHED - RCP EXIT (UNAFFECTED LOOP)
DOTTED - LOWER PLENUM

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Figure 14.2-53
LOCKED ROTOR - RCS OVERPRESSURE CASE STEAM GENERATOR PRESSURE

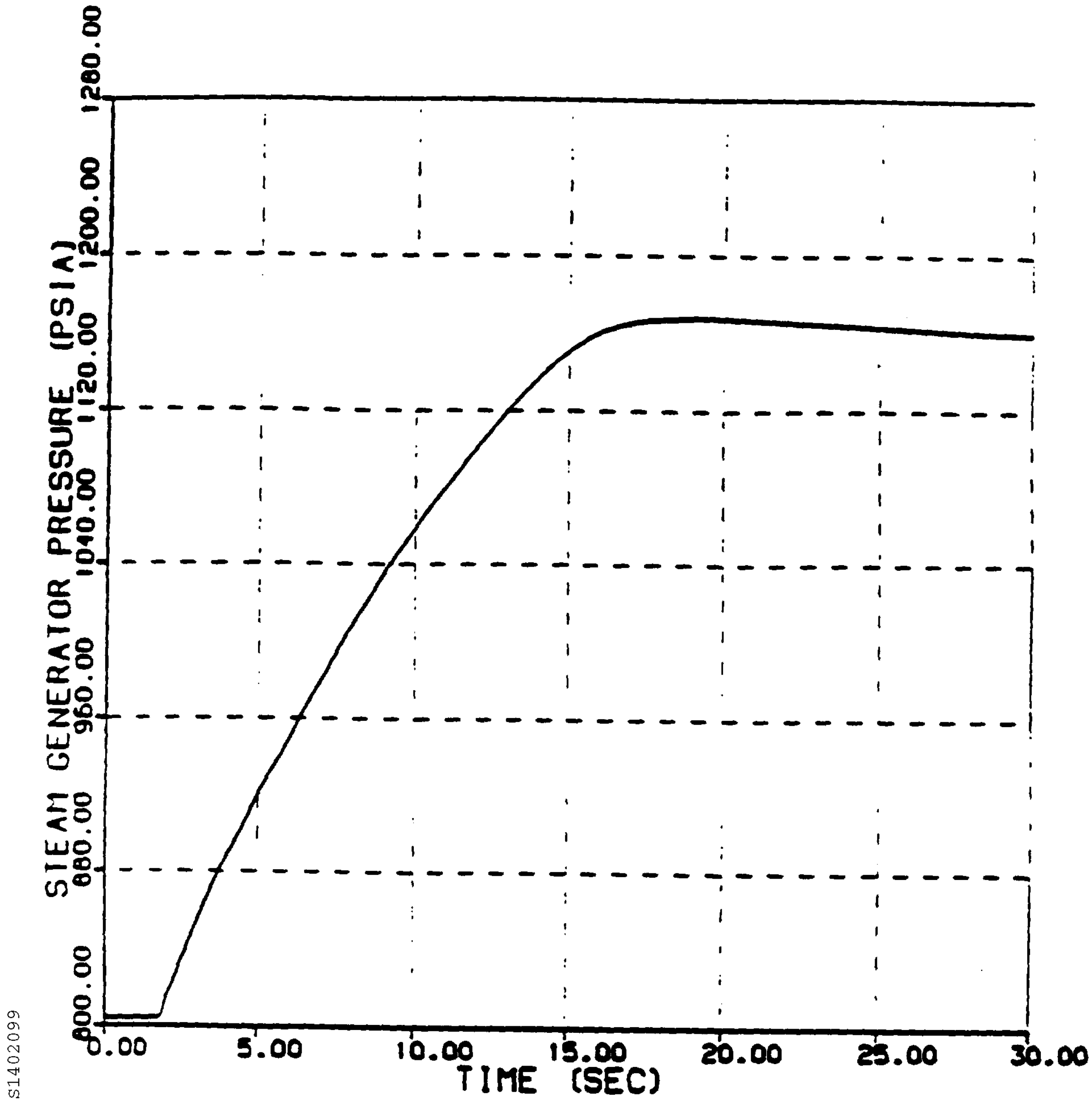


Figure 14.2-54
LOSS OF EXTERNAL LOAD - BOC WITH PRESSURIZER RELIEF & SPRAY
NUCLEAR POWER (% OF 2546 MWt)

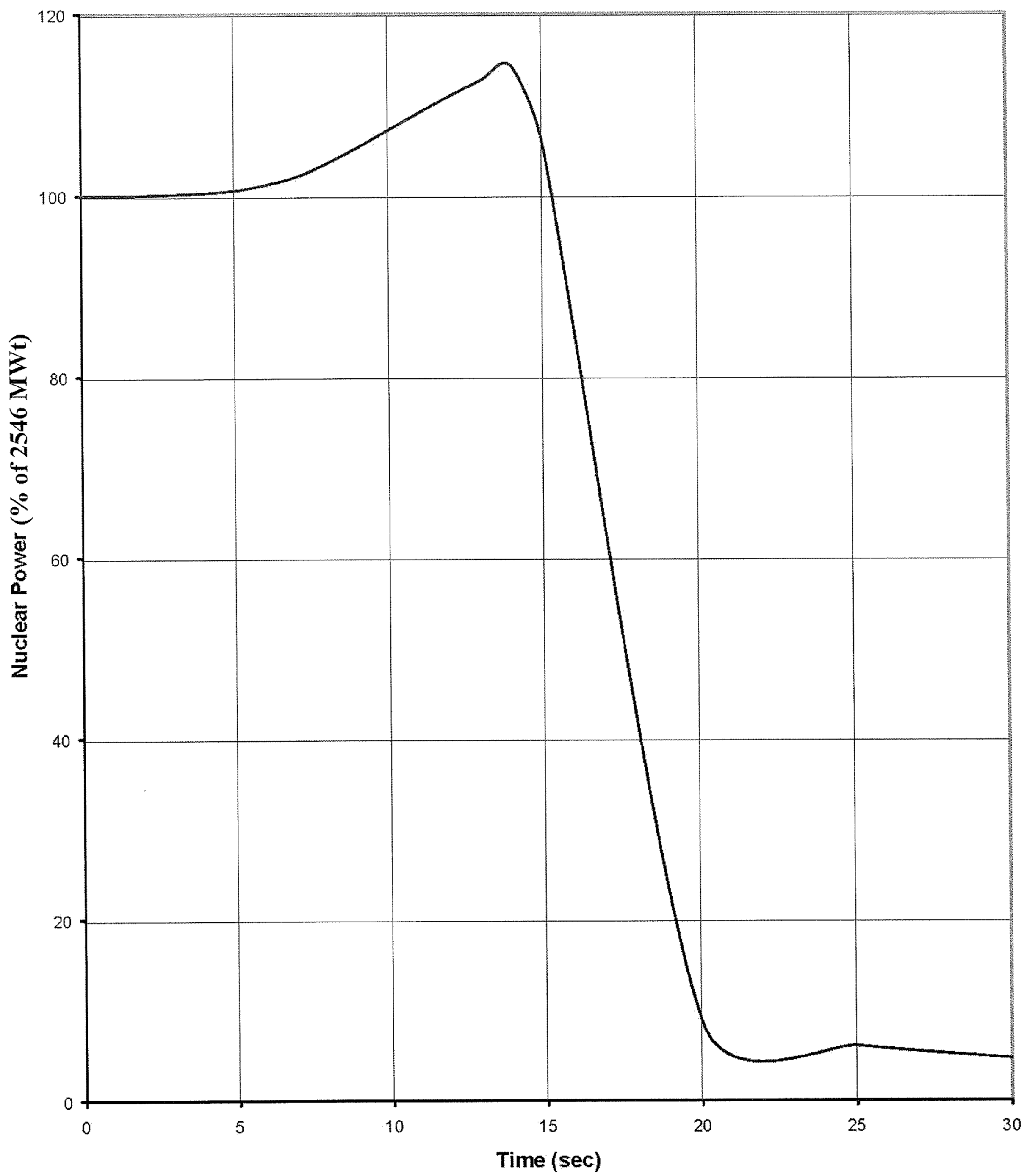


Figure 14.2-55
LOSS OF EXTERNAL LOAD - BOC WITH PRESSURIZER RELIEF & SPRAY
CORE INLET TEMPERATURE

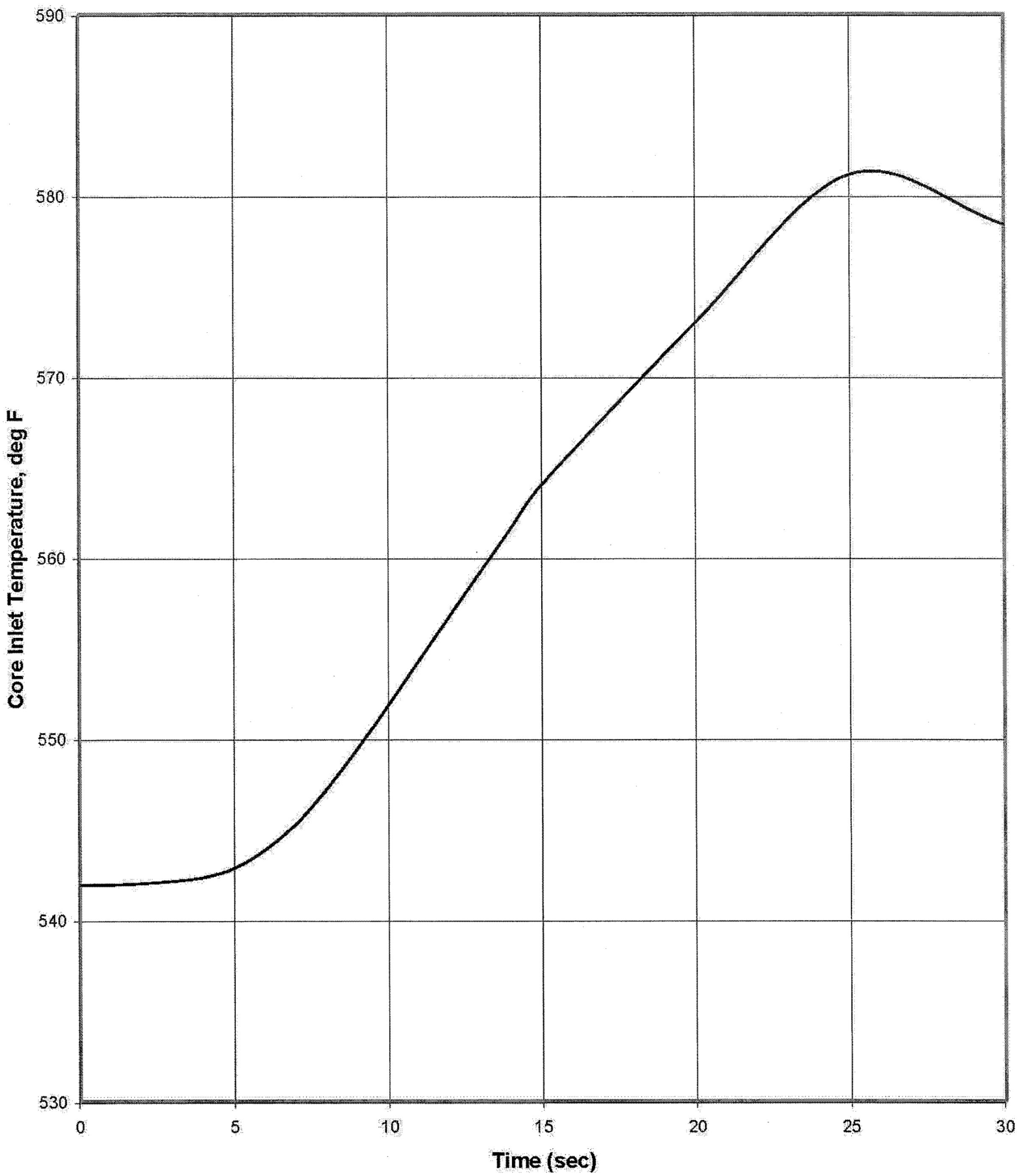


Figure 14.2-56
LOSS OF EXTERNAL LOAD - BOC WITH PRESSURIZER RELIEF & SPRAY
PRESSURIZER LIQUID VOLUME

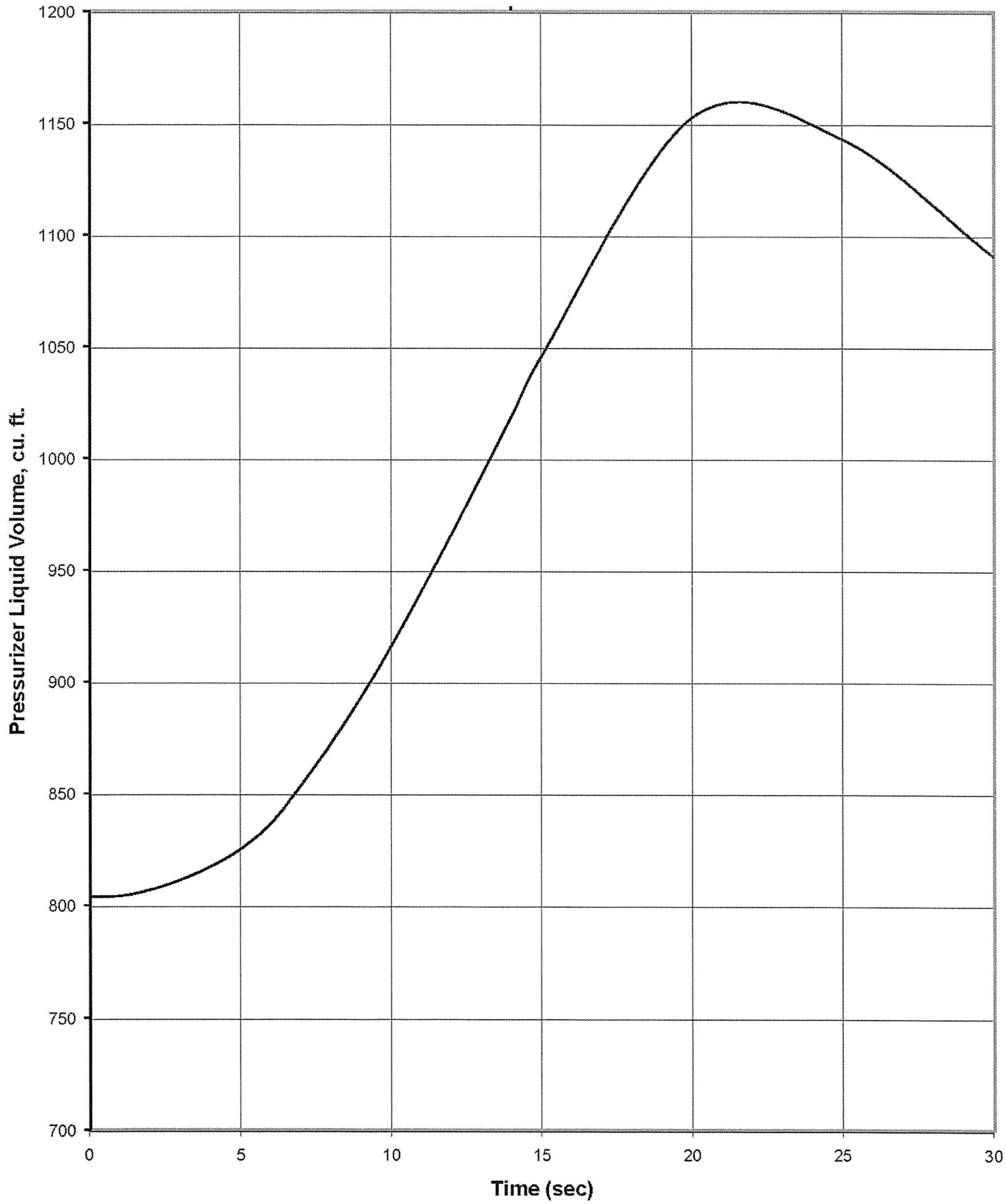


Figure 14.2-57
LOSS OF EXTERNAL LOAD - BOC WITH PRESSURIZER RELIEF & SPRAY
RCS COLD LEG PRESSURE

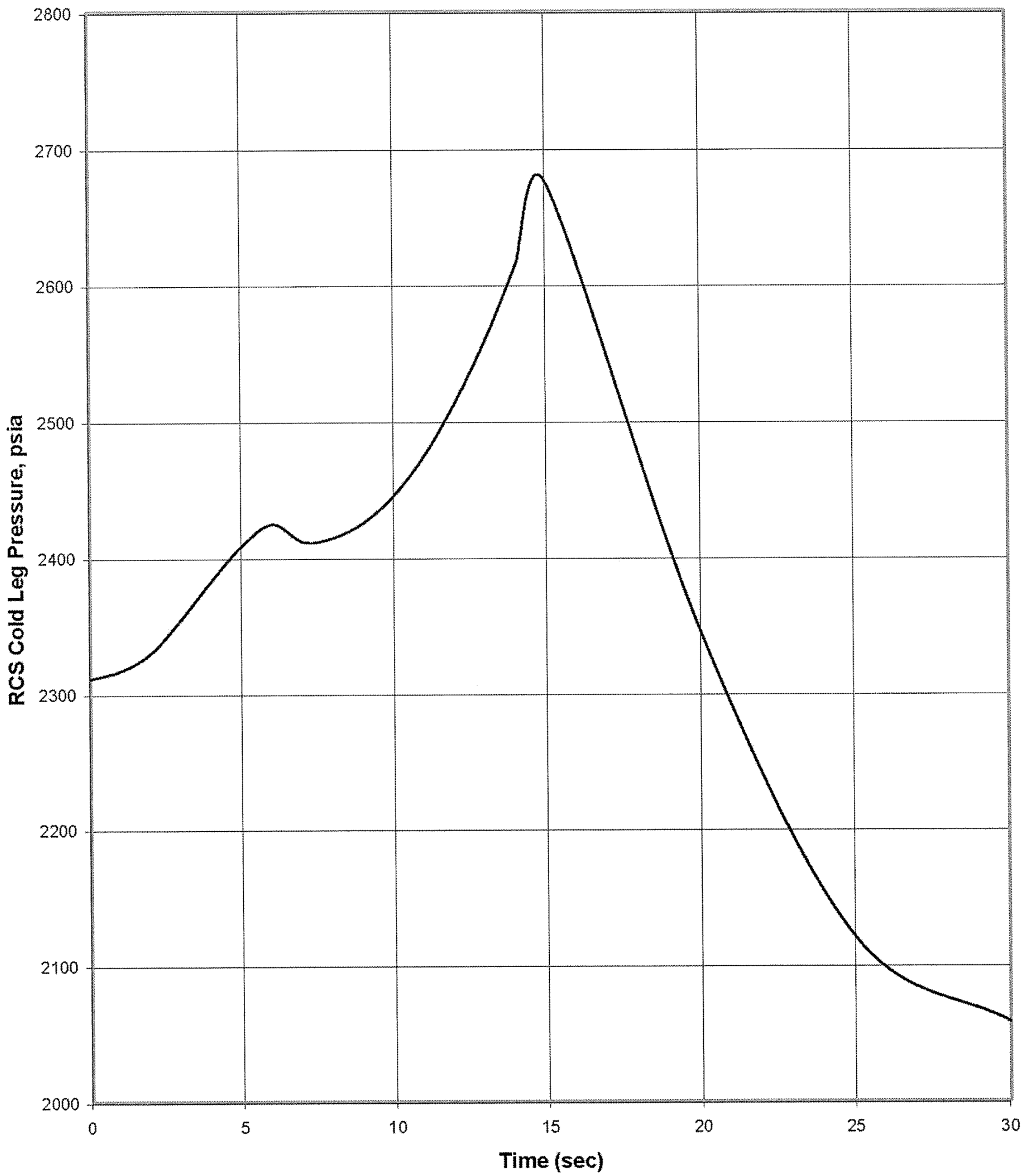


Figure 14.2-58
LOSS OF EXTERNAL LOAD - BOC WITH PRESSURIZER RELIEF & SPRAY
STEAM GENERATOR PRESSURE

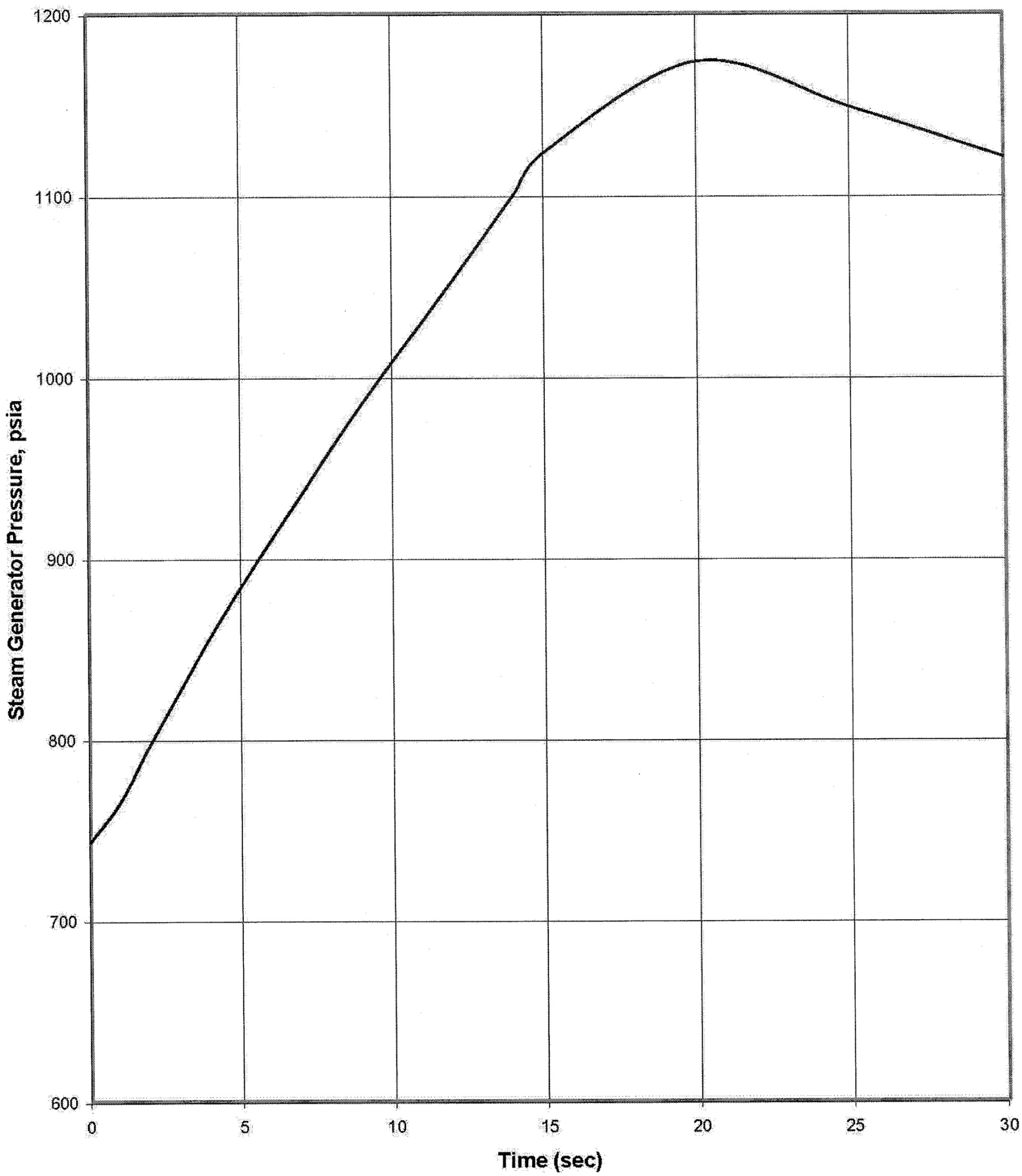


Figure 14.2-59
LOSS OF EXTERNAL LOAD - BOC WITHOUT PRESSURIZER RELIEF & SPRAY
NUCLEAR POWER (% OF 2546 MWt)

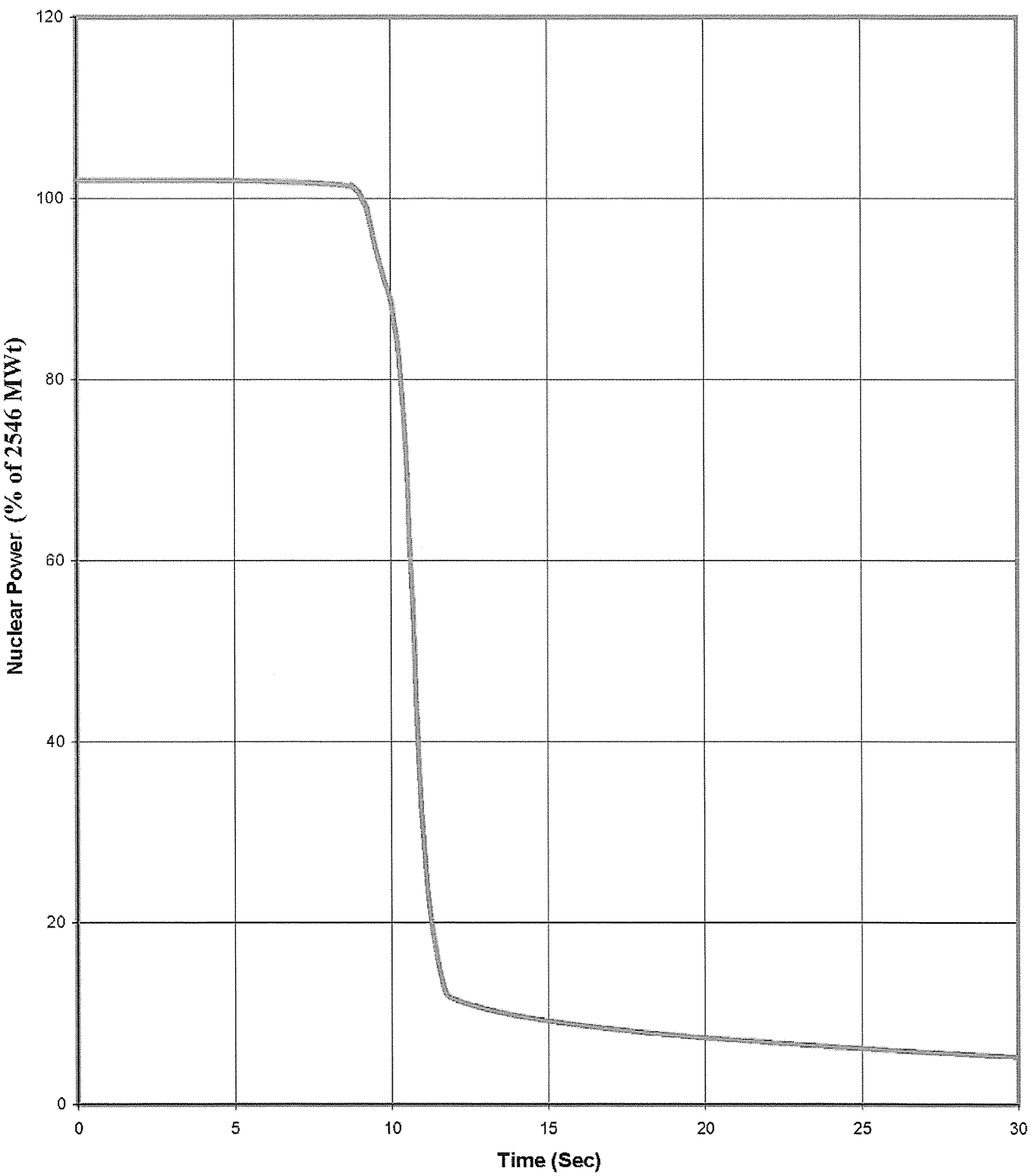


Figure 14.2-60
LOSS OF EXTERNAL LOAD - BOC WITHOUT PRESSURIZER RELIEF & SPRAY
CORE INLET TEMPERATURE

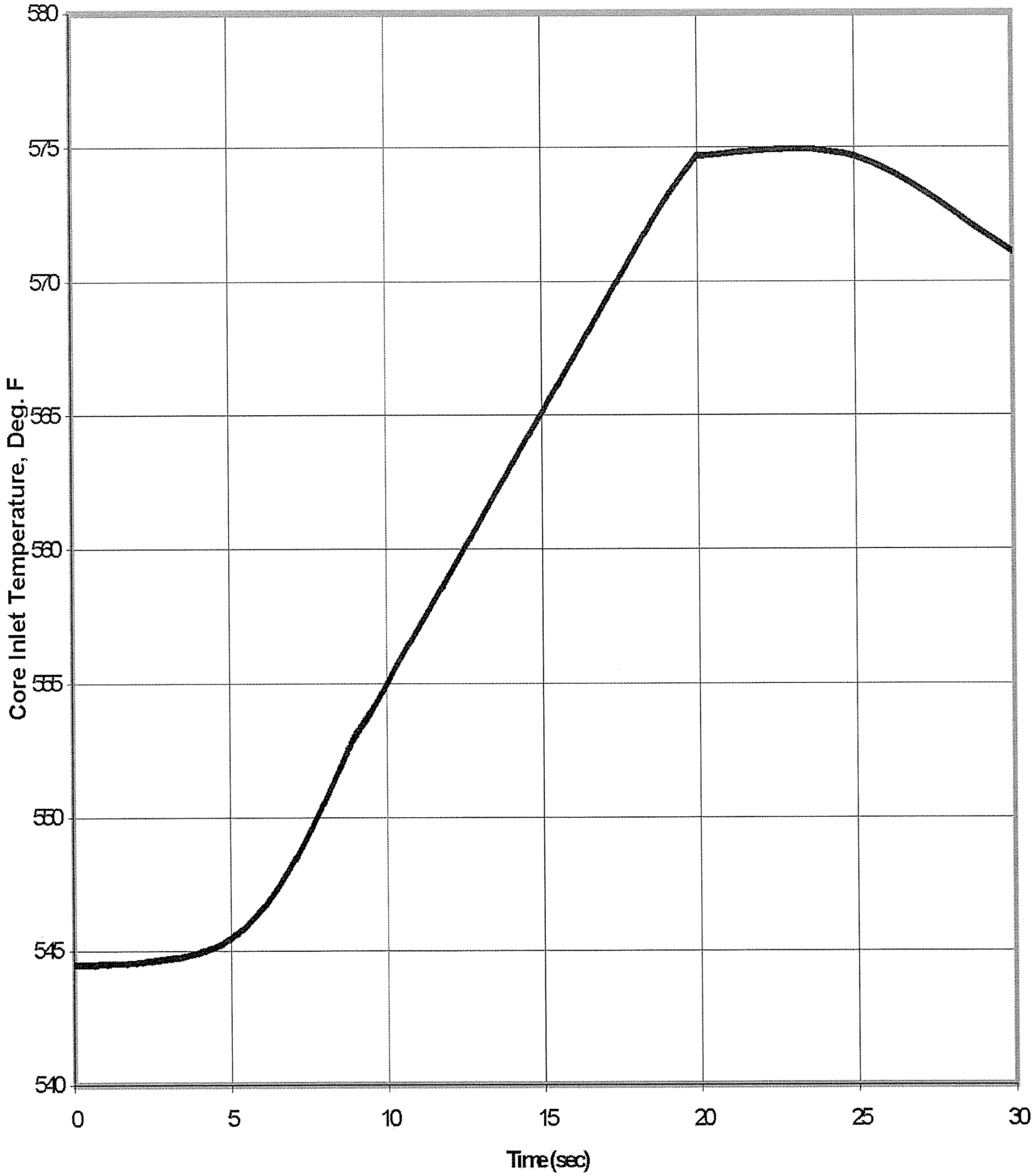


Figure 14.2-61
LOSS OF EXTERNAL LOAD - BOC WITHOUT PRESSURIZER RELIEF & SPRAY
PRESSURIZER LIQUID VOLUME

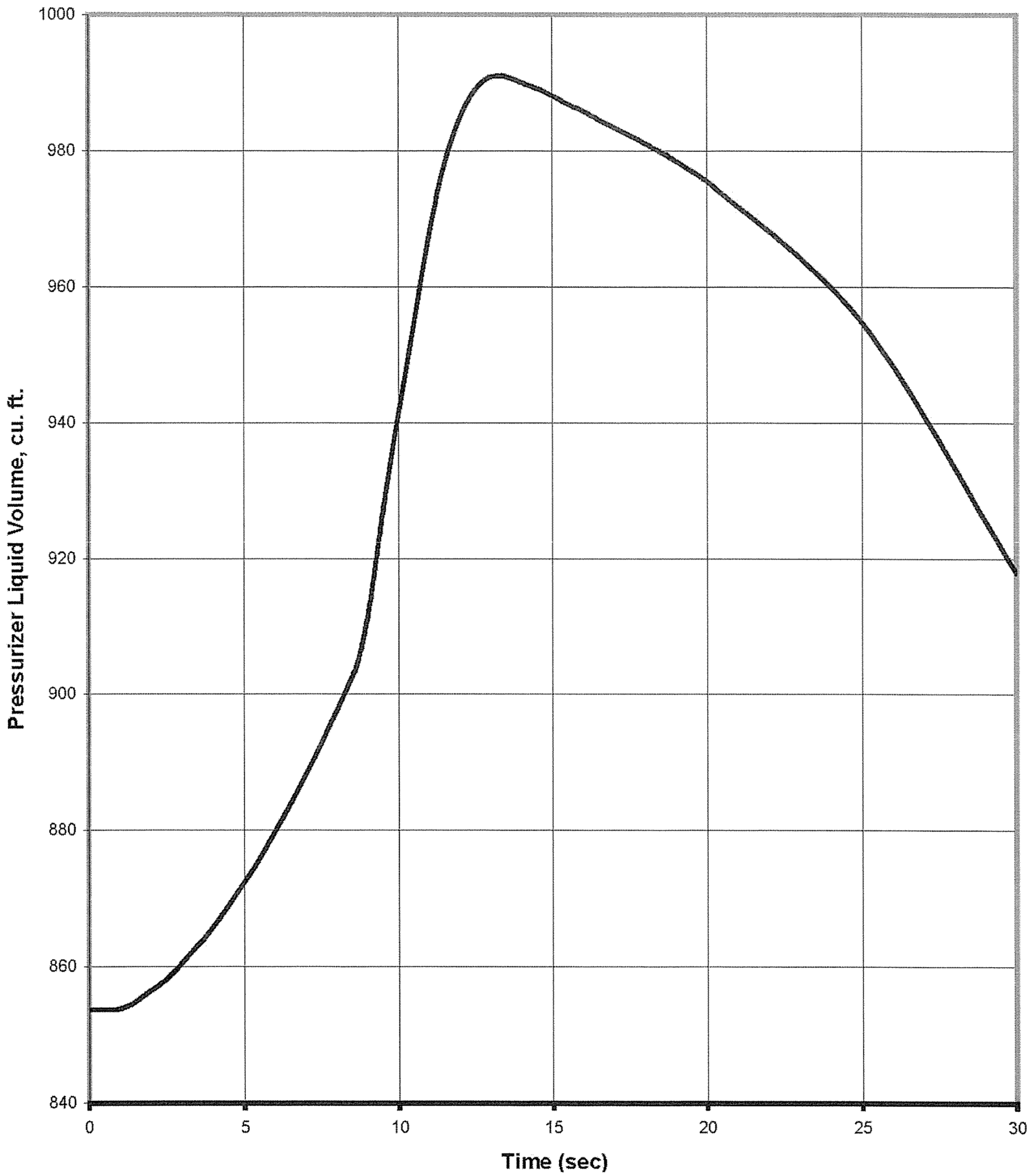


Figure 14.2-62
LOSS OF EXTERNAL LOAD - BOC WITHOUT PRESSURIZER RELIEF & SPRAY
RCS COLD LEG PRESSURE

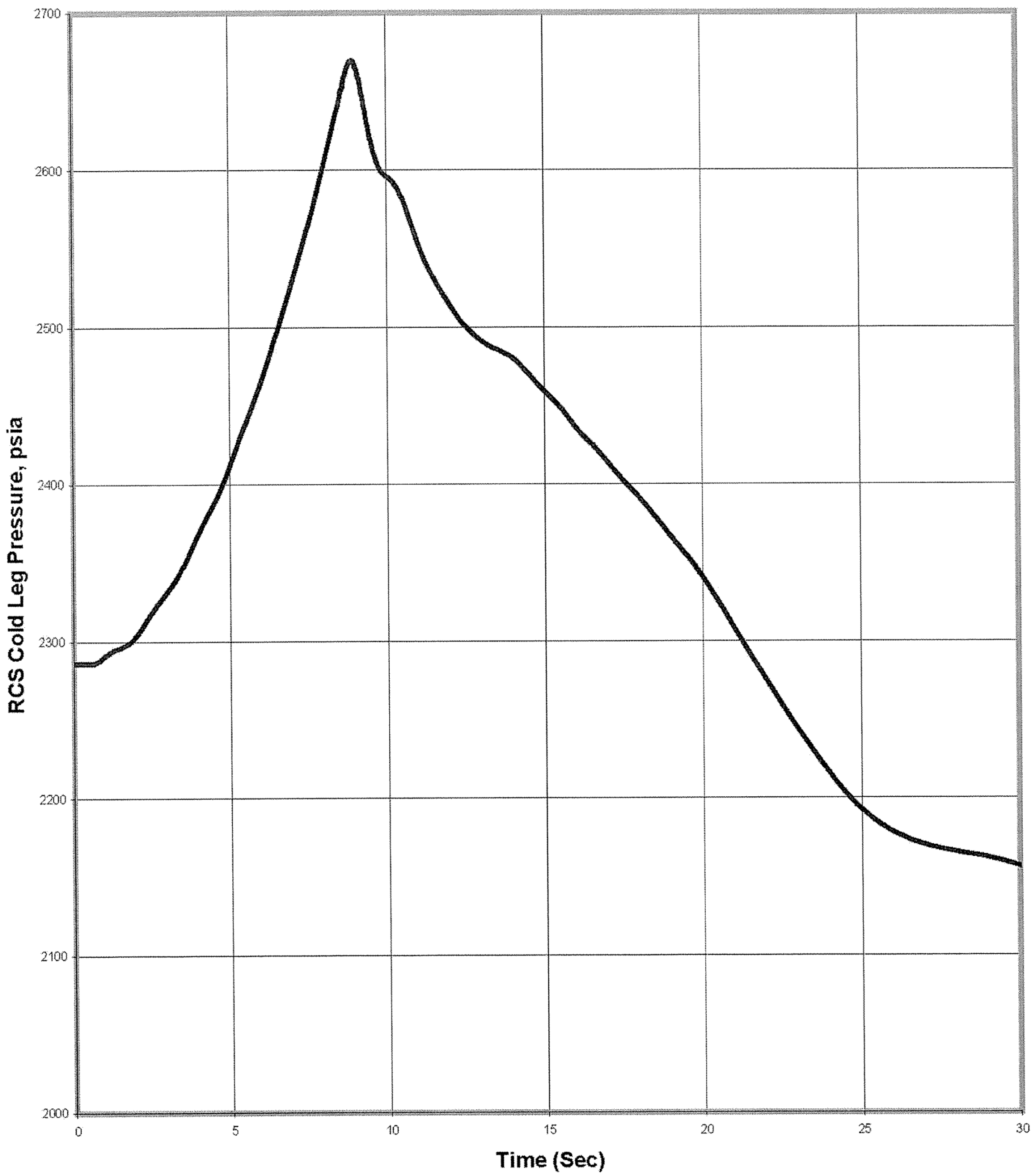


Figure 14.2-63
LOSS OF EXTERNAL LOAD - BOC WITHOUT PRESSURIZER RELIEF & SPRAY
STEAM GENERATOR PRESSURE

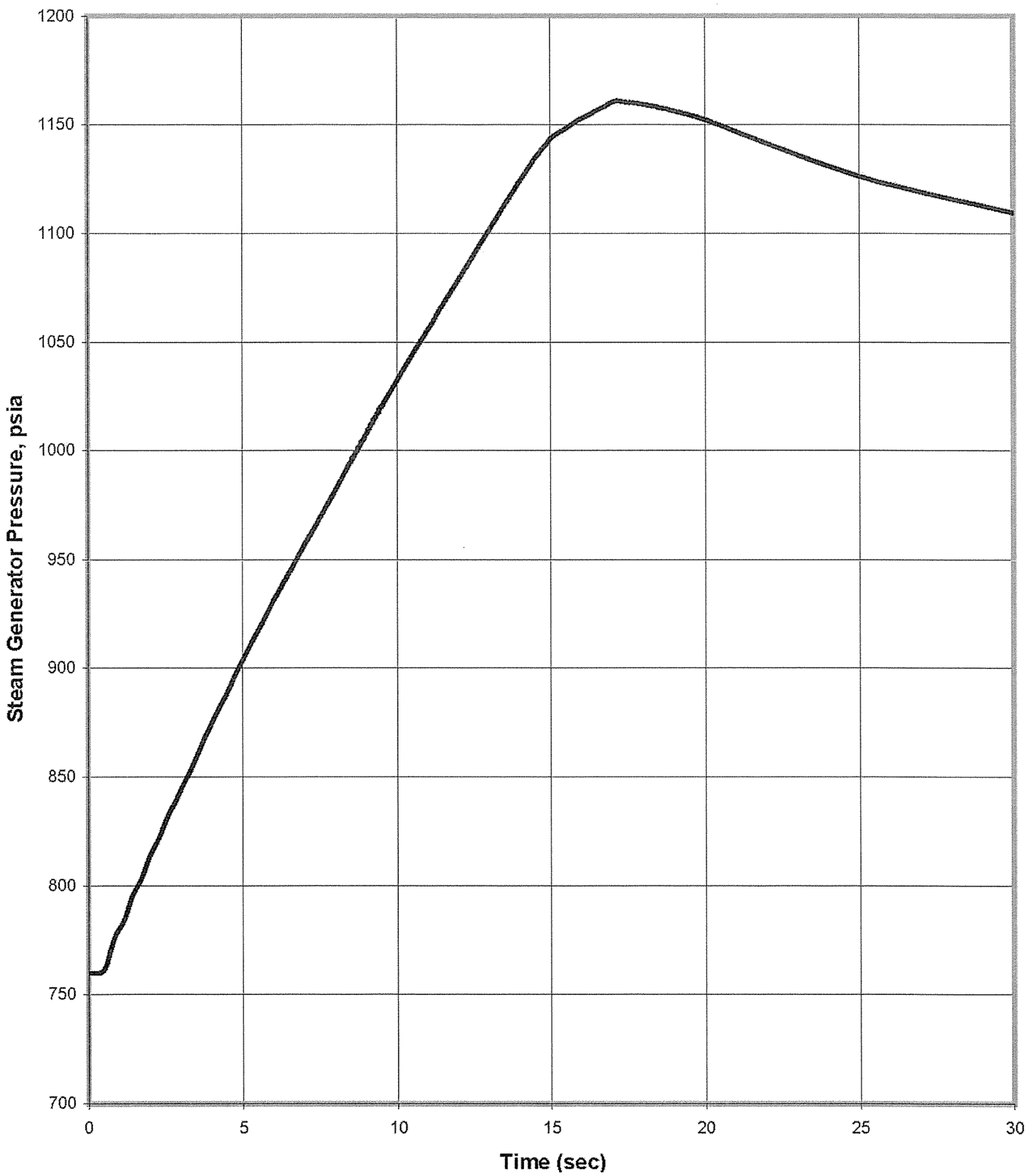


Figure 14.2-64
LOSS OF NORMAL FEEDWATER; PRESSURIZER PRESSURE
(OFFSITE POWER NOT AVAILABLE)

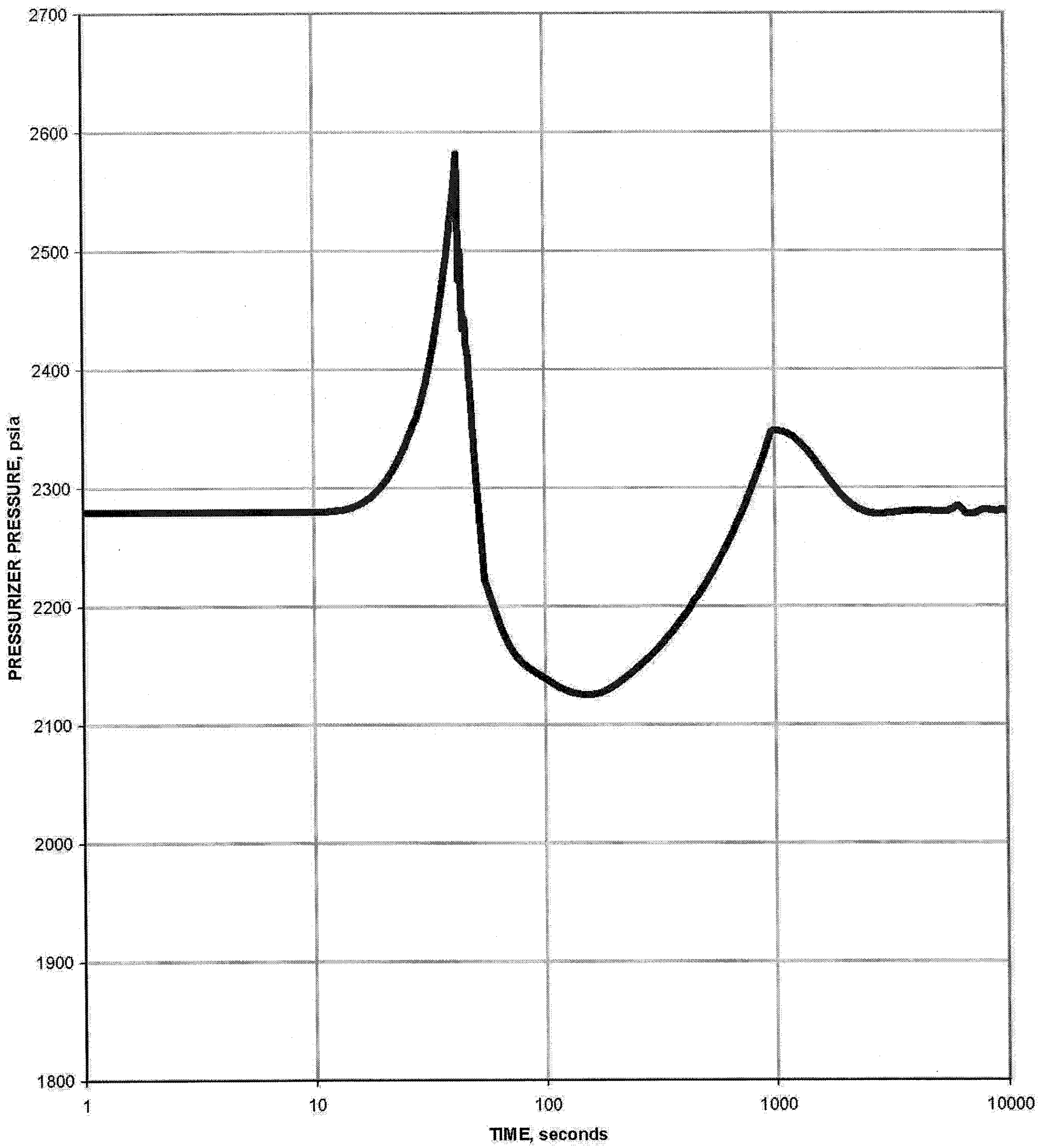


Figure 14.2-65
LOSS OF NORMAL FEEDWATER; PRESSURIZER WATER VOLUME
(OFFSITE POWER NOT AVAILABLE)

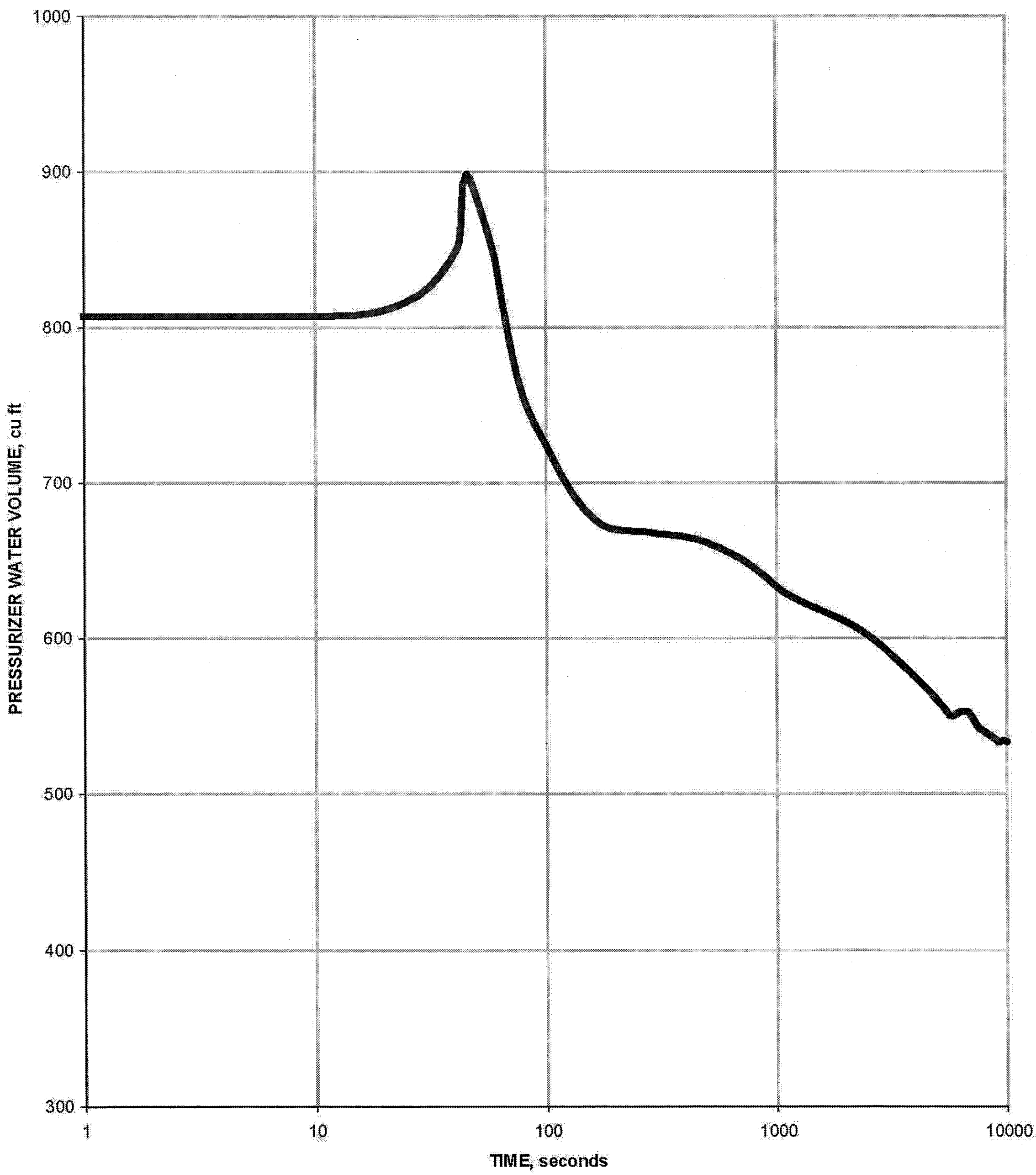


Figure 14.2-66
LOSS OF NORMAL FEEDWATER; RCS LOOP TEMPERATURE
(OFFSITE POWER NOT AVAILABLE)

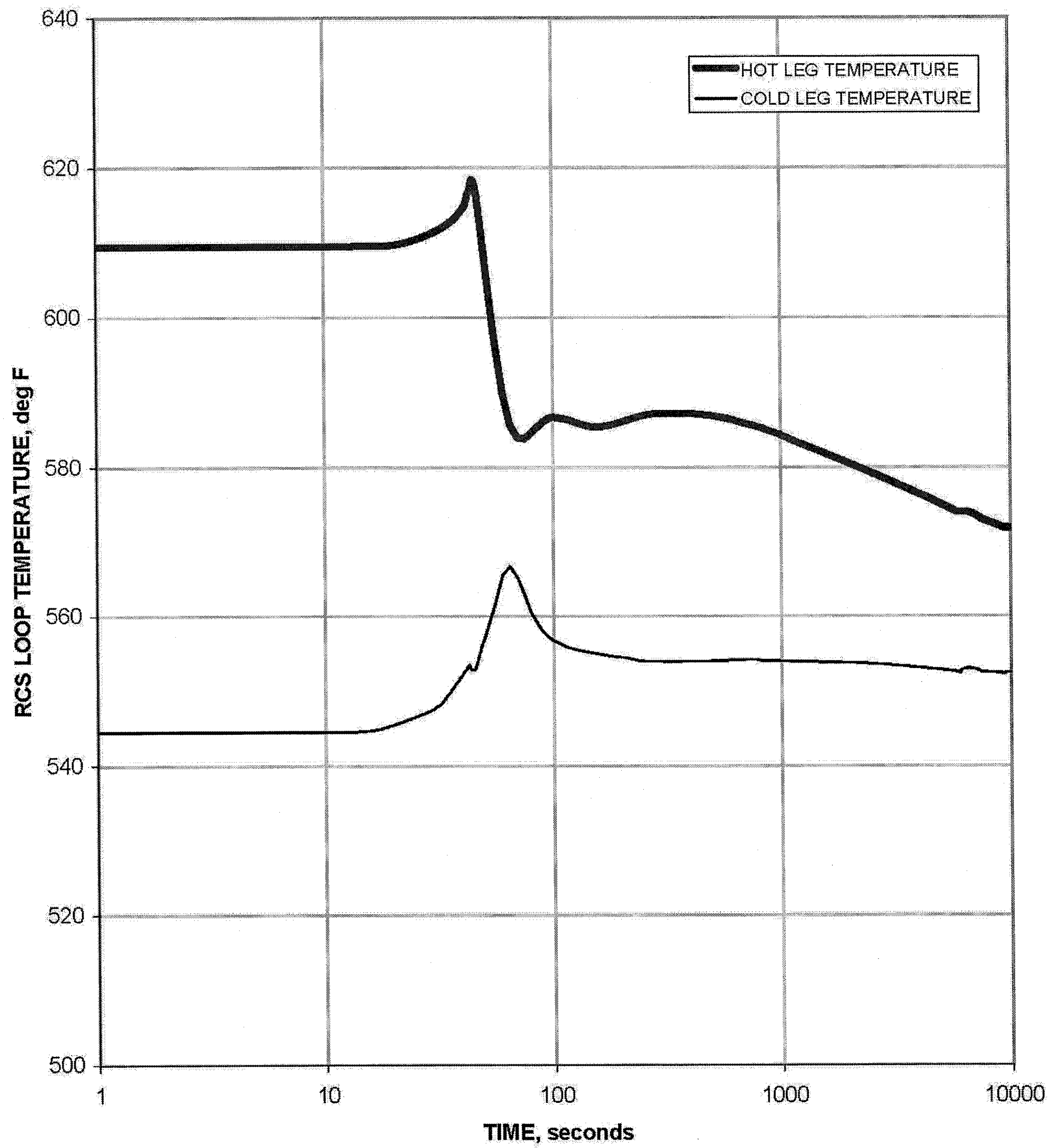


Figure 14.2-67
LOSS OF NORMAL FEEDWATER; CORE INLET FLOW
(OFFSITE POWER NOT AVAILABLE)

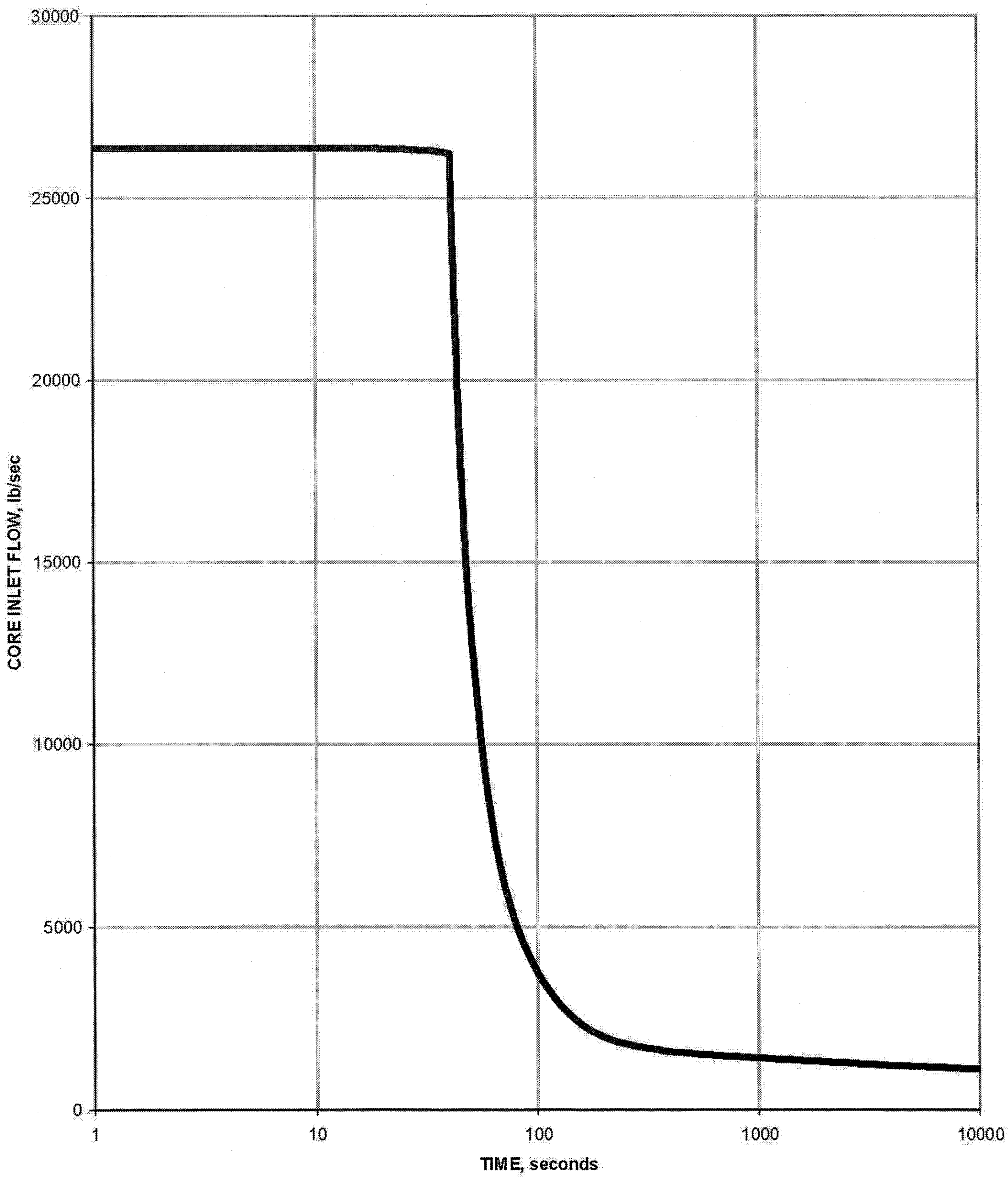


Figure 14.2-68
LOSS OF NORMAL FEEDWATER; PRESSURIZER PRESSURE
(OFFSITE POWER AVAILABLE)

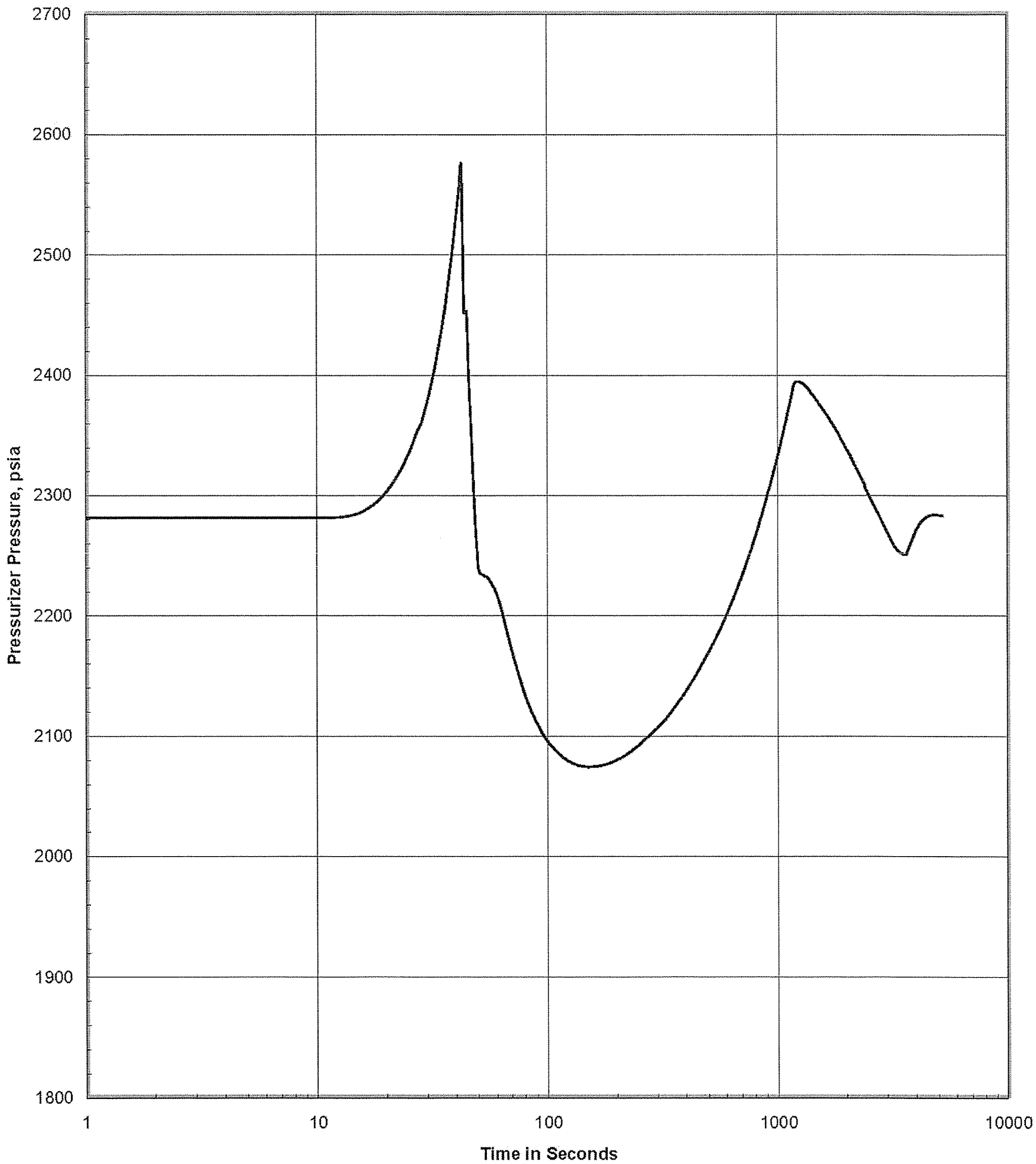


Figure 14.2-69
LOSS OF NORMAL FEEDWATER; PRESSURIZER WATER VOLUME
(OFFSITE POWER AVAILABLE)

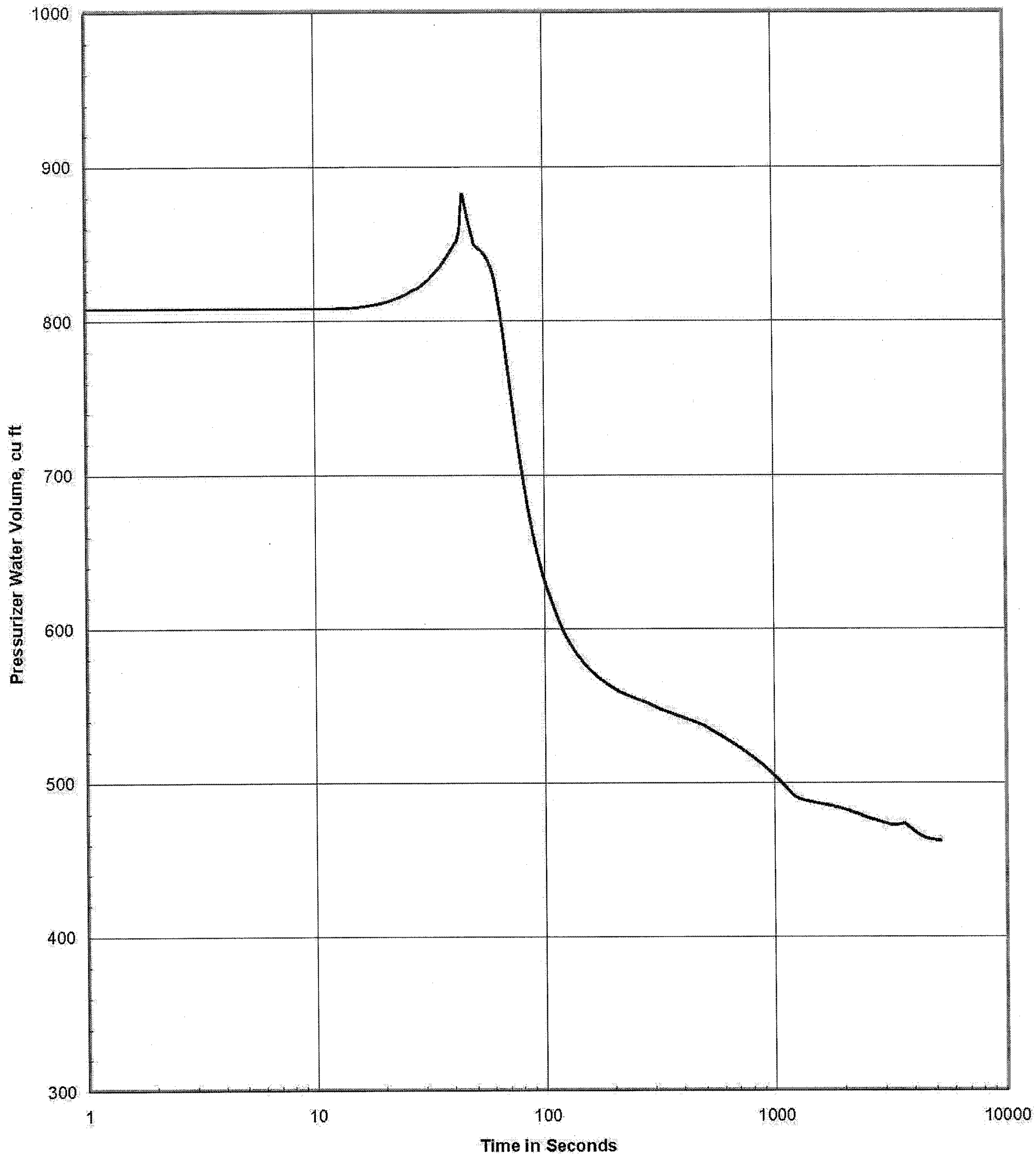


Figure 14.2-70
LOSS OF NORMAL FEEDWATER; RCS LOOP TEMPERATURE
(OFFSITE POWER AVAILABLE)

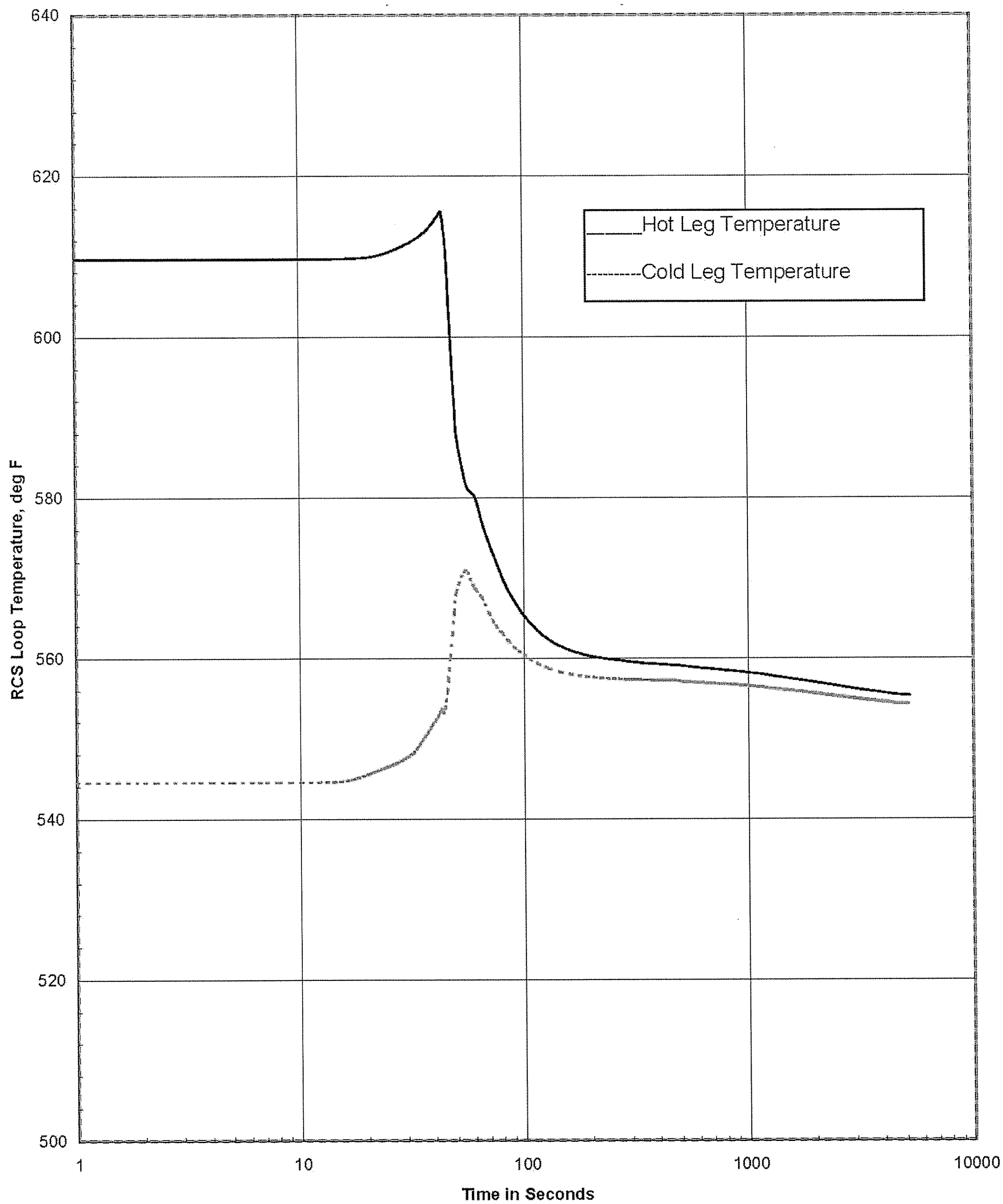


Figure 14.2-71
LOSS OF NORMAL FEEDWATER; CORE INLET FLOW
(OFFSITE POWER AVAILABLE)

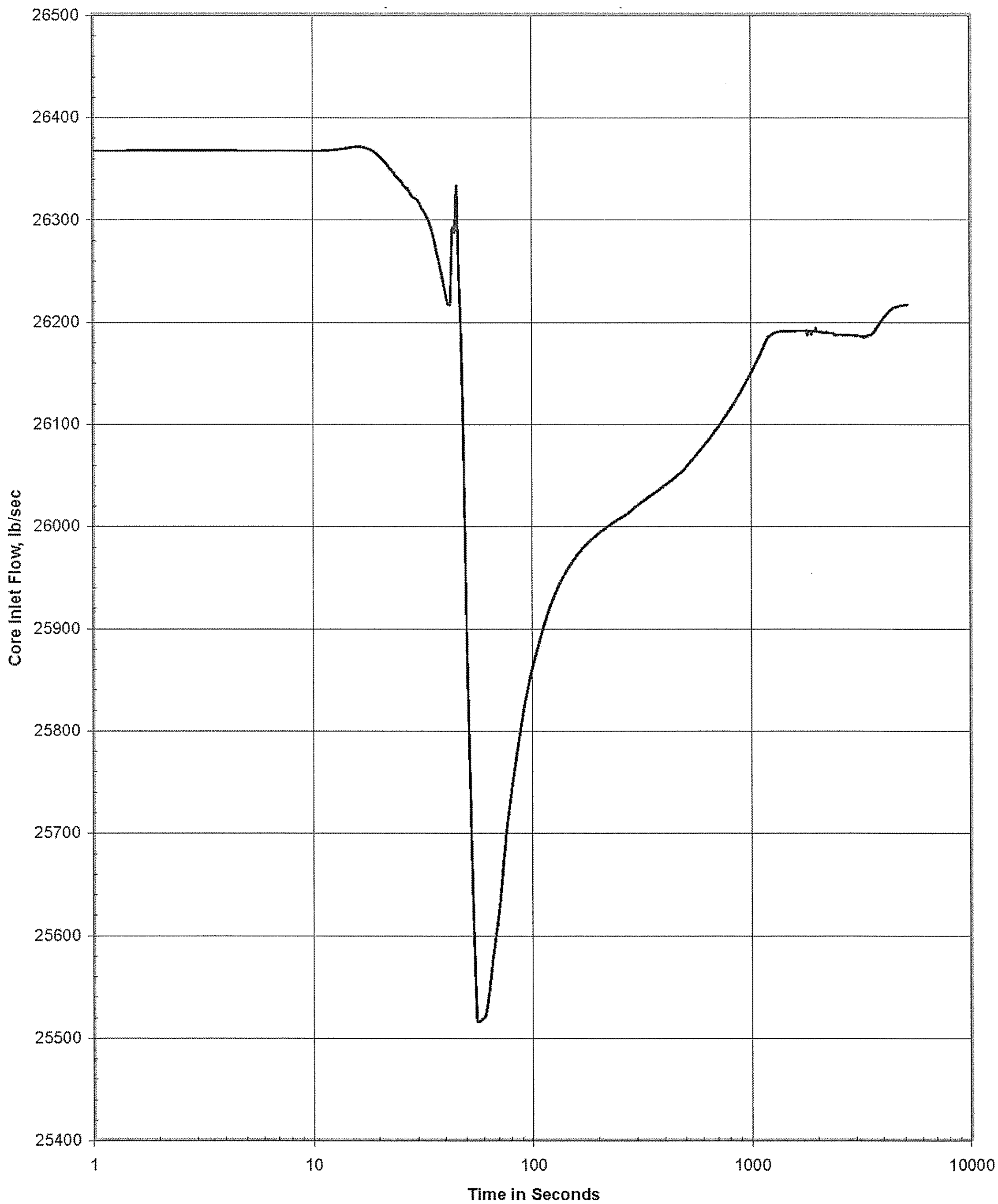


Figure 14.2-72
PROBABILITY DISTRIBUTION OF STRESS CORROSION
CRACK GROWTH RATE

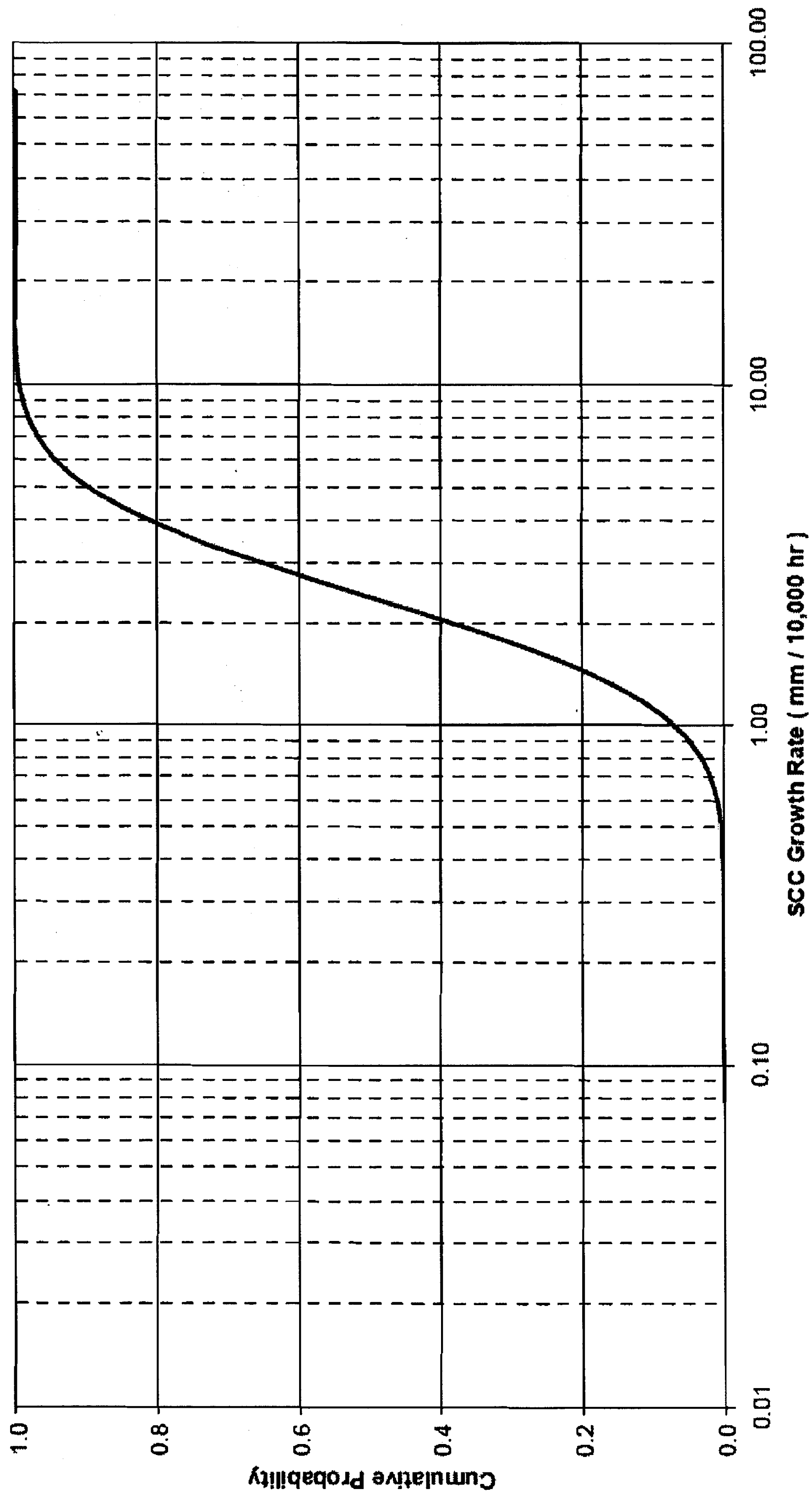


Figure 14.2-73
PROBABILITY DISTRIBUTION OF CRACK SHAPE FACTOR G

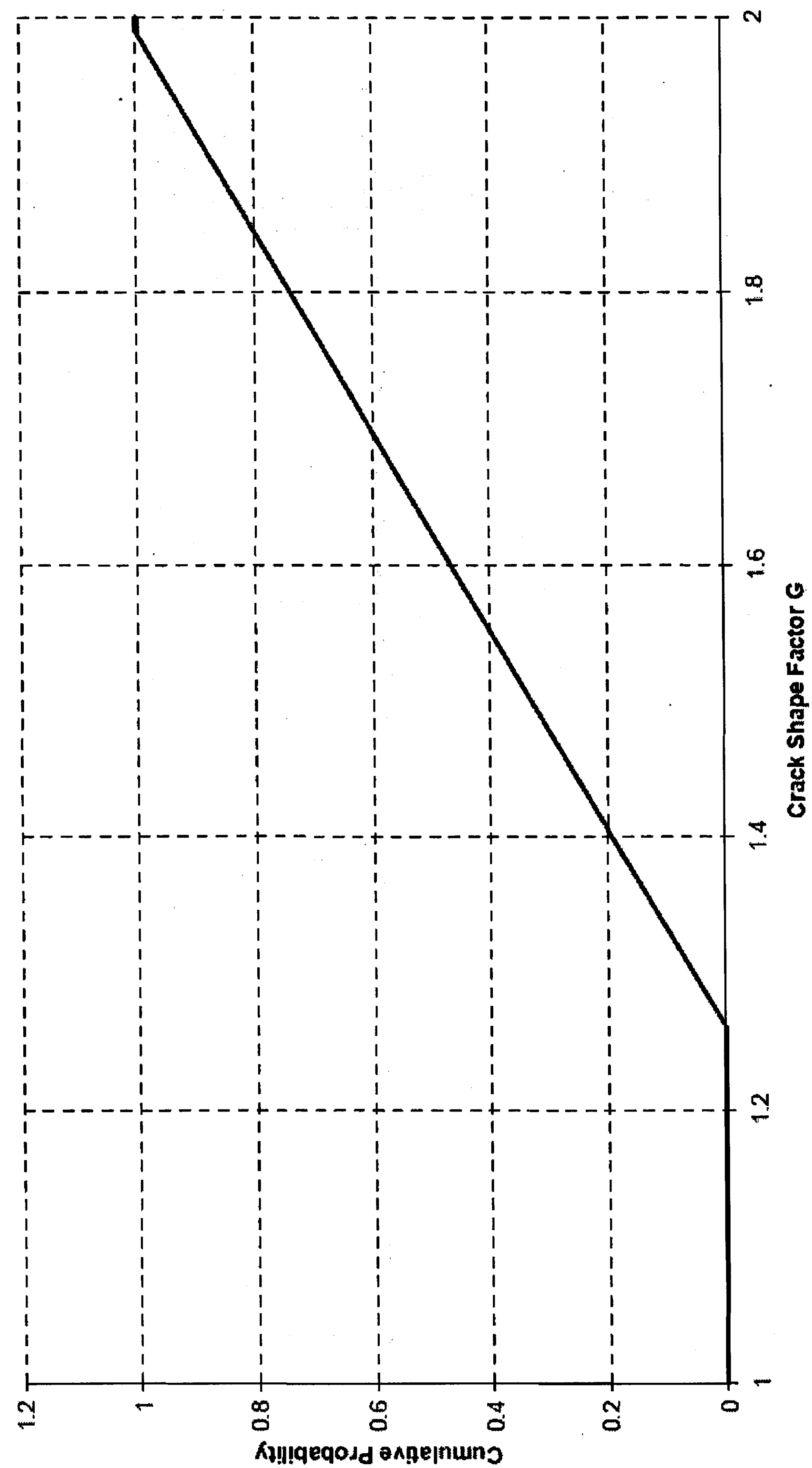
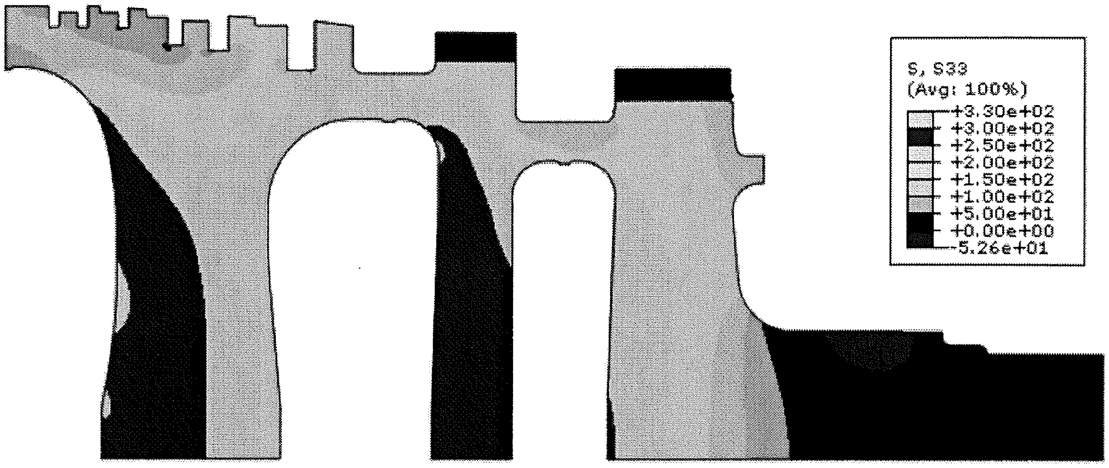
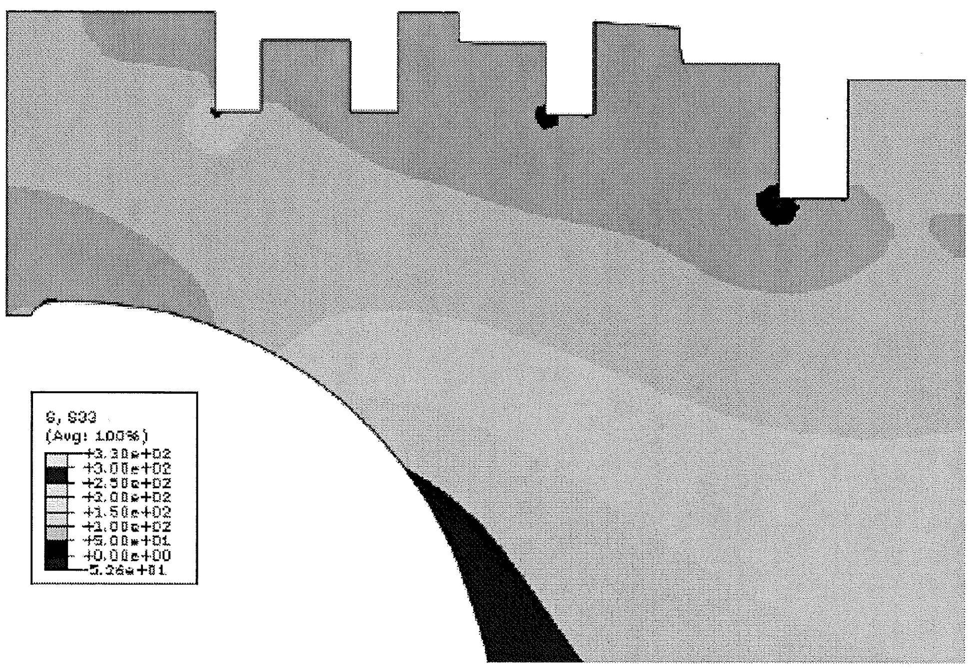


Figure 14.2-74
LP ROTOR TANGENTIAL STRESS CONTOURS



Tangential stress contours (MPa) in steady operation



Tangential stress contours (MPa) at outside of centre disc

Figure 14.2-75
VARIATION OF CRITICAL CRACK DEPTH WITH CRACK SHAPE FACTOR G

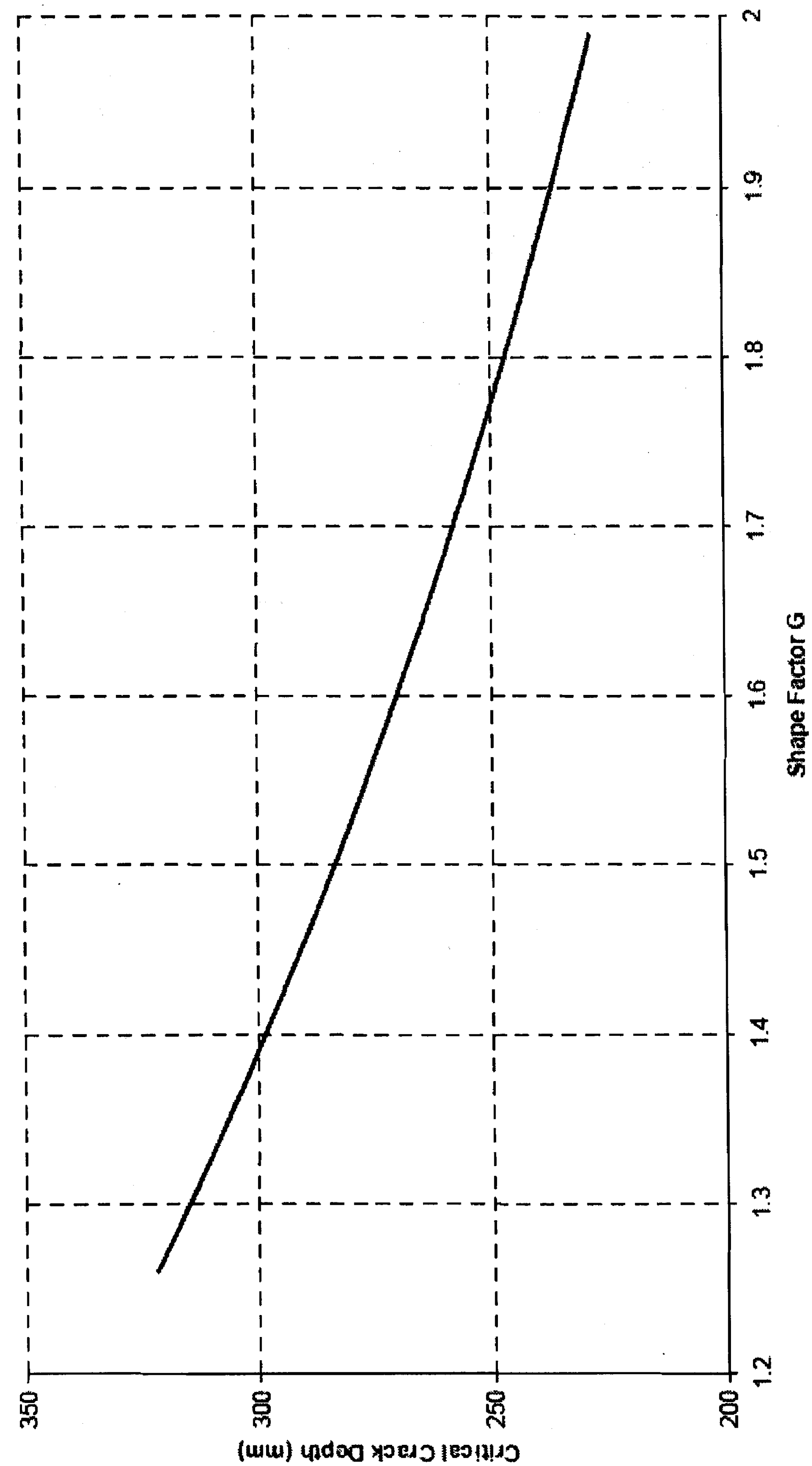
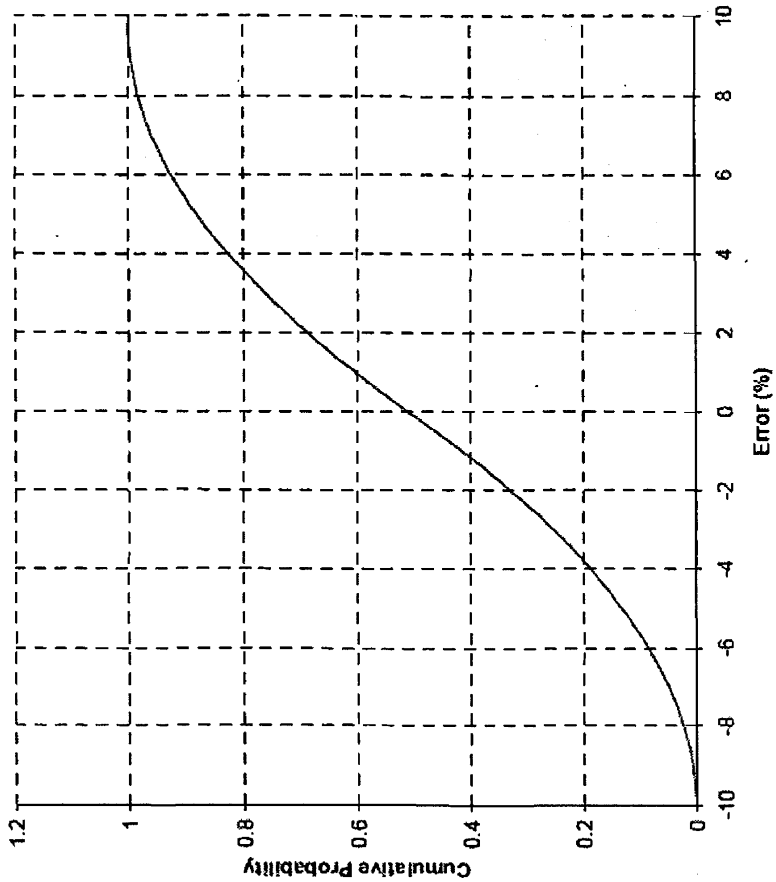
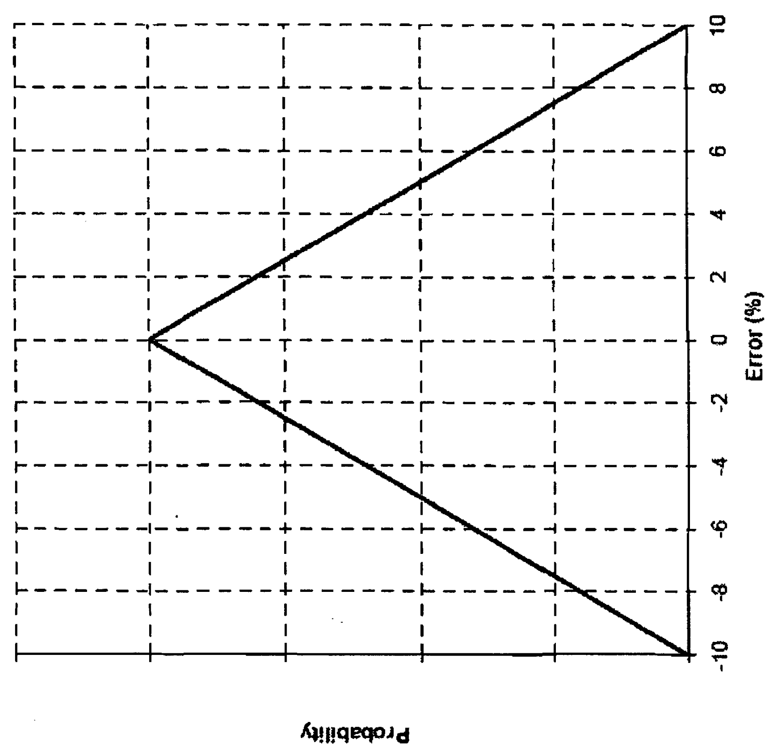


Figure 14.2-76
PROBABILITY DISTRIBUTION OF CALCULATED STRESSES





14.3 STANDBY SAFEGUARDS ANALYSIS

14.3.1 Steam Generator Tube Rupture

14.3.1.1 Identification of Causes and Description of Accident

The accident examined is the complete severance of single steam generator tube near the top of the tube bundle. It is assumed that the accident takes place at full power and while the reactor coolant is contaminated with the maximum concentrations allowable by the Technical Specifications, including the effects of pre-accident iodine spiking. The accident leads to the contamination of the secondary side due to the leakage of radioactive coolant from the Reactor Coolant System and, in the event of a coincidental loss of offsite power, a discharge of activity to the atmosphere through the steam generator safety valve and/or power operated relief valves. The analysis presented here conservatively assumes a single power operated relief valve on the header closest to the ruptured generator sticks open following reactor trip.

The steam generator tube material is Inconel 600, and, as the material is highly ductile, it is considered that the complete severance of a tube is extremely conservative. Surry Unit 1 and 2 steam generator tube bundles were replaced in 1979 and 1980, respectively, and operating experience since then has been extremely good.

The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the Steam and Power Conversion System is subject to continual surveillance, and primary to secondary leakage during unit operation is limited to a value that is much lower than that associated with a full tube rupture. For RCS leaks in excess of 50 gpm with indications of secondary activity, the reactor is tripped, and an abnormal procedure for large steam generator tube leak is implemented which directs the operator to (a) identify and isolate secondary release paths from the ruptured generator, (SG PORVs, steam flow to the turbine driven AFW pump, etc.), (b) align the condenser air ejector discharge to containment and (c) commence unit cooldown.

Once the operator has determined a tube rupture has occurred, his first priority is to identify and isolate the affected steam generator as soon as possible in order to minimize the contamination of the secondary system and ensure the termination of any activity discharge to the atmosphere. The recovery procedure can be carried out on such a time scale as to ensure that break flow to the secondary system is terminated before the water level in the affected steam generator can rise into the main steam pipe. Sufficient indications and controls are provided to enable the operator to perform these functions satisfactorily. Training on the tube rupture accident is a significant emphasis of the licensed operator requalification program.

The following sequence of events is initiated by a tube rupture:

1. Pressurizer low pressure and low-level alarms are actuated, and, before unit trip, charging pump flow increases in an attempt to maintain the pressurizer level. On the secondary side there is a steam flow-feedwater flow mismatch before the trip as feedwater flow to the affected steam generator is reduced owing to the additional break flow that is now being supplied to that steam generator.
2. The loss of reactor coolant inventory leads to falling pressure and level in the pressurizer, and eventually a reactor trip signal is generated by overtemperature ΔT or low pressurizer pressure. Automatic unit cooldown following a reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates the normal feedwater supply and initiates the addition of auxiliary feedwater.
3. The steam generator blowdown liquid monitor and the air ejector radiation monitor alarm, indicating the passage of reactor coolant into the secondary system. The air ejector radiation monitor high alarm causes the air ejector exhaust from the condenser to be discharged to the containment, thereby terminating any direct atmospheric release.
4. The unit trip automatically shuts off the steam supply to the turbine and, if outside power is available, the condenser bypass valves open to permit steam dump to the condenser. In the event of a coincidental station blackout, and loss of condenser vacuum, the condenser bypass valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase and discharge steam to the atmosphere through the steam generator safety valves and/or power-operated relief valves.
5. Following a unit trip, the continued auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink that eventually absorbs decay heat. Thus, steam bypass to the condenser, or, in the case of the loss of condenser vacuum, steam relief to the atmosphere, is discontinued on a time scale that is dependent on the exact amount of emergency equipment (safety injection pumps and auxiliary feedwater pumps) operating.
6. Safety injection flow results in an increasing pressurizer water level. The time after trip at which the operator can clearly see the returning level in the pressurizer is also dependent on the amount of operating auxiliary equipment.

14.3.1.2 **Method of Analysis and Description of the Accident**

A. Thermal-Hydraulic Analysis (Loss of Offsite Power Case)

The thermal hydraulic portion of the tube rupture accident is simulated with the Virginia Power RETRAN model (Reference 12). Key analysis assumptions were as follows:

1) A double ended tube rupture was modeled. Break flow was calculated by explicitly modeling friction losses in both segments of the ruptured steam generator tube and unchoked flow at the rupture site. This model overpredicts the actual break flows observed in the 1987 North Anna Unit 1 steam generator tube rupture. The resultant decrease in RCS pressure eventually reduces the overtemperature ΔT trip setpoint to the full power value resulting in a reactor and turbine trip.

2) Following reactor trip on overtemperature ΔT, the condenser dumps are assumed unavailable, and the secondary side pressurizes to steam generator power operated relief valve (PORV) setpoint following turbine trip.

3) The PORV on the main steam header nearest the ruptured generator is assumed to remain fully open from the time of PORV actuation until 30 minutes after event initiation. Thus, atmospheric releases are assumed over this interval. For the normal case of condensers available, a high air ejector radiation signal diverts the air ejector exhaust to containment. Following safety injection, this exhaust path is also isolated. When offsite power and the condenser are available, the volatile species undergo two stages of partitioning (i.e., in the steam generator and the condenser) prior to being released to the atmosphere. Loss of offsite power results in loss of the condenser and in coastdown of the reactor coolant pumps, which increases the break fluid flashing fraction. Flashed break flow is a major contributor to the release of radioisotopes, as discussed in Section 14.3.1.4. Thus, the case of loss of offsite power is the limiting case from the standpoint of site boundary dose, and the analysis for this case assumes loss of the condenser and coastdown of the reactor coolant pumps after reactor trip.

4) After reactor and turbine trip, the Reactor Coolant System continues to depressurize to the safety injection setpoint. Two high head safety injection pumps are assumed to operate. The Reactor Coolant System pressure stabilizes at the point where break flow and safety injection flow are essentially equal.

5) Reactor power level used in this analysis was set at 102% of 2546 MWt or 100.38% of 2587 MWt.

B. Thermal-Hydraulic Analysis—Control Room Dose (Offsite Power Available) Case

The thermal/hydraulic analysis performed to support the calculation of control room dose is similar to that presented above. However, the calculated control room dose is more limiting under the assumption of offsite power continuing to remain available. Continued availability of offsite power would result in a potentially larger forced intake of unfiltered

air from the normal control room air inlets prior to control room isolation than the case of concurrent loss of offsite power.

Therefore, the thermal/hydraulic analysis used to develop the control room calculation assumes continued operation of the reactor coolant pumps after reactor trip. However, no credit is taken for operation of the condenser dumps. As with the previous case, releases are assumed to be via a stuck open PORV on the main steam header leading from the ruptured generator.

14.3.1.3 **Results**

The thermal hydraulic results are shown in Figures 14.3-1 through 14.3-7 for the loss of offsite power case:

Figure 14.3-1—RCS Average Temperature: Following the rupture, RCS temperature is relatively stable until the unit trips on overtemperature ΔT at 74 seconds. The turbine stop valves are assumed to close within the next 2 seconds. Temperature continues to decrease in response to addition of cold safety injection water (safety injection occurs in response to low pressurizer pressure at 225 seconds) and the release of steam through the stuck open PORV (the PORV opens at 80 seconds). In actual operating practice, additional cooldown would be imposed by the operators as directed by the emergency procedures to support primary side depressurization to reduce the break flow (assuming the affected SG’s PORV was manually isolated).

Figure 14.3-2—Reactor Power: As discussed above, reactor trip is on overtemperature ΔT at 74 seconds. This figure shows a return to power since the boron injection coincident with safety injection was not modeled in the RETRAN analysis.

Figure 14.3-3—Ruptured Loop Steam Pressure: After the reactor and turbine trip, pressure in the steam generator initially increases. The expected response would be an increase followed by stabilization at the no-load pressure of about 1005 psig, but since the analysis assumes a steam generator PORV sticks open, there is a gradual depressurization. Flow through the stuck open PORV not shown in this figure follows the same trend as the ruptured loop steam pressure in Figure 14.3-3. The flow through the stuck open PORV represents the primary potential source of radioactivity transport to the environment. Figure 14.3-5 shows the integrated mass flow rate out of the faulted SG and the intact SG through the stuck open PORV and MSSV.

Figure 14.3-4—Pressurizer Pressure: The initial drop in pressurizer pressure results from excess of tube rupture flow over the charging flow. The pressurizer level controller, which would increase charging flow and tend to retard this initial depressurization, is not modeled. Immediately following reactor trip, the depressurization rate is accelerated. Safety injection is initiated on low pressurizer pressure, the depressurization drops significantly as a result.

Figure 14.3-5 – Integrated Mass Flow Rate: This figure shows the integrated mass flow rate out of the faulted SG and intact SG through the open PORV and MSSV. This represents the primary potential source of radioactivity transport to the environment.

Figure 14.3-6—Break Flow: The initial break flow through the two ends of the ruptured steam generator tube is about 90 lbm/sec. The flow drops off quickly in response to the RCS depressurization until safety injection is initiated. Then the flow stabilizes, as equilibrium between the break flow and safety injection is established, at about 65 lbm/sec. The slight increasing trend in mass flow beyond this point is a result of increased fluid density due to the RCS cooldown.

Figure 14.3-7—Integrated Break Flow: At one-half hour after initiation of the event, approximately 122,000 lbm of fluid has been transferred from the RCS to the secondary side of the ruptured steam generator.

14.3.1.4 **Environmental Consequences of Steam Generator Tube Rupture (SGTR)**

A steam generator tube rupture (SGTR) is a break in a tube carrying primary coolant through the steam generator. This postulated break allows primary liquid to leak to the secondary side of the steam generator with an assumed release to the environment through the steam generator Power Operated Relief Valves (PORVs) or the steam generator safety valves. Steam is assumed to be discharged from the affected generator to the environment until the generator is isolated at 30 minutes. The SGTR analysis used the alternative source term and followed the guidance of Regulatory Guide 1.183. Consistent with Regulatory Guide 1.183 the analysis assumed both a pre-accident iodine spike and a concurrent iodine spike.

14.3.1.4.1 SGTR Analysis Assumptions

It has been determined that tube bundle uncovery can affect doses from a Steam Generator Tube Rupture (SGTR). SGTR dose calculations follow the Westinghouse Owners Group (WOG) developed methodology (Reference 39) for this analysis. This methodology of dose calculations consists of four components:

1. Releases from secondary liquid boiling including allowance for a partition factor of 0.01 for iodine between secondary liquid and steam.
2. Releases from the fraction of primary liquid break flow that flashes to steam. A partition factor of 1 is assumed for this flashing fraction.
3. Releases from primary liquid bypassing the secondary side.
4. Releases caused by secondary moisture carryover.

As shown in Reference 39, releases from a SGTR are dominated by the first two terms above for a case with a stuck open PORV. A stuck open PORV also produces a larger radionuclide release than a cycling PORV or a PORV that fails closed, and causes the steam generator safety valves to open to relieve secondary side pressure. The LOCADOSE computer model for the SGTR analysis includes terms 1, 2, and 4 discussed above.

Uncovery of the tube bundle in a SGTR does not significantly increase radionuclide releases for the stuck open PORV case. If the tube bundle is uncovered in a SGTR and the PORV

is stuck open, the third release term described above increases, but it is still only a small part of the total release.

14.3.1.4.2 Initial Radioisotope Concentrations

The analyses of the SGTR accidents indicate that no additional fuel rod failures occur as a result of this transient. Thus, radioactive material releases are determined by the radionuclide concentrations initially present in primary liquid, secondary liquid and secondary steam, plus any releases from fuel rods that have failed before the transient. The analyses considered both pre-accident iodine spike cases and concurrent iodine spike cases. The discussion regarding the determination of the pre-accident and concurrent iodine spikes found in Section 14.3.2.4.2 is applicable to the SGTR as well as the main steam line break. The radionuclide inventories and concentrations for the concurrent and pre-accident iodine spike cases are shown in Tables 14.3-10 and 14.3-11, respectively.

14.3.1.4.3 Determination of χ/Q Values

The exclusion area boundary (EAB) and low population zone (LPZ) χ/Q values given in Table 14.3-13 were determined based on Regulatory Guide 1.145 methodology using meteorological data from 1994 to 1998.

During a SGTR with loss of offsite power, or the condenser otherwise unavailable, the steam generators release steam through the secondary system PORVs. The control room χ/Q values for releases from the steam generator PORVs to the control room emergency air inlet are given in Table 14.3-13. The limiting case for control room dose assumes loss of the main condenser; however, offsite power is assumed to remain available to power the normal control room ventilation fans.

The distance from the closest PORV to the control room normal air inlet is shorter than the distance to the emergency inlet, so a different χ/Q value is applicable for releases when the normal control room ventilation system is in use. Based on the Reference 34 methodology, the control room χ/Q for the normal inlet was determined to be $7.71 \times 10\text{-}3$ sec/m^3. This χ/Q is only used for the short time (247 seconds) before the control room is isolated from the normal inlet air by a SI signal. The χ/Q value (7.71×10^{-3} sec/m^3) is only used in the SGTR analysis and is not included in Table 14.3-13.

14.3.1.4.4 Steam Generator Tube Rupture LOCADOSE Models

The LOCADOSE computer code system (References 29 through 31) with Federal Guidance Report 11 and 12 dose conversion factors (References 7 & 41) was used to model the SGTR. Models were developed for both a pre-accident iodine spike case and a concurrent iodine spike case. The two models are identical except for the initial radioisotope inventories and the inclusion of modeling of an iodine release from the fuel rods for eight hours for the concurrent iodine spike case.

The primary system, steam generator, and control room volumes for the SGTR are given in Table 14.3-12. The release of the radionuclides in the steam from the unaffected steam generators was modeled as essentially a puff release occurring when the PORVs open. The affected steam generator release was modeled using numerical integration over the time period of release (30 minutes).

The primary coolant leakage to the unaffected steam generators was assumed to be 1 gpm, which is conservative with respect to Surry Technical Specifications. The maximum leakage allowed by Technical Specifications is 150 gpd through any one steam generator.

For conservatism, all of this leakage was assumed to occur into the two unaffected steam generators. This assumption is conservative because the unaffected generators release steam to the environment for 8 hours compared to 30 minutes for the affected generator.

The break flow rates through the ruptured tube to the affected steam generator were based on the thermal hydraulic analysis of a complete double-ended tube rupture. To be consistent with the guidance in Regulatory Guide 1.183 (Reference 32), the liquid and steam break flows are modeled separately. The break flow rates and release rates to the environment are summarized in Tables 14.3-7 and 14.3-8 for the cases with and without continued availability of offsite power (used for the control room and site boundary dose calculations, respectively).

The liquid break flow through the primary system is modeled as mixing with the secondary liquid in the affected steam generator. The flow from the secondary liquid to the secondary steam is then modeled assuming a partition factor of 0.01 for iodine and moisture carryover of 1% for particulates. The fraction of the break flow that flashes to steam is modeled as being transferred to the affected steam generator steam space with no credit for scrubbing by the steam generator liquid, i.e., equivalent to tube uncovery. Once in the steam generator steam space, the radionuclides in this part of the break are almost immediately released to the environment. This technique for modeling a SGTR with uncovery of the tube bundle was developed in a generic study by the Westinghouse Owners Group (Reference 39).

The primary and secondary system releases are replaced with safety injection and auxiliary feedwater flows. Therefore, the volume of the primary and secondary liquids remains relatively constant during this transient.

The radionuclide inventory in the steam generators is modeled based on the initial inventory, the primary to secondary leakage, and the break flow rates and release rates to the environment that are discussed above. Flow through the condenser was not modeled because it was unavailable for the loss of offsite power case and because modeling the condenser reduces dose consequences.

The model for the control room ventilation system for the SGTR is set up to accurately model the timing of the sequence of events of the SGTR accident. The start of the accident is the tube rupture itself. The PORV on the faulted steam generator was determined to open 263 seconds

(0.0731 hrs.) after the break, and the SI signal is generated at 365 seconds (0.1014 hrs.) for the case assuming continued availability of offsite power. For the case assuming loss of offsite power, the PORV on the faulted steam generator was determined to open 80 seconds (0.0222 hrs.) after the break, and the SI signal is generated at 225 seconds (0.0625 hrs.). The timing of these events was extracted from the thermal-hydraulic analysis. During this time, the control room is being supplied via the normal ventilation system, with a 3000 cfm intake air flow rate. The control room isolates automatically on initiation of the SI signal. Emergency ventilation is assumed to provide a filtered breathing air supply of 1000 cfm within 1 hour of control room envelope isolation until the end of the accident. An unfiltered inleakage of 10 or 500 cfm was assumed for the entire time the control room is isolated. The control room intake filter efficiency assumed was 90% and 70% for elemental and organic iodine, respectively. All other non-noble gas isotopes modeled were filtered at 99% efficiency.

14.3.1.4.5 Results of Dose Calculations for SGTR

Both pre-accident and concurrent iodine spike cases were analyzed for the steam generator tube rupture. The limiting case for the control room dose was determined to be a pre-accident iodine spike with continued availability of offsite power. This dose is shown in Table 14.3-9 and corresponds to 10 cfm of unfiltered control room inleakage. For 500 cfm of unfiltered control room inleakage the calculated dose actually goes down slightly. This result is attributed to the decreased average residence time for radionuclides in the control room with the higher inleakage.

Table 14.3-9 also shows a comparison of the doses calculated for the limiting SGTR accident scenario with the GDC-19 criteria. All calculated control room doses for the Surry steam generator tube rupture remain below the GDC-19 criteria.

The limiting case for the EAB and LPZ was determined to be a concurrent iodine spike with loss of offsite power. The EAB and LPZ doses shown in Table 14.3-9 are less than the Regulatory Guide 1.183 limits for concurrent iodine spike cases.

14.3.1.5 **Recovery Procedure**

The immediately apparent symptoms of a tube rupture accident, such as falling pressurizer pressure and level and increased charging pump flow, can also be symptoms of small steam-line breaks and loss-of-coolant accidents. It is therefore important that the operator determine that the accident is the rupture of a steam generator tube in order to carry out the correct recovery procedure. This accident is uniquely identified by a condenser air ejector radiation alarm or a steam generator blowdown radiation alarm, and the operator does not proceed with the following recovery procedure unless these alarms are observed. In the event of a relatively large rupture, such as that analyzed above and shown in Figures 14.3-1 to 14.3-6, it is clear soon after the trip that the level in one steam generator is rising more rapidly than those in the others. This indication is used in identifying the affected steam generator.

The analysis described above takes no credit for operator action for the first 30 minutes. In an actual event, within 30 minutes the operators would be expected to achieve the following:

1. Ensure that power is available to the emergency buses and that safety injection and auxiliary feedwater are actuated. Verify that main feedwater is isolated.
2. Control the reactor system cooldown to maintain no-load temperature. Stop the reactor coolant pumps if safety injection flow to the core is indicated and the minimum required RCS subcooling is not maintained.
3. If not already completed, identify the ruptured steam generator by rising water level or high steam line radiation indications and isolate flow from this steam generator. Adjust auxiliary feedwater flow to maintain the specified water levels in the affected and intact steam generators.

Completion of these steps terminates the release of radioisotopes. For the analysis case discussed above, following termination of flow from the stuck PORV in the ruptured generator, more than 15 additional minutes would elapse prior to repressurizing the steam generator to the nominal PORV relief setpoint (Reference 40), assuming no additional operator actions. In practice, upon completion of identification and isolation of the ruptured generator, the following additional actions are performed:

1. Initiate RCS cooldown through the intact steam generators by dumping steam to the main condenser or through the steamline PORV (depending on the availability of offsite power). Following a loss of offsite power, the steam generator PORVs can not be operated remotely from the main control room because the control circuits are powered from a semi-vital source. However, a backup bottled air system has been provided so that the SG PORVs can be operated from within the Containment Spray Pump House. A second backup bottled air system is also provided inside the Main Steam Valve House, which is a safety-related, seismic structure that is tornado missile protected.
2. Depressurize the RCS to minimize break flow and refill the pressurizer using the pressurizer spray or the pressurizer PORVs. Maintain the RCS pressure within the pressure-temperature limit curve for the Reactor Coolant System.
3. Terminate safety injection flow upon meeting the SI termination criteria.
4. Establish normal letdown and charging functions and control RCS pressure to minimize primary-to-secondary leakage.
5. Initiate appropriate post-SGTR cooldown procedures.

These additional actions limit the potential for any additional releases from the affected generator following the isolation step.

The generic analyses of Reference 40 show that the stuck PORV case yields higher releases than the case where the PORV on the affected generator cycles normally. For cases where less than a full double-ended tube rupture occurs, it may take the operator longer to perform the RCS

depressurization step. However, for this case, the break flow rate will also be lower. Therefore, although in specific event scenarios some limited additional relief from cycling of the affected generator's PORV might occur beyond 30 minutes, the analysis cases presented here are bounding in terms of total integrated release and therefore radiological consequences. Based on observation of the relative releases cited for the stuck PORV, cycling PORV and cycling main steam safety valves in Section 9 of Reference 40, the stuck PORV analysis presented here bounds the following scenarios:

1. SGTR with the ruptured SG's PORV cycling at its nominal setpoint; releases terminated at approximately 37 minutes.
2. SGTR with the ruptured SG PORV isolated and the associated main steam safety valve(s) cycling at the nominal setpoint. Releases terminated well beyond 1 hour.
3. Any case above with a break area corresponding to less than a double-ended tube rupture (initial break flow rate of 800 gpm).

14.3.2 Rupture of a Main Steam Pipe

A rupture of a main steam pipe (the pipes that carry steam from the steam generators to the main turbine) is assumed to include any accident that results in an uncontrolled steam release from a steam generator. The release can occur as a result of either a break in a pipe line or a valve malfunction. The steam release results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of reactor coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of the core shutdown margin. If the most reactive control-rod assembly is stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power, even with the remaining control-rod assemblies inserted. A return to power following a main steam pipe rupture is a potential problem mainly because of the high hot-channel factors that exist when the most reactive rod is stuck in its fully withdrawn position. Assuming the worst combination of circumstances that could lead to the resumption of power generation following a main steam line break, the core is ultimately shut down by the boric acid in the safety injection system.

The analysis of a main steam pipe rupture is performed to demonstrate that:

1. Assuming a stuck control-rod assembly with or without offsite power, and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.
2. There will be no DNB or clad perforation resulting from any single active failure in the main steam system. The single active failure is the opening, with failure to close, of the largest of any single steam bypass, relief, or safety valve.
3. Energy release to containment from the worst steam pipe break does not cause failure of the containment structure.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis shows that no DNB occurs for any rupture, even in the event that the most reactive control-rod assembly is stuck in its fully withdrawn position.

The following systems provide the necessary protection against a main steam pipe rupture:

1. Safety injection system actuation by any of the following (see Chapter 7 for logic details):
   a. Two out of three pressurizer low pressure signals.
   b. Two out of three differential pressure signals between any main steam line and the main steam header.
   c. High steam flow in two out of three main steam lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (two out of three) or low main steam line pressure (two out of three).
   d. Three out of four high containment pressure signals.
   e. Manual intervention.
2. The overpower reactor trips (neutron flux and delta T) and the reactor trip occurring upon actuation of the safety injection system.
3. Redundant isolation of the steam generator feedwater lines. Sustained high feedwater flow would cause additional cooldown; thus, in addition to the normal control action that closes the main feedwater valves, any safety injection signal rapidly closes all feedwater control valves, trips the steam generator feedwater pumps, and closes the feedwater pump discharge valves.

   The feedwater isolation function is primarily accomplished by safety grade feedwater control valves and feedwater control valve bypass valves. The feedwater isolation design does not include safety grade back-up isolation capability. However, the automatic trip of the main feedwater pumps and closure of the feedwater pump isolation valves accomplishes the back-up feedwater isolation function. The reliance upon commercial grade isolation equipment as back-up feedwater isolation has been accepted as a generic industry position as documented in NUREG-0138. The failure of a feedwater control valve or bypass valve to close upon a feedwater isolation signal has been evaluated and shown to be bounded by the assumptions in the limiting analysis described in this section.
4. The trip of the fast-acting main steam line trip valves (designed to close within 10 seconds from the time the process variable reaches the trip setpoint) on:
   a. High steam flow in two out of three main steam pipes (one out of two per line) in coincidence with either low reactor coolant system average temperature (two out of three) or low steam line pressure (two out of three).
   b. Three out of four high containment pressure signals.

Each main steam line has a fast-closing trip valve and a nonreturn valve. These six valves prevent the blowdown of more than one steam generator for any break location in a main steam pipe, even if one valve fails to close. For example, for a break upstream from the trip valve in one line, the closure of either the nonreturn valve in that line or the trip valves in the other lines prevents the blowdown of the other steam generators.

All Surry steam generators are equipped with integral flow restrictors at the generator outlet. The restrictors have a smaller flow area than the main pipe and serve to reduce the largest effective break area which must be considered to 1.4 ft^2.

A special case of this event is a main steam line break in the main steam valve house (outside containment). The transient is described in Section 14B.6.

14.3.2.1 **Method of Analysis**

The analysis of the main steam pipe rupture has been performed to determine:

1. The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam-line break. The analysis was performed with the RETRAN (Reference 12) computer code. The calculation describes the plant neutron kinetics, the reactor coolant system including natural circulation, the pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the break flow rate, core power and point kinetics reactivity and primary coolant temperatures.
2. The thermal and hydraulic behavior of the core following a steam-line break. A detailed COBRA (Reference 11) thermal and hydraulic digital computer calculation has been used to determine if DNB occurs for the core conditions computed in 1 above for the SIF product. The 15 x 15 Upgrade fuel design was analyzed to determine if DNB has occurred using the VIPRE-D code (Reference 42) for the core conditions computed in 1 above. These calculations solve the continuity, momentum, and energy equations of fluid flow in the core, and with the Westinghouse W-3 (Reference 2) or WLOP (Reference 42) correlations determines the DNB margin.

The following assumptions were made:

1. A 1.77% delta k/k shutdown reactivity from all but one control rod assembly at no-load conditions. This is the end-of-life design value, including design margins for the case in which the most reactive control-rod assembly is stuck in its fully withdrawn position. The actual shutdown capability is expected to be significantly greater.
2. The negative moderator coefficient corresponding to the end-of-life core with all but the most reactive control-rod assembly inserted. The variation of the coefficient with temperature has been included. In computing the power generation following a steam-line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control-rod assembly has been included in the overall reactivity balance. The local reactivity feedback is composed of Doppler reactivity from the high fuel temperatures near

the stuck control-rod assembly and moderator feed back from the high water temperature near the stuck control-rod assembly. The Doppler reactivity feedback corresponds to a most negative hot zero power Doppler temperature coefficeint. For the cases in which steam generation occurs in the high flux regions of the core, the effect of void formation on the reactivity has also been included. The effect of power generation in the core on overall reactivity is shown in Figure 14.3-8. The curve assumes end-of-life core conditions with all control-rod assemblies in except the most reactive control-rod assembly, which is assumed to be stuck in its fully withdrawn position (completely removed from the core).

3. Minimum safety injection capability corresponding to the operation of only one high head safety injection pump. The most restrictive single failure corresponds to the flow delivered by one charging pump delivering its full flow to the cold leg header. A boron concentration of 2300 ppm was assumed in the Refueling Water Storage Tank (RWST), from which the safety injection pumps take suction. Boron enters the safety injection system after the charging pump suction switches over from the volume control tank to the refueling water storage tank upon safety injection actuation.

   The assumed single failure for the steamline break analysis is the failure of one safeguards train to function, thus maximizing the delay time for boron to reach the core. Other failures that could affect the severity of the transient are bounded by the failure of a safeguards train.

   The initial boron concentration in the Boron Injection Tank (BIT) and the associated safety injection piping is assumed to be zero. The time delays incurred prior to the delivery of the 2300 ppm boron have been included in the analysis. These time delays are conservatively based on the SI system design which included a 900-gallon boron injection tank (BIT). The BIT has been subsequently removed from the SI system on both units (Reference 4). An evaluation of this change against the criteria of 10 CFR 50.59 showed that the main effect of removing the BIT was to significantly reduce the time delay required to sweep unborated water from the SI piping following a safety injection signal, which would be a benefit from a safety analysis standpoint. Thus, the steamline break analyses based on a BIT at 0 ppm are conservative and bound the current condition with the BIT removed.

4. Hot-channel factors corresponding to one stuck control-rod assembly, i.e. the control-rod assembly giving the highest factor at the end of life. The hot-channel factors account for the void existing in the locality of the stuck control-rod assembly at the pressure that occurs during the return-to-power phase following the steam break. This void, in conjunction with the large negative moderator coefficient, partially offsets the effect of the stuck control-rod assembly. The hot-channel factors depend on the core temperature, pressure, and flow, and are therefore different for each case studied. The calculations used to obtain the hot-channel factors again assume end-of-life core conditions with all control-rod assemblies in except the most reactive control-rod assembly.

5. Three combinations of break sizes and initial unit conditions were considered in determining the core power and reactor coolant system transient:

a. The complete severance of a main steam pipe, initially at no load conditions with offsite power available. The presence of the integral flow restrictors in the steam generators will control the steam release rates for all break locations, both inside and outside the containment.

b. Case A above with loss of offsite power immediately before the steam break.

c. A break larger than or equal to the capacity of any single steam dump or safety valve from one steam generator with offiste power available (credible break).

All the cases above assume initial hot-shutdown conditions with the control-rod assemblies inserted (except for one stuck control-rod assembly) at time zero. Should the reactor be just critical or operating at power at the time of a main steam line break, the reactor is tripped by the normal overpower protection system when the power level reaches a trip point or by the safety injection signal from the steamline break protection functions. Following a trip at power, the reactor coolant system contains more stored energy than at no load, the average coolant temperature is higher than at no load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the main steam line break before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes a no-load condition at time zero. However, since the initial steam generator mass is greatest at no load, the magnitude and duration of the reactor coolant system cooldown are less for main steam line breaks occurring at power.

1. In determination of the critical flux at which burnout could occur, the W-3 or WLOP Correlation may be used. This was considered to be the correlation that most accurately represented the range of parameters produced in the transients analyzed.
2. In computing the steam flow during a steam-line break, the Moody Critical Flow Model (Reference 3) was used.

14.3.2.2 **Results**

The results presented are a conservative indication of the events that would occur assuming a main steam line rupture, since it is postulated that all of the conditions above occur simultaneously.

14.3.2.2.1 Core Power and Reactor Coolant System Transient

Figures 14.3-9 through 14.3-13 show the reactor coolant system transient and core heat flux following a main steam pipe rupture (complete severance of a pipe) at initial no-load conditions (Case A). The break assumed is the largest break that can occur anywhere in the system. Offsite power is assumed available such that full reactor coolant flow exists. The transient shown assumes that the control-rod assemblies are inserted at time 0 (with one control-rod assembly stuck in its fully withdrawn position) and steam is released from only one steam generator after

closure of the steamline trip valves. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by high differential pressure between any steam generator and the main steam header or by high steam flow signals in coincidence with either low reactor coolant system temperature or low steam-line pressure. The current bounding main steam line break analysis assumes a 5-second delay from the time the measured process variables (e.g., steam line flow, steam line pressure) reach the main steam line setpoints to the initiation of main steam trip valve motion, followed by an additional 5-second ramp closure of the valves.

The acceptance criteria for a satisfactorily full closure test of a trip valve is defined in the Basis of Technical Specification 4.7-2 (Reference 26). With the high flow existing during a main steam line rupture, the valves will close considerably faster since their closure is flow assisted.

Tables 14.3-1 through 14.3-3 outline the sequence of events and Tables 14.3-4 and 14.3-5 the transient statepoint parameters for the three main steamline break cases. Figures 14.3-9 through 14.3-23 plot the transient results for several key parameters in the offsite power case, loss of offsite power case, and credible break case, respectively.

As shown in Figure 14.3-9 through 14.3-13, the core attains criticality with the control-rod assemblies inserted (with the design shutdown, assuming one stuck control-rod assembly) before boron solution enters the reactor coolant system from the safety injection system. The delay time consists of the time to receive and actuate the safety injection signal and the time to completely open or realign valve trains in the safety injection lines. The safety injection pumps are then ready to deliver flow. At this stage a further delay time is incurred before boron solution can be injected to the reactor coolant system, due to 0% boron concentration water being swept from the safety injection lines. No credit was taken for any boron in the safety injection lines entering the reactor coolant system prior to the 2300 ppm boric acid from the refueling storage tank. The case attains a peak core power well below the nominal full power value.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the reactor coolant system before entering the reactor core. The concentration after mixing depends on the relative flow rates in the reactor coolant system and in the safety injection system. The variation of mass flow rate in the reactor coolant system due to water density changes is included in the calculation, as is the variation of flow rate in the safety injection system due to changes in the reactor coolant system pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

Figures 14.3-14 through 14.3-18 show the responses of the core parameters for Case B, which corresponds to the case discussed above with loss of offsite power at the time the main steamline break occurs. The safety injection system delay time includes 10 seconds to start the diesel and 10 seconds for the safety injection pump to reach full speed. Criticality is reached later in the transient and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the reactor coolant

system is reduced by the decreased flow in the reactor coolant system. The peak core power remains well below the nominal full power value.

It should be noted that, following a main steam-line break, only one steam generator blows down completely. Thus, two steam generators are still available for the dissipation of decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere, the atmospheric safety valves having been sized to cover this condition.

Figures 14.3-19 through 14.3-23 show the responses of the core parameters resulting from a steam release with an initial steam flow typical of the capacity of any single steam dump or safety valve. In this case, safety injection is initiated automatically by low pressurizer pressure. The limited cooldown resulting from the stuck open valve results in the transient reaching criticality much later in the transient than in the other cases. Sufficient negative reactivity remains to limit the peak heat flux to approximately 7% of the rated power. With the reactor coolant pumps still providing full flow, the case is bounded in terms of minimum DNBR by the offsite power case in Case A, severance of a main steam pipe.

The evaluation of Reference 5 demonstrates that even with 0 ppm boron in the BIT, the containment design bases are met for steam line break. As discussed previously, removal of the BIT provides an analysis benefit with respect to the case of 0 ppm in the BIT, by reducing the time delay for introducing borated water to the core. This reduced time delay will result in a more rapid shutdown and, therefore, reduced mass and energy release to the containment.

14.3.2.3 **Margin to Critical Heat Flux**

Using the transients shown in Figures 14.3-9 through 14.3-23, the Westinghouse W-3 Correlation was used in conjunction with the COBRA core thermal hydraulics code to determine the margin to DNB for the SIF product. The 15 x 15 Upgrade fuel design is analyzed with the VIPRE-D/W-3 code/correlation pair or the VIPRE-D/WLOP code/correlation pair (Reference 42) to determine the margin to DNB. Carefully chosen points from each transient were examined, and the results showed that all three cases have a minimum DNBR greater than the applicable SAL (Section 3.2.3). The power and flow statepoint conditions are shown together with pressure and inlet core temperature in Table 14.3-4 and 14.3-5.

14.3.2.4 **Environmental Consequences of a Main Steam-Line Break (MSLB)**

A Main Steam Line Break (MSLB) involves the postulated double ended failure of one of the steam lines carrying steam from a steam generator to the turbine generator. Two cases have to be considered. Offsite doses are determined based on a case with minimal retention of radionuclides in the turbine building. The control room dose analysis assumes that the MSLB occurs in the turbine building, where the control room emergency air inlet is located, and that the turbine ventilation fans fail to operate. The MSLB analysis used the alternative source term and followed the guidance of Regulatory Guide 1.183.

Because the MSLB releases are assumed to occur in the turbine building, the normal χ/Q methodology used for the control room does not apply. χ/Q is used to determine the concentration of a radioisotope χ in Ci/m^3 from the release rate Q in Ci/sec. The control room χ/Q is normally determined with the methodology of Reference 34 based on the distance between release and receptor points and site meteorology. Depending on the type of release, building wake effects may also be considered. For the MSLB, the releases occur in the same building as the control room emergency inlet, so the Murphy and Campe χ/Q methodology does not apply. Therefore, the direct pathway from the steam line break to the turbine building was modeled along with the intake of control room air from the turbine building.

14.3.2.4.1 MSLB Analysis Assumptions

There is no control room χ/Q defined for a situation when the releases are into the same building where the inlet to the control room is located. Therefore, for the MSLB it was necessary to use a different approach to model the transport of radioactive steam releases from the broken steam line to the control room. (Normal χ/Q methodology is applicable to the modeling of the releases through the unaffected steam generators.)

The control room is normally modeled in the LOCADOSE computer code as a special volume "connected" only to the environment, with the inlet concentrations based on releases to the environment and the χ/Q for the control room. However, the control room radioisotope concentrations can be calculated with LOCADOSE by defining one of the user specified volumes as the control room and appropriately modeling the air flows, including the inlet air from the turbine building, to this control room volume.

As a starting point for the MSLB analysis, the concentrations of each radioisotope in the primary liquid, secondary liquid and secondary steam were determined. Radionuclides are released with the steam from these sources through the break. These MSLB release rates are shown in Table 14.3-14.

The flow rates used in this analysis considered the volume expansion that occurs when pressurized liquid or steam is discharged from the steam generator to the turbine building. The flow rate from the steam generator to the turbine building was based on the density of steam or liquid inside the steam generator, while the flow rate from the turbine building to the environment was based on the expansion of steam to atmospheric pressure inside the turbine building. This MSLB model is summarized below.

14.3.2.4.2 Initial Radioisotope Concentrations

For the MSLB, the radioactive material releases are determined by the initial radionuclide concentrations present in primary liquid, secondary liquid and secondary steam, plus any releases from failed fuel rods. The amount of activity in the primary and secondary coolant at the initiation of the MSLB is assumed to be the maximum levels allowed by the plant Technical Specifications.

Consistent with RG 1.183 (Reference 32), both a pre-accident iodine spike and a concurrent iodine spike were considered for the MSLB. For Surry, the maximum iodine concentration allowed in Surry Technical Specifications for an iodine spike is 10 μCi/gm dose equivalent I-131. RG 1.183 defines a concurrent iodine spike as an accident initiated increase in the release rate of iodine from failed fuel rods to a value 500 times the release rate corresponding to the Technical Specification dose equivalent iodine limit for normal operations. (For the SGTR accident, the concurrent spike release rate of iodine from failed fuel rods is set to a value 335 times the release rate corresponding to the Technical Specification dose equivalent iodine limit for normal operations.) In addition to the Technical Specification dose equivalent iodine concentrations and to bound the release rate expected during normal operations; the release rate was determined assuming hot full power conditions, a letdown flow rate of 120 gpm, a primary system leak rate of 11 gpm (10 gpm identified and 1 gpm unidentified), primary-to-secondary leakage of 450 gpd (150 gpd per steam generator) and a letdown decontamination efficiency of 100%. The concurrent iodine spike term of 500 times the release rate is also known as the concurrent iodine spike appearance rate. A concurrent iodine spike is more likely than a pre-accident spike since the pressure change caused by an accident can increase iodine releases from failed fuel rods. A pre-accident iodine spike is unlikely, since some independent event would have had to occur shortly before the accident to cause the spike.

The primary liquid, secondary liquid and secondary steam radionuclide inventories are given in Table 14.3-10. The pre-accident spike activity and the concurrent iodine spike rate are given in Table 14.3-11. The secondary side activity levels are initially the same (at the Technical Specification secondary activity limit) for the pre-accident and concurrent spike cases. Only the primary liquid activities differ. The concurrent iodine spike case assumes the primary coolant activity is initially at the steady state activity limit of 1 μCi/gm dose equivalent I-131 shown in Table 14.3-10, with iodine added at the appearance rates shown in Table 14.3-11. The pre-accident spike case assumes the primary coolant iodine activity is initially at the short term Technical Specifications limit of 10 μCi/gm dose equivalent I-131 and the remainder of the primary coolant isotopes are at the steady state activity limit of 1 μCi/gm dose equivalent I-131 shown in Table 14.3-10. The isotopic concentrations and iodine spike values in Tables 14.3-10 and 14.3-11 are derived from RCS 1% Failed Fuel concentrations shown in Table 14.3-16, which are based on a core power of 2605 MWt and nominal 18 month operating cycles, by normalizing to the Technical Specifications limits for primary and secondary liquid dose equivalent I-131 using dose conversion factors from RG 1.109. Technical Specifications allow the use of either RG 1.109 or TID-14844 dose conversion factors to determine dose equivalent I-131. RG 1.109 dose conversion factors were used to determine the coolant activities and iodine spike factors rather than TID-14844 dose conversion factors because use of RG 1.109 dose conversion factors results in higher allowable coolant activities and iodine spiking terms, and therefore a more limiting analysis. These inventories and appearance rates were input to the LOCADOSE code system to calculate doses from an MSLB. The volumes of the primary liquid, secondary liquid and secondary steam used in this MSLB dose analysis are listed in Table 14.3-12. The initial

inventory of isotopes in the secondary side steam was conservatively modeled based on the ratio of a primary to secondary leak rate of 1 gpm and the normal steam flow to the condenser.

14.3.2.4.3 Determination of χ/Q Values

The EAB and LPZ atmospheric dispersion factors (χ/Q) are given in Table 14.3-13 and were determined based on Regulatory Guide 1.145 methodology using meteorological data for 1994 to 1998.

The unaffected steam generators are assumed to release steam through the secondary system steam relief valves. Based on the Reference 34 methodology, a diffuse source - point receptor χ/Q can be used when the elevation difference between the point source (steam generator relief valves) and point receptor (control room emergency air inlet) is more than 30% of the height of the source building. The steam releases from the unaffected generators meet this criterion. Using the Murphy and Campe methodology, the control room χ/Q values shown in Table 14.3-13 were determined for releases from the unaffected generators.

14.3.2.4.4 Main Steam Line Break LOCADOSE Models

The LOCADOSE computer code system (References 29 through 31) with FGR 11 and FGR 12 dose conversion factors (References 7 & 41) was used to model the MSLB. Two LOCADOSE models were created, one for the pre-accident iodine spike and the other for the concurrent accident spike.

The flow rates from the primary coolant to the steam generators prior to the start of the accident were based on conservatively assumed leak rates with respect to Technical Specifications. The maximum leakage from one generator was assumed to be 500 gpd into the generator affected by the steam line break. The maximum leak rate allowed by the Technical Specifications is 150 gpd through any one steam generator.

The affected steam generator was modeled as discharging through the turbine building, while the other two generators were modeled as discharging directly to the environment. The flow rates from the affected steam generator liquid to the turbine building and from the turbine building to the environment are summarized in Table 14.3-14.

All of the iodine being released is conservatively assumed to be airborne. In practice, some of the steam generator discharge would be as water, which would retain some of the iodine in the liquid phase.

The mass release in thirty minutes (Table 14.3-14) is several times the initial mass of the affected steam generator. Therefore, the volume released from the steam generator to the turbine building was increased above the calculated values to ensure that substantially all of the radionuclides initially present in the affected steam generator were released.

Because the affected steam generator is essentially emptied of liquid during the MSLB, no partitioning of iodine between the liquid and steam is assumed for discharges from the affected

generator. A partition factor of 0.01 for the iodine isotopes and a moisture carryover of 1% for the particulate isotopes were assumed for the unaffected generators. The flow rate from the turbine building to the environment considered the expansion of the steam as the pressure is reduced to atmospheric in the turbine building. In addition, because the turbine building is not a sealed building, air flow through the building was considered. The building has a forced ventilation system capable of approximately one volume exchange every six minutes.

However, this forced ventilation system would not function after a loss of offsite power. One volume exchange per hour is a reasonable air flow rate for the turbine building without forced ventilation. For conservatism, the control room doses were calculated assuming only a 0.2 volume/hour air flow rate. A forced ventilation of 12 volumes/hour was also evaluated to provide a bounding case for offsite dose calculations.

The control room was assumed to be isolated. Leakage into the control room was assumed to be from the turbine building where the break occurs (no atmospheric dispersion). Filtered intake of 1000 cfm was modeled after the first hour. An unfiltered inleakage of 500 cfm was again assumed for the full duration of the accident (0 to 30 days).

14.3.2.4.5 Results of Dose Analysis for MSLB

The control room, EAB and LPZ doses calculated for the MSLB are shown in Table 14.3-15. The limiting accident scenario for the calculation of the doses was determined to be a concurrent iodine spike case. The calculated control room dose from a MSLB is below the GDC-19 criteria. The doses calculated for a MSLB at the EAB and LPZ are less than the RG 1.183 limits for concurrent iodine spike cases.

14.3.2.5 **Conclusions**

Although DNB and possible clad perforation (no clad melting or zirconium-water reaction) following a steam pipe rupture are not necessarily unacceptable, the above analysis, in fact, shows that DNB does not occur for any rupture, assuming the most reactive rod stuck in its fully withdrawn position.

The minimum DNBRs determined in the analysis of the steamline break are greater than the applicable SAL (Section 3.2.3).

14.3.3 Rupture of a Control Rod Drive Mechanism Housing (Control Rod Assembly Ejection)

14.3.3.1 **Identification of Causes and Accident Description**

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

14.3.3.1.1 Design Precautions and Protection

Certain features in Westinghouse pressurized water reactors are intended to preclude the possibility of a rod-ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, a thorough quality control (testing) program during assembly, and a nuclear design that lessens the potential ejection worth of rod cluster control assemblies and minimizes the number of assemblies inserted at power.

14.3.3.1.1.1 Mechanical Design. The mechanical design is discussed in Section 3.5. Mechanical design and quality control procedures intended to preclude the possibility of a rod cluster control assembly (RCCA) drive mechanism housing failure sufficient to allow a rod cluster control assembly to be rapidly ejected from the core are listed below:

1. Each Unit 1 control rod drive mechanism housing is completely assembled and shop-tested at 3450 psig. Each Unit 2 control rod drive housing is hydrostatically tested at the shop at 3107 psig.
2. The mechanism housings are checked during the system leak test of the reactor coolant system.
3. Stress levels in the mechanisms are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design-basis earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class A components.
4. The Unit 1 latch mechanism housing and rod travel housing are each a single length of forged type 304 stainless steel. The Unit 2 latch mechanism housing and rod travel housing are each a single length of forged type 316 stainless steel. These materials exhibit excellent notch toughness at all temperatures that will be encountered.

The Unit 1 joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds. The Unit 2 joints between the latch mechanism housing and head adapter housing are threaded joints reinforced by canopy type welds. The Unit 2 joints between the latch mechanism housing and rod travel housing are butt welds. Administrative regulations require periodic inspections of these (and other) welds.

14.3.3.1.1.2 Nuclear Design. Even if a rupture of a rod cluster control assembly (RCCA) drive mechanism housing is postulated, the operation of a plant using chemical shim is such that the severity of an ejected rod cluster control assembly is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated for by boron changes. Further, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a RCCA-ejection accident. Therefore, should a rod cluster control assembly be ejected

from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may occasionally be desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the rod cluster control assemblies above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all rod cluster control assemblies is continuously indicated in the control room. An alarm will occur if a bank of rod cluster control assemblies approaches its insertion limit or if one assembly deviates from its bank. There are low-low-level insertion monitors with visual and audio signals.

14.3.3.1.1.3 Reactor Protection. The reactor protection in the event of a rod-ejection accident has been described in Reference 13. The protection for this accident is provided by the high-neutron-flux trip (high and low setting). This protection function is described in detail in Section 7.2.

14.3.3.1.1.4 Effects on Adjacent Housings. A control-rod drive mechanism assembly is shown in Chapter 3. The operating coil stack assembly of this mechanism has a 10.718-inch by 10.718-inch cross section and is 39.875 inch in length. The position indicator coil stack assembly is located above the operating coil stack assembly. It surrounds the rod travel housing over nearly its entire 163.25-inch length. The rod travel housing outside diameter is 3.75 inch and the inside and outside diameters of the position indicator coil stack assembly are 3.75 and 7 inches, respectively. This assembly consists of a Micarta tube surrounded by a continuous stack of copper-wire coils. This assembly is held together by two end plates, an outer sleeve, and four axial tie rods.

14.3.3.1.1.4.1 Effects of Rod Travel Housing Longitudinal Failures. Should a longitudinal failure of the rod travel housing occur, the region of the Micarta tube opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the Micarta tube. The development of a radial free-water jet would be unlikely because of the small clearance between the Micarta tube and the rod travel housing, and the considerable resistance of the combination of the Micarta tube and the position indicator coils to internal pressure.

Calculations based on experimental data on the mechanical properties of Micarta and copper at reactor operating temperature show than an internal pressure of at least 2500 psia would be necessary for the combination of the Micarta tube and the coils to start leaking in a radial direction between the Micarta glass filaments.

The normal operating environment of the Micarta tube is strictly controlled during unit operation, and thus no deterioration of the Micarta is expected. Should for unknown reasons the mechanical strength of the Micarta tube be reduced and a longitudinal crack occur in a control-rod assembly housing, weepage flow between the Micarta filaments and the copper coil wires might

take place, but no free jet should be formed. The formation of a free jet implies a cracking of the Micarta tube, which could occur only with internal pressure substantially in excess of reactor operating pressure. Prolonged exposure to hot water might cause a deterioration of the Micarta and radial leakage might increase; even under these conditions, however, a net radial free jet would be improbable.

A position indicator coil assembly has to maintain its integrity after a housing failure only until the remaining control-rod assembly can be tripped into the core. Should for unknown reasons a failure of the position indicator coil assembly occur after reactor trip, the resulting free radial jet from the failed housing could cause the housing to bend and come into contact with adjacent rod travel housings. If the adjacent housings were on the periphery, they could conceivably bend outward from their bases. The housing material is quite ductile and plastic hinging without cracking could be expected. Rod travel housings adjacent to a failed housing in locations other than the periphery would not be bent because of the rigidity of multiple adjacent housings.

14.3.3.1.1.4.2 Effect or Rod Travel Housing Circumferential Failures. If a circumferential failure of a rod travel housing were to occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical. The position indicator coil stack assembly and the drive shaft would tend to guide the broken-off piece upward during its travel. Travel would be limited to less than 3 feet (Unit 1) or about 15 inches (Unit 2, due to the integral missile shield) by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield, it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing were short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top-end plates of the position indicator coil stack assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

14.3.3.1.2 Limiting Criteria

Due to the extremely low probability of a RCCA-ejection accident, limited fuel damage is considered an accepatable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 14). Extensive tests of Zirconium-clad UO_2 fuel rods representative of those in pressurized-water-reactor-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ

significantly from the TREAT (Reference 15) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10% with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, conservative criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.
2. Peak clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (2700°F). (Reference 25).
3. Peak reactor coolant pressure less than 3000 psi, which is much less than that which would cause damage to the reactor coolant system.
4. Fuel melting limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

14.3.3.2 **Analysis of Effects and Consequences**

14.3.3.2.1 Method of Analysis

Previous analyses of this event are documented in the original FSAR, and in References 9 and 16 through 21 and 27. The initial FSAR analysis was performed by Westinghouse. The calculation was done in two stages: an average core calculation, and then a hot region calculation. The nuclear power transients for the average core calculation were calculated using the CHIC-KIN code developed by the Bettis Atomic Power Laboratory (Reference 22) to solve the point kinetics equations. A detailed heat transfer code, which employed the Tong, Sandberg and Bishop correlation (Reference 23) to determine the film boiling heat transfer coefficient after DNB, was then used for the hot region calculations.

References 16 and 17 updated the original analysis to accommodate the higher end of life (EOL) ejected rod worths and peaking factors realized for Cycle 2 operation at both units.

The Reference 18 analysis was performed to reflect a positive moderator temperature coefficient at beginning of cycle (approximately 3 pcm/F at zero power, decreasing to 1.5 pcm/F at full power). These calculations were done by Westinghouse using the TWINKLE code for the nuclear power transient and the FACTRAN code for the hot spot heat transfer calculations. The method of analysis is given in WCAP-8117 (Reference 24), and the basis for the calculation and the Westinghouse limit criteria is given in WCAP-7588 (Reference 10).

These same models and methods were used for the reanalyses in References 19 and 20. These evaluations were performed because the Cycle 3 reload core design for each unit resulted in violations of one or more of the following design limits: the ratio of the rod worths to the delayed neutron fraction, the peaking factors; maximum ejected rod worths, or minimum delayed neutron fractions.

The analysis in Reference 9 was performed to establish new design limits when the minimum delayed neutron fractions for the Surry 1 Cycle 5 reload design were less than the applicable limits. The Westinghouse models and methods were again used for this analysis.

The rod ejection analysis in Reference 21 was performed to evaluate the impact of the increased drop time associated with the SIF assemblies.

The rod ejection analysis in Reference 27 was performed to establish new design limits to accommodate trends toward higher peaking factors. For the analysis of the rod ejection event, it was determined that the use of ZIRLO cladding results in a small reduction in both the fraction of fuel melting at the hot spot and the fuel peak stored energy when compared with the results for Zircaloy clad fuel (Reference 28). The use of Optimized ZIRLO cladding associated with the 15 x 15 Upgrade fuel design has a negligible effect on the results as compared to ZIRLO cladding (Reference 43). The analysis described below is therefore applicable for all of these clad materials.

A rod ejection analysis has been performed to establish design limits for the moderator temperature coefficient. A subsequent rod ejection analysis described below was performed to support the implementation of Integral Fuel Burnable Absorber (IFBA) fuel. The implementation of IFBA fuel core loading patterns at Surry results in several core physics parameters which exceed values previously analyzed. The current analysis employs increased key core physics parameter inputs which accommodate the predicted core behavior for IFBA core reload patterns. The current analysis is applicable for both IFBA and non-IFBA fuel types at Surry. The analysis performed for the IFBA transition is bounding for the 15 x 15 Upgrade fuel design.

The analysis of the RCCA ejection accident is performed in two stages: first, an average core nuclear power transient calculation, and then a hot-spot heat transfer calculation. The average core power calculation is performed using point neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core power generation by the hot-channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference 25.

14.3.3.2.1.1 Average Core Analysis. The point kinetics model of the RETRAN computer code (References 12 and 25) is used to perform the average core transient analysis. This code includes

the simulation of prompt and delayed neutrons (using the six group model), the thermal kinetics of the fuel and moderator and the balance of the NSS primary and secondary coolant system. Thermal feedback effects are modeled via temperature dependent reactivity coefficients with a detailed multiregion, transient fuel-clad-coolant heat transfer model. Reactivity insertion from the ejection of the control rod and the subsequent reactor trip are accounted for.

Since both the axial and radial dimensions are missing, it is necessary to use very conservative methods (described below) of calculating the ejected-rod worth and hot-channel factor.

14.3.3.2.1.2 Hot-Spot Analysis. The average core energy addition, calculated as described above, is multiplied by the appropriate hot-channel factors, and the hot-spot analysis is performed using a detailed fuel and clad transient heat transfer model of the RETRAN code termed the Hot Spot Model (Reference 25). This model calculates the transient temperature distribution in a cross section of a metal-clad UO_2 fuel rod and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power generation is used within the fuel rod.

The RETRAN Hot-Spot Model uses the Thom subcooled boiling correlation to determine the film heat transfer before departure from nucleate boiling, and the Bishop-Sandberg-Tong correlation (Reference 23) to determine the film-boiling coefficient after departure from nucleate boiling. The DNB heat flux is not calculated; instead, the code is forced into departure from nucleate boiling by specifying a conservative DNB heat flux. The gap heat transfer coefficient is adjusted to force the full-power steady-state temperature distribution to agree with that predicted by design fuel heat transfer codes presently used by Westinghouse.

For full-power cases, the design initial hot-channel factor (Fqt) is input to the code. The hot-channel factor during the transient is assumed to increase from the steady-state design value to the maximum transient value in 0.1 second and remain at the maximum for the duration of the transient. This is conservative, since detailed spatial kinetics models show that the hot-channel factor decreases shortly after the nuclear power peak due to the power flattening caused by the preferential feedback in the hot channel (Reference 10).

14.3.3.2.1.3 System Overpressure Analysis. Because safety limits for the fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot-spot heat flux versus time. Using these heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient, taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the

pressurizer spray and pressure relief valves. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing (Reference 10).

Due to the very conservative method of analysis, the peak surge rate is high enough to cause the reactor coolant pressure to exceed the pressurizer safety valve actuation pressure. However, this condition exists only for a few seconds; consequently, the pressurizer water volume does not change significantly (less than 150 ft^3). Therefore, the transient is not sensitive to the initial pressurizer level, and the programmed value is used.

14.3.3.2.2 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 14.3-6 presents the parameters used in this analysis. The MUR uprate does not impact 0% power initial condition case results and is bounded by analysis at 102% of 2546 MWt or 100.38% of 2587 MWt initial core power.

14.3.3.2.2.1 Ejected-Rod Worths and Hot-Channel Factors. The values for ejected-rod worths and hot-channel factors are calculated using a synthesis of one-dimensional, two-dimensional and three-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculations.

The total transient hot-channel factor, Fqt, is then obtained by combining the axial and radial factors.

Appropriate margins are added to the results to allow for calculational uncertainties, including an allowance for nuclear power peaking due to fuel densification.

14.3.3.2.2.2 Reactivity Feedback Weighting Factors. The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single-channel analysis. Physics calculations were carried out for a large number of radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers that, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape. In this analysis, although a point kinetics method is used, only a radial weighting factor is applied. In addition, no weighting is applied to the moderator feedback. This very conservative radial weighting factor is applied to the Doppler reactivity feedback of the fuel as a function of the post-ejection radial power peaking factor to account for the missing spatial effect. This weighting factor has been shown to be conservative compared to three-dimensional analysis (Reference 25).

14.3.3.2.2.3 Moderator and Doppler Coefficient. The critical boron concentration at the beginning of cycle (BOC) and end of cycle (EOC) were adjusted in the nuclear code to obtain moderator density coefficient curves that are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to this coefficient.

The Doppler reactivity coefficeint is determined as a function of fuel temperature using a two-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The resulting coefficient is conservative compared to design predictions for this plant. The weighting factor will increase under accident conditions as discussed above. The transient weighting factor used in the analysis is presented in Table 14.3-6.

14.3.3.2.2.4 Delayed Neutron Fraction, Beff. The accident is sensitive to Beff if the ejected-rod worth is nearly equal to or greater than Beff, as in zero-power transients. To allow for future fuel cycles, conservative estimates of Beff of 0.54% at beginning of cycle and 0.43% at end of cycle were used in the analysis.

14.3.3.2.2.5 Trip Reactivity Insertion. The trip reactivity insertion is assumed to be 4% from 102% of 2546 MWt or 100.38% of 2587 MWt and 1.77% from hot zero power, including the effect of one stuck rod, (i.e., the ejected rod). The shutdown reactivity is simulated by a conservative curve of trip reactivity insertion versus time after trip. The start of the rod motion occurs 0.5 second after the high-neutron-flux point is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 second for the coil to release the rods. The analyses presented are applicable for a rod insertion time of 2.4 second from coil release to entrance of the rod at the dash pot, although measurements indicate that this value should be closer to 1.8 second. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly significant conservatism for hot full-power accidents.

14.3.3.2.3 Results

The value of parameters used in the analysis, as well as the results of the most recent analysis are presented in Table 14.3-6 and discussed below.

14.3.3.2.3.1 Beginning of Cycle, Full Power. Control bank D was assumed to be inserted to its insertion limit. The ejected-rod worth, hot-channel factor, maximum fuel pellet average temperature, maximum fuel center temperature, maximum clad temperature, maximum fuel stored energy and percent of fuel melt are presented in Table 14.3-6.

14.3.3.2.3.2 Beginning of Cycle, Zero Power. For this condition, control bank D was assumed to be fully inserted and control bank C was at its insertion limit. The ejected-rod worth, hot-channel factor, maximum fuel pellet average temperature, maximum fuel center temperature, maximum clad temperature, maximum fuel stored energy and percent of fuel melt are presented in Table 14.3-6.

14.3.3.2.3.3 End of Cycle, Full Power. Control bank D was assumed to be inserted to its insertion limit. The ejected-rod worth, hot-channel factor, maximum fuel pellet average temperature, maximum fuel center temperature, maximum clad temperature, maximum fuel stored energy and percent of fuel melt are presented in Table 14.3-6.

14.3.3.2.3.4 End of Cycle, Zero Power. For this condition, control bank D was assumed to be fully inserted and control bank C was at its insertion limit. The ejected-rod worth, hot-channel factor, maximum fuel pellet average temperature, maximum fuel center temperature, maximum clad temperature, maximum fuel stored energy and percent of fuel melt are presented in Table 14.3-6.

A summary of the cases presented above is given in Table 14.3-6. The nuclear power and hot-spot fuel and clad temperature transients for the BOC full-power and zero-power cases are presented in Figures 14.3-24 through 14.3-27.

14.3.3.2.3.5 Fission Product Release. It is assumed that fission products are released from the gaps of all fuel rods entering DNB. Fission product release fractions, which show the expected fraction of the core inventory that is released to the gap, are summarized in Appendix 14A. These fractions are based on the steady-state fuel temperatures expected at full-power operation, and they include the effect of high fuel temperatures at the hot spot, using the design hot-channel factor. As a result of the rod ejection accident, the hot-spot fuel temperatures will increase, leading to an increase in the fraction of activity released to the gap. However, the results of the rod ejection analysis showed that even at the hot spot there is limited metal-water reaction and the clad is not expected to fail. Even if the rods entering DNB were to fail, only a small portion of the core is affected because of the strong localized peak typical of rod ejection accidents. For example, a fuel census performed from the results of a static, three-dimensional ejected-rod calculation (Figure 14.3-28) shows that, for this typical case, 90% of the fuel volume is operating at a power level less than half that at the hot spot. For this reason, less than 10% of the core enters DNB and a much smaller fraction will experience a fuel temperature nearly as high as that of the hot spot. Since there will be no massive failure of the fuel rods, the position with regard to fission product release, even taking into account the increased fuel temperatures in the area of the rod ejection, is that less than 10% of the core will release fission products. A gap-type release is expected. However, even assuming that a TID-14844-type release (100% of the noble gases and 50% of the halogens) occurs in less than 10% of the core, this release is much less than that for the double-ended severance of a reactor coolant pipe, in which 100% of the total core noble gases and 50% of the total core halogens are assumed to be released.

14.3.3.2.3.6 Pressure Surge. It is shown that there is no danger of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant. The most severe excess addition of energy to the coolant occurs for the high-power and end-of-life case. In order to estimate the magnitude of this pressure transient, average channel and hot-spot heat transfer calculations were performed using a high gap conductance and without assuming DNB. The power curves used for these calculations

represented a limiting case in which center melting was initiated at the hot spot. Using these heat flux data, a THINC 3 run was conducted to determine the volume surge without the benefit of pressure feedback. This volume surge was subsequently used as the basis-for a pressure calculation. The results indicated that, starting at 2250 psia, a peak pressure of about 2340 psia occurs some 1.5 seconds after a control-rod assembly ejection.

14.3.3.2.3.7 Lattice Deformation. A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods any produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot-spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot-spot region would produce a reduction in the total core moderator-to-fuel ratio and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

14.3.3.3 **Conclusions**

Even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the primary loop. The analyses have demonstrated that the upper limit in fission product release as a result of a number of fuel rods entering departure from nucleate boiling amounts to 10%.

14.3 REFERENCES

1. M. A. Styrikovich et al., *Transfer of Iodine from Aqueous Solutions to Saturated Vapor, Atommaya Energiya*, Vol. 17, No. 1, pp. 45-49, 1964.
2. L. S. Tong, *Boiling Heat Transfer and Two Phase Flow.*
3. Moody, F. J. *Maximum Flow Rate of a Single Component, Two-Phase Mixture,* Journal of Heat Transfer, 93 pp. 179-187, 1965.
4. Letter from W. L. Stewart (VEPCO), to H. R. Denton (NRC), Subject: Supplement 2 to a Request for an Amendment to Operating Licenses DPR-32 and DPR-37, Proposed Reduction in Boron Concentrations, Surry Power Station Units 1 and 2, dated December 19, 1983 (Serial No. 706).

5. Letter from W. L. Stewart (VEPCO) to H. R. Denton (NRC), Subject: Supplement to a Request for an Amendment to Operating Licenses DPR-32 and DPR-37, Proposed reduction in Boron Concentrations, Surry Power Station Units 1 and 2, dated November 30, 1983 (Serial No. 521B).

6. R. J. French et al., Indian Point Unit No. 2 Rod Ejection Analysis, WCAP-2940, 1966.

7. Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,* EPA 520/1-88-020, U.S. Environmental Protection Agency, 1988.

8. R. F. Barry, *The Revised LEOPARD Code - A Spectrum Dependent Non-Spatial Depletion Program, WCAP-2759*, 1965.

9. Letter from C. M. Stallings, Vepco, to E. G. Case, NRC, Subject: Surry 1 - Cycle 5 Reanalyses, dated March 15, 1978 (Serial No. 108).

10. Westinghouse Electric Corporation, *An Evaluation of the Rod Ejection Accident in Westinghouse PWRs Using; Spatial Kinetics Methods. WCAP-7588*, Rev. 1-A, 1975.

11. F. W. Sliz & K. L. Basehore, *VEPCO Reactor Core Thermal-Hydraulic Analysis Using the COBRA III C/MIT Computer Codes,* VEP-FRD-33A, October 1983.

12. *VEPCO Reactor System Transient Analysis Using RETRAN Computer Code*, VEP-FRD-41, Rev. 0.2-A, March 2015.

13. T. W. T. Burnett, *Reactor Protection System Diversity in Westinghouse Pressurized Water Reactor*, WCAP-7306, April 1969.

14. T. G. Taxelius, ed., *Annual Report - Spert Project. October 1968 and September 1969,* Idaho Nuclear Corporation IN-1370, June 1970.

15. R. C. Liimatainen and F. J. Testa, *Studies in TREAT of Zircaloy-2-Clad, UO^2-Core Simulated Fuel Elements,* ANL-7225, January-June 1966, p. 177, November 1966.

16. Letter from C. M. Stallings (VEPCO) to R. A. Purple (NRC), Serial No. 460, dated March 26, 1975, updating Letter from C. M. Stallings (VEPCO) to K. R. Goller (NRC), Serial No. 369, dated December 18, 1974.

17. Letter from C. M. Stallings (VEPCO) to K. R. Goller (NRC), Subject: Surry 2, Cycle 2 Reload Reanalysis, dated March 12, 1975 (Serial No. 458).

18. Letter from C. M. Stallings (VEPCO} to S. R. Goller (NRC), Subject: Positive Moderator Reanalysis, dated June 5, 1975, (Serial No. 553).

19. Letter from C. M. Stallings (VEPCO) to K. R. Goller (NRC), Subject: Surry 1, Cycle 3 Reload Reanalysis, dated September 8, 1975 (Serial No. 686).

20. Letter from C. M. Stallings (VEPCO) to B. C. Rusche (NRC), Subject: Surry 2, Cycle 3 Reload Reanalysis, dated March 11, 1976 (Serial No. 936).

21. Letter from W. L. Stewart, VEPCO, to U.S. Nuclear Regulatory Commission, Subject: Proposed Technical Specifications Change Surry Improved Fuel Assembly, dated May 26, 1987 (Serial No. 87-188).

22. J. A. Redfield, *CHIC-KIN - A FORTRAN Program for Intermediate and Fast Transients in a Water Moderated Reactor*, WAPD-TM-479, 1965.

23. A. A. Bishop, R. O. Sandberg, and L. S. Tong, *Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux,* ASME 65-HT-31, August 1965.

24. Westinghouse Electric Corporation, *Fuel Densification, Surry Units 1 and 2, Low Pressure Analysis*, WCAP-8117, 1973.

25. J. G. Miller and J. O. Erb, *Vepco Evaluation of the Control Rod Ejection Transient*, VEP-NFE-2-A, December 1984 and letter from C. O. Thomas (NRC) to W. L. Stewart (Vepco) dated September 26, 1984 Re: Acceptance for Referencing of License.

26. Letter from Chandu P. Patel, NRC, to W. L. Stewart, VEPCO, Subject: Issuance of Amendments (TAC Nos. 66359 and 66360), dated November 17, 1987.

27. P. J. Larouere, *Reanalysis of Surry Power Station UFSAR Chapter 14 Rod Ejection Transient,* Technical Report NE-635, May 1988.

28. S. L. Davidson (Ed.) et al, *VANTAGE+ Fuel Assembly Reference Core Report*, WCAP-12610-P-A (Proprietary), April 1995.

29. *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, Theoretical Manual, Revision 10, August 2004, Bechtel Power Corporation, San Francisco, CA.

30. *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, User's Manual, Revision 10, August 2004, Bechtel Power Corporation, San Francisco, CA.

31. *LOCADOSE NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation*, Validation Manual, Revision 12, August 2004, Bechtel Power Corporation, San Francisco, CA.

32. U.S. Nuclear Regulatory Commission, Office of Nuclear Research, Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,* July 2000.

33. Regulatory Guide 1.4, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactor*, Rev. 2, June 1974.

34. K. G. Murphy and K. M. Campe, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19*, 13th AEC Air Cleaning Conference, August 1974.

35. U.S. Nuclear Regulatory Commission, Office of Standards Development, *Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants*, Regulatory Guide 1.52, Revision 2, March 1978.

36. Letter from W. L. Stewart (Virginia Power) to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Control Room Habitability, Operator Dose Assessment*, Serial Number 89-381, June 1989.

37. Letter from C. M. Stalling (Virginia Power) to U.S. Nuclear Regulatory Commission, *Response to Request for Additional Information*, Serial Number 045A/020177, May 1977.

38. Letter from D. S. Cruden (Virginia Power) to NRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Safety Evaluation of Steam Generator Tube Rupture Accidents*, Serial No. 88-229, May 5, 1988.

39. Letter from L. A. Walsh (Westinghouse Owner's Group Steam Generator Tube Uncovery Task Team) to R. C. Jones, NRC, *Westinghouse Owner's Group Steam Generator Tube Uncovery Issue*, OG-92-25, March 31, 1992.

40. WCAP-13132, *The Effect of Steam Generator Tube Bundle Uncovery on Radioiodine Release*, January 1992.

41. Federal Guidance Report No. 12, *External Exposures to Radionuclides in Air, Water and Soil,* EPA 420-R-93-081, U.S. Environmental Protection Agency, 1993.

42. DOM-NAF-2-A, *Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code*.

43. H. H. Shah and P. Schueren, *Optimized ZIRLO™,* WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, July 2006.

Table 14.3-1
TIME SEQUENCE OF EVENTS FOR MAJOR SECONDARY STEAM PIPE RUPTURE
1.4 FT^2 BREAK WITH OFFSITE POWER

Event	Time, sec
Steamline rupture	1.01
High Steamline ΔP	1.16
High Steam Flow	1.58
Pressurizer Empties	11.80
Lo-Lo T_{avg}	13.86
Main Feedwater Isolation	15.07
Safety Injection Initiation	15.87
Main Steamline Isolation	18.86
Critically Reached	29.80
Boron Enters Core	243.6
Peak Heat Flux Reached	244.8

Table 14.3-2
TIME SEQUENCE OF EVENTS FOR MAJOR SECONDARY STEAM PIPE RUPTURE
1.4 FT^2 BREAK WITHOUT OFFSITE POWER

Event	Time, sec
Steamline rupture	1.01
High Steamline ΔP	1.16
High Steam Flow	1.58
Pressurizer Empties	13.80
Main Feedwater Isolation	14.96
Lo-Lo T_{avg}	16.86
Lo Steamline Pressure	19.18
Main Steamline Isolation	21.86
Safety Injection Initiation	25.08
Critically Reached	45.0
Boron Enters Core	253.6
Peak Heat Flux Reached	267.2

Table 14.3-3
TIME SEQUENCE OF EVENTS FOR MAJOR SECONDARY STEAM PIPE RUPTURE CREDIBLE BREAK

Event	Time, sec
Steamline rupture	1.01
Main Feedwater Isolation	8.96
Pressurizer Empties	26.40
Lo-Lo T_{avg}	31.91
Lo-Lo Pressurizer Pressure	68.79
Safety Injection Initiation	73.79
Critically Reached	207.6
Peak Heat Flux	316.8
Boron Enters Core	328.8

Table 14.3-4
SURRY MAIN STEAMLINE BREAK ANALYSIS
1.4 FT^2 BREAK WITH OFFSITE POWER AND CREDIBLE BREAK CASE

Statepoint		1	2	3	4	5 [a]
Time, sec		86.0	174.0	244.8	298.0	316.8
Loop A Cold Leg Temperature, °F		400.7	396.88	396.74	395.86	459.02
Loop B Cold Leg Temperature, °F		471.06	466.76	465.42	463.94	482.33
Loop C Cold Leg Temperature, °F		471.59	466.92	465.44	463.94	482.33
Pressurizer Pressure, psia		820.40	842.60	869.81	882.98	1006.93
Volumetric RCS Flow, % nominal		99.68	99.67	99.67	99.67	99.87
Heat Flux,	% of 2546 MWt	18.33	22.61	23.74	21.90	7.07
	% of 2587 MWt	18.04	22.25	23.36	21.55	6.46

a. Credible Break Statepoint

Table 14.3-5
SURRY MAIN STEAMLINE BREAK ANALYSIS
1.4 FT^2 BREAK WITHOUT OFFSITE POWER

Statepoint		6	7	8	9	10
Time, sec		246.0	267.2	338.0	382.0	470.8
Loop A Cold Leg Temperature, °F		291.32	285.48	271.34	265.74	257.31
Loop B Cold Leg Temperature, °F		467.87	466.18	462.68	460.93	457.27
Loop C Cold Leg Temperature, °F		473.47	471.36	466.86	464.63	460.04
Pressurizer Pressure, psia		848.23	852.61	858.85	862.42	881.58
Volumetric RCS Flow, % nominal		6.12	6.08	5.70	5.44	5.22
Heat Flux,	% of 2546 MWt	5.69	5.94	5.15	5.01	5.01
	% of 2587 MWt	5.60	5.85	5.07	4.93	4.93

Table 14.3-6
CONTROL ROD ASSEMBLY EJECTION DATA

	Time In Cycle			
Parameter	Beginning	Beginning	End	End
Power Level (% of 2587 MWt)	100.38%	0%	100.38%	0%
Ejected Rod Worth, %Δk/k	0.215	0.830	0.215	0.830
Delayed Neutron Fraction, %	0.54	0.54	0.43	0.43
Feedback Reactivity Weighting	1.325	3.258	1.310	3.258
Trip Reactivity, %Δk/k	4.0	1.77	4.0	1.77
Fq Before Rod Ejection	2.508	-	2.508	-
Fq After Rod Ejection	6.60	20.0	6.50	20.0
Number of Operational Pumps	3	2	3	2
Maximum Fuel Pellet Average Temperature, °F	4080	3470	3991	3533
Maximum Fuel Center Temperature, °F	4903	4015	4804	4076
Maximum Clad Temperature, °F	2681	2593	2602	2641
Maximum Fuel Stored Energy, cal/gm	189.2	148.1	183.5	151.5
Percent Fuel Melting	<4	0	<4	0

Table 14.3-7
STEAM GENERATOR TUBE RUPTURE BREAK FLOW RATES AND RELEASES
OFFSITE POWER UNAVAILABLE
(All flow rates are in cubic feet per minute)

From Primary Coolant to Unaffected SG Liquid	0.1337 cfm (1 gpm)
From Unaffected Steam Generator Steam to Environment	1 cfm

Time (hr)	RCS to Affected SG Liquid	RCS to Affected SG Steam	Affected SG Liquid to Steam[a]	Affected SG Steam to Environment
0 – 0.0222	105	15.2	1330	0
0.0222 – 0.0625	104	5.48	168	4673
0.0625 – 0.5	85.2	6.61	96.9	2701

Time (hr)	Unaffected SG Liquid to Environment[a]			
0 – 0.0228	0			
0.0228 – 0.0625	179			
0.0625 – 0.1186	66.0			
0.1186 – 0.5	0			
0.5 – 2	87			
2 – 8	33			

a. Partitioning and Moisture Carryover are modeled in the iodine and particulate releases by decreasing these flow rates by 100.

Table 14.3-8
STEAM GENERATOR TUBE RUPTURE BREAK FLOW RATES AND RELEASES
OFFSITE POWER AVAILABLE
(All flow rates are in cubic feet per minute)

From Primary Coolant to Unaffected SG Liquid	0.1337 cfm (1 gpm)
From Unaffected Steam Generator Steam to Environment	1 cfm

Time (hr)	RCS to Affected SG Liquid	RCS to Affected SG Steam	Affected SG Liquid to Steam[a]	Affected SG Steam to Environment
0 – 0.0731	91.4	9.83	1330	0
0.0731 - 0.1014	78.8	6.39	182	5088
0.1014 - 0.5	72.2	0.84	127	3541

Time (hr)	Unaffected SG Liquid to Environment[a]			
0 - 0.0947	0			
0.0947 – 0.1014	746			
0.1014 - 0.1497	173			
0.1497 - 0.5	0			
0.5 - 2	97			
2 – 8	49			

a. Partitioning and Moisture Carryover are modeled in the iodine and particulate releases by decreasing these flow rates by 100.

Table 14.3-9
STEAM GENERATOR TUBE RUPTURE CONTROL ROOM AND OFFSITE DOSES

		Control Room 30-day (Rem TEDE)	EAB worst 2-hour (Rem TEDE)	LPZ 30-day (Rem TEDE)
Concurrent Spike	Dose Consequences	1.3	1.7	0.2
	GDC 19 and RG 1.183 dose limits	5	2.5	2.5
Pre-Accident Spike	Dose Consequences	4.3	1.2	0.2
	GDC 19, RG 1.183 and 10 CFR 50.67 dose limits	5	25	25

Table 14.3-10
PRIMARY COOLANT AND SECONDARY SIDE RADIONUCLIDE INVENTORIES
TECHNICAL SPECIFICATION LIMITS FOR DOSE EQUIVALENT I-131

Isotope	Primary Concentration for 1 μCi/gm DE I-131 (μCi/gm)	Primary Activity (Ci)	Unaffected SG Liquid Activity (Ci)	Affected SG Liquid Activity (Ci)	Unaffected SG Steam Activity (Ci)	Affected SG Steam Activity (Ci)
Kr-83m	1.56E-01	2.85E+01			2.99E-05	1.49E-05
Kr-85m	5.70E-01	1.04E+02			1.09E-04	5.46E-05
Kr-85	2.31E+00	4.22E+02			4.42E-04	2.21E-04
Kr-87	3.64E-01	6.65E+01			6.97E-05	3.48E-05
Kr-88	1.03E+00	1.88E+02			1.97E-04	9.86E-05
Kr-89	3.02E-02	5.51E+00			5.78E-06	2.89E-06
Xe-131m	1.03E+00	1.88E+02			1.97E-04	9.86E-05
Xe-133m	1.58E+00	2.88E+02			3.03E-04	1.51E-04
Xe-133	1.00E+02	1.83E+04			1.91E-02	9.57E-03
Xe-135m	3.60E-01	6.57E+01			6.89E-05	3.45E-05
Xe-135	3.78E+00	6.90E+02			7.24E-04	3.62E-04
Xe-137	7.65E-02	1.40E+01			1.46E-05	7.32E-06
Xe-138	2.60E-01	4.75E+01			4.98E-05	2.49E-05
I-130	1.94E-02	3.54E+00	1.73E-01	8.67E-02	1.18E-04	5.90E-05
I-131	7.45E-01	1.36E+02	6.66E+00	3.33E+00	4.53E-03	2.26E-03
I-132	3.76E-01	6.86E+01	3.36E+00	1.68E+00	2.29E-03	1.14E-03
I-133	1.23E+00	2.25E+02	1.10E+01	5.50E+00	7.48E-03	3.74E-03
I-134	2.42E-01	4.42E+01	2.16E+00	1.08E+00	1.47E-03	7.35E-04
I-135	7.90E-01	1.44E+02	7.06E+00	3.53E+00	4.80E-03	2.40E-03
Cs-134m	1.94E-02	3.54E+00	1.73E-01	8.67E-02	1.18E-04	5.90E-05
Cs-134	1.35E+00	2.46E+02	1.21E+01	6.03E+00	8.21E-03	4.10E-03
Cs-136	2.69E-01	4.91E+01	2.40E+00	1.20E+00	1.63E-03	8.17E-04
Cs-137	8.67E-01	1.58E+02	7.75E+00	3.88E+00	5.27E-03	2.63E-03

Table 14.3-10 (CONTINUED)
PRIMARY COOLANT AND SECONDARY SIDE RADIONUCLIDE INVENTORIES
TECHNICAL SPECIFICATION LIMITS FOR DOSE EQUIVALENT I-131

Isotope	Primary Concentration for 1 μCi/gm DE I-131 (μCi/gm)	Primary Activity (Ci)	Unaffected SG Liquid Activity (Ci)	Affected SG Liquid Activity (Ci)	Unaffected SG Steam Activity (Ci)	Affected SG Steam Activity (Ci)
Cs-138	4.07E-01	7.43E+01	3.64E+00	1.82E+00	2.47E-03	1.24E-03
Cs-139	3.76E-02	6.86E+00	3.36E-01	1.68E-01	2.29E-04	1.14E-04
Ba-137m	8.15E-01	1.49E+02	7.29E+00	3.64E+00	4.95E-03	2.48E-03
Ba-139	3.04E-02	5.55E+00	2.72E-01	1.36E-01	1.85E-04	9.24E-05
Br-83	2.80E-02	5.11E+00	2.50E-01	1.25E-01	1.70E-04	8.51E-05
Br-84	1.40E-02	2.56E+00	1.25E-01	6.26E-02	8.51E-05	4.25E-05
Rb-86	1.22E-02	2.23E+00	1.09E-01	5.45E-02	7.42E-05	3.71E-05
Rb-88	1.06E+00	1.94E+02	9.48E+00	4.74E+00	6.44E-03	3.22E-03
Rb-89	6.21E-02	1.13E+01	5.55E-01	2.78E-01	3.77E-04	1.89E-04
Co-58	1.38E-02	2.52E+00	1.23E-01	6.17E-02	8.39E-05	4.19E-05
Tc-99m	4.47E-01	8.16E+01	4.00E+00	2.00E+00	2.72E-03	1.36E-03
Tc-101	8.32E-03	1.52E+00	7.44E-02	3.72E-02	5.06E-05	2.53E-05
Tc-102	6.25E-03	1.14E+00	5.59E-02	2.79E-02	3.80E-05	1.90E-05
Te-131m	7.14E-03	1.30E+00	6.38E-02	3.19E-02	4.34E-05	2.17E-05
Te-131	4.89E-03	8.93E-01	4.37E-02	2.19E-02	2.97E-05	1.49E-05
Te-132	7.78E-02	1.42E+01	6.96E-01	3.48E-01	4.73E-04	2.36E-04
Te-133m	6.14E-03	1.12E+00	5.49E-02	2.74E-02	3.73E-05	1.87E-05
Te-133	3.46E-03	6.32E-01	3.09E-02	1.55E-02	2.10E-05	1.05E-05
Te-134	1.10E-02	2.01E+00	9.83E-02	4.92E-02	6.69E-05	3.34E-05
Mo-99	1.06E+00	1.94E+02	9.48E+00	4.74E+00	6.44E-03	3.22E-03
Mo-101	8.66E-03	1.58E+00	7.74E-02	3.87E-02	5.26E-05	2.63E-05
Mo-102	6.25E-03	1.14E+00	5.59E-02	2.79E-02	3.80E-05	1.90E-05

Table 14.3-11
PRIMARY COOLANT TECHNICAL SPECIFICATION PRE-ACCIDENT SPIKE AND CONCURRENT IODINE SPIKE ACTIVITIES

Nuclide	10 µCi/gm DE I-131 Pre-Accident Iodine Spike Concentrations (µCi/gm)	10 µCi/gm DE I-131 Pre-Accident Iodine Spike Activity (Ci)	SGTR Concurrent Spike (Ci/hr)	MSLB Concurrent Spike (Ci/hr)
I-131	7.45	1.36E+03	7.550E+03	1.127E+04
I-132	3.76	6.86E+02	1.011E+04	1.508E+04
I-133	12.3	2.25E+03	1.447E+04	2.160E+04
I-134	2.42	4.42E-02	1.315E+04	1.962E+04
I-135	7.90	1.44E+03	1.254E+04	1.872E+04

Table 14.3-12
VOLUMES USED IN ANALYSIS OF MAIN STEAM LINE BREAK (MSLB), LOCKED ROTOR ACCIDENT (LRA), STEAM GENERATOR TUBE RUPTURE (SGTR)

Description	Volume (ft^3)	Notes
Environment		
Primary Coolant	8902	
Secondary Liquid	2052	Per steam generator
Secondary Steam [a], [b]	3889	Per steam generator
Control Room	2.23×10^5	
Turbine Building [c]	6.00×10^6	

a. The MSLB analysis modeled the steam compartments of all three steam generators as a single 1 ft^3 volume.
b. The SGTR analysis modeled the steam compartment of the two intact steam generators as a single 1 ft^3 volume
c. The Turbine Building volume is only used in the MSLB analysis.

Table 14.3-13

χ/Qs USED IN THE SGTR AND MSLB ANALYSES

Location	Time	χ/Q (sec/m^3)
Control Room	0 to 720 hours	3.79×10^{-3}
EAB	0 to 2 hours	1.76×10^{-3}
LPZ	0 to 8 hours	2.01×10^{-4}
LPZ	8 to 24 hours	1.22×10^{-4}
LPZ	24 to 96 hours	4.18×10^{-5}
LPZ	96 to 720 hours	8.94×10^{-6}

Table 14.3-14
FLOW FROM AFFECTED SG TO THE TURBINE BUILDING AND FROM THE TURBINE BUILDING TO THE ENVIRONMENT

Time Seconds	Volume Released (cfm)[a] SG to Turbine Building	Volume Released (cfm) Turbine Building to Environment
0 - 41	1.632×10^3	2.396×10^6
41 - 181	3.818×10^3	1.132×10^6
181 - 1800	2.511×10^3	4.096×10^5
>1800	0.0	0.0

a. Note that the release rates from the steam generator to the turbine building are increased above the values calculated by the thermal hydraulic analysis by a factor of 5 after 41 seconds, and by a factor of 10 after 181 seconds. This ensures that all of the radionuclides initially present in the Steam Generator are released.

Table 14.3-15
MAIN STEAM LINE BREAK CONTROL ROOM AND OFFSITE DOSES

		Control Room 30-day (Rem TEDE)	EAB worst 2-hour (Rem TEDE)	LPZ 30-day (Rem TEDE)
Concurrent Spike Cases	Dose Consequences	1.6	0.5	0.1
	GDC 19 and RG 1.183 dose limits	5	2.5	2.5
Pre-Accident Spike Cases	Dose Consequences	1.4	0.4	0.1
	GDC 19, RG 1.183 and 10 CFR 50.67 dose limits	5	25	25

Table 14.3-16
FISSION PRODUCT CONCENTRATIONS IN THE REACTOR COOLANT WITH SMALL CLADDING DEFECTS IN ONE PERCENT OF THE FUEL RODS
(2605 MWt, 18 month cycles, 574.4°F)

Isotope	Concentration µCi/gm
Kr-83m	3.06E-0l
Kr-85m	1.12E+00
Kr-85	4.52E+00
Kr-87	7.l2E-01
Kr-88	2.02E+00
Kr-89	5.9IE-02
Xe-131m	2.02E+00
Xe-133m	3.08E+00
Xe-133	1.96E+02
Xe-135m	7.04E-01
Xe-135	7.40E+00
Xe-137	1.50E-01
Xe-138	5.08E-01
Br-83	5.47E-02
Br-84	2.74E-02
Br-85	3.22E-03
Br-87	1.72E-03
I-129	8.8lE-08
I-130	3.80E-02
I-131	1.46E+00
I-132	7.36E-01
I-133	2.40E+00
I-134	4.74E-01

Isotope	Concentration µCi/gm
Y-9l	1.94E-03
Y-92	7.98E-04
Y-93	4.36E-04
Y-94	2.l6E-05
Y-95	9.25E-06
Zr-95	3.17E-04
Zr-97	2.31E-04
Nb-95m	2.24E-06
Nb-95	3.20E-04
Nb-97m	2.19E-04
Nb-97	2.46E-04
Mo-99	2.07E+00
Mo-101	1.69E-02
Mo102	1.22E-02
Mo105	5.32E-04
Tc-99m	8.74E-01
Tc-101	1.63E-02
Tc-l02	1.22E-02
Tc-l05	5.82E-04
Ru-103	2.86E-04
Ru-105	8.75E-05
Ru-l06	1.10E-04
Ru-107	1.43E-06
Rh-l03m	2.58E-04

Isotope	Concentration µCi/gm
Te-131	9.56E-03
Te-132	1.52E-01
Te-133m	1.20E-02
Te-133	6.76E-03
Te-134	2.15E-02
Cs-134m	3.79E-02
Cs-134	2.64E+00
Cs-136	5.27E-01
Cs-137	1.70E+00
Cs-138	7.95E-01
Cs-139	7.35E-02
Cs-140	7.47E-03
Cs-142	9.02E-05
Ba-137m	1.59E+00
Ba-139	5.94E-02
Ba-140	2.06E-03
Ba-14l	9.77E-05
Ba-142	1.38E-04
La-140	5.56E-04
La-14l	1.83E-04
La-l42	1.83E-04
La-143	1.12E-05
Ce-141	3.03E-04
Ce-143	2.43E-04

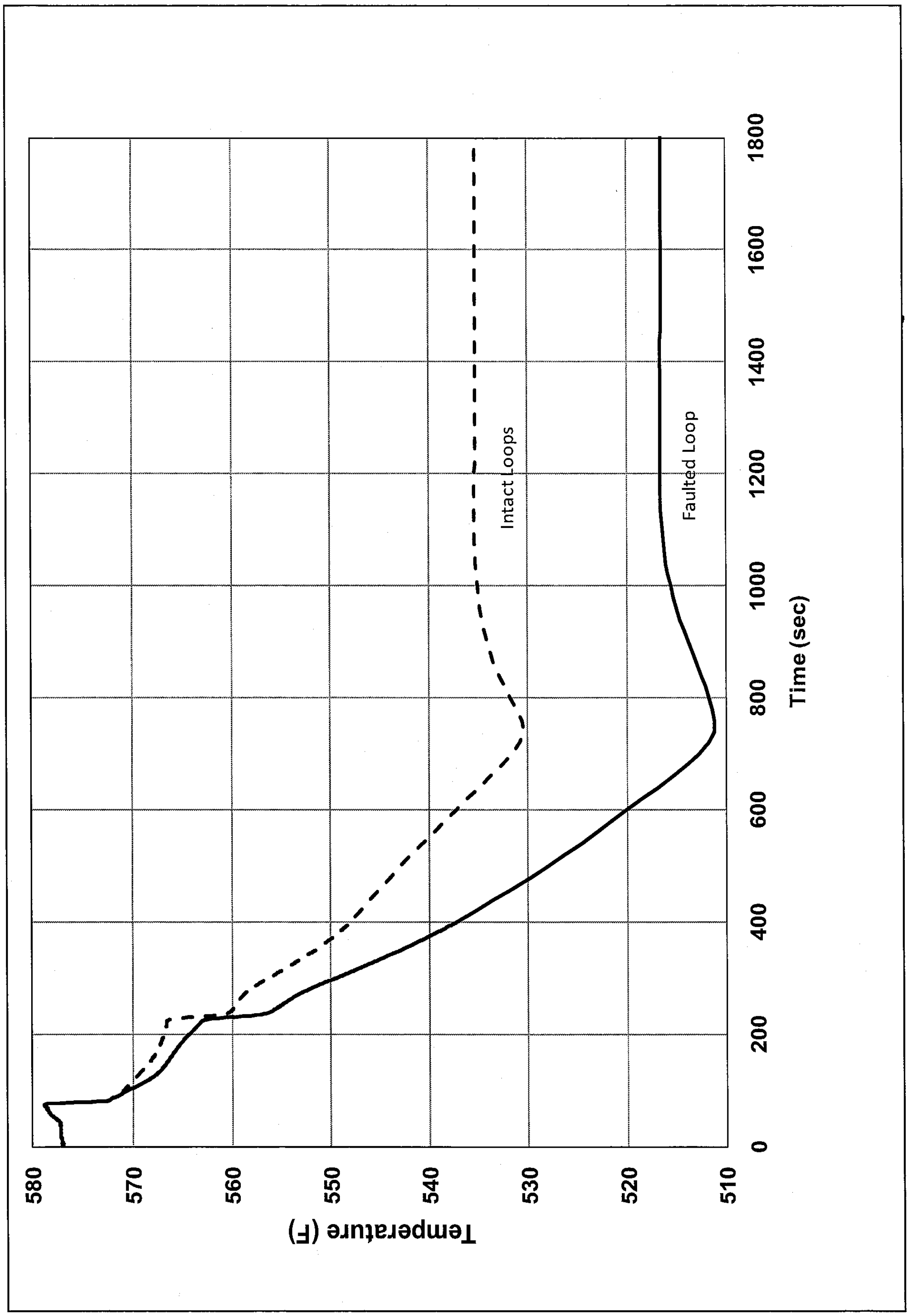
Table 14.3-16 (CONTINUED)
FISSION PRODUCT CONCENTRATIONS IN THE REACTOR COOLANT WITH SMALL CLADDING DEFECTS IN ONE PERCENT OF THE FUEL RODS
(2605 MWt, 18 month cycles, 574.4°F)

Isotope	Concentration µCi/gm
I-135	1.54E+00
I-136	5.56E-03
Se-8l	5.l4E-07
Se-83	5.53E-07
Se-84	3.66E-07
Rb-86	2.38E-02
Rb-88	2.07E+00
Rb-89	1.2lE-01
Rb-90	9.96E-03
Rb-9l	4.72E-03
Rb-92	3.18E-04
Sr-89	1.72E-03
Sr-90	1.10E-04
Sr-91	8.50E-04
Sr-92	7.06E-04
Sr-93	3.64E-05
Sr-94	5.89E-06
Y-90	1.37E-04

Isotope	Concentration µCi/gm
Rh-l05m	2.46E-05
Rh-l05	1.82E-04
Rh-l06	1.2lE-04
Rh-l07	8.39E-06
Sn-127	3.43E-06
Sn-128	4.48E-06
Sn-130	7.l2E-07
Sb-127	1.87E-05
Sb-128	1.64E-06
Sb-129	2.58E-05
Sb-130	2.05E-06
Sb-131	9.86E-06
Sb-132	7.54E-07
Sb-133	7.57E-07
Te-125m	4.l2E-04
Te-127m	1.68E-03
Te-127	8.35E-03
Te-129m	5.42E-03
Te-129	9.20E-03

Isotope	Concentration µCi/gm
Ce-144	2.53E-04
Ce-145	1.70E-06
Ce-146	6.22E-06
Pr-143	2.8IE-04
Pr-144	2.55E-04
Pr-145	1.00E-04
Pr-146	1.62E-05
Nd-147	1.19E-04
Nd-149	1.68E-05
Nd-151	1.30E-06
Pm-l47	5.65E-05
Pm-149	7.8IE-05
Pm-151	3.10E-05
Sm-151	3.12E-07
Sm-153	7.52E-05

Figure 14.3-1
STEAM GENERATOR TUBE RUPTURE - RCS AVERAGE TEMPERATURE



s1403001

s1403002

Figure 14.3-2
STEAM GENERATOR TUBE RUPTURE - REACTOR
(FRACTION OF 2546 MWt) POWER

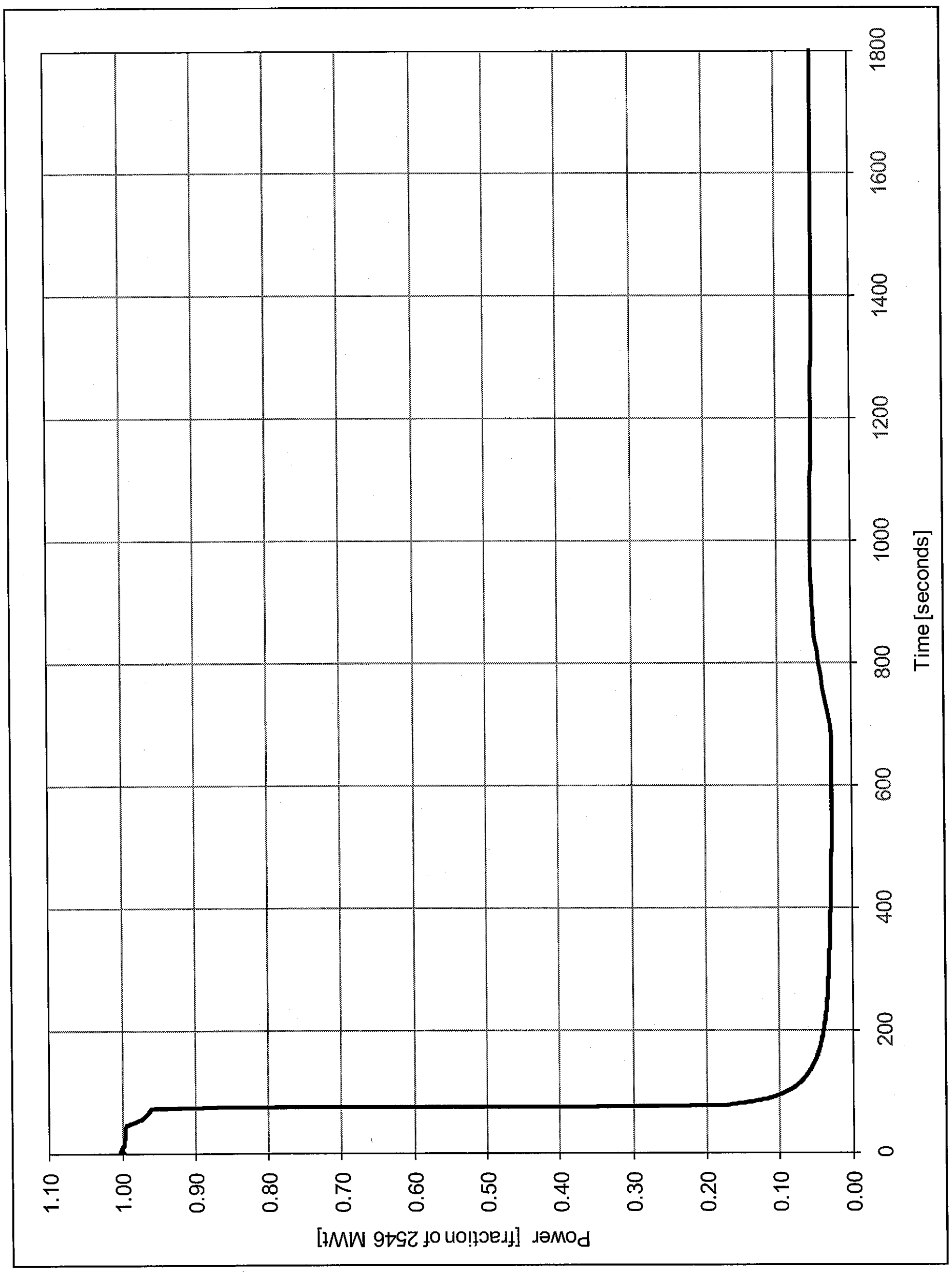
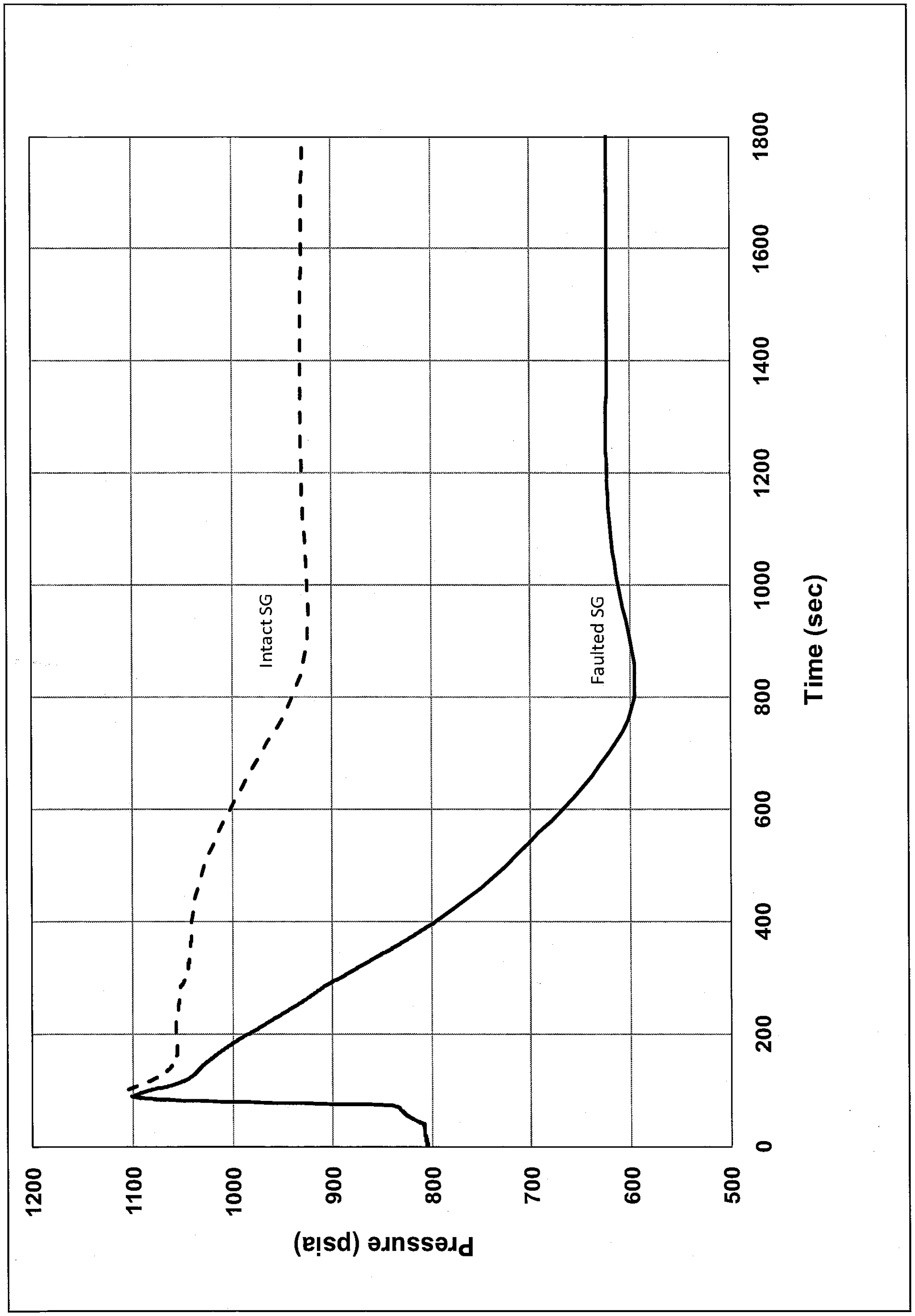


Figure 14.3-3
STEAM GENERATOR TUBE RUPTURE - STEAM GENERATOR PRESSURE



s1403003

s1403026

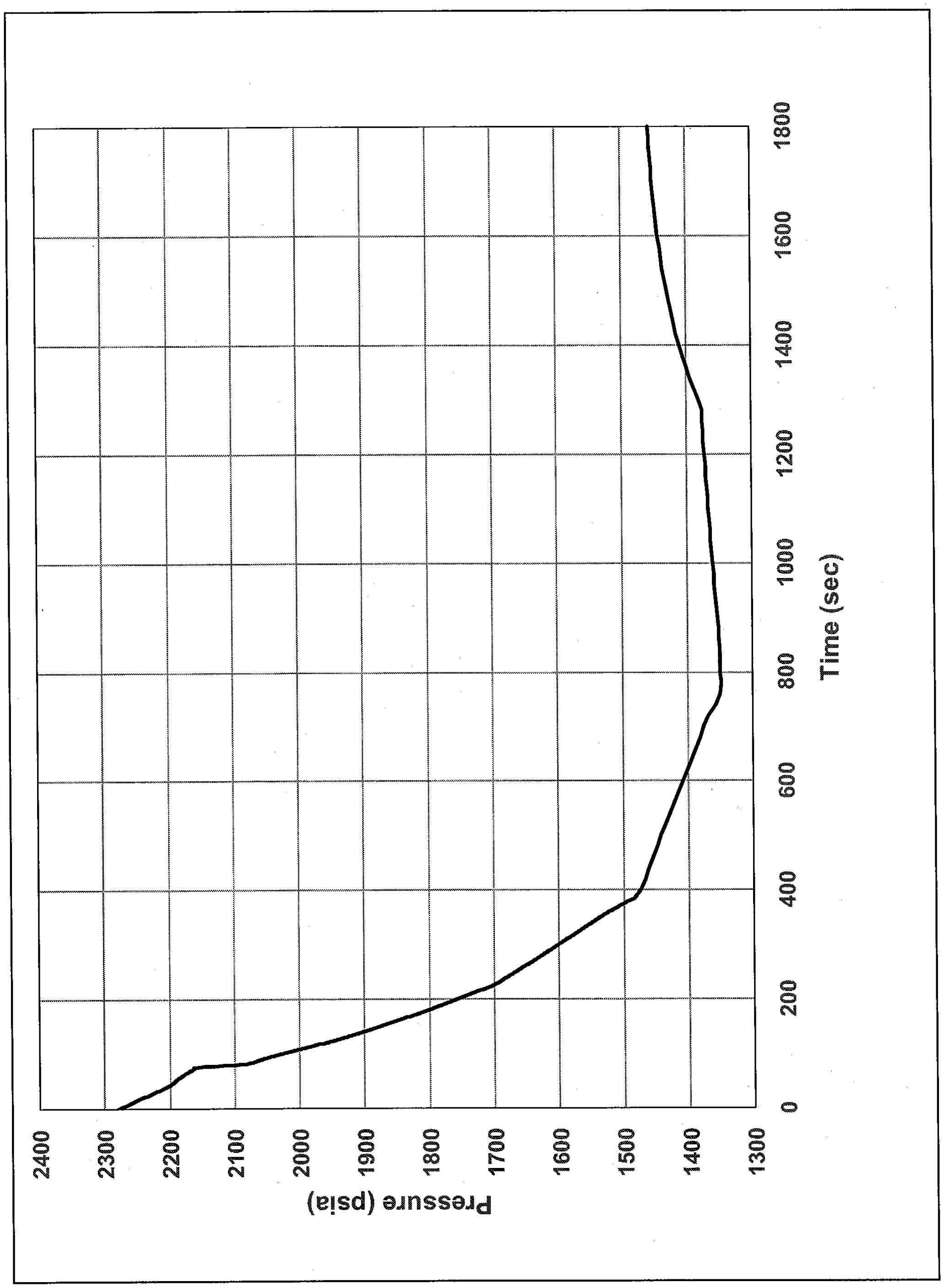
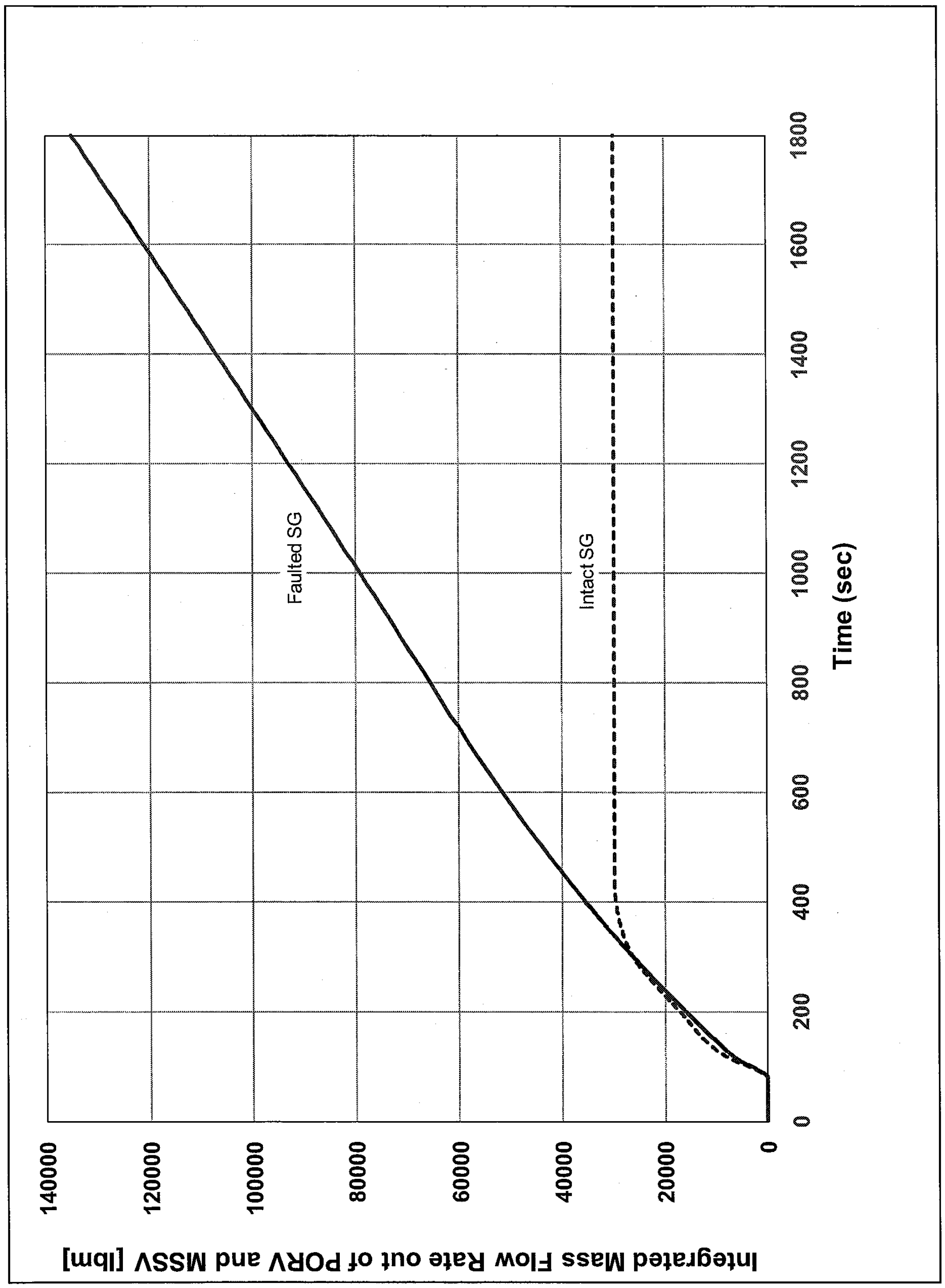


Figure 14.3-4
STEAM GENERATOR TUBE RUPTURE - PRESSURIZER PRESSURE

Figure 14.3-5
STEAM GENERATOR TUBE RUPTURE - OPEN PORV AND MSSV INTEGRATED MASS RELEASE



s1403027

s1403028

Figure 14.3-6
STEAM GENERATOR TUBE RUPTURE - BREAK MASS FLOW

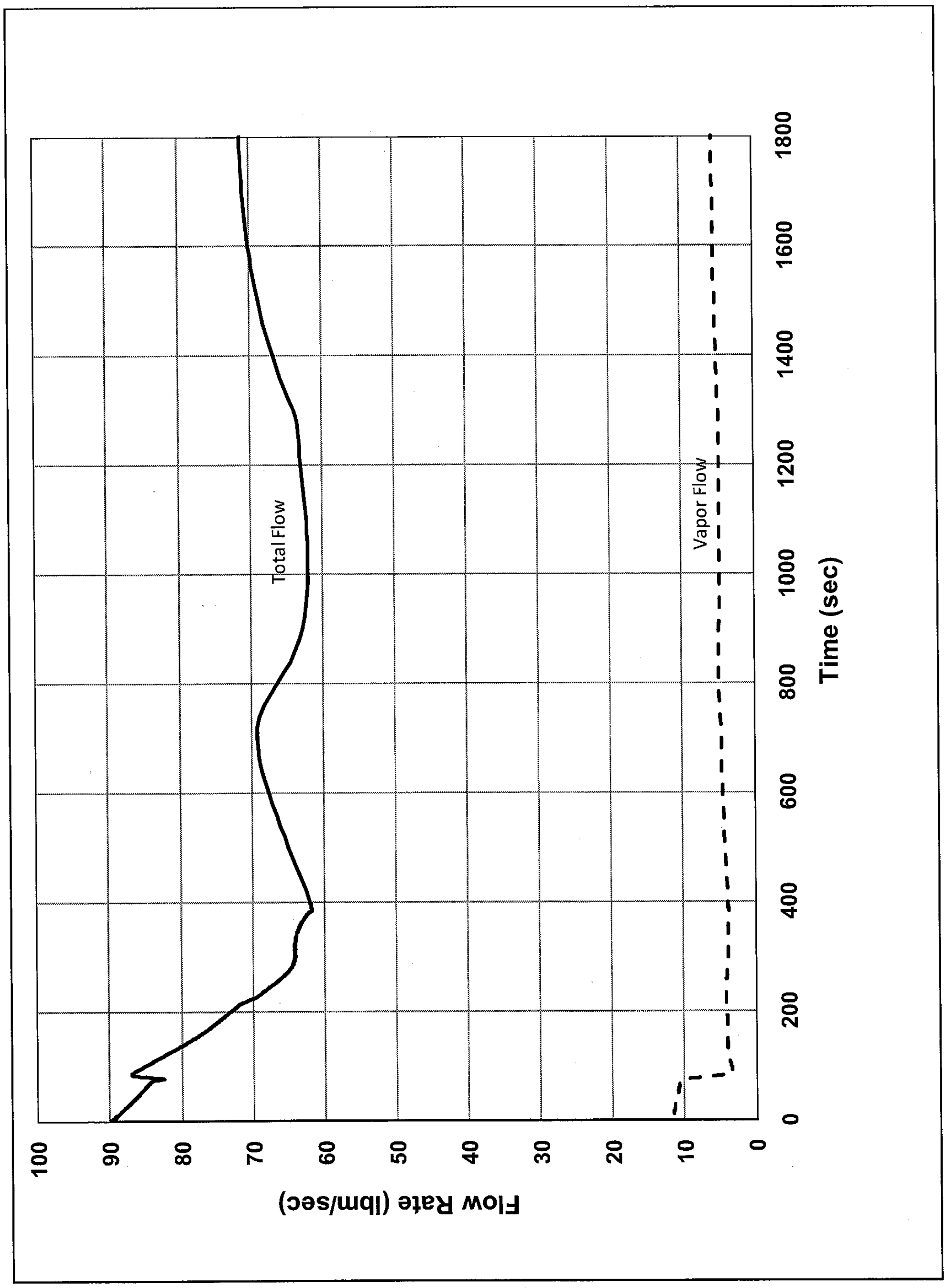
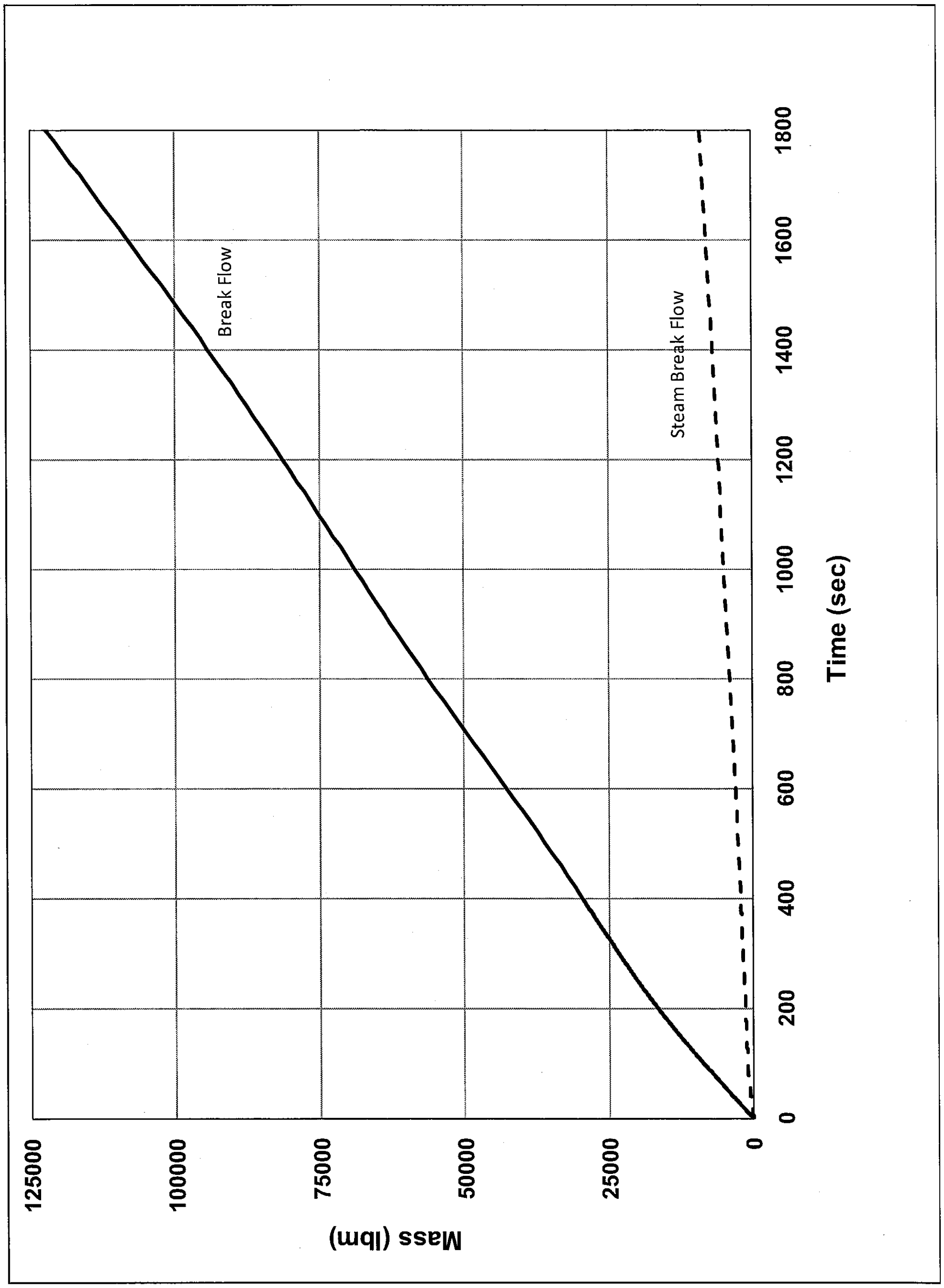


Figure 14.3-7
STEAM GENERATOR TUBE RUPTURE - BREAK INTEGRATED MASS FLOW



s1403029

Figure 14.3-8
VARIATION OF REACTIVITY WITH POWER

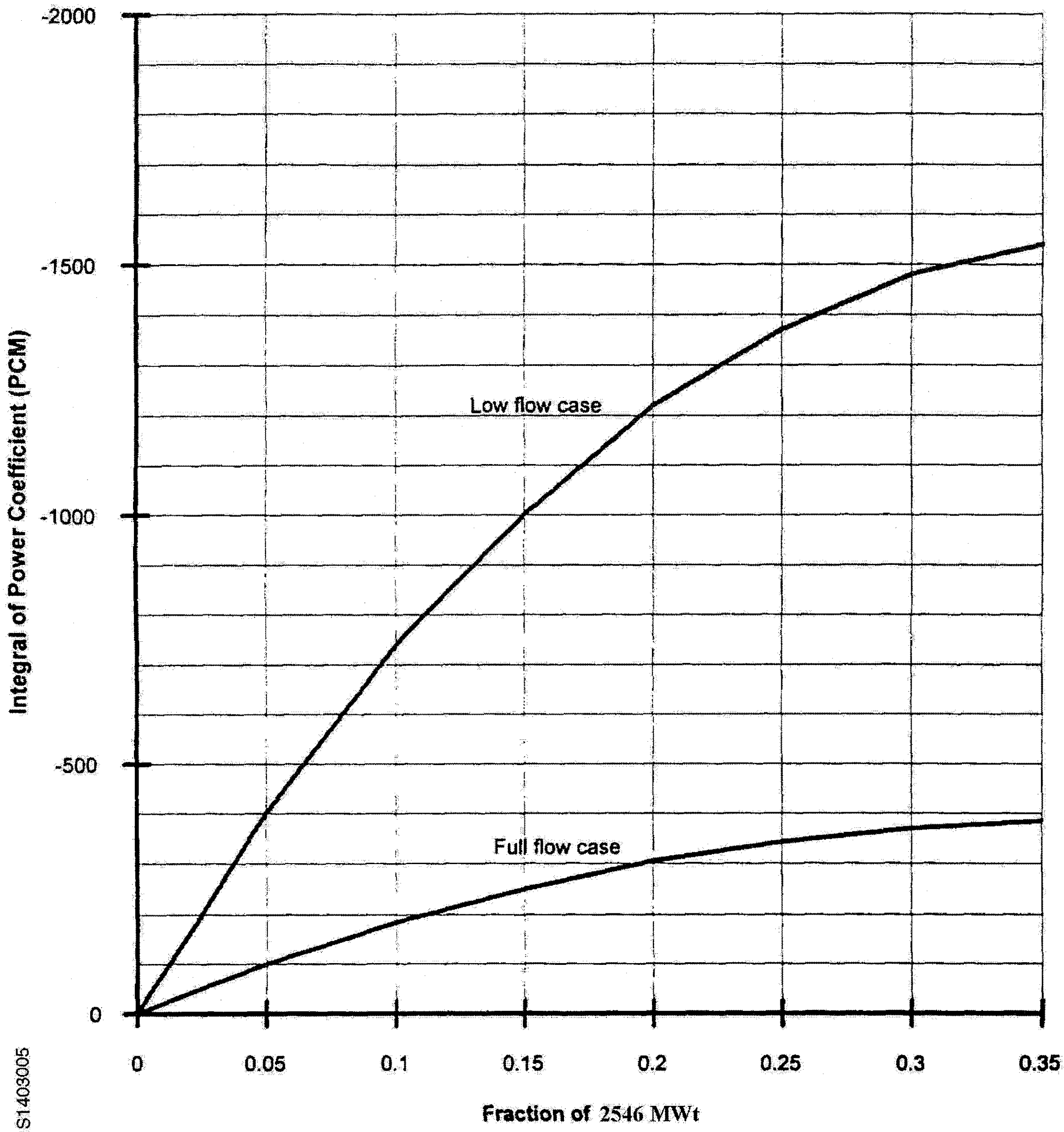


Figure 14.3-9
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK, OFFSITE POWER AVAILABLE NORMALIZED CORE HEAT FLUX (FRACTION OF 2546 MWt)

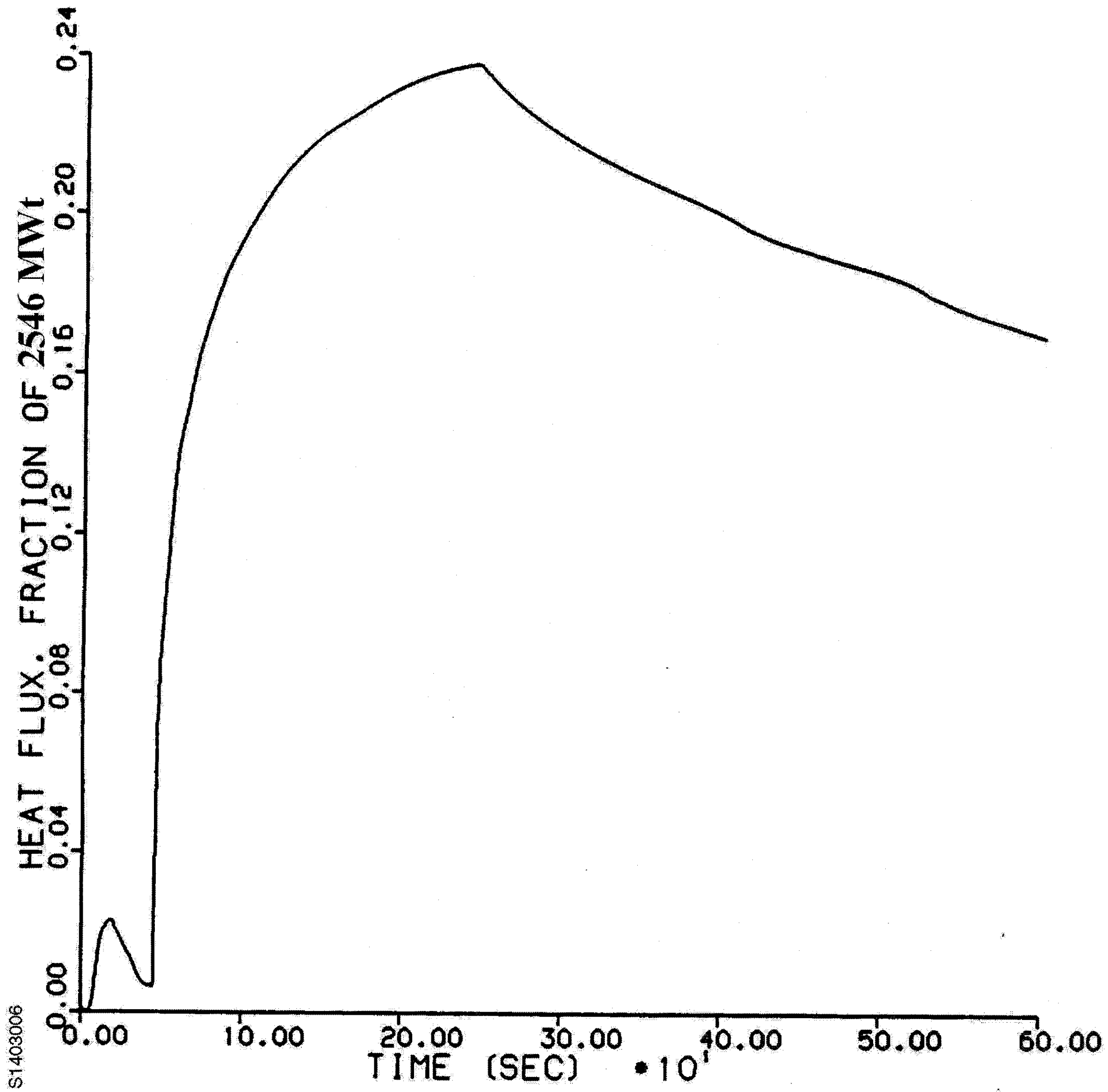


Figure 14.3-10
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
OFFSITE POWER AVAILABLE PRESSURIZER PRESSURE

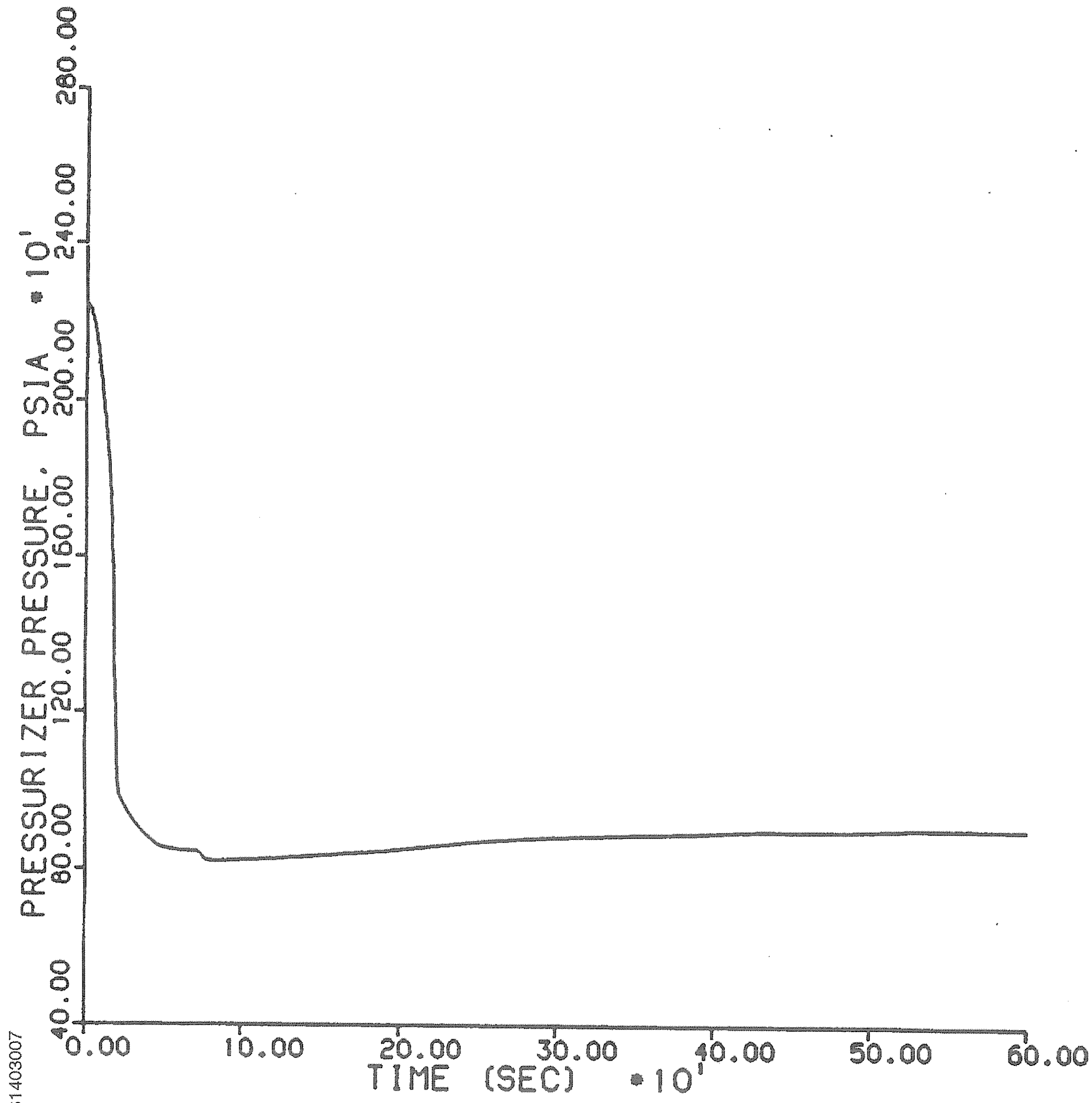


Figure 14.3-11
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
OFFSITE POWER AVAILABLE CORE REACTIVITY,% $\Delta K/K$

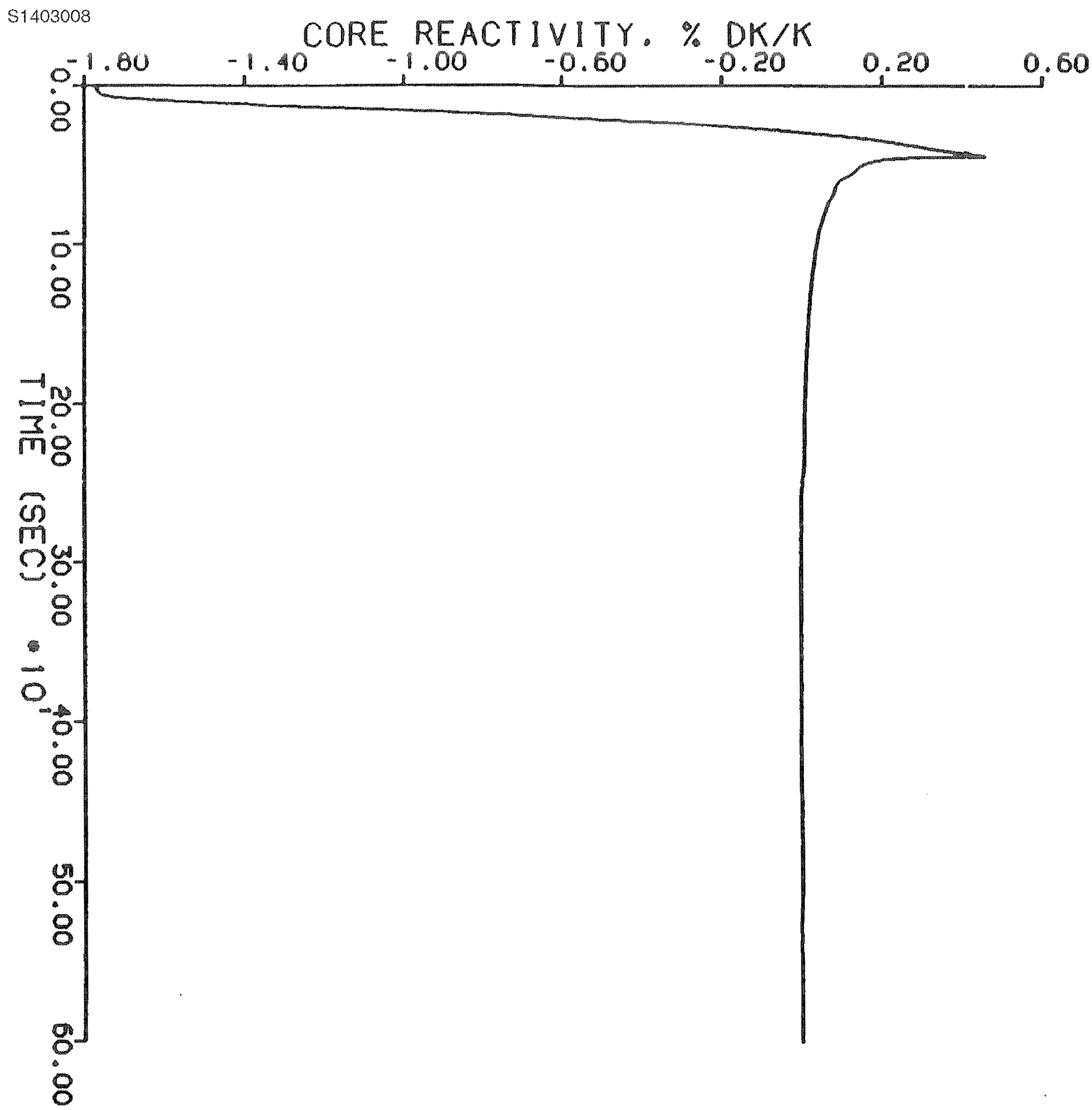


Figure 14.3-12
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
OFFSITE POWER AVAILABLE CORE INLET BORON CONCENTRATION

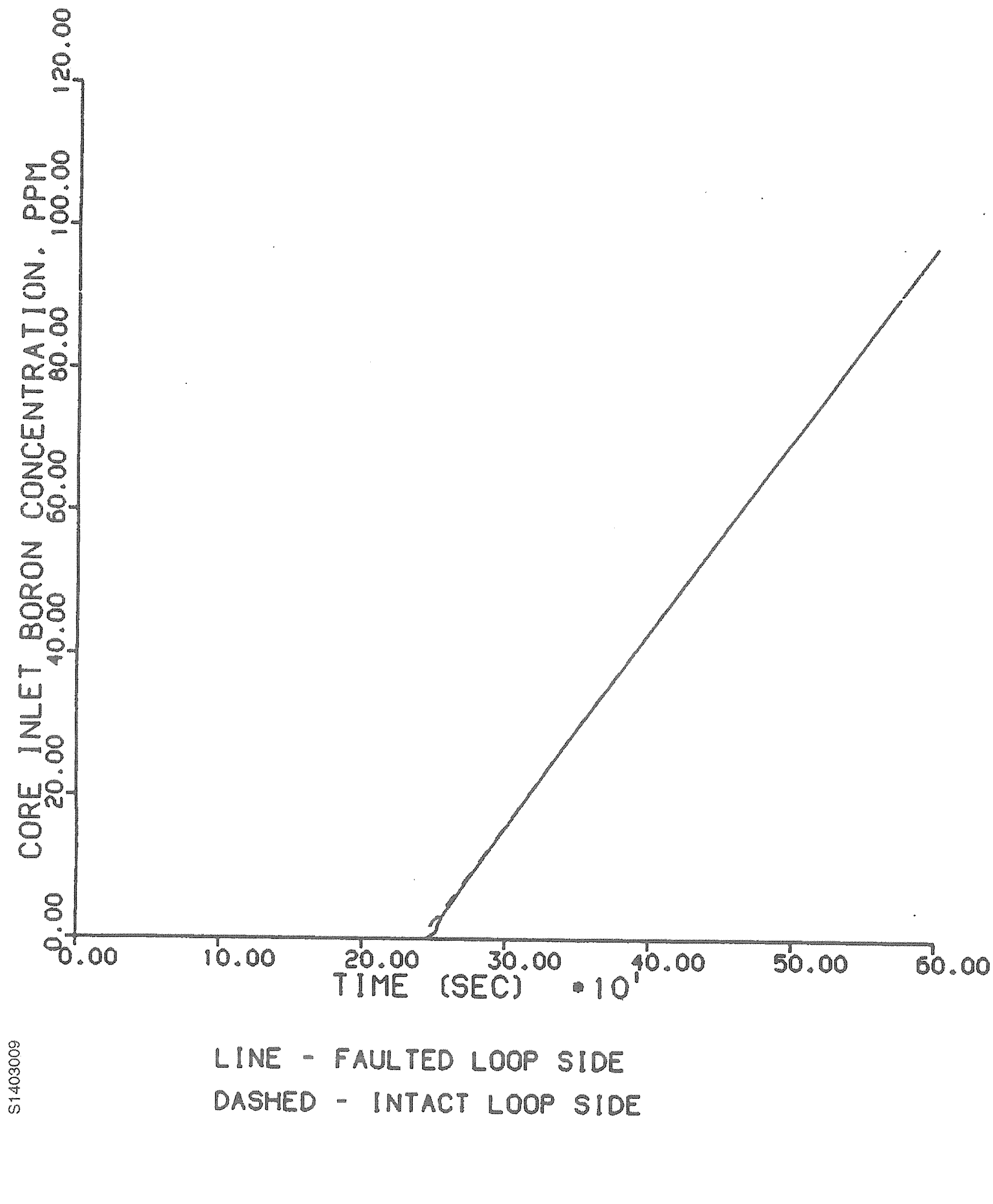


Figure 14.3-13
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
OFFSITE POWER AVAILABLE ACTUAL LOOP AVERAGE TEMPERATURES

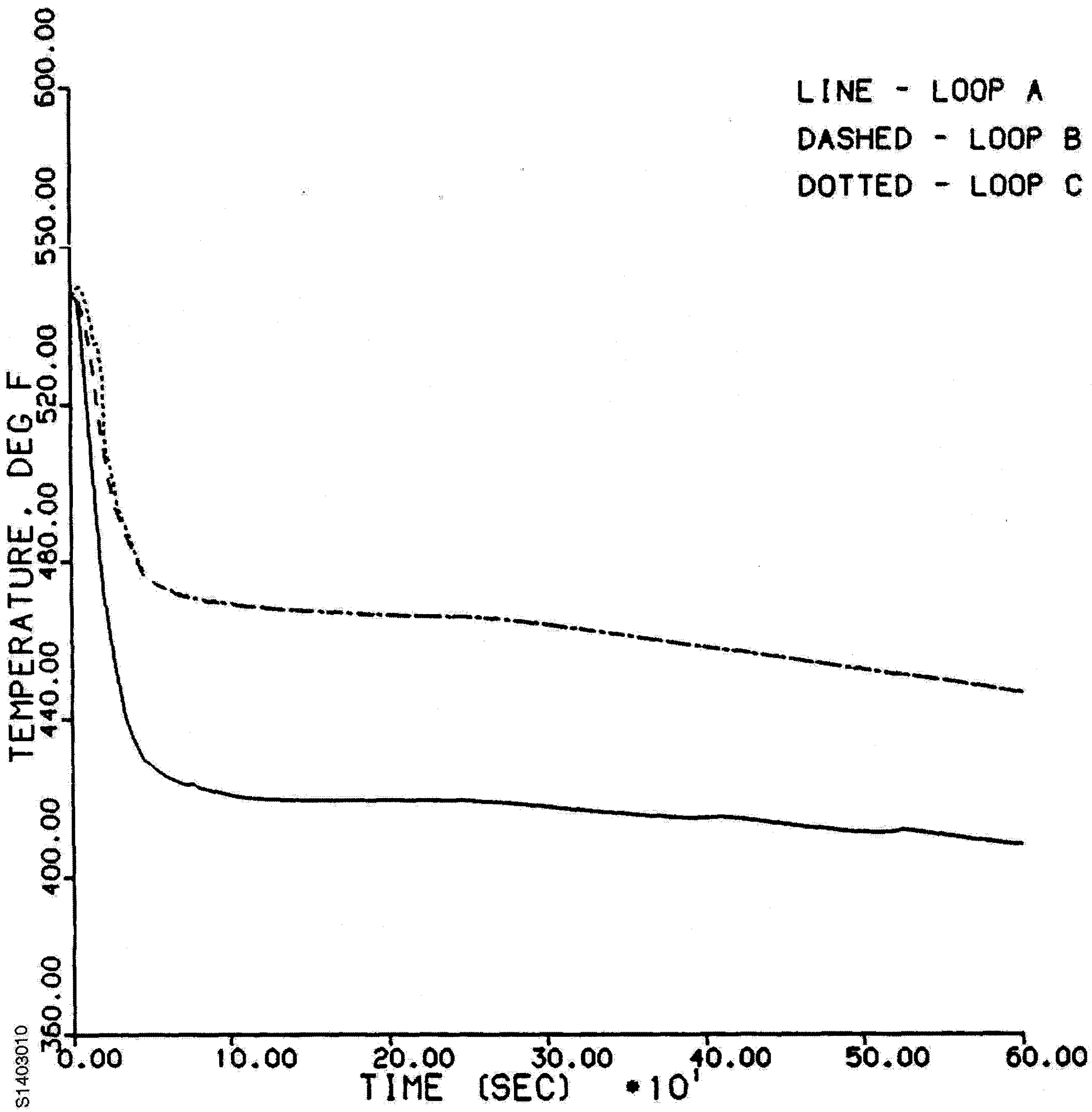


Figure 14.3-14
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK, W/O OFFSITE POWER AVAILABLE NORMALIZED CORE HEAT FLUX (FRACTION OF 2546 MWt)

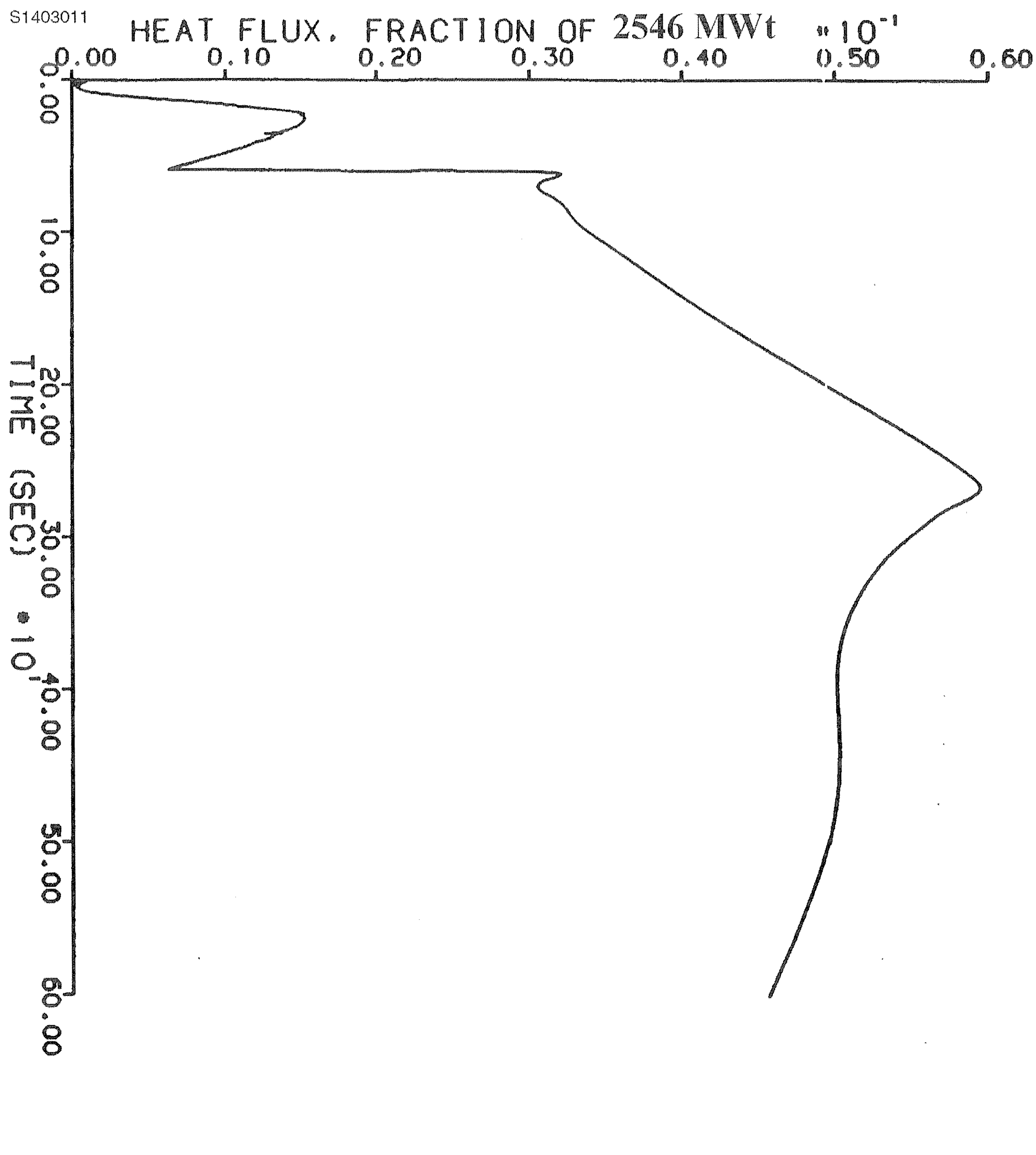


Figure 14.3-15
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
W/O OFFSITE POWER AVAILABLE PRESSURIZER PRESSURE

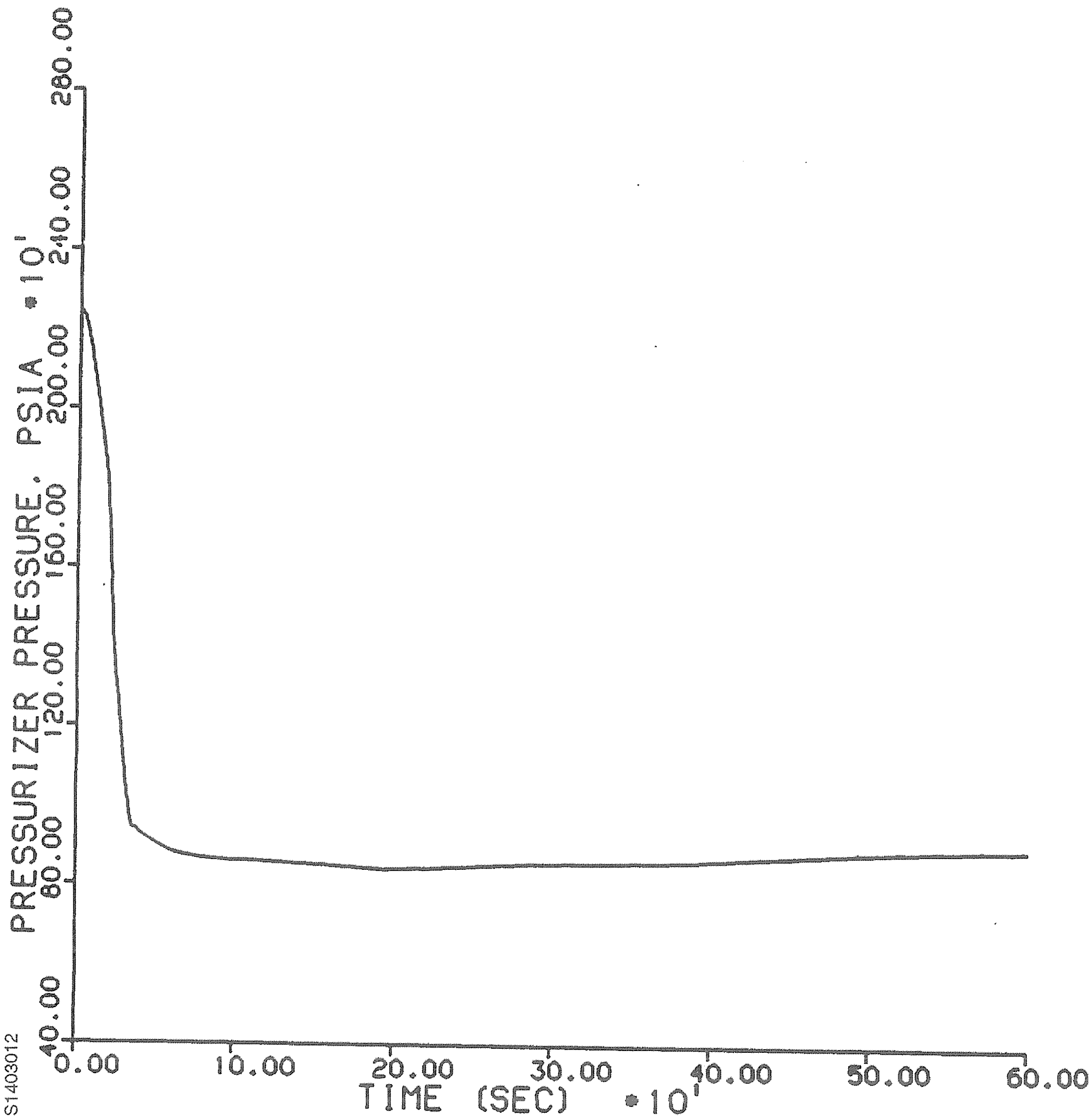


Figure 14.3-16
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
W/O OFFSITE POWER AVAILABLE CORE REACTIVITY,% ΔK/K

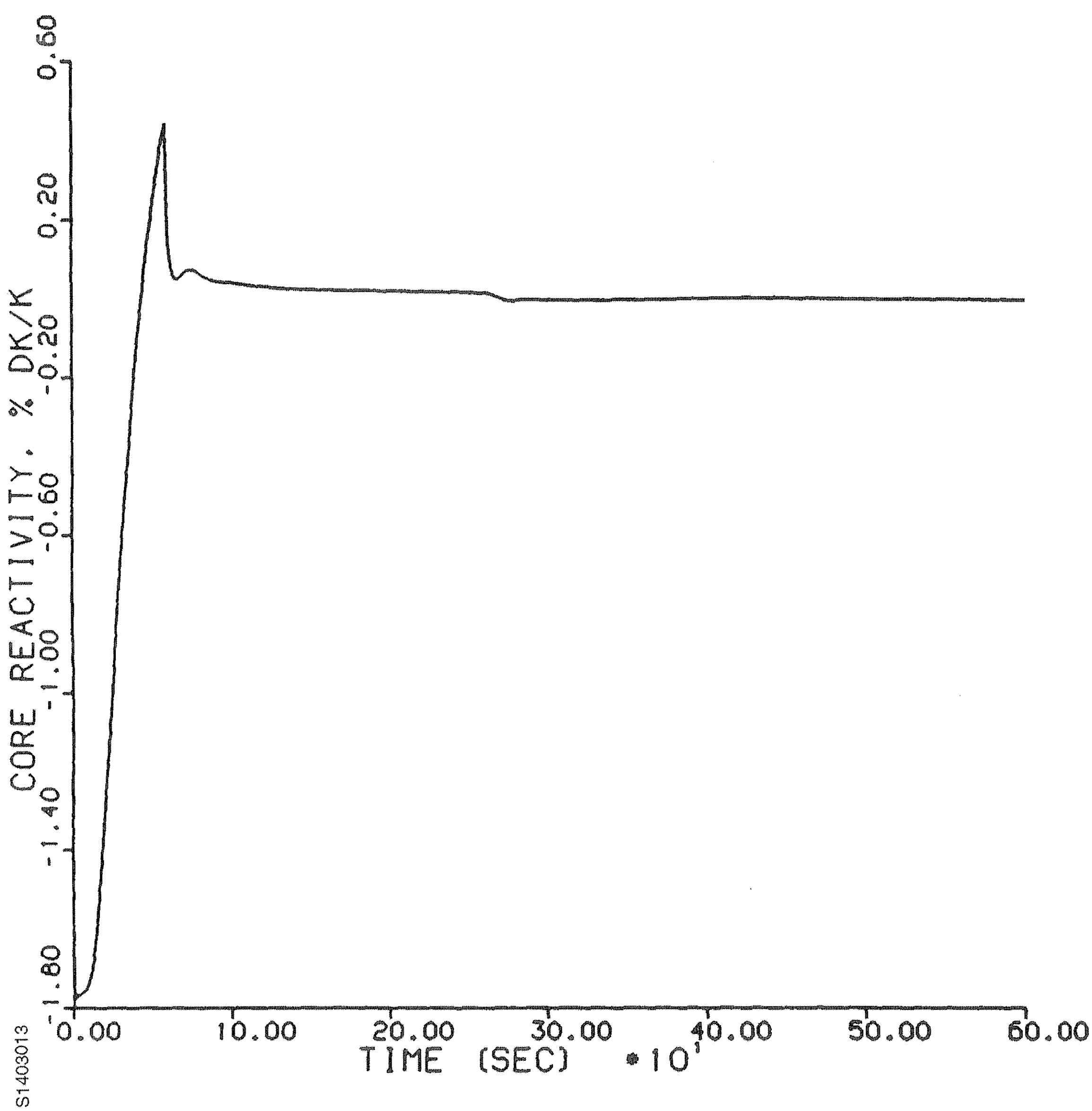


Figure 14.3-17
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
W/O OFFSITE POWER AVAILABLE CORE INLET BORON CONCENTRATION

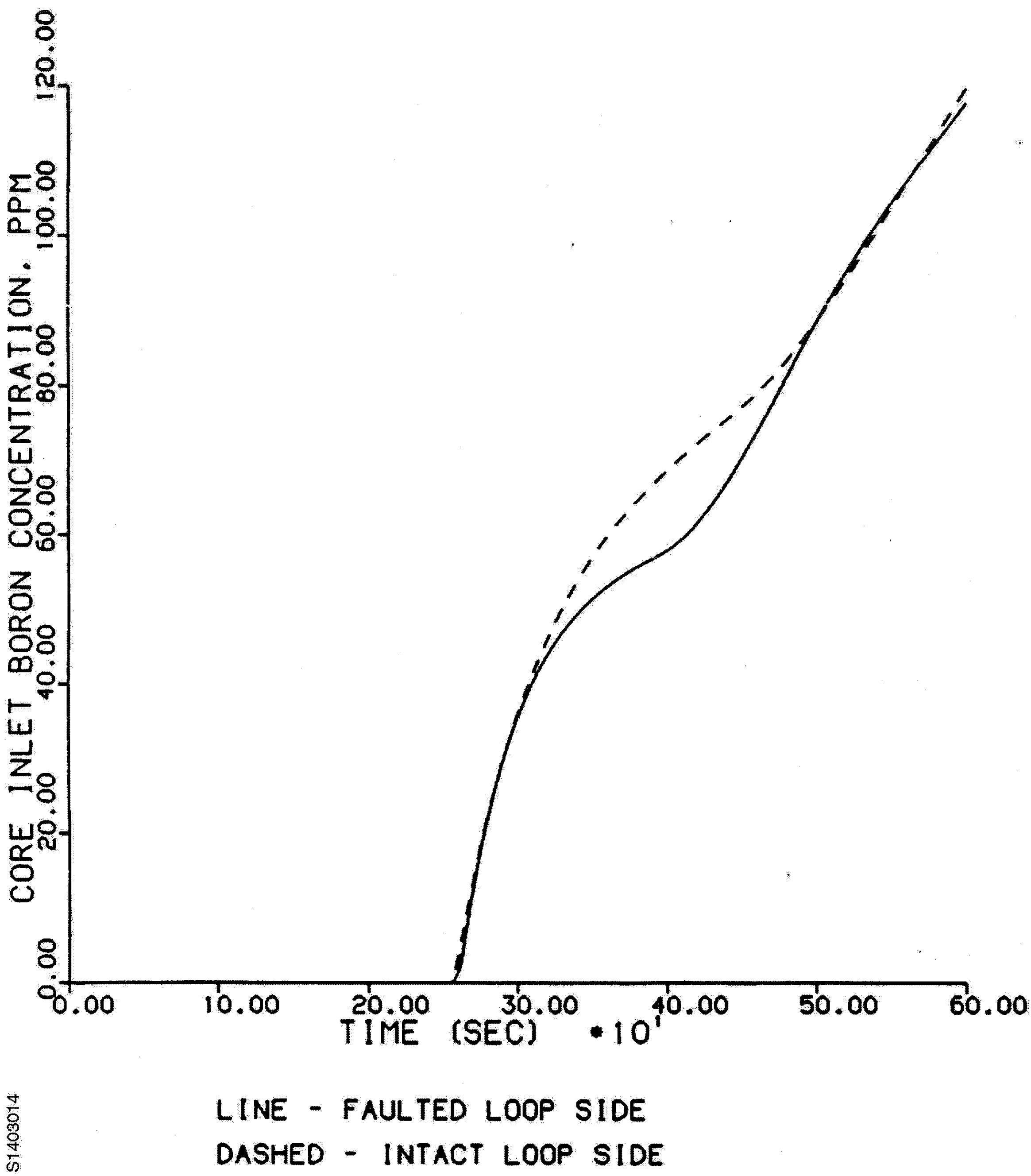


Figure 14.3-18
SPS MAIN STEAMLINE BREAK ANALYSIS 1.4 FT^2 BREAK,
W/O OFFSITE POWER AVAILABLE ACTUAL LOOP AVERAGE TEMPERATURES

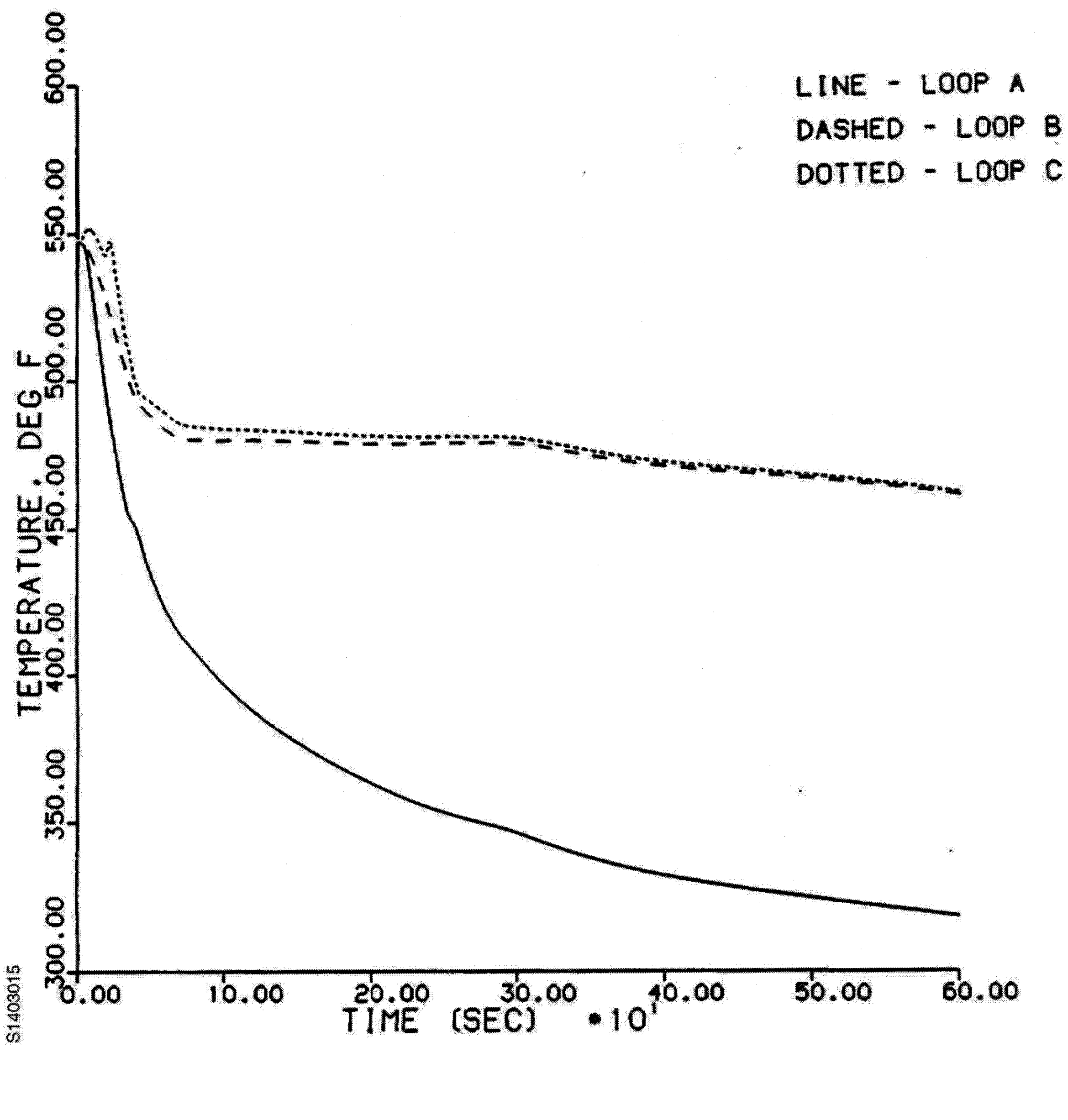


Figure 14.3-19
SPS MAIN STEAMLINE BREAK ANALYSIS CREDIBLE BREAK
NORMALIZED CORE HEAT FLUX (FRACTION OF 2546 MWt)

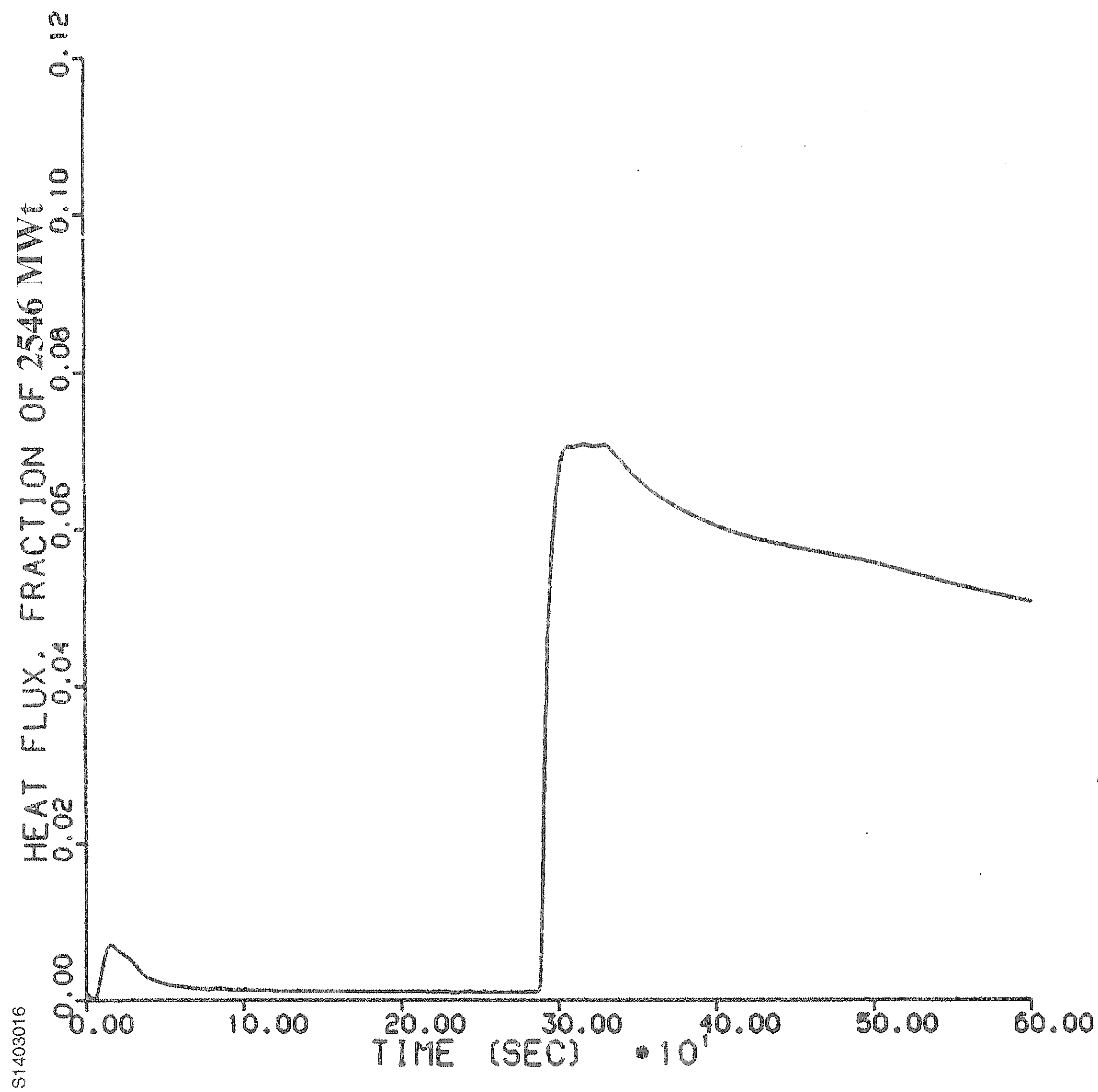


Figure 14.3-20
SPS MAIN STEAMLINE BREAK ANALYSIS
CREDIBLE BREAK PRESSURIZER PRESSURE

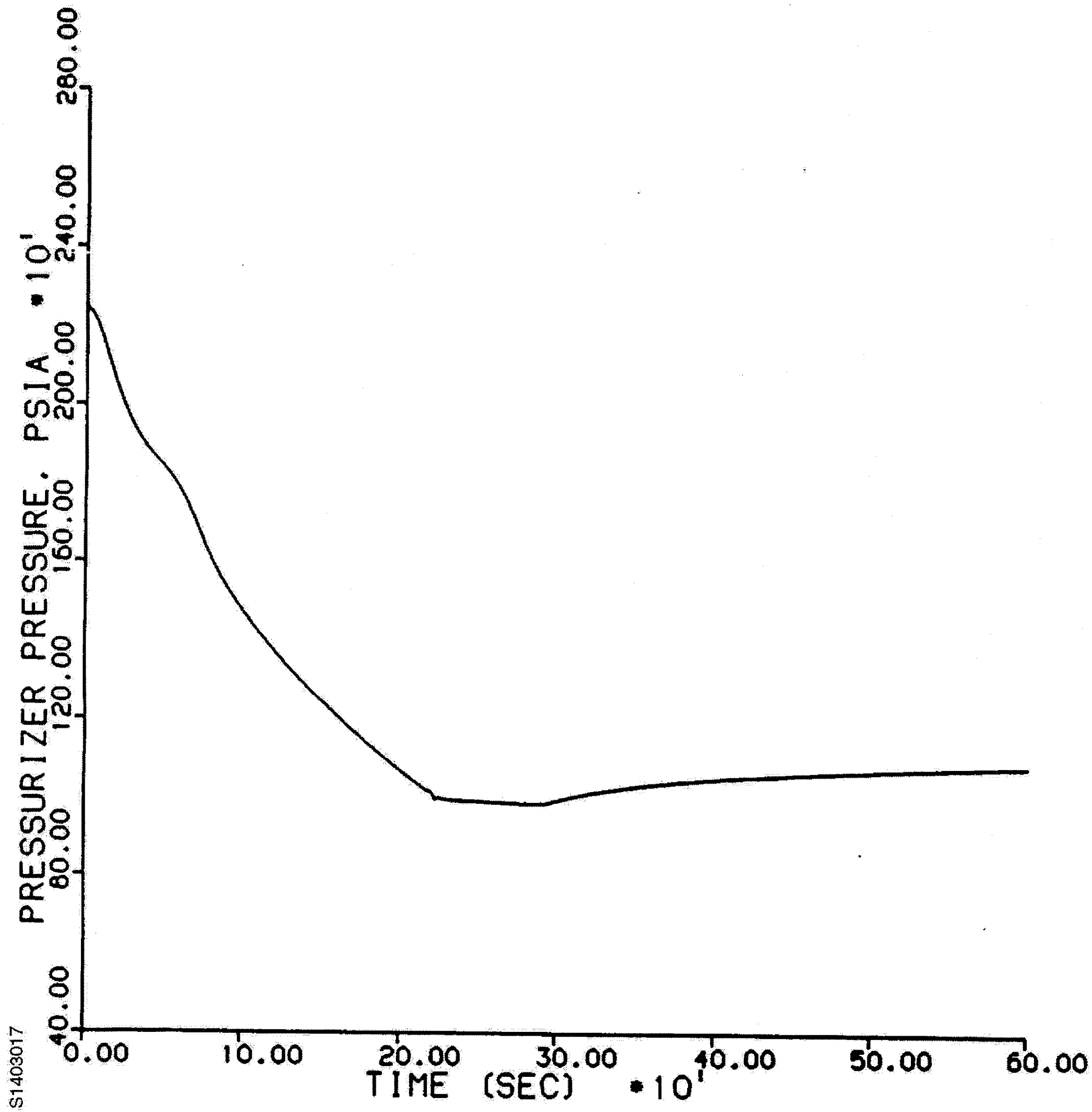


Figure 14.3-21
SPS MAIN STEAMLINE BREAK ANALYSIS
CREDIBLE BREAK CORE REACTIVITY,% ΔK/K

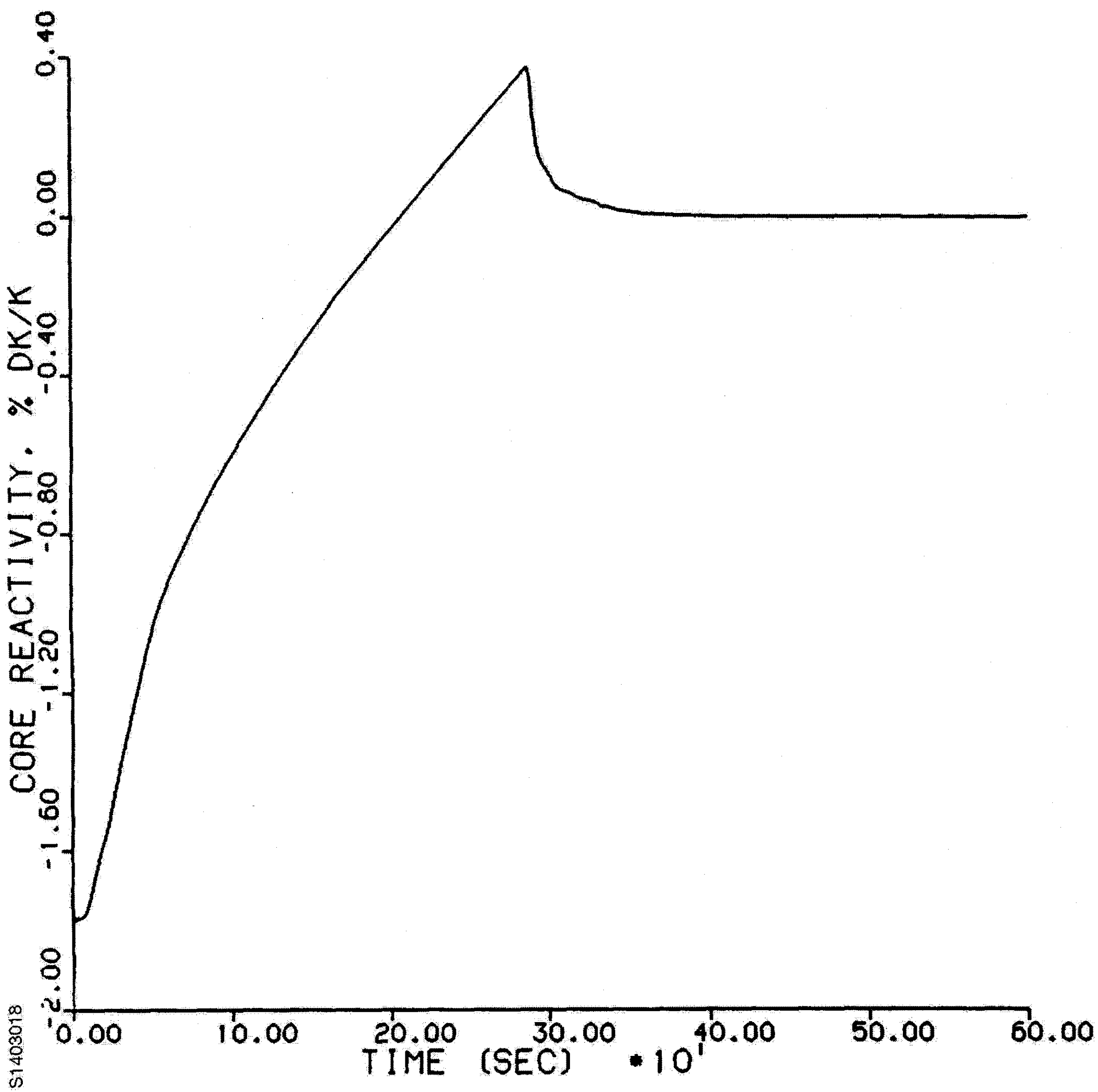


Figure 14.3-22
SPS MAIN STEAMLINE BREAK ANALYSIS
CREDIBLE BREAK CORE INLET BORON CONCENTRATION

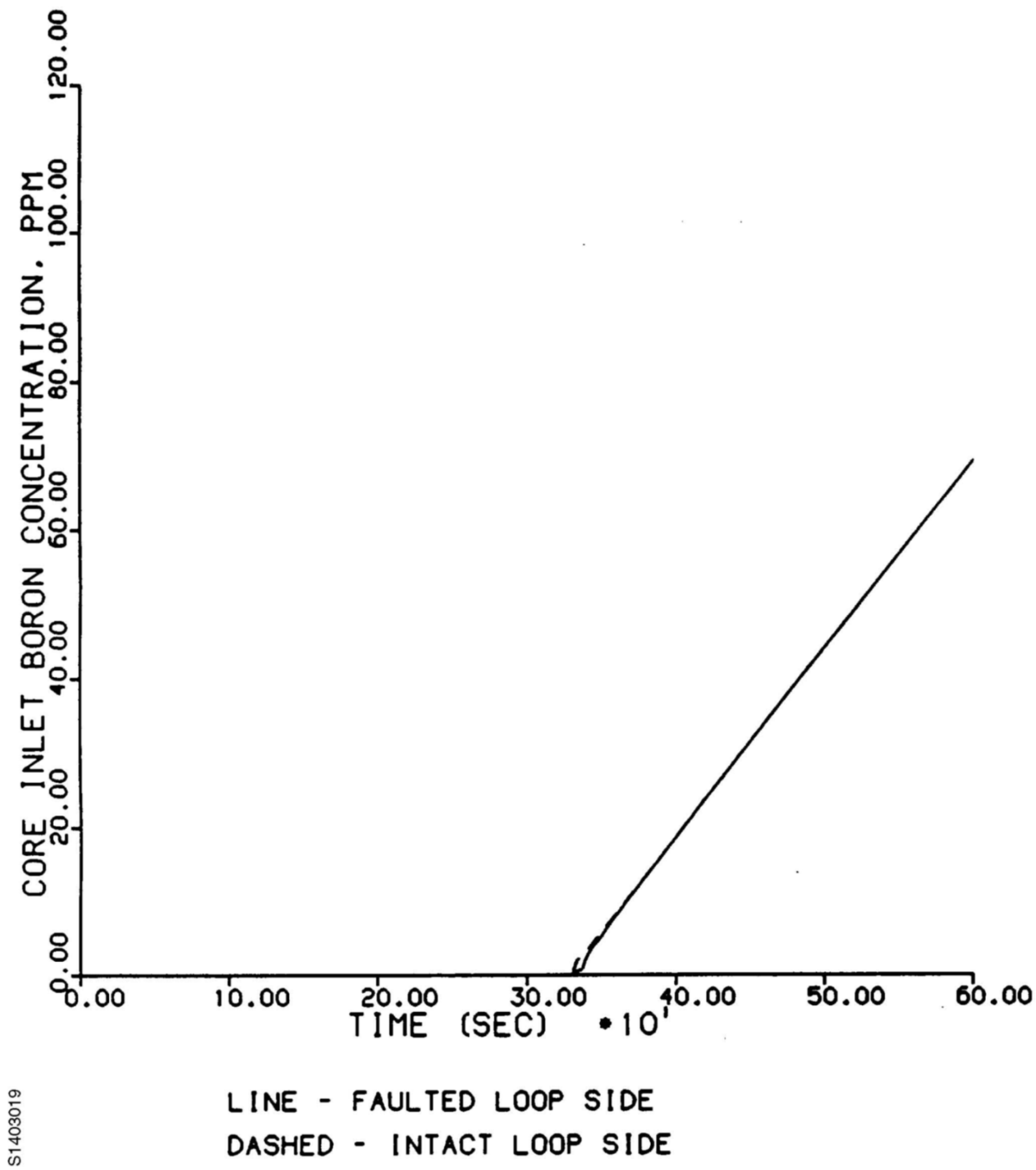


Figure 14.3-23
SPS MAIN STEAMLINE BREAK ANALYSIS
CREDIBLE BREAK ACTUAL LOOP AVERAGE TEMPERATURES

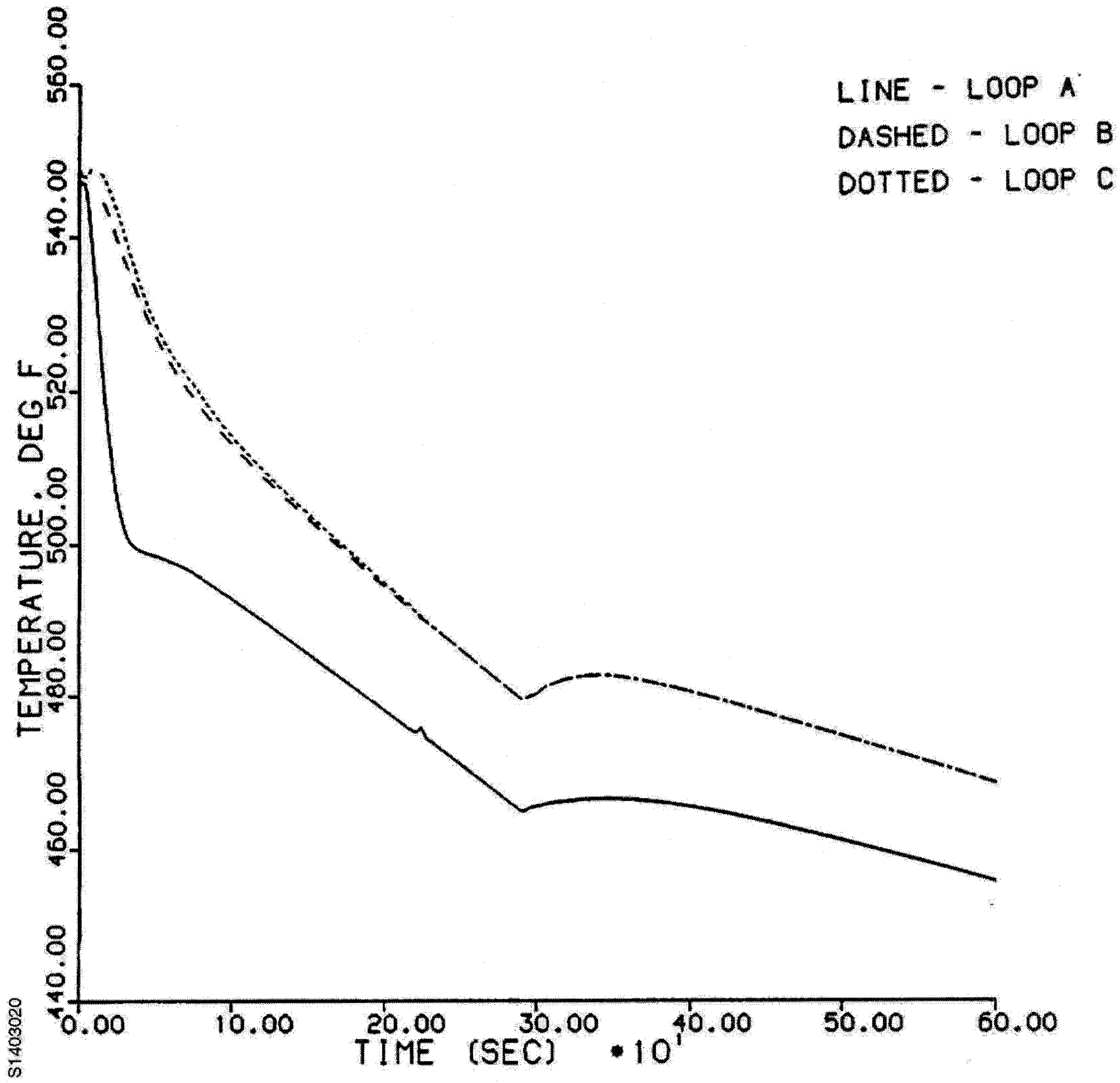


Figure 14.3-24
NUCLEAR POWER TRANSIENT - BOC HFP ROD EJECTION ACCIDENT

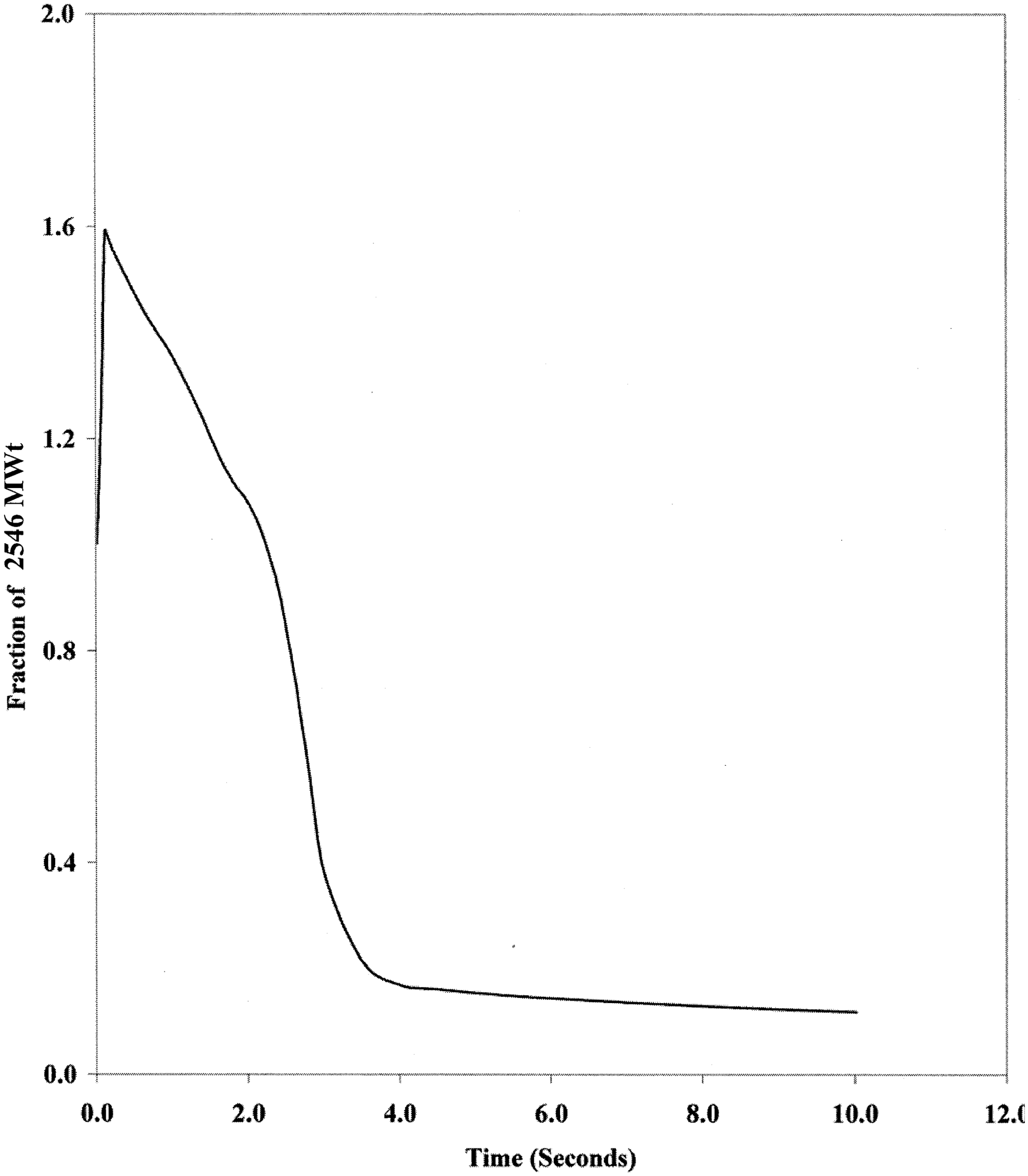


Figure 14.3-25
HOT SPOT FUEL AND CLAD TEMPERATURE VERSUS TIME -
BOC HFP ROD EJECTION ACCIDENT

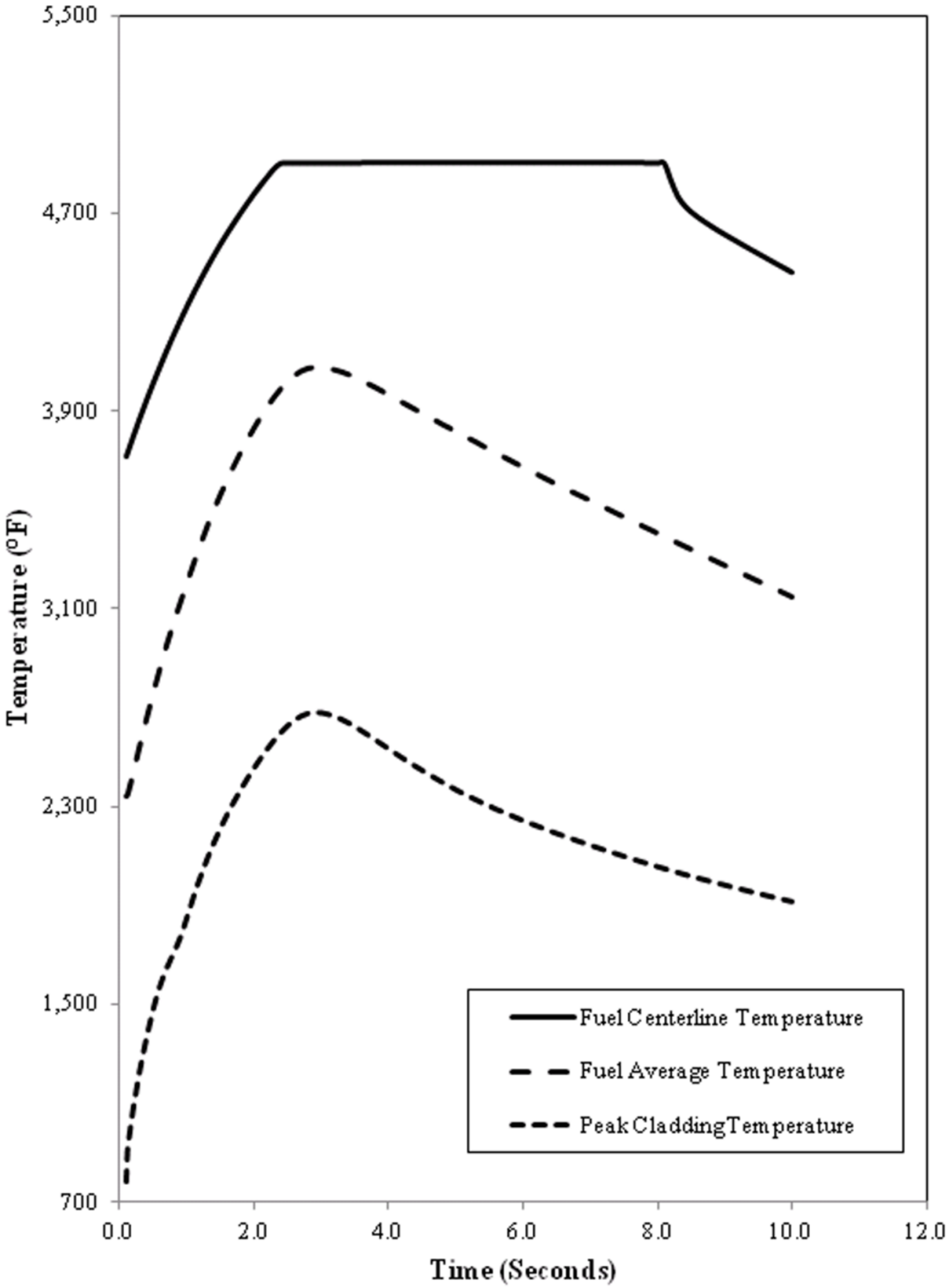


Figure 14.3-26
NUCLEAR POWER TRANSIENT - BOC HZP ROD EJECTION ACCIDENT

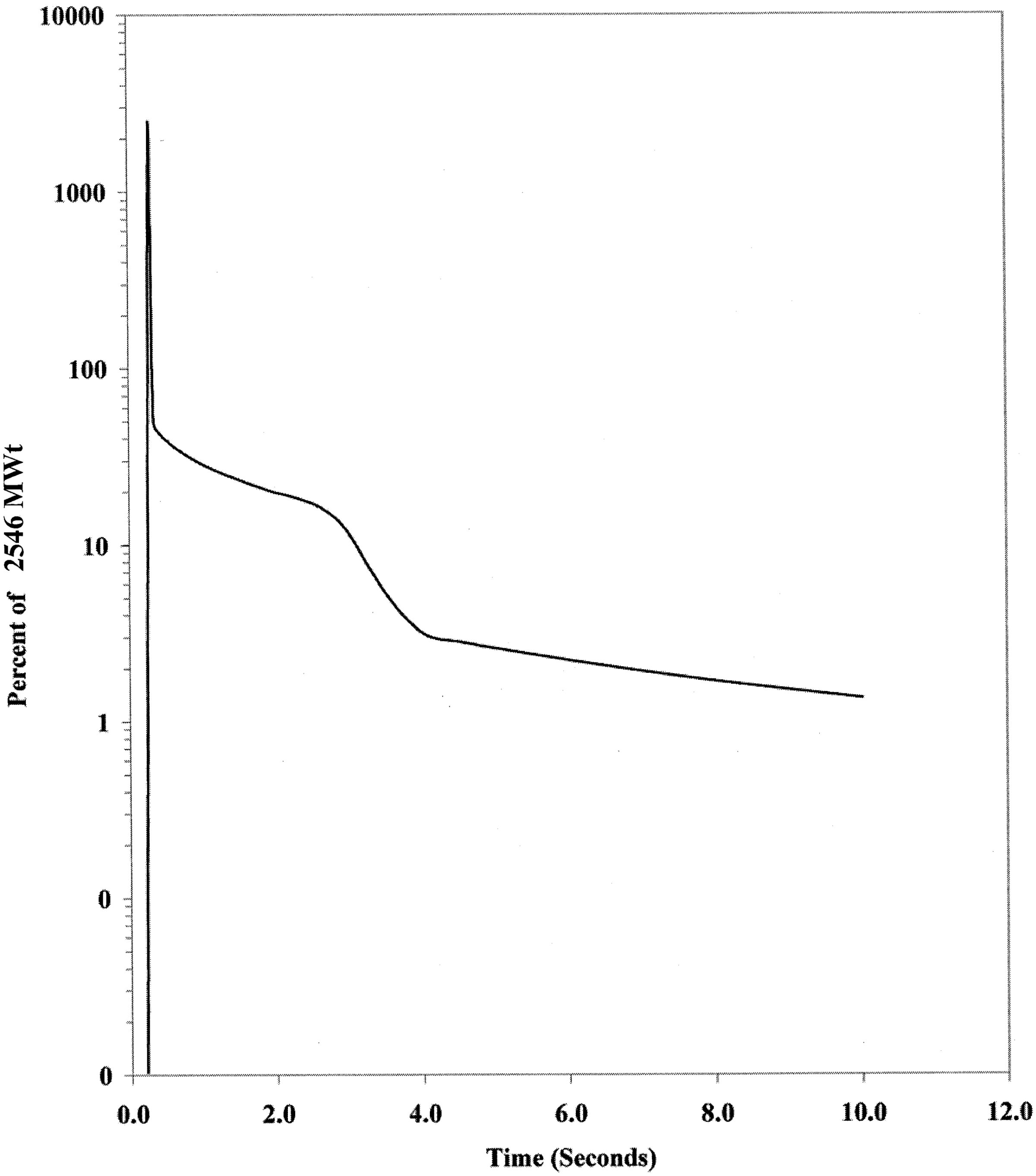


Figure 14.3-27
HOT SPOT FUEL AND CLAD TEMPERATURE VERSUS TIME
BOC HZP ROD EJECTION ACCIDENT

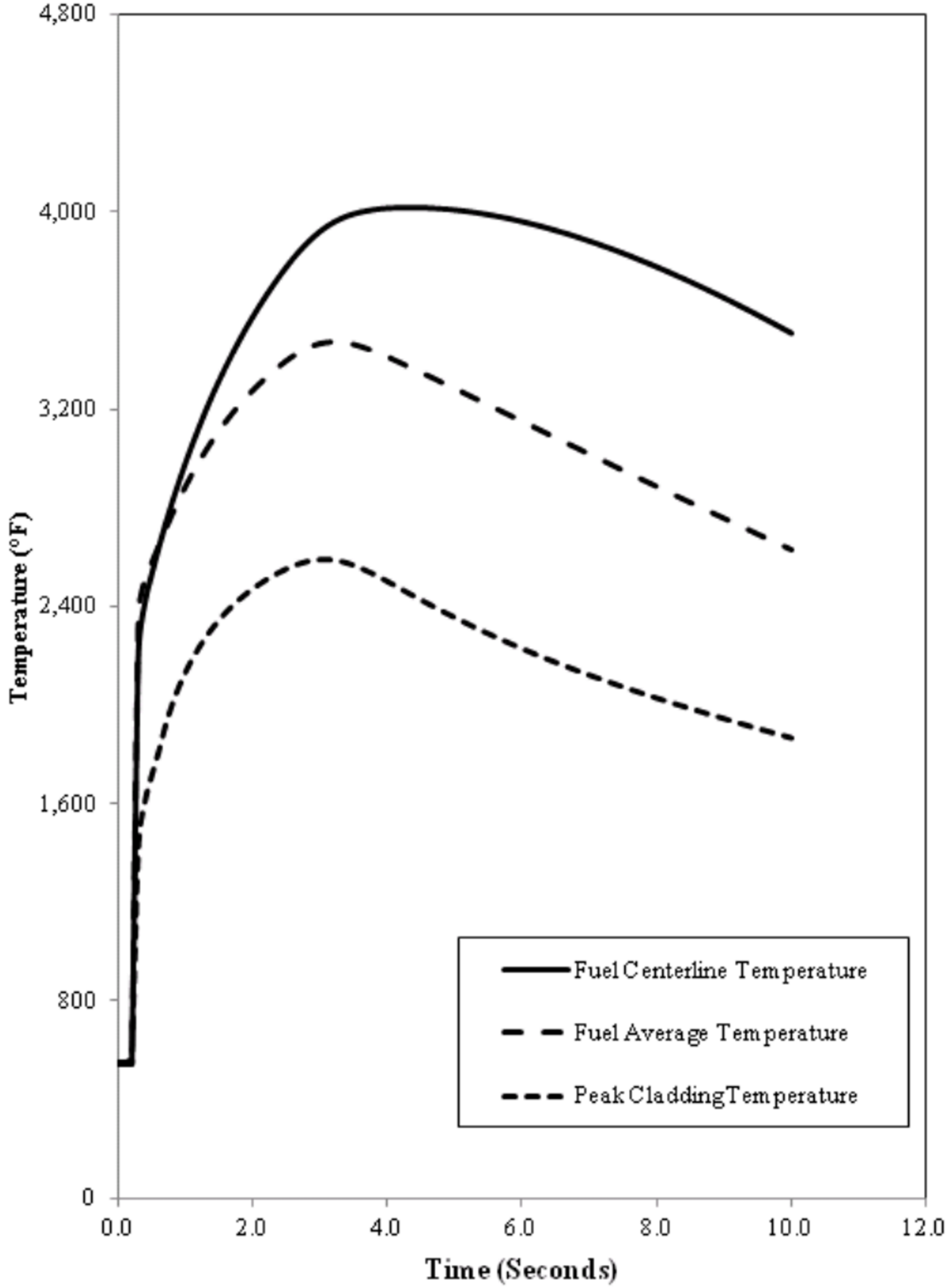
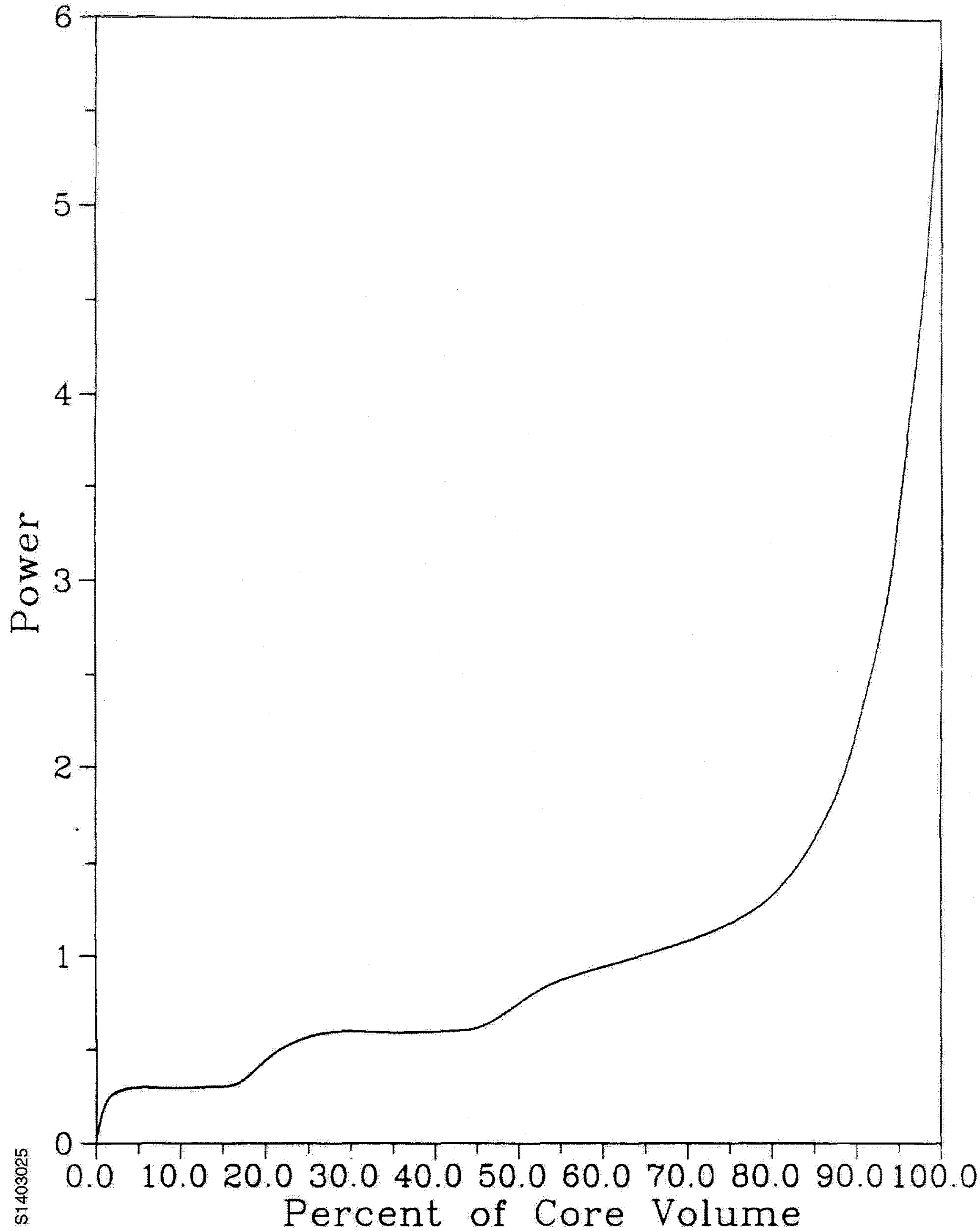


Figure 14.3-28
FUEL ROD POWER LEVEL VERSUS PERCENT OF CORE
VOLUME ROD EJECTION CASE



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14.4 GENERAL STATION ACCIDENT ANALYSIS

14.4.1 Fuel-Handling Accidents

The following fuel-handling accidents are evaluated:

1. A fuel assembly becomes stuck inside the reactor vessel.
2. A fuel assembly becomes stuck in the containment penetration valve (fuel transfer tube).
3. A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.
4. A fuel assembly in the reactor cavity becomes damaged (fuel-handling accident in containment).
5. A fuel assembly in the spent-fuel pool becomes damaged (fuel-handling accident in the spent-fuel pool).
6. A spent-fuel shipping cask is dropped into the cask laydown area of the spent-fuel pool (cask-drop accident).

14.4.1.1 **Accident Prevention or Mitigation**

The possibility of a fuel-handling accident is remote because of the stringent administrative controls and physical limitations imposed on fuel-handling operations. All refueling operations are conducted in accordance with prescribed procedures under the direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, the verification of complete control-rod assembly insertion is obtained by tripping the control-rod banks and obtaining indication of rod drop and disengagement from the control-rod drive mechanisms. The boron concentration in the reactor coolant is raised to the relatively high refueling concentration and verified by sampling. The refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all control-rod assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications. As the vessel head is raised, a visual check is made to verify that the control-rod assembly drive shafts are free in the mechanism housing.

After the vessel head is removed, the control-rod assembly drive shafts are removed from their respective assemblies using the manipulator crane hoist and the shaft unlatching tool. A spring scale is used to indicate that the drive shaft is free of the control-rod assembly as the lifting force is applied.

The fuel-handling manipulators and hoists are designed so that fuel cannot be raised above a position that provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and the spent-fuel pool area. In the spent-fuel pool, the design of storage racks and manipulation facilities is such that:

1. Fuel at rest is held in position by positive restraints in a safe, always subcritical, geometrical array, with no credit for boric acid in the water.

2. Fuel can be manipulated only one assembly at a time.
3. A violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.
4. Crane facilities do not permit the handling of heavy objects, such as a spent-fuel shipping cask, above the fuel racks.

Adequate cooling of spent-fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent-fuel pool.

Even if a spent-fuel assembly becomes stuck in the transfer tube, natural convection maintains adequate cooling. The fuel-handling equipment is described in detail in Chapter 9.

Two nuclear instrumentation system source range channels are continuously in operation and provide a warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and it would annunciate a local horn and a horn and light in the control room if the count rate increased above a preset low level.

The refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5% delta k/k with all control-rod assemblies inserted (Reference 8). At this boron concentration the core would also be subcritical with all control-rod assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

All these safety features make the probability of a fuel-handling accident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this accident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel-handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent-fuel pool and during their installation in the reactor. All irradiated fuel-handling actions are conducted under water. The handling tools used in the fuel-handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern that prevents any possibility of a criticality accident. The motions of the cranes that move the fuel assemblies are limited to a relatively low maximum speed. Caution is exercised during fuel-handling to prevent a fuel assembly from striking another fuel assembly or structures in the containment or fuel building.

The fuel-handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips that provide a total restraining force of approximately 60 lb on each fuel rod. If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited by the restraining force of the grid clips. The force transmitted to the fuel rods during fuel-handling is not sufficient to breech the fuel-rod cladding. If the fuel rods were not in contact with the bottom plate of the assembly, the rods would have to slide against the 60-lb friction force. This would absorb the shock and thus limit the force on the individual fuel rods.

Considerable assembly deformation would have to occur before the rod would make contact with the top plate and place any appreciable load on the fuel rod. In view of the above, it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would affect the integrity of the cladding.

If during handling the fuel assembly were to strike against a flat surface, the loads would be distributed across the fuel assembly and grid clips and essentially no damage would be expected in any fuel rods.

If the fuel assembly were to strike a sharp object, it would be possible for the sharp object to damage the fuel rods with which it comes in contact, but a breech of the cladding would be unlikely. On this basis, assuming the failure of an entire row of fuel rods (15) is a conservative upper limit.

Preliminary analyses in support of the initial FSAR assumed three extremely remote situations: a fuel assembly is dropped 14 feet and strikes a flat surface; one assembly is dropped onto another; and one assembly strikes a sharp object. The analysis of a fuel assembly assumed to be dropped and striking a flat surface considered the stresses the fuel cladding was subjected to and any possible buckling of the fuel rods between the grid clip supports. The results showed that the axial load at the bottom section of the fuel rod, which would receive the highest loading (approximately 100 lb) was below the critical buckling load (250 lb) and the stresses were relatively low and below the yield stress. For the case where one assembly is dropped on top of another fuel assembly, the loads would be transmitted through the end plates and the control-rod assembly guide tubes of the struck assembly before any of the loads reached the fuel rods.

The end plate and guide thimbles absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. Also, energy is absorbed in the struck assembly top end plate before any load can be transmitted to the fuel rods. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling loads and the stresses in the cladding were relatively low and below yield.

The experience that has been gained in Westinghouse reactor refueling operations indicates that fuel cladding integrity failures would not be expected to occur during fuel-handling operations.

For the initial FSAR, the rupture of one complete outer row of fuel rods in a withdrawn spent-fuel assembly was assumed as a conservative limit for evaluating the environmental consequences of a fuel-handling accident. The remaining fuel assemblies are protected by the storage rack structure so they are not subjected to lateral bending loads. No damage resulted from the axial application of a load of 2200 lb to a fuel assembly. The maximum load expected to be experienced in service is approximately 1000 lb. This information was used in the fuel-handling equipment design to establish the limits for inadvertent axial loads.

The spent fuel cask drop analysis is discussed in Reference 1. The fuel handling accident in the containment and the fuel handling accident in the spent fuel pool are described below in more detail. These analyses were performed as part of implementing the alternate source term that is described in RG 1.183 (Reference 13). It should be noted that Surry Power Station has been licensed for fuel burnups up to 62,000 MWD/MTU lead rod burnup beginning with Surry Improved Fuel Assemblies with ZIRLO cladding (Reference 20). Older fuel assemblies with Zircaloy-4 cladding are limited to a lead rod average burnup of 60,000 MWD/MTU (Reference 20). The Optimized ZIRLO cladding was approved by the NRC for use at Surry in Reference 21 as part of the 15 x 15 Upgrade fuel design for lead rod average burnups up to 62,000 MWD/MTU. For this extended burnup it has been shown that the radiological consequences of the fuel handling accidents discussed below remain unchanged (References 2, 3, 4, 19, & 20).

Virginia Power conducted a spent fuel cask drop evaluation in support of the use of spent fuel casks in the fuel building area (References 5 & 6). As a result of this evaluation, cask impact pads were installed in the cask loading area of the spent fuel pool, and the spent fuel pool was divided into two regions for the storage of spent fuel (Reference 7). Region 1 comprises the first three rows of fuel racks (324 storage locations) adjacent to the Fuel Building Trolley Load Block. Region 2 comprises the remainder of the fuel racks in the fuel pool. During spent fuel cask handling, Region 1 is limited to storage of spent fuel assemblies which meet the criteria delineated in Surry Power Station Technical Specification 5.3, *Fuel Storage*.

14.4.1.2 **Fuel-Handling Accident in the Containment**

The fuel handling accident (FHA) in the containment has three postulated release paths. These three pathways are the ventilation system (through Vent Stack No. 2), the open personnel airlock, and the open equipment access hatch. The analysis models the release flow and atmospheric dispersion factors to bound the radiological effects of release from any combination of the three release paths and from penetrations that terminate in the Auxiliary Building and Safeguards. Filtration of the containment release to atmosphere is not credited in the FHA analysis.

14.4.1.2.1 Assumptions

During refueling, the containment purge system may be aligned to exhaust through either the non-safety related or safety-related ventilation filters in the Auxiliary Building or a

combination thereof, but no filtration is credited in the analysis. If exhaust is being filtered, more than one filter bank may be on line because the fuel building exhaust could also be aligned through the filters. The containment purge design flow rate to the non-safety-related filters is 20,000 cfm. The design flow rate through a safety-related filter is 36,000 cfm. The containment was modeled at various release flow rates between 2000 cfm and 80,000 cfm in order to bound all credible releases and maximize dose consequences. The fuel building modeling assumptions are discussed in Section 14.4.1.3.1. The analysis results are not sensitive to release flows exceeding 36,000 cfm.

While the purge system is in operation, the air flow in the containment is as follows. Air enters the containment through two 14,500-cfm fans and two 36-inch butterfly supply valves, and is dispersed through the ring header outside the crane wall at Elevation 39 ft. 6 in. The air is continuously recirculated inside the containment by three 75,000-cfm recirculation fans. The air is purged from the containment through the ring header at Elevation -20 ft. outside the crane wall. The air discharges through two 36-inch butterfly valves in series. The air then passes through the auxiliary building filter banks and the two 36,000 cfm filter exhaust fans. Air is also assumed to flow through the personnel airlock, equipment access hatch, and other containment penetrations (if these are open).

The worst single failure would be either the inability to close one of the hatches or the loss of the valve-closing circuit that closes the valves and secures the purge fans on an alarm from either the manipulator crane monitor or the containment gas and particulate monitors. The two output relays are sufficiently redundant to secure purge flow; however, a loss of power to this circuit would cause them not to function. Failure to isolate containment or establish containment closure could cause a release to the atmosphere with a boundary dose as calculated below. Even though containment isolation, containment closure, and filtering of the release are not credited in the analysis, the dose is still within allowable limits (Table 14.4-5).

The transit time for any released activity from the radiation detection point to the control room normal ventilation system intake is assumed to be sufficiently long such that control room manual isolation was modeled as occurring before any radioactive material reached the control room air inlet. This assumption relies upon the operability of the manipulator crane area monitor and the containment gas and particulate monitors in conjunction with communications to provide a timely and valid indication of a FHA.

Within 1 hour of control room envelope isolation, procedures require the alignment of the control room emergency ventilation system to provide a filtered breathing air supply to the control room envelope. This analysis considered that one fan was operational which provides a control room intake flow rate of 1000 cfm from 1 hour through the end of the 30-day dose calculation period. Operation of additional fans will not increase the consequences of the FHA. An unfiltered inleakage of 500 cfm is assumed when the control room is isolated (0-30 days).

The control room χ/Q values were determined with ARCON96 (Reference 11) methodology and meteorological data for the 1982 through 1986 time period. These values are listed in Table 14.4-3. The control room occupancy factors in Table 14.4-4 were also incorporated into the dose calculations to reflect that personnel would not be exposed to the released activity 100% of the time over the entire 30 day period. The breathing rate used for the control room dose calculations was 3.5×10^{-4} m^3/sec.

More specific conservative assumptions are:

1. A puff release of radioactivity occurs as the result of the rupture of a fuel assembly in the reactor fuel cavity. The puff release is instantaneously and uniformly distributed through one-half the containment volume.
2. The manipulator crane area monitor is gamma radiation sensitive, so that it is not necessary for it to be immersed in a radioactive cloud to detect radioactivity. Its position above the fuel cavity (approximately 10 feet), unshielded from direct gamma rays from the cavity, enhances its capability to detect an accident release immediately.
3. The containment closure is not credited even though the equipment hatch and the personnel airlock will be capable of being closed, and all other containment penetrations will either be closed, capable of being closed, or have an operable isolation valve.
4. The containment effluent flow rate is varied from 2000 to 80,000 cfm to bound all credible flow rates for the ventilation exhaust system or natural circulation flow processes and to maximize the dose consequences.
5. The assumed volume of containment air with which the radioactive release is mixed is 931,500 ft3, or 50% of the containment volume.
6. The delay time from reactor shutdown to the initiation of fuel assembly transfer operations is at least 100 hours.
7. The assembly radial peaking factor is 1.70, which is the appropriate peaking factor, including uncertainties, for events (e.g., FHA) that do not employ the Statistical DNBR Evaluation Methodology. This value is a multiplier applied to the batch average isotopic activity of assemblies in the first and second cycle of irradiation to determine bounding activity in those batches. Since assemblies in their third cycle can not achieve these power levels, a peaking factor of 1.188 was applied to determine the bounding activity of that batch. The accident analysis was performed using the most limiting fuel assembly, which was determined to be from the second burned batch.
8. The number of fuel assemblies in the core is 157, which is distributed as 56 first burned, 56 second burned, and 45 third burned. However, the evaluation results are not sensitive to this distribution.
9. 5.35 percent of the fuel assembly Iodine-131 activity is assumed to be released into the reactor cavity water, as are five percent of the other iodine isotopes present in the fuel

assembly, 99.85% being elemental and 0.15% in the organic form. The decontamination factor (DF) for elemental iodine is 500 while the DF for organic iodine is 1.

10. Five percent of each of the noble gases present in the fuel assembly is released to the reactor cavity pool, with the exception of Kr-85; 10% of Kr-85 is released. The DF of the water for noble gases is 1.
11. The calculational method includes dose conversion factors for each isotope.
12. The χ/Q values used in the offsite dose analysis were calculated using the PAVAN (Reference 10) methodology and are based on site specific meteorological data for the 1994 through 1998 time period. These χ/Q values are listed in Table 14.4-2.
13. For the first 8 hours, the breathing rate for offsite dose calculations was 3.5×10^{-4} m^3/sec. The breathing rates for the time periods from 8 to 24 hours and from 24 hours until the end of the accident, were 1.8×10^{-4} m^3/sec and 2.3×10^{-4} m^3/sec respectively.
14. The distance from the possible release point to the site boundary varied from a minimum of 1642 feet to a maximum of 5066 feet. The minimum distance of 1642 feet was used.
15. The assumed volume of the 45-ft elevation of the Auxiliary Building is 100,000 ft^3. This volume is used to dilute the release from the Containment Personnel Airlock prior to exiting through the louvers on the east and west sides of the 45-ft elevation of the Auxiliary Building.

The activity for the limiting fuel assembly is calculated using the following equation:

Fuel assembly activity (Ci) = Total activity in the second-burned batch normalized to core average power after a 100-hr decay $\times \frac{1}{56} \times 1.70$.

14.4.1.2.2 Results

As was mentioned above, the containment effluent flow rate is varied from 2000 to 80,000 cfm to bound all credible flow rates for the ventilation exhaust system or natural circulation flow processes and to maximize the dose consequences. The results of the FHA dose calculations are shown in Table 14.4-5. The doses in Table 14.4-5 are a composite of the worst doses from an FHA in either the containment of the fuel building. A fuel-handling accident in the containment will not lead to EAB and LPZ doses exceeding the dose limits as specified in Regulatory Guide 1.183. Also, the control room doses will not exceed the 10 CFR 50.67 dose limit.

14.4.1.3 **Fuel-Handling Accident in the Spent-Fuel Pool**

If a fuel assembly is dropped in the spent-fuel pool in the fuel building, the increase in radiation level as these radionuclides mix with the fuel building air will be detected by the two radiation monitors located in the ventilation vent no. 2 or by the fuel pool bridge area monitor.

The fuel building exhaust may be diverted through the particulate and activated charcoal filter banks during refueling operations but no filtration is credited in the analysis (Section 9.13).

The monitors alarm on a high radiation level to indicate a possible dropped-fuel-assembly incident.

14.4.1.3.1 Assumptions

To determine the quantity of radioactive material available for release, it is conservatively assumed that the fuel assembly with the peak fission product inventory is the one damaged. The inventory is based on maximum full power operation at the end of core life immediately preceding shutdown and a conservative radial peaking factor which is applied to all fuel rods in the assembly. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions of the fuel rods during normal operation is considered to be available for immediate release into the water in the event of clad damage. The quantity of radioactive material released subsequent to the immediate release is considered to be negligible compared to the quantity released immediately after the Fuel Handling Accident (FHA).

The fuel radionuclide inventory was based on a core power level of 2605 MWt. This core power level is conservative compared to 100.38% of the uprated power level of 2587 MWt (i.e., 2596.9 MWt).

For analyses employing alternative source terms, the FHA is discussed in Section 15.0.1 of the NRC's Standard Review Plan and Regulatory Guide 1.183. The following assumptions were made for the evaluation of the Surry control room and offsite doses due to a FHA.

1. The accident occurs 100 hours after shutdown. Surry Technical Specification 3.10 requires a minimum 100-hour period between the shutdown of a unit and initiation of fuel movement, so the use of a 100-hour time period is conservative. Radioactive decay of the fission product inventory during the 100-hour interval between shutdown and the assumed commencement of fuel handling is incorporated into the analysis.
2. The minimum water depth between the top of the damaged fuel rods and the water surface is 23 feet.
3. All of the gap activity in the damaged rods is released and consists of 5% of the total noble gases other than Kr-85, 10% of the Kr-85, and 5% of the total radioactive iodine other than I-131, and 5.35% of the I-131 in the rods at the time of the accident.
4. The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown.
5. The iodine gap inventory is composed of 99.85% inorganic species and 0.15% organic species.
6. The pool decontamination factors for the inorganic and organic species are 500 and 1, respectively, giving an overall decontamination factor of 286 (i.e., 99.65% of the total iodine released from the damaged rods is retained by the water).

This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 57% elemental and 43% organic species.

7. The retention of the noble gases in the water is negligible.
8. After escaping from the spent fuel pool the effluent mixes with the fuel building atmosphere before entering the ventilation system and being released to the atmosphere through ventilation vent No. 2.

The amount of radioactive material which is released to the fuel building or containment during a FHA at 100-hour period of decay is determined from this core inventory using the following assumptions:

1. All rods in one fuel assembly are damaged.
2. There are 157 fuel assemblies in the Surry core, which is distributed as 56 first burned, 56 second burned, and 45 third burned. However, the evaluation results are not sensitive to this distribution.
3. The assembly radial peaking factor is 1.70, which is the appropriate peaking factor, including uncertainties, for events (e.g., FHA) that do not employ the Statistical DNBR Evaluation Methodology. This value is a multiplier applied to the batch average isotopic activity of assemblies in the first and second cycle of irradiation to determine bounding activity in those batches. Since assemblies in their third cycle can not achieve these power levels, a peaking factor of 1.188 was applied to determine the bounding activity of that batch. The accident analysis was performed using the most limiting fuel assembly, which was determined to be from the second burned batch.
4. Gap fractions as defined above.
5. Decontamination factors as defined above.

The resulting activities released to the fuel building or containment are given in Table 14.4-1.

The LOCADOSE computer code system (Reference 9) was used to calculate doses for the FHA. The model for this accident considered three distinct volumes: the environment, the spent fuel building and the control room. The spent fuel building volume assumed is a conservatively small value, which minimizes mixing of the radioactive material with the building atmosphere. The following volumes were used in the FHA control room dose calculations:

Spent Fuel Building Volume	1.11×10^5 ft^3
Control Room Volume	2.23×10^5 ft^3

The release to the environment from the fuel building was varied from 3500 to 80,000 cfm to bound all credible flow rates for the ventilation exhaust system and to maximize the dose consequences. No credit was taken for filtration of these releases.

The χ/Q values which were used to calculate the exclusion area boundary (EAB) and low population zone (LPZ) doses were calculated using the PAVAN (NUREG/CR-2858) methodology and were based on site specific meteorological data for the 1994 through 1998 time period. The χ/Q values are listed in Table 14.4-2.

The transit time for any released activity from the radiation detection point at the water surface to the control room normal ventilation system intake is assumed to be sufficiently long such that control room manual isolation was modeled as occurring before any radioactive material reached the control room air inlet. This assumption relies upon the operability of the fuel pit bridge area monitor and the ventilation vent No. 2 gas and particulate monitors in conjunction with communications to provide a timely and valid indication of a FHA.

Within 1 hour of control room envelope isolation, procedures require the alignment of the control room emergency ventilation system to provide a filtered breathing air supply to the control room envelope. This analysis considered that only one fan was operational which provides a control room intake flow rate of 1000 cfm from 1 hour through the end of the 30-day dose calculation period. An unfiltered inleakage of 500 cfm is assumed when the control room is isolated (0-30 days).

The control room χ/Q values were determined with the ARCON96 (Reference 11) methodology and meteorological data for the 1982 through 1986 time period. These values are listed in Table 14.4-3. The control room occupancy factors in Table 14.4-4 were also incorporated into the dose calculations to reflect that personnel would not be exposed to the released activity 100% of the time over the entire 30 day period. The breathing rate used for the control room dose calculations was 3.5×10^{-4} m^3/sec which is consistent with Reference 13.

14.4.1.3.2 Results

The results of the FHA dose calculations are shown in Table 14.4-5. The doses in Table 14.4-5 are a composite of the worst doses for an FHA in either the containment or the fuel building.

The EAB and LPZ doses for a FHA are less than the dose limits presented in Regulatory Guide 1.183 as shown in Table 14.4-5. The control room doses for the FHA are less than the 10 CFR 50.67 limit, which is also indicated in Table 14.4-5.

14.4.1.3.3 Analysis for High-Density Spent-Fuel Racks

The use of high density fuel racks does not affect the dose consequences resulting from a fuel handling accident in the spent fuel pool. Therefore the analysis provided in the Fuel Handling Accident in the Spent Fuel Pool in Section 14.4.1.3 remains bounding for the use of high density fuel storage racks.

14.4.2 Radioactive Gas Release

The concentration of radioactive waste gases in the primary and auxiliary systems is a function of the rate of fission gas release to the coolant from defective fuel and the rate of gas removal by auxiliary systems. The components that retain significant concentrations of radioactive gases are the volume control tank and the waste gas decay tanks. The radioactive release analysis considers the rupture of the volume control tank and a waste gas decay tank with an instantaneous release of the radioactive gas inventories of each tank to the environment.

14.4.2.1 **Volume Control Tank Rupture**

In this analysis, the volume control tank (VCT) is assumed to rupture and release to the atmosphere all the gases that have collected in the vapor space of the tank. Also released are all the gases in the liquid inventory of the tank and in the volume of liquid that continues to flow into the tank until it is isolated. Isolation is assumed to take 25 minutes, and the flow rate of the entering liquid is assumed to be 160 gpm, a conservatively high letdown flow rate.

The maximum activities of the gases in the vapor space with 1% failed fuel are listed in Table 14.4-6. The activities of the gases in the liquid are based on the reactor coolant equilibrium activities with 1% failed fuel as listed in Table 9.1-4. For the accident analysis, activities in the liquid have been corrected for density. The analysis follows the guidance of NRC Branch Technical Position ETSB (Effluent Treatment Systems Branch) 11-5.

The impact of the measurement uncertainty recapture (MUR) uprate on the VCT rupture dose consequences was evaluated by comparing reactor coolant equilibrium noble gas activities for 1% failed fuel between Table 9.1-4 (original design and licensing basis) and Table 14.3-16 It was determined that replacing the Table 9.1-4 1% failed fuel inventory with the Table 14.3-16 1% failed fuel inventory would result in a decrease in dose from a VCT rupture. This decrease in the dose is largely due to the smaller Kr-88 and Xe-133 content.

Using these sources and an atmospheric dispersion factor of 1.16×10^{-3} sec/m^3, and assuming a puff ground level release, the two-hour whole-body dose at the EAB is below the 10 CFR 100 limit, and below the 0.5 REM limit contained in Branch Technical Position 11-5.

14.4.2.2 **Waste Gas Decay Tank (WGDT) Rupture**

Surry has two Waste Gas Decay Tanks that collect the gases stripped from the primary coolant system by the primary coolant clean-up systems. One tank is charged with waste gases being removed from the primary system while the other tank is used to hold up the gases for decay and controlled release. The analysis of doses from rupture of a WGDT assumes rupture of a WGDT with the release of the maximum inventory allowed by Technical Specifications.

14.4.2.2.1 WGDT Analysis Assumptions

The whole body EAB dose from the rupture of a WGDT was determined based on a puff release as the product of the (l) curies released, (2) dose conversion factor for Xe-133 and

(3) EAB χ/Q. This analysis does not require any computer code. As explained in Reference 15, the WGDT control room dose was bounded by doses determined for other accident conditions. Although some iodine may be present in the tank, the amount are orders of magnitude below those considered for other accidents.

14.4.2.2.2 Dose Analysis for WGDT Rupture

The maximum WGDT inventory allowed by Surry Technical Specification 3.11 is 24,600 curies, (considered as Xe-133). The χ/Q for the EAB is 1.16×10^{-3} sec/m^3. The whole body dose conversion factor for Xenon-133 is 9.316×10^{-3} rem-m^{-3}/Ci-sec. A puff release of the maximum WGDT inventory allowed by Technical Specifications results in a whole body EAB dose less than 0.5 Rem.

14.4.3 Radioactive Liquid Release

Accidents in the auxiliary systems that could result in the release of waste liquids must necessarily involve the rupture or leaking of various pipelines, valves, tanks, and pumps.

All liquid processing components are located within the auxiliary building, fuel building, decontamination building, radwaste facility, and station yard area. Any liquid leakage or release from these components is collected in sumps and pumped to the liquid waste disposal systems (Section 11.2.3) or flows directly to the vent and drain system (Section 9.7). The auxiliary building and fuel building are of Class I design. The below ground levels of the radwaste facility are seismically designed to the requirements of RG 1.143.

The boron recovery tanks are located in the station yard area in separately diked enclosures, each of which is of sufficient capacity to retain the total liquid volume resulting from the rupture of one boron recovery tank without any overflow to areas outside the enclosure. The collected liquid is pumped either to the unruptured boron recovery tanks or to the liquid waste disposal systems. The diked enclosure is of Class I design.

Piping running between the auxiliary building and the reactor containment, between the auxiliary and fuel buildings, between the fuel building and the tanks in the yard area, and between the auxiliary building and the radwaste facility is situated below grade in concrete trenches or in special piping conduits. Liquids spilled or released from such piping are collected in sumps and pumped into the liquid waste disposal system. Accordingly, a release of waste liquids would be contained within the station and would not result in an uncontrolled release to the environment.

14.4 REFERENCES

1. Letter from C. M. Stallings, Vepco, to V. Stello, NRC, Subject: *Movement of Heavy Loads Near Spent Fuel,* dated October 9, 1978.
2. Letters from B. C. Buckley and L. B. Engle (U. S. NRC) to W. L. Stewart (Virginia Electric and Power Company), *Surry, Units 1 and 2, and North Anna, Units 1 and 2 - Removal of 45,000 MWD/MTU Batch Average Burnup Restriction (TAC Nos. M87767, M87768, M87812, and M87813)*, December 14, 1993 and April 20, 1994.
3. *Extended Burnup Evaluation of Westinghouse Fuel*, Westinghouse Electric Corporation, WCAP-10125-P-A, December 1985.
4. D. A. Walker, et al., *Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors*, NUREG/CR-5009, PNL-6258, February 1988.
5. Letter from R. H. Leasburg, Vepco, to H. R. Denton, NRC, Subject: Amendment to Operating Licenses DPR-32 and DPR-37, Surry Power Station Unit Nos. 1 and 2. Proposed Technical Specification Changes, dated September 23, 1982 (Serial No. 543).
6. Letter from R. H. Leasburg, Vepco, to H. R. Denton, NRC, Subject: Supplemental Information for Proposed Operating License Amendment, dated January 17, 1983 (Serial No. 543A).
7. Letter from J. D. Neighbors (NRC) to W. L. Stewart (Virginia Electric and Power Company) Serial No. 131, March 4, 1983.
8. Letter from Chandu P. Patel (NRC) to W. L. Stewart (Virginia Electric and Power Company) *Approval of increase the maximum allowable k-effective during refueling*, April 1986.
9. *LOCADOSE*, Bechtel Standard Computer Program NE319, Version 7.0 (Users's Manual Revision 10, Theoretical Manual Revision 10 and Validation Manual Revision 12).
10. NUREG/CR-2858, *PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations*, USNRC, 1982.
11. NUREG/CR-6331, Rev. 1, *Atmospheric Relative Concentrations in Building Wakes, ARCON96*, USNRC, 1997.
12. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, *Standard Review Plan*, NUREG-0800, Revision 2, July 1981.
13. U.S. Nuclear Regulatory Commission Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000.
14. U.S. Nuclear Regulatory Commission, Office of Standards Development, *Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere*

*Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plan*ts, Regulatory Guide 1.52, Revision 2, March 1978.

15. Letter from W. L. Stewart (Virginia Power) to U. S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Control Room Habitability, Operator Dose Assessment*, Serial Number 89-381, June 1989.
16. Letter from C. M. Stalling (Virginia Power) to U. S. Nuclear Regulatory Commission, *Response to Request for Additional Information*, Serial No. 045A/020177, May 1977.
17. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.25 (Safety Guide 25), *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, March 23, 1972.
18. Technical Report NE-1227, Rev. 1, *Assessment of Accident Radiological Consequences Using NUREG-1465 Methodology—Surry Power Station Units 1 and 2*, July 2001.
19. Letter from G.T. Bischof (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed Increase In The Lead Rod Average Burnup Limit*, Serial No. 07-0024, March 6, 2007.
20. Letter from S.P. Lingam (NRC) to D.A. Christian (Dominion), "*Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Increase in the Lead Rod Average Burnup Limit (TAC Nos. MD4716 and MD4717)*," Serial No. 07-0854, December 19, 2007.
21. Letter from K. Cotton (NRC) to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding the Use of Optimized ZIRLO™ Fuel Rod Cladding (TAC Nos. ME3343 and ME3344)*, Serial No. 10-669, December 22, 2010; as amended by Letter from K. Cotton (NRC) to D. A. Heacock (Dominion), *Surry Power Station, Unit Nos. 1 and 2, Issuance of Corrections to Amendments Regarding the Use of Optimized ZIRLO™ Fuel Rod Cladding (TAC Nos. ME3343 and ME3344)*, Serial No. 10-755, December 23, 2010.

Table 14.4-1
ACTIVITY RELEASED TO THE CONTAINMENT OR FUEL BUILDING

Isotope	Activity Released (Ci)
Elemental I-130	1.253E-02
Elemental I-131	5.333E+01
Elemental I-132	4.153+01
Elemental I-133	5.062E+00
Elemental I-135	3.648E-03
Organic I-130	9.415E-03
Organic I-131	4.006E+01
Organic I-132	3.119E+01
Organic I-133	3.842E+02
Organic I-135	2.740E-03
Kr-85	8.420E+02
Kr-88	5.261E-07
Kr-83m	4.544E-09
Kr-85m	1.540E-03
Xe-133	4.808E+04
Xe-135	8.824E+01
Xe-131m	3.752E+02
Xe-133m	8.950E+02
Xe-135m	2.927E-01

Table 14.4-2
ATMOSPHERIC DISPERSION FACTORS (χ/Qs) FOR OFFSITE CALCULATIONS

Receptor	Time Period	χ/Q value
EAB	0-2 hr	1.76E-03
LPZ	0-8 hr	2.01E-04
	8-24 hr	1.22E-04
	24-96 hr	4.18E-05
	96-720 hr	8.94E-06

Table 14.4-3
ATMOSPHERIC DISPERSION FACTORS (χ/Qs) FOR CONTROL ROOM CALCULATIONS

Time Period	Containment Equipment Access Hatch χ/Q value [a]	Containment Personnel Airlock χ/Q value [a]	Fuel Building & Containment Purge χ/Q value [b]
0-2 hours	6.74E-04	1.07E-03	6.97E-04
2-8 hours	5.18E-04	9.03E-04	5.43E-04
8-24 hours	2.22E-04	3.87E-04	2.31E-04
24-96 hours	1.66E-04	2.73E-04	1.71E-04
96-720 hours	1.20E-04	1.87E-04	1.22E-04

a. Releases through penetrations that terminate in the Auxiliary Building and Safeguards have been modeled by a bounding release from either the Containment Equipment Access Hatch or the Containment Personnel Airlock depending upon the location of the penetration.

b. Both the Fuel Building and Containment Purge exhaust through Ventilation Vent No. 2. The Containment Purge release is bounded by releases from the Containment Personnel Airlock because the Atmospheric Dispersion Factors are more limiting.

Table 14.4-4
CONTROL ROOM OCCUPANCY FACTORS

0-8 hours	1.0
8-24 hours	1.0
24-96 hours	0.6
96-720 hours	0.4

Table 14.4-5
FHA CONTROL ROOM AND OFFSITE DOSES[b]

	Control Room 30-day Dose (REM TEDE)	EAB Worst 2-hour Dose (REM TEDE)	LPZ 30-day Dose (REM TEDE)
	0.5	2.6	0.3
Regulatory Guideline Value [a]	5.0	6.3	6.3

a. 10 CFR Part 50.67 establishes TEDE dose limits for the EAB, the outer boundary of the LPZ, and for the control room for use with the alternate source term. The specified offsite dose limits are stated for evaluating reactor accidentS of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. For events with a higher probability of occurrence, e.g., FHA postulated EAB and LPZ doses should not exceed the limits established in RG 1.183. The 10 CFR 50.67 control room criterion applies to all accidents.

b. Composite results from the limiting or worst case containment and fuel building releases.

Table 14.4-6
MAXIMUM VOLUME CONTROL TANK NOBLE GA CONCENTRATION IN VAPOR PHASE WITH SMALL CLADDING DEFECTS IN ONE PERCENT OF THE FUEL RODS

Isotope	Vapor Phase Activity Concentration μCi/gm
Kr-85	66.9
Kr-85m	18.1
Kr-87	3.0
Kr-88	32.5
Xe-133	3464.2
Xe-133m	33.8
Xe-135	80.6
Xe-135m	0.004
Xe-138	0.005

14.5 LOSS-OF-COOLANT ACCIDENT

14.5.1 Major Reactor Coolant System Pipe Ruptures (Large Break Loss-of-Coolant Accident)

14.5.1.1 General

An analysis of the Emergency Core Cooling System (ECCS) performance for the postulated large-break loss of coolant accident (LOCA) has been performed in compliance with the methods and acceptance criteria of 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, *ECCS Evaluation Models,* to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Regulatory Guide 1.157 (Reference 3).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 4). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC (Reference 5).

The best-estimate large-break loss-of-coolant accident (BE-LBLOCA) analysis for the Surry Units 1 and 2 uses the Westinghouse statistical treatment of uncertainty methodology called ASTRUM, which stands for Automated Statistical Treatment of Uncertainty Method (Reference 6). The ASTRUM methodology is based on the use of the WCOBRA/TRAC computer code. The same code was adopted in the original Westinghouse Best-Estimate LBLOCA methodology approved in 1996 by the NRC. Similar to the 1996 Westinghouse methodology, ASTRUM follows the steps in the Code Scaling, Applicability and Uncertainty (CSAU) methodology of Reference 4, where the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A (Reference 6).

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95th percentile values of three parameters with a 95-percent confidence level.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A (Reference 6), as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this topical report.

14.5.1.2 **Description of Event**

A LBLOCA is a hypothetical, design-basis accident that is considered in the sizing of ECCS components. The accident is initiated by an instantaneous rupture of a Reactor Coolant System (RCS) pipe. The break type considered is either a double-ended guillotine, defined as a complete severance of the pipe resulting in unimpeded flow from either end, or a split break, defined as a partial tear. Consistent with prior Westinghouse LBLOCA methodologies, the break sizes considered vary from 1 ft^2 up to two times the pipe area. A break in the cold-leg piping between the reactor coolant pump and the reactor vessel inlet nozzle has been concluded to be the most limiting location for a large break in a PWR.

A revision to General Design Criterion 4 (GDC-4) was issued by the NRC effective May 12, 1986. In accordance with the revised rule, consideration of the dynamic effects of RCS pipe rupture may be eliminated as a design basis provided the "Leak Before Break" (LBB) analyses demonstrate that any flaw in the RCS primary loop piping which grew would become a through-wall crack with detectable leakage allowing shutdown of the plant long before a rupture would occur. LBB fracture mechanics analyses applicable to Surry have been accepted by the NRC and, in accordance with Amendment 108 to the Surry operating license, consideration of the dynamic effects of a LOCA is no longer part of the design basis. However, this change to the design basis does not affect the ECCS design basis or engineered safety feature system response. Therefore, the pipe rupture LOCA condition will still remain as a design basis for safety related systems in the RCS and conservatively envelopes all other accidents.

Should a LBLOCA occur, rapid depressurization of the reactor coolant system occurs. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the appropriate setpoint is reached and the high-head safety injection pumps are activated. The actuation and subsequent activation of the Emergency Core Cooling System, which occurs with the SIS signal, assumed the most limiting

single-failure event. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. No credit is taken in the analysis for the insertion of control rods to shut down the reactor.
2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperature.

Before the break occurs, the unit is in an equilibrium condition and the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continue to be transferred to the reactor coolant system. When the reactor coolant system depressurizes to 600 psia (nominal), the accumulators begin to inject borated water into the reactor coolant loops. As the system pressure continues to fall, the break flow, and consequently the downward core flow, are reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water. As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water. This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins. During the reflood phase, the accumulators empty and the ECCS water refills the core causing the core to be quenched. A more detailed description of the thermal-hydraulic conditions in the RCS and the fuel rod heat transfer is provided in Section 14.5.1.4.

During the refill period, the heat transfer between the reactor coolant system and the secondary system may be in either direction, depending on the relative temperatures. For the case of continued heat addition to the secondary side, secondary-side pressure increases and the main safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps.

The operation of the low-head safety injection pumps supplies water for long-term cooling. When the refueling water storage tank (RWST) is nearly empty, long-term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low-head safety injection pumps and returned to the reactor vessel. The containment spray system and the recirculation spray system operate to return the containment environment to subatmospheric pressure.

14.5.1.3 **Method of Analysis**

The methods used in the application of WCOBRA/TRAC to the large break LOCA with ASTRUM are described in References 5 and 6. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were

used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis. WCOBRA/TRAC MOD7A was used for the execution of ASTRUM for Surry Units 1 and 2 (Reference 6).

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
4. Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood proceeds continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (Reference 7) and mass and energy releases from the WCOBRA/TRAC calculation.

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, Local Maximum Oxidation (LMO), and Core-Wide Oxidation (CWO) at 95-percent probability, is described in the following sections.

1. Plant Model Development:

   In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions:

In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient". Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties.

3. Assessment of Uncertainty:

   The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of Reference 6. The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMO, CWO).

   The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or double-ended guillotine), and break size for the split break are sampled as uncertainty contributors within the ASTRUM methodology.

   Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

4. Plant Operating Range:

   The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, the methodology allows for the narrowing of the assumed ranges for some parameters if necessary to achieve additional margin.

14.5.1.4 **Analysis Assumptions**

The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 14.5-1 summarizes the operating ranges for Surry Units 1 and 2 as defined for the proposed operating conditions, which are supported by the Best-Estimate LBLOCA analysis. The operating ranges are equal to or greater than the actual plant operating conditions and thus bound plant operations. The fuel analyzed is 15 x 15 Upgraded Fuel Design with ZIRLO cladding. Table 14.5-2 summarizes the LBLOCA containment data used for

calculating containment pressure. If operation is maintained within these ranges, the LBLOCA results developed in this report using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (T_{avg}).

14.5.1.5 **Results**

The Surry Units 1 and 2 PCT-limiting transient is a double-ended cold leg guillotine break which analyzes conditions that fall within those listed in Table 14.5-1. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (Reference 5).

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figures 14.5-1 to 14.5-12. Table 14.5-17 summarizes the important sequence of events for the LBLOCA. The PCT-limiting case was chosen to show a conservative representation of the response to a large break LOCA.

1. Critical Heat Flux (CHF) Phase:

   Immediately following the cold leg rupture, the break discharge rate is subcooled and high (Figures 14.5-2 and 14.5-3). The regions of the RCS with the highest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses and the fuel rods begin to go through departure from nucleate boiling (DNB). The fuel cladding rapidly heats up (Figure 14.5-1) while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figures 14.5-7 and 14.5-12, respectively). The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

2. Upward Core Flow Phase:

   Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 14.5-4 shows the void fraction for one intact loop pump and the broken loop pump. The figure shows that the intact loop remains in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

3. Downward Core Flow Phase:

   The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top of core vapor flow (Figure 14.5-5) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure (Figure 14.5-6), the accumulators begin to inject cold borated water into the intact cold legs (Figure 14.5-9). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow, are reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water (Figure 14.5-8).

4. Refill Period:

   As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water (Figures 14.5-9 and 14.5-10). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

5. Reflood Period:

   During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system re-pressurization, and the lower core region begins to quench (Figure 14.5-11). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region. As the vessel continues to fill, the PCT location is cooled the early reflood period is terminated.

A second cladding heatup transient may occur due to boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel and vessel metal, reduces the subcooling of ECCS water in the lower plenum and downcomer. The saturation temperature is dictated by the containment pressure. If the liquid temperature in the downcomer reaches saturation, subsequent heat transfer from the vessel and other structures will cause boiling and level swell in the downcomer. The downcomer liquid will spill out of the broken cold leg and reduce the driving head, which can reduce the reflood rate, causing a late reflood heatup at the upper core elevations. Figure 14.5-12 shows only a slight reduction in downcomer level and Figure 14.5-1 indicates that a late reflood heatup does not occur.

14.5.1.6 **Conclusions**

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1853°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., “Peak Clad Temperature less than 2200°F”, is satisfied. The results are shown in Table 14.5-3.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 2.45 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., “Local Maximum Oxidation of the cladding less than 17 percent,” is satisfied. The results are shown in Table 14.5-3.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.30 percent. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than 0.30 percent. Since the resulting CWO is 0.30 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., “Core-Wide Oxidation less than 1 percent,” is satisfied. The results are shown in Table 14.5-3.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits have been satisfied. The approved methodology (Reference 5) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the peripheral assemblies (i.e., at least one face to the baffle). This conclusion is based on taking credit for the low power generation in the peripheral assemblies, and the observation that any flow redistribution that may occur would tend to benefit the inboard assemblies. For Surry Units 1 and 2, grid crushing has been predicted to occur only on the core periphery. Therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 6).

Based on the ASTRUM Analysis results (Table 14.5-3), it is concluded that Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

14.5.1.7 **Post Analysis of Record Evaluations**

In addition to the analyses presented in this section, evaluations and reanalyses may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT. The PCT, including all penalties and benefits is presented in Table 14.5-6 for the large break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

The pre-transient oxidation was not factored into the local maximum oxidation results presented for Surry Units 1 and 2 in Table 14.5-3. The maximum expected total of the normal operation (pre-transient) and LOCA transient oxidation, for any time in life, was considered for Surry Units 1 and 2. The pre-transient oxidation increases with burnup, from zero at beginning-of-life (BOL) to a maximum value at the discharge of fuel (end-of-life, or EOL). The transient oxidation has been calculated throughout the entire life of the fuel. It has been confirmed that the sum of the pre-transient plus transient oxidation remains below 17% at all times in life for Surry Units 1 and 2.

The effects of Optimized ZIRLO cladding on the BE LBLOCA analysis described herein have been considered. It has been concluded that the LOCA ZIRLO models are acceptable for application to Optimized ZIRLO cladding in ECCS performance analyses. Therefore, the use of Optimized ZIRLO cladding is deemed acceptable for Surry Units 1 and 2. No PCT penalty is required for the Surry Units 1 and 2 BE LBLOCA analysis with 15 x 15 Upgrade fuel design and Optimized ZIRLO cladding.

Adjustments were made to seven specific Containment Structural Heat Sink areas as identified in Table 14.5-5 to accommodate future structural changes. These heat sink changes were evaluated against the existing PCT limit and have shown that there is a 0°F PCT impact for a LBLOCA event as reported in Table 14.5-6.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation model used in the large break LOCA analysis. These reports constitute addenda to the analysis of record provided in the UFSAR until the overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model results in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC.

Finally, the criteria of 10 CFR 50.46 requires that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the PWR Owner's Group (PWROG), has developed an approach for compliance with the reporting requirements. This approach is documented in WCAP-13451, Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (Reference 30). Dominion provides the NRC with annual and 30-day reports, as applicable, for Surry Power Station. Dominion intends to provide future reports required by 10 CFR 50.46 consistent with the approach described in Reference 30.

14.5.2 Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes, Which Actuates Emergency Core Cooling System (Small Break Loss-of-Coolant Accident Analysis)

14.5.2.1 General

A reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated small-break LOCA (SBLOCA) has been performed in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are in compliance with 10 CFR 50.46. This analysis was performed with the NRC-approved NOTRUMP code (Reference 26) of the Westinghouse LOCA-ECCS evaluation model (Reference 27). The thermal behavior of the fuel was analyzed using the LOCTA-IV code (Reference 28).

14.5.2.2 Identification of Causes and Accident Description

A LOCA can result from a rupture of the reactor coolant system (RCS) or of any line connected to that system up to the first isolation valve. Ruptures of small cross section will cause

expulsion of the coolant at a rate that can be accommodated by the charging pumps. Breaks of greater size (up to 1 ft^2 area) are defined as small breaks and are analyzed with the NOTRUMP computer code. A rupture in the reactor coolant system results in the discharge to the containment of reactor coolant and associated energy. The result of this discharge is a decrease in coolant pressure in the reactor coolant system and an increase in containment temperature and pressure. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the pressurizer low-low pressure setpoint is reached, activating the high head safety injection pumps. The SIS actuation and subsequent activation of the Emergency Core Cooling System, which results from the SIS signal, assumes the most limiting single failure of ECCS equipment.

Before the break occurs, the unit is assumed to be in an equilibrium condition, (i.e., the heat generated in the core is being removed via the secondary system). In the small break LOCA, the blowdown phase of the small break occurs over a long time period. Thus for a small break LOCA, there are three characteristic stages: (1) a gradual blowdown in which the decrease in water level is checked by the inventory replenishment associated with safety injection, (2) core recovery, and (3) long-term recirculation. The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending on the relative temperature. For the case of continued heat addition to the secondary side, the secondary side pressure increases and the main steam safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary side flow aids in the reduction of RCS pressure. When the reactor coolant system depressurizes to approximately 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. Reflecting the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip, and the effects of pump coastdown are included in the blowdown analysis.

14.5.2.3 **Analysis Assumptions**

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the Small Break LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur and include such items as the core peaking factors, core decay heat, and the performance of the Emergency Core Cooling System. Table 14.5-12 presents the values assumed for several key parameters in this analysis. Assumptions and initial operating conditions that reflect the requirements of Appendix K to 10 CFR 50 have been used in this analysis. These assumptions include:

- The break is located in the cold leg between the pump discharge and the vessel inlet.
- Safety injection occurs both in the intact loop and the broken loop.

- Accumulator injection occurs both in the intact loop and the broken loop.
- 120% of 1971 ANS decay heat is assumed following reactor trip.
- Initial power is 2597 MWt, which is 102% of 2546 MWt (100.38% of 2587 MWt) to account for the calorimetric uncertainty
- 7% tube plugging in each steam generator.
- Safety injection system delivers borated water to the reactor coolant system 25 seconds after actuation of the SIS signal. The 25-second delay includes sufficient time to allow startup of the emergency diesel generators and loading of the charging pumps onto the emergency buses.
- Minimum assumed auxiliary feedwater flow is provided from one motor-driven and one turbine-driven auxiliary feedwater pump.

Several additional assumptions have been incorporated into the SBLOCA reanalysis described below to provide margin in key input parameters. These changes are described below.

In the previous NOTRUMP evaluation model, safety injection is delivered only in the intact loop, and the least resistance safety injection line is assumed to spill onto the containment floor. This modeling was assumed to be conservative since the additional safety injection was considered to be a benefit. This assumption was based on older evaluation models which employed a homogenous equilibrium assumption for the mixing of different phases. Sensitivity studies with the previous evaluation model determined that safety injection in the broken loop, in conjunction with the existing condensation model, resulted in a PCT penalty. Reference 27 has documented this change to the NOTRUMP evaluation model. This modeling is used in the current analysis.

To offset the penalties associated with the revised safety injection assumption, Westinghouse has incorporated a new condensation model in the NOTRUMP evaluation model. This model, referred to as the COSI model, is based on tests which modelled the configuration of the SI piping to the RCS cold leg. Use of this more realistic model for condensation of steam by pumped SI is demonstrated to provide a benefit larger than any penalty associated with injecting into the broken loop (Reference 27).

The analysis assumed a peak Heat Flux Hot Channel Factor, FQ(z), value of 2.50 and a peak Nuclear Enthalpy Hot Channel Factor, FΔh, value of 1.70. These values bound the current and anticipated power peaking limits.

This analysis also employed a K(z) envelope, the hot channel factor normalized operating curve shown in Figure 14.5-17. K(z) is a multiplier on the allowable 3-D peaking factor FQ, and by nature cannot exceed 1.0. The corresponding hot rod and hot assembly average rod axial power shapes (Figure 14.5-18) were used in the LOCTA-IV code. These power shapes have been

chosen from a generic database of potential shapes achievable during power operation by assessing the characteristics which yield limiting small break LOCA results.

The flow rates for the HHSI are provided by an engineering model of the HHSI subsystem that is based on the system configuration and measured data from the plant. This model includes allowances for imbalance between the separate injection lines, HHSI pump degradation, and instrument accuracy. The HHSI pump curves used in the model are based on the actual measured plant data for the installed HHSI pumps in each unit. For the calculated HHSI flows, it is assumed that the HHSI flow recirculation line is open above RCS pressures of 1000 psig and that it is closed below that RCS pressure. This is consistent with previous assumptions used to calculate HHSI flow rates versus RCS pressure for small break LOCAs. Other assumptions regarding HHSI system configuration, such as water levels and back pressures, are set to provide limiting conditions for the specified test condition. HHSI flow testing performed during refueling outages assesses the condition of the HHSI pumps to ensure that the actual system performance is bounded by the assumptions in the current analysis.

The analysis assumes a full core of Surry Improved Fuel (SIF) with ZIRLO™ cladding material and Performance+ design features.

14.5.2.4 **Analysis of Effects and Consequences**

14.5.2.4.1 Method of Analysis

A small break LOCA analysis was performed using the NOTRUMP computer code following the methodology and the model delineated in WCAP-10079-P-A (Reference 26) and WCAP-10054-P-A (Reference 27). The code calculates the transient depressurization of the RCS as well as describing the mass and enthalpy of flow through the break.

NOTRUMP is a general one-dimensional network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations.

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

The peak clad temperature in the core during a transient is calculated by utilizing a version of the Westinghouse LOCTA-IV code (Reference 28) for a small break analysis. The transient thermal hydraulic NOTRUMP code writes data to a file for the LOCTA-IV code. The clad thermal analysis code uses the core pressure, core mixture level, core inlet enthalpy, fuel rod power history, and uncovered core steam flow from the thermal hydraulic code NOTRUMP as input.

The assumed SIS pumped injection flowrate, shown in Figures 14.5-15 and 14.5-16, is a function of reactor coolant system pressure. The analysis assumes minimum safeguards and emergency core cooling system capability and operability.

The analysis considered cases assuming 1.5-inch, 2-inch, 2.25-inch, 2.5-inch, 2.75-inch, 3-inch, 4-inch, and 5.5-inch effective diameter cold leg break sizes. A refined break spectrum is used where any integer break size (e.g. 2 and 3 inch effective diameter) PCT is approximately equal to or greater than 1700°F, or if the PCT results are close to or greater than the corresponding LBLOCA PCT results. Several of the integer break sizes noted above exceeded 1700°F and as such a more refined break spectrum was used for this analysis with size spacing of 0.25 inches between the more limiting breaks.

14.5.2.4.2 Results

For this analysis, cases were run assuming 1.5-inch, 2-inch, 2.25-inch, 2.5-inch, 2.75-inch, 3-inch, 4-inch, and 5.5-inch effective diameter cold leg break sizes. Results of key parameters for the cases analyzed are presented in Figures 14.5-19 through 14.5-74. Table 14.5-13 presents the time sequence of events, and Table 14.5-14 summarizes the peak clad temperature for each beginning-of-life (BOL) case analyzed. Table 14.5-15 summarizes the limiting results for the limiting break burnup studies. The 2.75-inch cold leg break was found to be the most limiting BOL break size for a small break LOCA and a burnup study was performed to determine the limiting time-in-life results. For ZIRLO™–clad fuel, the time-in-life analysis resulted in a limiting peak clad temperature of 2012°F, a maximum transient local cladding oxidation level of 11.86% (cladding oxidation is also referred to as metal-water reaction and ZrO_2 reaction), and a total core oxidation of less than 1.0%. In addition, the maximum total local oxidation is less than 17% for the life of the fuel. The identified figures show the following:

- Pressurizer Pressure—Figures 14.5-19 through 14.5-26 show the calculated pressure for the different break sizes.
- Core Mixture Level—Figures 14.5-27 through 14.5-34 show that the core mixture level decreases, accompanied by the RCS depressurization, until the combined rate of the Safety Injection and the Accumulator Injection exceeds the break flow.
- Pumped SI Flow—Figures 14.5-35 through 14.5-42 show the pumped safety injection flow to the intact loops and to the broken loop.
- Core Exit Vapor Flow—Figures 14.5-43 through 14.5-50 show the core exit vapor flow.

- Hot Assembly Fluid Temperature at PCT Elevation—The fluid temperature in the hot assembly peaks at the same time as the clad temperature, with approximately the same magnitude, and is shown in Figures 14.5-51 through 14.5-58.
- Hot Rod Heat Transfer Coefficient at PCT Elevation—Figures 14.5-59 through 14.5-66 show the calculated heat transfer coefficient on the hot rod at the PCT Elevation.
- Hot Rod Clad Average Temperature at PCT Elevation—Figures 14.5-67 through 14.5-74 show the calculated hot-spot clad temperature transient. The peak clad temperature for the limiting 2.75-inch break size is 2012°F at the 12.00 foot core elevation.

14.5.2.5 **Post Analysis of Record Evaluations**

In addition to the analyses presented in this section, evaluations and reanalyses may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the PCT is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT. The PCTs, including all penalties and benefits, are presented in Table 14.5-16 for the small break LOCA. The resultant PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

An evaluation was performed to support the transition from 15 x 15 Surry Improved Fuel with ZIRLO cladding to 15 x 15 Upgrade fuel with Optimized ZIRLO cladding. The LOCA ZIRLO models utilized in the SBLOCA analysis were demonstrated to be applicable for Optimized ZIRLO. Comparisons of the hydraulic characteristics and pressure loss data used in the NOTRUMP evaluation model for the two fuel types have shown that the fuel transition will have no significant impact on the SBLOCA analysis results. This evaluation has shown that the transition to 15 x 15 Upgrade fuel, including consideration of the 15 x 15 Upgrade-specific fuel temperature and rod internal pressure data, will have minimal or negligible effect on the SBLOCA analysis. All pertinent 10 CFR 50.46 criteria continue to be satisfied.

As discussed in Section 14.5.1.7, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation models used in LOCA analyses. The requirements discussed in Section 14.5.1.7 are also applicable to the small break LOCA analysis.

14.5.2.6 **Conclusions**

The fuel clad heatup summary in Tables 14.5-14 and 14.5-15 presents results for ZIRLO™-clad fuel that are within the acceptance criteria specified by 10 CFR 50.46. The calculated peak clad temperature for the limiting 2.75-inch break is 2012°F, which is less than the 2200°F limit. The maximum transient local metal-water reaction is 11.86%, and the total local metal-water reaction is less than the embrittlement limit of 17%. The total core-wide metal-water reaction is less than the 1% limit. The results show that the clad temperature transient has peaked and sufficiently stabilized while the core is still amenable to cooling. Consequently, it is concluded that the Surry ECCS will be capable of mitigating the effects of a small break LOCA

with a maximum F_Q of 2.50 and a $F\Delta h$ of 1.70, at a thermal core power of 2597 MWt, for cores containing SIF with the ZIRLO™ cladding and Performance+ features.

An evaluation performed for a transition from 15 x 15 SIF with ZIRLO cladding to 15 x 15 Upgrade fuel with Optimized ZIRLO cladding concluded that there is no significant impact on the SBLOCA analysis results and all pertinent 10 CFR 50.46 acceptance criteria continue to be satisfied (see Section 14.5.2.5).

For the small break LOCA, the emergency core cooling system will thus meet the acceptance criteria as presented in 10 CFR 50.46, as follows:

1. The calculated peak fuel element clad temperature provides margin to the limit of 2200°F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.
3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17% is not exceeded.
4. The core remains amenable to cooling during and after the break.
5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

14.5.3 Core and Internals Integrity Analysis

The methodology presented in Sections 14.5.3.1 through 14.5.3.4 has been replaced in part by the methodology of WCAP-9401. Also, the BLODWN-2 program has been replaced in part by the MULTIFLEX computer code. Refer to Section 14.5.3.3.4 for a description of the MULTIFLEX code and its use in blowdown and force models and the WCAP-9401 methodology.

14.5.3.1 Internals Evaluation

The forces exerted on the reactor internals and the core following a LOCA are computed by employing the BLODWN-2 digital computer program developed for the space-time-dependent analysis of multiloop PWR plants.

14.5.3.2 Design Criteria

Following a LOCA, the basic requirement is that the plant shall be shut down and cooled down in an orderly manner so that fuel cladding temperature is kept within the specified limits. This implies that the deformation of the reactor internals must be kept sufficiently small so that the core geometry remains substantially intact to allow core cooling and insertion of a sufficient number of control-rod assemblies.

After the break, the reduction in water density greatly reduces the reactivity of the core, thus shutting down the core independent of the control-rod assemblies. In other words, the core is shut down whether or not the control-rod assemblies are tripped. (The subsequent refilling of the core

by the emergency core cooling system uses borated water to maintain the core in a subcritical state). Therefore, insertion of most of the control-rod assemblies gives further assurance of the ability to shut the unit down and keep it in a safe-shutdown condition. Note that the methodologies of Reference 45 and Reference 52 have been used to verify acceptability for crediting control rod insertion following a cold leg break in the assessment of long term core cooling.

Maximum allowable deflection limitations are established for those regions of the internals that are critical for unit shutdown. Allowable stress limits are adopted to ensure physical integrity of the components.

In the event of a sudden double-ended reactor coolant system pipe rupture[1] (complete severance in a few milliseconds), pressure waves are produced in the reactor, causing vertical and horizontal excitation of the components. A study has been made to analyze the response of the reactor vessel internal structures under these conditions.

14.5.3.3 **Blowdown and Force Models**

14.5.3.3.1 Blowdown Model

BLODWN-2 is a digital computer program used for calculation of local fluid pressure, flow, and density transients that occur in the reactor coolant system during a LOCA. This program applies to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM (Reference 9), which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place.

BLODWN-2 is based on the method of characteristics, wherein the resulting set of ordinary differential equations obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although one-dimensional conversation laws are employed, the code can be applied to describe three-dimensional system geometries through the use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in a pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

14.5.3.3.2 Comparison With Experimental Data

BLODWN-2 predictions have been compared with data obtained by Phillips Petroleum Company from their loss-of-flow test (LOFT) semi-scale and 1/4-scale blowdown experiments.

1. As discussed in Section 15.6.2, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop piping. However, pipe ruptures of reactor coolant branch lines are still postulated.

An example of these comparisons is shown in Figure 14.5-75, which illustrates the pressure history in the blowdown pipe for the semi-scale test #522. This was a bottom blowdown test for the "Bettis Flask No. 1" geometry, with initial uniform fluid conditions of 1268 psia and 445°F. It can be seen that the BLODWN-2 digital computer program gives good agreement in both the subcooled and the saturated regimes.

14.5.3.3.3 Force Model

BLODWN-2 evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE, which uses a detailed geometric description in evaluating the loading in reactor internals.

Each reactor component for which force calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated, summing the effects of:

1. The pressure differential across the element.
2. Flow stagnation on, and unrecovered orifice losses across, the element.
3. Friction losses along the element.

Input to the code, in addition to the BLODWN-2 pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

14.5.3.3.4 Method of Blowdown Re-Analysis

Re-analysis of the blowdown forces on the reactor vessel and internals structures for Units 1 and 2, such as the one performed for the vessel head replacements and the control rod insertion analysis following a cold leg break (Reference 45), has made use of the MULTIFLEX (References 46 & 47) computer code, rather than BLODWN-2 described above, and the methodology of WCAP-9401 (Reference 48). MULTIFLEX is an extension of the BLODWN-2 code and includes mechanical structure models and their interactions with the thermal-hydraulic system. Both versions of the MULTIFLEX code share a common hydraulic modeling scheme, with the differences confined to a more realistic downcomer hydraulic network, and a more realistic core barrel structural model that accounts for non-linear boundary conditions and vessel motion. Generally, this improved modeling results in lower, more realistic, but still conservative hydraulic forces on the core barrel. The NRC staff has accepted (References 49, 50, & 53) the use of MULTIFLEX (including MULTIFLEX 3.0) for calculating the hydraulic forces on reactor vessel internals, including the reactor core (References 51 & 52). MULTIFLEX is used in the analysis to calculate the thermal-hydraulic transient (primarily transient pressures) within the reactor vessel. The re-analysis uses the FORCE2 computer code (described in Reference 46) to post process MULTIFLEX hydraulic transient results into vertical forces as described above for

the FORCE code. Lateral forces are computed using the LATFORC code (described in Reference 46). The WCAP-9401 methodology utilizes a 3-dimensional structural model of the reactor vessel, internals, reactor core, and vessel support mechanism. LOCA forces acting on internals components are generated using the calculated transient pressures from the MULTIFLEX computer code and the FORCE2 and LATFORC codes. Horizontal and vertical responses are calculated simultaneously from the 3-dimensional model for both LOCA and seismic loading conditions.

14.5.3.4 **Response of Reactor Internals to Blowdown Forces**

14.5.3.4.1 Reactor Equipment System Model - LOCA Analysis

The response of reactor internals components due to an excitation produced by complete severance of a branch line pipe is analyzed. Assuming a pipe break occurs in a very short period of time of 1 millisecond, the rapid drop of pressure at the break produces a disturbance that propagates along the primary loop and excites the internal structures.

The LOCA break considered for Surry Units 1 and 2 consist of breaks located at the accumulator (ACC) line, pressurizer surge (PZR) line, and the residual heat removal (RHR) line. The LOCA hydraulic forcing functions (horizontal and vertical forces) that were used in the analyses were generated using MULTIFLEX 3.0 computer code described by Reference 47.

Mathematical Model of the Reactor Pressure Vessel (RPV) System

The mathematical model of the RPV system is a three-dimensional nonlinear finite element model, which represents the dynamic characteristics of the reactor vessel/internals/fuel in the six geometric degrees of freedom. The RPV system model was developed using the WECAN (Westinghouse Electric Computer Analysis) computer code. The WECAN finite element model consists of three concentric submodels connected by the nonlinear impact elements and stiffness matrices. The first submodel represents the reactor vessel shell and associated components. The reactor vessel is restrained by reactor vessel supports and by the attached primary coolant piping. The reactor vessel support system is represented by stiffness matrices.

The second submodel represents the reactor core barrel assembly (core barrel and thermal shield), lower support plate, tie plate, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel represents the upper support plate, guide tubes, support columns, upper and lower core plates, and the fuel. This submodel includes the specific properties of the Westinghouse 15 x 15 Upgraded fuel assemblies. The third submodel is connected to the first and second by a stiffness matrices and nonlinear elements.

The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure:

$$[M]\{\ddot{U}\}+[D]\{\dot{U}\}+[K]\{U\}=\{F\}$$

Where,

$[M]$ = Global inertia matrix
$[D]$ = Global damping matrix
$[K]$ = Global stiffness matrix
$\{\ddot{U}\}$ = Acceleration array
$\{\dot{U}\}$ = Velocity array
$\{U\}$ = Displacement array
$\{F\}$ = Force array, including impact, thrust forces, hydraulic forces, constraints and weight.

WECAN solves the above equation using the nonlinear modal superposition theory. An initial computer run is made to calculate the eigenvalues (frequencies) and eigenvectors (mode shapes) for the mathematical model. This information is stored, and is used in a subsequent computer run which solves the equation. The first time step performs a static solution of the equation to determine the initial displacements of the structure due to deadweight and normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of the equation. Time-history nodal displacements and impact forces are stored for postprocessing.

The following typical discrete elements from the WECAN finite element library are used to represent the reactor vessel and internals components:

- Three-dimensional elastic pipe
- Three-dimensional mass with rotary inertia
- Three-dimensional beam
- Three-dimensional linear spring
- Concentric impact element
- Linear impact element
- 6 x 6 stiffness matrix
- 18 Card stiffness matrix

- 18 Card Mass matrix
- Three-dimensional friction element

Analytical Methods

The RPV system finite element model as described above was used to perform the LOCA analysis. Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of the pressurized coolant results in the traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to the reactor vessel system consist of (a) reactor internal hydraulic loads (vertical and horizontal), and (b) reactor coolant loop mechanical loads. All of the loads are calculated individually and combined in a time-history manner.

RPV Internal Hydraulic Loads -

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle.

After a postulated break in cold leg, the depressurization path for the waves entering the reactor vessel is through the nozzle into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial wares propagate up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel; that is the fuel the region.

The region of the downcomer annulus close to the break depressurizes rapidly; however, because of the restricted flow areas and finite wave speed (approximately 3,000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and reactor pressure vessel. As the depressurization wave propagates around the downcomer annulus and up through the core, the barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of the postulated break in the hot leg, the waves follow a dissimilar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the downcomer annulus from the bottom exit of the core barrel. Thus, after a break in the hot leg, the downcomer annulus would depressurize with very little difference in pressure across the outside of the diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the downcomer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg break. Since the differential

pressure is less for a hot leg break, the downcomer annulus would be depressurized with very little difference in the pressure across the outside diameter of the core barrel.

Reference 47 describes how the MULTIFLEX computer code calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and two-phase (saturated) blowdown regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid structure interaction by accounting for the deflection of the constraining boundaries, which are represented by separate spring mass oscillator systems. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into various segments and the pressure as well as the wall motions are projected onto the plane parallel in the broken inlet nozzle. Horizontally, the barrel is divided into 10 segments; each segment consists of three separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces, which act on the 10 mass points of the beam model. Each flexible wall bounded on either side by a hydraulic flow path. The motion of the flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RPV nozzles by the primary coolant loop piping. The loop mechanical loads results from the release of normal operation forces present in the pipe prior to the separation as well as transient hydraulic forces in the reactor coolant system. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The loads existing in the pipe at the postulated break location are calculated and are “released” at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with a ramp time of 1 millisecond because of the assumed instantaneous break opening time. For breaks in the branch lines, the force applied at the reactor vessel would be insignificant. The restraints on the main coolant piping would eliminate any force to the reactor vessel caused by a break in the branch line.

Results of the Analysis

The severity of a postulated break in a reactor vessel is related to three factors: the distance from the reactor vessel to the break location, the break opening area, and the break opening time. The nature of the decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates

as it propagates toward the reactor vessel. The LOCA hydraulic and mechanical loads described in the previous sections were applied to the WECAN model of the reactor pressure vessel system.

The results of LOCA analysis include time-history displacements and nonlinear impact forces for all major components. The time-history displacements of upper core plate, lower core plate, and core barrel at the upper core plate elevation are provided as input for the reactor core evaluations. The impact forces calculated at the vessel-internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. Using appropriate postprocessors, component linear forces are also calculated.

RPV Sliding Foot Support Analysis

For analysis of the RPV sliding foot supports for both Unit 1 and 2, an NRC approved methodology (References 55, 56, & 57) is used which credits increased break opening times for RCS branch line breaks in order to better characterize LOCA forces acting on the sliding foot supports. Using these alternatively developed LOCA forces, combined with existing design basis loads, the RPV sliding foot supports are capable of supporting postulated design basis loads without the need for crediting any of the twenty cap screws between the socket plate and the nozzle pad. The alternative methodology approach utilized is described in more detail below.

For purposes of lowering LOCA forces acting on the RPV sliding foot supports, increased break opening times (BOT) shown in Table 14.5-18 are developed using an NRC approved methodology described in References 55, 56, & 57. Crediting these increased BOTs, a new thermal-hydraulic model coded in AREVAs CRAFT2 software (Reference 58) is used to develop pressure and force time-histories corresponding to postulated breaks at the following RCS branch line connections: [1] 14” residual heat removal (RHR) line (Loop 1 Hot Leg); [2] 12” safety injection (SI) line (Loop 1 Cold Leg); and [3] 12” pressurizer surge (PZR) line (Loop 3 Hot Leg). AREVAs CRAFT2 software (Reference 58) is a thermal hydraulic code that is functionally equivalent to Westinghouse’s MULTIFLEX software. Asymmetric cavity pressure (ACP) force time histories acting on the reactor coolant pumps and steam generators are also developed. AREVAs BWHIST software (Reference 59) is used to develop LOCA force time histories from the pressure time histories developed from the CRAFT2 thermal-hydraulic model and the ACP analysis. AREVAs BWHIST program (Reference 59) is used to develop LOCA force time histories from the pressure time histories developed from the CRAFT2 thermal-hydraulic model and the ACP analysis. AREVAs BWHIST program (Reference 59) is a postprocessor of CRAFT2 that converts the CRAFT2 results to input for AREVAs BWSPAN structural analysis software (Reference 60), which is functionally equivalent to Westinghouse’s WECAN software. BWSPAN is used to model the RPV and three reactor coolant loops (including large bore piping, reactor coolant pumps, steam generators, and supports) and LOCA loads at the RPV sliding foot support (and at other major RCL equipment supports) are generated. These revised LOCA forces are combined with other design basis loads and the RPV support is re-evaluated in order to demonstrate that the aforementioned twenty cap screws are no longer required to be credited. Note that LOCA reaction forces developed for major RCL equipment supports (i.e., steam

generator and RCP supports) are checked against design basis limitations for these supports to evaluate acceptability for the increased BOT and revised modeling approach. However, the Westinghouse-developed LOCA forces (using the 1 ms BOT) are still valid for the design of the major RCL equipment supports.

14.5.3.4.2 Reactor Equipment System Model Seismic Analysis

Nonlinear dynamic seismic analysis of the RPV system (reactor pressure vessel, internals, and fuel) includes the development of the system finite element model and the synthesized time-history accelerations.

The basic mathematical model for seismic analysis is essentially similar to the LOCA model except in that the seismic model includes the hydrodynamic mass matrices in the vessel/barrel annulus to account for the fluid-interactions. The RPV system finite element model for the nonlinear time-history seismic analysis consists of three concentric structural submodels connected by nonlinear impact elements and linear stiffness matrices. The first submodel represents the reactor vessel shell and its associated components. The reactor vessel is restrained by reactor vessel support system in the system finite element model was represented by stiffness matrices.

The second submodel represents the reactor core barrel, thermal shield, lower support plate, tie plates, and the secondary core support components. These submodels are physically located inside the first, and are connected to them by stiffness matrices at the vessel-internals interfaces. Core barrel to reactor vessel shell impact is represented by nonlinear elements at the core barrel flange, upper support plate flange, core barrel outlet nozzles, and the lower radial restraints.

The third and innermost submodel represents the upper support plate assembly consisting of guide tubes, upper support columns, upper and lower core plates, and the fuel. The fuel assembly simplified structural model incorporated in to the RPV system model. The third submodel is connected to the first and second submodel by stiffness matrices and nonlinear elements.

As mentioned earlier, fluid-structure or hydroelastic interaction is included in the reactor pressure vessel model for seismic evaluations. The horizontal hydroelastic interaction is significant in the cylindrical fluid flow region between the core barrel and the reactor vessel annulus. Mass matrices with off-diagonal terms (horizontal degrees of freedom only) attach between nodes on the core barrel, thermal shield, and the reactor vessel. The diagonal terms of the mass matrix are similar to the lumping of water masses to the vessel shell, thermal shield, and core barrel. The off-diagonal terms reflect the fact that all of the water mass does not participate when there is no relative motion of the vessel and the core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight-forward, quantitative manner.

The matrices are a function of the properties of two cylinders with the fluid in the cylindrical annulus, specifically, inside and outside radius of the annulus, density of the fluid and

length of the cylinders. Vertical segmentation of the reactor vessel and the core barrel allows inclusion of radial variations along their heights and approximates the effects of beam mode deformation. These mass matrices were inserted between the selected nodes on the core barrel, thermal shield, and the reactor vessel.

The seismic evaluations are performed by including the effects of simultaneous application of time-history accelerations in three orthogonal directions. The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of discrete members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformations, the element global matrices and arrays are assembled into global structural matrices and arrays, and used for dynamic solution of the system equations.

14.5.3.4.3 Allowable Deflection and Stability Criteria

14.5.3.4.3.1 *Fuel Assemblies.* The limitations for this case are related to the stability of the thimbles in the upper end. The upper end of the thimbles cannot experience stresses above the buckling compressive stresses, because any buckling of the upper end of the thimbles distorts the guide line and could affect the free fall of the control-rod assembly. The buckling stress for the thimbles is 62,300 psi, and the yield stress is 62,500 psi.

14.5.3.4.3.2 *Upper Core Package.* The local deformation of the upper core plate where a guide tube is located shall be less than 0.100 inch. This deformation causes the plate to contact the guide tube, since the clearance between plate and guide tube is 0.1 inch. This limit prevents the guide tubes from being put in compression.

For a plate local deformation of 0.150 inch, the guide tube is compressed and deformed transversely to the established upper limit, and consequently the value of 0.150 inch. is adopted as the maximum core plate local deformation, with an allowable of 0.100 inch.

14.5.3.4.3.3 *Upper Core Barrel.* The upper barrel deformation has the following limits:

1. To ensure reactor trip and to avoid disturbing the control-rod assembly guide structure, the barrel cannot interfere with any guide tubes. This condition requires a stability check to ensure that the barrel does not buckle under the accident loads. The minimum distance between guide tube and barrel is 9 inches. This value is adopted as the limit above which "no loss of function" can no longer be guaranteed. An allowable deflection of 4.5 inches has been selected.

2. To ensure core cooling, the outward movement of the upper barrel must be such that the inlet flow from the unbroken cold legs is not impaired. From this condition an outward barrel deflection of 6 inches in front of the inlet nozzle has been established as the no-loss-of-function value. An allowable deflection of 3 inches has been selected.

14.5.3.4.3.4 *Control-Rod Assembly Guide Tubes.* The guide tubes in the upper core support package housing control-rod assembly required for unit shutdown have the following deflection limit: the maximum horizontal deflection of a beam should not exceed 1.75 inches over the length of the guide tube. An allowable distortion of 1.0 inch has been selected.

14.5.3.4.3.5 *Allowable Stress Criteria.* The allowable stress criteria fall into two categories depending on the nature of the stress state (membrane or bending). A direct or membrane state of stress has a uniform stress distribution over the cross section. The allowable (maximum) membrane or direct stress is taken to be equal to the stress corresponding to 20% of the uniform material strain or the yield strength, whichever is higher. For unirradiated type 304 stainless steel at operating temperature, the stress corresponding to 20% of the uniform strain is 39,500 psi. For irradiated type 304 stainless steel, the stress limit is higher.

For a bending state of stress, the strain is linearly distributed over a cross section. The average strain value is one-half of the outer fiber strain where the stress is a maximum. Thus, by requiring the average bending stress to satisfy the allowable criterion for the direct state of stress, the average absolute strain may be 20% of the uniform strain. Consequently, the outer fiber strain may be 40% of the uniform strain. The maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated type 304 stainless steel operating temperatures, the stress-strain curve gives the maximum stress intensity as 50,000 psi. For irradiated type 304 stainless steel, the stress limit is higher; therefore, it is conservative to use the unirradiated value.

For combinations of membrane and bending stresses, the maximum allowable stress is taken to be equal to the maximum stress corresponding to the strain distribution having the maximum outer fiber strain not in excess of 40% uniform strain and average strain not in excess of 20% uniform strain. Analogous to the uniaxial case, the maximum allowable membrane and total stress intensities for multiaxial stress distributions are 39,500 psi and 50,000 psi.

14.5.3.5 **Effects of LOCA and Safety Injection on the Reactor Vessel**

The effects of injecting safety injection water into the reactor coolant system following a postulated LOCA have been analyzed. WCAP 7304L gives a description of the program associated with this analysis. Below is a summary of the conditions that were considered.

For the reactor vessel, three modes of failure are considered: the ductile mode, the brittle mode, and the fatigue mode.

14.5.3.5.1 Ductile Mode

The failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall, using the material yield stress specified in Section III of the ASME Code. The combined pressure and thermal stresses during safety injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at various times during the safety injection transient.

The results of the analyses showed that local yielding may occur in approximately the inner 12% of the base metal and in the cladding.

14.5.3.5.2 Brittle Mode

The possibility of a brittle fracture of the irradiated core region has been considered from both a transition temperature approach and a fracture mechanics approach.

The failure criterion used for the transition temperature evaluation is that a local flaw cannot propagate beyond any given point where the applied stress remains below the critical propagation stress at the applicable temperature at that point.

The results of the transition temperature analysis showed that the stress-temperature condition in the outer 65% of the base metal wall thickness remains in the crack arrest region at all times during the safety injection transient. Therefore, if a defect were present in the most detrimental location and orientation (i.e., a crack on the inside surface and circumferentially directed), it could not propagate any farther than approximately 35% of the wall thickness, even considering the worst-case assumptions used in this analysis.

Both a local crack effect and a continuous crack effect have been considered, with the latter requiring the use of a rigorous finite element axisymmetric code. The results of the fracture mechanics analysis, considering the effects of water temperature, heat transfer coefficients, and fracture toughness of the material as a function of time, temperature, and irradiation show that the integrity of the reactor vessel is maintained throughout the life of the unit.

14.5.3.5.3 Fatigue Mode

The failure criterion used for the failure analysis is the one presented in Section III of the ASME Code. In this method, the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowance usage factor of one.

The results of this analysis show that the combined usage factor never exceeds 0.2, even after assuming that the safety injection transient occurs at the end of unit life.

In order to cause a fatigue failure during the safety injection transient at the end of unit life, it has been estimated that a wall temperature of approximately 1100°F is needed at the most critical area of the vessel (instrumentation tube welds in the bottom head).

The design basis of the emergency core cooling system ensures that the maximum cladding temperature does not exceed the clad melting temperature. This is achieved by prompt recovery of the core through flooding, with the passive accumulator and the active injection systems. Under these conditions, a vessel temperature of 1100°F is not considered a credible possibility, and the evaluation of the vessel under such elevated temperatures is a hypothetical case.

For the ductile failure mode, such hypothetical rise in the wall temperature would increase the depth of local yielding in the vessel wall.

The results of these analyses show that the integrity of the reactor vessel is never violated.

The safety injection nozzles have been designed to withstand 10 postulated safety injection transients without failure. This design and associated analytical evaluation were made in accordance with the requirements of Section III of the ASME Code.

The maximum calculated pressure plus thermal stress in the safety injection nozzle during the safety injection transient was calculated to be approximately 50,900 psi. This value compares favorably with the code-allowable stress of 80,000 psi.

These 10 safety injection transients are considered along with all the other design transients for the vessel in the fatigue analysis of the nozzles. This analysis shows the estimated usage factor for the safety injection nozzles to be 0.47, which is well below the code-allowable value of 1.0.

The safety injection nozzles are not in the highly irradiated region of the vessel, and thus they are considered ductile during the safety injection transient.

The effect of the safety injection water on the fuel assembly grid springs has been evaluated and, due to the fact that the springs have a large surface-area-to-volume ratio, and are in the form of thin strips, they are expected to follow the coolant temperature transient with very little lag; hence, no thermal shock is expected, and the core cooling is not compromised.

Evaluations of the core barrel and thermal shield have also shown that core cooling is not jeopardized under the postulated accident conditions.

14.5.4 Containment Iodine Removal by Spray System

The spray system is designed to reduce post-accident containment pressure by condensing steam and to adsorb inorganic or particulate iodine present in the containment atmosphere by chemical spray. The spray system design bases and description are discussed in Section 6.3.1.

The analyses establishing the amount of radioiodine in the containment following a LOCA are documented in References 44 and 54. The spray removal coefficients used in the analyses for elemental and organic iodine were assumed to be 10 hr-1 and 0 hr-1, respectively, consistent with NUREG-0800 (Standard Review Plan) Section 6.5.2 (Reference 15). The spray removal coefficients used in the analyses for particulate iodine and other aerosols were as indicated in Table 14.5-8. The spray removal coefficients in Table 14.5-8 were developed in accordance with the methodology of NUREG/CR 5966 (Reference 14). A maximum decontamination factor of 200 was applied to the elemental iodine. A two-region model was used to calculate the effective spray removal coefficients. The spray is effective over 60% of the containment volume until containment spray is terminated, at which time the sprayed volume is reduced to 18%. There is assumed to be a mixing rate of 2 unsprayed volumes per hour.

14.5.5 Environmental Consequences of Loss-of-Coolant Accident (LOCA)

The assumed design basis accident LOCA is defined as the double-ended guillotine failure of a cold leg reactor coolant pipe, the total loss of coolant through such a double-ended failure, a total loss of offsite station power, where that is conservative, the availability of only minimum safeguards, and release of the core fission product inventory indicated in Table 14.5-10 to the reactor containment atmosphere. The core iodines released during the LOCA take the following chemical and physical forms:

1. 4.85% elemental
2. 0.15% organic
3. 95.0% particulate

This section describes the method and results of the radiological analyses for the design basis accident. The analyses include TEDE doses from two sources: dose from the containment leakage plume and the dose due to ECCS leakage for 30 days following the accident. Doses were calculated at the exclusion area boundary, at the low population zone boundary, and in the control room. The LOCA dose analyses discussed below assume operation at the uprated power.

The methodology used to evaluate the control room and offsite doses resulting from a LOCA was consistent with the NRC Standard Review Plan (References 15, 17, 18, & 34), and the Regulatory Guide 1.183 (Reference 35). Core radionuclide inventory was based on a power level of 2605 MWt which is slightly conservative compared to the uprated power level for Surry of 2587 MWt plus 0.38% for instrument uncertainty.

Regulatory Guide 1.183 (Reference 35), provides detailed guidelines for calculating doses from a LOCA in a PWR. Doses from all postulated release paths to the environment are calculated as described in the SRP, and compared with 10 CFR 50.67 exposure criteria. Radiological consequences of both containment leakage and post-LOCA leakage from Engineered Safety Feature (ESF) system components outside containment, (including backleakage into the RWST) were considered.

To account for manual realignment of the safeguards area ventilation system to filtered exhaust, a 30-minute delay in filtration, which corresponds to the earliest time for recirculation mode transfer, is included in the analysis of doses resulting from a LOCA. Surry Units 1 and 2 share a single fuel building. Prior analyses of the fuel handling accident required that the auxiliary ventilation system be aligned in the refueling mode during fuel handling. This alignment was necessary unless sufficient decay had occurred since reactor shutdown to eliminate the need for filtration of radioiodine releases from postulated fuel handling accidents. With this alignment, a manual action was required to enable filtration of the exhaust from the safeguards area after a safety injection (SI) signal. Currently, air filtration is not required to mitigate a fuel handling accident, and procedural controls have been established to eliminate operating with automatic alignment defeated, but this action is still conservatively modeled. Additionally, while modeled as

a 30-minute delay, Reference 54 indicates that realignment of the safeguards area ventilation system is not actually required until just prior to recirculation mode transfer, when contaminated sump water is recirculated outside containment. The Surry core radionuclide inventory was determined from calculations using the ORIGEN2 computer code that conservatively modeled a representative Surry core-loading plan.

Surry has a subatmospheric containment system. During the first hour following the accident containment pressure is analyzed to remain less than 45 psig. A 0.1 volume percent per day containment leak rate was used for the first hour after the LOCA, which corresponds to a maximum containment pressure of 45 psig. The original design criterion required that within one hour, containment pressure be calculated to return to subatmospheric conditions. This original design criterion was modified in conjunction with the analyses for implementation of the alternative source term and, subsequently, Generic Letter 2004-02. The modified criteria require that, following the LOCA, the containment pressure be less than 1.0 psig within 1 hour and less than 0.0 psig within 4 hours. Therefore, from 1 hour to 4 hours after a LOCA, a 0.029 volume percent per day leak rate was assumed, which corresponds to a maximum containment pressure of 1.0 psig. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

When the containment pressure is subatmospheric, any leakage would be into the containment. Therefore, no containment leakage is assumed after the fourth hour. Surry does not have a vent purge system that has to be considered as a LOCA release pathway.

The following containment spray removal coefficients were used:

Elemental Iodine = 10 per hour

Organic Iodine = 0 per hour

Particulate Iodine and other Particulates = per Table 14.5-8

Sprayed Volume = 60% (reduced to 18% at 1.14 hrs)

Mixing Rate = 2 unsprayed volumes/hour

Spray Start Time = 100 seconds (recirculation spray is not credited until 1.14 hrs)

Based on a review of the Basis for Surry Technical Specification 4.20, Regulatory Guide 1.25, and Regulatory Guide 1.52, the following control room and auxiliary building ventilation system filters efficiencies were used for Surry:

Iodine Type	Filter Efficiency
Elemental	90%
Methyl	70%
Particulate	99%

The 90% elemental and 70% methyl iodine removal efficiencies are indicated in Regulatory Guides 1.25 and 1.52 (References 19 & 37) as being appropriate for 2-inch charcoal filters without humidity control. The 99% efficiency for particulate iodine is based on the use of HEPA filters.

The ESF leakage was assumed to be 2 times the total allowable ECCS leakage of 4800 cc/hour (Section 14.5.5.3) and the total allowable back-leakage into the RWST of 100 cc/minute or 6000 cc/hour. The RWST release to the atmosphere is modeled at 1000 cfm through the safeguards building and out ventilation vent No. 2. Filtration by the auxiliary building ventilation system was credited for the portion of the ECCS leakage that occurs in the Safeguards Building, primarily the 964 cc/hour of Outside Recirculation Spray System leakage identified in Section 14.5.5.3, and the RWST backleakage. The release of radionuclides to the environment was determined both for containment leakage and Engineered Safety Feature components leakage.

For the ECCS leakage dose calculation, forty percent of the core iodine inventory is assumed to be released to the sump. The water in the ESF system at Surry, including RWST back-leakage, is taken from the containment sump at temperatures less than 212°F except for a short period at the beginning of the accident. During the period of time that the water temperature in the sump exceeds 212°F, the flashing fraction is less than 10%. Therefore, Surry meets the requirements for assuming 10% of the iodine in the ESF system leakage becomes airborne (Reference 35). Of the 10% of the iodine that becomes airborne, 97% is modeled as elemental and 3% organic (Reference 35). All of the other radionuclides as indicated in Table 14.5-10 as being in the containment sump remain in the liquid phase (Reference 35) of ESF leakage. Therefore only the airborne iodine portion of radionuclides released through ESF leakage has any consequence for EAB. LPZ, and control room doses.

14.5.5.1 Methodology to Determine Atmospheric Dispersion Factors, Control Room Occupancy, and Breathing Rates

The parameter χ/Q defines the ratio of radionuclide concentration (χ in curies/m^3) to release rate (Q in curies/second). It depends on the site meteorology (average wind speed and atmospheric stability) and distance between source and receptor. The EAB and LPZ atmospheric dispersion factors (χ/Q) are given in Table 14.5-7 and were determined based on the PAVAN (NUREG/CR-2858) methodology (Reference 33) using meteorological data for 1994 to 1998. The "wake-credit not allowed" scenario of the PAVAN results was used, since the closest point of both the EAB and LPZ from the onsite release points is greater than 10 'building heights' of the containment dome (the tallest wake-producing structure).

The control room χ/Q values were determined with the ARCON96 (Reference 32) methodology and meteorological data form the 1982 through 1986 time period. These values are listed in Table 14.5-7. Wake effects were considered in calculating the atmospheric dispersion factors for all the onsite receptor points. It was conservatively assumed that only the portion of the

reactor containment dome that is higher than the auxiliary building roof is accounted for in determining the magnitude of the wake effects. Additionally, further conservatism was introduced by only considering one containment dome for wake effect impacts. All releases were modeled as ground-level releases even when the source point was elevated (e.g., ventilation vent No. 2).

The control room occupancy factors in Table 14.5-9 were also incorporated into the dose calculations to reflect that personnel would not be exposed to the released activity 100% of the time over the entire 30 day period. These occupancy factors are based on the guidance from Reference 35. The breathing rate used for the control room dose calculations was 3.5×10^{-4} m^3/sec. For the first 8 hours, the breathing rate for offsite dose calculations was 3.5×10^{-4} m^3/sec. The offsite breathing rates for the time periods from 8 to 24 hours and from 24 hours until the end of the accident, were 1.8×10^{-4} m^3/sec and 2.3×10^{-4} m^3/sec respectively.

14.5.5.2 **LOCADOSE Model for Surry LOCA Analysis**

LOCADOSE (Reference 31) was used to model the release of radionuclides for a LOCA at Surry. This computer code system first calculates radionuclide concentrations and releases to the environment. The LOCADOSE computer code system modeled a LOCA at Surry with six volumes: 1) the environment, 2) the containment sump, 3) the portion of the containment covered by the Containment Chemical Spray System, 4) the portion of the containment not covered by the Chemical Spray System, 5) the RWST, and 6) the control room. The volumes used in the computer model were:

		0–1.14 hours	> 1.14 hours
Unsprayed containment volume	=	7.45×10^5 ft^3	1.528×10^6 ft^3
Sprayed containment volume	=	1.12×10^6 ft^3	3.353×10^5 ft^3
Sump volume	=	5.83×10^4 ft^3	
RWST Volume	=	5.335×10^4 ft^3	
Control room volume	=	2.23×10^5 ft^3	

The transport of radionuclides to the environment is modeled by specifying flow rates between the various volumes modeled. The mixing between the sprayed and unsprayed containment volumes was modeled based on 2 unsprayed volumes per hour. The containment leakage to the environment was modeled as 0.1 volume percent per day for the first hour. From 1 hour until 4 hours after the LOCA, a 0.029 volume percent per day leak rate was used. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment. The appropriate containment leakage rates, based upon time in the accident, were applied proportionately to both the sprayed and unsprayed containment volumes during the first four hours of the LOCA.

ESF leakage to the environment is modeled at twice the potential leakage rates specified in Tables 6.2-6 and 6.3-2 and twice the RWST allowable back-leakage of 6000 cc/hour:

Twice the Allowable Leak Rate

Time Period	ECCS filtered leakage (cc/hour)	ECCS unfiltered leakage (cc/hour)	RWST filtered back-leakag e (cc/hour)	Comments
0–0.25 hours	0	0	0	
0.25–0.5 hours	0	1928	0	earliest start of the RS System
0.5–720 hours	1928	7672	12,000	earliest RMT

RWST back-leakage is the modeled ESF system leakage through ECCS check valves into the RWST after switching to the recirculation cooling mode.

LOCADOSE utilizes the χ/Q values discussed in Section 14.5.5.1 to determine the radionuclide concentrations outside the control room and at the EAB and LPZ dose points. Radionuclide transport into the control room is then modeled by specifying flow rates from the environment outside the control room.

The control room ventilation system was modeled consistent with Reference 32 and Surry control room ventilation system design and operation. An unfiltered inleakage of 500 cfm was modeled from time 0 to 30 days. The control room was assumed to be isolated at the start of the event based on a SI signal. After the first hour, a filtered intake of 1000 cfm was modeled. Control room ventilation filter efficiencies are indicated in Section 14.5.5.

The LOCADOSE code system calculates radionuclide releases to the environment and radionuclide concentrations versus time in each volume. Dose conversion factors, occupancy factors, and breathing rates are then used along with the radionuclide concentrations to calculate doses. The breathing rates and occupancy factors used for the dose calculations were discussed in Section 14.5.5.1. The LOCADOSE code system uses dose conversion factors based on Federal Guidance Report Nos. 11 and 12 (References 38 & 39) to determine the TEDE doses.

14.5.5.3 **Results of Dose Calculations for LOCA**

The calculated LOCA doses are given in Table 14.5-11. It should be noted that the control room TEDE dose does not include the dose due to direct shine from containment due to the control room wall thickness being at least 18 inches thick (Reference 17). The calculated doses are less than the 10 CFR 50.67 limits for the EAB and LPZ, and the control room.

Filtration by the auxiliary building ventilation system was credited for the portion of the ECCS leakage that occurs in the Safeguards Building, primarily the Outside Recirculation Spray System leakage and RWST backleakage. Filtration was not credited for the portion of the ECCS leakage that occurs in the Auxiliary Building, primarily the SI and Charging System leakage.

However, only certain areas (the charging pump cubicles and safeguards) are provided with dedicated exhaust paths to the filters. This has the potential to lead to releases to the environment that may bypass the auxiliary building filters. However, since no filtration credit is taken (in areas without dedicated exhaust paths to filters) and all atmospheric dispersion factors were modeled as ground releases (Section 14.5.5.1) this analysis remains conservative. Therefore, the maximum allowable unfiltered leakage is limited to the total allowable ECCS leakage of 4800 cc/hour, and the total allowable RWST back-leakage of 6000 cc/hour. The total allowable ECCS leakage of 4800 cc/hour includes SI and Charging Systems leakage of 3836 cc/hour and Outside Recirculation Spray System leakage of 964 cc/hour.

14.5.6 Summary

For breaks up to and including the double-ended guillotine break of a cold leg reactor coolant pipe, the emergency core cooling system with minimum safeguards will limit the clad temperature to below the melting temperature of the cladding and ensure that the core will remain in place and substantially intact, with its essential heat transfer geometry preserved. The emergency core cooling system design meets the core cooling criteria for all cases. This is confirmed by the results of the limiting cases for the small break and large break LOCA sensitivity analyses. The emergency core cooling system components meet the acceptance criteria throughout the range of break sizes with the high-head safety injection pumps mitigating the smaller breaks and the accumulators in conjunction with the pumped safety injection flow mitigating the larger breaks.

The design of the fuel assemblies and the core support structures is such that the pressure oscillations and flow transients resulting from any LOCA can be accommodated without changes that would affect the capability of the safety injection system to perform its required function.

The calculated TEDE doses at exclusion area boundary, low population zone boundary, and in the control room resulting from a design basis LOCA are within the regulatory limits stated in 10 CFR 50.67.

14.5 REFERENCES

1. Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, *Surry Power Station Units 1 and 2—Proposed Technical Specifications Change—Surry Improved Fuel Assembly,* Serial No. 87-188, May 26, 1987.
2. Letter from C. P. Patel (NRC) to W. L. Stewart (Va. Electric & Power Co.), *Surry Units 1 and 2—Issuance of Amendments Re: Control Rod Assemblies and Surry Improved Fuel,* Serial 88-209, January 6, 1988.
3. Regulatory Guide 1.157, *Best-Estimate Calculations of Emergency Core Cooling System Performance,* USNRC, May 1989.

4. NUREG/CR-5249, *Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident,* B. Boyack, et. al., 1989.

5. Bajorek, S.M., et. al., 1998, *Code Qualification Document for Best-Estimate LOCA Analysis,* WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1.

6. WCAP-16009-P-A, *Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),* (Proprietary), January 2005.

7. *Containment Pressure Analysis Code (COCO),* WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.

8. Letter from C. M. Stallings, Virginia Electric and Power Company, to E. G. Case, NRC, Serial No. 344, August 9, 1977.

9. S. Fabric, *Computer Program WHAM for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks*, Kaiser Engineers Report No. 67-49-R, November 1967.

10. J. J. DiNunno et al., *Calculation of Distance Factors for Power and Test Reactor Sites,* USAEC, TXD 14844, March 1962.

11. Title 10 Code of Federal Regulations, Part 100 - Reactor Site Criteria.

12. Letter from C. M. Stallings, Vepco, to E. G. Case, NRC, Serial No. 587, dated December 28, 1977.

13. Letter from A. K. Banerjee, Stone & Webster, to W. C. Spencer, VEPCO, Letter No. NUS-10, 645, dated February 2, 1984.

14. NUREG/CR-5966, *A Simplified Model of Aerosol Removal by Containment Sprays,* D. A. Powers and S. B. Burson, Sandia National Laboratories, Albuquerque, New Mexico, 1993.

15. Standard Review Plan 6.5.2, *Containment Spray as a Fission Product Cleanup System*, Rev. 2, 1988.

16. Letter from J. Stolz, Chief Light Water Reactors, Branch 1 to W. L. Proffitt, dated July 9, 1976.

17. Standard Review Plan 6.4, *Control Room Habitability System*, Rev. 2, 1981.

18. Standard Review Plan 15.6.5, *Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary*, Rev. 2, 1981.

19. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.25 (Safety Guide 25), *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors,* March 23, 1972.

20. Letter from W. L. Stewart, Vepco, to U.S. NRC, Serial No. 89-381, dated June 1, 1989.

21. Letter from W. L. Stewart, Vepco, to U.S. NRC, Serial No. 89-381A, dated October 26, 1989.

22. Letter from C. M. Stallings, Vepco, to U.S. NRC, Serial No. 045A/020177, dated May 9, 1977.

23. Letter from B. C. Buckley, U.S. NRC, to Vepco, dated December 26, 1991.

24. Letter from S. A. Varga, U.S. NRC, to Vepco, dated August 1, 1988.

25. Letter from S. A. Varga, U.S. NRC, to Vepco, dated June 23, 1981.

26. Meyer, P. E.: *NOTRUMP, A Nodal Transient Small Break and General Network Code,* WCAP-10079-P-A, August 1985.

27. Lee, N., et al.: *Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code,* WCAP-10054-P-A, August 1985, and Addendum 2, Revision 1, *Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code*, *Safety Injection Into the Broken Loop and COSI Condensation Model,* July 1997.

28. Bordelon, F. M., et al.: *LOCTA-IV Program: Loss of Coolant Transient Analysis,* WCAP-8301, June 1974.

29. Letter from W. L. Stewart, Virginia Electric and Power Company, to NRC, Surry Power Station Units 1 and 2—Proposed Technical Specification Changes—$F\Delta h$ Increase/Statistical DNBR Methodology, Serial No. 91-374, dated July 8, 1991.

30. J. S. Ivey, M. Y. Young, *Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting*, WCAP-13451, October 1992.

31. LOCADOSE, Bechtel Standard Computer Program NE319, Version 4.1a (User's Manual Revision 4A and Theoretical Manual, Revision 4).

32. NUREG/CR-6331, Rev. 1, *Atmospheric Relative Concentrations in Building Wakes, ARCON96,* USNRC, 1997.

33. NUREG/CR-2858, *PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations,* USNRC, 1982.

34. Standard Review Plan 15.0.1, *Radiological Consequence Analyses Using Alternative Source Terms,* Rev. 0, July 2000.

35. U. S. Nuclear Regulatory Commission Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,* July 2000.

36. NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants,* L. Soffer et al., February 1995.

37. U.S. Nuclear Regulatory Commission, Office of Standards Development, *Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants,* Regulatory Guide 1.52, Revision 2, March 1978.

38. Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,* EPA 520/1-88-020, Environmental Protection Agency, Washington D.C., 1988.

39. Federal Guidance Report No. 12, *External Exposures to Radionuclides in Air, Water and Soil,* EPA 420-r-93-081, Environmental Protection Agency.

40. *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR Part 50, Appendix I,* Regulatory Guide 1.109, Revision 1, October 1977.

41. Letter from J. P. O'Hanlon to NRC, Serial No. 94-509, *Virginia Electric and Power Company, Surry Power Station, Units 1 and 2, Proposed Technical Specification Changes to Accommodate Core Uprating*, August 30, 1994.

42. Letter from J. P. O'Hanlon to NRC, Serial No. 94-509A, *Virginia Electric and Power Company, Surry Power Station, Units 1 and 2, Core Uprating Licensing Report Revision Steam Generator Tube Rupture Reanalysis*, February 6, 1995.

43. Letter from B. C. Buckley (NRC) to J. P. O'Hanlon, *Surry Units 1 and 2—Issuance of Amendments RE: Uprated Core Power (Serial No. 94-509) (Tac Nos. M90364 and M90365)*, August 3, 1995.

44. Technical Report NE-1227, Rev. 1, *Assessment of Accident Radiological Consequences Using NUREG-1465 Methodology—Surry Power Station Units 1 and 2*, July 2001.

45. *Control Rod Insertion Following a Cold Leg LOCA Generic Analyses for 3-Loop and 4-Loop Plants,* WCAP-15704-P, October 2001.

46. K. Takeuchi, et al., *MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics*, WCAP-8708-P-A, Westinghouse Proprietary Class 2/WCAP-8709-A, NES Class 3 (Non-Proprietary), September 1977.

47. K. Takeuchi, et al., *MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model*, WCAP-9735, Revision 2, Westinghouse Proprietary Class 2/WCAP-9736, Revision 1, Non-Proprietary, February 1998.

48. WCAP-9401-P-A (Proprietary), WCAP-9402-NP-A (Non-proprietary), *Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly,* August 1981.

49. Letter, T. H. Essig (USNRC) to Lou Liberatori (WOG), *Safety Evaluation of Topical Report WCAP-15029, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,' (TAC NO. MA1152),* November 10, 1998 (Enclosure 1 - Safety Evaluation Report).

50. Letter from John F. Stang (USNRC) to Robert P. Powers (Indiana Michigan Power Company), *Issuance of Amendments - Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474),* December 23, 1999.

51. R. E. Schwirian, et al., *Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,* WCAP-15029-P-A, Westinghouse Proprietary Class 2/WCAP-15030-NP-A, Revision 0, non-proprietary, January 1999.

52. WCAP-15245 (proprietary), WCAP-15246 (non-proprietary), *Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2,* May 28, 1999.

53. Letter, Robert L. Tedesco (USNRC) to T. M. Anderson, (Westinghouse), Acceptance for Referencing Topical Report WCAP-9401 (Proprietary)/WCAP 9402 (Non-Proprietary).

54. Technical Report NE-1460, Rev. 1, *Implementation of GOTHIC Containment Analyses and Revisions to the LOCA Alternate Source Term Analysis to Support Resolution of NRC GL 2004-02 for Surry Power Station,* July 2006.

55. AREVA Topical Report 43-BAW-10132P-A, *Analytical Methods Description, Reactor Coolant System Hydrodynamic Loading During a Loss-of-Coolant Accident*, 1979 (AREVA Proprietary).

56. AREVA Topical Report 43-1621-00, *Effects of Asymmetric LOCA Loads-Phase II*, July 1980 (BAW-1621) (AREVA Proprietary).

57. AREVA Topical Report 77-1126594-00, *Effects of Asymmetric LOCA Loadings-Phase II, Supplement 1, Responses to NRC Questions,* June 1981 (BAW-1621, Supp. 1) (AREVA Proprietary).

58. CRAFT2, Program Version 31.0 (AREVA Proprietary), *Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant.*

59. BWHIST, Version 3.0 (AREVA Proprietary), *Force Evaluation and Pressure Integration*.

60. BWSPAN, Version 13.0 (AREVA Proprietary), *Linear Static and Dynamic Analysis*.

Table 14.5-1

PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Parameter	As-Analyzed Value or Range	Operating Range or Target Value
1.0 Plant Physical Description		
a. Dimensions	Nominal	N/A
b. Pressurizer location	On an intact loop	N/A
c. Hot assembly location	Anywhere in core[a]	N/A
d. Hot assembly type	15 x 15 Upgrade fuel design	15 x 15 Upgrade fuel design
e. Steam generator tube plugging level	≤ 7%	0% ≤ SGTP ≤ 7% (in any one SG)
f. Fuel assembly type	15 x 15 Upgrade fuel with ZIRLO or Optimized ZIRLO cladding[f], non-IFBA or IFBA, with IFMs	15 x 15 Upgrade fuel with ZIRLO or Optimized ZIRLO cladding[f], non-IFBA or IFBA, with IFMs
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core power	≤ 100% of 2597 MWt	≤ 2597 MWt
b. Peak heat flux hot channel factor (F_Q)	≤ 2.5	≤ 2.5
c. Peak hot rod enthalpy rise hot channel factor ($F_{\Delta H}$)	≤ 1.7	≤ 1.7
d. Hot assembly radial peaking factor ($\overline{P}_{HA}$)	≤ 1.7/1.04	≤ 1.7/1.04
e. Hot assembly heat flux hot channel factor (F_{QHA})	≤ 2.5/1.04	≤ 2.5/1.04
2.0 Plant Initial Operating Conditions (continued)		
2.1 Reactor Power (continued)		
f. Axial power distribution (P_{BOT}, P_{MID})	Figure 14.5-13	Figure 14.5-13

a. 28 peripheral locations will not physically be lead power assemblies.
b. Minimum Measured flow is verified per Technical Specifications 3.12.F, DNB parameters.
c. Please note that the fuel temperature and rod internal pressure data is only provided up to 62,000 MWD/MTU. In addition, the hot assembly/ hot rod will not have a burnup this high in ASTRUM analyses.
d. Plant control systems are designed to control these parameters to the stated values.
e. fL/D based on average L/D of 516.2.
f. The use of Optimized ZIRLO cladding has been deemed acceptable with no PCT penalty. See further discussion in Section 14.5.1.7.

Table 14.5-1 (CONTINUED)
PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Parameter	As-Analyzed Value or Range	Operating Range or Target Value
g. Low power region relative power (P_{LOW})	$0.2 \leq P_{LOW} \leq 0.6$	$0.2 \leq P_{LOW} \leq 0.6$
h. Hot assembly burnup	≤ 75,000 MWD/MTU, lead rod[a], [c]	≤ 62,000 MWD/MTU, lead rod[a], [c]
i. MTC	≤ 0 at hot full power (HFP)	≤ 0 at hot full power (HFP)
j. Typical cycle length	18 months	18 months
k. Minimum core average burnup	≥ 10,000 MWD/MTU	≥ 10,000 MWD/MTU
l. Maximum steady state depletion, F_Q	2.0	≤ 2.0
2.2 Fluid Conditions		
a. T_{AVG}	570–5.6°F ≤ T_{AVG} ≤ 576 + 5.6°F	570 to 576°F[d]
b. Pressurizer pressure	2250–60 psia ≤ P_{RCS} ≤ 2250 + 60 psia	2250 psia[d]
c. Loop flow	TDF ≥ 88,500 gpm/loop	MMF ≥ TS Limit[b]
d. Upper head design	T_{HOT}	T_{HOT}
e. Pressurizer level	53.7% of span (694 ft^3)	53.7% of span @ 573°F T_{AVG}[d]
f. Accumulator temperature	89°F ≤ T_{ACC} ≤ 110°F	89.5°F ≤ T_{ACC} ≤ 109.5°F
2.0 Plant Initial Operating Conditions (continued)		
2.2 Fluid Conditions (continued)		
g. Accumulator pressure	580 psia ≤ P_{ACC} ≤ 700 psia	600 psia ≤ P_{ACC} ≤ 680 psia
h. Accumulator liquid volume	970 ft^3 ≤ V_{ACC} ≤ 1030 ft^3	975 ft^3 ≤ V_{ACC} ≤ 1025 ft^3
i. Accumulator fL/D	7.227[e]	Current line configuration

a. 28 peripheral locations will not physically be lead power assemblies.
b. Minimum Measured flow is verified per Technical Specifications 3.12.F, DNB parameters.
c. Please note that the fuel temperature and rod internal pressure data is only provided up to 62,000 MWD/MTU. In addition, the hot assembly/ hot rod will not have a burnup this high in ASTRUM analyses.
d. Plant control systems are designed to control these parameters to the stated values.
e. fL/D based on average L/D of 516.2.
f. The use of Optimized ZIRLO cladding has been deemed acceptable with no PCT penalty. See further discussion in Section 14.5.1.7.

Table 14.5-1 (CONTINUED)
PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Parameter	As-Analyzed Value or Range	Operating Range or Target Value
j. Minimum accumulator boron	≥ 2000 ppm	≥ 2000 ppm
3.0 Accident Boundary Conditions		
a. Minimum safety injection flow	Table 14.5-4	≥ Table 14.5-4
b. Safety injection temperature	37.5°F ≤ SI Temp ≤ 62.5°F	40°F ≤ SI Temp ≤ 60°F
c. Safety injection delay	25 seconds (with offsite power) 40 seconds (with LOOP)	≤ 25 seconds (with offsite power) ≤ 40 seconds (with LOOP)
d. Containment modeling	See Figure 14.5-14 and raw data in Tables 14.5-2 and 14.5-5	See Figure 14.5-14 and raw data in Tables 14.5-2 and 14.5-5
e. Minimum containment air partial pressure	9.85 psia	≥ 9.85 psia
f. Containment spray initiation delay	59 seconds	≥ 59 seconds
g. Recirculation spray initiation delay	900 seconds	≥ 900 seconds
h. Single failure	Loss of one ECCS train	Loss of one ECCS train

a. 28 peripheral locations will not physically be lead power assemblies.
b. Minimum Measured flow is verified per Technical Specifications 3.12.F, DNB parameters.
c. Please note that the fuel temperature and rod internal pressure data is only provided up to 62,000 MWD/MTU. In addition, the hot assembly/ hot rod will not have a burnup this high in ASTRUM analyses.
d. Plant control systems are designed to control these parameters to the stated values.
e. fL/D based on average L/D of 516.2.
f. The use of Optimized ZIRLO cladding has been deemed acceptable with no PCT penalty. See further discussion in Section 14.5.1.7.

Table 14.5-2
CONTAINMENT BACK PRESSURE ANALYSIS INPUT PARAMETERS USED FOR BEST-ESTIMATE LBLOCA ANALYSIS

Containment Net Free Volume	1,819,000 ft^3
Initial Conditions	
Minimum air initial containment partial pressure at full power operation	9.85 psia
Minimum steam initial containment partial pressure at full power operation	0.00 psia
Minimum initial containment temperature at full power operation	100.0°F
RWST temperature	37.5°F
Temperature outside containment	9.0°F
Initial spray temperature	37.5°F
Spray System	
Number of containment spray pumps operating	2
Post-accident containment spray system initiation delay	59 sec
Maximum spray system flow from all containment spray pumps	4500 gal/min
Fan Coolers	
Maximum number of containment fan coolers in operation	0
Recirculation Spray*	
Recirculation spray heat exchanger UA	5.0 MBTU/hr-°F per heat exchanger
Recirculation spray heat exchanger service water flow	10,000 gal/min each
Number of recirculation spray pumps operating	4
Post-accident recirculation spray system initiation delay	900 sec
Maximum spray system flow from all recirculation spray pumps	13,700 gal/min

*Not modeled because recirculation spray begins after transient has ended.

Table 14.5-3
BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS RESULTS

	Results	Criterion
95/95 Peak Clad Temperature	1853°F	< 2200°F
95/95 Local Maximum Oxidation	2.45%	< 17%
95/95 Core Wide Oxidation	0.30%	< 1%

Table 14.5-4
TOTAL MINIMUM INJECTED SAFETY INJECTION FLOW USED IN BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

RCS Pressure (psia)	Flow Rate (gpm)
14.7	2269.0
52.7	2269.0
64.7	2098.6
69.7	1994.9
89.7	1648.0
114.7	1164.3
139.7	613.5
149.7	488.2
154.7	356.3

Table 14.5-5
LARGE BREAK LOCA CONTAINMENT DATA USED FOR CALCULATION OF CONTAINMENT PRESSURE

Structural Heat Sinks

	Wall (feet)	T_{Air} (°F)	Area (ft^2)	Height (ft)	$T_{initial}$ (°F)
1.	0.00025 paint, 0.5 concrete	100	8393	10	100
2.	0.00025 paint, 1.0 concrete	100	62,271	10	100
3.	0.00025 paint, 1.5 concrete	100	55,365	10	100
4.	2.0 concrete	100	11,591	10	100
5.	2.25 concrete	100	9404	10	100
6.	3.0 concrete	100	3636	10	100
7.	0.00025 paint, 0.03125 carbon steel, 4.5 concrete	9	1354	10	100
8.	0.00025 paint, 0.03125 carbon steel, 4.5 concrete	9	45,135	10	100
9.	0.00025 paint, 0.04167 carbon steel, 2.5 concrete	9	747	10	100
10.	0.00025 paint, 0.04167 carbon steel, 2.5 concrete	9	24,905	10	100
11.	0.00025 paint, 2.2 concrete	45	12,110	10	100
12.	0.0005 paint, 0.01967 carbon steel	100	9200 11,700[1]	10	100
13.	0.0005 paint, 0.03608 carbon steel	100	69,362	10	100
14.	0.0005 paint, 0.07458 carbon steel	100	14,126 14,626[1]	10	100
15.	0.0005 paint, 0.14167 carbon steel	100	2668 3,168[1]	10	100
16.	0.0005 paint, 0.24167 carbon steel	100	11,880	10	100
17.	0.03608 carbon steel	100	11,811	10	100
18.	0.005 carbon steel	100	104,525	10	100
19.	0.0097 stainless steel	100	40,000 45,000[1]	10	100

Table 14.5-5 (CONTINUED)
LARGE BREAK LOCA CONTAINMENT DATA USED FOR CALCULATION
OF CONTAINMENT PRESSURE

Structural Heat Sinks

	Wall (feet)	T_{Air} (°F)	Area (ft^2)	Height (ft)	$T_{initial}$ (°F)
20.	0.03567 stainless steel	100	14,500 15,500[1]	10	100
21.	0.12783 stainless steel	100	600 1,600[1]	10	100
22.	0.012 carbon steel	100	2000 2,300[1]	10	100

1. Adjustments were made to the following heat sinks to accommodate future structural changes. These post-analysis of record surface area changes result in a 0°F PCT impact.

Table 14.5-6
PEAK CLAD TEMPERATURE INCLUDING ALL PENALTIES AND BENEFITS, BEST-ESTIMATE LARGE BREAK LOCA (BE LBLOCA)

PCT for Analysis-of-Record (AOR)	1853°F
PCT Assessment Allocated to AOR	—
Assessment of Optimized ZIRLO	0°F
Containment Structural Heat Sink Assessment	0°F
Thermal Conductivity Degradation Assessment (including peaking factor burndown)	183°F
Fuel Pellet Radial Power Profile	-13°F
Revised Heat Transfer Multiplier Distributions	-7°F
Hotspot Burst Strain Error	51°F
Decay Group Uncertainty Error	4°F
BE LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	2071°F

Table 14.5-7
SURRY CONTAINMENT AND ECCS X/Q VALUES

Leakage Source	Receptor	χ/Q (sec/m3)				
		0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
Containment	Control Room	6.74E-04	5.18E-04	2.22E-04	1.66E-04	1.20E-04
	EAB	1.79E-03	1.79E-03	1.79E-03	1.79E-03	1.79E-03
	LPZ	2.01E-04	2.01E-04	1.22E-04	4.18E-05	8.94E-06
ECCS & RWST (Ventilation Vent No. 2)	Control Room	6.95E-04	5.40E-04	2.30E-04	1.71E-04	1.22E-04
	EAB	1.79E-03	1.79E-03	1.79E-03	1.79E-03	1.79E-03
	LPZ	2.01E-04	2.01E-04	1.22E-04	4.18E-05	8.94E-06

Table 14.5-8
COMBINED CONTAINMENT AND RECIRCULATION SPRAY AEROSOL (PARTICULATE) REMOVAL COEFFICIENTS

Time (hr)		λ_{mf} (hr^{-1})
From	To	
2.78E-02	1.94E-01	3.59E+00
1.94E-01	5.56E-01	3.69E+00
5.56E-01	1.00E+00	4.16E+00
1.00E+00	1.14E+00	4.40E+00
1.14E+00	1.80E+00	2.53E+01
1.80E+00	1.84E+00	1.61E+01
1.84E+00	1.88E+00	1.12E+01
1.88E+00	2.07E+00	5.99E+00
2.07E+00	2.77E+00	3.25E+00
2.77E+00	3.57E+00	2.90E+00
3.57E+00	4.37E+00	2.86E+00
4.37E+00	7.20E+02	2.85E+00

Table 14.5-9
SURRY CONTROL ROOM OCCUPANCY FACTORS[a]

Time	Occupancy Factor
0 - 8 hr	1.0
8 - 24 hr	1.0
24 - 96 hr	0.6
96 - 720 hr	0.4

a. These values also used for LRA, SGTR, and MSLB accident dose analyses.

Table 14.5-10
CORE FISSION PRODUCT RELEASE FRACTIONS AND PRODUCTION RATES FOR THE LOCA AS SPECIFIED BY NUREG-1465 (REFERENCE 36)

	Core Release Fraction		Production Rate (Fraction/hr)[a]	
Group	Gap	Early In-Vessel	Gap	Early In-Vessel
Noble Gases[b]	0.05	0.95	0.1	7.31E-01
Halogens	0.05	0.35	0.1	2.69E-01
Alkali Metals	0.05	0.25	0.1	1.92E-01
Tellurium	0	0.05	0	3.85E-02
Barium, Strontium	0	0.02	0	1.54E-02
Noble Metals	0	0.0025	0	1.92E-03
Cerium	0	0.0005	0	3.85E-04
Lanthanides	0	0.0002	0	1.54E-04
Duration (hr)[a]	0.5	1.3		

a. Release duration and production rates apply only to the Containment release. The ECCS leakage portion of the analysis conservatively assumes that the entire core release fraction is in the containment sump from the start of the LOCA.

b. Noble Gases are not scrubbed from the containment atmosphere and therefore are not found in either the sump or ECCS fluid.

Table 14.5-11
LOCA CONTROL ROOM AND OFFSITE TEDE DOSE CONSEQUENCES COMPARED TO THE TEDE DOSE LIMITS OF 10 CFR 50.67

	Control Room (Rem TEDE)	Exclusion Area Boundary (Rem TEDE)	Low Population Zone (Rem TEDE)
Total Dose Consequences including contributions from containment, ECCS and RWST leakage	3.9	15.6	3.5
10 CFR 50.67 dose limits	5	25	25

a. 10 CFR Part 50.67 establishes TEDE dose limits for the EAB, the outer boundary of the LPZ, and for the control room for use with the alternate source term.

Table 14.5-12
SIGNIFICANT INPUTS AND ASSUMPTIONS IN THE SMALL BREAK LOCA ANALYSIS

Parameter	Value	
Core Power	2597 MWt	(102% of 2546 MWt)
		(100.38% of 2587 MWt)
Total Peaking Factor, F_Q	2.50	
Core Enthalpy Rise Factor, $F\Delta h$	1.70	
Fuel Enrichment	3.2%	
Fuel Pellets	Chamfered	
Fuel Assembly Array	15 x 15 SIF[a]	
Accumulator Water Volume	1000 ft^3/accumulator	
Accumulator Tank Volume	1450 ft^3/accumulator	
Accumulator Gas Pressure	580 psia	
Safety Injection Flow	Figures 14.5-15 and 14.5-16	
Initial Loop Flow[b]	9317 lbm/sec	
Vessel Average Temperature[b]	577.6°F	
Reactor Coolant Pressure[b]	2310 psia	
Steam Pressure[b]	795.5 psia	
Steam Generator Tube Plugging (uniform)	7%	
Low Pressurizer Pressure Setpoint	1840 psia	
Low-Low Pressurizer Pressure Setpoint	1715 psia	

a. This analysis was performed assuming the SIF fuel product with ZIRLO™ cladding and Performance+ design features. An evaluation was performed assuming the 15 x 15 Upgrade fuel product with Optimized ZIRLO cladding.
b. Initial conditions assumed in the analysis.

Table 14.5-13
SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Break Size, in	**1.5**	**2**	**2.25**	**2.5**	**2.75**	**3**	**4**	**5.5**
Transient Initiated, sec	0	0	0	0	0	0	0	0
Reactor Trip Signal, sec	65.8	34.3	26.6	20.9	17.1	14.4	8.4	5.4
Safety Injection Signal, sec	85.7	45.6	35.9	29.0	24.2	20.7	11.4	7.5
Safety Injection Begins, sec[a]	110.7	70.6	60.9	54.0	49.2	45.7	36.4	32.5
Loop Seal Clearing Occurs, sec	1855	1050	803	658	548	466	236	138
Top of Core Uncovered, sec	5144	1410	1068	856	674	711	443	205
Accumulator Injection Begins, sec	NA[b]	NA[b]	2496	1831	1414	1215	623	296
RWST Low Level Time, sec	3086.2	3073.6	3064.2	3057.1	3051.6	3048.0	3037.0	2458.6
Top of Core Recovered, sec	NA[c]	NA[c]	NA[c]	NA[c]	NA[c]	NA[c]	2568	422

a. Safety Injection is assumed to begin 25.0 seconds after the Safety Injection Signal.
b. RCS pressure does not reach the accumulator injection pressure setpoint before transient end.
c. Long-term RCS inventory recovery was established in the transient calculation prior to reaching this condition.

Table 14.5-14
SMALL BREAK LOCA RESULTS - FUEL CLADDING DATA[a]

Break Size, in	**1.5**	**2**	**2.25**	**2.5**	**2.75**	**3**	**4**	**5.5**
PCT, °F	1132	1756	1802	1870	1907	1766	1442	1438
PCT Time, sec	7796.2	3303.1	2948.3	2174.3	1687.6	1451.6	700.5	363.2
PCT Elevation, ft	11.50	12.00	12.00	12.00	12.00	11.75	10.75	10.50
Maximum Hot Rod (HR) Transient ZrO_2, %	0.12	3.75	5.06	4.87	4.64	2.47	0.25	0.14
Maximum HR ZrO_2 Elevation, ft	11.50	12.00	12.00	12.00	11.75	11.75	11.00	10.50
HR Average ZrO_2, %	0.02	0.50	0.65	0.65	0.66	0.38	0.05	0.03

a. Cladding burst was not calculated to occur for any of these break sizes.

Table 14.5-15
LIMITING 2.75 INCH SBLOCTA BURNUP STUDY RESULTS

Burnup, MWD/MTU	5,000[a]	7,000
PCT, °F	2012	1943
PCT Time, sec	1649.2	1685.2
PCT Elevation, ft	12.00	11.75
Maximum HR Transient ZrO_2, %	11.65	11.86
Maximum HR ZrO_2 Elevation, ft	12.00	11.75
HR Average Transient ZrO_2, %	0.67	0.73

a. Case modeled annular pellets.

Table 14.5-16
PEAK CLAD TEMPERATURE INCLUDING ALL PENALTIES AND BENEFITS, SMALL BREAK LOCA

PCT for Analysis of Record	2012°F
PCT Assessments Allocated to AOR	NONE
15 x 15 Upgrade Fuel	+0°F
SBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	2012°F

Table 14.5-17
SURRY UNITS 1 AND 2 BEST-ESTIMATE LARGE-BREAK LOCA SEQUENCE OF EVENTS FOR LIMITING PCT TRANSIENT (RUN 078)

Event	Time After Break (sec)
Start of transient	0.0
Safety Injection Signal	4.9
Accumulator Injection Begins	10.0
End of Blowdown	24.0
Safety Injection Begins	29.9
Bottom of Core Recovery	32.5
Accumulator Empty[a]	36.2
PCT Occurs	
WCOBRA/TRAC PCT (Global)	~41.8
HOTSPOT PCT (Global + Local)	~ 80.0
Core Quenched	450
End of analysis time	600

Note:
a. Accumulator liquid injection ends.

Table 14.5-18
SURRY UNITS 1 AND 2 INCREASED BREAK OPENING TIMES USED FOR EVALUATION OF RPV SLIDING FOOT SUPPORTS

RCS Branch Line Break Case	Break Opening Time (milliseconds)
RHR Line Break	26.6
SI Line Break	23.4
PZR Surge Line Break	20.6

Figure 14.5-1
LIMITING PCT CASE PCT AND PCT LOCATION

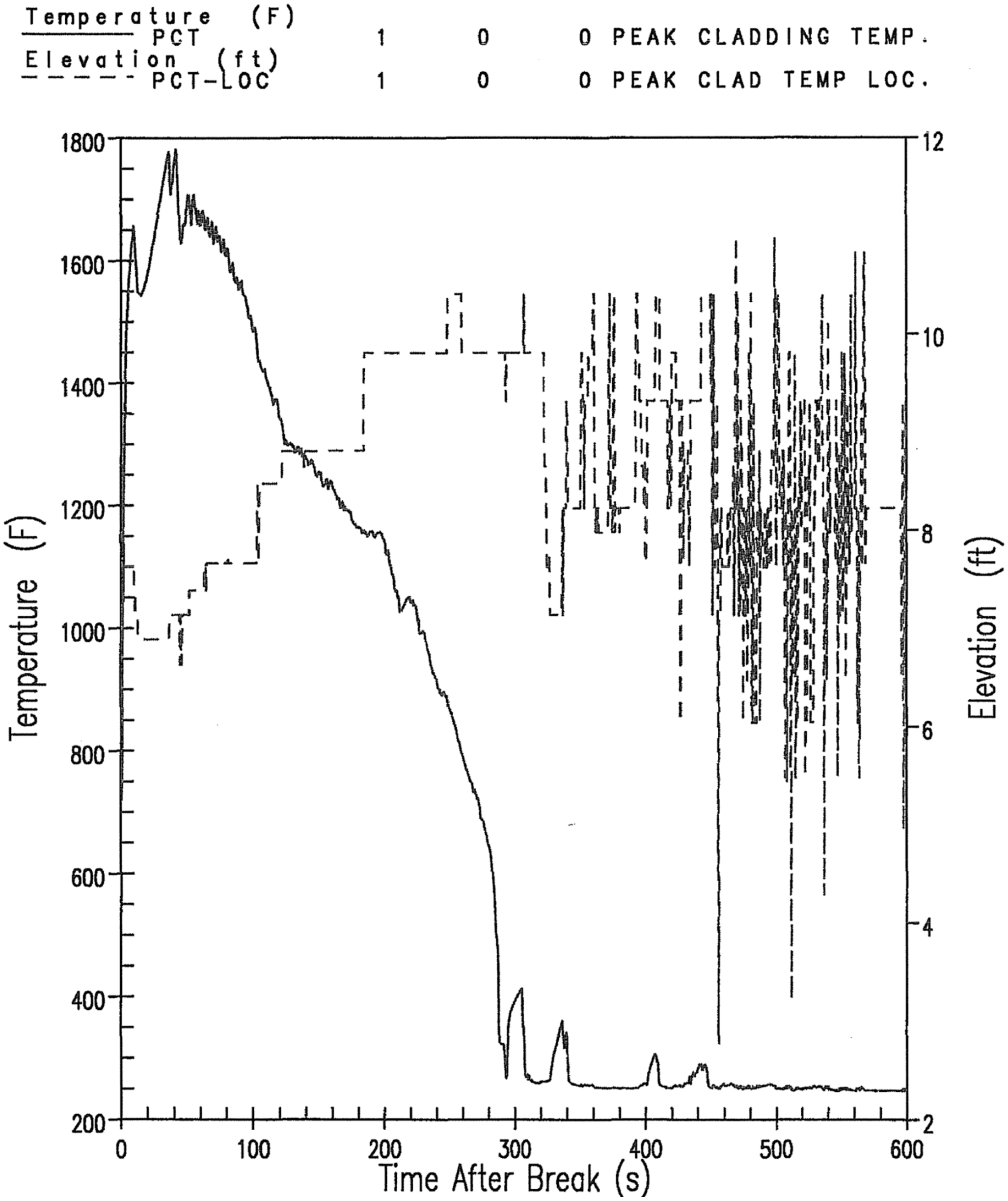


Figure 14.5-2
LIMITING PCT CASE VESSEL SIDE BREAK FLOW

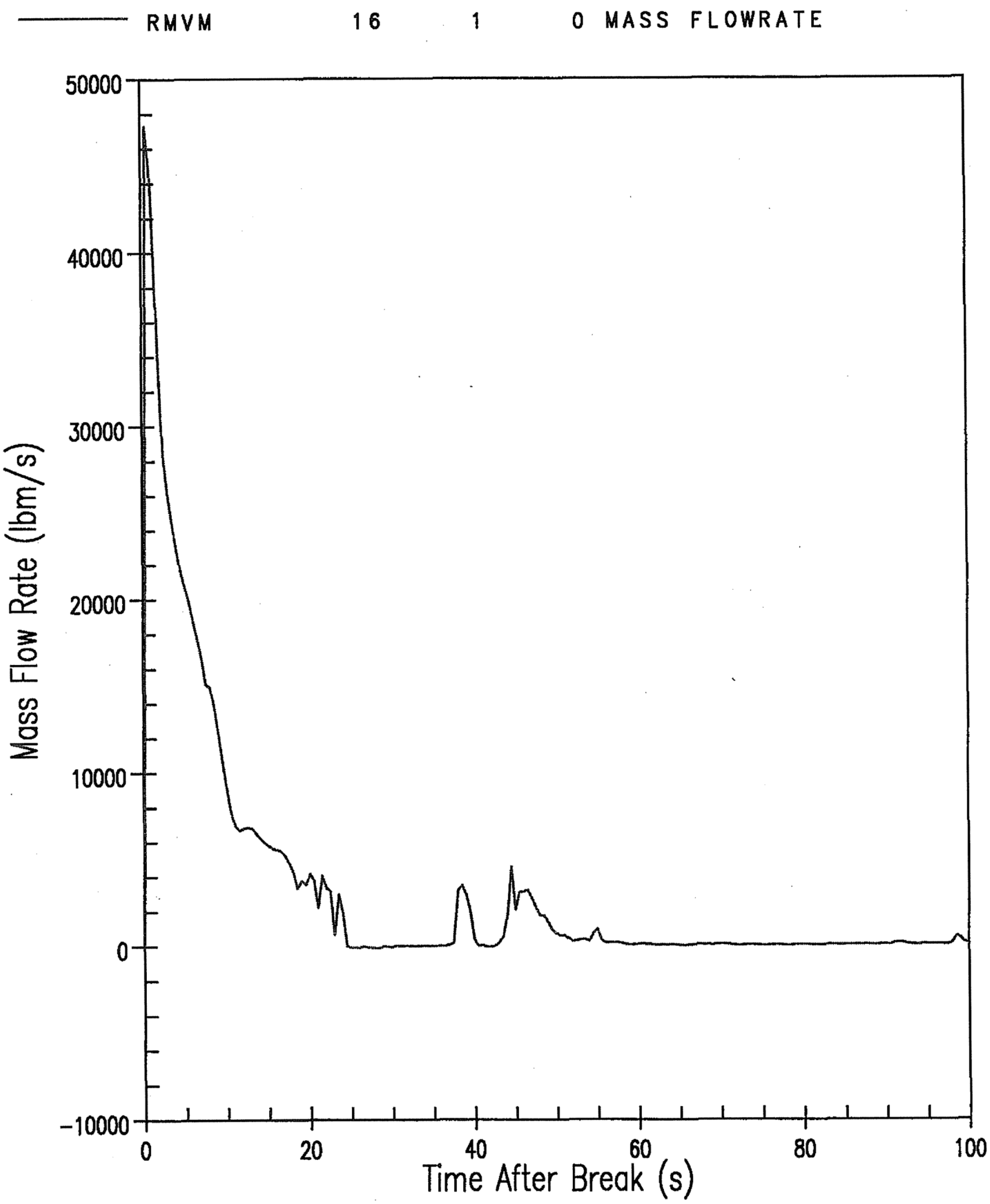


Figure 14.5-3
LIMITING PCT CASE LOOP SIDE BREAK FLOW

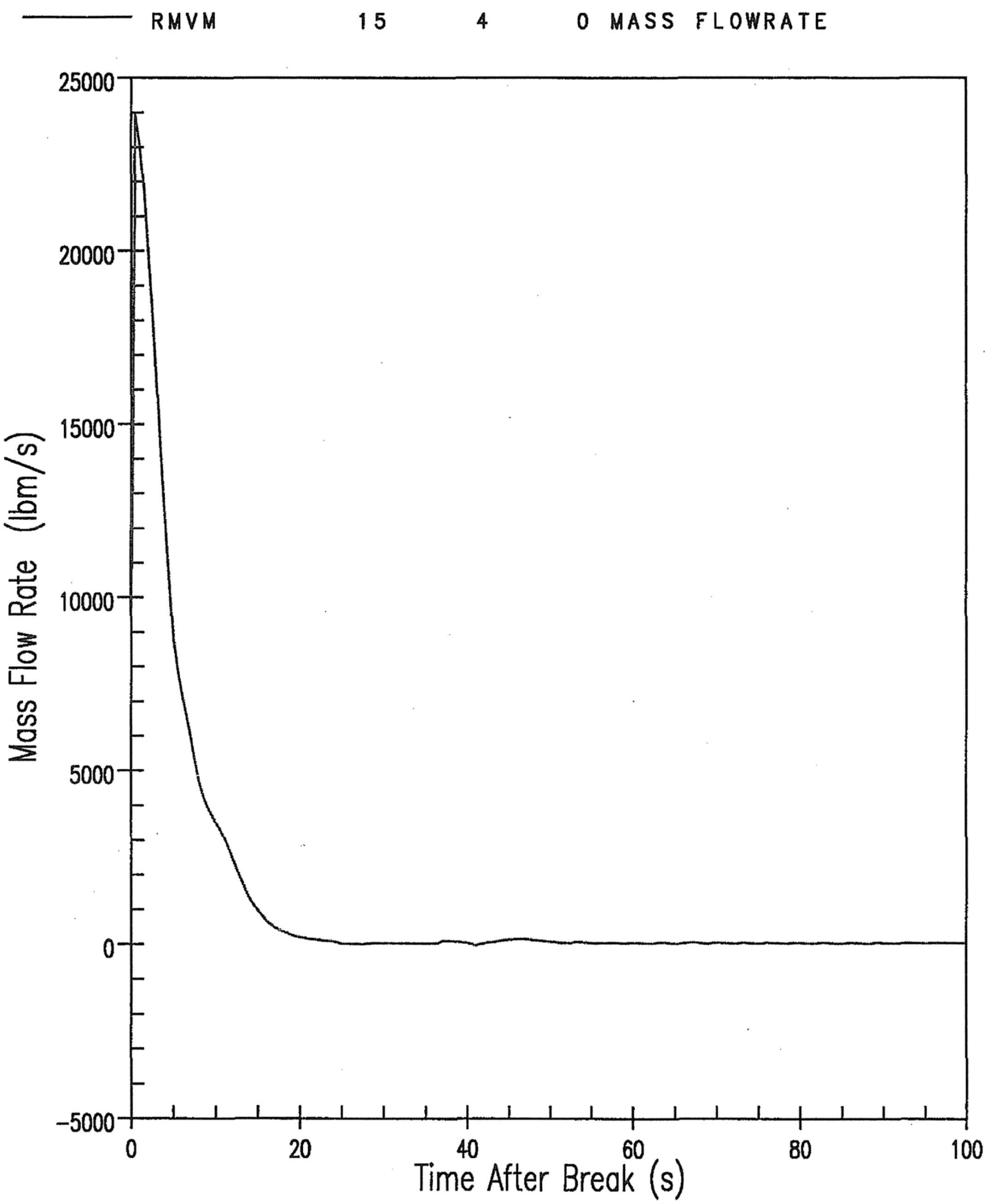


Figure 14.5-4
LIMITING PCT CASE BROKEN AND INTACT LOOP PUMP VOID FRACTION

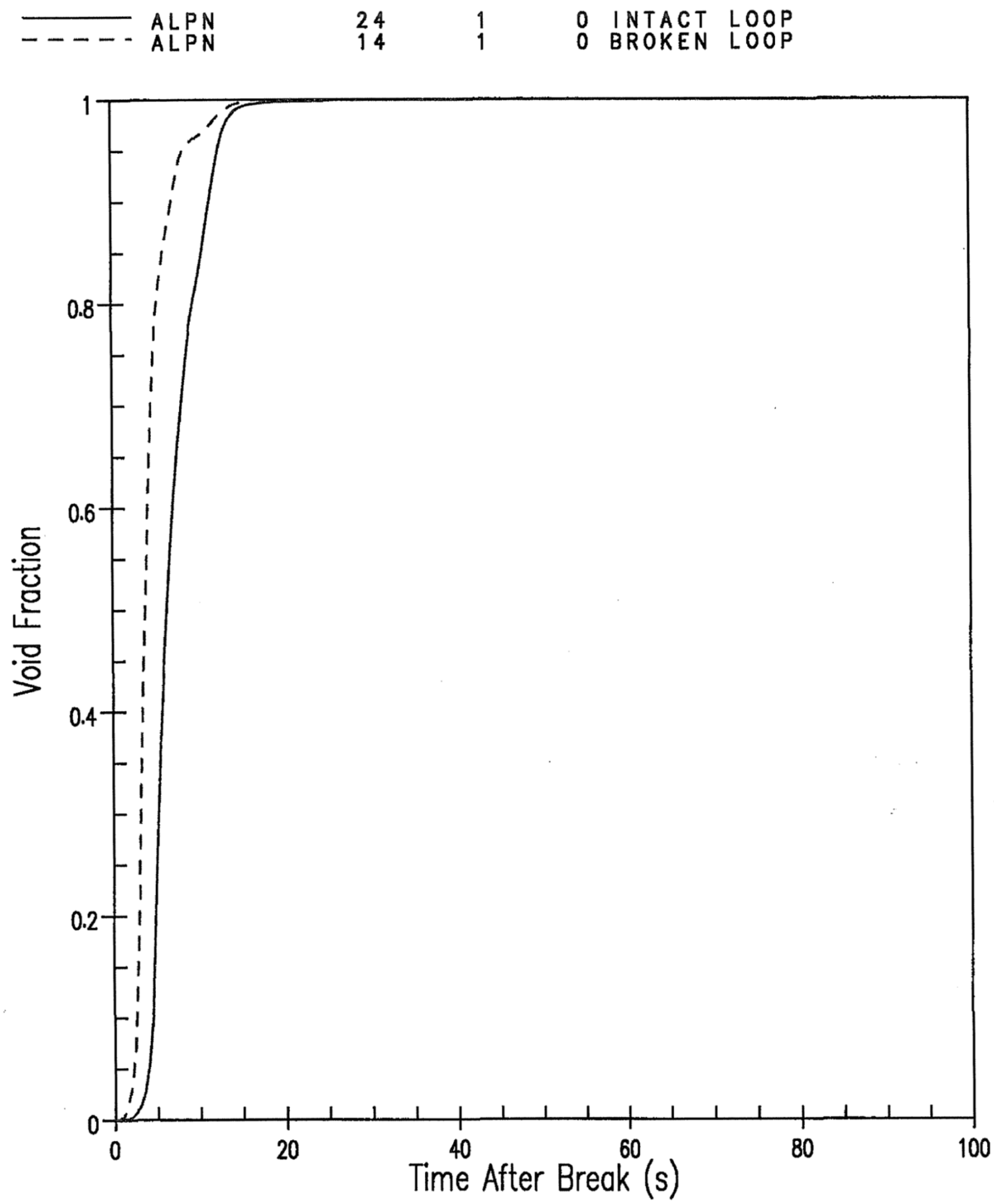


Figure 14.5-5
LIMITING PCT CASE HOT ASSEMBLY TOP OF CORE VAPOR FLOW

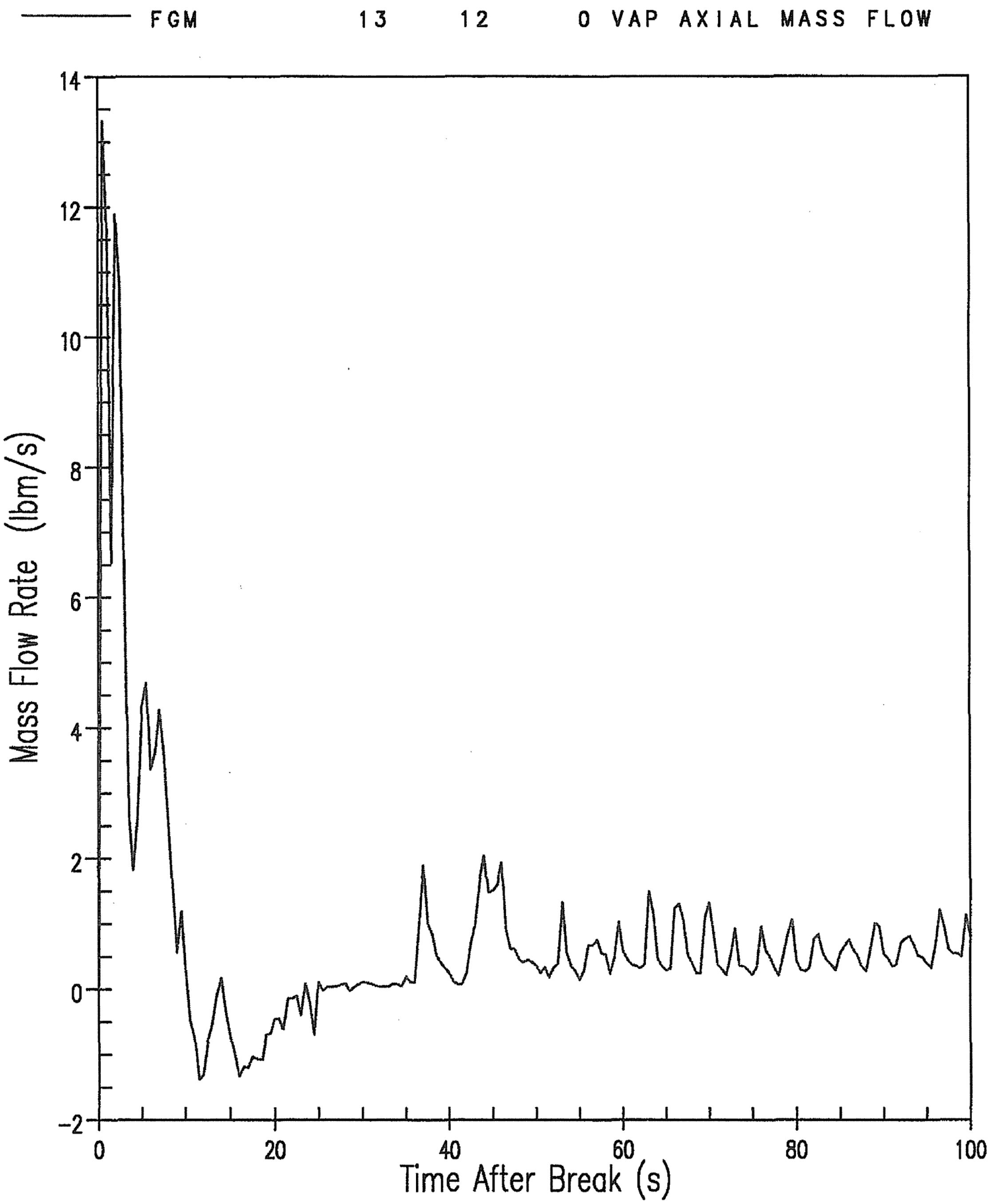


Figure 14.5-6
LIMITING PCT CASE PRESSURIZER PRESSURE

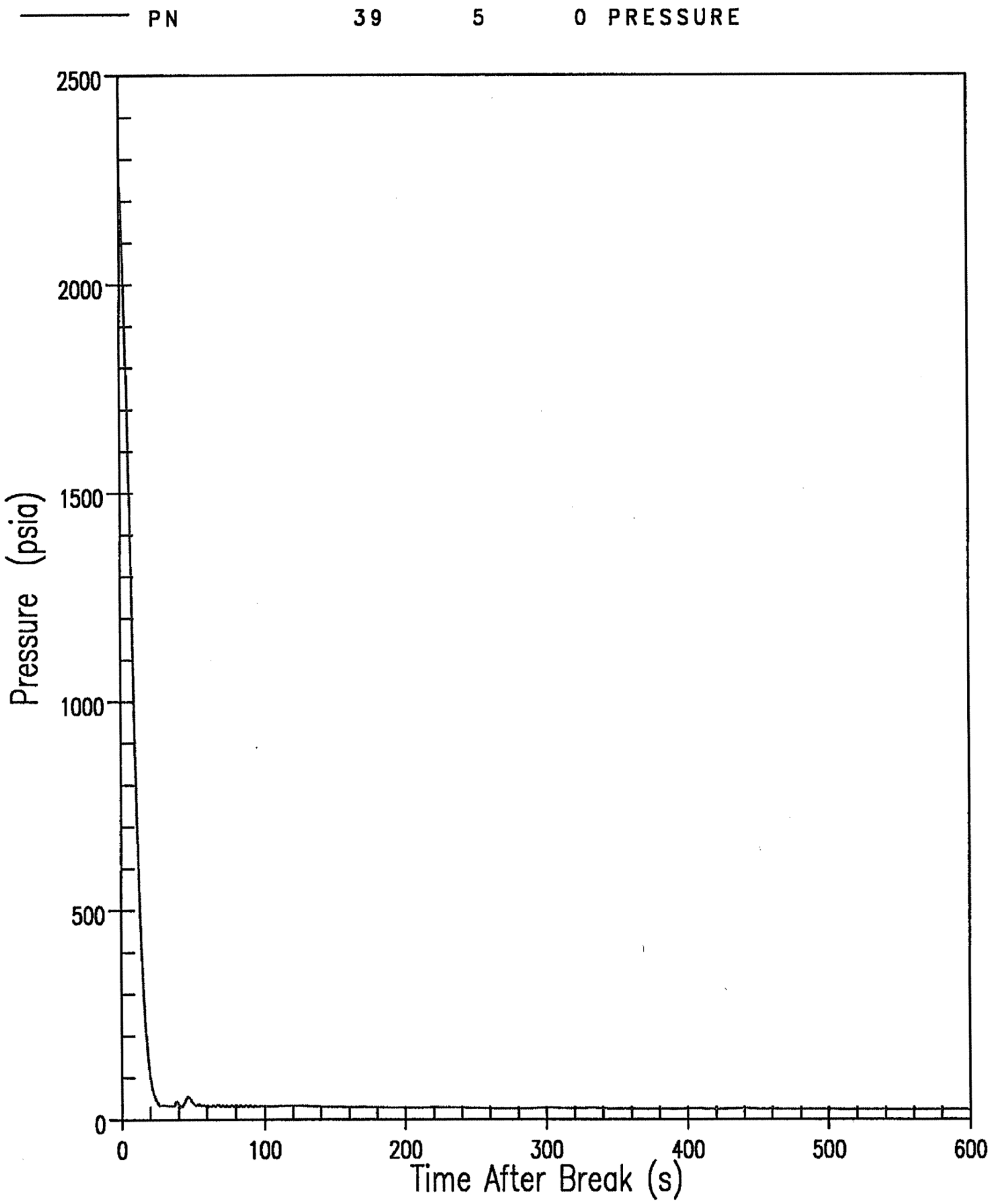


Figure 14.5-7
LIMITING PCT CASE LOWER PLENUM COLLAPSED LIQUID LEVEL

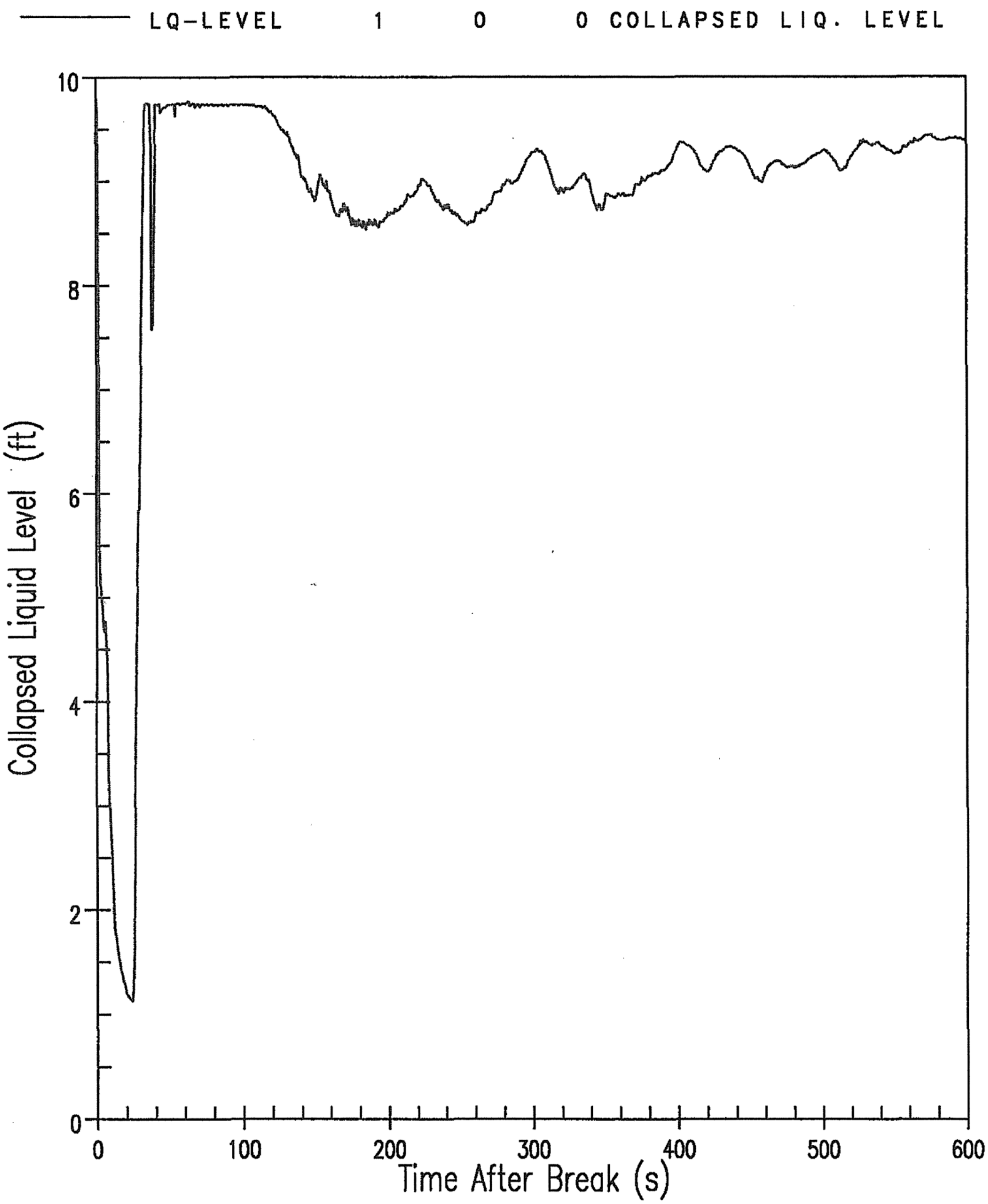


Figure 14.5-8
LIMITING PCT CASE VESSEL FLUID MASS

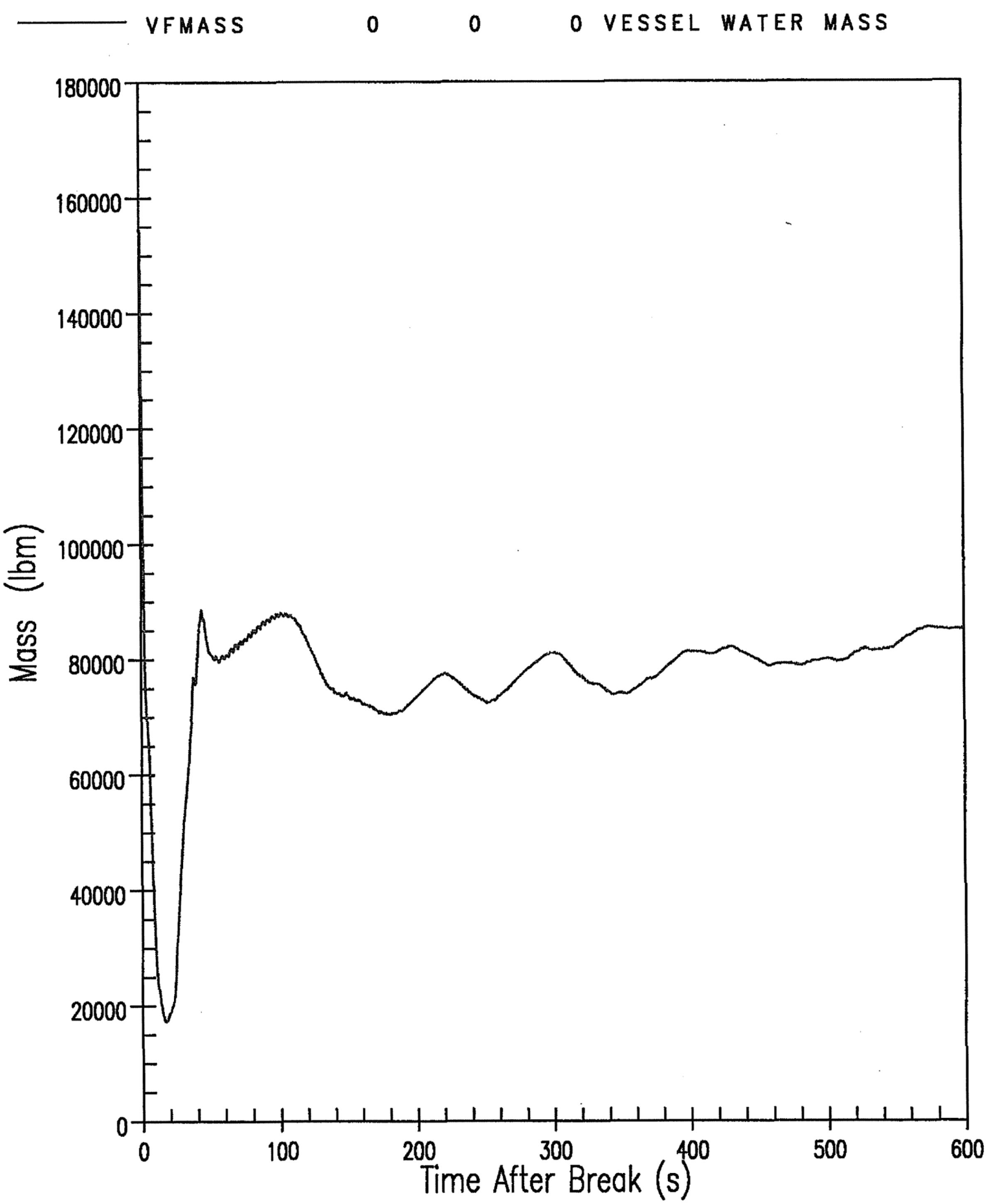


Figure 14.5-9
LIMITING PCT CASE LOOP 2 ACCUMULATOR FLOW

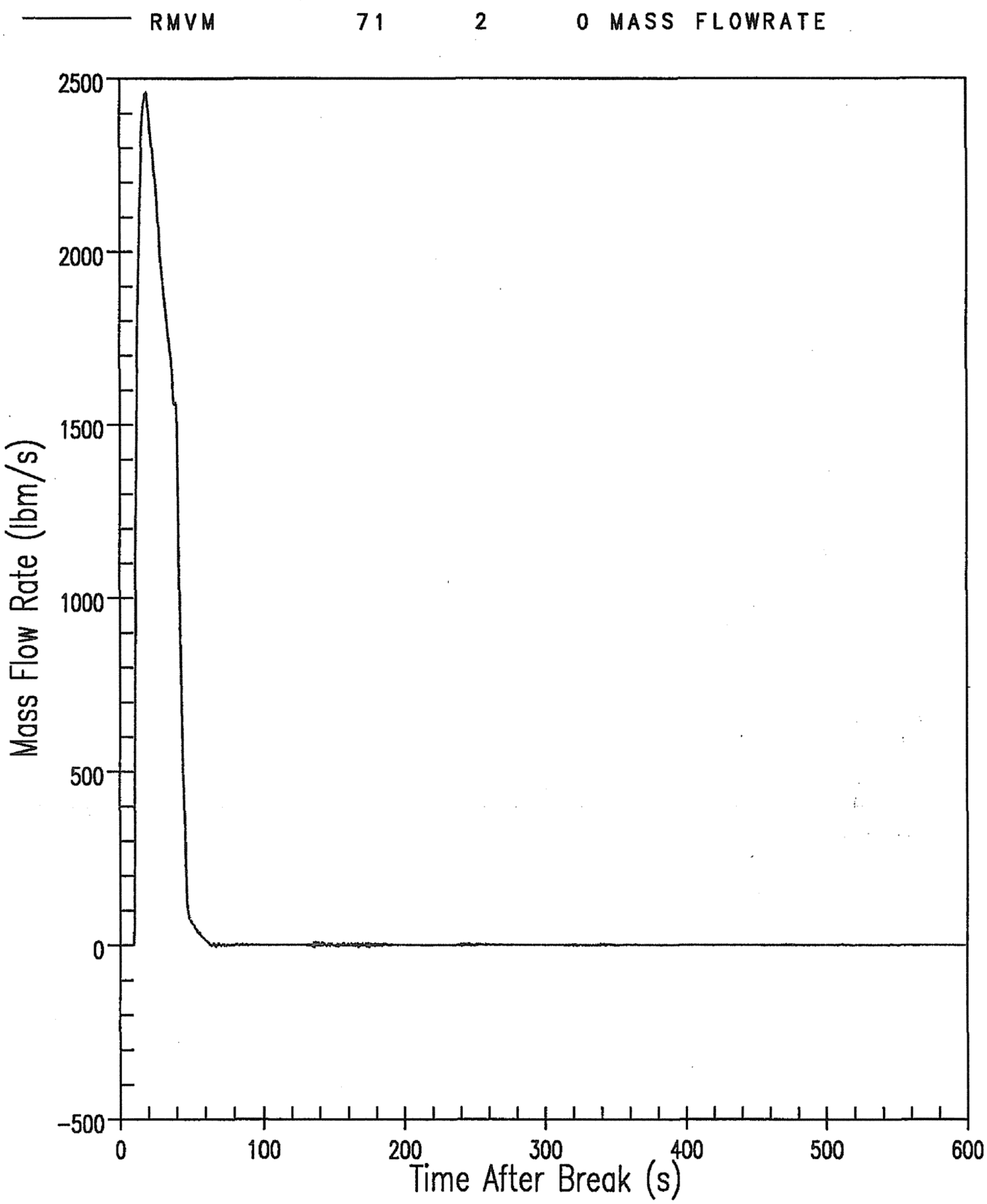


Figure 14.5-10
LIMITING PCT CASE LOOP 2 SAFETY INJECTION FLOW

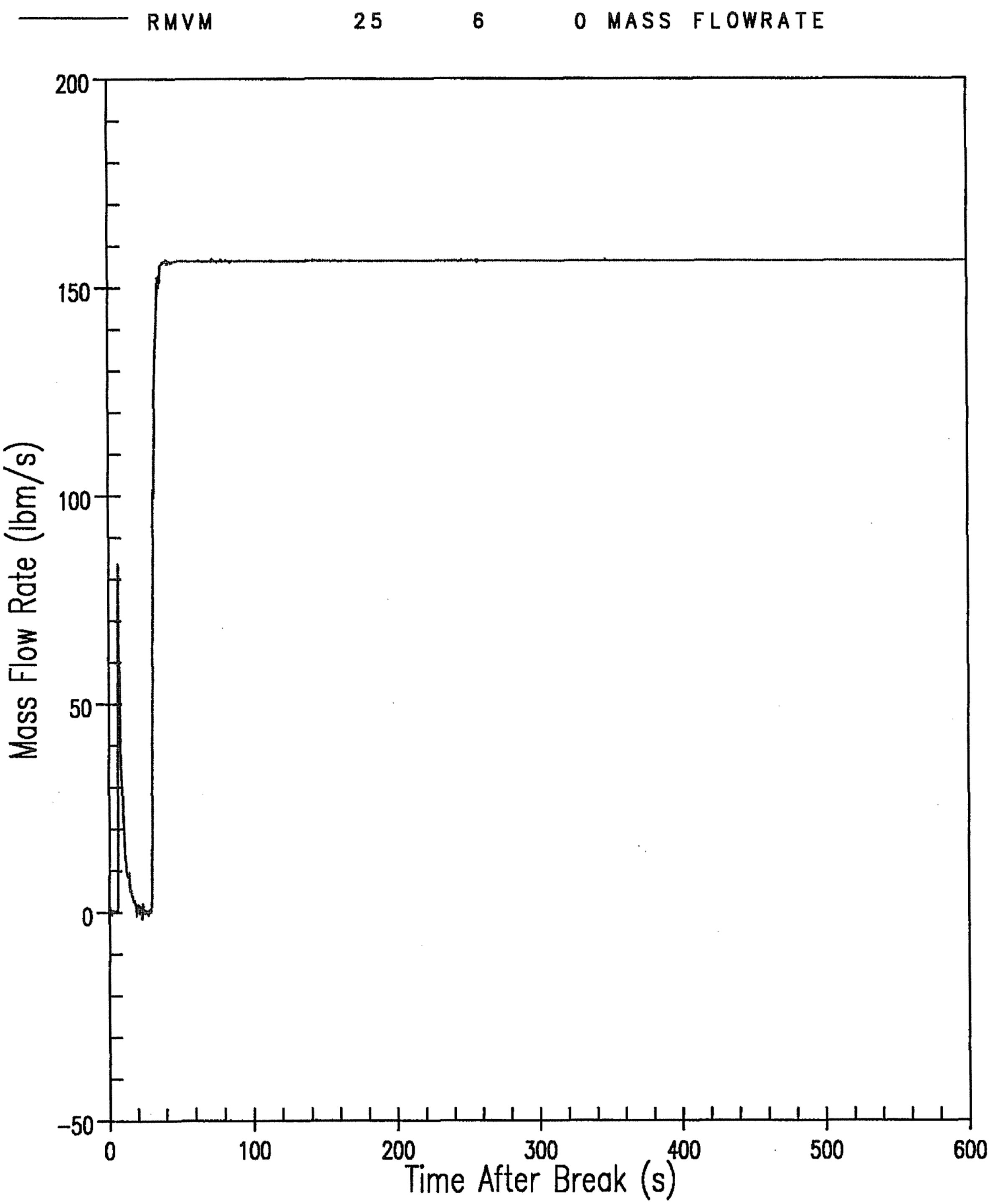


Figure 14.5-11
LIMITING PCT CASE CORE AVERAGE CHANNEL COLLAPSED LIQUID LEVEL

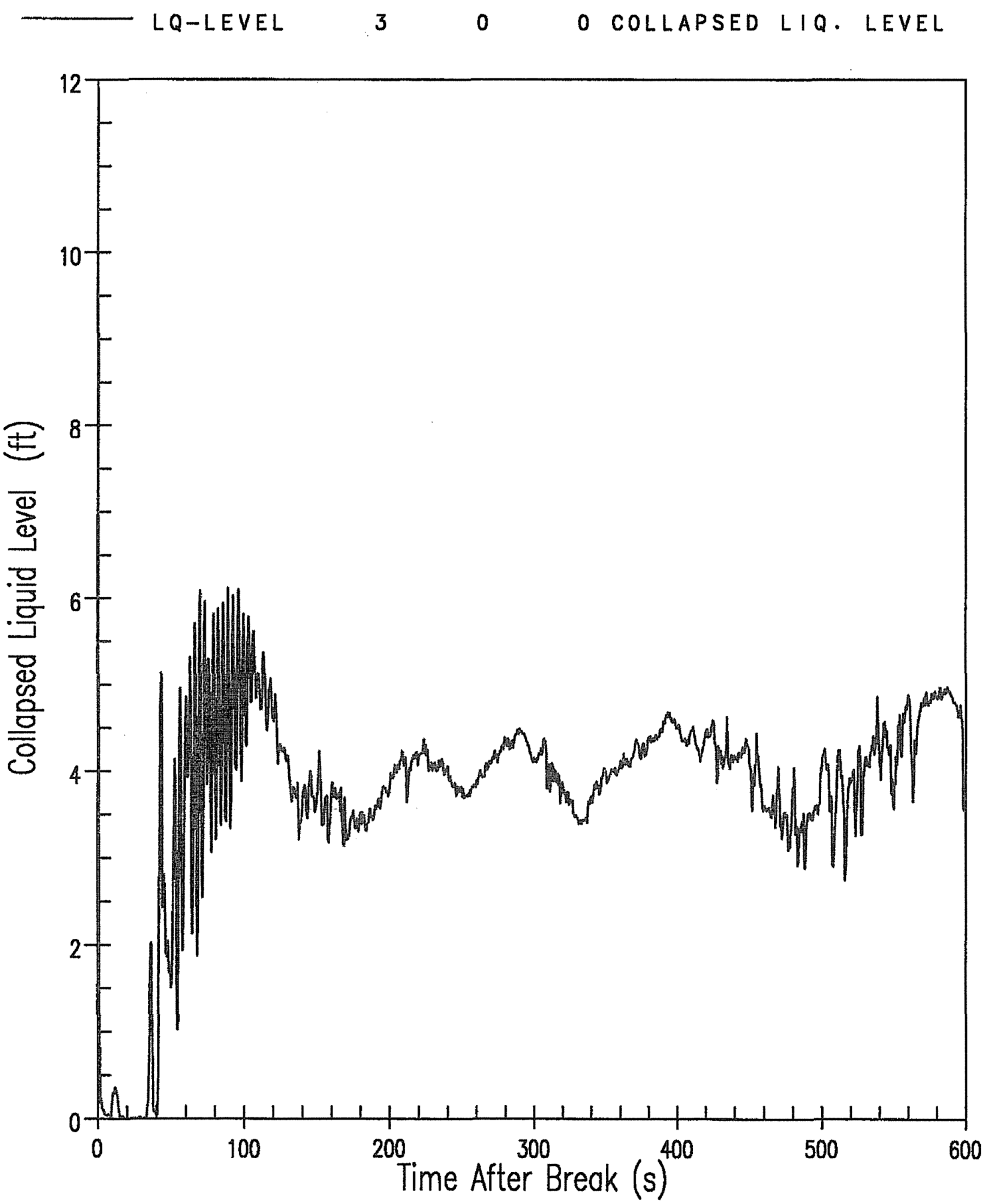


Figure 14.5-12
LIMITING PCT CASE LOOP 2 DOWNCOMER COLLAPSED LIQUID LEVEL

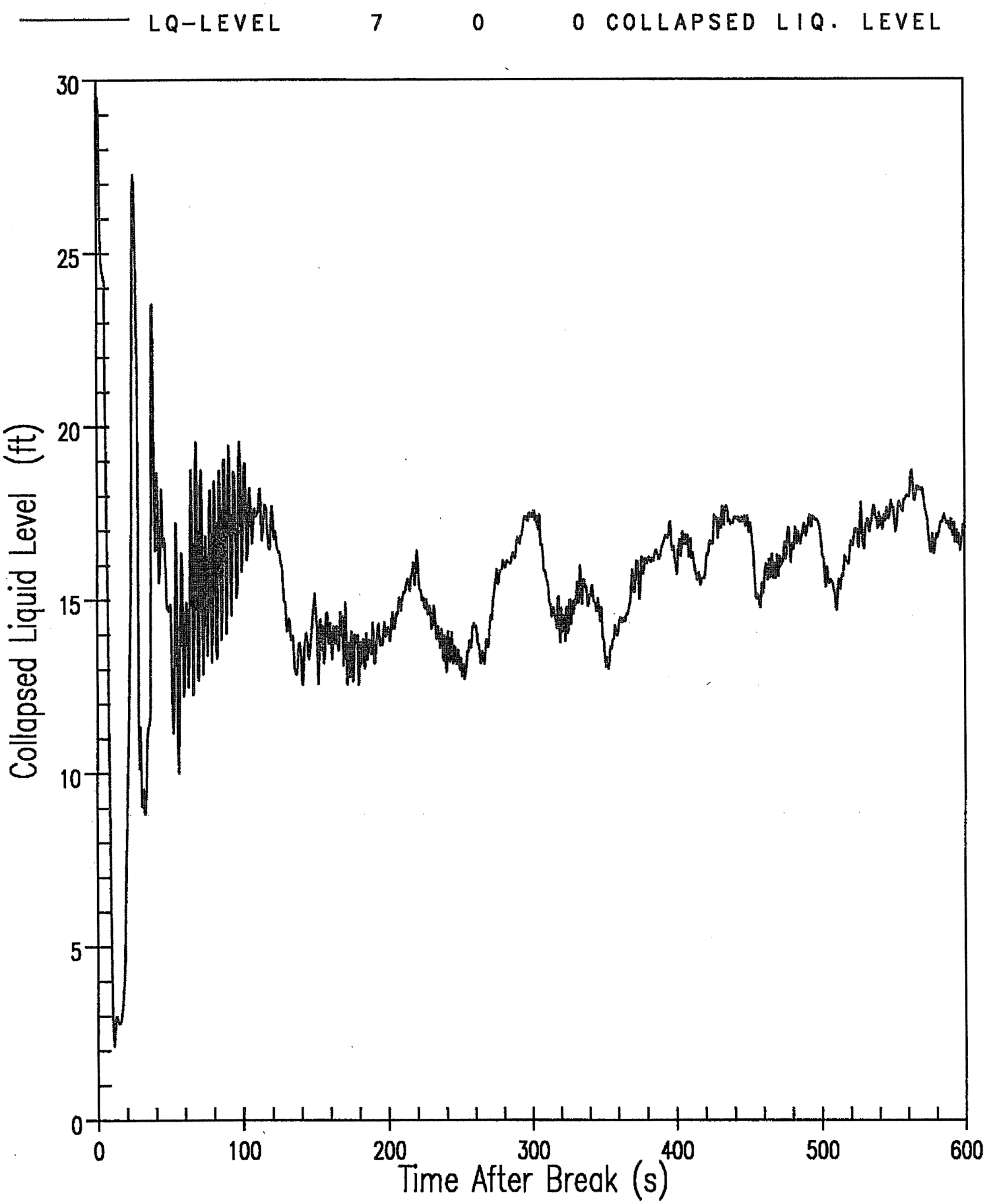


Figure 14.5-13
BELOCA ANALYSIS AXIAL POWER SHAPE OPERATING SPACE ENVELOPE

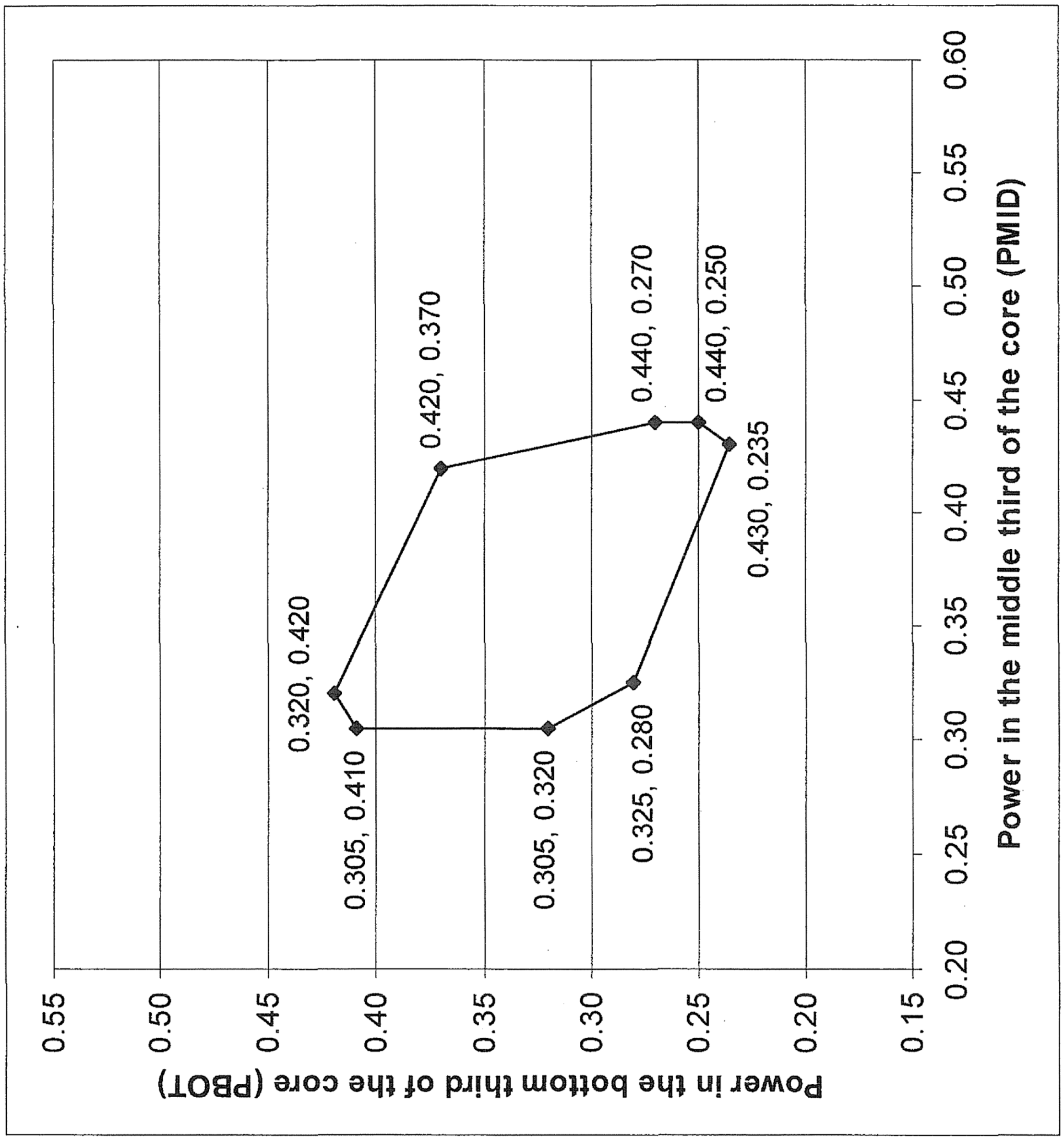


Figure 14.5-14
LOWER BOUND CONTAINMENT PRESSURE

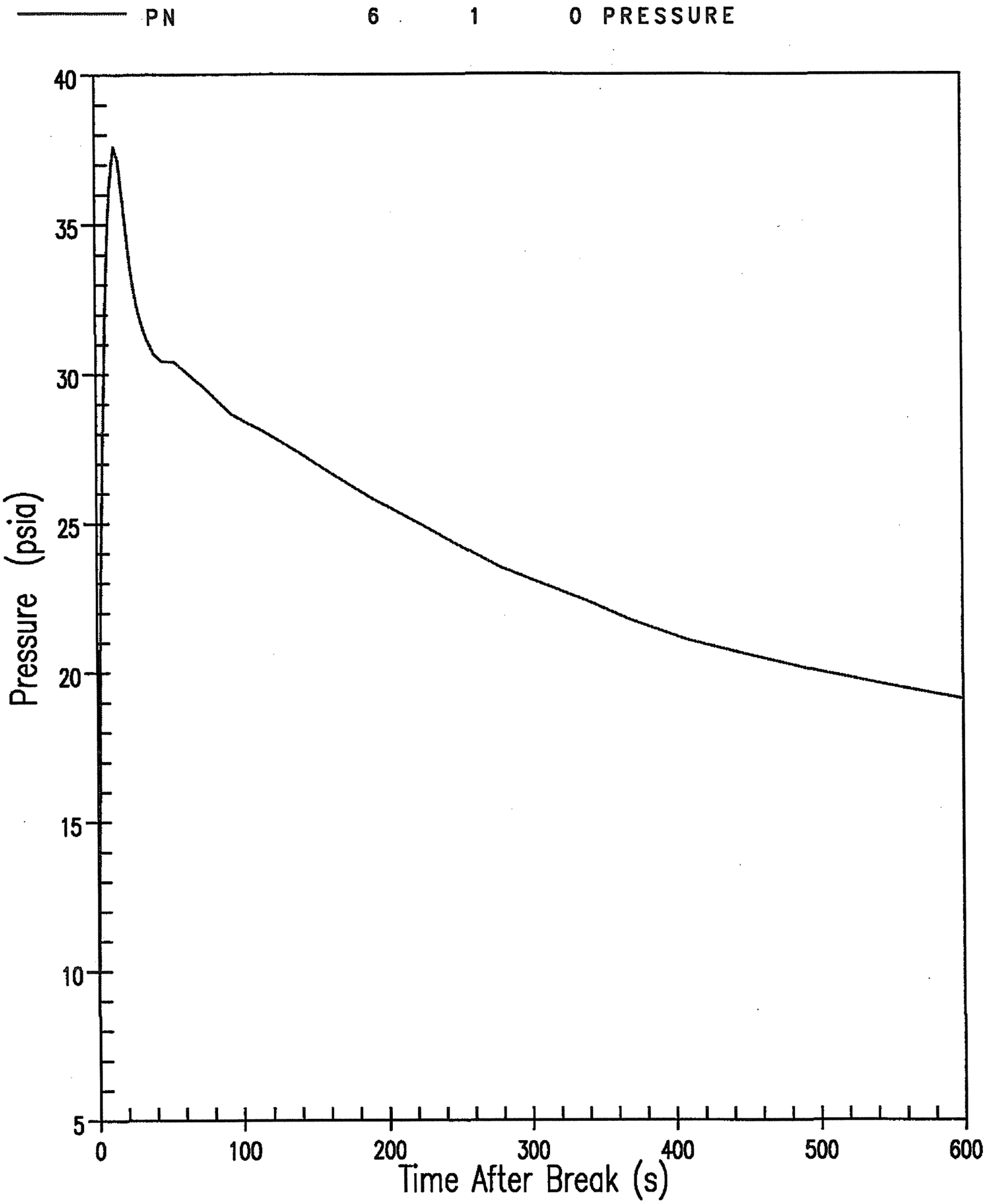


Figure 14.5-15
HHSI FLOWS WITH THE FAULTED LOOP SPILLING TO RCS PRESSURE
(PRESSURE VS. FLOW RATE)

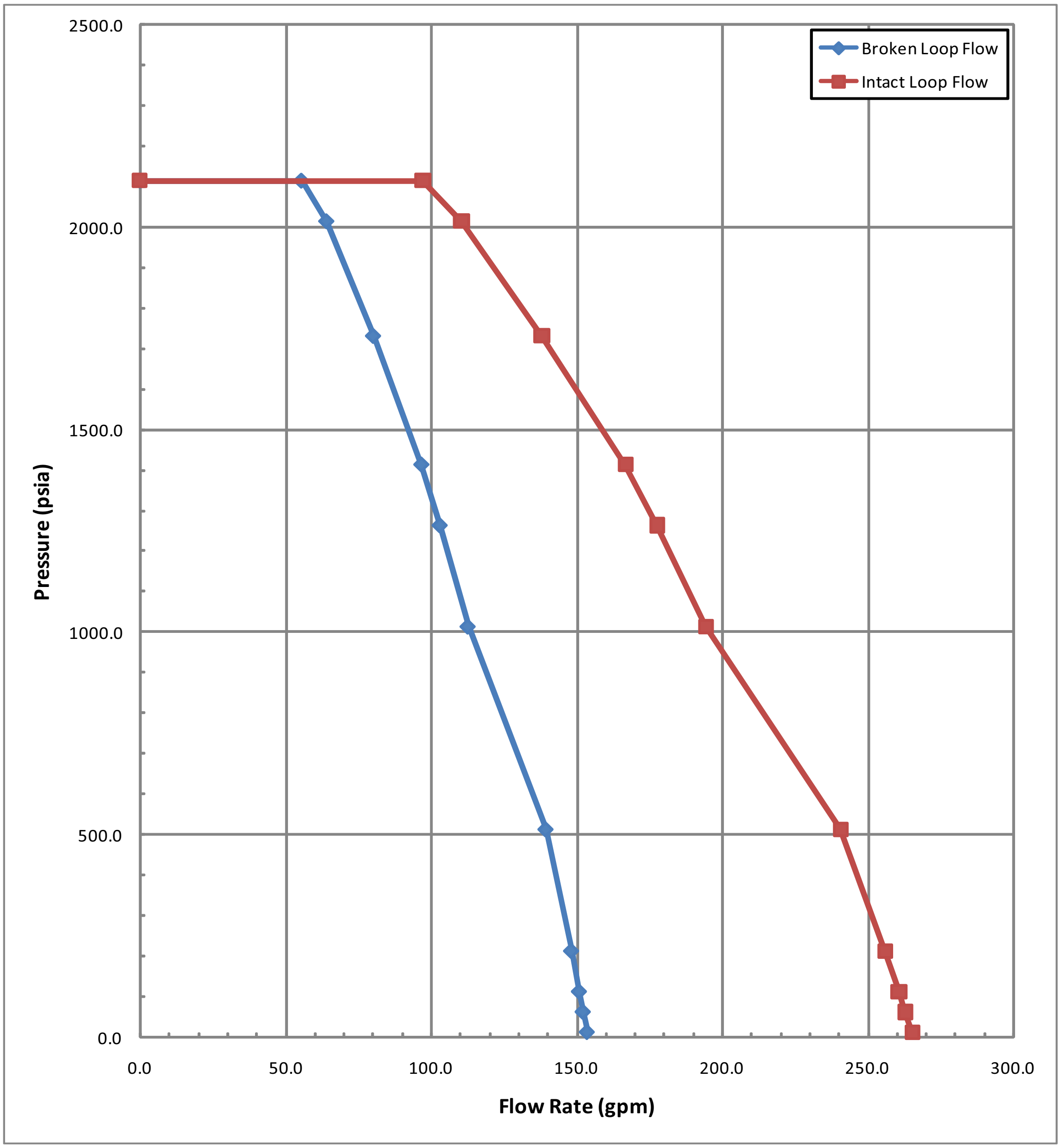


Figure 14.5-16
HHSI AND LHSI COMBINED FLOWS WITH THE FAULTED LOOPS SPILLING TO CONTAINMENT PRESSURE (0 PSIG) (PRESSURE VS. FLOW RATE)

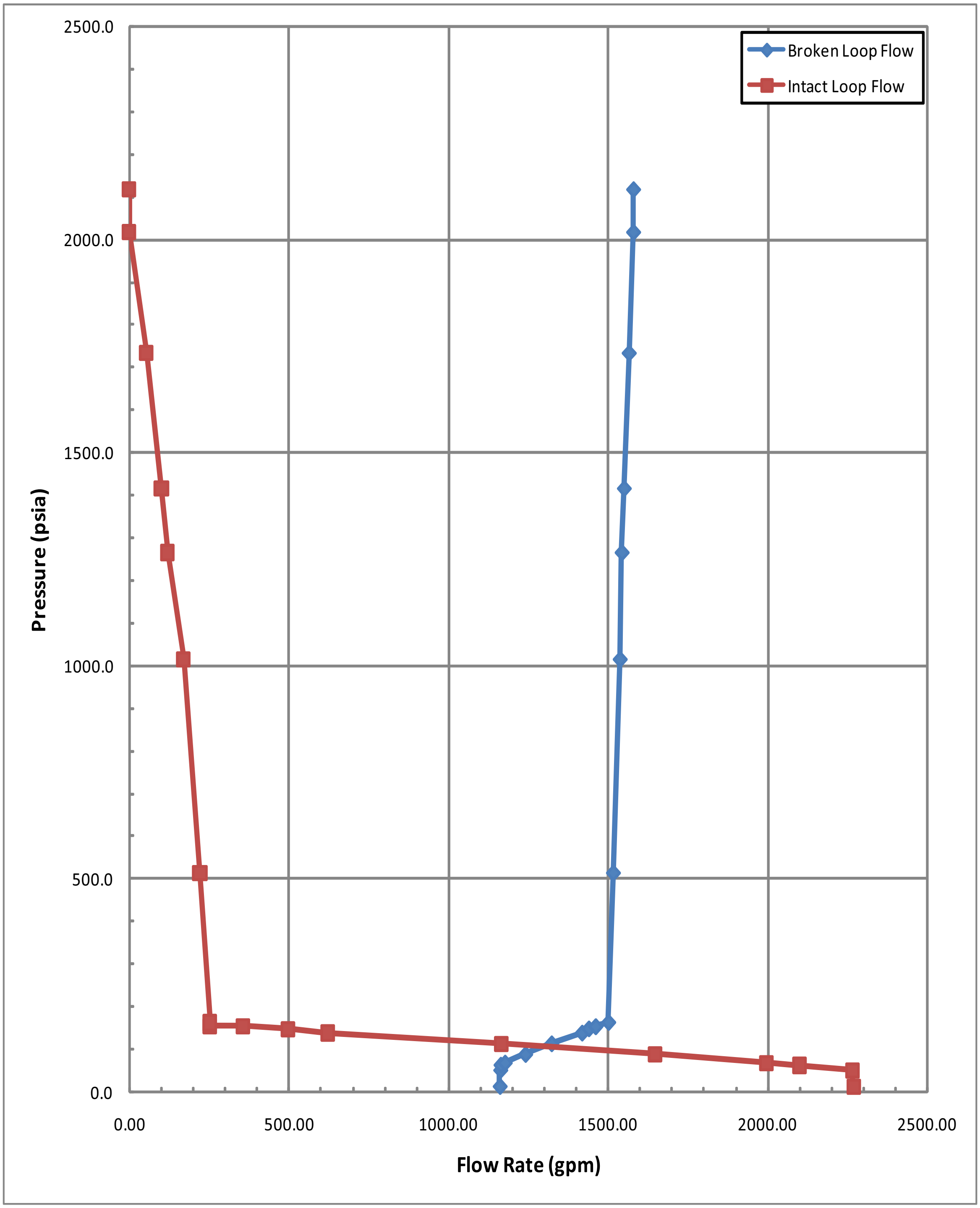


Figure 14.5-17
SBLOCA HOT CHANNEL FACTOR NORMALIZED K(Z) OPERATING ENVELOPE

FQ	K(z)	Elevation (ft)
2.50	1.0	0.0
2.50	1.0	6.0
2.50	1.0	12.0

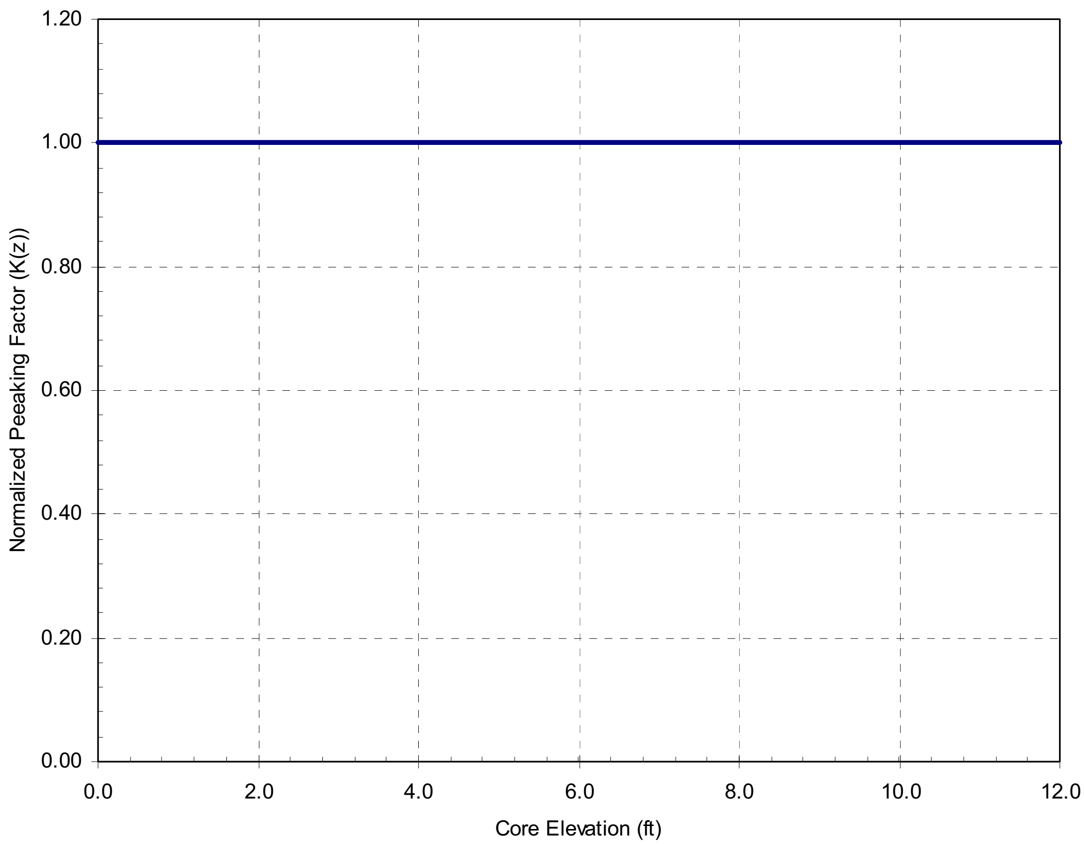


Figure 14.5-18
HOT ASSEMBLY AVERAGE ROD AND HOT ROD AXIAL POWER SHAPES

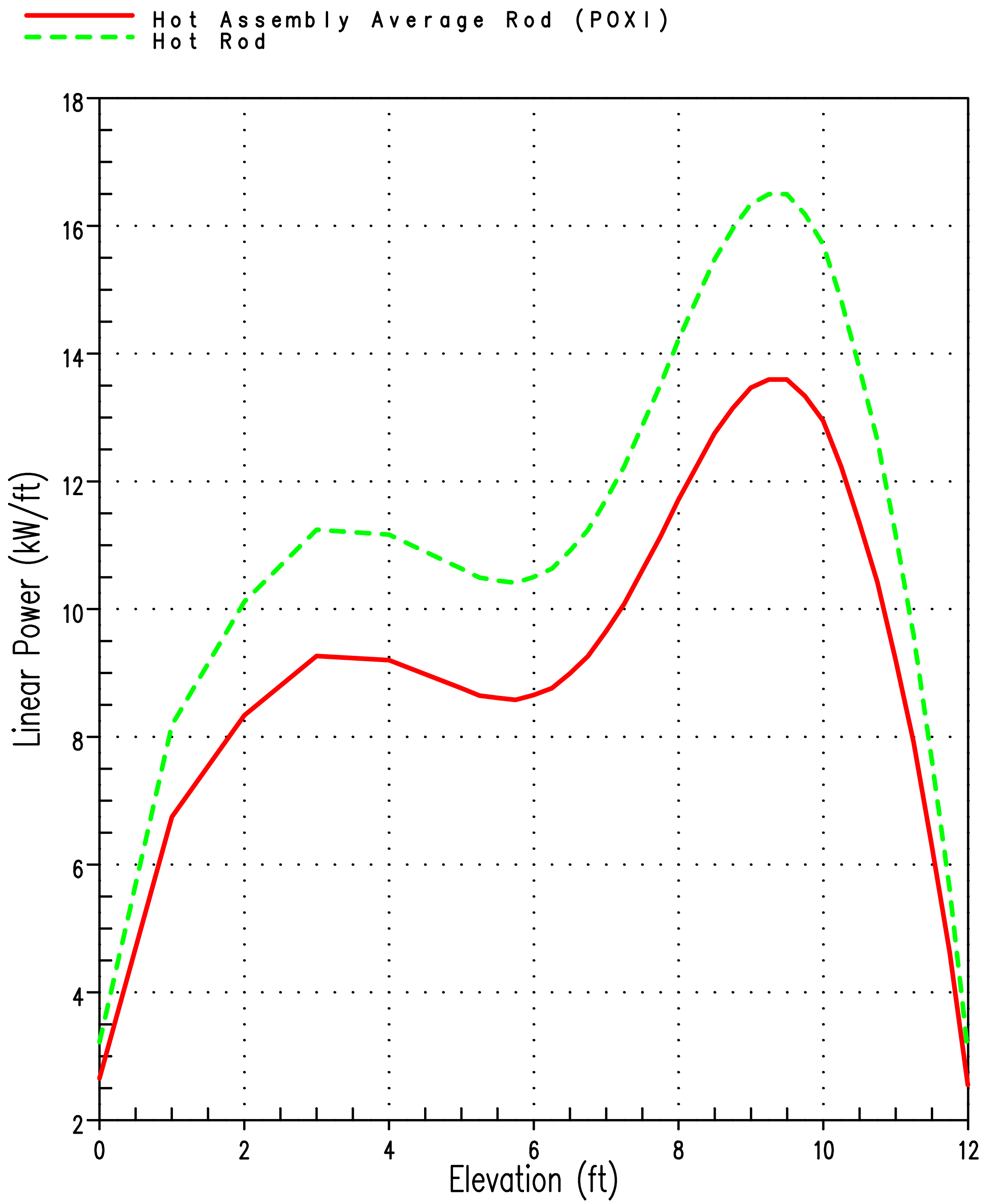


Figure 14.5-19
PRESSURIZER PRESSURE (PSIA), 1.5-INCH BREAK

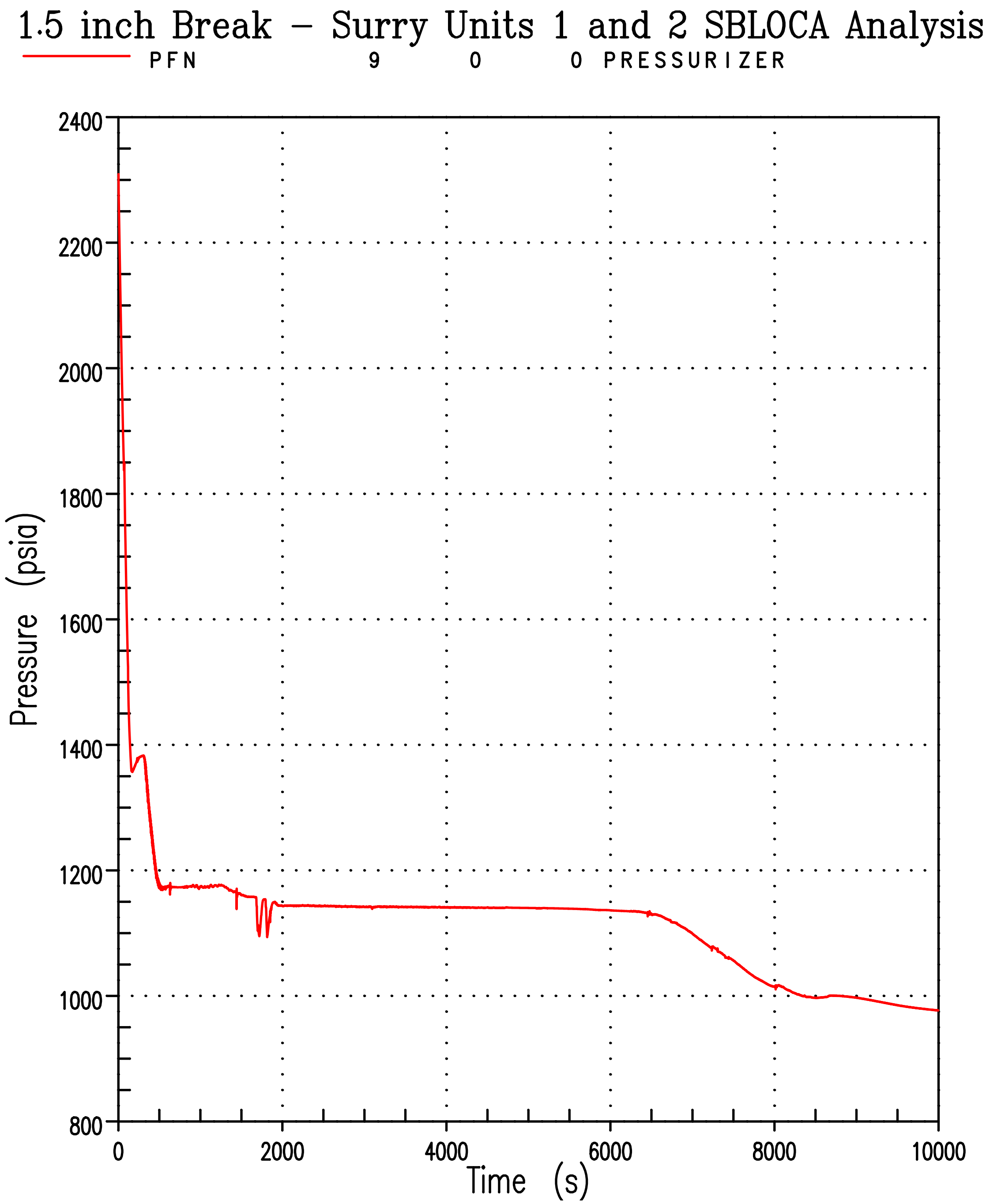


Figure 14.5-20
PRESSURIZER PRESSURE (PSIA), 2-INCH BREAK

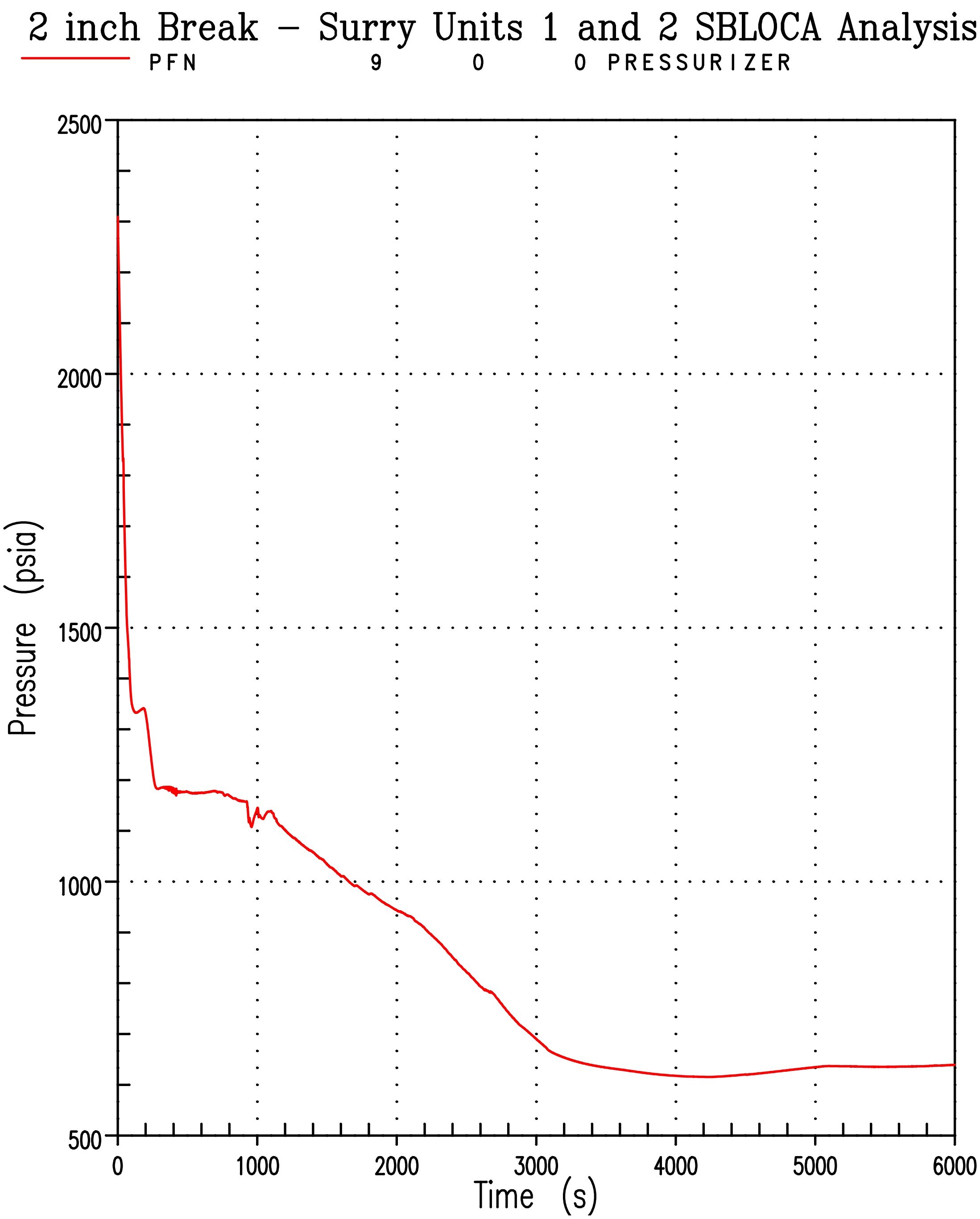


Figure 14.5-21
PRESSURIZER PRESSURE (PSIA), 2.25-INCH BREAK

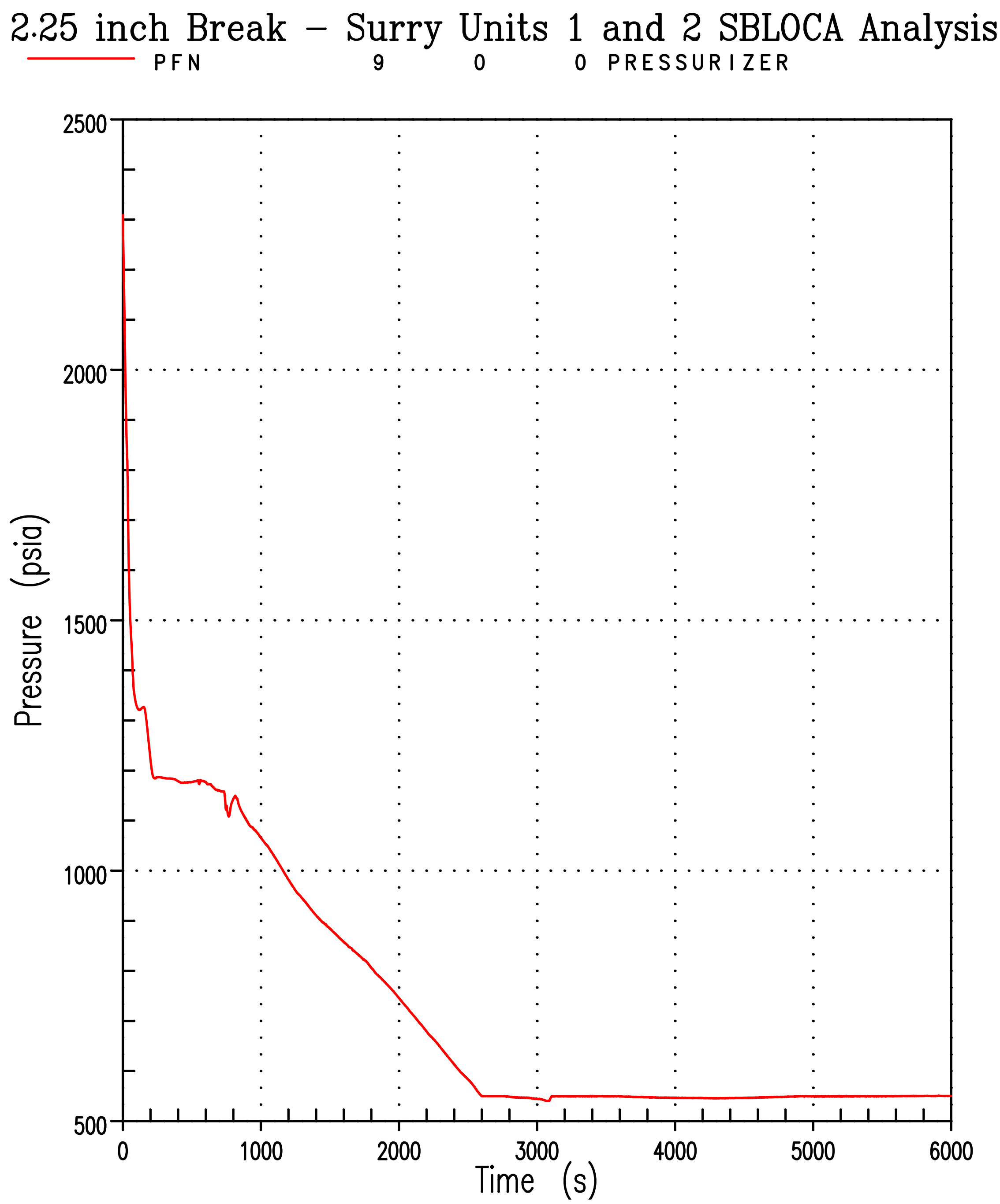


Figure 14.5-22
PRESSURIZER PRESSURE (PSIA), 2.5-INCH BREAK

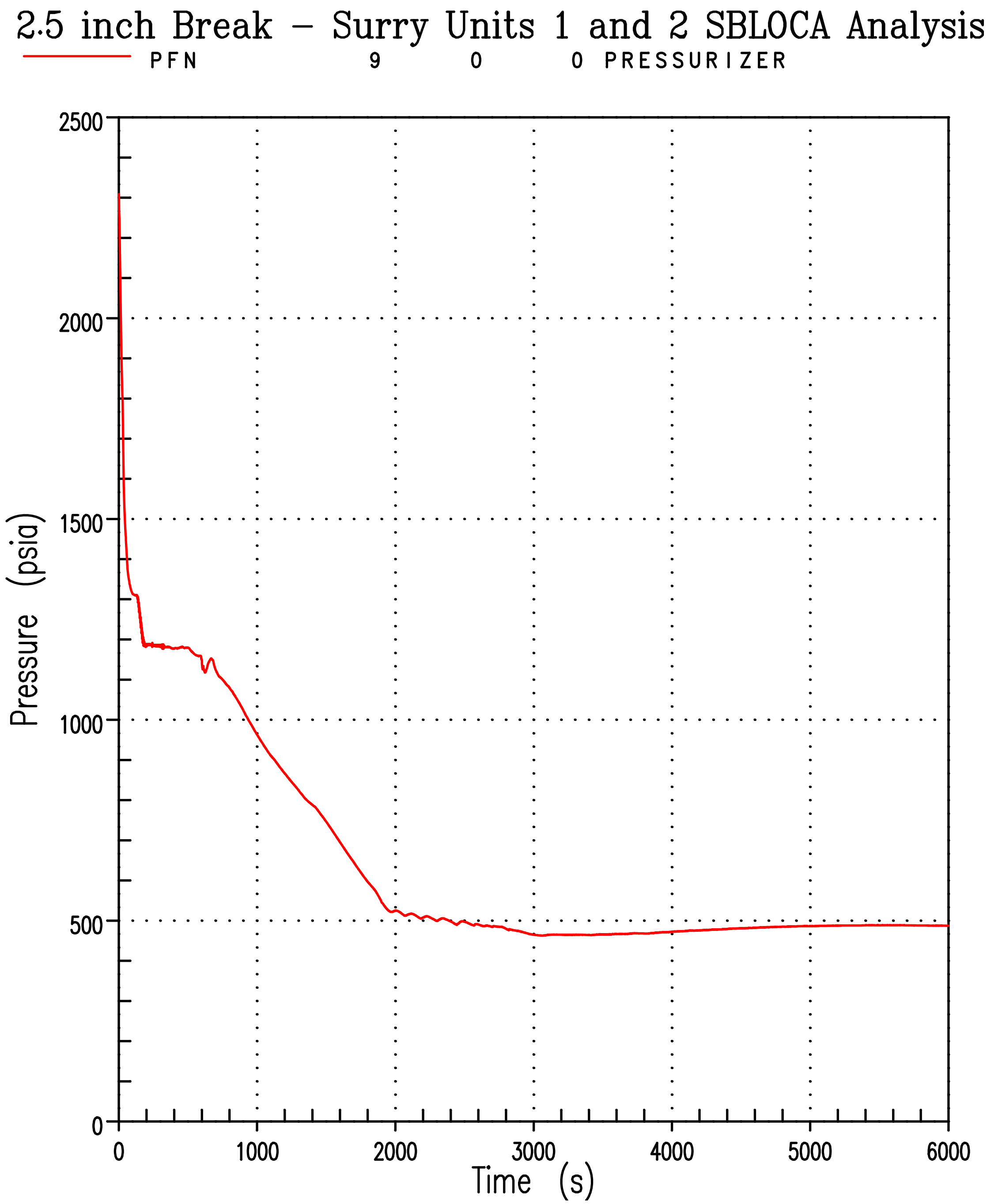


Figure 14.5-23
PRESSURIZER PRESSURE (PSIA), 2.75-INCH BREAK

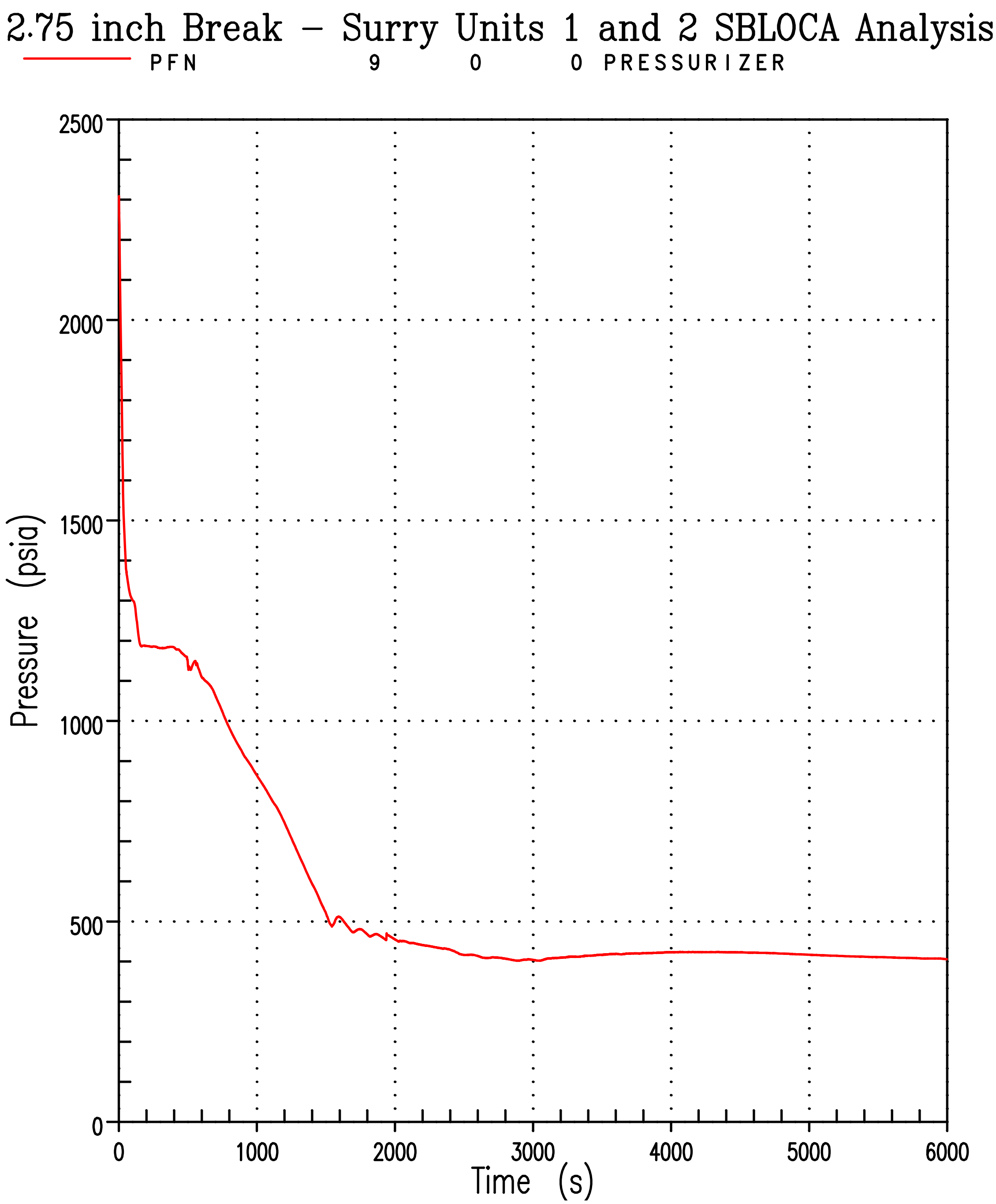


Figure 14.5-24
PRESSURIZER PRESSURE (PSIA), 3-INCH BREAK

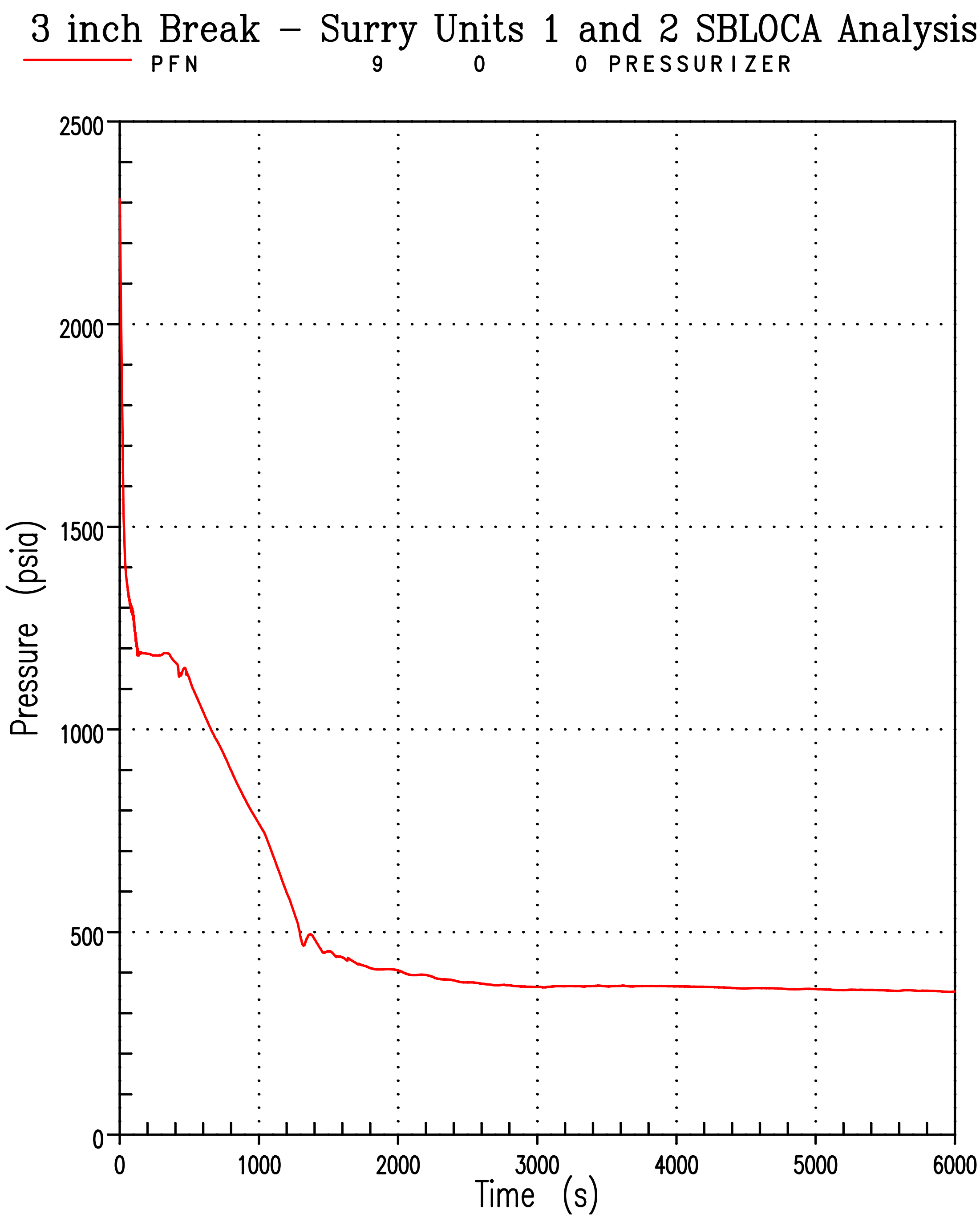


Figure 14.5-25
PRESSURIZER PRESSURE (PSIA), 4-INCH BREAK

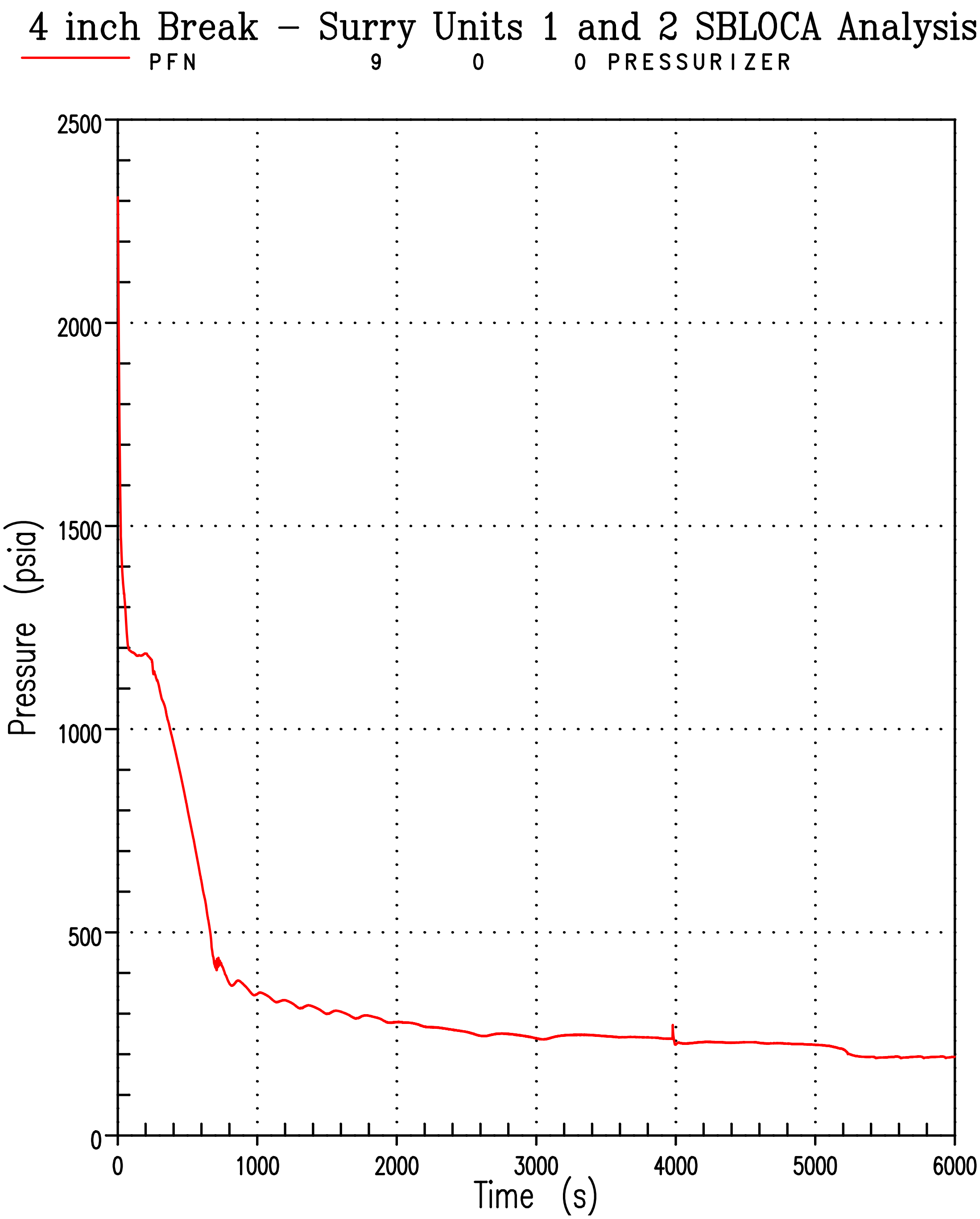


Figure 14.5-26
PRESSURIZER PRESSURE (PSIA), 5.5-INCH BREAK

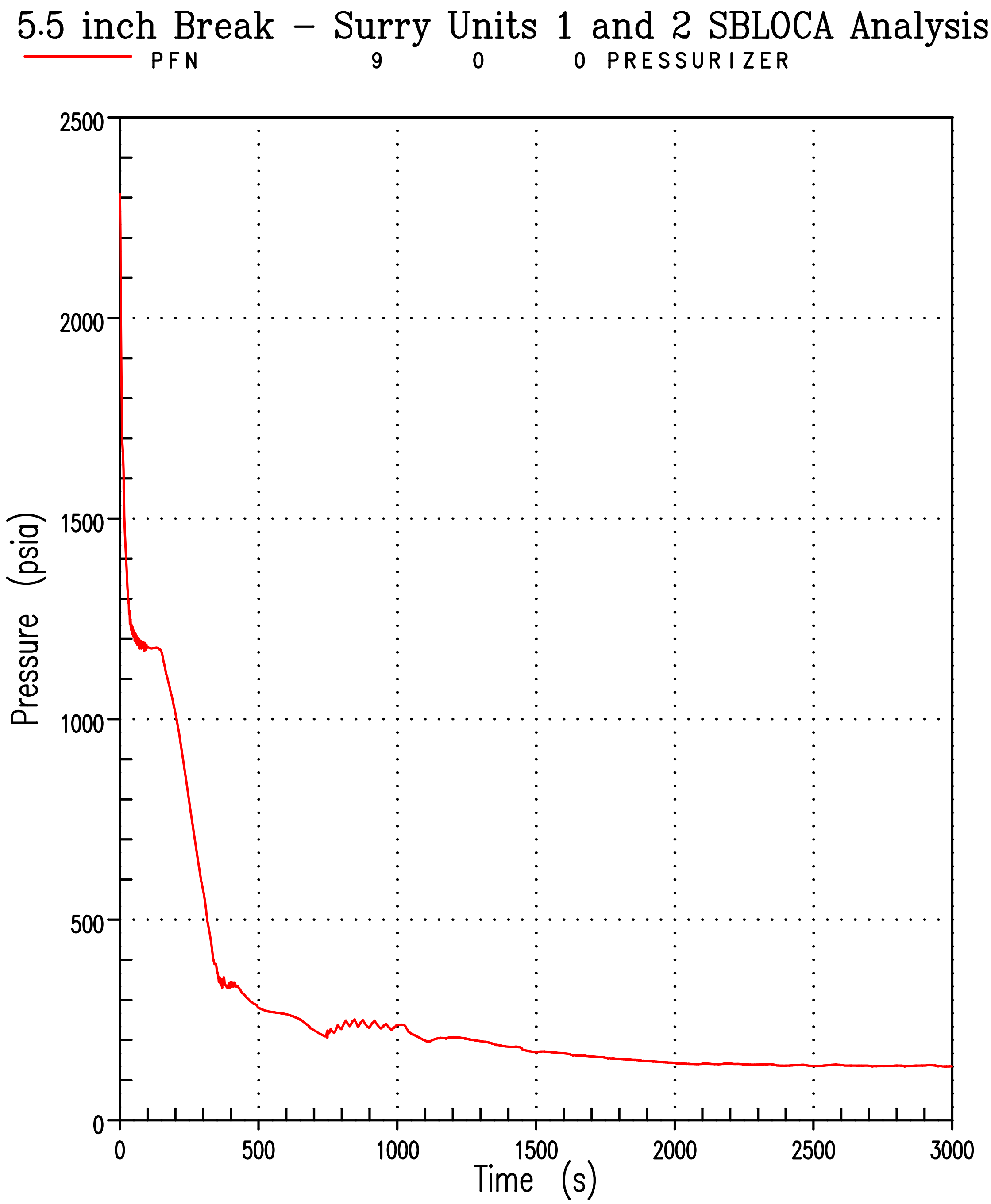


Figure 14.5-27
CORE MIXTURE LEVEL, 1.5-INCH BREAK

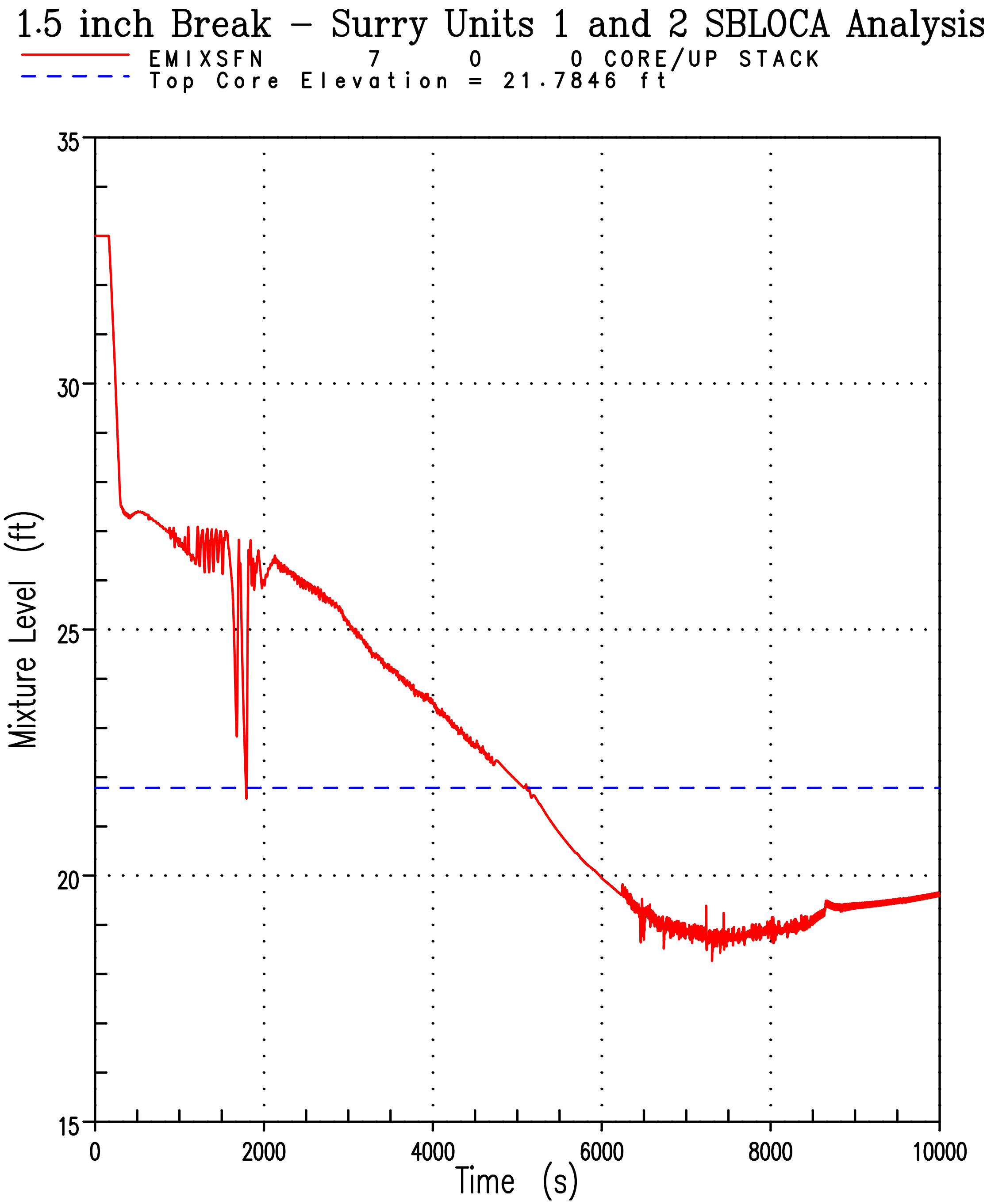


Figure 14.5-28
CORE MIXTURE LEVEL, 2-INCH BREAK

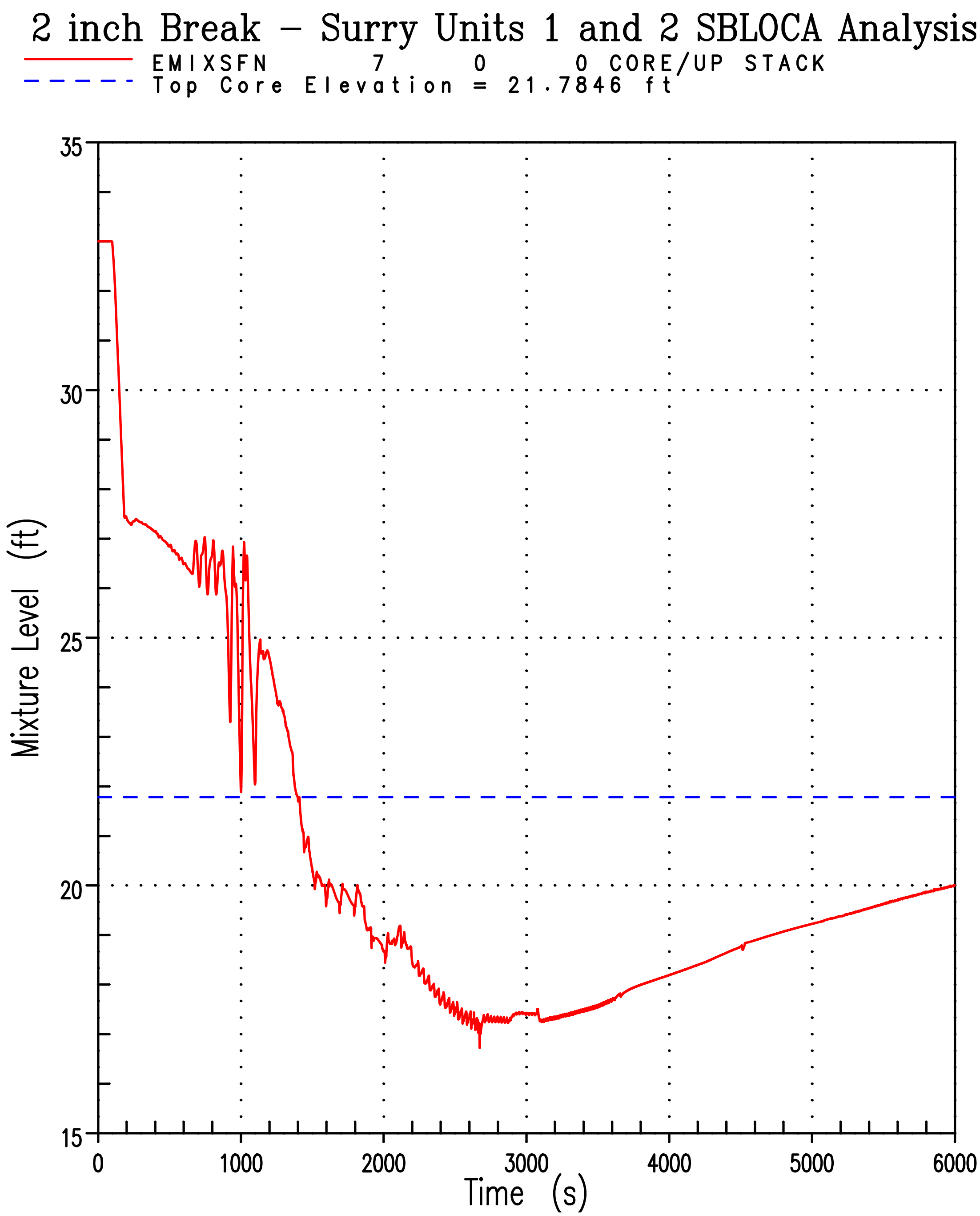


Figure 14.5-29
CORE MIXTURE LEVEL, 2.25-INCH BREAK

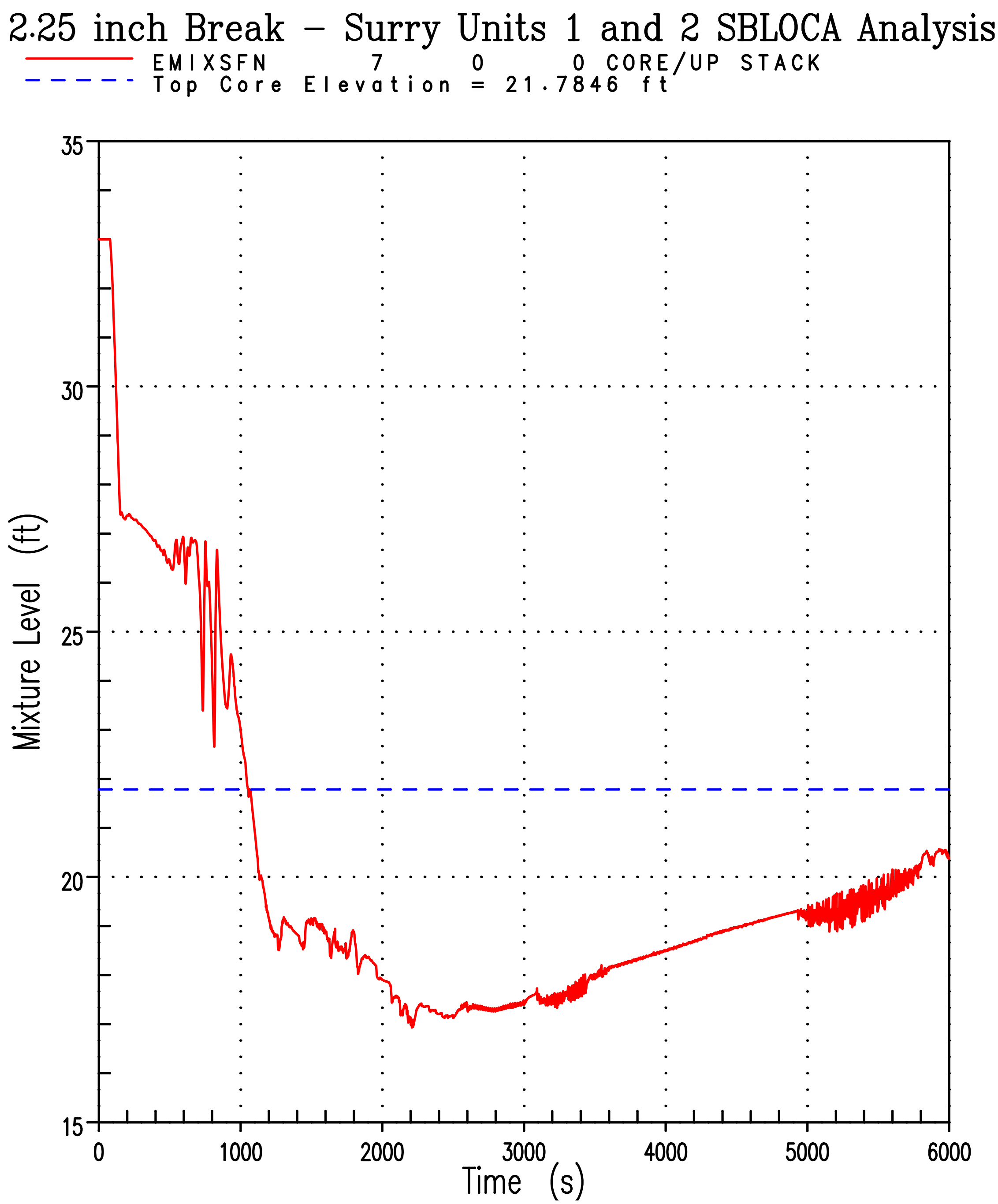


Figure 14.5-30
CORE MIXTURE LEVEL, 2.5-INCH BREAK

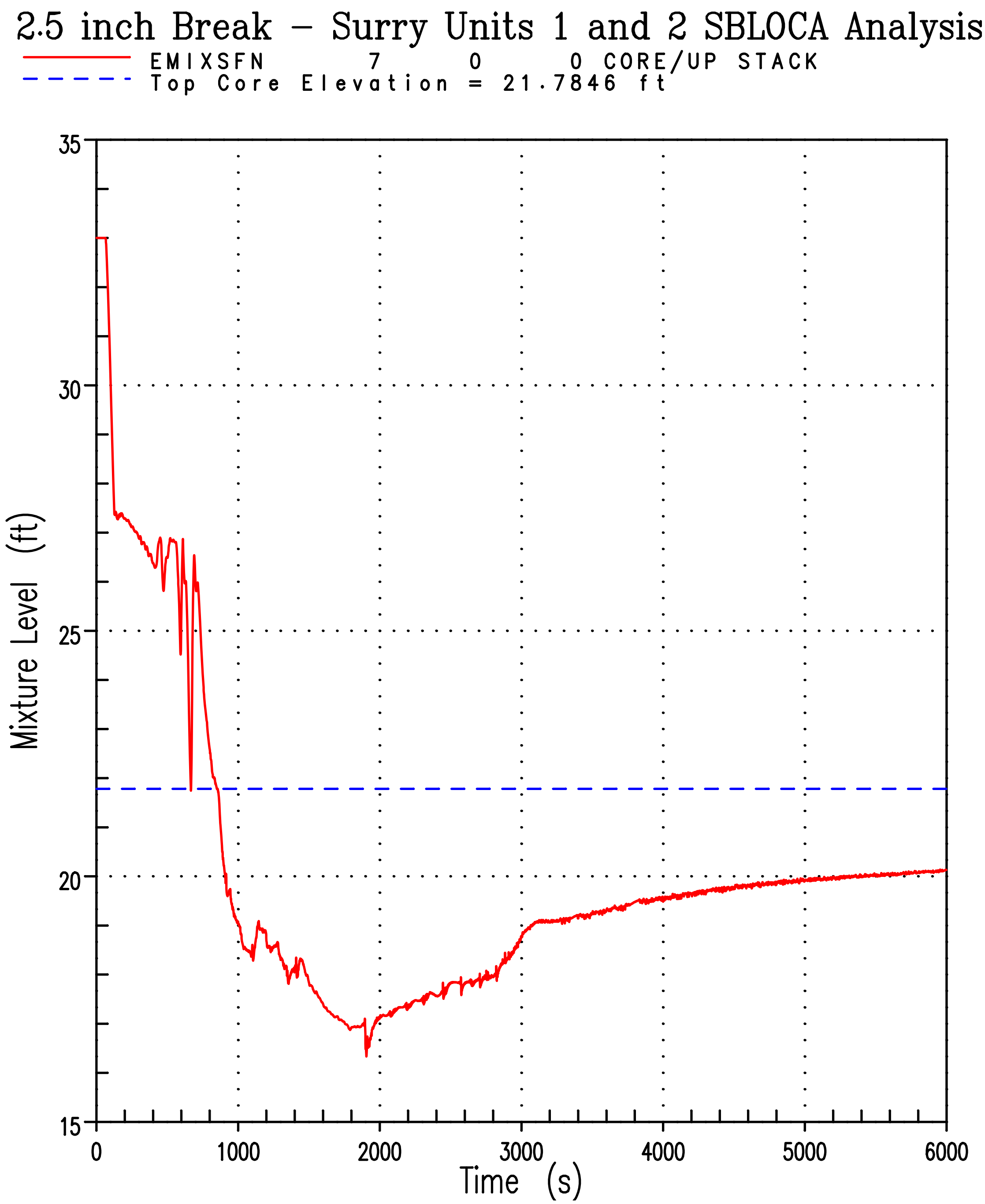


Figure 14.5-31
CORE MIXTURE LEVEL, 2.75-INCH BREAK

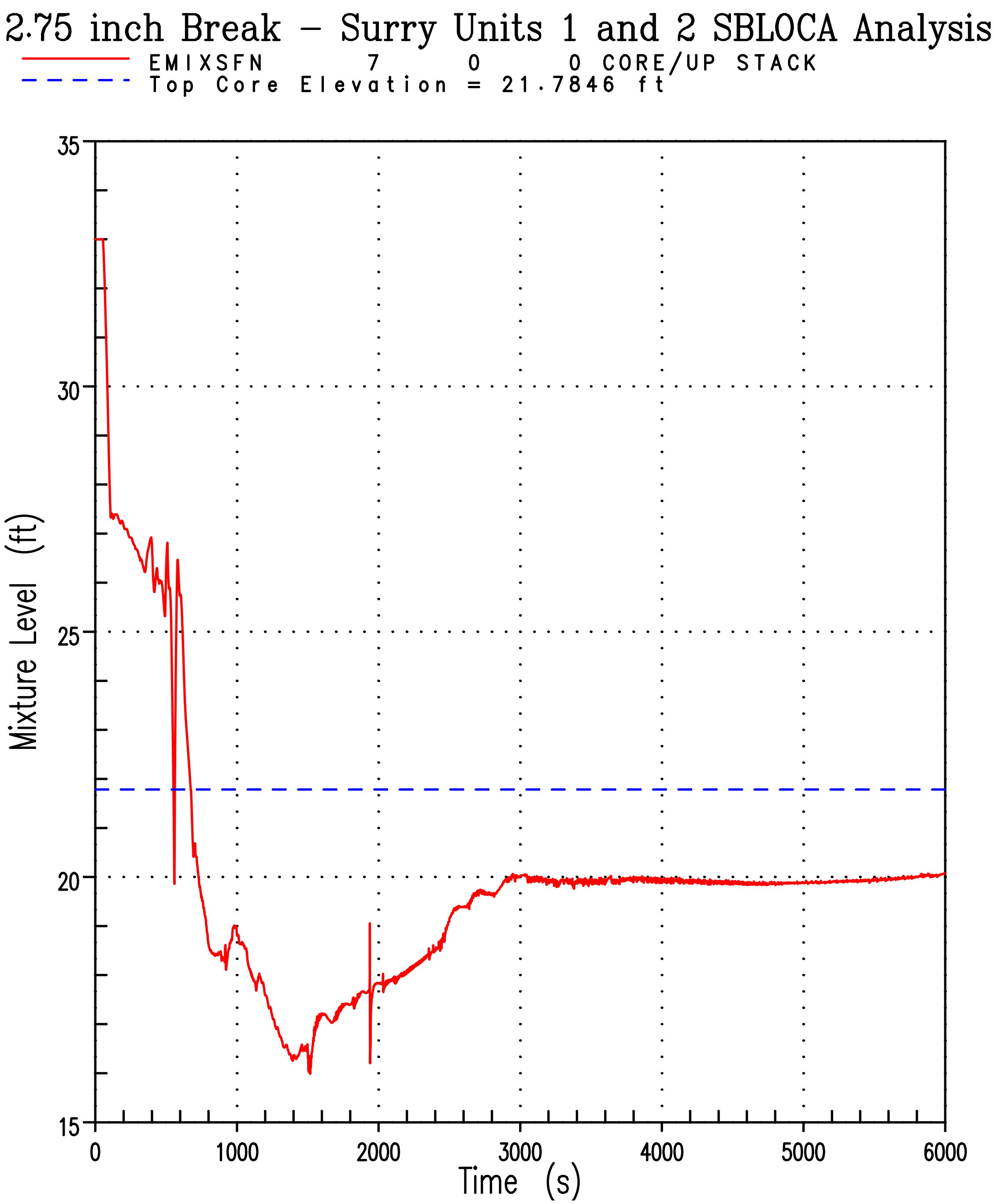


Figure 14.5-32
CORE MIXTURE LEVEL, 3-INCH BREAK

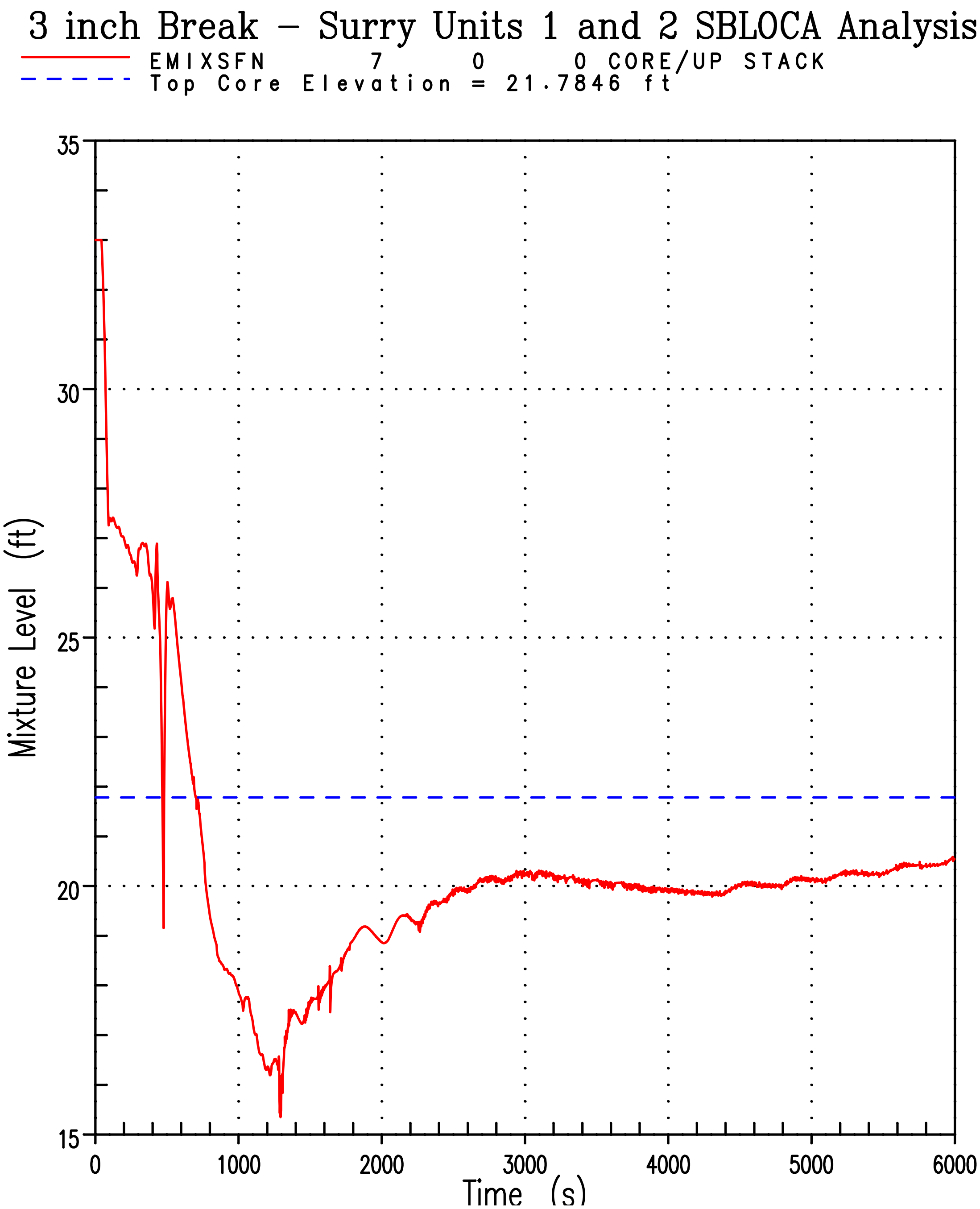


Figure 14.5-33
CORE MIXTURE LEVEL, 4-INCH BREAK

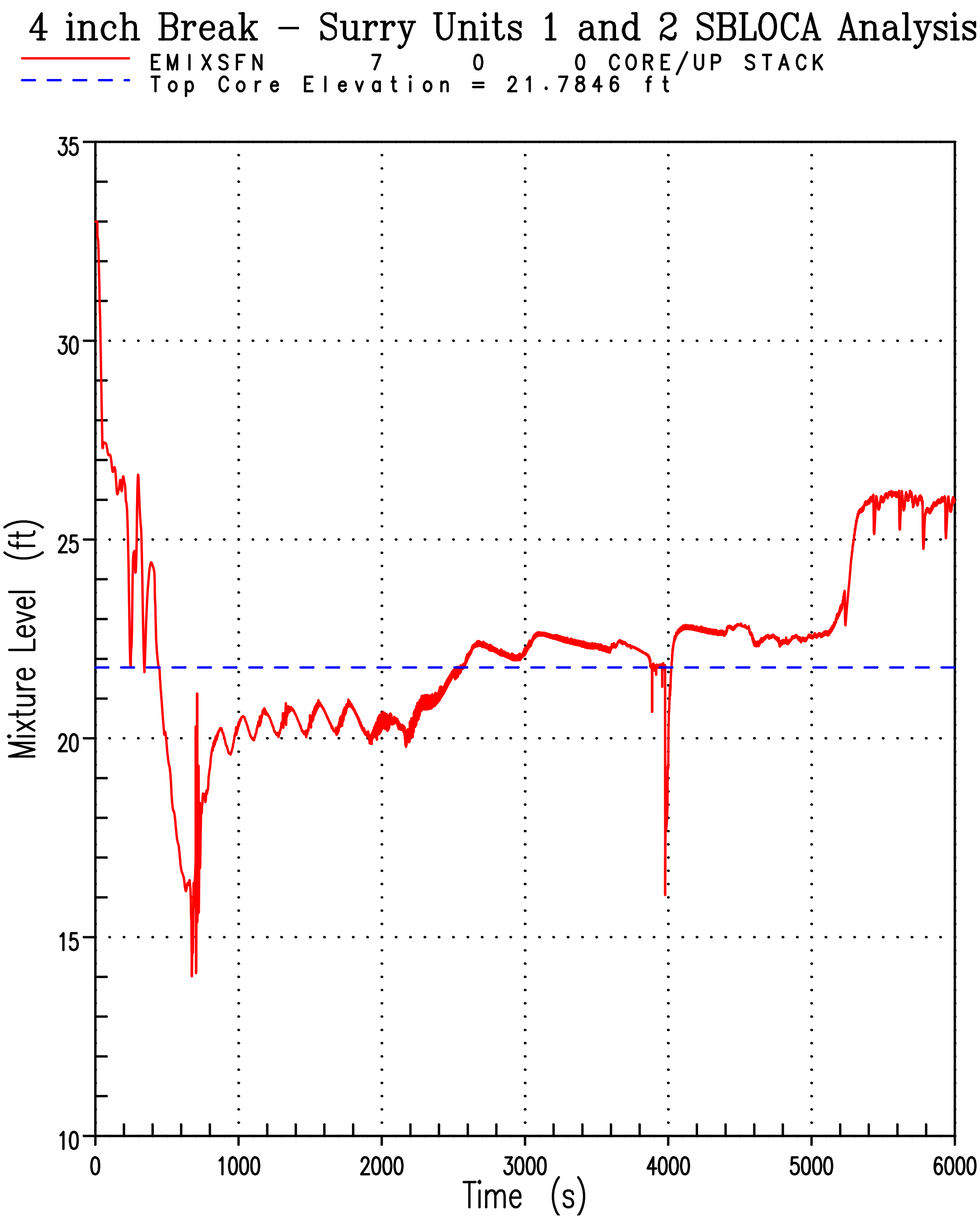


Figure 14.5-34
CORE MIXTURE LEVEL, 5.5-INCH BREAK

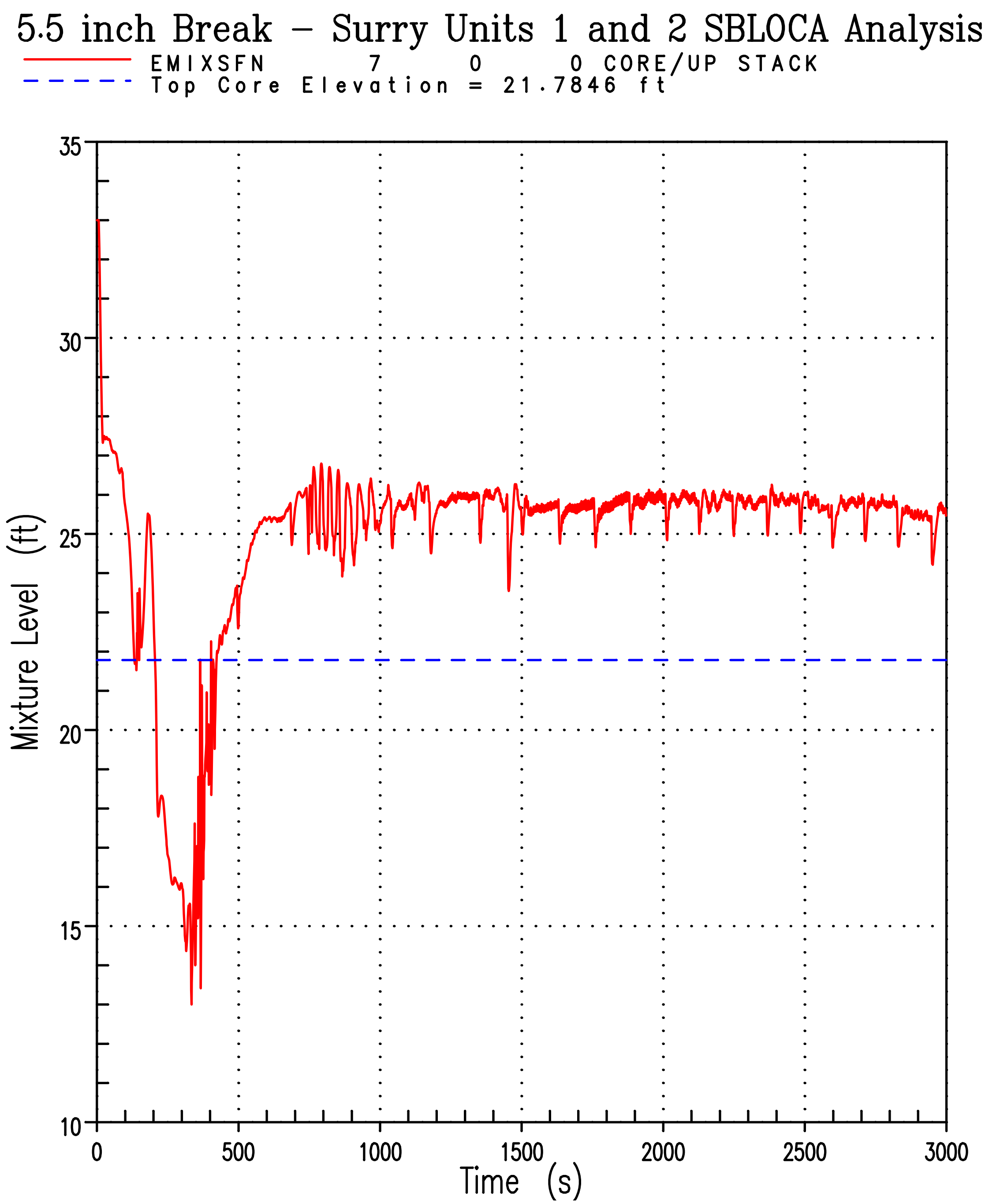


Figure 14.5-35
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 1.5-INCH BREAK

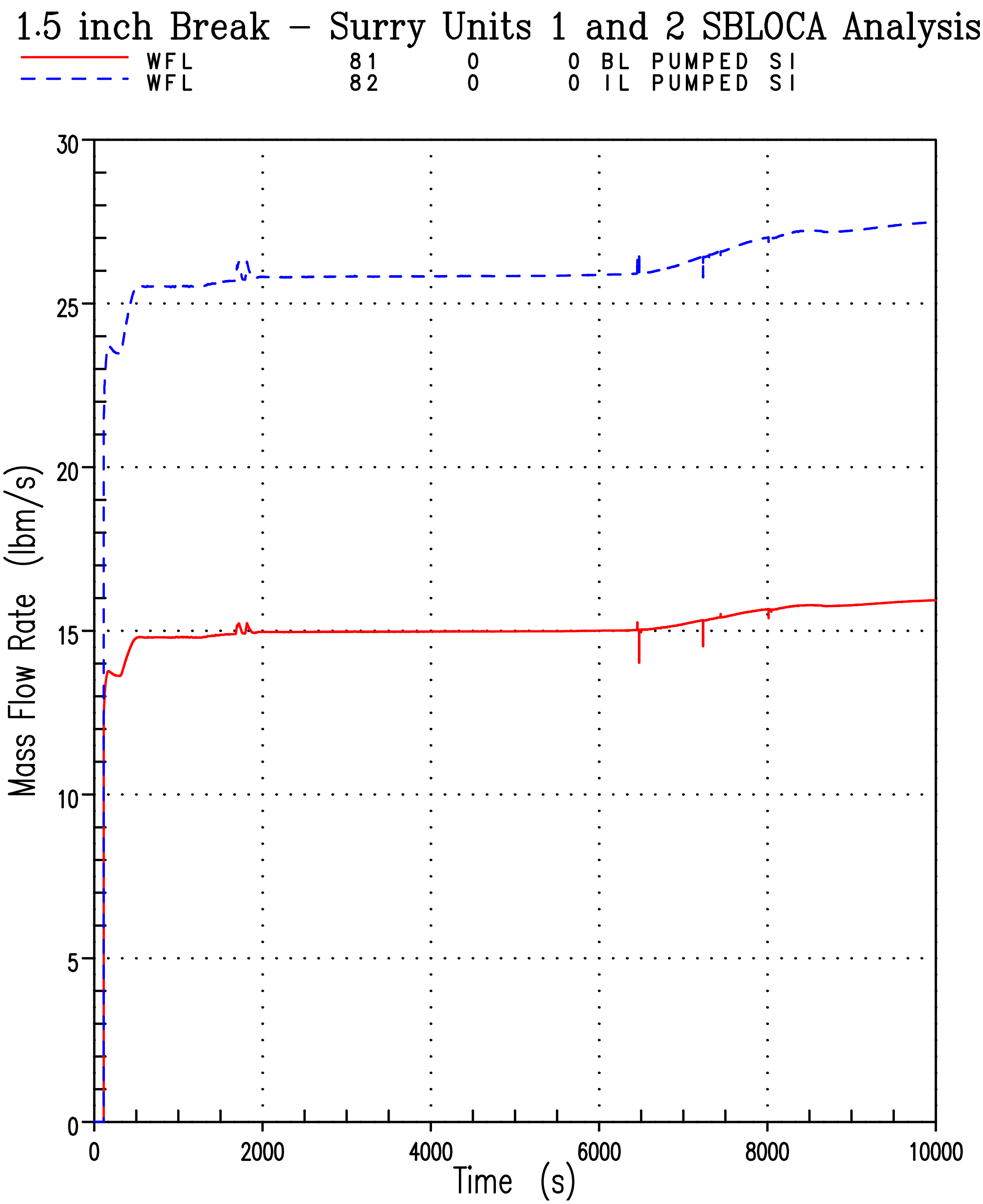


Figure 14.5-36
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 2-INCH BREAK

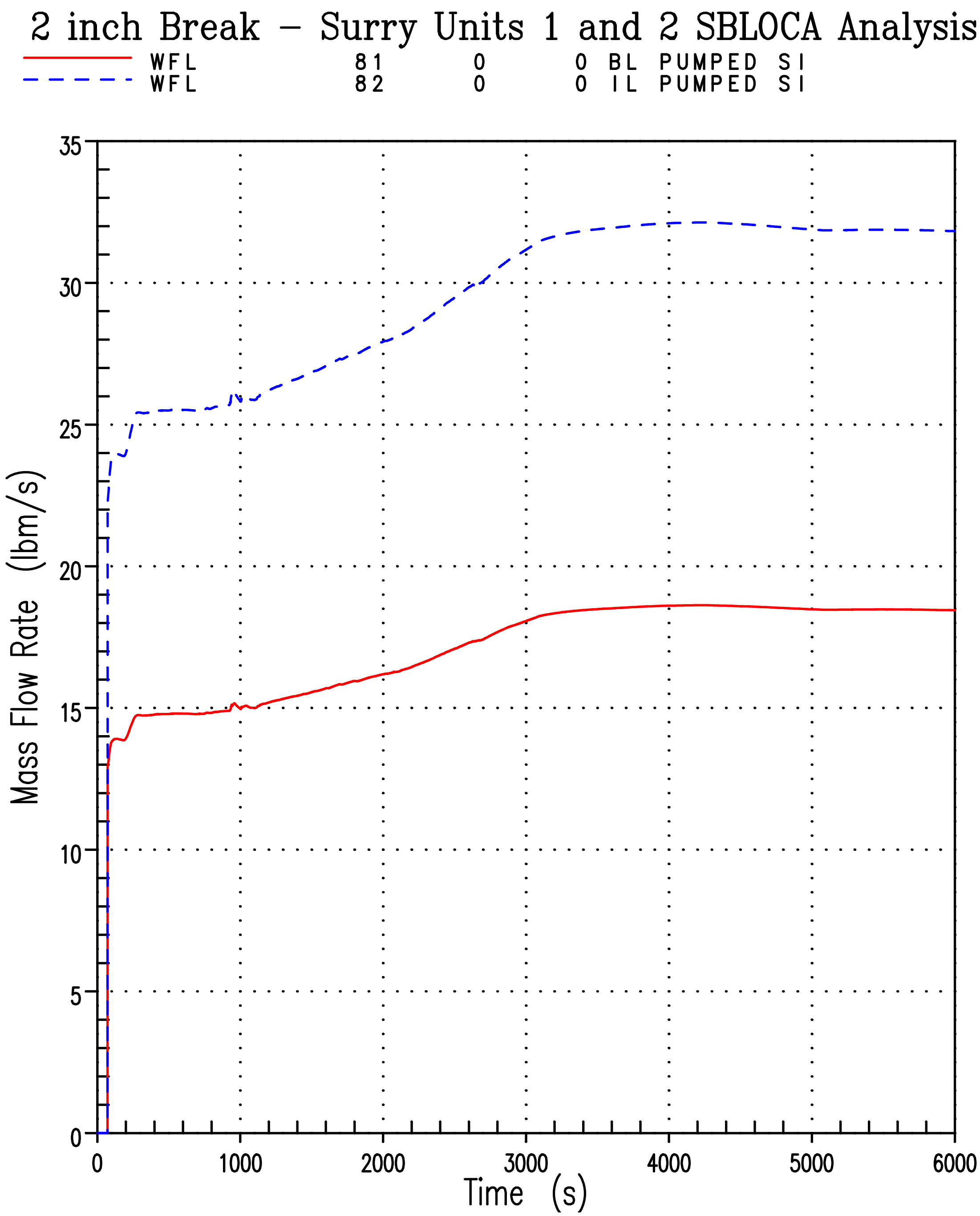


Figure 14.5-37
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 2.25-INCH BREAK

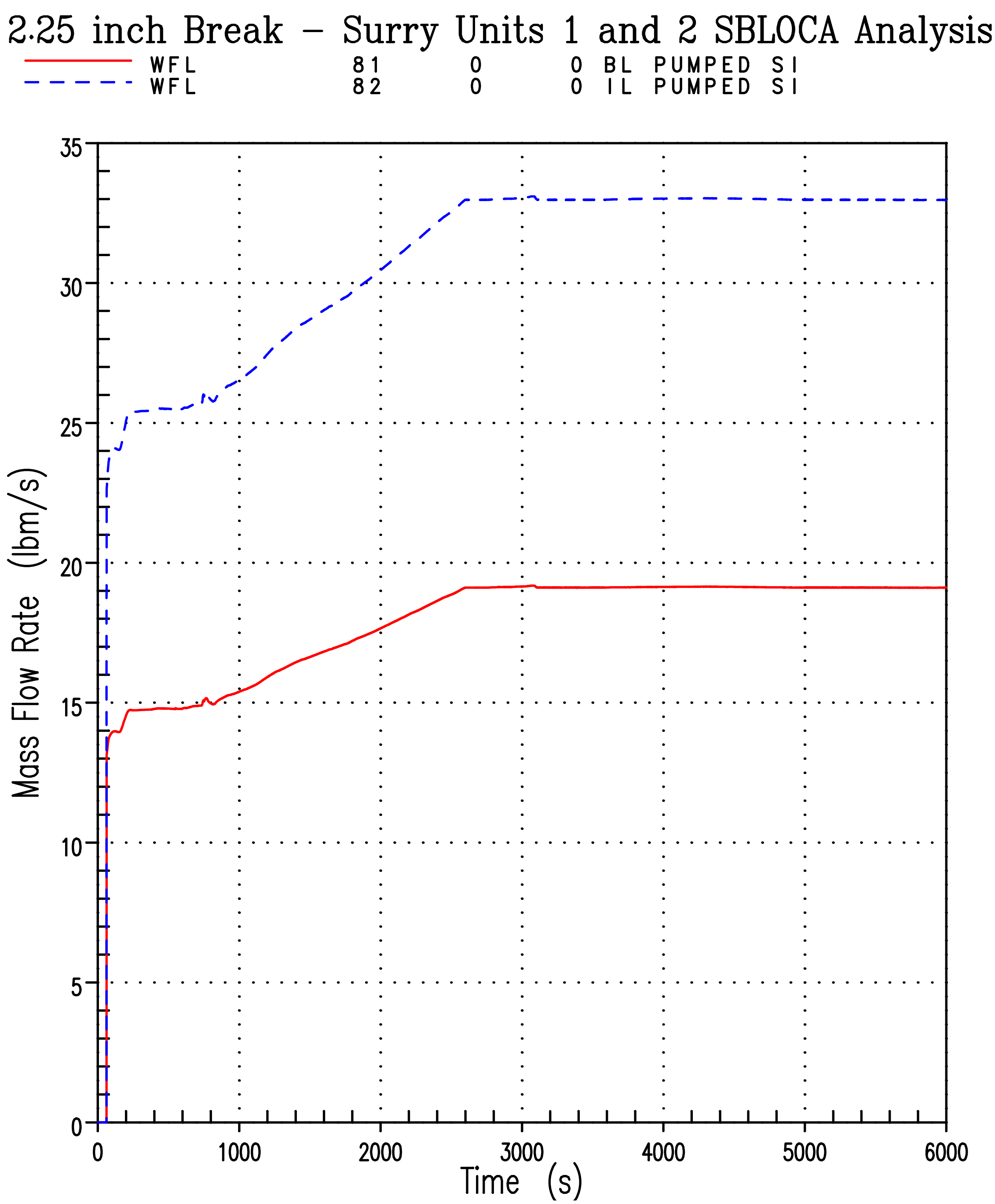


Figure 14.5-38
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 2.5-INCH BREAK

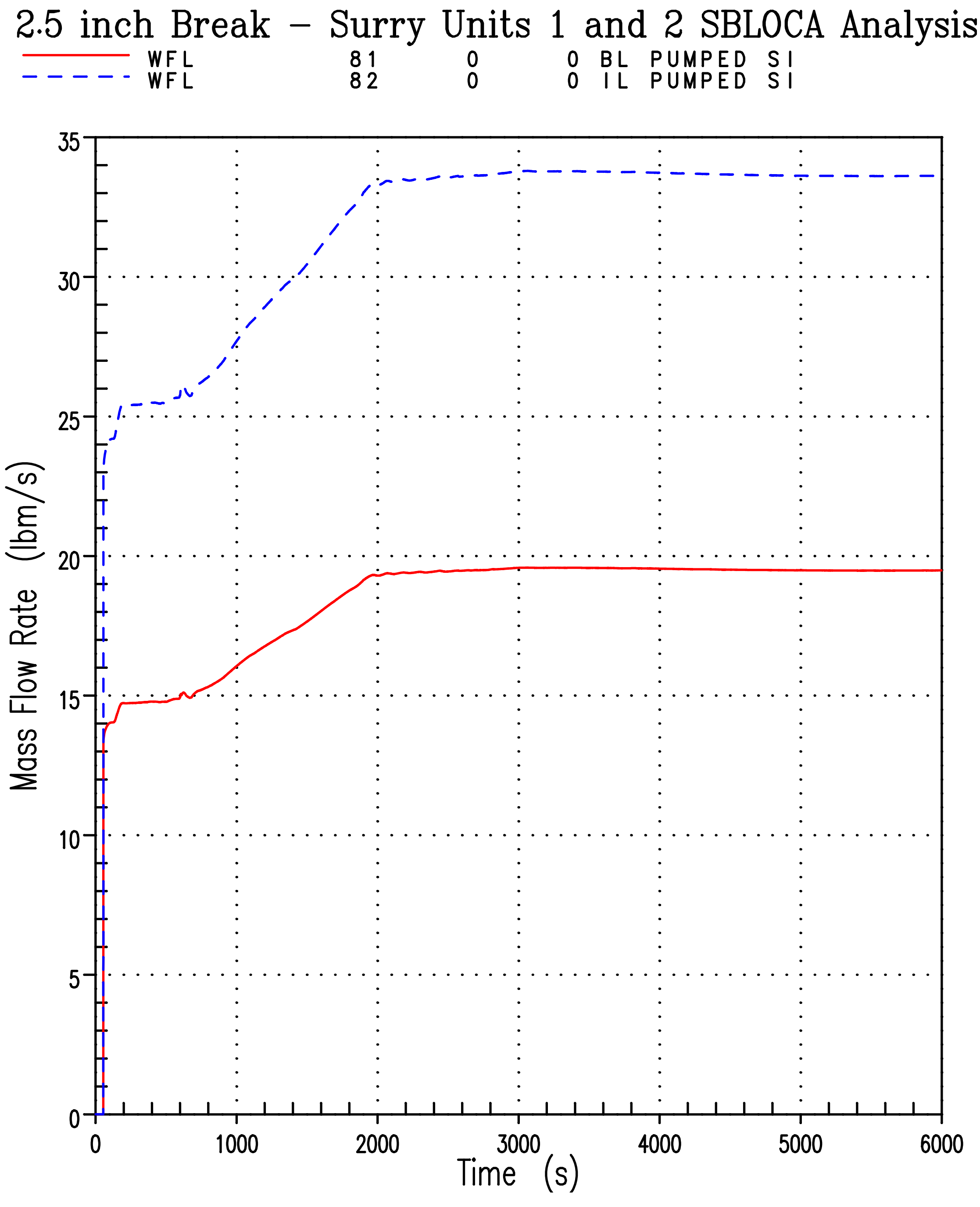


Figure 14.5-39
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 2.75-INCH BREAK

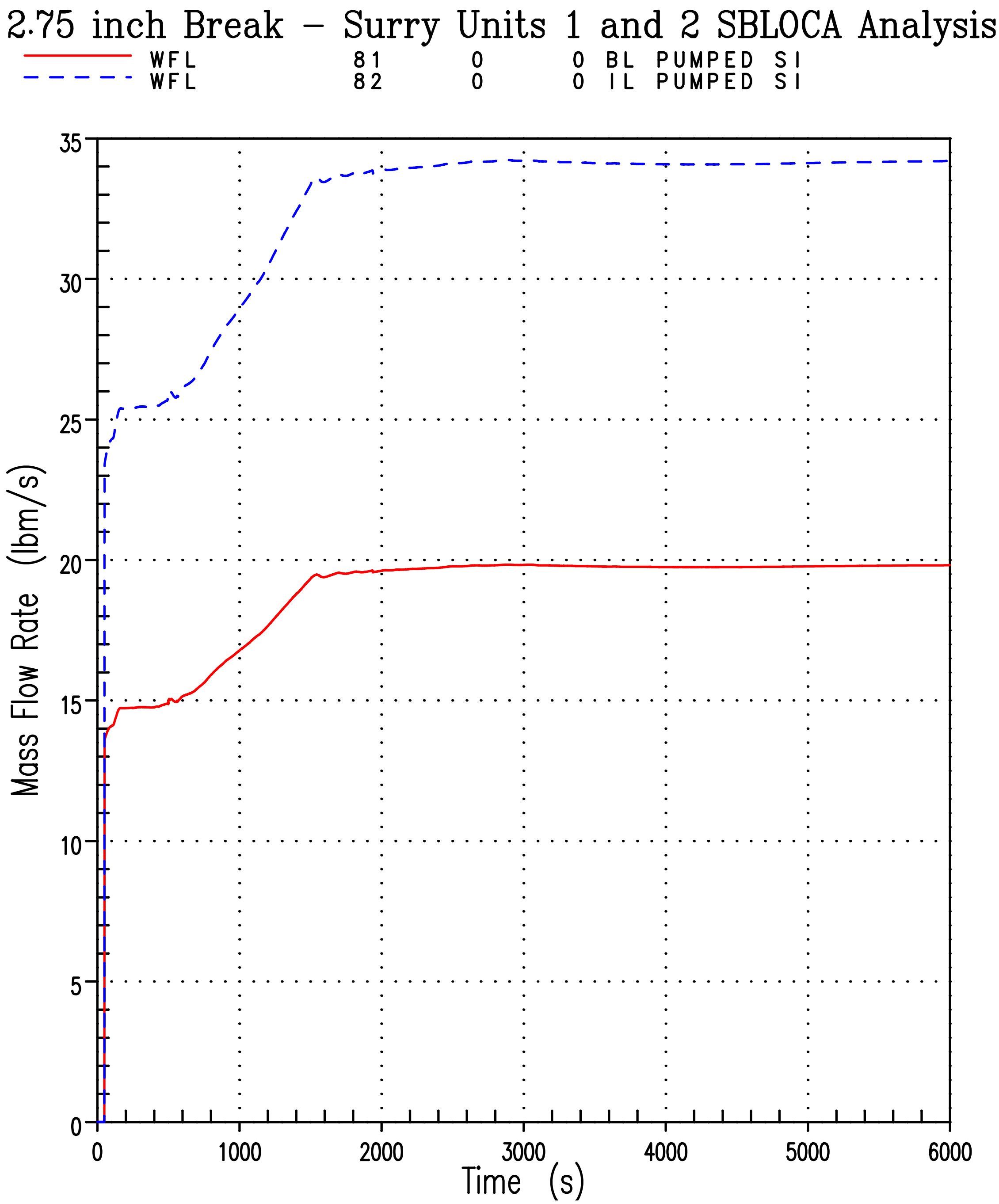


Figure 14.5-40
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 3-INCH BREAK

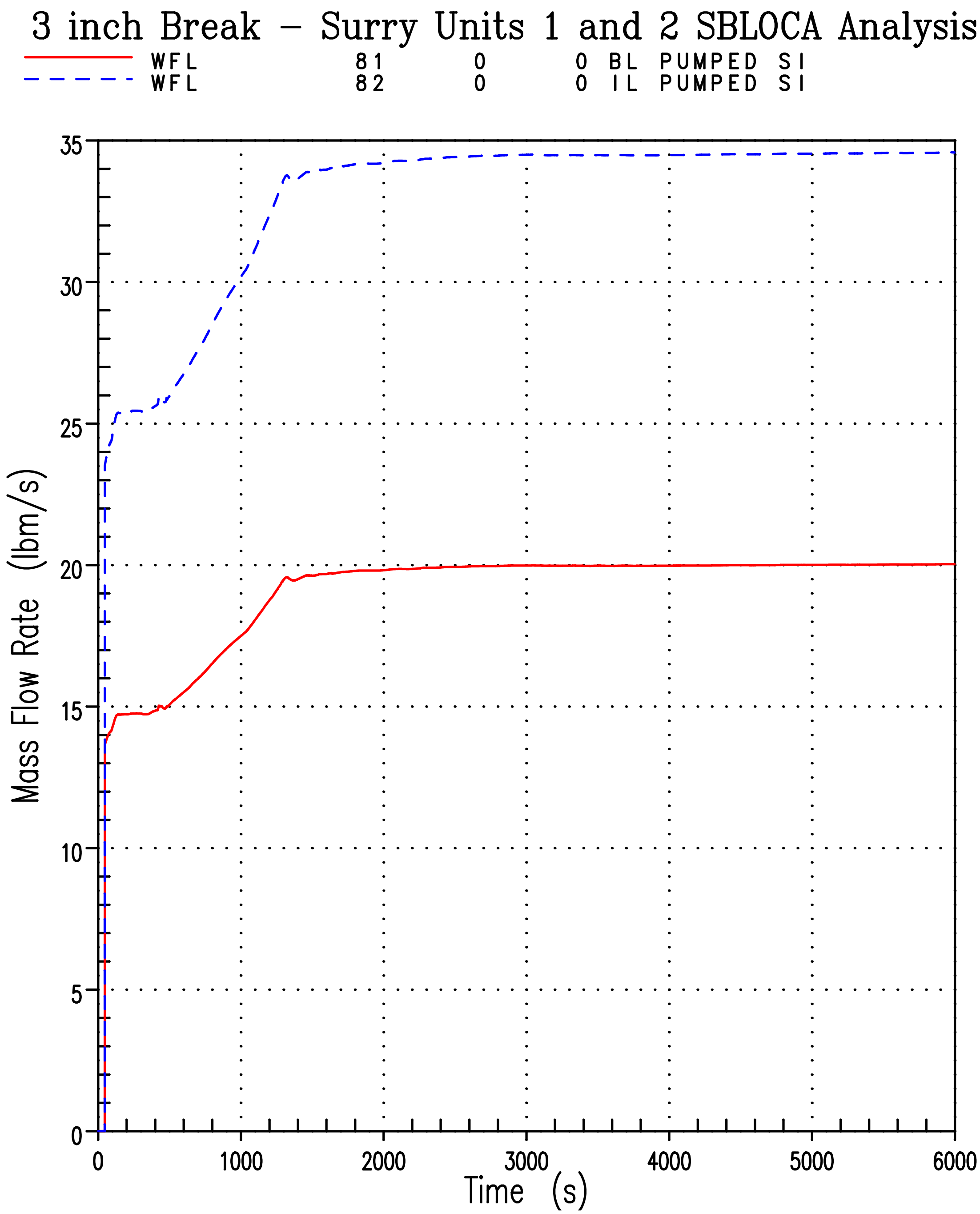


Figure 14.5-41
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 4-INCH BREAK

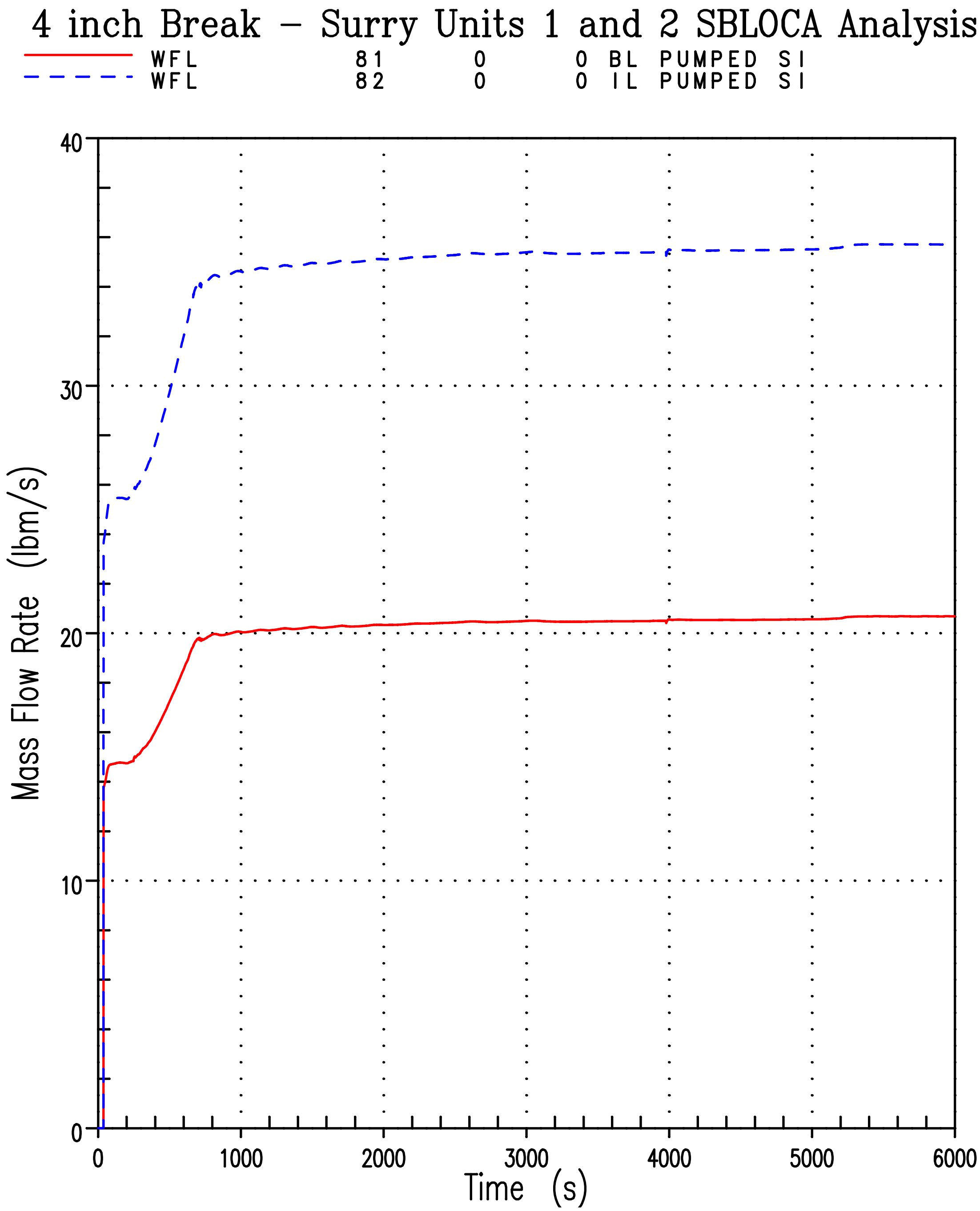


Figure 14.5-42
BROKEN AND INTACT LOOP PUMPED SI FLOW RATE, 5.5-INCH BREAK

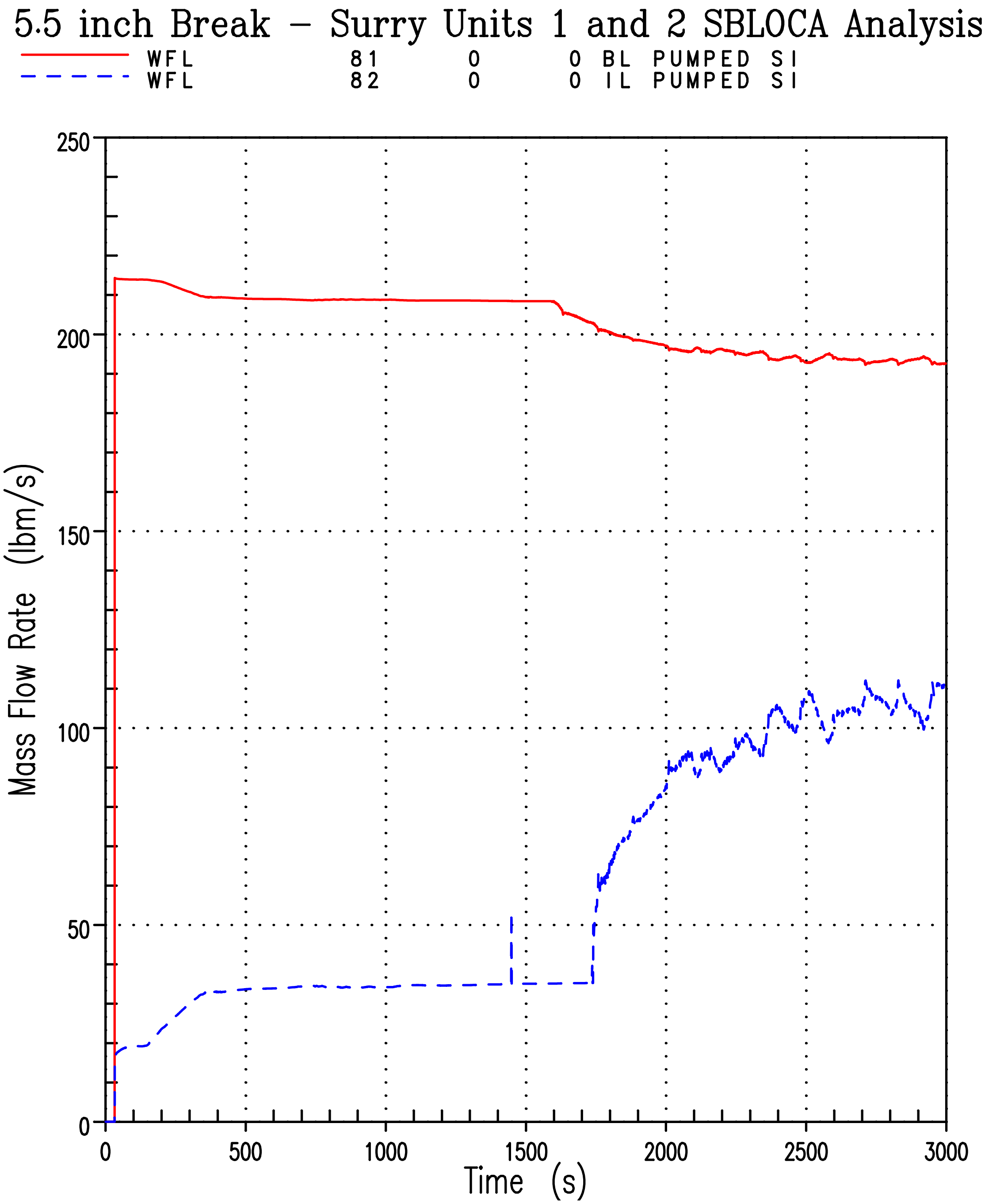


Figure 14.5-43
CORE EXIT VAPOR FLOW RATE, 1.5-INCH BREAK

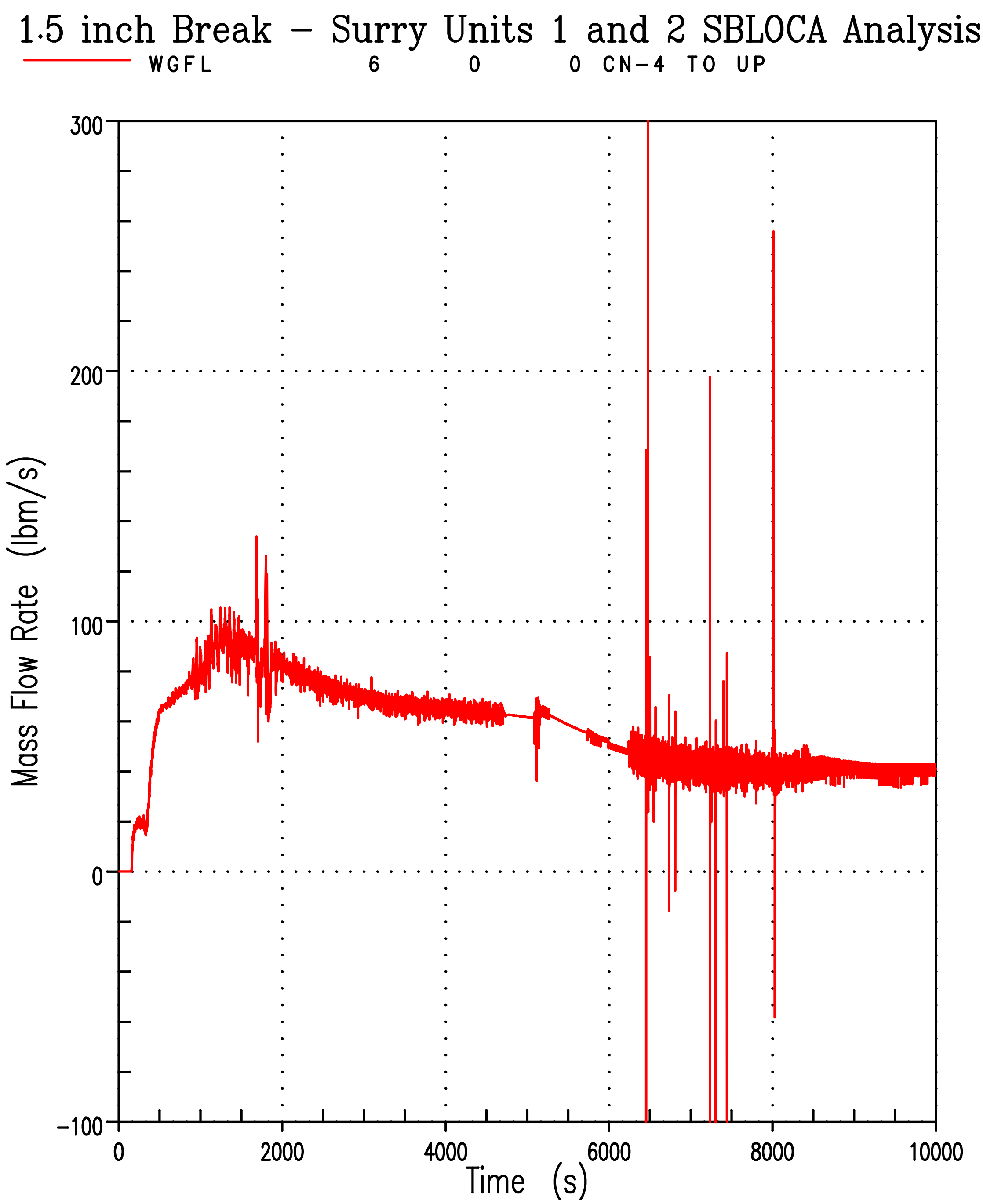


Figure 14.5-44
CORE EXIT VAPOR FLOW RATE, 2-INCH BREAK

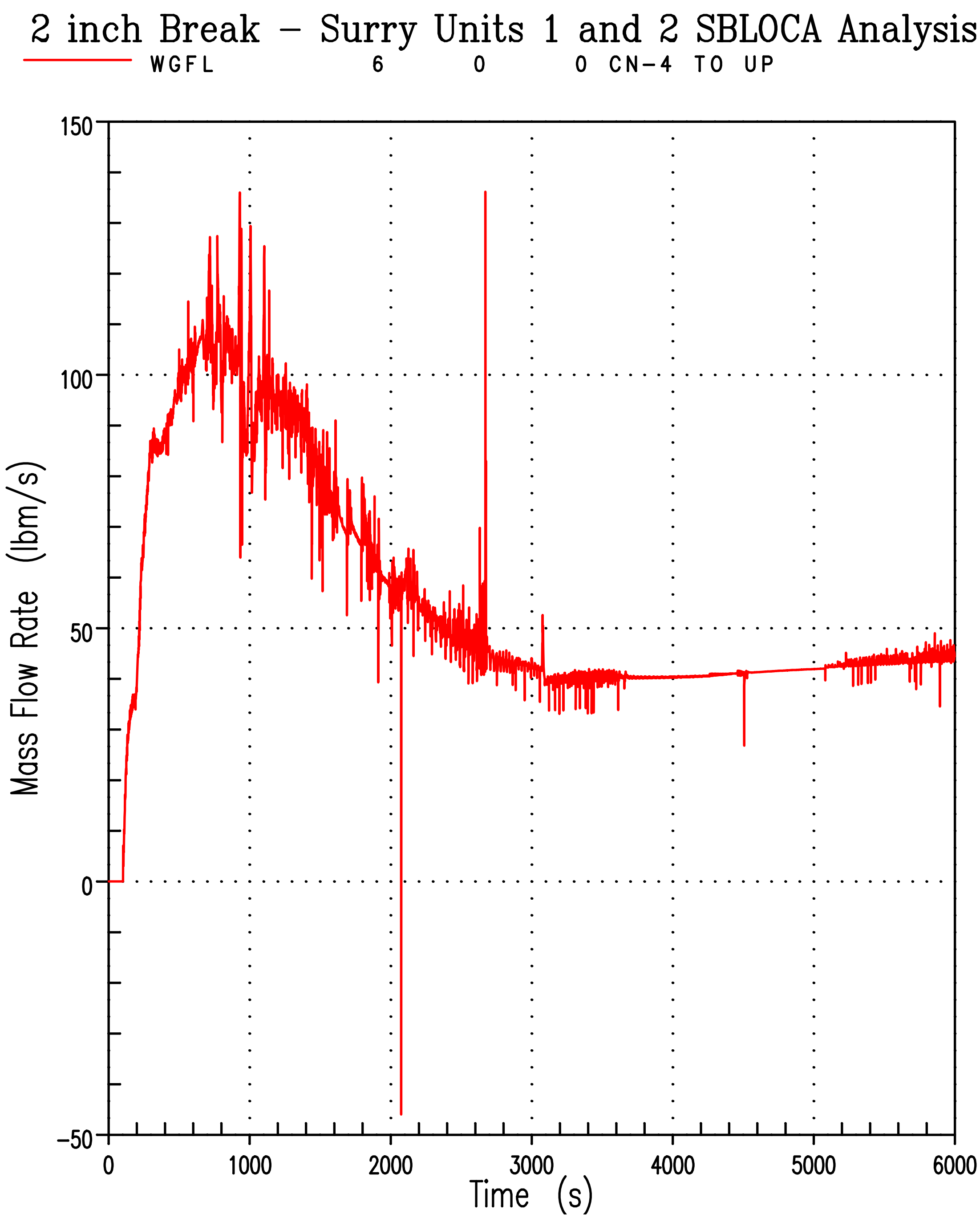


Figure 14.5-45
CORE EXIT VAPOR FLOW RATE, 2.25-INCH BREAK

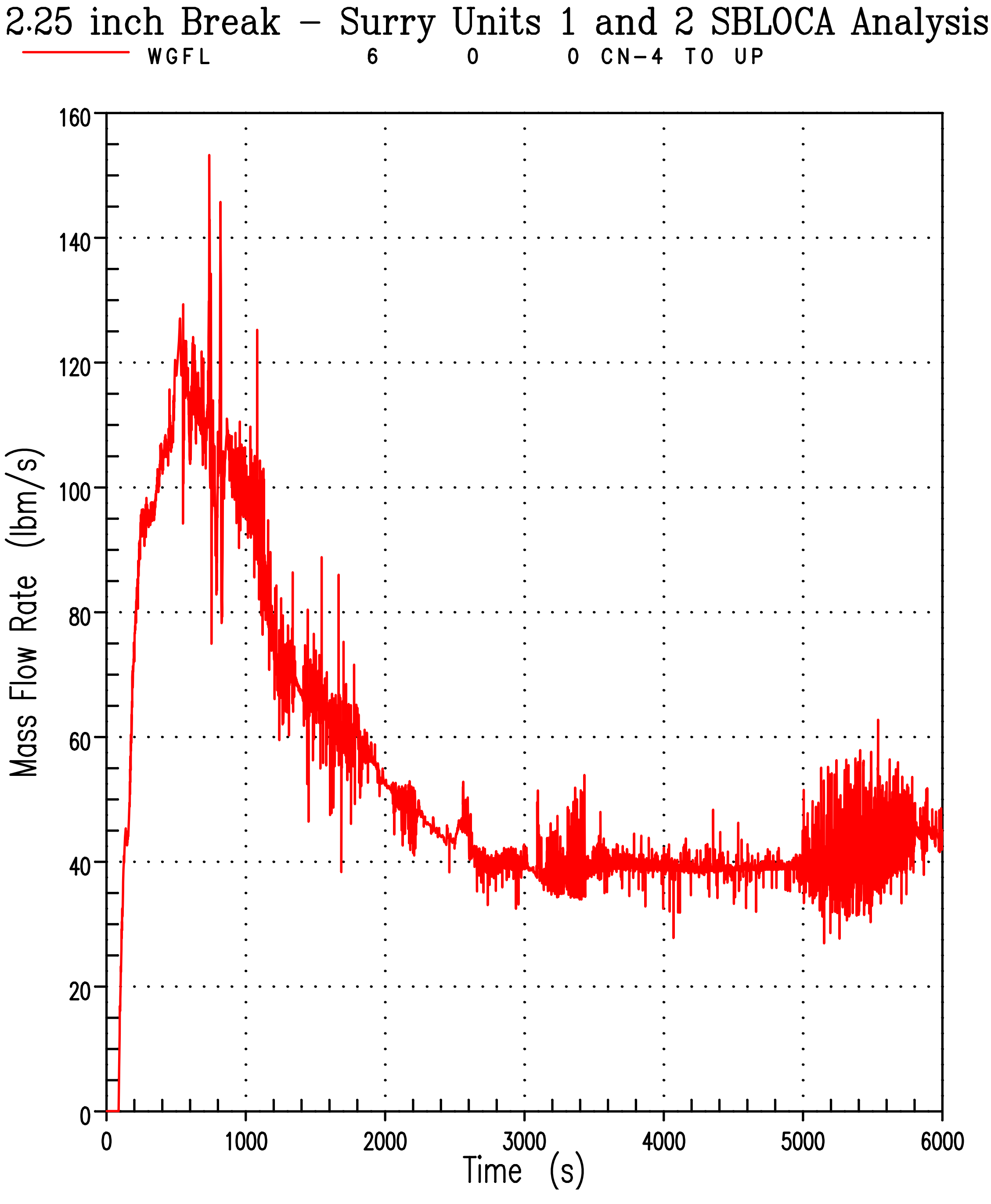


Figure 14.5-46
CORE EXIT VAPOR FLOW RATE, 2.5-INCH BREAK

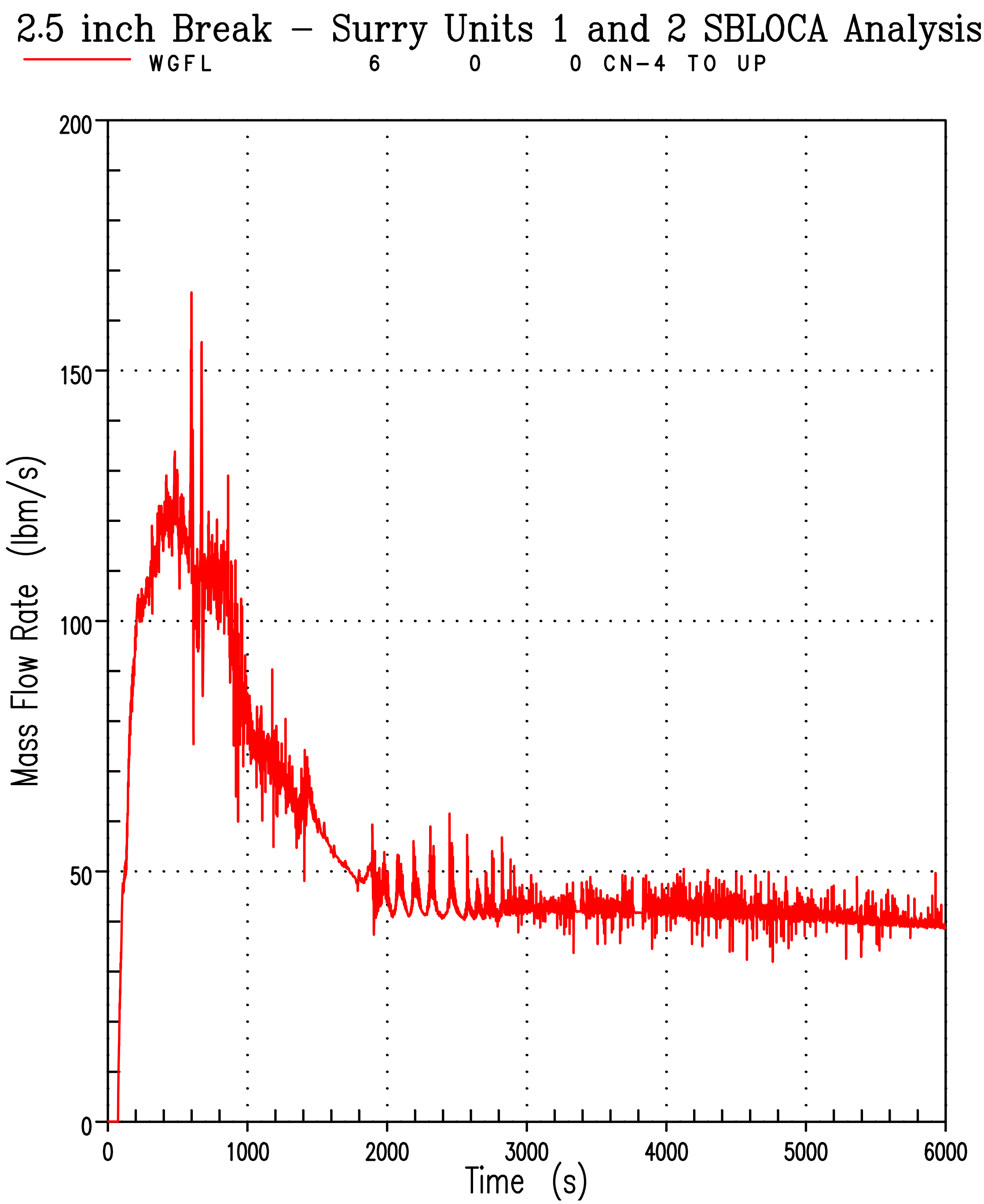


Figure 14.5-47
CORE EXIT VAPOR FLOW RATE, 2.75-INCH BREAK

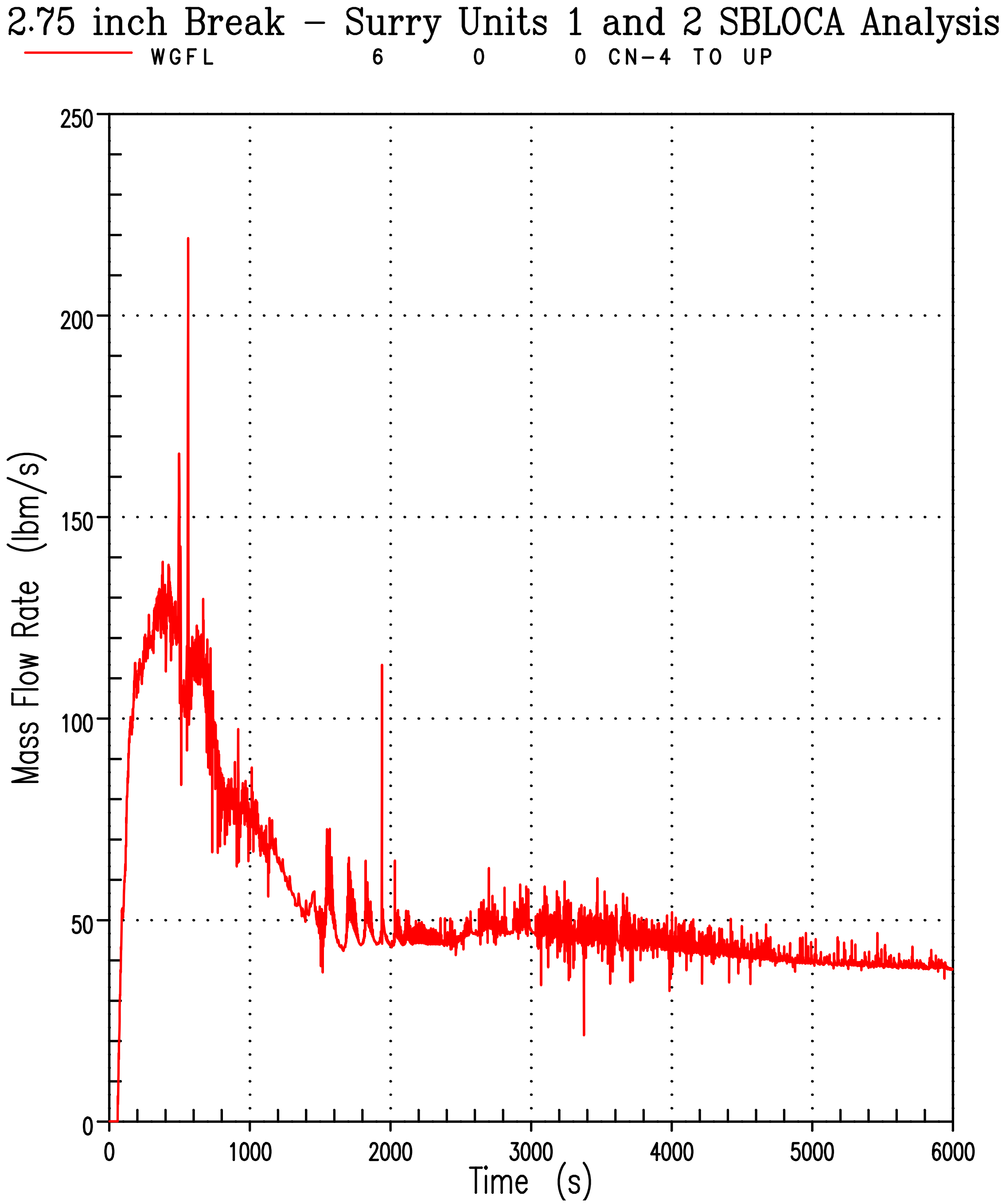


Figure 14.5-48
CORE EXIT VAPOR FLOW RATE, 3-INCH BREAK

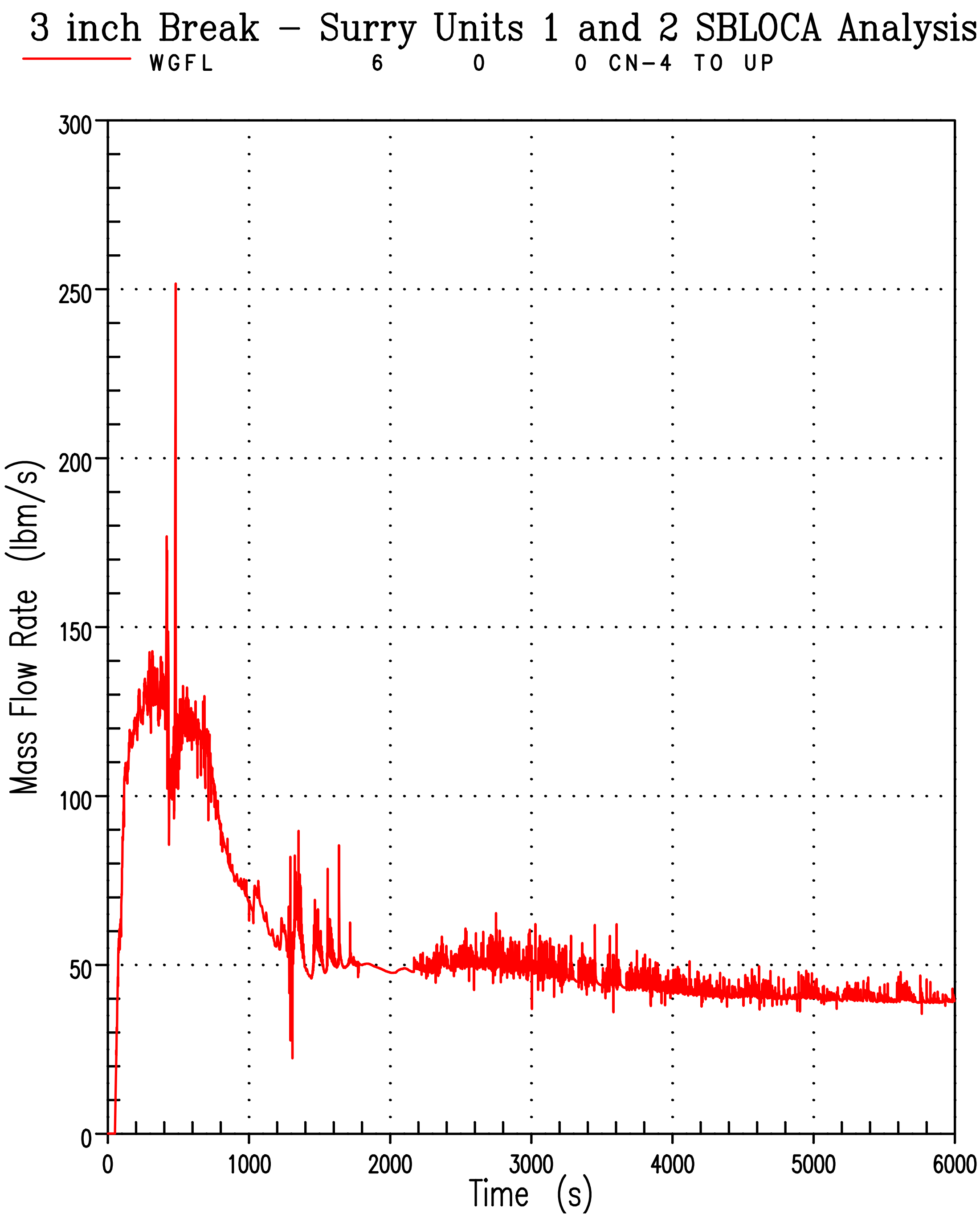


Figure 14.5-49
CORE EXIT VAPOR FLOW RATE, 4-INCH BREAK

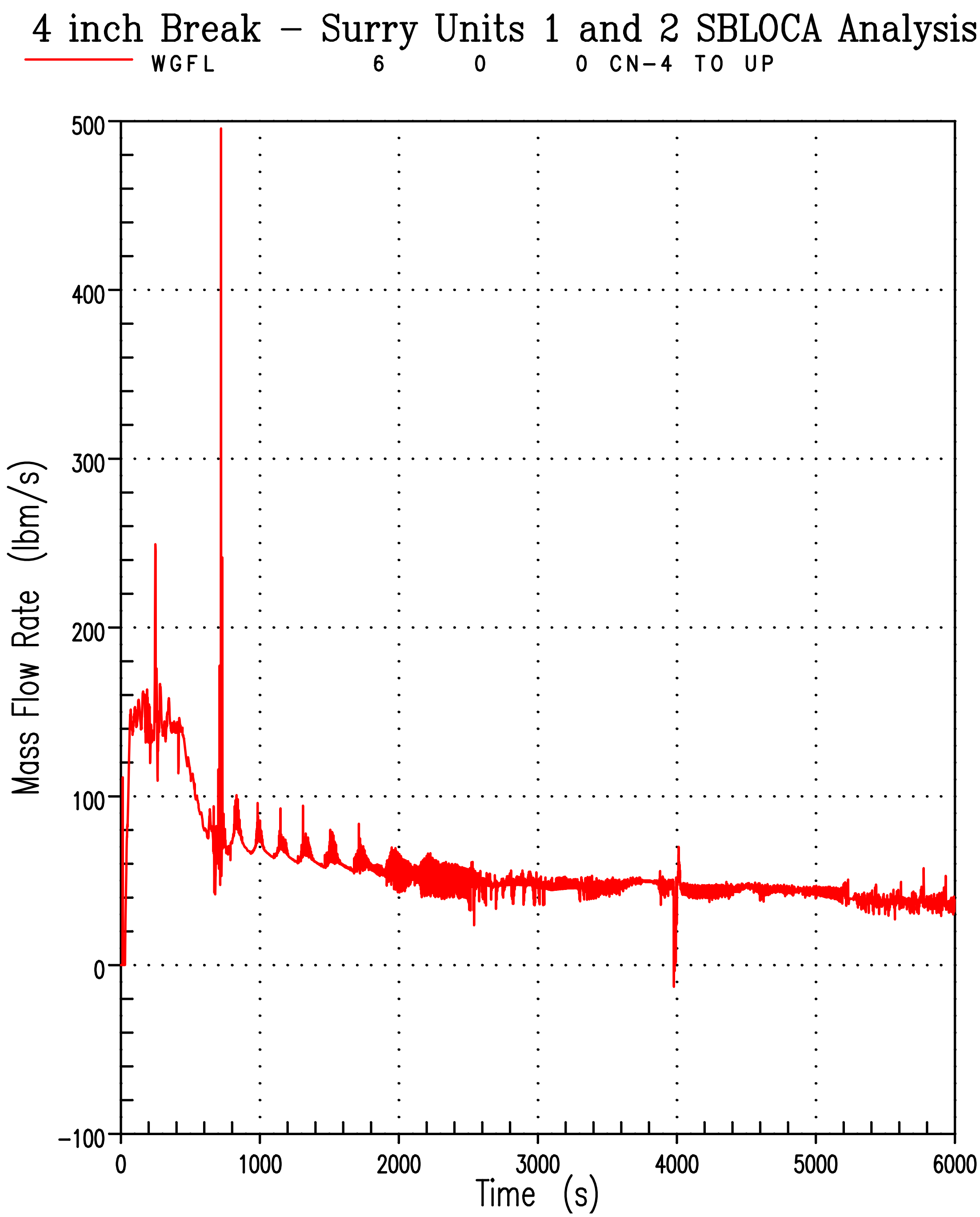


Figure 14.5-50
CORE EXIT VAPOR FLOW RATE, 5.5-INCH BREAK

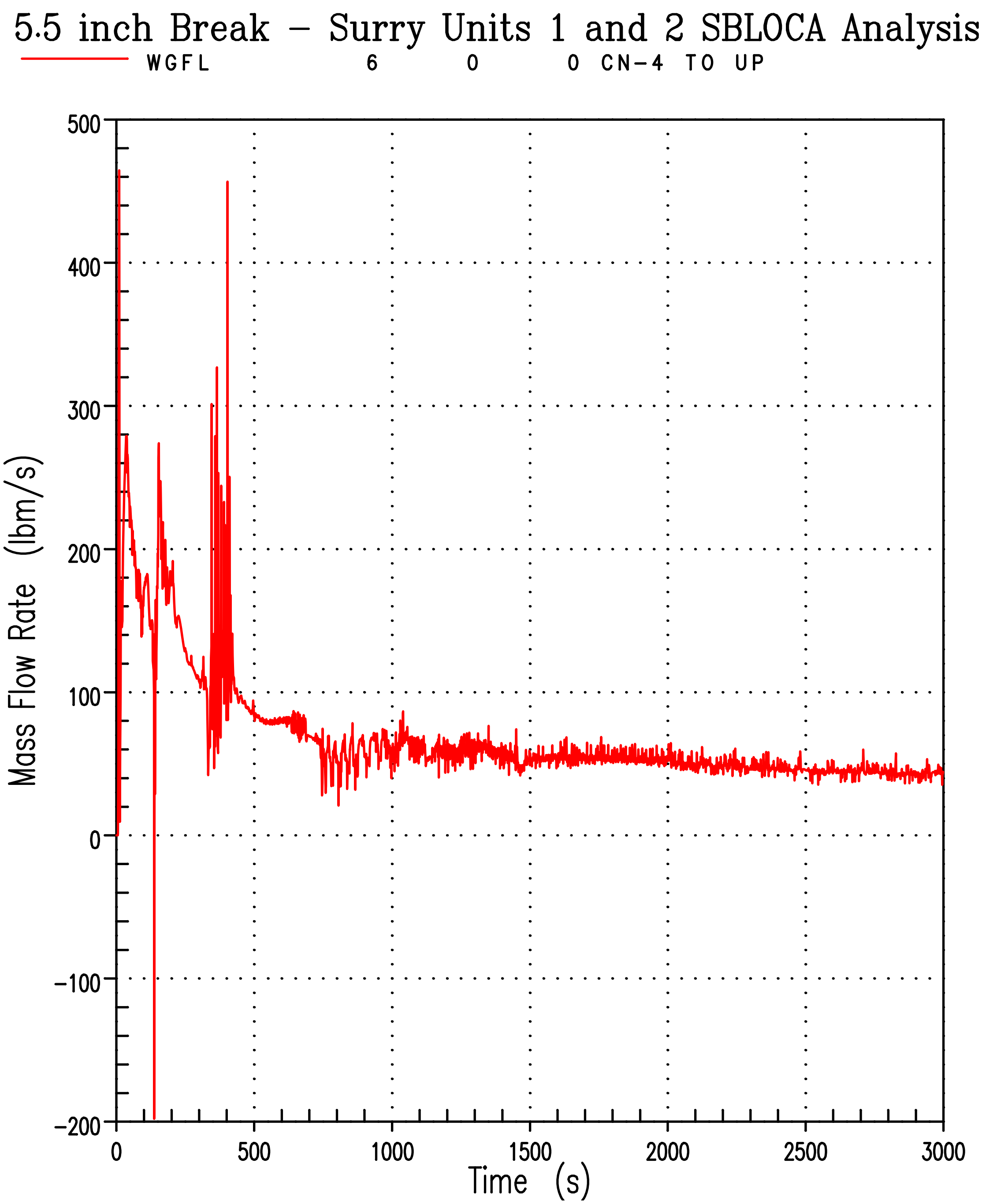


Figure 14.5-51
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 1.5-INCH BREAK

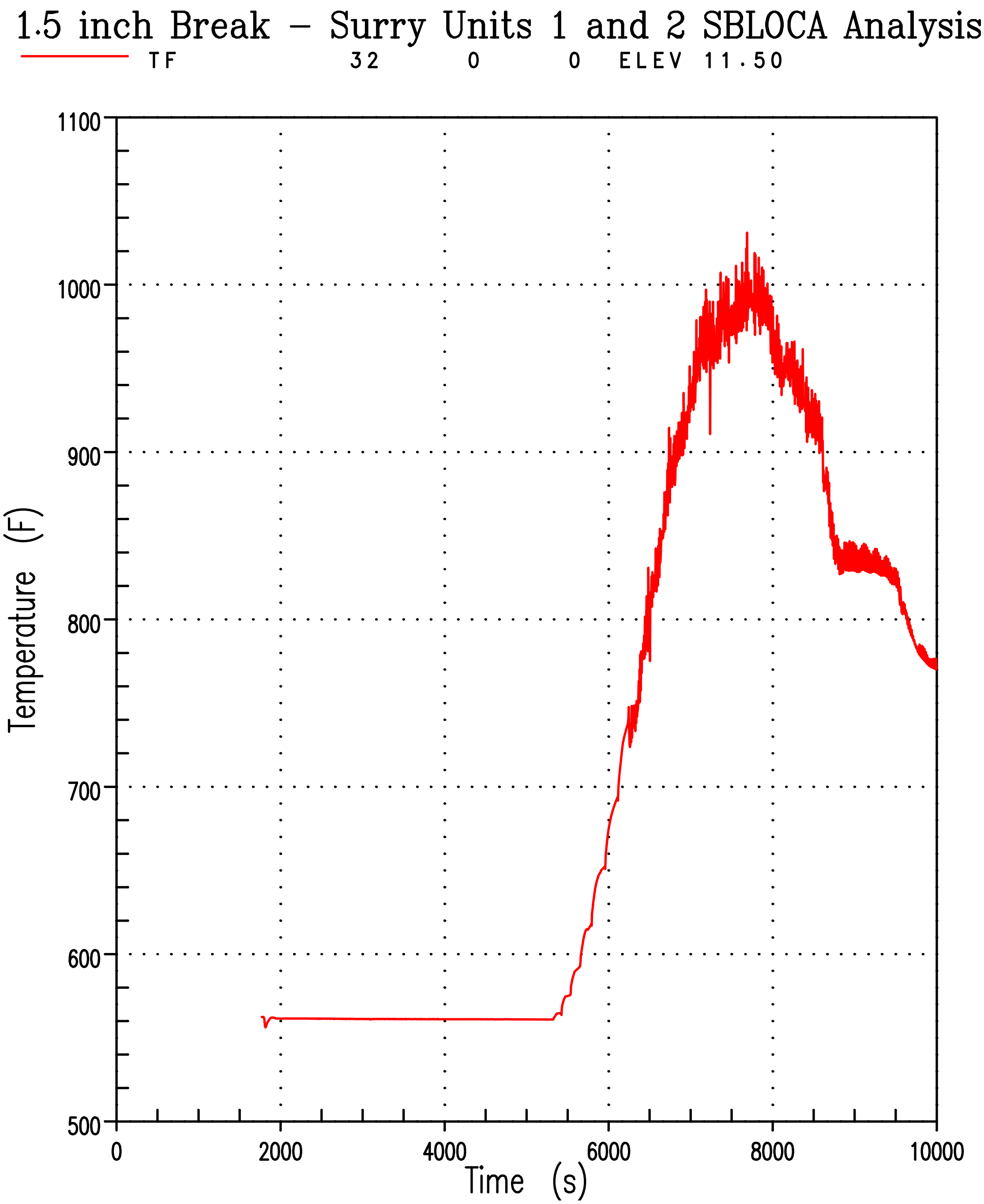
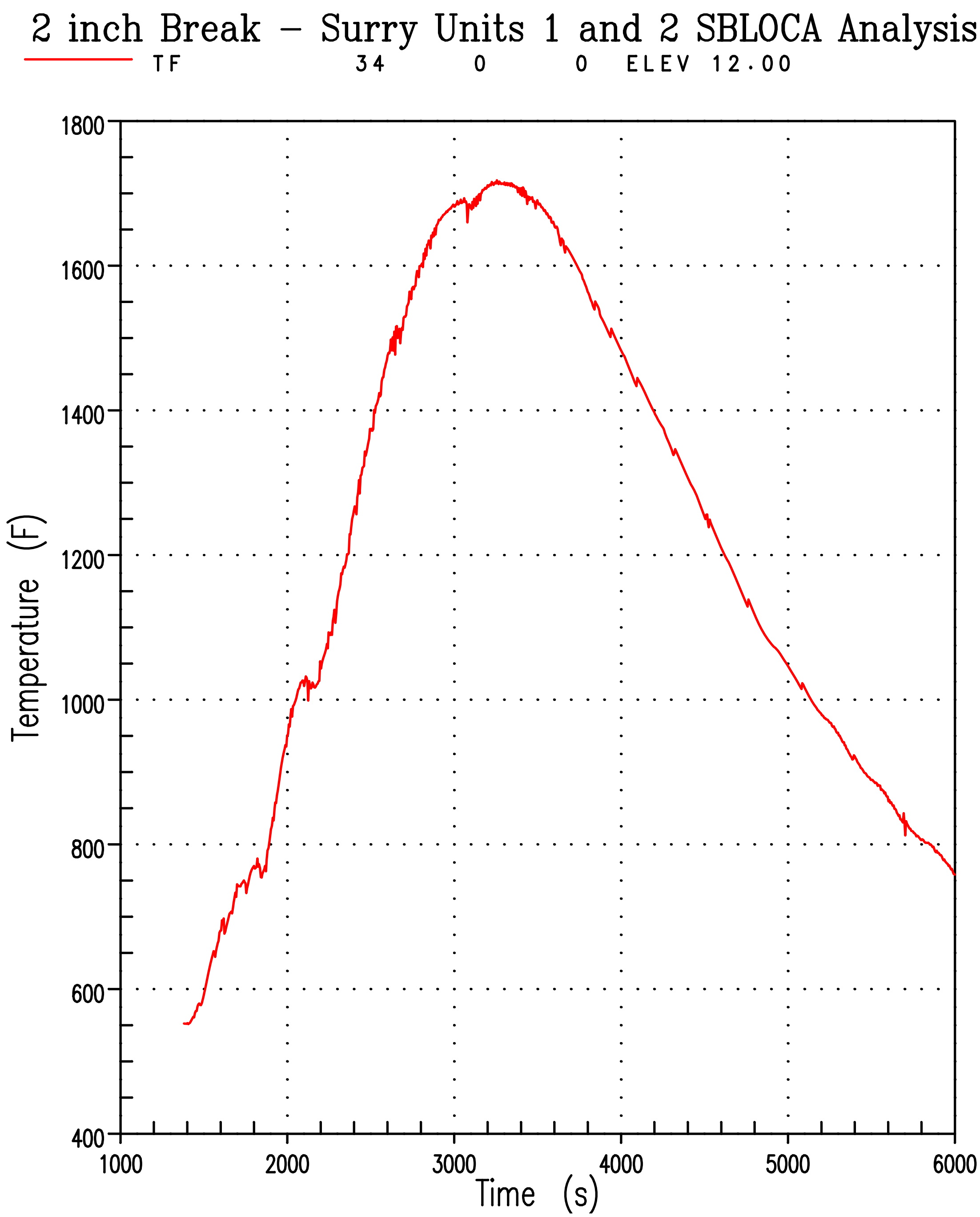


Figure 14.5-52
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 2-INCH BREAK



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HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 2.25-INCH BREAK

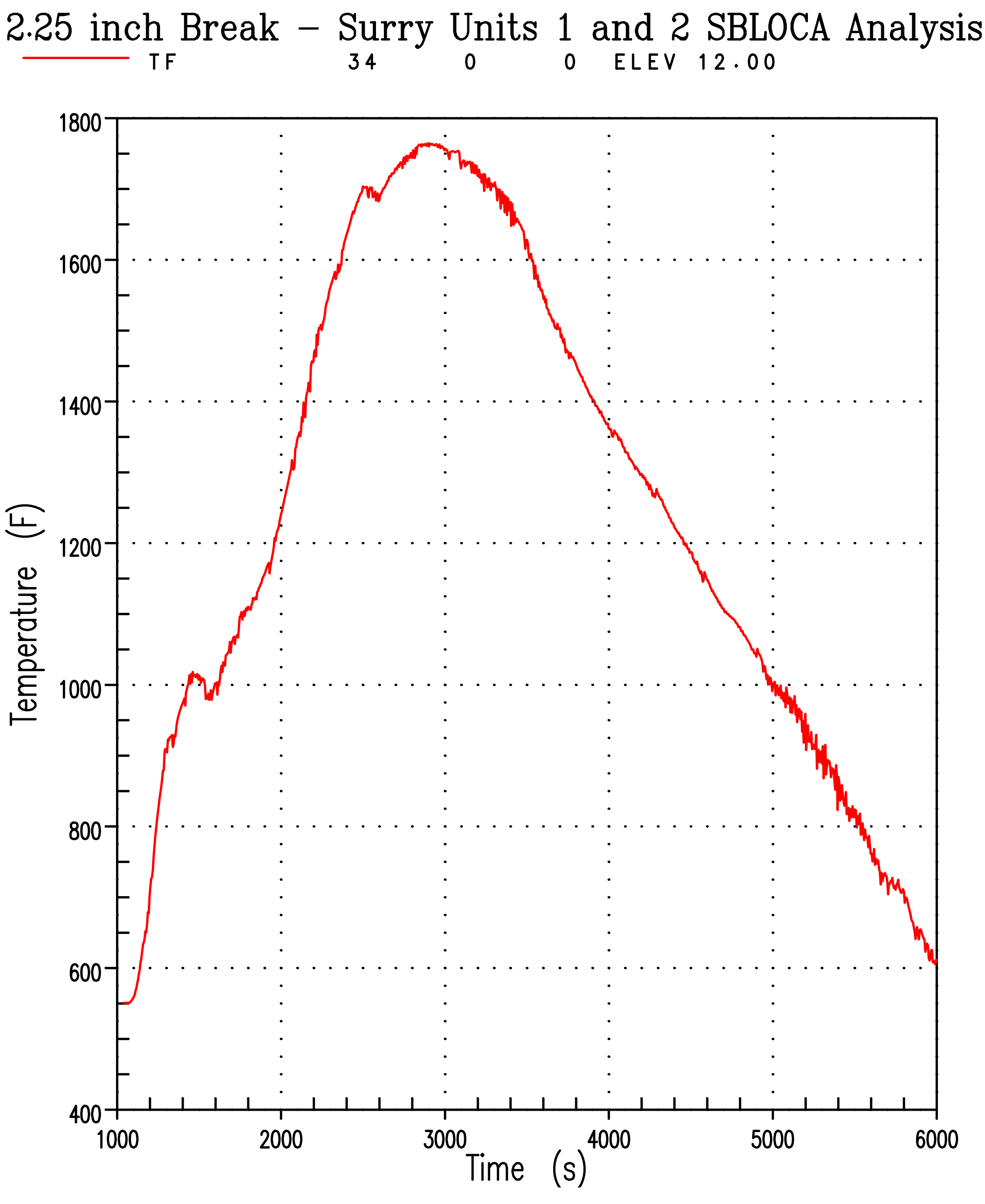


Figure 14.5-54
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 2.5-INCH BREAK

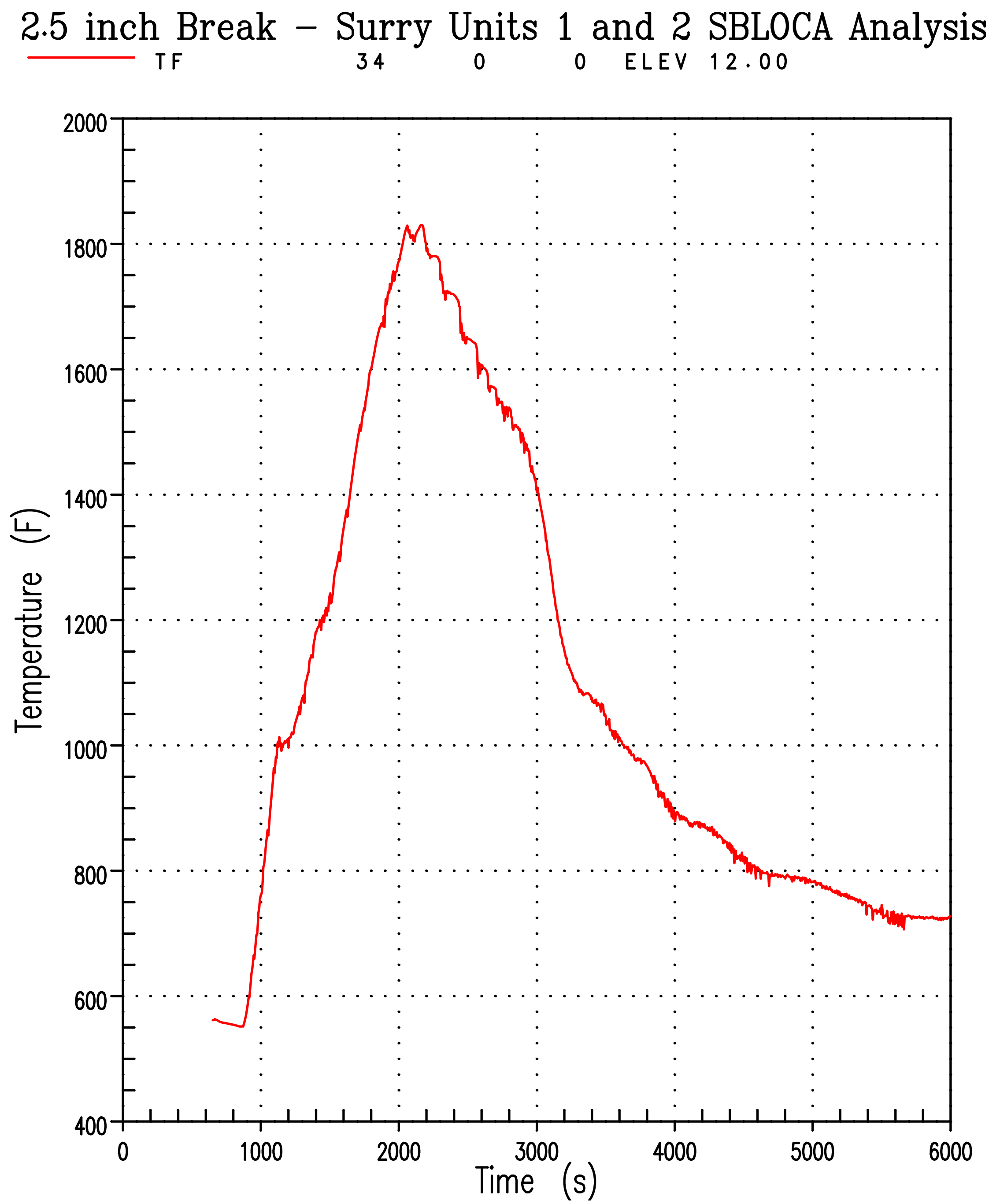


Figure 14.5-55
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION
(5,000 MWD/MTU BURNUP), 2.75-INCH BREAK

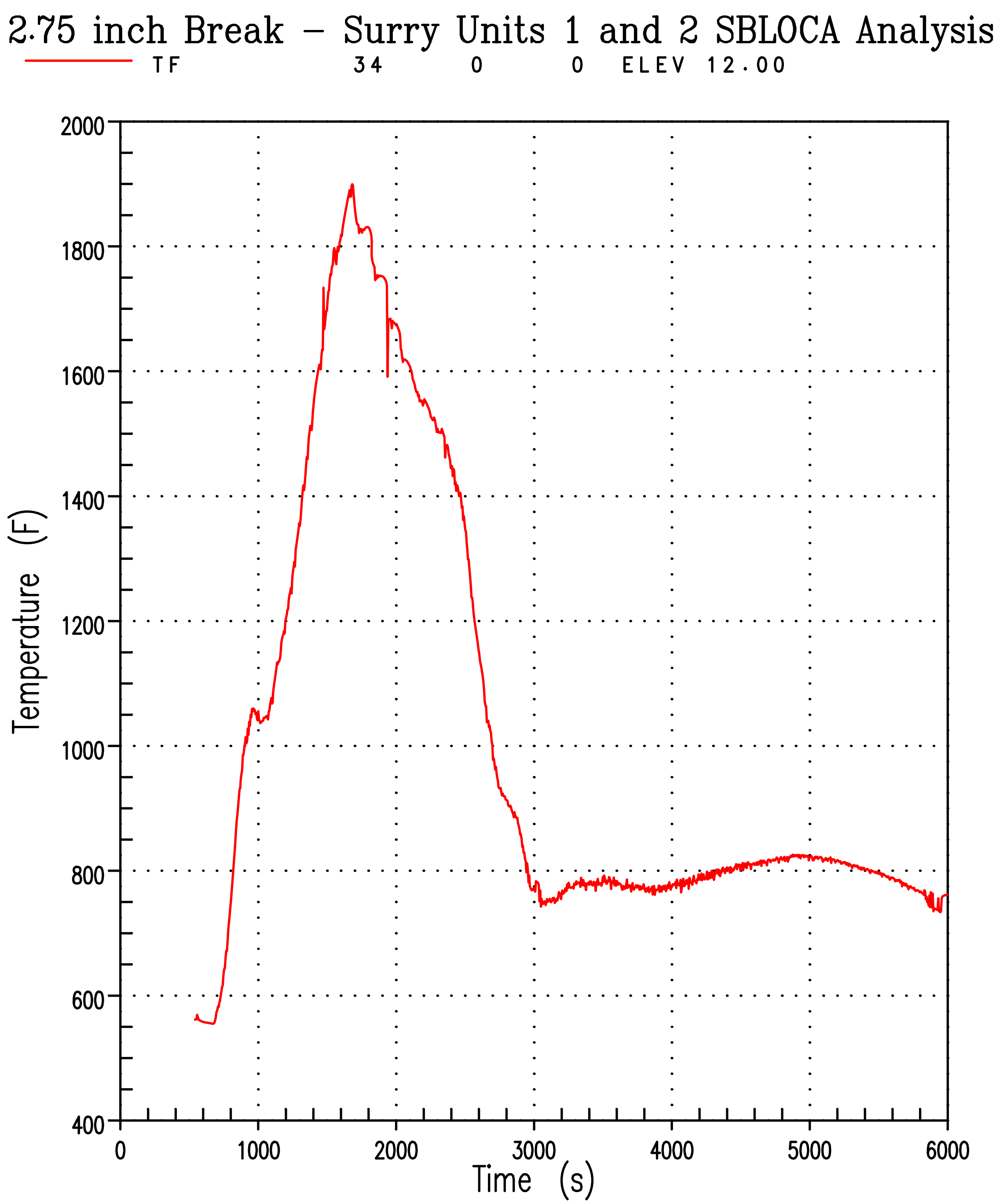


Figure 14.5-56
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 3-INCH BREAK

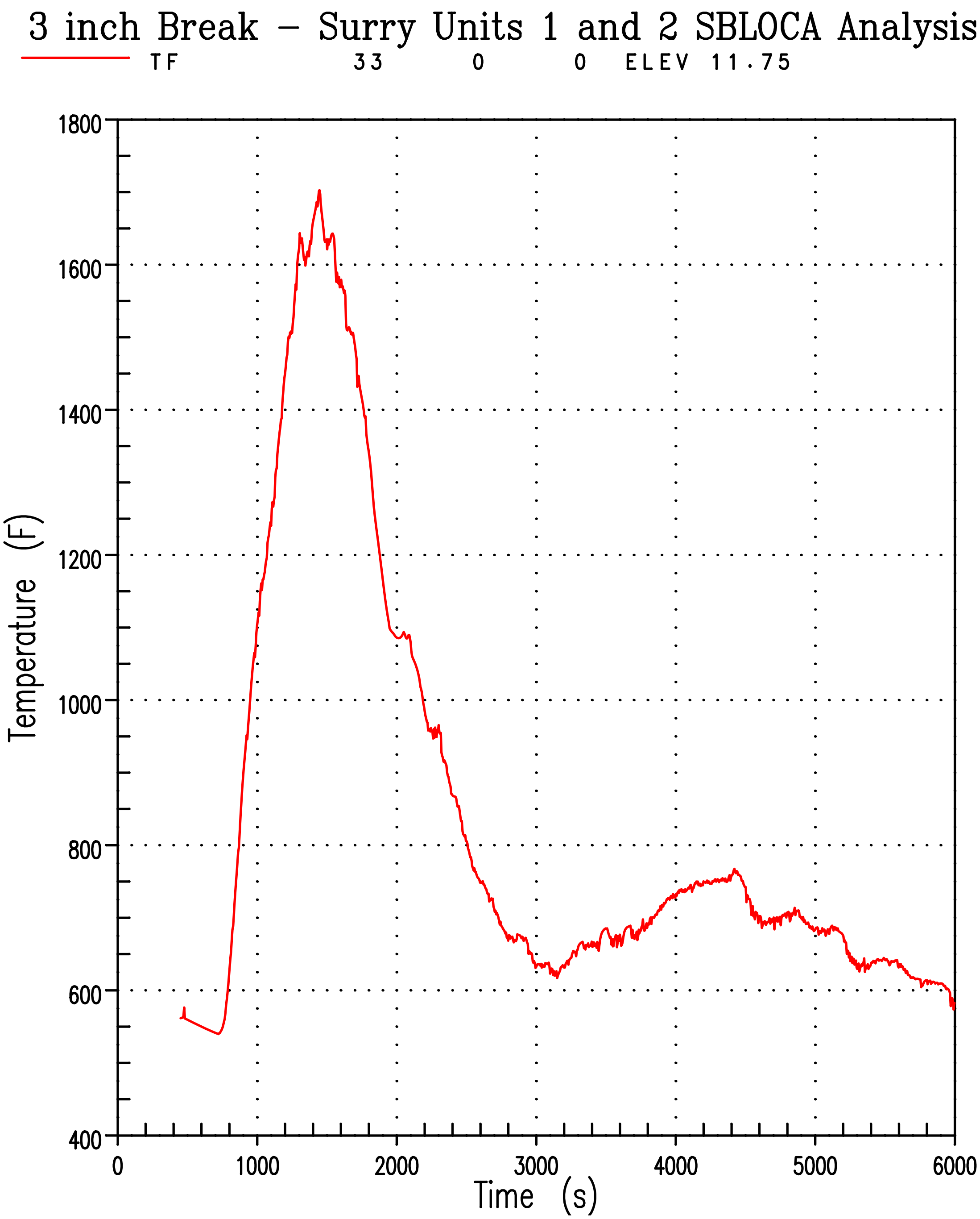


Figure 14.5-57
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 4-INCH BREAK

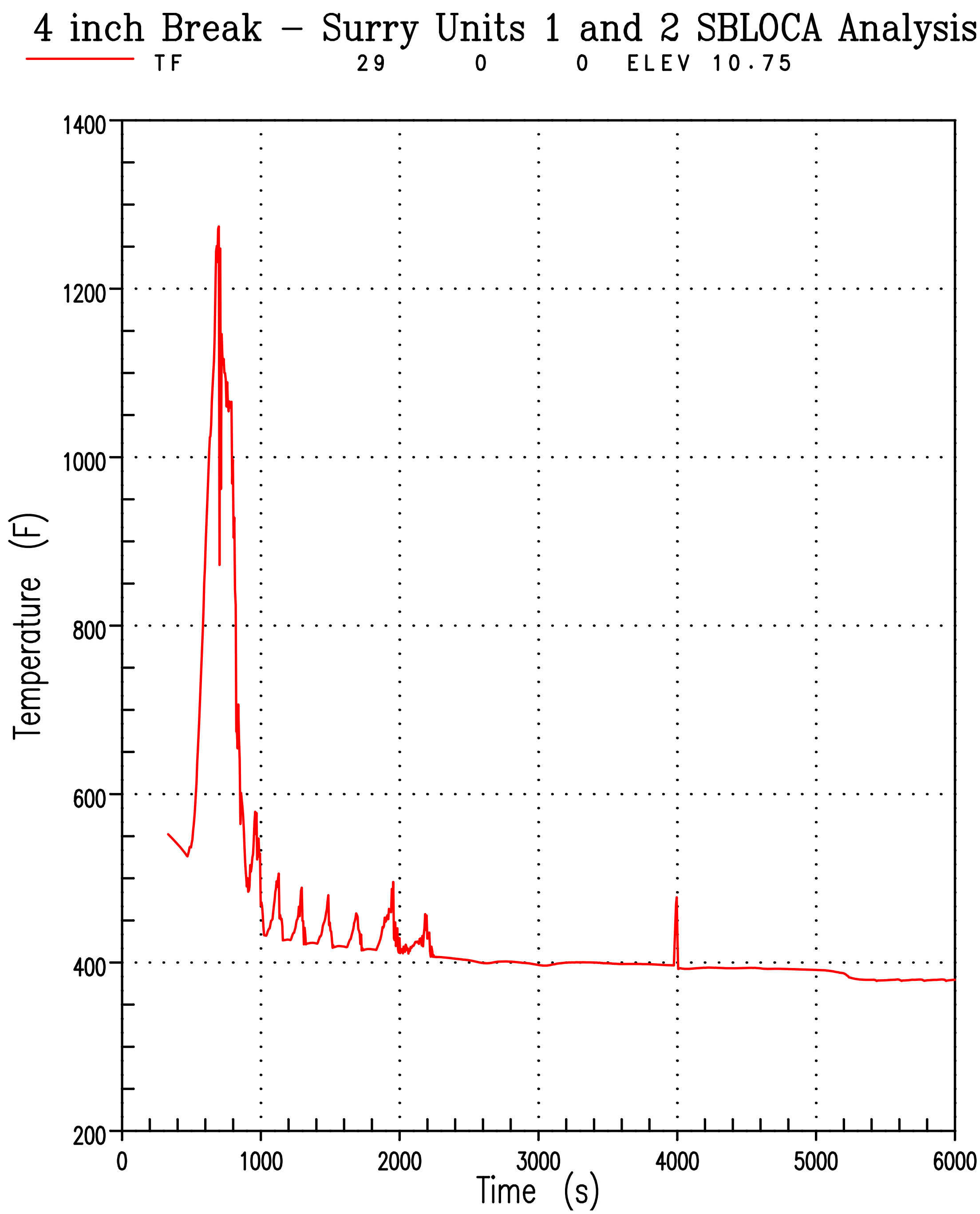


Figure 14.5-58
HOT ASSEMBLY FLUID TEMPERATURE AT PCT ELEVATION, 5.5-INCH BREAK

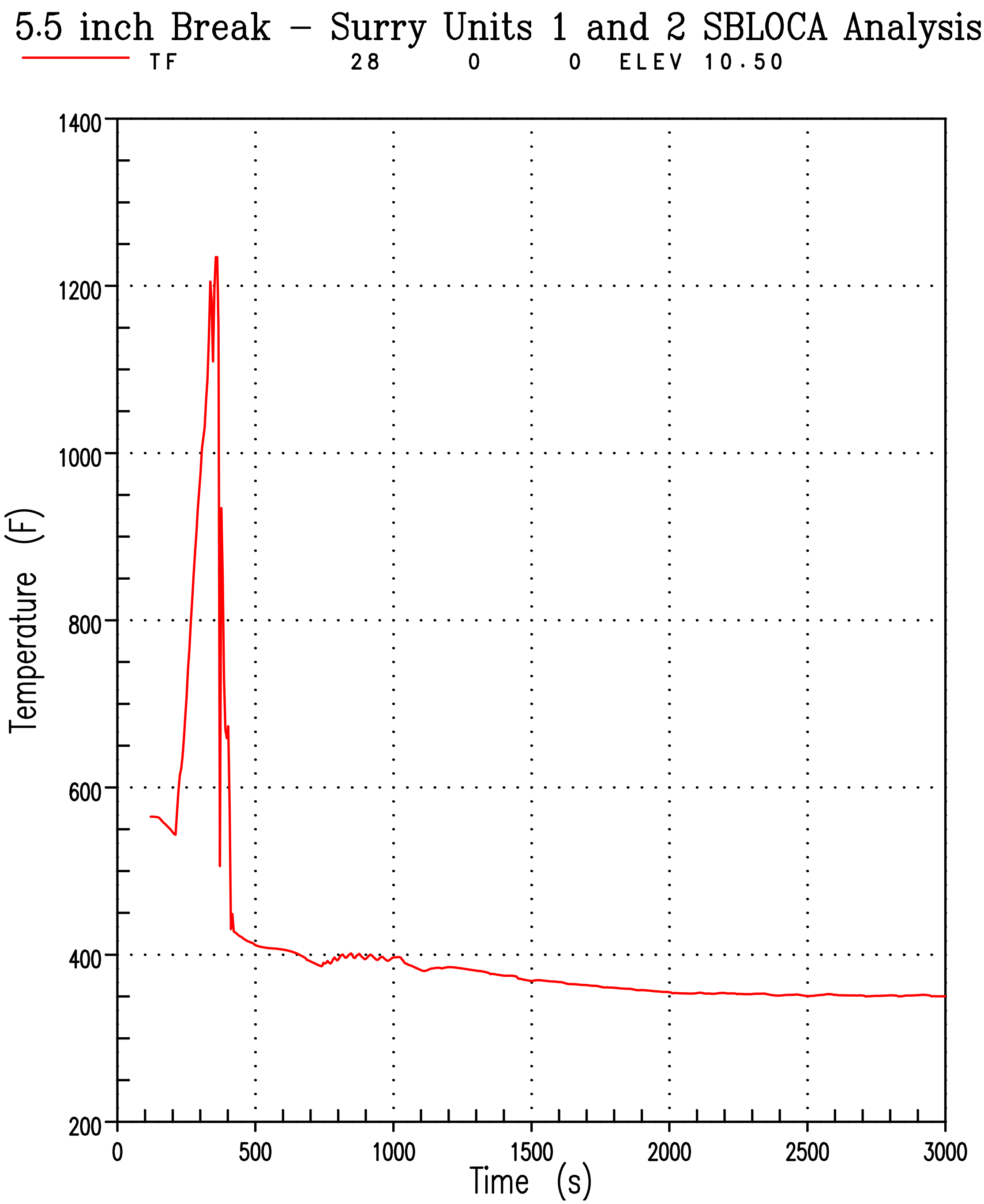


Figure 14.5-59
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 1.5-INCH BREAK

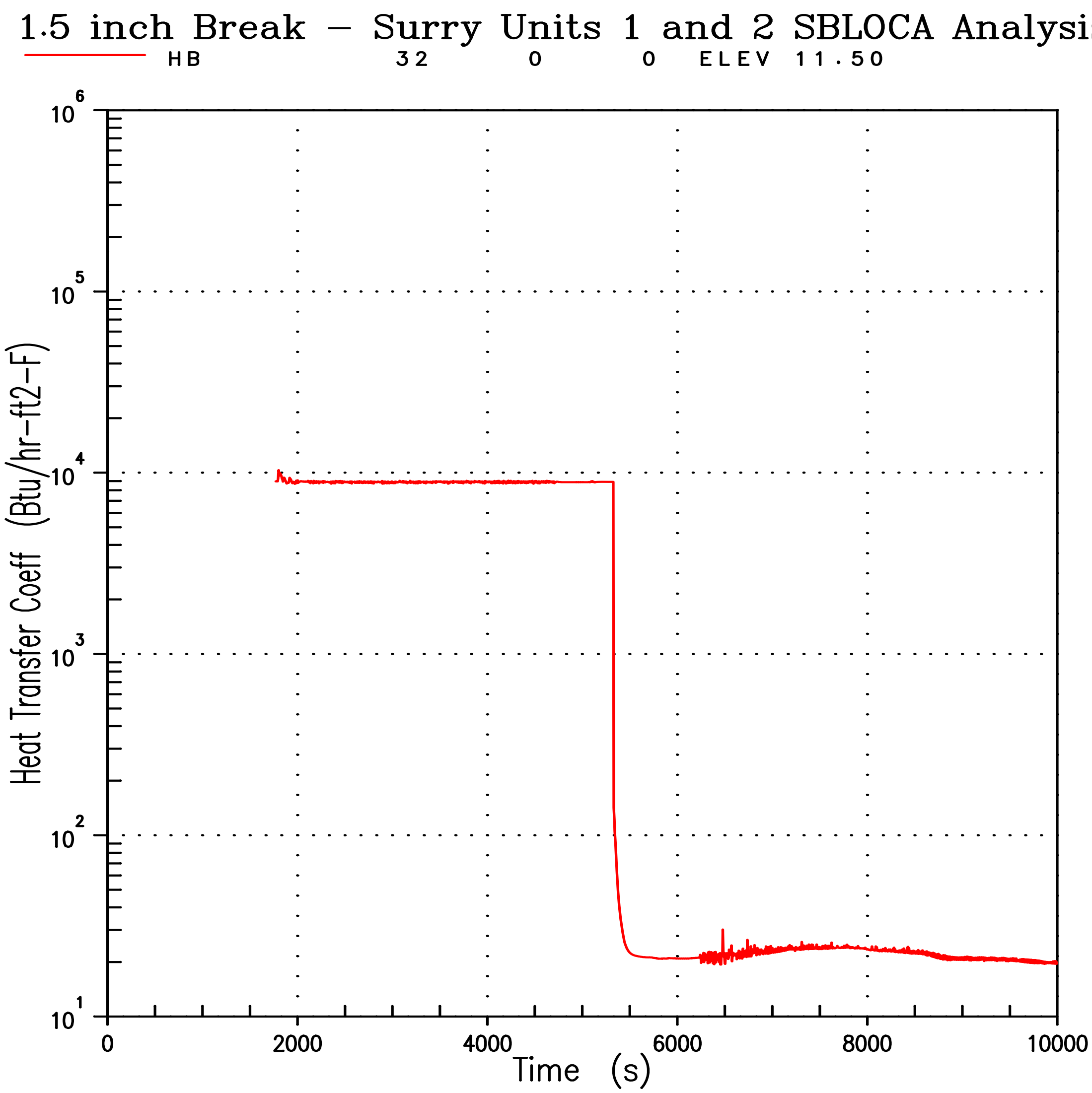


Figure 14.5-60
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 2-INCH BREAK

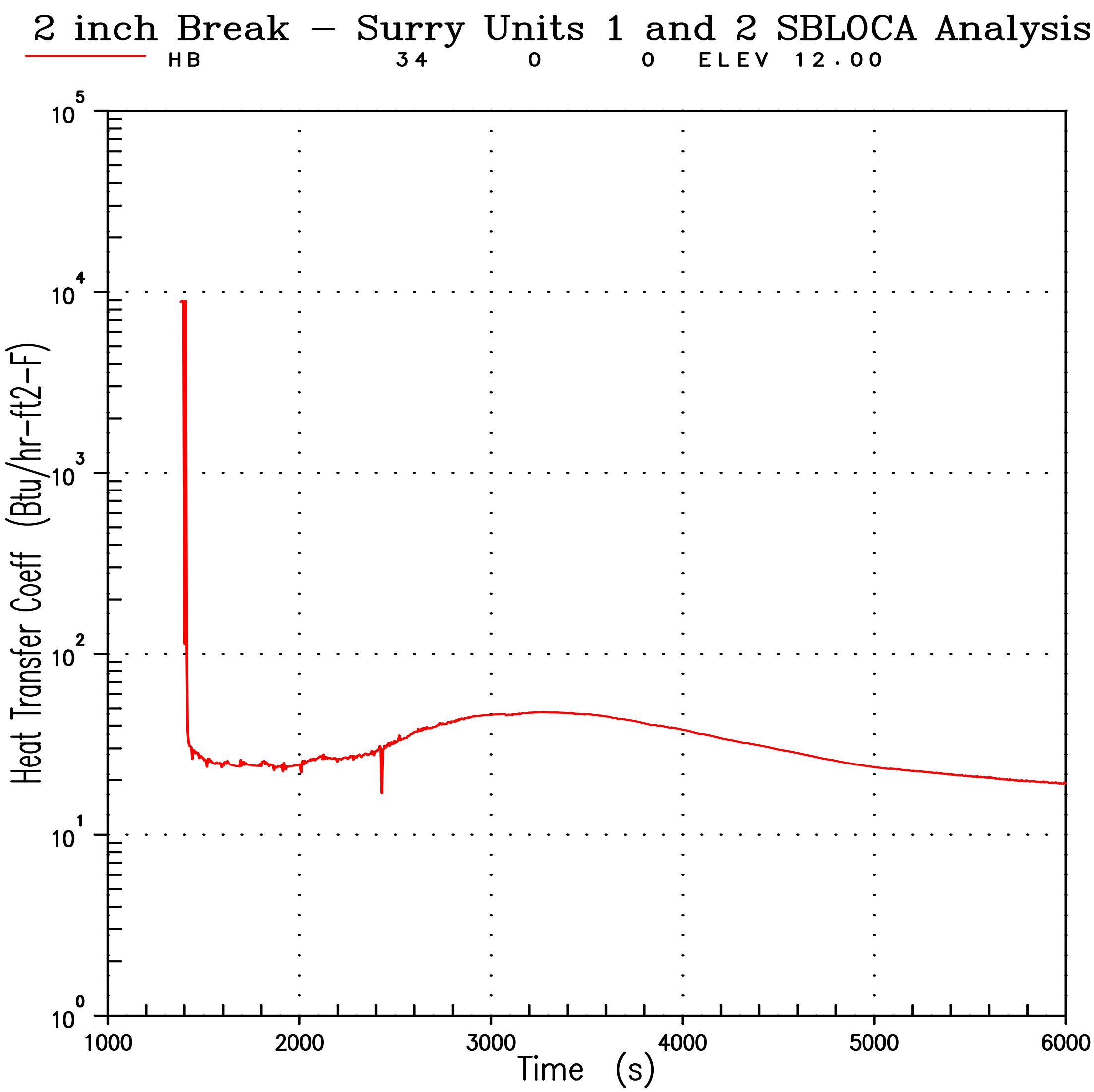


Figure 14.5-61
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 2.25-INCH BREAK

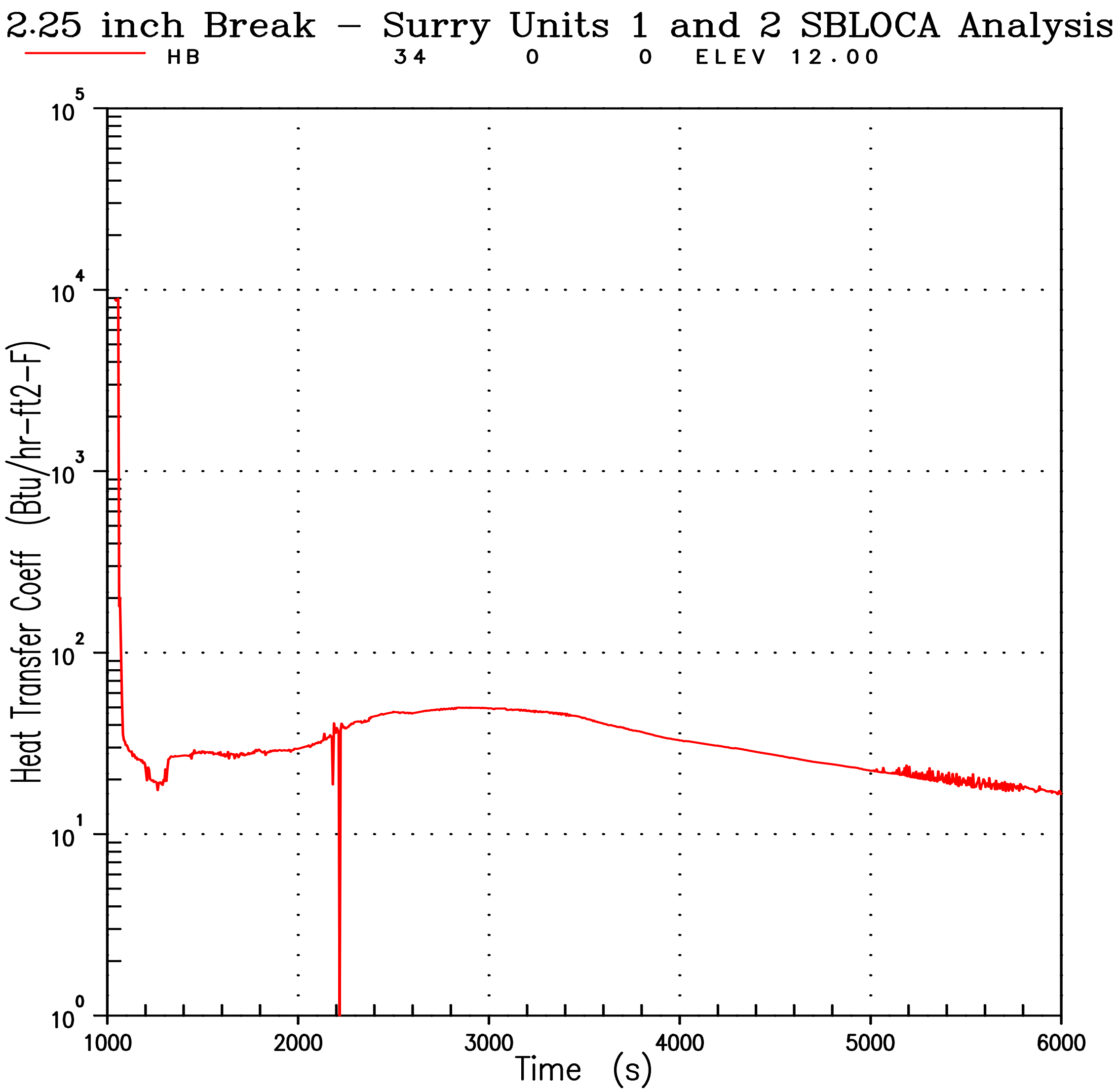


Figure 14.5-62
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 2.5-INCH BREAK

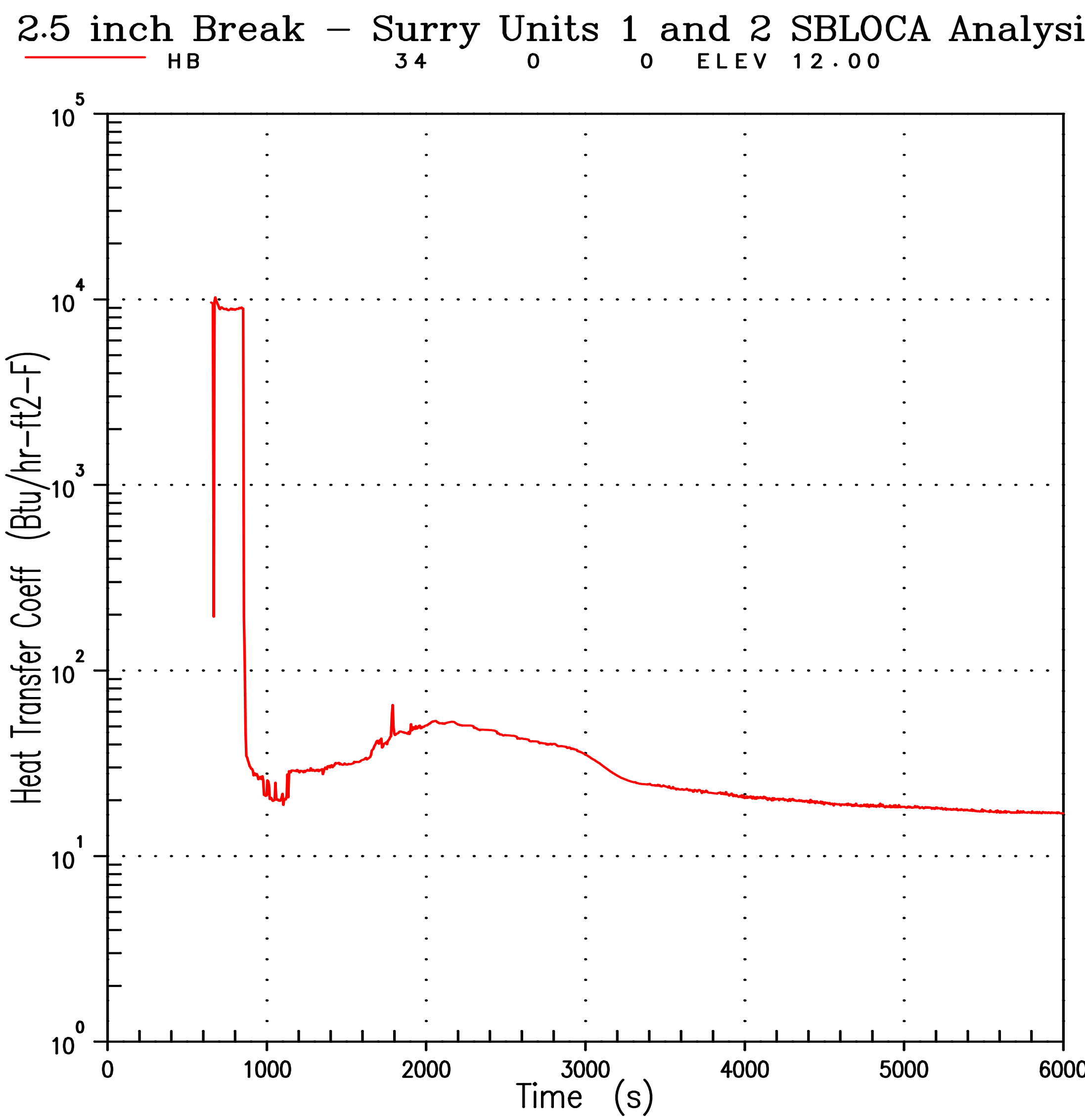


Figure 14.5-63
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION
(5,000 MWD/MTU BURNUP), 2.75-INCH BREAK

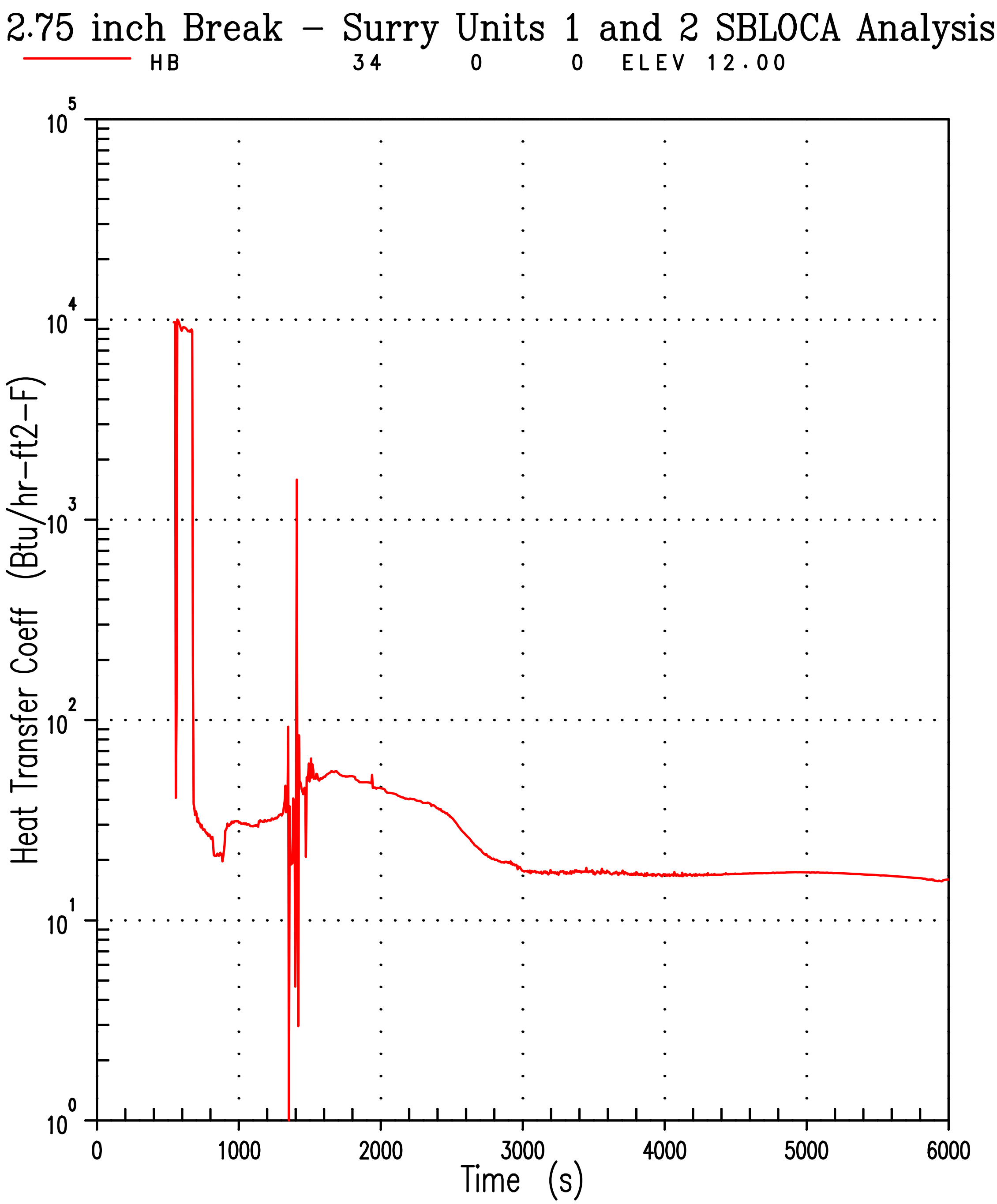


Figure 14.5-64
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 3-INCH BREAK

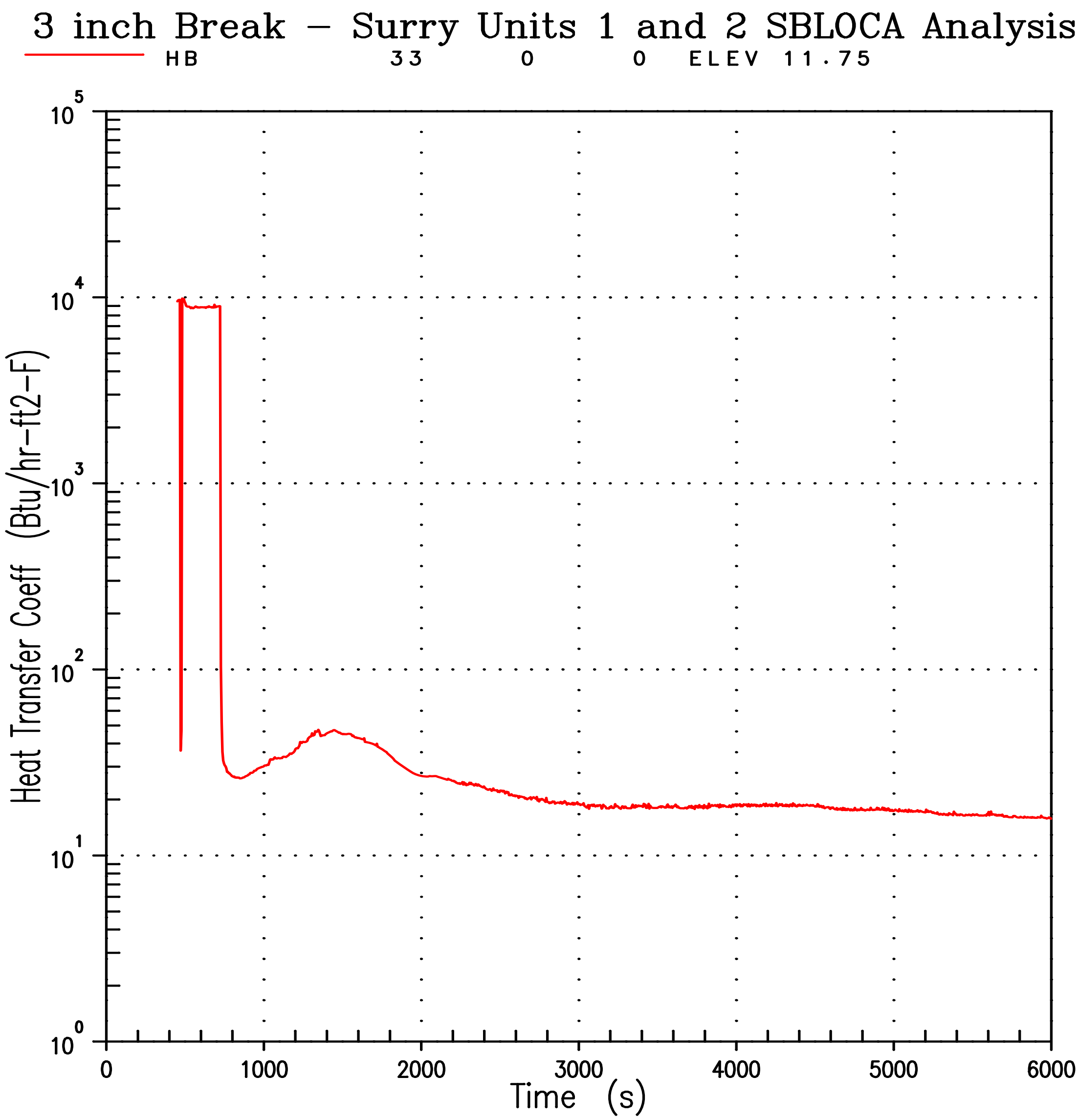


Figure 14.5-65
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 4-INCH BREAK

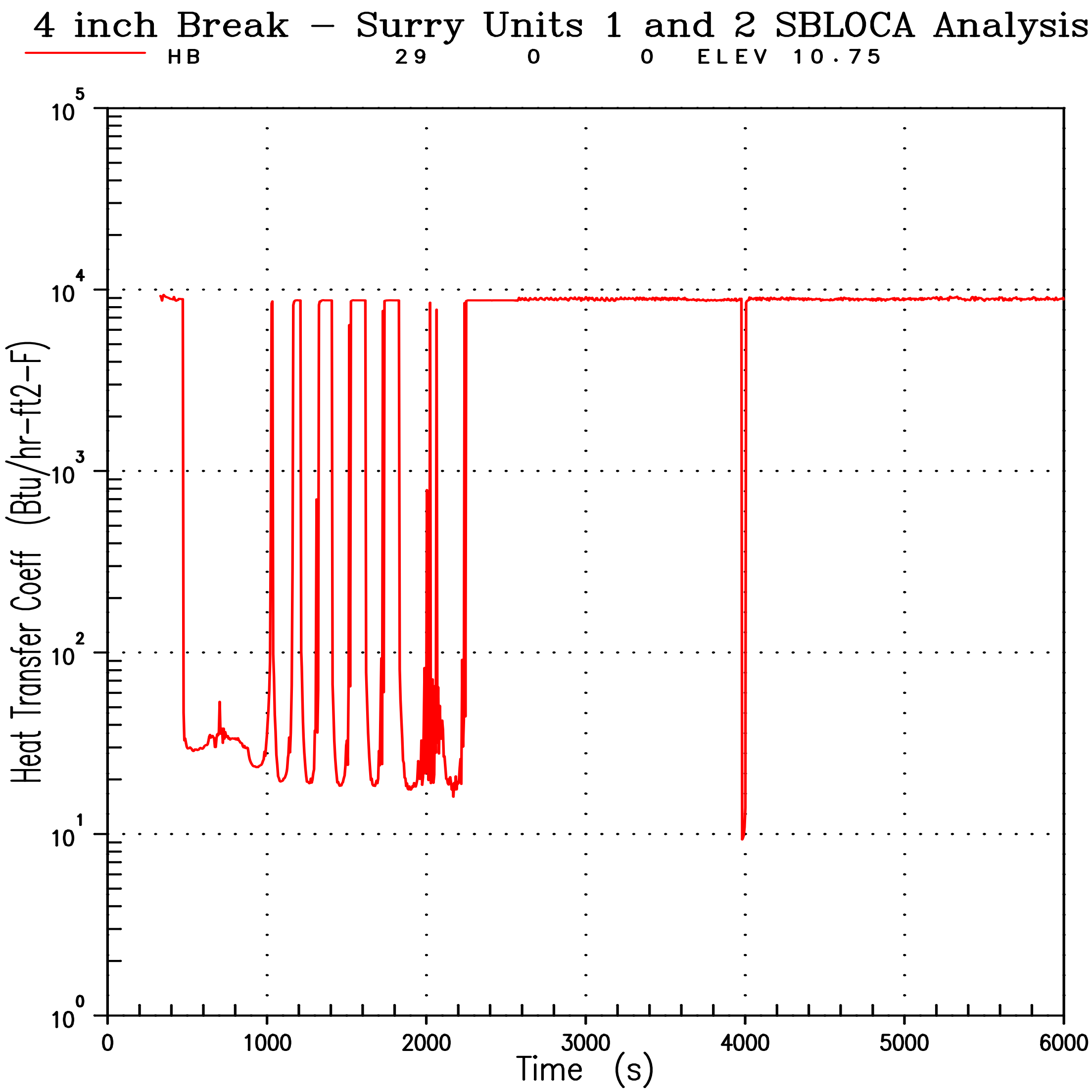


Figure 14.5-66
HOT ROD HEAT TRANSFER COEFFICIENT AT PCT ELEVATION, 5.5-INCH BREAK

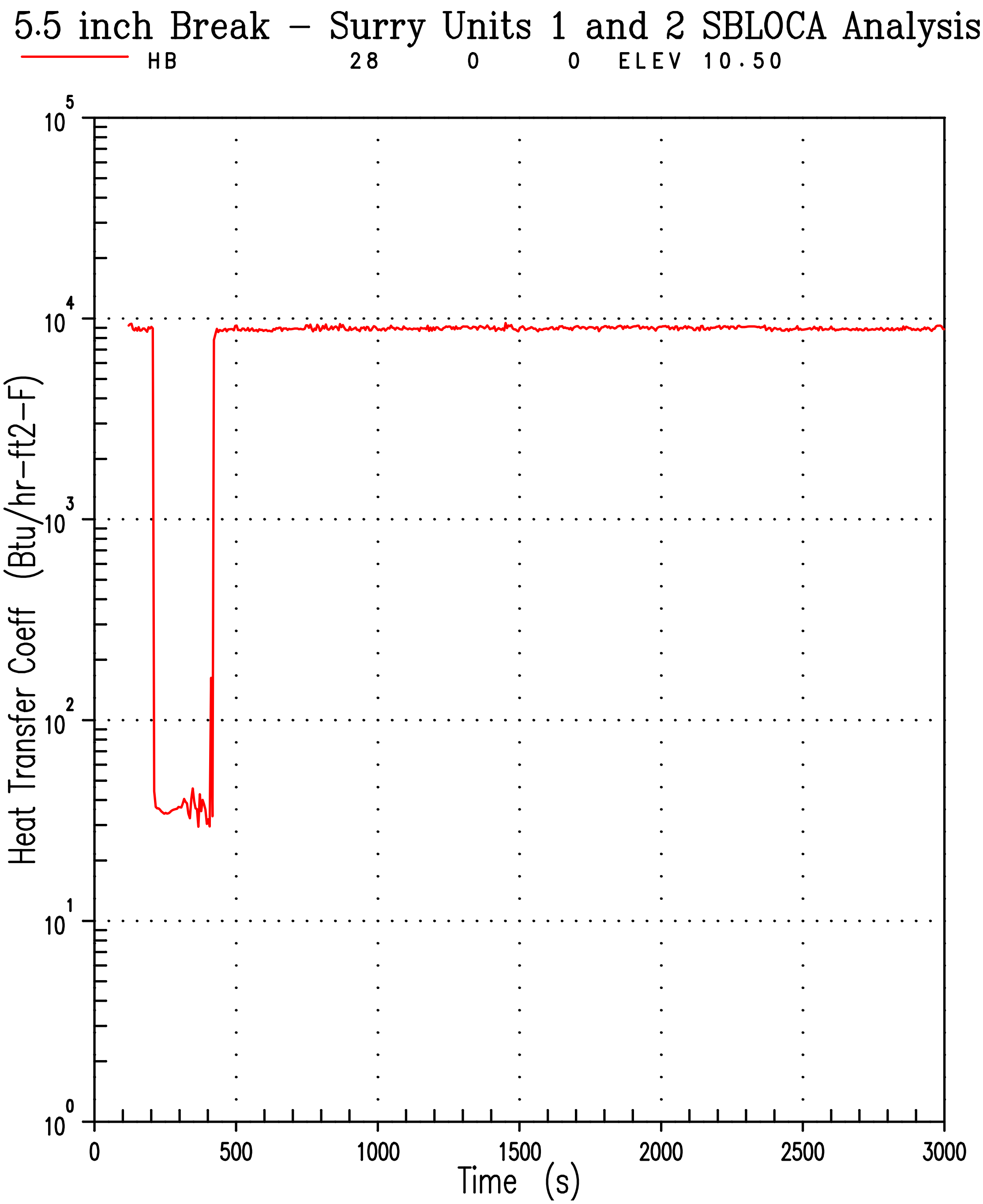


Figure 14.5-67
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION, 1.5-INCH BREAK

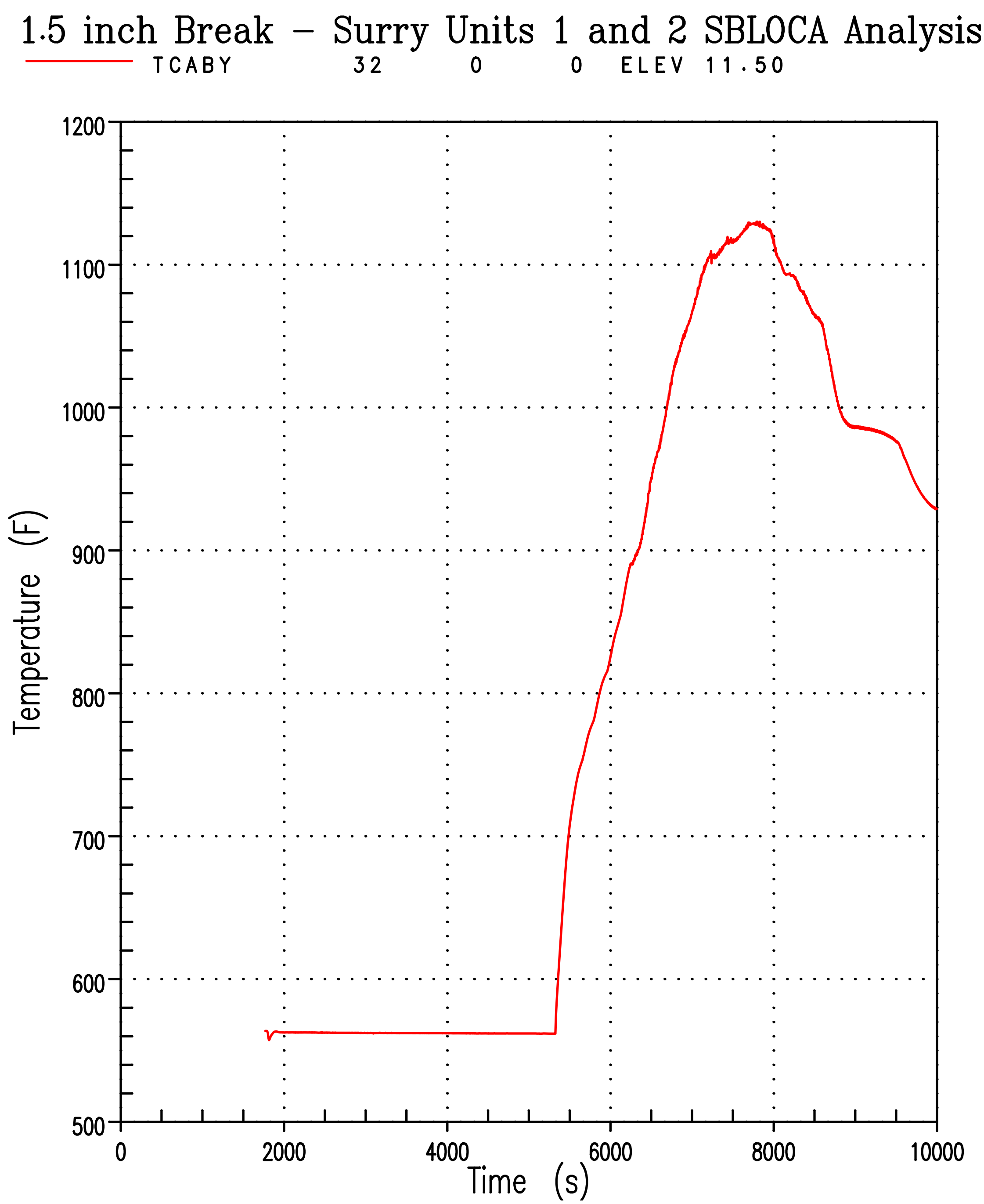


Figure 14.5-68
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION, 2-INCH BREAK

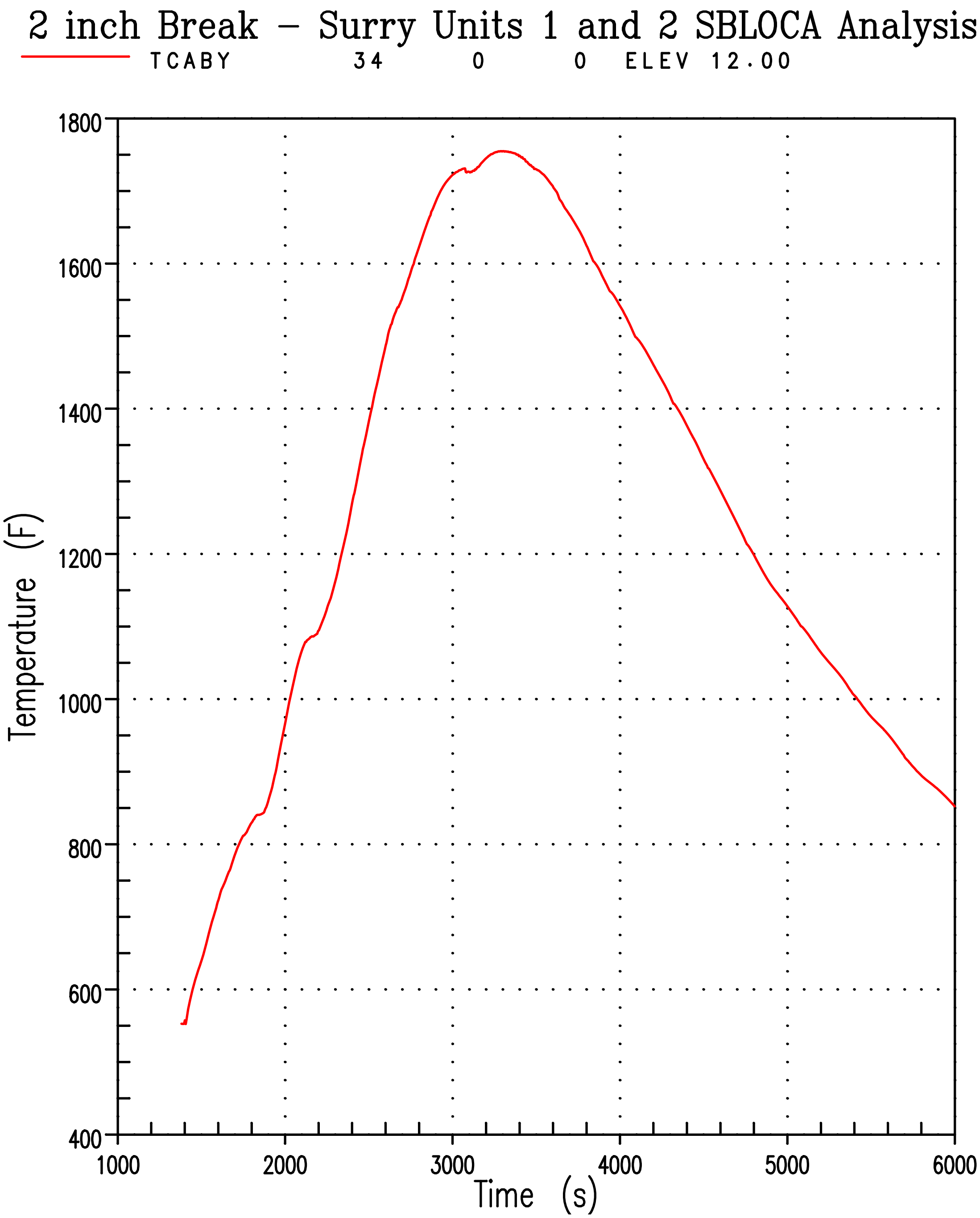


Figure 14.5-69
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION,
2.25-INCH BREAK

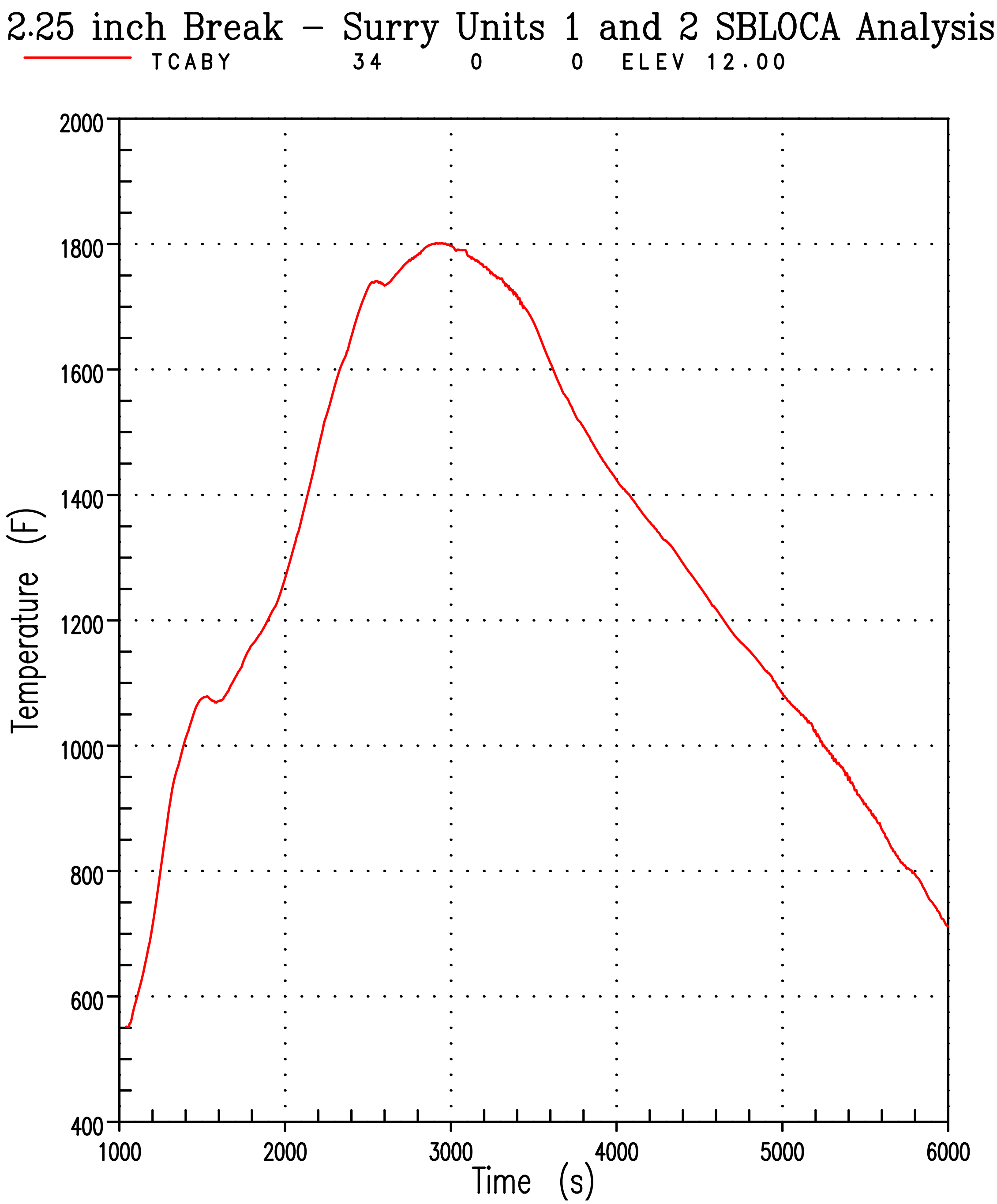


Figure 14.5-70
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION, 2.5-INCH BREAK

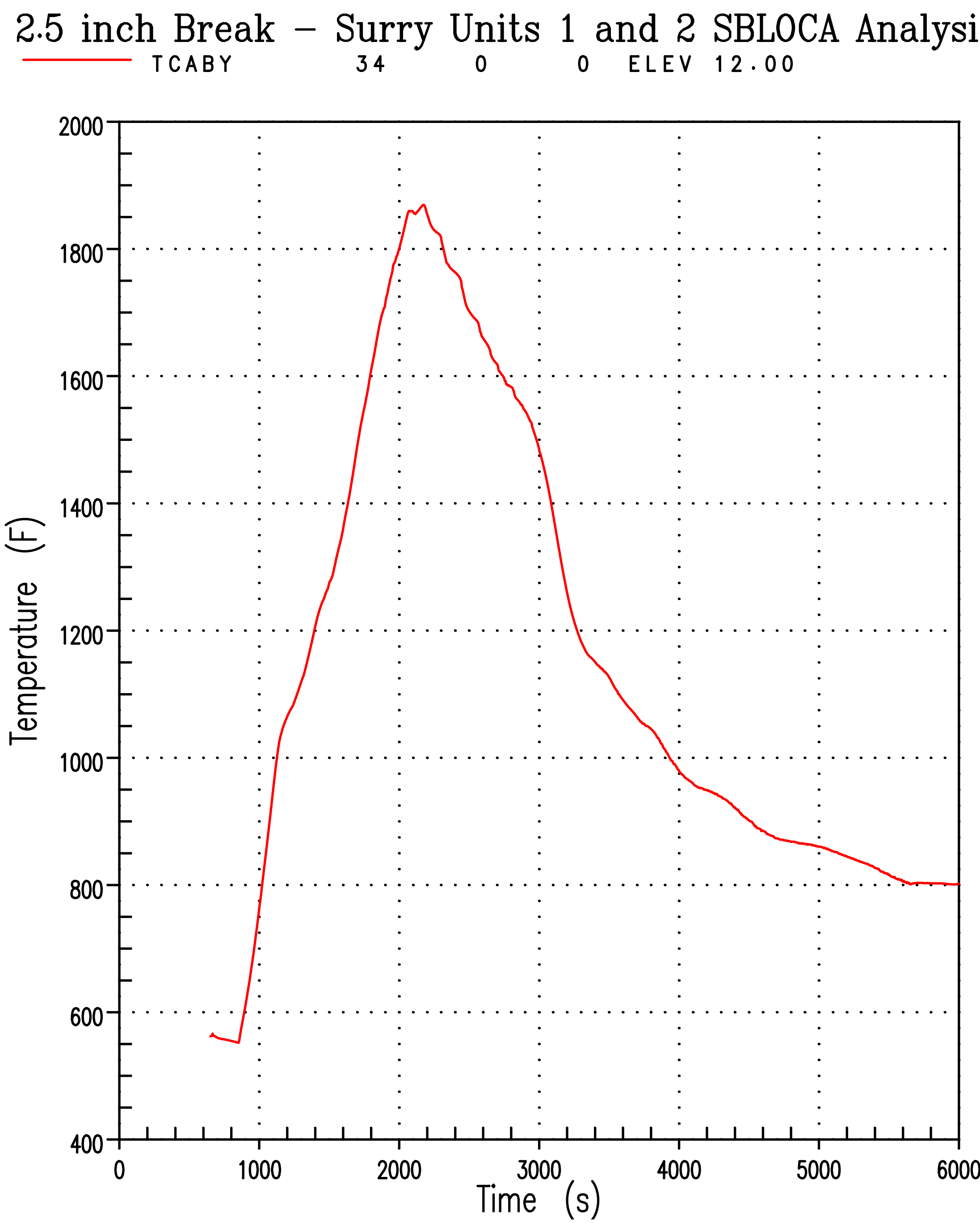


Figure 14.5-71
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION
(5,000 MWD/MTU BURNUP), 2.75-INCH BREAK

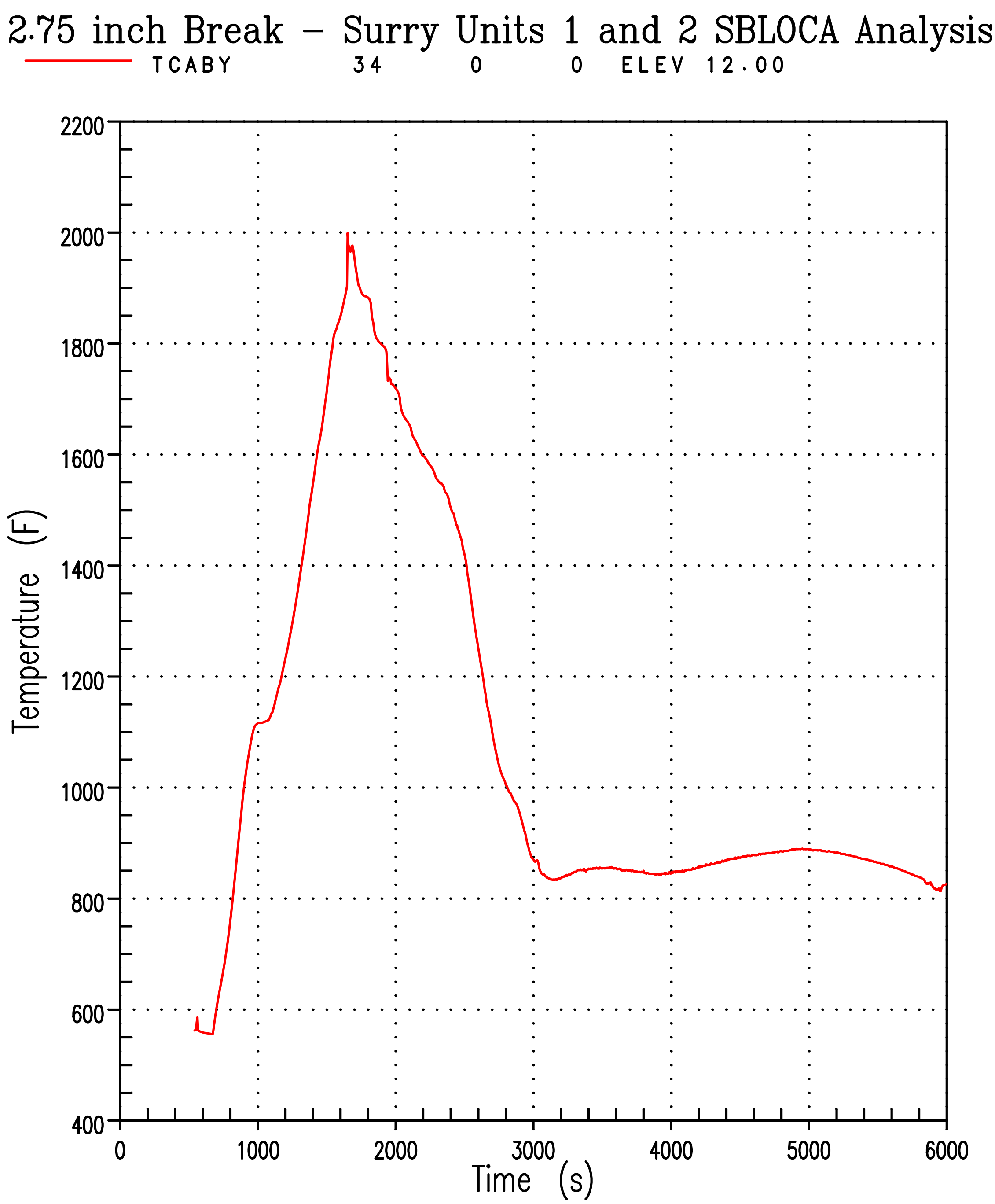


Figure 14.5-72
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION, 3-INCH BREAK

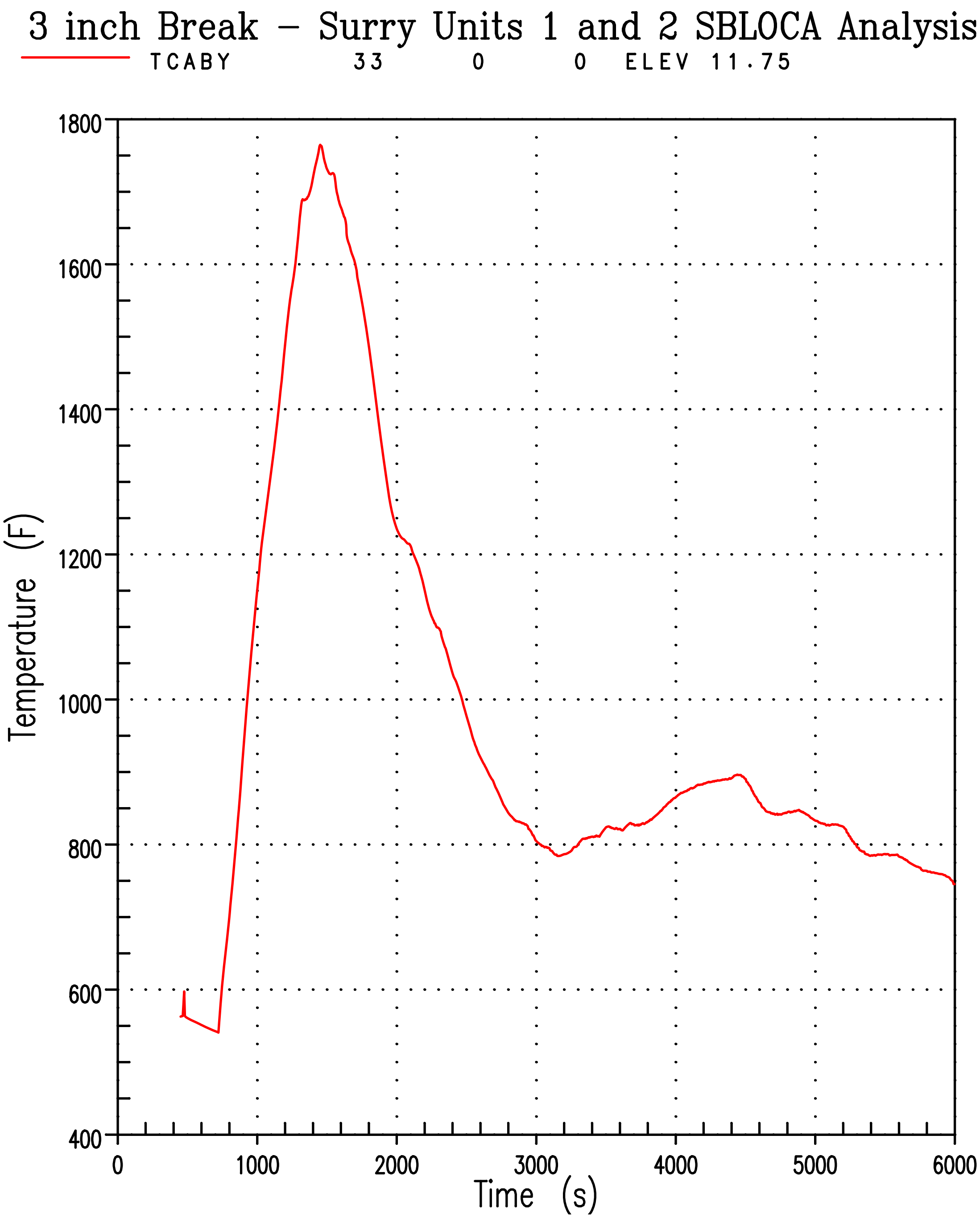


Figure 14.5-73
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION, 4-INCH BREAK

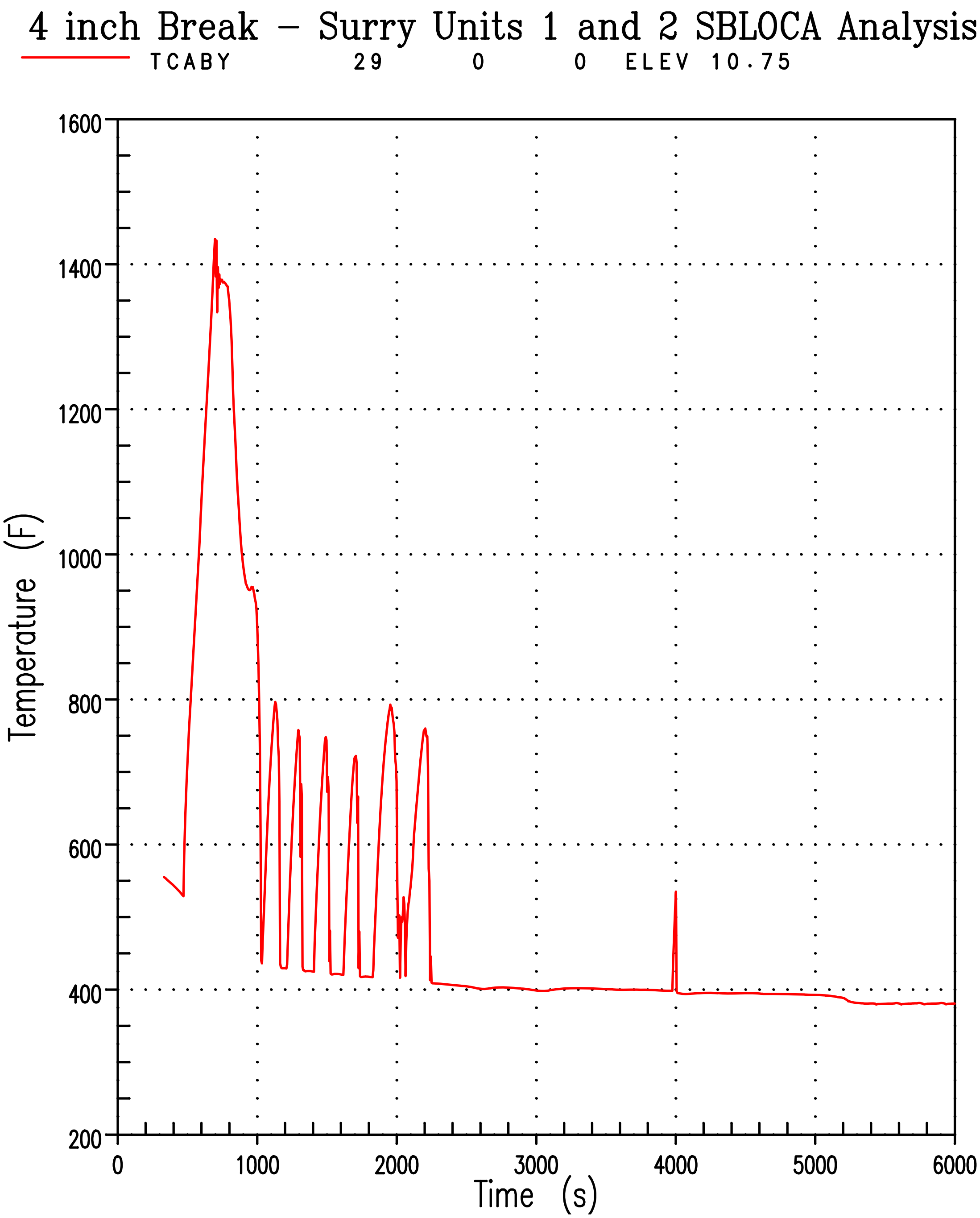


Figure 14.5-74
HOT ROD CLAD AVERAGE TEMPERATURE AT PCT ELEVATION, 5.5-INCH BREAK

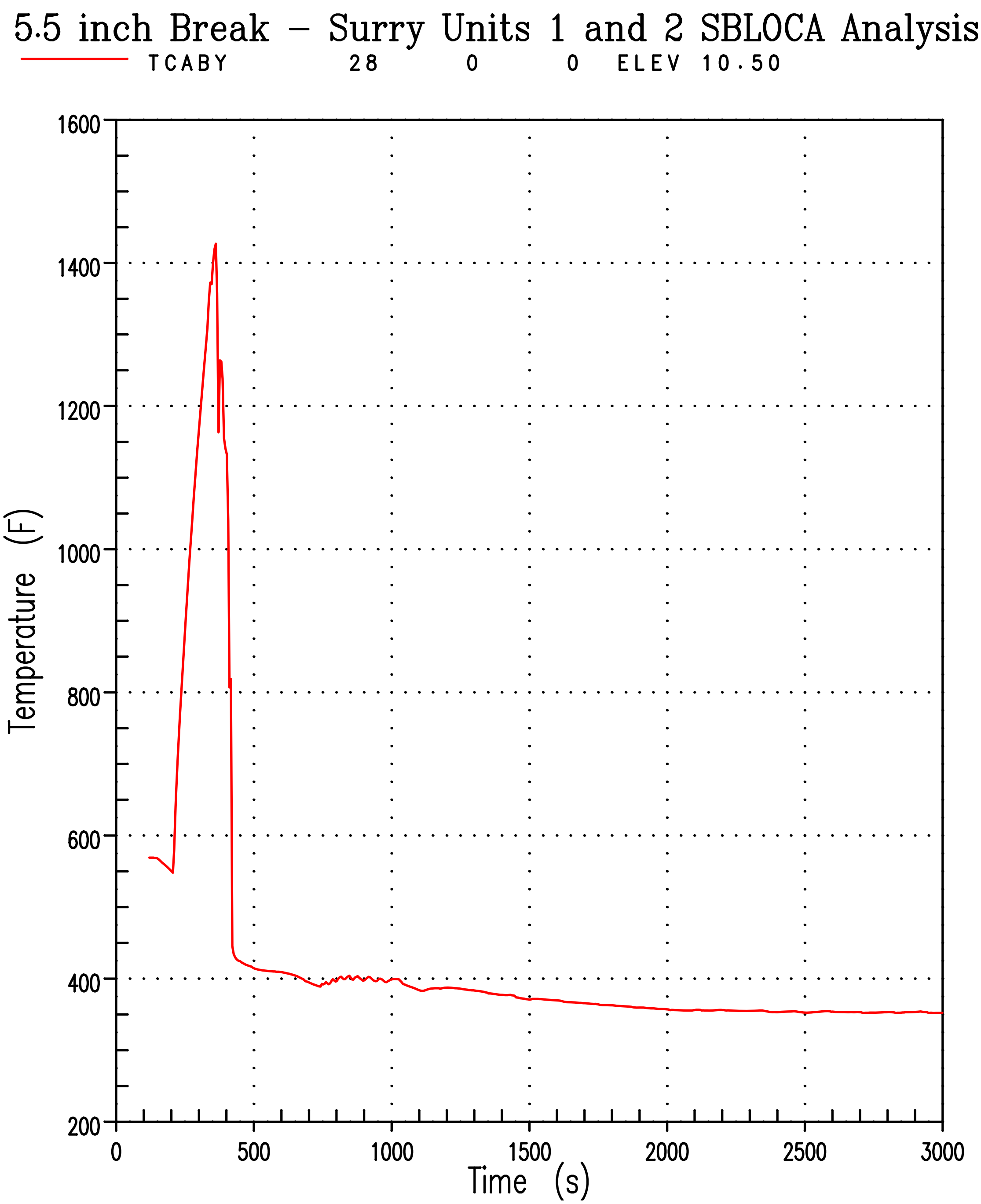


Figure 14.5-75
LOFT SEMISCALE VESSEL BLOWDOWN RUN #522

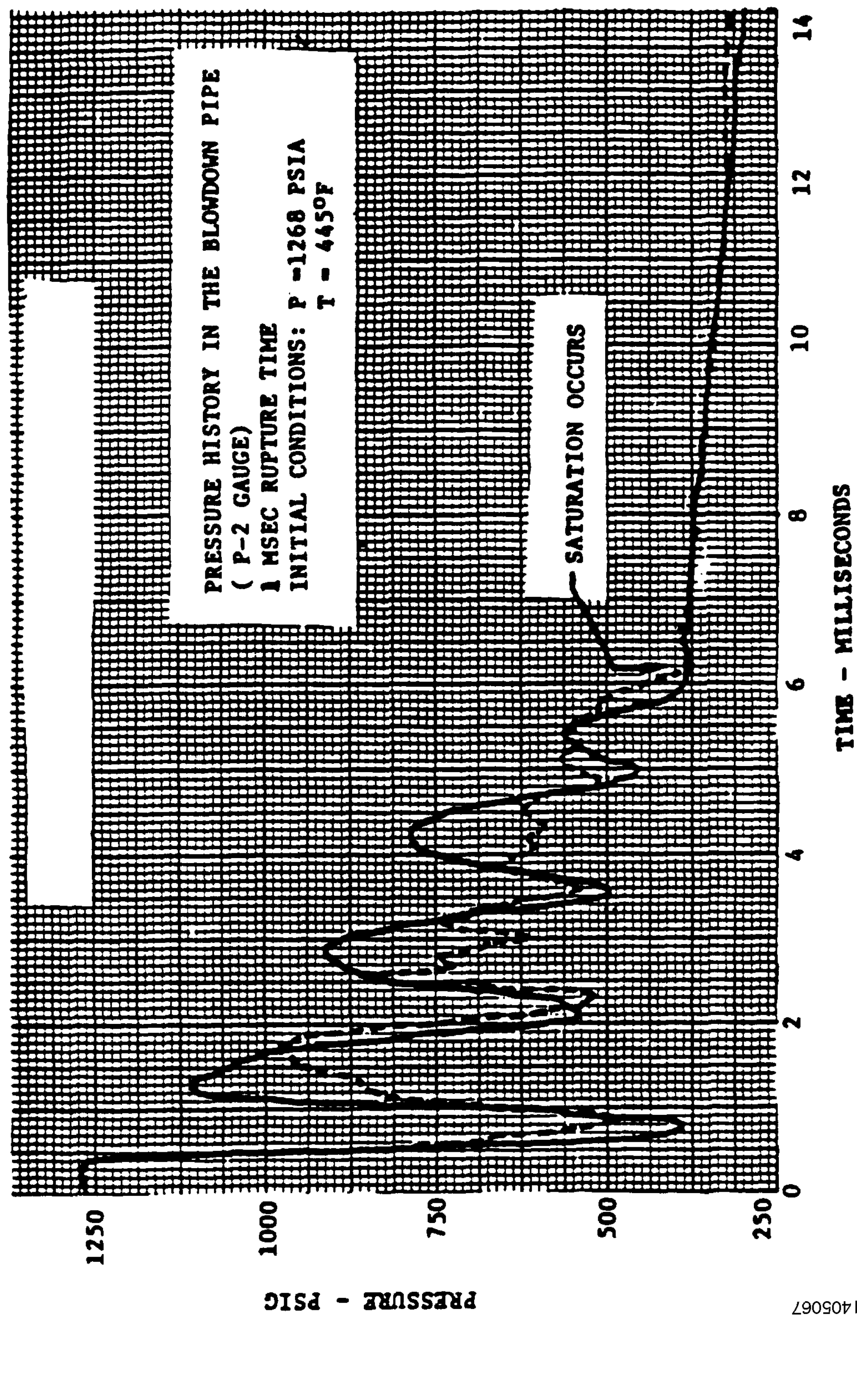
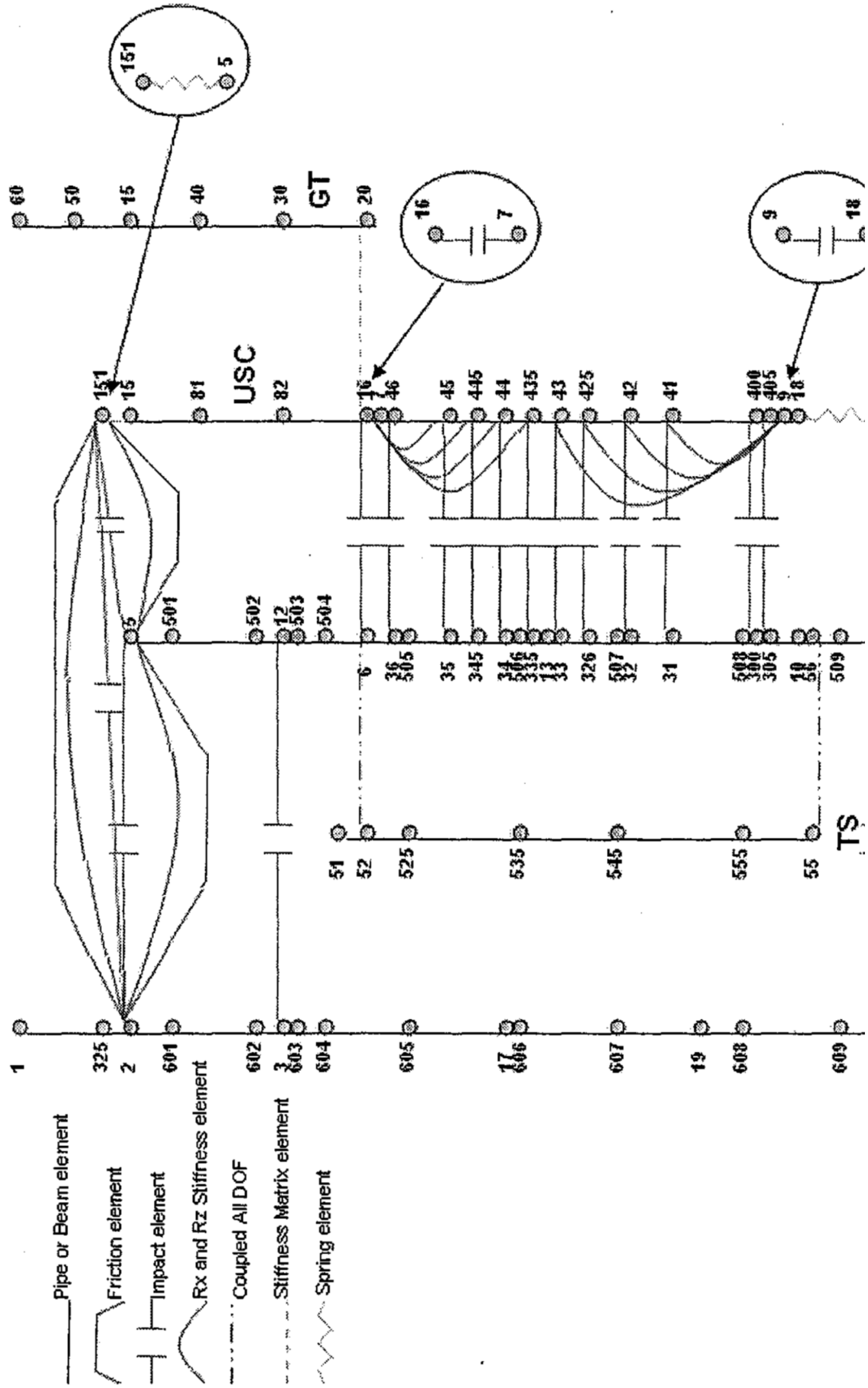


Figure 14.5-76
WECAN REACTOR EQUIPMENT SYSTEM MODEL



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Appendix 14A
Radiation Sources

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The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

APPENDIX 14A RADIATION SOURCES

14A.1 INTRODUCTION

This appendix presents the quantities of radioactive isotopes present in the core and the fuel rod gap. A general discussion of the derivations is also provided.

14A.2 TOTAL ACTIVITY IN THE CORE

The total core activity calculation is consistent with TID 14844 and data from ORNL-2127 (Reference 1). Numerical values for certain significant isotopes are given in Table 14A-1.

14A.3 ACTIVITY IN THE FUEL ROD GAP

The gap activity is computed based on buildup in the fuel from the fission process and diffusion to the fuel rod gap at rates dependent on the operating temperature. For analysis, the fuel pellets are considered divided into five concentric rings, each with release rate dependent on the mean fuel temperature within that ring. The diffusing isotope is assumed present in the gas gap when it has diffused to the boundary of its ring.

The diffusion coefficient, D', for Xe and Kr in UO_2 varies with temperature in accordance with the following expression:

$$D'(T) = D'(1673)\exp\left[-\frac{E}{R}\left(\frac{1}{T} - \frac{1}{1673}\right)\right]$$

Where:

E = activation energy

D'(1673) = diffusion coefficient at 1673 K = 1×10^{-11} sec^{-1}

T = temperature, K

R = gas constant

The above expression is valid for temperatures above 1473 K. Below 1473 K, fission gas release occurs, mainly by two temperature-independent phenomena, recoil and knock-out, and is predicted by using D' at 1473 K. The value used for D'(1673 K), based on data at burnups greater than 10^{19} fissions/cc, accounts for possible fission gas release by other mechanisms and pellet cracking during irradiation.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The diffusion coefficient for iodine isotopes is assumed to be the same as for Xe and Kr. Toner and Scott (Reference 2) observed that iodine diffuses in UO_2 at about the same rate as Xe and Kr and has about the same activation energy. Data surveyed and reported by Belle (Reference 3) indicate that iodine diffuses at slightly slower rates than do Xe and Kr.

For a full core cycle at 2546 MWt, the above analysis results in a pellet-clad gap activity of less than 3% of the dose equivalent equilibrium core iodine inventory. The noble gas activity present in the pellet-clad gap is about 2.5% of the core inventory.

The percentage of the total core activity present in the gap for each isotope is also listed in Table 14A-1.

The core temperature distribution used in this analysis is presented in Table 14A-2.

14A REFERENCES

1. J. O. Blomeke and Mary F. Todd, *Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time,* ORNL-2127, August 19, 1957.
2. D. F. Toner and J. S. Scott, “Fission Product Release From UO_2,” *Nuclear Safety*, Vol. 3, No. 2, December 1961.
3. J. Belle, *Uranium Dioxide: Properties and Nuclear Applications*, Naval Reactors, Division of Reactor Development, United States Atomic Energy Commission, 1961.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14A-1
CORE AND GAP ACTIVITY

Isotope	Ci in the core ($\times 10^7$)	Ci in the gap ($\times 10^5$)
I-131	6.27	16.9
I-132	9.57	3.1
I-133	14.4	14.0
I-134	17.3	3.53
I-135	12.8	7.08
Kr-85	.092	1.46
Xe-133	14.3	32.2
Xe-133m	.388	.602
Xe-135	5.43	.437

Note: Operation at 2546 MWt for 500 days. Temperature distribution specified in Table 14A-2.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14A-2
CORE TEMPERATURE DISTRIBUTION

Percent of Core Fuel Volume Above Given the Temperature	Local Temperature, °F
0.01	4100
0.40	3700
2.20	3300
5.90	2900
11.30	2500

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Appendix 14B
Effects of Piping System Breaks Outside Containment

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APPENDIX 14B EFFECTS OF PIPING SYSTEM BREAKS OUTSIDE CONTAINMENT

14B.1 INTRODUCTION

14B.1.1 Appendix Coverage and Summary

This appendix is based on Appendix D to the initial FSAR and provides the response to a Commission letter dated December 18, 1972 (Reference 1), which contained a document entitled *General Information Required for Consideration of the Effects of a Piping System Break Outside Containment* (later revised in January 1973).

Since Surry Units 1 and 2 are similar in design, and to avoid unnecessary repetition, the analysis within this Appendix is oriented to Unit 2. However, wherever Unit 1 is unique with respect to Unit 2, an additional analysis is made for the unique portions.

This appendix presents an analysis of the consequences of postulated pipe failures outside the containment. In addition to the direct effects on safety resulting from the postulated break of a high-energy line, it is shown in this analysis that Surry Units 1 and 2 can be shut down and maintained in a shutdown condition. The postulated break of a pipe is shown not to negate any safety function as a result of the postulated failure.

The analysis ensures that the Commission's General Design Criterion 4 is met, i.e., that all structures, systems, and components important to safety are designed to accommodate the effects of and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These structures, systems, and components are protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result in equipment failures and from events and conditions outside the nuclear power unit.

14B.1.2 Appendix Organization

The sectional organization of this Appendix is delineated in Figure 14B-1.

The approach used to analyze the consequences of pipe failure is to identify and locate the high-energy sources, identify and locate the safety-related targets, and determine and evaluate the physical effects. The criteria for determining pipe breaks and methods of analysis are presented in Section 14B.2. The identification and location of high-energy systems are found in Section 14B.3. The safety-related and shutdown equipment is identified, and the location listed in Section 14B.4. Calculation results and the evaluation of the physical effects from a pipe system break are found in Section 14B.5. The conclusions are found in Section 14B.7.

14B.2 CRITERIA FOR PIPE BREAKS AND METHODS FOR ANALYSIS

14B.2.1 General Discussion

High-energy systems that require analysis for the consequences of pipe breaks are identified based on the fluid in the pipe, and the pressure and temperature during normal station operation.

In pressurized water reactors, the fluids are water, steam, and water solutions. High-pressure nonflashing gas lines are not included in this analysis.

The temperatures and pressures used for determination of high-energy systems are the maximum normal operating temperatures and pressures. The type of analysis that is required is based on the temperature and pressure conditions as shown in Figure 14B-2. The lines that are both high-temperature and high-pressure are analyzed for pipe whip and environmental effects. The pipes that are low-pressure and high-temperature, or low-temperature and high-pressure, are postulated to crack and are analyzed for environmental effects.

The analysis of these effects (environmental, pipe whip, steam jets, etc.) involves consideration of the source and the target. The source includes the postulated pipe failure and the resulting reactions of the failure. The target includes components or systems that are considered essential in shutting down and maintaining the reactor in a safe-shutdown condition in the event of a postulated break outside containment of a pipe containing high-energy fluid, and which provide protective functions such that a loss of redundancy can be permitted but a loss of function cannot be permitted. The approach taken involved the determination of the effects of the source on the target.

After the high-energy lines are identified in accordance with the above definition, the function of each line is determined. Failure of lines that do not serve a safety function do not require the plant to be shut down. The criterion to which these lines are analyzed is that all safety functions must be protected. Failure of one out of two redundant components is acceptable if the safety function is not degraded. It is assumed that the plant must be shut down to repair damage to safety equipment in accordance with the Technical Specifications.

Failures in lines that serve a safety function require the plant to be shut down and maintained in a shutdown condition. The criterion under which these lines were analyzed is that all redundant components required to operate to recover from this failure are to be protected, including all redundant equipment required to bring the plant to shutdown and to maintain the plant in the shutdown condition.

To analyze the consequences of the postulated break, the targets must be identified. Targets are identified on the various drawings within this appendix.

After the high-energy break points and targets are located, the consequences of pipe whip and jet impingement are determined. The criteria and methods of analysis for determining these effects are discussed below. As a part of the analysis of each break point, it is determined that

either the consequences are acceptable, or pipe whip protection and/or jet impingement protection is required.

14B.2.2 Criteria for Pipe Breaks and Cracks

14B.2.2.1 **Definition of High-Energy Lines**

Design-basis pipe breaks are postulated in piping for which the maximum operating pressure exceeds 275 psig and the maximum operating temperature equals or exceeds 200°F. The critical crack size is taken to be one-half the pipe diameter in length (d/2) and one-half the wall thickness in width (t/2). Pipe cracks (d/2 x t/2) are postulated in piping for which either the operating pressure exceeds 275 psig or the operating temperature equals or exceeds 200°F. If both operating pressure and temperature are below these values, breaks and cracks are not postulated (Figure 14B-2).

Operating temperature and pressure are defined as the maximum temperature and pressure in the piping system, during occurrences that are expected frequently or regularly in the course of power operation, start-up, shutdown, standby, refueling, or maintenance of the plant.

Protection from pipe whip is not provided if any of the following conditions exists:

1. The piping is physically separated by protective barriers or is otherwise isolated from structures, systems, or components important to safety, or is restrained from whipping by plant design features such as concrete encasement.
2. Following a single break, the unrestrained pipe movement of either end of the broken pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety.
3. The internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

   The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiters) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to break an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

14B.2.2.2 **Pipe Break Criteria**

Design-basis break locations are postulated in accordance with the following pipe whip protection criteria. However, where pipes carrying high-energy fluids are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of these structures and systems is provided to cope with the environmental effects (including effects of jet impingement) of a single postulated open crack at the most adverse location with

regard to these essential structures and systems, the length of the crack being chosen not to exceed the critical crack size.

1. ASME Section III, Class I piping breaks are postulated to occur at certain locations in each piping run or branch run. Piping is defined as a pressure-retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers). A piping run is defined as piping that interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movements beyond the restraint required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

   The postulated locations of piping breaks are:

   a. The terminal ends.

   b. Any intermediate locations between terminal ends where the primary-plus-secondary stress intensities S_n (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half of the safe shutdown earthquake and operational plant conditions exceed 2.0 S_m for ferritic steel and 2.4 S_m for austenitic steel.

      Operational plant conditions include normal reactor operation, upset conditions (anticipated operational occurrences), and testing conditions. S_m is the design stress intensity as specified in Section III of the ASME Code.

   c. Any intermediate locations between terminal ends where the cumulative usage factor (U) derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1.

      U is the cumulative usage factor as specified in Section III of the ASME Code.

   d. At intermediate locations in addition to those determined by l.a and l.b above, selected on a reasonable basis as necessary to provide protection. As a minimum, there are two intermediate locations for each piping run or branch run.

2. ASME Section III, Class 2 and 3, and ANSI-B31.1.0 (1967 Edition) piping breaks are postulated to occur at the following locations in each piping run or branch run:

   a. The terminal ends.

   b. Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 (S_h + S_A), or the expansion stresses exceed 0.8 S_A.

S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code, Section III, Winter 1972 Addenda. S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III-1971, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

c. Intermediate locations in addition to those determined by 2.b above selected on a reasonable basis as necessary to provide protection. As a minimum, there are two intermediate locations for each piping run or branch run, selected on the basis of maximum combined primary and secondary stress. For nonseismic piping systems, the intermediate locations are selected on the basis of maximum thermal stress.

3. For systems meeting maximum operating conditions of either pressures greater than 275 psig or temperatures greater than 200°F, piping cracks were postulated at the most adverse points with respect to targets.

14B.2.2.3 **Pipe Break Orientation**

The criteria used to determine the pipe break orientation at the break locations as specified in Section 14B.2.2.2 above are equivalent to the following:

1. Longitudinal breaks in piping runs and branch runs, 4-inch nominal pipe size and larger, and/or
2. Circumferential breaks in piping runs and branch runs exceeding 1-inch nominal pipe size.

A tee-joint that connects a branch run and main piping is not necessarily a break location for the main piping if it does not qualify as a high-stress and/or high cumulative usage factor location in this main piping run; however, at its welding junction to the branch run, which is a terminal point of the branch run, a break location has to be postulated.

If one of the computed stresses and/or cumulative usage factors of the various points of an elbow (tee or reducer) is high enough to be qualified as an intermediate break location, and the other(s) varies within ±l0% of it, all these points are considered as a single break location.

Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe ends.

Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the broken pipe. The dynamic (blowdown) forces resulting from a circumferential break act to separate the piping axially; there is no transverse force during a circumferential break event.

14B.2.3 Methods of Analysis and General Results

14B.2.3.1 Whipping Pipes and Interactions With Concrete Walls

The velocity of a whipping pipe is dependent on:

1. The blowdown forces.
2. The pipe, break geometry, and size.
3. The distance traveled.

A typical mathematical model is shown in Figure 14B-3. At time zero, before the break occurs, the system is in a state of stress due to internal pressure, but these pressure forces are in static equilibrium with the axial loads in the pipe. As the circumferential crack propagates, the load-carrying metal area decreases, so a force imbalance results (Figure 14B-3, Part A). The axial load at the break is assumed to drop linearly to zero in 1 millisecond. After the break, the forces exerted on the pipe by the fluid are determined by time-dependent pressure and momentum effects. The combined behavior of these two terms is equivalent to a pressure drop to 0.7 of the initial value after the passage of the decompression wave (Reference 2). A wave velocity, assumed to be 1600 fps, results in the forcing functions as shown in Figure 14B-3, Part B.

The results of the above analysis indicate that, during most of the pipe displacement, the applied forces are only 0.7 of the initial forces and that approximately 30% of the energy is dissipated by plastic deformation in the pipe before impact. Due to strain hardening and strain rate effects, a distinct hinge may not form, but rather an extended region of large plastic deformation occurs. The plastic hinge lengths are also determined by this analysis (Table 14B-1) for the initial condition of 1050 psi.

Effects from whipping pipes on concrete walls were analyzed as follows. The local crushing stiffness of the pipe elbow may be readily determined in the elastic range, but only with difficulty once plastic deformation begins. The case of the actual elliptical contact area between the pipe elbow and wall has been considered, as well as an idealized case in which the portion of the elbow near the contact area is modeled as an equivalent sphere. PISCES (Reference 3) computer runs indicate that once crushing (or denting) is initiated, a flat area forms on the elbow. (Without internal pressure, bounce-back or “oil canning” occurs.) The forces transmitted by the wall to the pipe occur mainly at the circumference of the contact area. Thus, analyses presented in the literature for the stiffness of a sphere intersected by a pipe with normal loading may be used to get an approximate stiffness (Reference 4).

The crushing resistance of the elbow is modeled as a spring (connected to ground) in the mathematical model. This is acceptable, since the great inertia of the wall prevents any appreciable movement prior to the moment that the peak forces occur. The peak force in this spring is the maximum load transmitted to the wall during the impact. The effects of the continuing blowdown forces and the inertia of the pipe away from the impact point are automatically included in the analysis. Typical examples of these peak forces as a function of

impact velocity are plotted in Figure 14B-4.

Since the load is applied to the concrete wall in a short time compared to the natural period of a concrete wall, the application of a dynamic load factor of two is required when using static design equations. The model used for punch shear is shown in Figure 14B-5. The equation used for punching shear (Reference 5) in a concrete wall is:

$$F = 4\sqrt{f_c}d2\pi\left(r + \frac{d}{2}\right)$$

where:

F = applied force

f_c = compressive strength of concrete

d = wall thickness

r = radius of contact area

In all cases, wall thicknesses employed in normal plant construction are sufficient to stop whipping pipes.

14B.2.3.2 **Fluid Jets and Interactions With Reinforced-Concrete Walls**

14B.2.3.2.1 Assumptions

1. The pipe break location is very close to the pressure reservoir(s). The pressure drop in the pipe due to flow friction is negligible.
2. The total jet force remains constant throughout its traveling distance; i.e., the friction force between the jet stream and ambient air is negligible.
3. The jet stream is totally intercepted by the concrete wall.
4. The jet impingement is a suddenly applied load to the concrete wall.

14B.2.3.2.2 Jet Force

The maximum value of the initial jet pressure from a pipe break can be expressed as:

$P_J = C_J P_o$

where:

P_o = fluid pressure inside pipe

C_J = jet coefficient

C_J = 1.26 for steam

C_J = 2.0 for subcooled nonflashing water

(If the pressure drop due to friction is taken into consideration, the values of C_J can be reduced.)

The total jet force is then:

$F_J = P_J A$

where:

A = pipe break area $= \frac{\pi}{4} D_p^2$

D_p = inside diameter of pipe

As the jet stream progresses away from the pipe break area, the width of the jet increases with the axial distance. The angle of divergence is assumed to be 20 degrees (Reference 6).

14B.2.3.2.3 Punch Shear Failure of Concrete Wall

The punch shear failure mechanism of a concrete wall due to jet impingement from a pipe break is shown in Figure 14B-6. The failure of a concrete wall is a diagonal cracking along the surface of a truncated cone or pyramid around the jet impingement area. The area of the shearing surface is:

$A_s = \pi D_w W$

where:

$D_w = D_P + 2L \tan 10° + W$

W = wall thickness

L = distance between wall and pipe break location

Without shear reinforcement, the shear strength is:

$$V_c = 4\sqrt{f'_c}$$

where:

f'_c = specified compressive strength of concrete

The total shear resistance of the reinforced-concrete wall is:

$R = ø A_s V_c$

where:

ø = capacity reduction factor = 0.85 for shear

The total jet impingement load seen by the wall is:

$F_T = C_D F_J$

where:

C_D = dynamic load factor = 2.0 for suddenly applied load

If R is greater than or equal to F_T, there will be no punch shear failure.

Curves relating dimensionless wall thickness ($X = W/D_P$) and dimensionless distance ($Y = L/D_P$) are shown in Figures 14B-7 and 14B-8 for steam and water lines, respectively. The specified concrete compressive strength, f'_c, is assumed to be 3000 psi.

These curves are extremely conservative. A more realistic analysis to determine the effective jet impingement force requires additional parameters, such as pipe lengths from sources, elbows, and flow restrictors and fluid characteristics. A conservative approach was used in the analysis for this appendix.

14B.2.3.2.4 Fluid Jets and Interaction With Steel Plates

For a fluid jet issuing from a crack (one-half the pipe diameter times one-half the pipe thickness) in a pipe wall, the magnitude of the jet force is small because the break area is small. It can be shown that either a 1-foot reinforced-concrete wall or a 1/8-inch steel plate being hit by a jet from close distance from a crack in a 32-inch, 2300-psi water line will not experience a local failure by punch shear. Therefore, it is not necessary to analyze the local punch shear failure of concrete walls or steel plates due to fluid jets from cracks in pipe walls.

14B.2.3.2.5 Fluid Jet Range

Any safety-related structure, piping component, and equipment located in the fluid jet traveling path is considered susceptible to jet impingement. As the jet propagates away from the pipe break area, it expands at a diverging angle. Therefore, the jet intensity decreases with distance from the break location to the target, whereas the total jet force is assumed to remain constant.

14B.2.3.3 **Pressure and Environment**

The pressure buildup from the postulated break of a high-energy pipe in a compartment or building is calculated using the computer program CUPAT (Reference 7).

14B.2.3.3.1 Introduction

CUPAT is a computer program used to calculate pressure and temperature transients in various nuclear power plant compartments resulting from a postulated high-energy pipe break.

The output is used mainly for design purposes in establishing the peak pressure differentials across the compartment walls.

This program was derived from the LOCTIC computer program (Reference 8) which was used to calculate pressure and temperature transients for the primary containment. Chapter 5 discusses the current method for primary containment analysis. There are two major differences between LOCTIC and CUPAT:

1. LOCTIC includes the effects of heat transfer by providing subroutines to handle sources and sinks. CUPAT assumes a volume that receives heat and mass from a broken piping source and discharges heat and mass to its surroundings, but aside from that there are no other heat sources or sinks (adiabatic assumption).
2. CUPAT allows for flow out of the volume considered as well as flow in. There is no provision in LOCTIC for mass outflow from the containment volume.

In order to calculate the transients within a compartment, CUPAT numerically solves finite difference equations defining heat and mass flows into and out of the compartment. The program uses the same basic assumptions as those used in LOCTIC, namely:

1. Mass and energy added or removed during each small time step are based on rates determined at the start of the time step; i.e., during any time interval, the thermodynamic state is assumed to be steady and the response of the flow out of the volume to changes in the thermodynamic state is instantaneous (quasi-steady-state assumption).
2. The atmosphere in the compartment mixes instantaneously and homogeneously, i.e., at each point in time, the atmosphere is in a state of thermodynamic equilibrium.

A detailed description of the approach to the problem is presented below.

The calculational approach used in CUPAT is summarized in the block diagram shown in Figure 14B-9. Blocks (1) through (5) in the figure are traversed once for each time step.

14B.2.3.3.2 Calculational Approach

14B.2.3.3.2.1 *Quasi-Steady-State Assumption.* The problem of analyzing the transient effects of a LOCA is very complex. The thermodynamic state of the compartment atmosphere is continuously changing. This state depends on the mass and energy flows into and out of the compartment. The flows, in turn, are dependent on the thermodynamic state within the compartment. In order to solve such a problem numerically, some simplifying assumptions must be made.

First, the system is defined as the compartment atmosphere at any given time. This includes any air, steam, and water droplets present, but not the walls, equipment, or internal structure of the compartment itself. If the time step is small enough, the net rate of mass and energy addition to the system will not vary appreciably during the time step. Thus, the flow rates are calculated assuming that the thermodynamic state does not change during the time step, and this assumption

eliminates the need to iterate and converge on the inflow and outflow for each time step. This approach was used in LOCTIC (which also includes heat flows) for the primary containment transients, and is also used in CUPAT.

14B.2.3.3.2.2 *Mass and Energy Flow Rates into Compartment.* The mass and energy flow rates into the compartment are supplied as input to the program in tabulated form. These blowdown rates into the compartment may be obtained from the output of a LOCTIC or LOCTVS (Reference 9) computer run or from the assumption of Moody flow (Reference 10) with a known pressure blowdown.

The flow of fluid from a piping break is relatively insensitive to the back pressure in the compartment, since the pressure in the high-energy line is above 275 psig. Thus, the mass and energy inflow data specified as input are close to the actual flow, but are conservatively high.

14B.2.3.3.2.3 *Calculation of the Thermodynamic State of the Compartment.* In each time step of the numerical calculation, equilibrium temperature and pressure in the compartment are determined based on new values of mass and internal energy. Properties of water are obtained from the steam tables. The detailed procedure by which the pressure and temperature of the compartment atmosphere are found from the updated values of mass and internal energy is described below.

Initially the equilibrium state is considered to be a two-phase mixture of air, saturated steam, and saturated liquid. However, if the energy content for the given mass is greater than that required for saturation, a single-phase mixture of air and superheated steam is determined.

To arrive at the correct equilibrium conditions, a curve of internal energy of the air, steam, and liquid in the volume versus temperature is generated. The basis for the curve is that the mass of water present in the compartment is at a saturated equilibrium state for each temperature, and the total internal energy of the system at this temperature is calculated accordingly. The actual total internal energy is then used to enter this curve and find the true temperature. The total pressure is then determined by adding the vapor pressure to the air partial pressure, which is calculated by the ideal gas flow at this temperature.

In the case where the contents form a superheated vapor, the superheat section of the steam tables is used to match the specific volume of the steam and the internal energy to find the equilibrium temperature and pressure.

14B.2.3.3.2.4 *Calculation of Flow Rate Out of Compartment.* The CUPAT computer program uses the LOCTVS vent flow (Reference 11) to determine the flow rate out of the compartment. A homogeneous flow model is used in LOCTVS to calculate flow out of the drywell through the vents of a pressure suppression containment. Although flow through the vents is characterized by slip between the gaseous and liquid phases, a homogeneous model yields lower flow rates and is used for conservatism. The ability of the vent flow model to conservatively predict flow through the vents has been checked against the Bodega and Humboldt Bay pressure suppression tests.

14B.2.4 Protection Against Pipe Whip

A combination of three basic approaches was used for the protection of targets from whipping pipes. These approaches include:

1. Separation of redundant features by distance or location so that at least one feature remains intact.
2. The incorporation of many redundant features into the design of the safety-related systems for assurance of reliability.
3. For the largest main steam and feedwater lines, an extensive inspection program was devised for each postulated break point. By means of ultrasonic and/or radiographic testing in addition to a visual surveillance program, defect propagation can be detected at any early stage and repaired accordingly, thereby ensuring the integrity of each postulated break point.

14B.2.5 Analysis of Seismic Category I Structures

Analysis of Seismic Category I structures for loads other than pipe break in the main steam valve house is given in Section 15.2.

14B.3 HIGH-ENERGY SYSTEMS

14B.3.1 System Identification

The following systems contain high-energy lines, as defined in Section 14B.2:

1. Main steam.
2. Feedwater.
3. High-pressure heater drains and vents.
4. Moisture separator drains.
5. Auxiliary steam.
6. Condensate.
7. Low-pressure heater drains and vents.
8. Boron recovery.
9. Liquid waste.
10. Chemical volume and control.
11. Safety injection.
12. Steam generator blowdown.
13. Extraction steam.

14. Sample.

The high-energy lines in these systems were reviewed in conjunction with safety-related and safe-shutdown equipment (Table 15.2-1) by means of a detailed drawing review and onsite inspection. Those portions of the high-energy lines in proximity to the safety-related and safe-shutdown equipment have been identified. These portions of the high-energy lines are defined as sources and are presented in Table 14B-2.

The safety-related equipment and plant shutdown equipment in proximity to these sources (identified as targets) are listed in Section 14B.4.1.

Table 14B-2 presents a listing of the high-energy line sources with their maximum operating conditions, locations, and seismic classifications. These lines were individually analyzed for adverse effects on targets. Sources such as smaller lines located in the target areas were not individually analyzed, since the sources listed were the worst cases for their respective areas.

14B.3.2 Quality Assurance and Inspection

Piping presently installed was designed and fabricated in accordance with the criteria described in Section 1.4, *Compliance with Criteria*.

14B.3.3 Detection of Failures

As described in Section 7.2 and delineated in Table 7.2-1, reliable and redundant systems have been incorporated into the present plant design for detection of failures in the main steam and feedwater systems.

As described in Section 11.3.4, the area radiation monitoring system is designed to alarm when radiation levels in their associated areas are slightly above background. This system detects pipe failures in systems containing radioactive fluids.

Detection for breaks in lines routed through the Auxiliary Building is discussed in Section 14B.5.3.3.

14B.4 PLANT SHUTDOWN AND SAFETY-RELATED EQUIPMENT

14B.4.1 Introduction

Table 14B-3 lists the systems and major equipment locations that constitute postulated targets among the plant shutdown and safety-related equipment. Associated cables and controls are considered along with this equipment.

14B.4.2 Emergency Procedures

Main steam or feedwater breaks outside the containment are discussed in Section 14B.6. Subsequent to a main steam or feedwater break, assuming offsite power is unavailable, plant

shutdown is achieved by actuation of the emergency core cooling system and removal of core decay and sensible heat via steam release through the steam generator power operated relief valves, and maintenance of steam generator water inventories by means of the auxiliary feedwater system.

Section 9.3 details the operation of the residual heat removal system necessary for long-term cooling and cold shutdown of the reactor.

Shutdown equipment is normally controlled from the control room. However, in the event that evacuation of the control room is necessary, shutdown equipment can be controlled from an auxiliary shutdown panel.

Emergency procedures direct the operators to perform mitigating actions in the event of a high-energy line break outside containment. The operator response to a break in the main steam valve house is described in Section 14B.6.

14B.4.3 Relationship of High-Energy Lines to Safe-Shutdown and Safety-Related Equipment

Figures 14B-10 through 14B-17 show the high-energy systems and the safe-shutdown and safety-related equipment.

14B.5 EFFECTS OF PIPE BREAKS AND CRACKS

14B.5.1 Main Steam

14B.5.1.1 Break Locations

Break locations were postulated in the main steam lines from the containment to the turbine building in accordance with Section 14B.2.2. For the main steam line, 0.8 of the allowable thermal stress is 22,500 psi, and 0.8 of the allowable combined primary and secondary stress is 0.8 (S_A + S_h) = 37,500 psi. Since 0.8 of the allowable stresses was not exceeded, the two intermediate locations between terminal points were selected on the basis of maximum primary-plus-secondary stress. Piping downstream of the manifold common to the three steam lines was not analyzed seismically. For this piping, pipe breaks were assumed; however, because of separation, no further analysis is required. At all break points, both circumferential and longitudinal breaks were postulated to occur.

The break points are listed in Table 14B-4 along with the thermal and combined stress levels. The break locations are shown on Figure 14B-10.

Cracks were selected in the vicinity of all targets.

14B.5.1.2 **Separation**

The steam lines in the turbine building were analyzed, and satisfactory separation was found to exist between steam lines and any safety-related features.

The control room and emergency diesel-generator rooms are separated by sufficient distance from all high-energy lines, so that whipping pipes or steam jets will not adversely affect their respective functions. These conclusions were based on results given in Section 14B.2.3.

The auxiliary feedwater modification described in Section 14B.5.1.7 provides a system widely separated from postulated breaks.

14B.5.1.3 **Pipe Whip**

An extensive nondestructive testing program, as described in Section 14B.5.1.6, is used to preclude breaks, thereby making pipe whip a noncredible incident.

Since the guideline referenced in Section 14B.1.1 requires a postulated failure, each postulated main steam line break has been evaluated for the effects of pipe whip. Because feedwater supply to the steam generators is the ultimate requirement for a safe shutdown, the evaluation was based on maintaining the feedwater function. The results of this investigation are shown in Table 14B-5. These results were based on the plastic hinge lengths established in Section 14B.2.3.1 and on discussions with the turbine-driven auxiliary feedwater pump manufacturer indicating that the pump can operate in a steam environment.

Since loss of offsite power must be assumed, and the turbine drives are not environmentally qualified by tests, additional assurance that feedwater will be maintained is obtained by the auxiliary feed cross-connect system. As shown in Table 14B-5, there are no effects on the auxiliary feed cross-connect system, considering all postulated breaks. The auxiliary feed cross connect was a modification to the initial plant design and is discussed in Section 14B.5.1.7.

14B.5.1.4 **Fluid Jet Effects**

Jet impingement loadings on the walls, valves, and pipes inside the main steam valve house have been calculated. The time-history results of jet force from a pipe break at the most adverse location in the steam line within the valve house is calculated as follows (Section 14B.2.3.2):

$F(t) = 1.0\ P_oA$ for $t \leq 0.020$ sec

$F(t) = 0.7\ P_oA$ for 0.020 sec $< t \leq 0.113$

$F(t) = 0.19\ P_oA$ for $t > 0.113$ sec

The initial jet force on the walls was calculated as:

$F_o = K\ P_o\ A$

where:

K = initial thrust coefficient = 1.0

P_o = hot standby pressure = 1005 psig

A = flow area = 616 in^2

F = 619 kips

For the longitudinal breaks, the break size was taken as 60 x 10.3 inches with the jet diverging at a 20-degree inclusive angle. The impingement areas, jet pressures, and loads corresponding to each of the postulated breaks are indicated in Table 14B-6.

Local damage to the walls and floors was checked based on Section 14B.2.3.2. The conservative calculation, which assumes a dynamic factor of two and no energy loss, indicates that the floor at Elevation 27 ft. 6 in. is subject to some local damage from jet impingement. However, the containment, the walls, and the roof withstand the effects of jet impingement with no punch shear damage.

For breaks of the main steam lines, jet impingement loads on the valves were calculated. The maximum normal force on the valves is given by:

$$F_v = C\ P_t\ A_t \cos^2 \alpha / 1000 \text{ (kips)}$$

where:

C = shape factor for flow around the valve, cylinder, $C = 0.6$

P_t = initial jet pressure at the target (psi)

A_t = impingement area (in^2)

α = incident angle

The maximum normal jet forces on the valves are listed in Table 14B-7. It should be noted that these loads drop instantaneously to a fraction of the levels recorded. Available calculations indicate that the nonreturn valve will continue to function as a check valve, preventing blowdown of the undamaged loops. Calculations indicate that jet impingement will not break a main steam trip valve housing, but can cause the valve to fail open. However, with the nonreturn valve operable, a broken line could still be isolated, so that only one steam generator would blow down.

14B.5.1.5 **Pressure and Environment**

The main steam valve house and the reactor trip switchgear room, as shown in Figure 14B-10, are the only target areas that can be affected by steam following a postulated break or a crack in a main steam line, as discussed in the following sections.

As shown on Figure 14B-10, the reactor trip switchgear room is far removed from the source lines; therefore, the probability of having a steam environment within the room is extremely remote.

With the existence of the nondestructive testing program, as described in Section 14B.5.1.6, only smaller steam-line breaks need to be considered for pressure effects on the main steam valve house. The pressure buildup within the valve house, following a smaller steam-line break, is negligible.

In order to comply with the criteria, as referenced in Section 14B.1.1, the pressure in the main steam valve house has been calculated for the largest steam-line break. Frictionless Moody flow and the CUPAT computer program, as described in Section 14B.2.3.3, were used in these calculations. The results are shown in Figure 14B-18.

The main steam valve house contains many targets, as shown in Table 14B-3. As detailed below, all the targets either fail in the safe position or operate mechanically:

1. Targets that fail in the safe position.
   a. Main steam trip valves (close).
   b. Auxiliary feed pump steam isolation valve (open).
2. Targets that operate mechanically.
   a. Feedwater isolation check valves.
   b. Main steam nonreturn valves.
   c. Main steam safety valves.
   d. Turbine driver for auxiliary feed pump.

Furthermore, all electrical cables considered as targets are the same as the cables used inside the containment. Since this cable has been qualified by test for post-design-basis-accident conditions inside the containment, the cabling is not subject to common mode environmental failure.

The blowdown rates used to obtain the above results were based on blowdown of one steam generator, following the postulated steam-line break. The blowdown of only one steam generator can be justified by taking no credit for the main steam trip valves, but taking credit for the nonreturn valve (NRV-201A, B, C for Unit 2) in the affected steam line. Credit for the nonreturn valve is justified as follows:

1. As described in Section 14B.5.1.4, jet impingement will not affect the intended performance of this valve.
2. There is no instrumentation or electrical component required for operation of the valves. The nonreturn valves require only reverse steam flow for their intended operation.

3. In the worst case, where blowdown is the greatest following the postulated steam-line break, the steam system is in the hot standby condition. Blowdown is greatest for this case, since the steam-line pressures are at a maximum. In this condition, there is little or no steam flow to hold the valve disk in an open position; therefore, the valve is performing its required function even before the postulated failure. In all cases, when the system pressure is high, with respect to the pressure at 100% power, the flow rates are low and the valve is in a nearly closed position before the postulated incident occurs.

14B.5.1.6 **Inspection Program for Large Steam Lines**

An extensive nondestructive testing program is provided for the main steam postulated break points in the main steam valve house. These points, a total of 12 for each unit, are shown in Figure 14B-10.

14B.5.1.6.1 Procedures

A program of periodic examination exceeding the requirements of ASME Section XI, Winter Addenda 1972, as instituted by Regulatory Guide 1.51, was originally provided as follows:

1. A baseline examination was performed including 100% coverage of all subject points.
2. Inservice examinations were performed including 100% coverage of all subject points for the initial 3 years of the 10-year inspection interval as defined by ASME Section XI.
3. Examinations were performed including 100% coverage of all subject points for two subsequent inservice examinations.

Currently, the Augmented Inspection Program includes periodic examinations that meet the requirements of the Technical Specifications and the Technical Requirements Manual. This program requires ultrasonic and surface examinations to be performed on 1/3 of the welds every 40 months, with a cumulative 100% coverage of all welds by the end of the interval. Repairs are made as required. Upon completion of any repairs, the program, as described above, will be reinstated for the repaired postulated break point.

In addition to the above testing program, a visual inspection of the surface of the insulation at the main steam break point locations in question is performed weekly for detection of leaks. If a leak is detected, it is immediately investigated and subsequently repaired if the leakage is caused by a through-wall crack. The investigation allows for evaluation of system functionality to determine if continued plant operation can or cannot be justified, with consideration of the ASME Section XI Code for applicable Class piping, prior to the repair.

14B.5.1.6.2 Methods

The ultrasonic test procedures include the examination of the postulated break points and heat-affected zones. Consideration of weld thickness, geometry, material, and curvature parameters results in establishing the appropriate transducer sizes, optimum beam angles, and

frequencies for test reliability and repeatability. Where appropriate, additional techniques are used for evaluating reflectors and obtaining characterization data. Test sensitivity is in accordance with ASME Section XI, which defines reference calibration requirements.

14B.5.1.6.3 Basis for the Inspection Program

As shown in a PVRC report (Reference 12), and a Virginia Power Technical Report (Reference 15), toughness of nuclear power plant piping materials is high enough to prevent brittle fracture at operating conditions. This conclusion can be supported by fracture mechanics calculations.

Furthermore, from the following fracture mechanics techniques and calculations, the critical size of surface and internal flaws far exceeds the thickness of the piping material. Consequently, a surface or an internal flaw will extend through the wall thickness and form a subcritical through-wall crack, which will leak before it reaches its critical size.

The main steam line material is SA155, grade CMS 75, Class 1, outside diameter 30 inches, wall thickness 1 inch. Plate material for piping is SA299. Fittings were fabricated from ASTM A299 steel plate stock, using the ASTM A234 Grade WPB specification. Fitting material equivalent to ASTM A691, Grade CMS 75, Class 32: carbon-manganese-silicon alloy steel can be used as replacement material for the main steam line pipe and fittings.

Feedwater line material is ASTM A106 Grade B: Carbon steel. Fittings are ASTM A234 WPB. ASTM A335 Grade P11 or P22: Chromium-Molybdenum steel can be used as replacement material for feedwater seamless pipe. ASTM A234 WP11 or WP22 can be used a replacement material for fittings.

For both main steam and feedwater piping, the ASME SA equivalent materials can be used as a preferred substitute for ASTM materials.

14B.5.1.6.4 Fracture Mechanics

The application of fracture mechanics techniques allows prediction of the critical flaw size that can cause fast or unstable fracture in a stressed structure.

When the critical flaw size is established for a nominal stress level, it is possible to decide the acceptable defect size. One of the criteria is the leak-before-fracture criterion, which requires that the defect wall propagate slowly through the wall of the pipe and that the pipe will leak before the crack is large enough to trigger the fast fracture.

Fabricated structures usually contain several types of defects, including surface flaws, internal flaws, and through-the-thickness cracks. The critical flaw size will be calculated for each of these flaws using fracture mechanics relationships. These formulas were used in two recently published papers (References 12 & 13) treating similar problems. As emphasized in the PVRC Recommendations on Toughness Requirements for Ferritic Materials (Reference 12), the pipe wall section is usually not thick enough to support plane strain fracture propagation, which can be

properly analyzed by the fracture mechanics methods. In other words, the load limits and critical flaw size calculated using fracture mechanics will in general be more conservative for pipe than for the thick section structures where the plane stress conditions can exist. Fracture will occur when the value of the stress intensity factor K_I reaches the critical value K_{IC}. The critical flaw size is related to the K_{IC} in several different formulas depending on geometry of structures, flaws, shape, and environmental factors. In this work, the following assumptions were made about factors affecting the relation between the K_{IC} and the critical flaw size:

1. Material properties (toughness and strength) of the weld metal and the heat-affected zone in the longitudinal and circumferential weldments are the same as in the base material.

2. The lowest and the highest temperatures in the main steam line are 510°F and 547°F. The lowest and the highest temperatures in the feedwater line are 438°F and 450°F. However, only the lowest temperatures are used in calculations of fracture toughness because they give more conservative values for critical crack size.

3. Because of uncertainty involved in evaluating the possible stress state, Irwin's (Reference 14) suggestion was accepted that the membrane stress is equal to the yield strength. For SA106B pipes, Class 1 data are given in Section III. For SA155 (plate SA299) material, the elevated temperature yield strength was not given in Section III, Class 2, and the allowable design stress data for Class 1 were used to get the yield stress.

4. The stress intensity factor of 300,000 psi $\sqrt{\text{in.}}$ was used (Reference 13) for SA106B pipe. In this work, a lower value of 200,000 psi $\sqrt{\text{in.}}$ was accepted, which would correspond to the reference stress intensity factor K_{IR} at the temperature NDT + 180°F. The lowest temperature for SA106B pipe is 438°F, and for SA155 pipe, 510°F, which means that the NDT temperature in the first case would be 438–180 = 258°F, and in the second case, 510–180 = 330°F. This is of course a very conservative assumption, because the NDT temperature for these materials is below room temperature.

Toughness of replacement materials is documented in Reference 15. This reference provides technical justification for use of replacement materials based upon fracture toughness of materials. The replacement materials are assessed using linear-elastic fracture mechanics, elastic-plastic fracture mechanics, and limit load methods.

14B.5.1.6.4.1 *Internal Flaw.* The internal flaw is an ellipsoid, as shown in Figure 14B-19, Part A. The flaw is located in the center of the pipe wall. The flaw can be axial (major axis parallel to the pipe axis) or circumferential (major axis perpendicular to the pipe axis). A further assumption is that the flaw is small compared to the pipe radius. Thus the curvature effect can be neglected and the pipe can be approximated with an infinite plate under uniform applied stress. The stress intensity factor K_I for this model is given (Reference 13) as:

where:

σ = applied stress

$$K_I = \left[\sin^2\beta + \frac{a^2}{b^2}\cos^2\beta\right]^{1/2} \frac{\sigma(\pi a)^{1/2}}{\phi}$$

β = angle at which the stress intensity is calculated

ø = the following elliptical integral:

$$ø = \int_o^{\pi/2}\left[1 - \frac{b^2 - a^2}{b^2}\sin^2\theta\right]^{1/2} d\theta$$

At the tip of the major axis $\beta = 0$, while at the tip of the minor axis, $\beta = \frac{\pi}{2}$.

If it is assumed that the major axis of the ellipsoid is twice as long as the minor axis, the equation for K_I becomes:

$$K_{IC} = 0.826\ \sigma\ (\pi a_{cr})^{1/2}$$

It has been shown that, for an elongated crack (b>>a), the critical stress intensity factor is given by:

$$K_{IC} = 1.2\ \sigma\ (\pi a_{cr})^{1/2}$$

Substituting the values for the stress intensity factor and applied stress (yield strength at the temperature) in the first equation for K_{IC} results in the following:

Material	Temperature °F	2 a_{cr}(Critical Flaw Size), in.
SA106B	400	42.0
SA155	500	46.5

Substituting the values for the stress intensity factor and applied stress in the second equation for K_{IC} results in the following:

Material	Temperature °F	2 a_{cr}(Critical Flaw Size), in.
SA106B	400	19.6
SA155	500	22.8

As can be seen, all critical flaw size values are much greater than the wall thickness, which means that the flaws would extend through the wall without becoming critical. In other words, the internal flaw will become a through-the-thickness crack and will leak.

14B.5.1.6.4.2 *Surface Flaw.* The surface flaw is a semi-ellipsoid, as shown in Figure 14B-19, Part B. The flaw can be axial or circumferential, as in the previous case. Again the curvature effect is neglected, and the stress intensity factor is given (Reference 13) as:

$$K_{IC} = 1.12\sigma\,(\pi a_{cr})^{1/2}$$

Material	Temperature °F	a_{cr}(Critical Flaw Size), in.
SA106B	400	11.3
SA155	500	13.2

As in the case of the internal flaw, the surface flaw will penetrate the pipe wall without becoming critical.

14B.5.1.6.4.3 *Axial Through-Wall Crack.* The simplest formula for axial through-wall cracks is obtained when the pipe is assumed to be infinite plate; that is, the diameter is much greater than the thickness. The critical crack size for such a simple case is (Reference 12):

$$\underline{K}_{IC} = \sigma\,(\pi b_{cr})^{1/2}$$

where 2 b_{cr} is the critical crack length. The geometry is shown in Figure 14B-19, Part C. When the pipe diameter decreases, corrections are necessary. As a result of tests at Battelle Memorial Institute on SA106B piping, the critical size of the axial through-wall crack is given (Reference 13) as:

$$b_{cr} = \left\{ \frac{Rt}{1.61}\left[\left(\frac{\sigma^*}{\sigma}\right) -1\right]\right\}^{1/2}$$

where:

b_{cr} = critical half-length

σ^* = flaw stress

R = average pipe radius

t = thickness

Material	Temperature °F	Equation	2 b_{cr} (Critical Flaw Size), in.
SA106B	400	K_{IC}	28.4
SA106B	400	b_{cr}	5.5 (14-inch o.d. Schedule 80)
SA106B	400	b_{cr}	7.5 (18-inch o.d. Schedule 100)
SA155	500	K_{IC}	33.2

Material	Temperature °F	Equation	2 b_{cr} (Critical Flaw Size), in.
SA155	500	b_{cr}	9.5

Flow stress data were not available for SA155 piping. The strength of this material is higher than the strength of SA106B steel. Consequently its flow stress must be greater than the flow stress of SA106B steel. To calculate b_{cr}, the flow stress value used was based on the ratio of ultimate tensile strength of SA106B material to SA155 material.

14B.5.1.6.4.4 *Circumferential Through-Wall Crack.* It is shown (Reference 13) that the critical length of a circumferential through-wall crack is greater than the critical length of an axial crack.

14B.5.1.6.4.5 *Flow Growth.* Under the influence of cyclic loads, small defects can grow to critical size. It has been shown that an empirical expression accurately describes the flaw growth:

$$\frac{da}{dn} = C\,(\Delta K)^m$$

where $\frac{da}{dn}$ is the flaw growth rate, ΔK is the change in stress intensity factor per cycle; C and m are constants.

The following calculation (Reference 13) describes the growth of the code allowable internal and surface flaws into through-wall cracks. Since the size of these flaws is small, pipe curvature can be neglected and there is no difference between axial and circumferential flaws. Surface defects in Seismic Category I piping allowed by the code are defects with a maximum depth of 5% of the wall thickness (t). Therefore the maximum flaw depth will be 0.05t. The material constants have values of $C = 1.6 \times 10^{-4}\ in^{-1}$ and $m = 4$ (at 550°F). Note that the value of the exponent m is conservative. The exponent varies between 2 and 4 for different steels and, using its maximum value, the growth rate will be the fastest.

Integration of the previous equation gives the number of cycles:

$$n = \int_{a_i}^{a_f} \frac{da}{C(\Delta K)^4}$$

where $a_i = 0.05$ t is the initial flaw depth (the code allowable defect) and a_f is the final flaw depth. For a surface flaw, the integral becomes:

$$n = \frac{1}{C}\int_{a_i}^{a_f} \frac{da}{[1.12\sigma(\pi a)^{1/2}]^4}$$

If a_f = thickness, then n is the number of cycles to develop a through-wall crack. When the equation for n is applied to SA155 pipe, $a_i = 0.05 \times 1 = 0.05$ in. and $a_f = 1$ in. σ = yield stress at 550°F (the flaw growth will be faster at higher temperatures).

$$n = 4.13 \times 10^9 \times \frac{1}{(27.7)^4}\left(\frac{1}{0.05} - 1\right) = 132,000 \text{ cycles}$$

The additional growth of this defect to reach critical size is not important because the pipe will leak and the leak will be repaired.

It has been shown that the growth of an internal flaw is even slower than in the above case (Reference 13). The number of cycles during the lifetime of a nuclear power plant can be obtained taking into account daily and weekly power reductions, start-ups, shutdowns, and other changes in pressure. An estimate (Reference 13) gives the number of cycles at about 13,000, which is much smaller than the value for the formation of a through-wall crack.

14B.5.1.7 **Modifications**

The following modifications to the initial plant design were made to further ensure safe-shutdown reliability and the operation of plant protective features:

1. The pump discharge piping of the auxiliary feedwater systems in Units 1 and 2 were cross connected so that the unaffected system will have the capability of maintaining both units in a shutdown condition. Furthermore, an additional source of makeup water for the auxiliary feedwater systems was installed. An in-ground 100,000-gallon emergency condensate makeup tank and two booster pumps can supply the suction of the unaffected auxiliary feedwater pumps. These modifications were designed and installed in accordance with ANSI B31.1-1967, Seismic Category I criteria, and are also tornado-protected. These modifications are shown in Figure 14B-20.

   As described in Section 14B.6 one auxiliary feedwater pump is required to remove stored and residual heat. Therefore, no redundancy requirements were lost for either unit since each unit is equipped with two motor driven auxiliary feedwater pumps (350 gpm nominal flow rating) and one turbine driven auxiliary feedwater pump (700 gpm nominal flow rate).

   Since, with the modifications, the unaffected auxiliary feedwater system can be required to supply both units, the residual heat removal capacity from the original 110,000-gallon emergency condensate storage tank is halved. Another reliable source of water is the fire protection system main, which has the capability of supplying 500,000 gallon of water to the suction side of the unaffected auxiliary feedwater pumps. In addition, the 300,000-gallon condensate storage tank used for normal condensate makeup can be used to supply the 110,000-gallon emergency condensate storage tank utilizing gravity flow. The 100,000-gallon emergency condensate makeup tank was added to enhance the reliability of the modified system.

2. The turbine drivers for the containment spray pumps were disconnected from their steam supply lines and the three-inch lines were removed from the containment spray pump room. These modifications eliminated the containment spray pumps as a target.

   Since the turbines were used only as redundant pump drivers for the two full-size containment spray pumps, the pumps with their motor drivers still maintain the redundancy requirements of the containment spray system.

3. In response to an Atomic Energy Commission (AEC) inquiry concerning the ability to shut down the reactor following the postulated loss of the main steam valve house (MSVH),

Virginia Power committed to a SI system modification (Reference 16). The AEC's inquiry was made during the original review of FSAR Appendix D (currently Figure 14B). The modification installed a SI system piping cross-connect between the Unit #1 and Unit #2 RWSTs and associated valves and controls (refer to Section 6.2.2.1.4). The modification ensures a supply of RWST water to each unit's SI charging pumps in the event the normal supply line is rendered inoperable due to the postulated loss of the associated main steam valve house. It should be noted that charging pump suction piping from the RWST was not identified as a potential target for a high energy line break (HELB) in the MSVH.

14B.5.2 Feedwater

14B.5.2.1 Break Locations

Break locations were postulated in the feedwater lines from the containment to the feedwater pumps in the turbine room in accordance with Section 14B.2.2. For the feedwater lines, 0.8 of the allowable thermal stress was 18,000 psi. For each line considered, none of the calculated thermal stresses exceeded 0.8 of their allowables. Piping upstream of the containment was not analyzed seismically, so that intermediate points were selected on the basis of maximum thermal stress. For piping from the containment to the turbine building, two intermediate locations were selected. Breaks were assumed for piping in the turbine building; however, because of physical separation, no detailed analysis was required. At all break points, both circumferential and longitudinal breaks were considered.

The break points are listed in Table 14B-8 along with thermal stress levels, as calculated. The break locations are shown in Figure 14B-10.

Cracks were selected at all locations in the vicinity of targets.

The environmental impact on the adjacent EQ rooms resulting from the worst case turbine building high energy line break (HELB) have been determined. The temperatures in these rooms were calculated as a function of EQ barrier breach size. These rooms include the control room envelope, MER-5, and the emergency diesel generator rooms. The size of these breaches into the above rooms is limited based on the average internal room temperature of 120°F (see Section 7.7.1).

14B.5.2.2 Separation

The same degree of separation provided between targets and a postulated steam-line break is found for the postulated feedwater line break.

14B.5.2.3 Pipe Whip and Fluid Jet Effects

The effects of pipe whip and fluid jets from a postulated feedwater line break are similar but less severe than a main steam line break.

Table 14B-9 contains the results of the evaluation for maintaining the feedwater system functional. All the assumptions required for the main steam system as described in Section 14B.5.1.3 also apply to the feedwater system.

14B.5.2.4 **Pressure and Environment**

The main steam valve house will withstand the pressure buildup from a postulated feedwater line break, which is less than the steam-line break pressure buildup. Environmental effects are similar to the main steam line break but less severe.

14B.5.2.5 **Inspection Program for Larger Lines**

An extensive nondestructive testing program was initiated for the main feedwater postulated break points in the main steam valve house. These points, a total of eight for each unit, are shown in Figure 14B-10.

14B.5.3 Other High-Energy Lines That Maintain Maximum Operating Temperatures Greater Than 200°F and Maximum Operating Pressures Greater Than 275 psig

14B.5.3.1 **Break Locations**

Figures 14B-11, 14B-12, and 14B-13 show the break locations that were postulated for circumferential and longitudinal breaks. Postulated break locations were selected in accordance with the criteria specified in Section 14B.2.2. Tables 14B-10 and 14B-11 list the stresses in the piping systems at the postulated break locations. Designated numbers for these high-energy lines are:

1. Steam Generator Blowdown

Unit 1	Unit 2
3"-WGCB-1-601	3"-WGCB-101-601
3"-WGCB-2-601	3"-WGCB-102-601
3"-WGCB-3-601	3"-WGCB-103-601

2. Letdown from Regenerative Heat Exchanger

Unit 1	Unit 2
2"-CH-6-602	2"-CH-306-602

Cracks were postulated throughout the length of the lines for any adverse effects on targets. For computation of crack size, the diameter and wall thickness of these pipes are given in Table 14B-2.

14B.5.3.2 **Pipe Whip and Fluid Jet Effects**

The letdown line and the steam generator blowdown lines will be permitted to whip in the event of a postulated circumferential break. It has been determined by an extensive drawing review, onsite inspection, and pipe break analysis sheets that no additional restraints are required in order to meet the criteria for pipe breaks. An example of the pipe break analysis sheets for these lines is given in Table 14B-12. Target protection is maintained in the following manner:

1. By means of physical separation, including distance, building support columns, concrete walls, and larger sized pipes, and/or
2. By means of the many redundant features originally designed into the existing systems.

In all cases, the postulated pipe break will not jeopardize a safe plant shutdown.

14B.5.3.3 **Pressure and Environment**

For the high energy line break analysis, the flow from the broken letdown line was 60 gpm with an operating temperature of 287°F and a pressure of 289 psig; therefore only local effects were considered. Although the actual letdown flow and pressure may be higher than these values, the conclusions of the analysis remain bounding.

The limiting case for pressure buildup and auxiliary building environmental conditions is the break of a 3-inch steam generator blowdown line. The maximum flow rate through this line is 140 lb/sec at a maximum operating temperature of 515°F and pressure of 775 psig. Calculations were made considering the area in which the blowdown lines are located as the worst case. This area is shown in Figure 14B-11 and extends to the charging pump cubicle walls as shown in Figure 14B-12. Because of the large volume and the large vent areas, there will be negligible pressure buildup within this volume, but the temperature in this area can essentially reach 212°F if blowdown is not stopped. Therefore, temperature sensors are provided in various areas of the Auxiliary Building to provide individual temperature indication and an alarm in the main control room to alert the operators to a potential problem.

An excess flow-measuring device mitigates the consequences of a steam generator blowdown line break outside the containment. This device is located upstream of the inside containment isolation valve. If blowdown line flow exceeds a predetermined value, a signal will close the inside containment isolation valve for that blowdown line. Automatic isolation of the steam generator blowdown lines is accomplished within 30 seconds in the event of a piping break. Also, manual isolation associated with other breaks or cracks must be made within 15 minutes to meet the environmental qualification requirements of certain Class 1E components in the Auxiliary Building.

Also, excess flow is annunciated in the control room. Indication of isolation valve closure presently exists in the control room.

Detection and subsequent isolation of the affected line will inhibit the increase of temperature and humidity of the environment in the areas adjacent to postulated cracks or breaks.

14B.5.4 High-Energy Lines That Maintain a Maximum Operating Temperature of Greater Than 200°F or a Maximum Operating Pressure of Greater Than 275 psig

14B.5.4.1 Break Locations

Figures 14B-11 through 14B-15 show the locations of the high-energy lines in question. In addition, Table 14B-2 gives the maximum operating conditions of each of the lines shown. Cracks were postulated throughout the entire length of each line shown, and evaluated for any adverse effects on targets. The diameter and thickness of these pipes are given in Table 14B-2 for computation of crack size.

14B.5.4.2 Separation

Source lines not located within the confines of the areas shown in Figures 14B-10 through 14B-15 have no adverse effects on targets because of the physical separation provided by the building arrangement. For this reason, high-energy lines outside these areas were not considered as sources.

14B.5.4.3 Local Environmental Effects and Jet Impingement

In many cases, shutdown and other protective features are far enough removed from the lines in question that local effects of a postulated crack will have no effect on even the redundant features.

In other cases, target protection is maintained with the many redundant features designed into the present systems and the separation of these by means of distance, walls, and location. An example of one of the pipe break/mini-crack analysis sheets is shown in Table 14B-13.

Except for minor modifications, as discussed in Section 14B.5.4.4, the plant will always maintain shutdown capability, and at least one each of the redundant protective features will remain operable following a crack in these high-energy lines.

14B.5.4.4 Modifications

In order to ensure a safe cold shutdown, the following modifications to the initial plant design were made:

1. The charging pump cooling water pumps are shielded from direct impingement, in accordance with Figure 14B-21.
2. One of the four component cooling pump cables is rerouted (shown in Figure 14B-14), so that two of the pumps are always available (only one is required for hot standby).

14B.6 TRANSIENT ANALYSIS OF A HIGH-ENERGY LINE BREAK IN THE MAIN STEAM VALVE HOUSE

A break in a main steam or main feedwater (MFW) pipe in the main steam valve house (MSVH) could be postulated to disable the auxiliary feedwater (AFW) pumps, resulting in a loss of all feedwater on the accident unit. The main feedwater line break (MFLB) is shown to be more severe than the steam line break with respect to core cooling and steam generator inventory reduction. The MFLB in the MSVH is assumed to disable all main and auxiliary feedwater to one of the units. The only source of AFW is from the opposite unit via the AFW cross-connect. This is a limiting assumption that requires operator action within a specified time. Emergency procedures direct the operators to perform mitigating actions for this event. A further limiting assumption is that only a single motor-driven AFW pump (most degraded) is available from the other unit.

This transient is characterized by a rise in the reactor coolant system (RCS) temperature and pressure and the pressurizer water volume due to a reduction in the capability of the secondary system to remove the heat generated in the reactor core and by the reactor coolant pumps (RCPs). AFW delivery via the cross-connect becomes the critical factor for maintaining a secondary heat sink and preventing core damage. The key safety analysis parameters for this transient are the core decay heat, time of AFW cross-connect flow delivery to the accident unit's steam generators, AFW flow rate and enthalpy, and time of RCP trip.

14B.6.1 Method of Analysis

The high-energy line break event is evaluated by modeling a loss of all MFW and AFW to the affected unit and the initiation of AFW via the cross-connect by operator action as directed by the station emergency procedures. The event was explicitly analyzed at hot full power deterministic conditions. The transient analysis code RETRAN (Reference 17) was used to simulate the reactor coolant system, core kinetics, and the feedwater and steam systems. The 1979 ANS decay heat standard with a two-sigma uncertainty was used to calculate post-trip reactor core heat based on long-term operation at the initial power level preceding the trip. An assumed AFW cross-connect flow rate was selected to be conservatively bounded by the design flow rate from the most degraded AFW pump. AFW enthalpy was based on the highest allowable design temperature in the emergency condensate storage tank.

Two cases were analyzed to demonstrate the plant behavior for a loss of all feedwater event. One case provided the limiting results with respect to the fuel integrity and steam generator dryout (i.e., minimum secondary side liquid inventory) criteria. The other case provided the limiting results for the primary and secondary system overpressure criteria. Reactor trip on a low-low water level in any steam generator provides the necessary protection. Simulator verification runs were performed to provide assurance that the operators can satisfy the analysis assumptions to cross-connect AFW from the unaffected unit and trip the RCPs to maintain the secondary heat sink and to ensure a safe plant shutdown. The analyses demonstrate that the event acceptance criteria are met.

14B.6.2 Results and Conclusions

The loss of all feedwater due to a high-energy line break in the MSVH was analyzed to demonstrate a long-term increase in steam generator inventory after AFW was provided and the secondary system heat removal rate exceeded the heat production by the reactor coolant system. The effects of a high-energy line break in the MSVH are mitigated by operator action in accordance with the emergency procedures. The accident analysis meets all event acceptance criteria (no RCS bulk boiling, no steam generator dryout, peak RCS and main steam system pressures less than the limits).

14B.7 CONCLUSIONS

Surry Units 1 and 2 are designed with highly reliable and redundant systems for the purpose of safe shutdown, considering normal and accident conditions. Furthermore, with the modifications described in the text of this appendix, safe plant shutdown is ensured for all postulated failures of high-energy piping outside of the containment.

The control room, which serves Units 1 and 2, will remain habitable and functional following a failure of any high-energy line.

The emergency diesel generators, which are required to satisfy loss-of-offsite-power criteria, will maintain integrity throughout a postulated high-energy piping failure incident.

14B REFERENCES

1. A. Giambusso, Atomic Energy Commission letter, *General Information Required for Consideration of the Effects of a Piping System Break Outside Containment,* December 18, 1972.
2. General Electric Company, *System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Break,* Document No. 22A2625, Nuclear Energy Division, 1971.
3. PISCES Computer Code, Physics International Multispacial Codes for Engineering and Science, Physics International Company.
4. K. R. Wichman, A. G. Hopper, and J. L. Mershon, *Local Stresses in Spherical and Cylindrical Shells Due to External Loadings*, Welding Research Council.
5. *Building Code Requirements for Reinforced Concrete (ACI 318-71),* American Concrete Institute, 1971.
6. G. Berkhoff and E. H. Zarantonello, *Jets, Wakes, and Cavities*, Academic Press Inc., New York, 1967.

7. *CUPAT - A Computer Program to Calculate Pressure and Temperature in Various Nuclear Power Plant Compartments Resulting from a Postulated High Energy Pipe Break,* Stone & Webster Engineering Corp., 1973.

8. *LOCTIC - A Computer Code to Determine the Pressure and Temperature Response of Dry Containments to a Loss-of-Coolant Accident,* SWND-1, Stone & Webster Engineering Corp., September 1971.

9. *LOCTVS - A Computer Code to Determine the Pressure and Temperature Response of Pressure Suppression Containments to a Loss-of-Coolant Accident,* SWND-2, Stone & Webster Engineering Corp., October 1969.

10. F. J. Moody, *Maximum Two-Phase Vessel Blowdown from Pipes*, APED-4827, General Electric Co., April 20, 1965.

11. *LOCTVS - A Computer Code to Determine the Pressure and Temperature Response of Pressure Suppression Containments to a Loss-of-Coolant Accident,* SWND-2, Supplement No. 1, Stone & Webster Engineering Corp., March 1973.

12. *PVRC Recommendations on Toughness Requirements for Ferritic Materials*, WRC Bulletin 175, August 1972.

13. W. E. Senchak and O. E. Widera, *Application of Fracture Mechanics to Nuclear Piping Systems, Engineering Fracture Mechanics*, 1972, Vol. 4, P. 877.

14. H. Liebowitz, ed., *Fracture - An Advanced Treatise,* Vol. V, 211.

15. *Evaluation of the Toughness of Alternate Materials for Feedwater and Main Steam Piping at Surry and North Anna Power Stations,* Technical Report No. MT-0003, Rev. 0, Material Engineering, Nuclear Engineering Services, Virginia Power, March 1992.

16. Letter dated July 16, 1973, Serial No. 03873, from Virginia Electric and Power Company to U. S. Atomic Energy Commission, Docket Nos. 50-280 and 50-281.

17. *Vepco Reactor System Transient Analyses Using the RETRAN Computer Code*, VEP-FRD-41, Rev. 0.2-A, March 2015.

Table 14B-1
LENGTH OF PLASTIC HINGE POINT

Nominal Pipe Size, in.	Schedule	Length, in.
10	80	97.4
14	80	122.6
24	80	197.8
30	l-inch wall	167.5
32	l-inch wall	155.6
3[a]	80	77.4

Notes: 1. Carbon steel pipe (A106, Grade B).
2. Break at elbow.
3. Steam initially at 1050 psi.

a. Saturated water - 775 psig, 515°F.

Table 14B-2
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
WGCB - 101 Steam generator blowdown	3	0.300	775	515	Pipe whip, jet impingement, environmental	I	Auxiliary building
WGCB - 102 Steam generator blowdown	3	0.300	775	515	Pipe whip, jet impingement, environmental	I	Auxiliary building
WGCB - 103 Steam generator blowdown	3	0.300	775	515	Pipe whip, jet impingement, environmental	I	Auxiliary building
SHP - 133 Capped line	3	0.438	700	505	Pipe whip, jet impingement, environmental	I	Main steam valve house
SLPD - 56 Low-pressure drip line	1	0.179	15	250	Environmental, jet impingement	NA	Auxiliary building
SLPD - 50 Low-pressure drip line	4	0.237	110	344	Environmental, jet impingement	NA	Auxiliary building
SA - 21 Auxiliary steam	6	0.280	150	338	Environmental, jet impingement	NA	Auxiliary building
SA - 29 Auxiliary steam	4	0.237	150	338	Environmental, jet impingement	NA	Auxiliary building

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
SA - 31 Auxiliary steam	1	0.179	150	338	Environmental, jet impingement	NA	Auxiliary building
SHP - 101 Main steam piping - from steam generators to and including main steam trip valve	30	1.0	1005	547	Pipe whip, jet impingement, environmental	I	Containment, main steam valve house
SHP - 101 Main steam piping - from main steam trip valve to main steam manifold	30	1.0	1005	547	Pipe whip, jet impingement, environmental	NA	Main steam valve house, service building, and turbine building
SHP - 102 Main steam piping - from steam generators to and including main steam trip valve	30	1.0	1005	547	Pipe whip, jet impingement, environmental	I	Containment, main steam valve house
SHP - 102 Main steam piping - from main steam trip valve to main steam manifold	30	1.0	1005	547	Pipe whip, jet impingement, environmental	NA	Main steam valve house, service building, and turbine building

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
SHP - 103 Main steam piping - from steam generators to and including main steam trip valve	30	1.0	1005	547	Pipe whip, jet impingement, environmental	I	Containment, main steam valve house
SHP - 103 Main steam piping - from main steam trip valve to main steam manifold	30	1.0	1005	547	Pipe whip, jet impingement, environmental	NA	Main steam valve house, service building, and turbine building
SHP - 122 Main steam piping riser - to and including the safety valve manifold	30	1.0	1005	547	Pipe whip, jet impingement, environmental	I	Containment, main steam valve house
SHP - 123 Main steam piping riser - to and including the safety valve manifold	30	1.0	1005	547	Pipe whip, jet impingement, environmental	I	Main steam valve house
SHP - 124 Main steam piping riser - to and including the safety valve manifold	30	1.0	1005	547	Pipe whip, jet impingement, environmental	I	Main steam valve house

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
WFPD - 109 Steam generator feed line - inside containment to and including first isolation check valve outside containment	14	0.750	1032	450	Pipe whip, jet impingement, environmental	I	Containment, main steam valve house
WFPD - 108/109 Steam generator feed line - from feedwater manifold to but not including first isolation check valve outside containment	14	0.750	1032	450	Pipe whip, jet impingement, environmental	NA	Service building, main steam valve house
WFPD - 113 Steam generator feed line - inside containment to and including first isolation check valve outside containment	14	0.750	1032	450	Pipe whip, jet impingement, environmental	I	Containment, main steam valve house

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
WFPD - 112/113 Steam generator feed line - from feedwater manifold to but not including first isolation check valve outside containment	14	0.750	1032	450	Pipe whip, jet impingement, environmental	NA	Service building, main steam valve house
WFPD - 117 Steam generator feed line - inside containment to and including first isolation check valve outside containment	14	0.750	1032	450	Pipe whip, jet impingement, environmental	I	Main steam valve house
WFPD - 107/117 Steam generator feed line - from feedwater manifold to but not including first isolation check valve outside containment	14	0.750	1032	450	Pipe whip, jet impingement, environmental	NA	Turbine building, service building, main steam valve house

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
WFPD - 106 Steam generator feed line - feedwater manifold	18	1.156	1032	450	Pipe whip, jet impingement, environmental	NA	Service building
WFPD - 104 Steam generator feed line - feedwater pumps to feedwater manifold	18	1.156	1032	450	Pipe whip, jet impingement, environmental	NA	Turbine building, service building
CH - 391 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 313 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building
CN - 312 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 414 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
CH - 381 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 369 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 321 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 302 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 370 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH -322 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building
WFPD - 105 Steam generator feed line - feedwater pumps to feedwater manifold	18	1.156	1032	450	Pipe whip, jet impingement, environmental	NA	Turbine building, service building

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
CH - 306 Chemical and volume control - letdown from regenerative heat exchanger	2	0.154	289	287	Pipe whip, jet impingement, environmental	I	Auxiliary building
CH - 380 Chemical and volume control - charging header	4	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 389 Chemical and volume control - charging header	4	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 379 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CN - 311 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 308 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
CH - 390 Chemical and volume control	2	0.343	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 303 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
CH - 371 Chemical and volume control	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
SI - 347 Safety injection charging pump - to and including the first isolation valve	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building
SI - 257 Safety injection charging pump - to and including the first isolation valve	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building

a. NA - not applicable

Table 14B-2 (CONTINUED)
HIGH-ENERGY LINES SOURCES

Service	Nominal Line Size, in.	Wall Thickness, in.	Maximum Operating Pressure, psig	Maximum Operating Temperature, °F	Pipe Break Effects Evaluated	Seismic Classification[a]	Location
SI - 272 Safety injection charging pump - to and including the first isolation valve	3	0.438	2500	130	Environmental, jet impingement	I	Auxiliary building

a. NA - not applicable

Table 14B-3
POSTULATED TARGETS

System	Major Equipment	Mark No.	Location
Auxiliary feedwater	Auxiliary feed pumps	2-FW-P-2 2-FW-P-3A 2-FW-P-3B	Main steam valve house (MSVH)
	Auxiliary feed pump oil coolers (1 per feed pump)	2-FW-E-7 2-FW-E-8 2-FW-E-9	MSVH
	Auxiliary feed check valves (1 per feed pump)	2-FW-142 2-FW-157 2-FW-172	MSVH
Chemical and volume control	Charging pumps	2-CH-P-1A 2-CH-P-1B 2-CH-P-1C	Auxiliary building
	Boric acid tanks	1-CH-TK-1A l-CH-TK-1B 1-CH-TK-1C	Auxiliary building
	Boric acid tank heaters	1-CH-E-6A 1-CH-E-6B 1-CH-E-6C	Auxiliary building
	Boric acid transfer pumps	1-CH-P-2C 1-CH-P-2D	Auxiliary building
	Boric acid isolation valves	MOV-2350	Auxiliary building
	Volume control tank isolation check valve	2-CH-230	Auxiliary building
	Cold-leg isolation valves (normal charging)	MOV-2289A MOV-2289B	Auxiliary building
	Charging pump discharge valves	MOV-2286A MOV-2286B MOV-2286C MOV-2287A MOV-2287B MOV-2287C	Auxiliary building
	Reactor coolant pump seal isolation valve	MOV-2370	Auxiliary building
	Refueling water storage tank isolation valves	LCV-2115B LCV-2115D	Auxiliary building

Table 14B-3 (CONTINUED)
POSTULATED TARGETS

System	Major Equipment	Mark No.	Location
Chemical and volume control (continued)	Alternate charging paths - isolation valves (redundancy)	FCV-2160 MOV-2867A MOV-2867B MOV-2867C MOV-2867D MOV-2869A MOV-2869B MOV-2842	Auxiliary building
	Charging pump discharge pressure (not required because of the availability of pressurizer level)	PT-2121	Auxiliary building
	Charging pump flow transmitter (not required because of the availability of pressurizer level)	FT-2122	Auxiliary building
	Flow control valve for the charging pump (fails open)	FCV-2122	Auxiliary building
Main feedwater	Main feedwater isolation check valves (3)	2-FW-12 2-FW-43 2-FW-74	MSVH
Main steam	Main steam nonreturn valves	NRV-MS201A NRV-MS201B NRV-MS201C	MSVH
	Main steam trip valves	TV-MS201A TV-MS201B TV-MS201C	MSVH
	Auxiliary feed pump steam isolation valve	PCV-MS202A PCV-MS202B (F/open)	MSVH MSVH
	Main steam safety valves (required only for main steam line rupture)	SV-MS201A SV-MS201B SV-MS201C SV-MS202A SV-MS202B SV-MS202C SV-MS203A SV-MS203B SV-MS203C	MSVH
	Main steam power operated relief valves (control cooling of the RCS)	RV-MS201A RV-MS201B RV-MS201C	MSVH

Table 14B-3 (CONTINUED)
POSTULATED TARGETS

System	Major Equipment	Mark No.	Location
Component cooling system	Component cooling water pumps	l-CC-P- lA 1-CC-P-1B 1-CC-P-1C 1-CC-P-1D	Auxiliary building
	Charging pump cooling water pumps	2-CC-P-2A 2-CC-P-2B	Auxiliary building
	Charging pump lube-oil coolers	2-CH-E-5A 2-CH-E-5B 2-CH-E-5C	Auxiliary building
	Charging pump seal coolers	2-CH-E-7A 2-CH-E-7B 2-CH-E-7C 2-CH-E-7D 2-CH-E-7E 2-CH-E-7F	Auxiliary building
	Flow transmitters, component cooling pump discharge (Unit 2)	FT-CC200A FT-CC200B	Auxiliary building

Table 14B-4
MAIN STEAM BREAK LOCATIONS AND STRESSES

Line Designation	Break Point	Thermal Stress, psi	Combined Stress, psi	Description
30-SHP-101	1	3850	15,937	Terminal - containment
	79	2560	11,727	Terminal - manifold
	3	6680	19,322	Intermediate - elbow valve house
	37	5330	14,718	
	40	5670	15,068	Intermediate location - bend
30-SHP-102	135	4380	17,484	Terminal - containment
	206	1965	10,335	Terminal - manifold
	138	7355	20,671	Intermediate - elbow valve house
	163	5690	15,116	
	166	6060	15,240	Intermediate location - bend
30-SHP-103	275	4365	17,528	Terminal - containment
	341	1180	9508	Terminal - manifold
	277	7210	18,257	Intermediate - elbow valve house
	309	6565	15,485	
	305	5990	14,910	Intermediate location - bend

Table 14B-5
FEEDWATER SYSTEMS AVAILABILITY AFTER MAIN STEAM PIPE BREAK IN VALVE HOUSE

Break No.	Location	Main Feed	Auxiliary Feed		Auxiliary Feed Cross-Connect
			Turbine	Motors (2)	
C1, C135, C275	Containment	0	0	X-S	0
C2, C136, C276, C461	Riser	0	0	X-S	0
C3, C138, C277	Elbow	0	0	X-S	0
L1, L135, L275	Containment	0	0	X-S	0
L2, L136, L276, L461	Riser	(0)-J	0	X-S	0
L3, L138, L277	Elbow (vertical)	0	X-J	X-S	0
	Elbow (lateral)	0	0	X-S	0

Key: 0 = operable
X = inoperable
(0) = 2 lines operable
W = whip damage
J = jet damage
S = steam damage
C = circumferential
L = longitudinal

Table 14B-6
JET IMPINGEMENT ON WALLS - FORCE = 619 kips

Target	Impingement Area, ft^2	Jet Pressure, psig	Wall Thickness, in.	Min. Wall-Punch Shear, in.
Floor	44.90	95.7	9	22
Roof	142.05	30.2	24	24
Containment (N)	10.23	420.0	54	54
Shield (S)	102.63	41.9	36	36
East wall	18.85	228.1	24	24
West wall	42.23	102.1	24	24
Shield (S)	26.97	159.4	36	36

Table 14B-7
JET IMPINGEMENT ON VALVES

Valves	Pt Jet Pressure, psig	At Impingement Area, in^2	Incident Angle	Fv Normal Force-kips
NRV/MS201B	74.6	3927	0	175
TV/MS201B	126.3	2182	0	165
SV/MS203A	37.8	673	60	3.8
SV/MS201A	37.8	312	60	1.8
RV/MS201A	34.2	200	25	2.8

Table 14B-8
FEEDWATER BREAK LOCATIONS AND STRESSES

Line Designation	Break Point	Thermal Stress, psi	Description
14-WFPD-117	1	1565	Terminal - containment
	76	6563	Terminal - manifold
	72	5837	Intermediate - valve
	4	5383	Intermediate - elbow valve house
14-WFPD-113	101	2393	Terminal - containment
	100	4722	Terminal - manifold
	107	5333	Intermediate - elbow valve house
	171 172	5466 5046	Intermediate - at valve
14-WFPD-109	241	2860	Terminal - containment
	195 210	4200 4169	Terminal - manifold
	247 249	11,171 11, 962	Intermediate - elbow valve house
	244 246	8814 7989	Intermediate - elbow valve house
18-WFPD-104	90	3539	Terminal - manifold
	93	5333	Terminal - 2-PW-E1A
	108	4798	Intermediate - bend
	95	6368	Intermediate - bend - valve
18-WFPD-105	190	5209	Terminal - manifold
	14	3260	Terminal - 2-PW-E1B
	48	3220	Intermediate - bend
	16	4460	Intermediate - bend - valve

Table 14B-9

FEEDWATER SYSTEMS AVAILABILITY AFTER MAIN FEEDWATER PIPE BREAK IN VALVE HOUSE

			Auxiliary Feed		
Break No.	Location	Main Feed	Turbine	Motors (2)	Auxiliary Feed Cross-Connect
C5, C6	Containment or elbow	(0)	0	X-S	0
C101, C241, C244	Containment or elbow	(0)	X-W	X-S	0
C107, C247, C249	Elbow	(0)	0	X-S	0
L5, L101, L241	Containment	(0)	0	X-S	0
L6, L244	Elbow	(0)	0	X-S	0
L107	Elbow	(0)	0	X-S	0
L247, L249	Elbow	(0)	0	X-S	0

Key: 0 = operable
X = inoperable
(0) = 2 lines operable
W = whip damage
J = jet damage
S = steam damage
C = circumferential
L = longitudinal

Table 14B-10
STEAM GENERATOR BLOWDOWN BREAK LINE LOCATIONS AND STRESSES

Line Designation	Break Point	Thermal Stress, psi	Description
		Unit 1	
3-WGCB-1	60	3952	Terminal - containment
	61	4991	Intermediate - elbow
3-WGCB-2	63	1414	Terminal - containment
	66	4411	Intermediate - elbow
3-WGCB-3	67	3251	Terminal - containment
	70	4942	Intermediate - elbow
		Unit 2	
3-WGCB-101	22	3952	Terminal - containment
	21	4991	Intermediate - elbow
3-WGCB-102	23	1414	Terminal - containment
	26	4411	Intermediate - elbow
3-WGCB-103	27	3251	Terminal - containment
	30	4942	Intermediate - elbow

Table 14B-11
LETDOWN LINE FROM REGENERATIVE HEAT EXCHANGER
BREAK LOCATIONS AND STRESSES

Line Designation	Break Point	Thermal Stress, psi	Description
		Unit 1	
2"- CH-6-602	238	5102	Terminal - anchor
	236	17,175	Intermediate - elbow
	71	2487	Terminal - containment
	73	2727	Elbow
	74	2774	Elbow
	64	90	Anchor
		Unit 2	
2"- CH-306-602	38	5102	Terminal - anchor
	36	17,175	Intermediate - elbow
	35	2487	Terminal - containment
	32	2727	Elbow
	33	2774	Elbow
	95	90	Anchor

Table 14B-12
PIPE BREAK/MINI-CRACK ANALYSIS SUMMARY

Stone & Webster Engineering Corporation
Calculation Sheet
Page No. 6

VEPCO — SURRY NO. 2
3-21-73
Checked 3-22-73
ONSITE INSPECTION

REF DWG No:
MSK No:

PIPE BREAK/ MINI-CRACK ANALYSIS SUMMARY

* CB - CIRCUMFERENTIAL BREAK; LB - LONGITUDINAL BREAK; MC - MINI CRACK (1/2 OD x 1/2 THICK)

** NR - NOT REQUIRED; R - REQUIRED

Max.Oper.Press. 775 psig
Max.Oper.Temp. 515 F

			PIPE WHIP			PIPE JET					
LINE	POINT	FAILURE TYPE *	BLOWDOWN FORCE (KIPS)	RESTRAINT **	DESCRIPTION AND COMMENTS	TARGET	TARGET DISTANCE (FT)	JET FORCE ON TARGET (KIPS)	BARRIER **	DESCRIPTION AND COMMENTS	RECOMMENDATIONS
MOCR-102	26	CB	7.3		Column TH-9' prevents the pipe from whipping in the eastward direction. For movement in other directions the hinge length of 77.4 in. was used to determine "target" damage. There were no "targets" within this distance. The normal charging line 3"-CH-379-1503 and the normal seal injection line for the reactor coolant pumps 3"-CH-311-1503 will not be damaged because of the criteria for whipping pipes. Line 2"-CH-390-1503 is only an alternate seal injection line to the reactor coolant pumps, and is therefore not required.	MOV-2289A MOV-2289B	10 13			Possible damage from direct impingement. These are redundant isolation valves which are normally open and will fail open. Furthermore this line can be isolated at the charging pumps, if necessary. The line which includes these valves is the normal charging path for chemical and volume control. An alternate path is 2"-CH-308-1503 (located at drawing coordinate E-2), could be used to shut the unit down in a normal manner. Therefore, there are no adverse effects on safe shutdown and no analysis on the effects of the two valves is required.	Plant must be shut down and repairs made.
						MOV-2370 MOV-2867C MOV-2867D	8 15 18			Valves are protected from jet impingement by column TH-9', distance, and location. Failure by fluid jets is therefore not credible. The power cables to these mot r operated valves could be severed. MOV-2370 is an isolation valve for isolation of normal seal water injection to the reactor coolant pump seals. This valve will fail as is (open). Valve operation is not required for safe shutdown. MOV-2867C & D are redundant isolation valves which require operation only in the event of a loss-of-coolant accident (13A) or a major main steam line rupture. Therefore, there are no adverse effects on safe plant shutdown.	Plant must be shut down and repairs made.

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Table 14B-13
PIPE BREAK/MINI-CRACK ANALYSIS SUMMARY

STONE & WEBSTER ENGINEERING CORPORATION
CALCULATION SHEET

Page No. 73

Client VEPCO   Location SURRY NO. 2   J.O. No. 12358
Date 3-21-73
Checked 3-22-73
Based on ON-SITE INSPECTION

REF DWG No:
MSK No:

PIPE BREAK/MINI-CRACK ANALYSIS SUMMARY

* CB - CIRCUMFERENTIAL BREAK
LB - LONGITUDINAL BREAK
MC - MINI CRACK (1/2 OD x 1/2 THICK)

** NR - NOT REQUIRED
R - REQUIRED

Max Oper. Press. 150 psig
Max Oper. Temp 338 °F

			PIPE WHIP			PIPE JET					
LINE	POINT	FAILURE TYPE *	BLOWDOWN FORCE (KIPS)	RESTRAINT **	DESCRIPTION AND COMMENTS	TARGET	TARGET DISTANCE (FT)	JET FORCE ON TARGET (KIPS)	BARRIER **	DESCRIPTION AND COMMENTS	RECOMMENDATIONS
BA-29 4"	#26	MC	None	NR	Line only analyzed for environmental effects.	CC-P-1A CC-P-1B CC-P-1C	0.5 0.5 0.5	NR NR NR	NR NR NR	Power cables for these component cooling pumps could be severed or permanently damaged. The fourth component cooling pump will not be affected. One pump is sufficient to maintain both units in a hot shutdown condition. Since the single active failure criteria must be applied; see Recommendations to assure a safe shutdown.	Reroute one of the three component cooling pump cables; so that the most adverse crack could only severe or permanently damage only two pump cables.
						CH-P-1A	4	NR	NR	If crack is moved to the most adverse location and orientation, there may be possible damage on the power cable to this charging pump. Since there are two other pumps available and only one is required, safe shutdown is still assured.	NR
						FCV-1113A	11	NR	NR	Because of the low steam line pressure and distance, valve is probably not subject to environmental failure. Of course, this valve is normally open and fails open, so even a failure of this valve will not effect safe shutdown. Valve used for normal boric acid transfer to charging pumps.	NR
										Point 26 denotes the area surround TW16 on Fig. D.4-3. Cracks are assumed anywhere in this area.	

S14B0025

Table 14B-13 (CONTINUED)
PIPE BREAK/MINI-CRACK ANALYSIS SUMMARY

STONE & WEBSTER ENGINEERING CORPORATION
CALCULATION SHEET

Page No. P76
Preliminary
Client VEPCO | Location SURRY NO. 2 | J.O. No. 12354
Subject or Apparatus | Date 3-21-73
Checked 3-22-73
Based on ON-SITE INSPECTION | Revised

* CB - CIRCUMFERENTIAL BREAK
LB - LONGITUDINAL BREAK
MC - MINI CRACK (1/2 OD x 1/2 THICK)

** NR - NOT REQUIRED
R - REQUIRED

PIPE BREAK/MINI-CRACK ANALYSIS SUMMARY

REF DWG No:
MSK No:

Max. Oper. Press. 150 psig
Max. Oper. Temp 338°F

			PIPE WHIP			PIPE JET					
LINE	POINT	FAILURE TYPE *	BLOWDOWN FORCE (KIPS)	RESTRAINT **	DESCRIPTION AND COMMENTS	TARGET	TARGET DISTANCE (FT)	JET FORCE ON TARGET (KIPS)	BARRIER **	DESCRIPTION AND COMMENTS	RECOMMENDATIONS
SA-29 4"	#26	MC	None	NR	Line only analyzed for environmental effects.	MOV-1350	12	NR	NR	Because of low steam line pressure and distance, valve is probably not subject to environmental failure. Of course, this valve is only an isolation valve for one of the many alternate methods for boric acid feed to the charging pumps. Therefore, failure will not effect a safe shutdown.	NR
						CH-E-6A	8	NR	NR	Because of low steam line pressure and distance, failure of the power cable to this heater is not likely. If failure occurred, there is a redundant power cable for this heater north of the boric acid tank in question. Furthermore, boric acid tanks 1-CH-TK-1B and 1-CH-TK-1C are available backup sources of boric acid. Therefore, failure will not effect a safe shutdown.	NR
										Point 25 denotes the area surrounding SN16 on Fig. D.4-3. Cracks are assumed anywhere in this area.	

S14B0026

Figure 14B-1

APPENDIX ORGANIZATION—EFFECTS OF HIGH ENERGY PIPING SYSTEM BREAK OUTSIDE CONTAINMENT

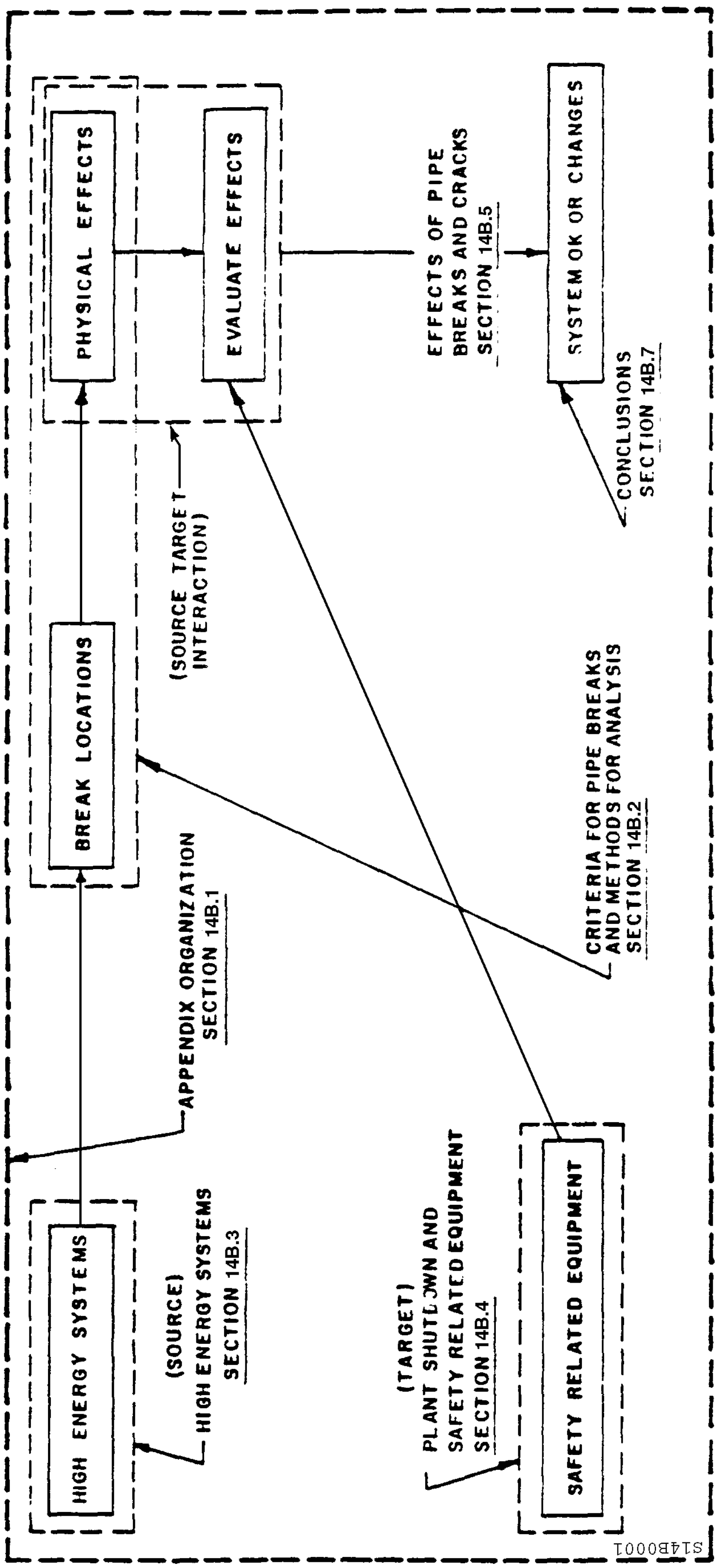


Figure 14B-2
PIPE SPLIT, CRACK AND BREAK ANALYSIS REQUIRED FOR HIGH ENERGY PIPING

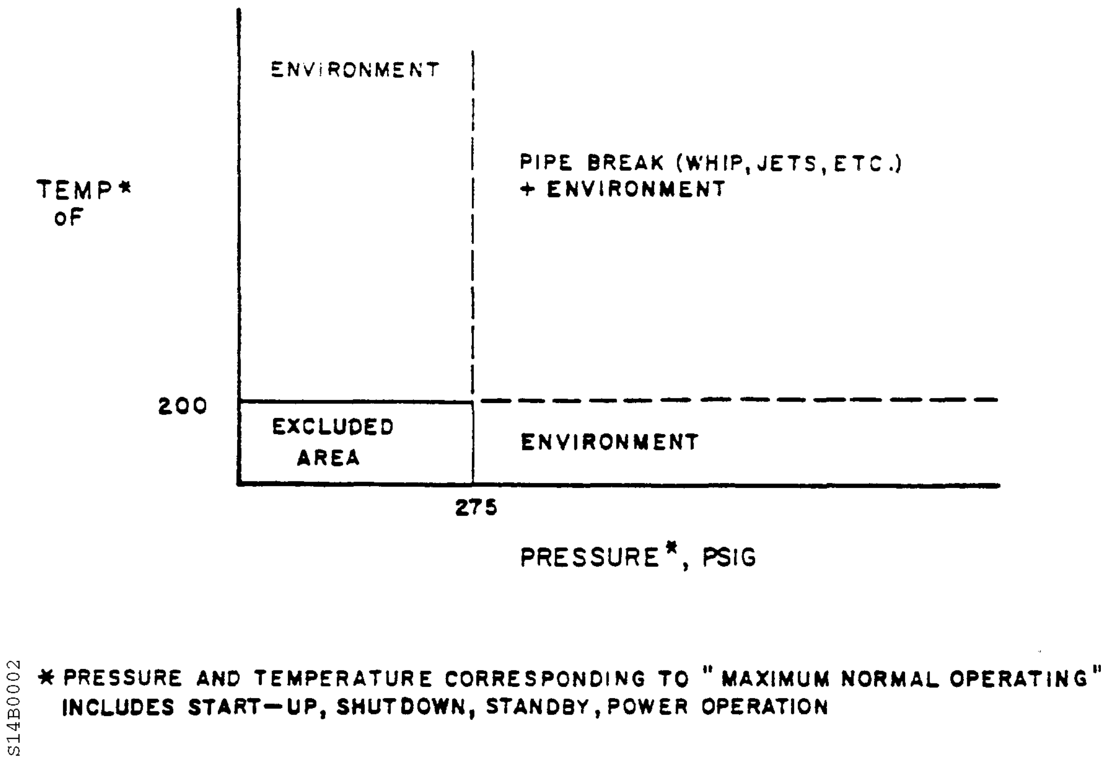
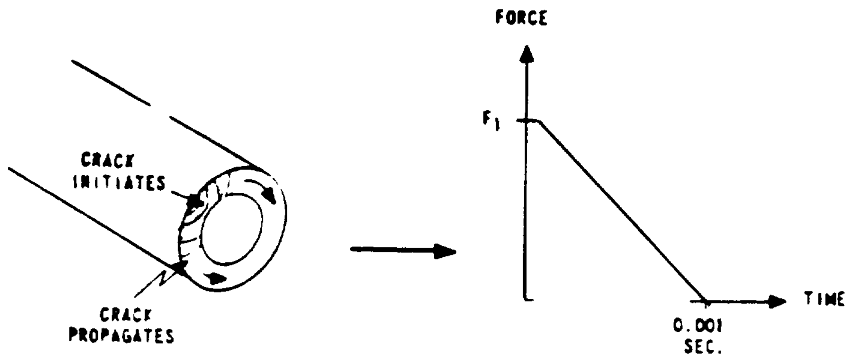
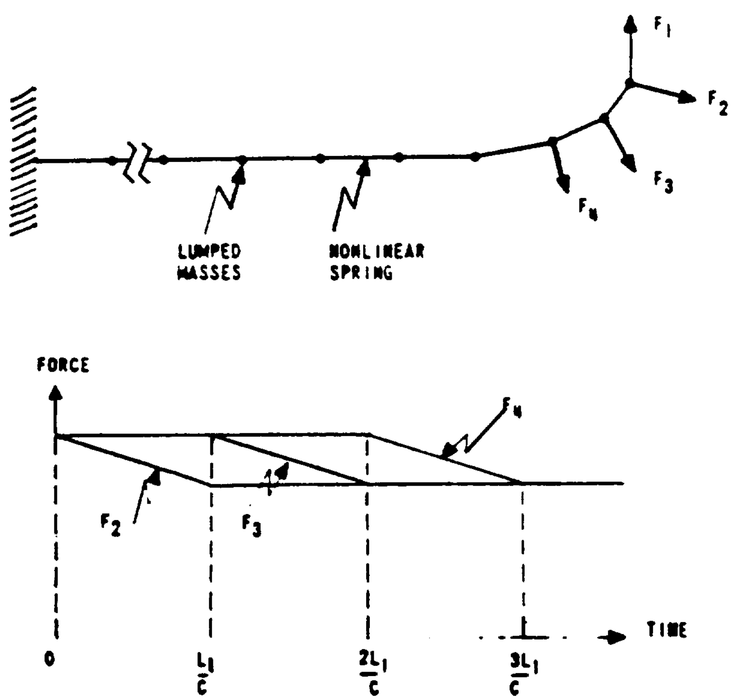


Figure 14B-3
MATHEMATICAL MODEL AND FORCING FUNCTIONS

PART A TIME DEPENDENCE OF MECHANICAL FORCE AT BREAK



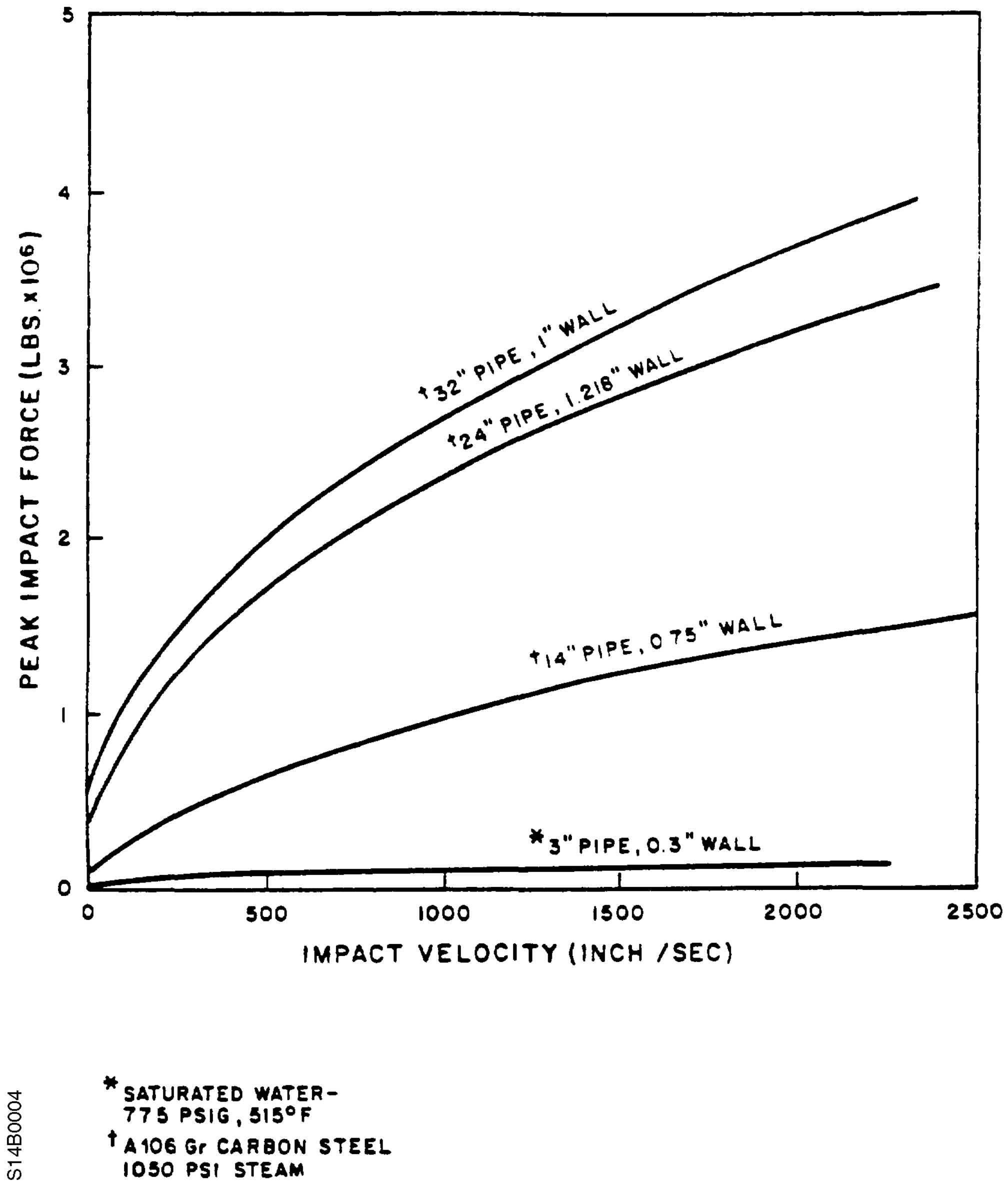
PART B TIME DEPENDENCE OF FLUID FORCES



NOTE:
L_1 = 1/3 X CENTERLINE LENGTH ALONG ELBOW
C = SPEED OF SOUND

S14B0003

Figure 14B-4
FORCE DUE TO PIPE IMPACT (TYPICAL EXAMPLES)



* SATURATED WATER-
775 PSIG, 515°F
† A106 Gr CARBON STEEL
1050 PSI STEAM

Figure 14B-5
PUNCHING SHEAR IN CONCRETE WALL

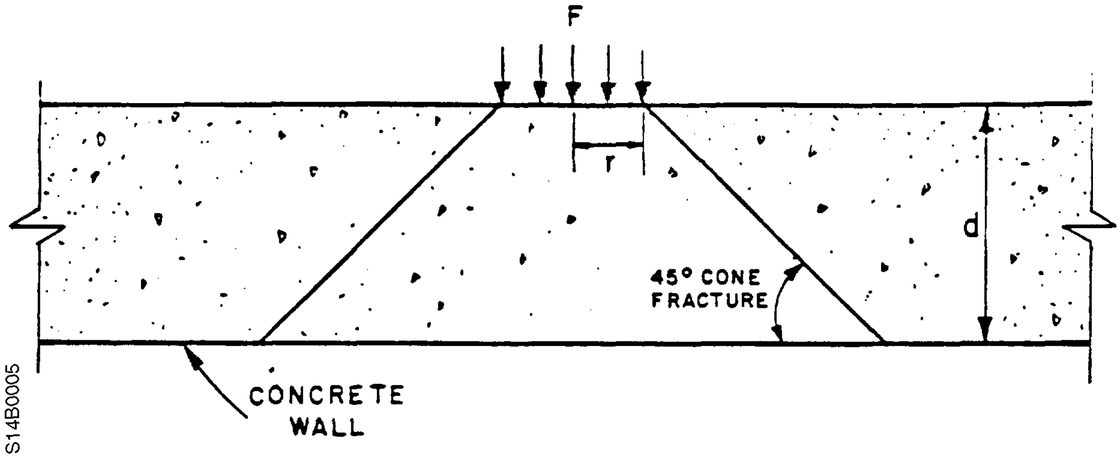
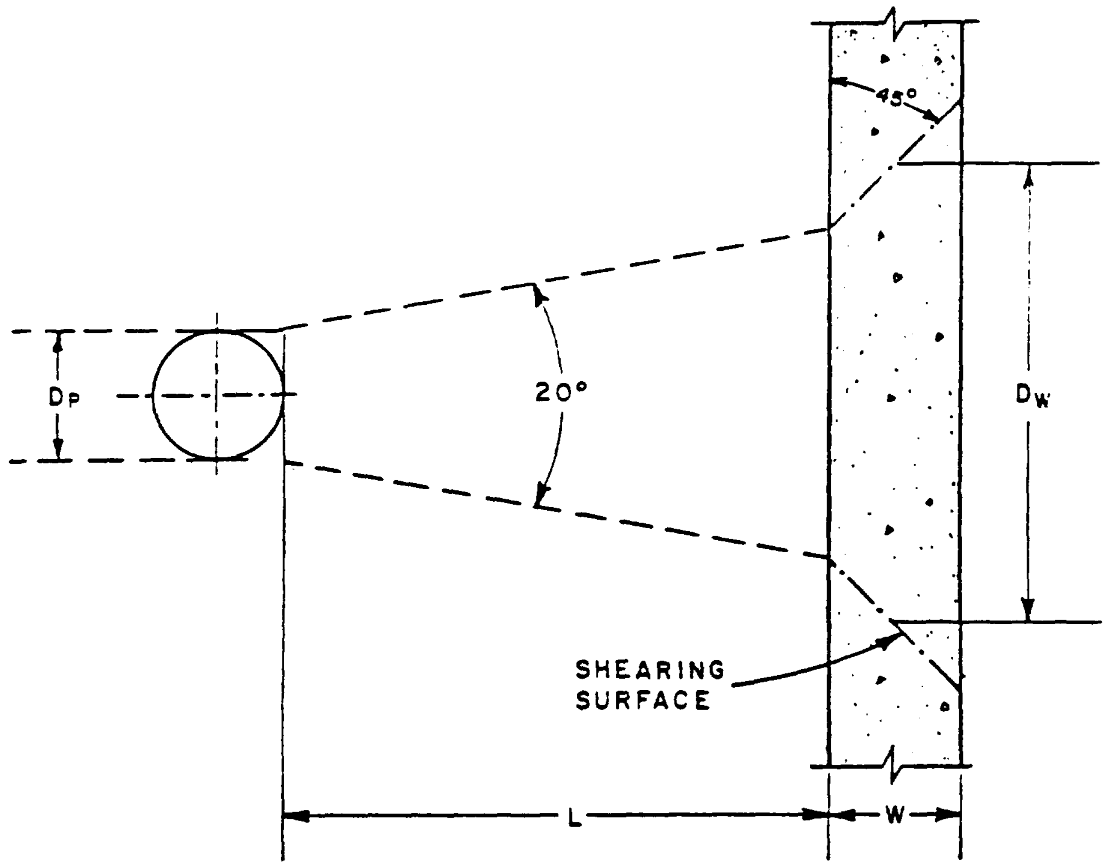


Figure 14B-6
PUNCH SHEAR FAILURE OF CONCRETE WALL



NOTE:
THIS ANALYSIS IS BASED ON THE DOUBLE ENDED BREAK MODEL WHICH GIVES MORE CONSERVATIVE RESULT THAN THE LONGITUDINAL BREAK MODEL DOES.

S14B0006

Figure 14B-7
PUNCH SHEAR OF REINFORCED CONCRETE WALL DUE TO JET IMPINGEMENT FROM STEAM PIPE BREAK

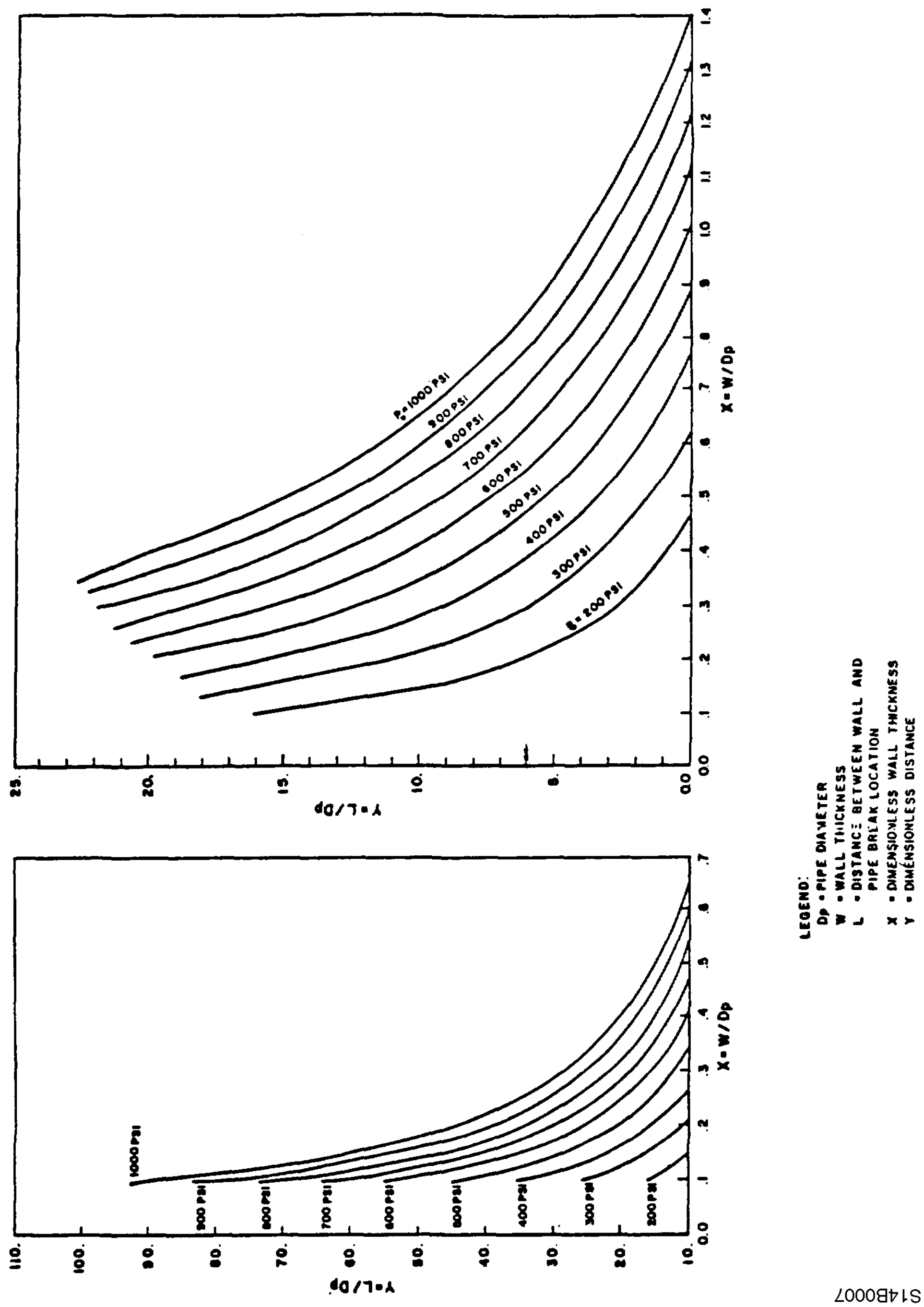
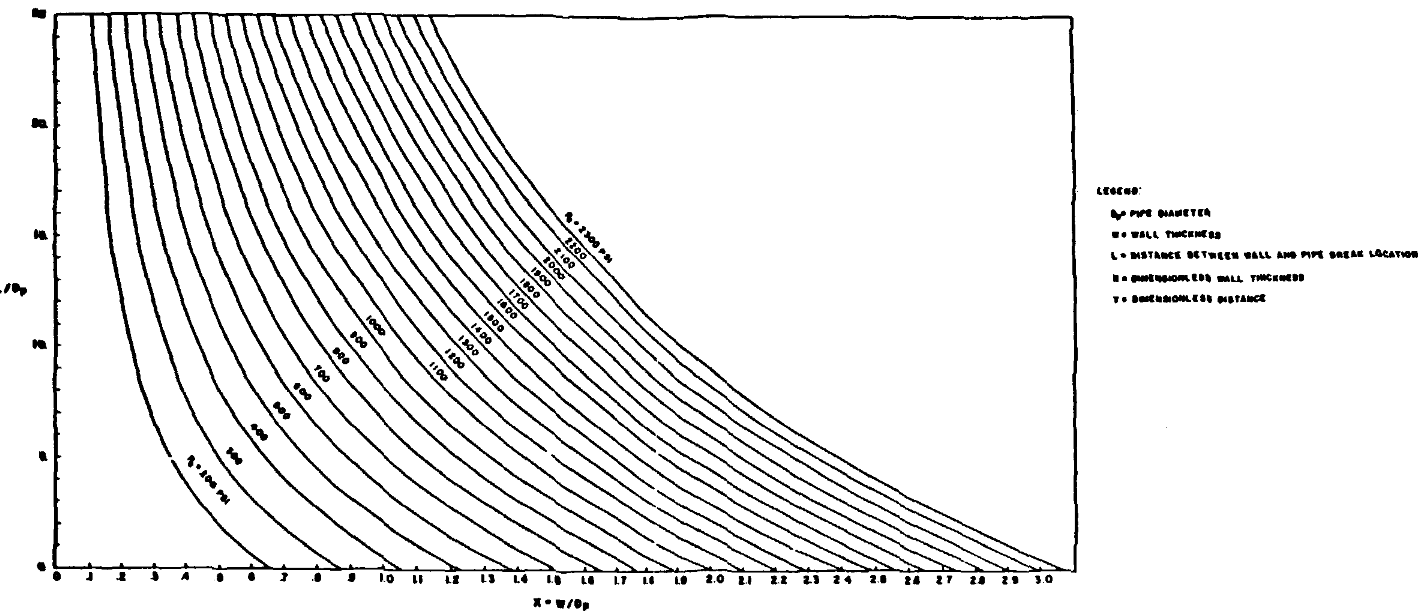
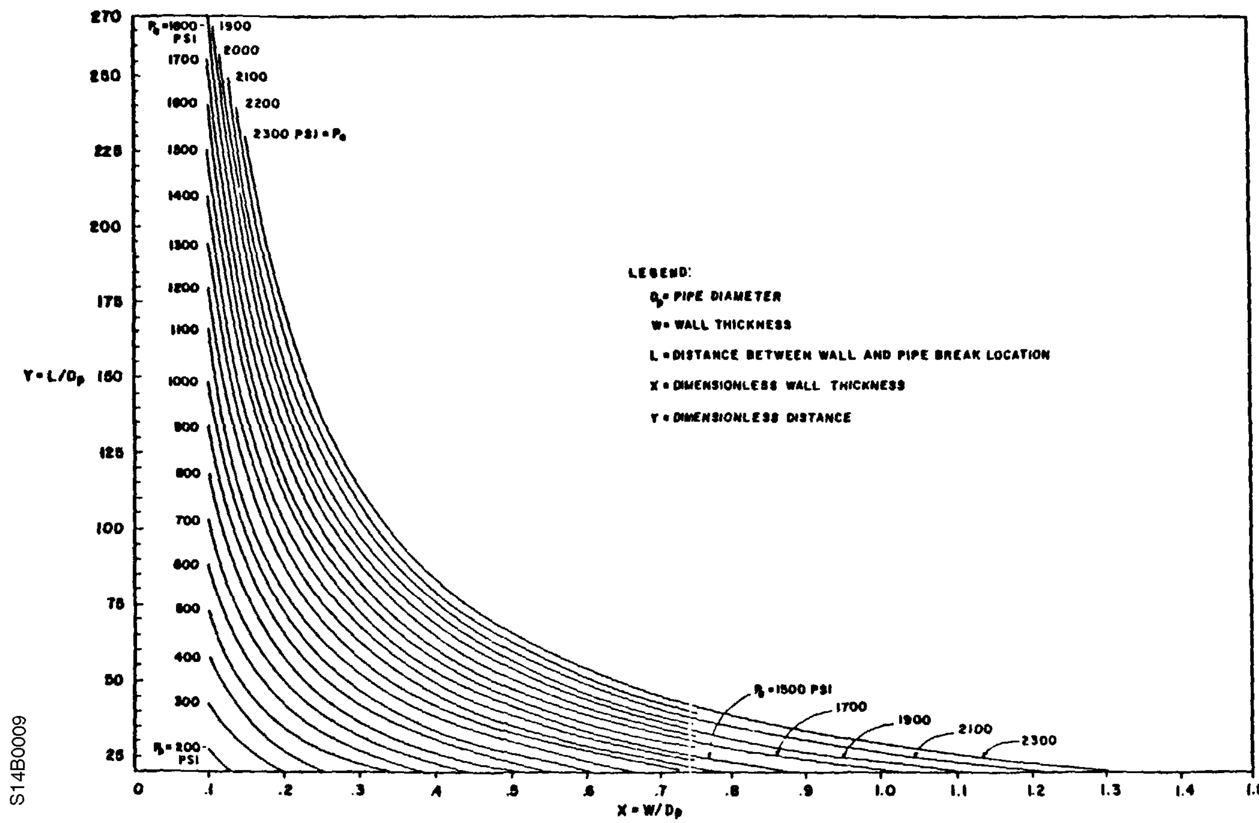


Figure 14B-8 (SHEET 1 OF 2)
PUNCH SHEAR OF REINFORCED CONCRETE WALL DUE TO JET IMPINGEMENT
FROM WATER PIPE BREAK



S14B0008

Figure 14B-8 (SHEET 2 OF 2)
PUNCH SHEAR OF REINFORCED CONCRETE WALL DUE TO JET IMPINGEMENT FROM WATER PIPE BREAK



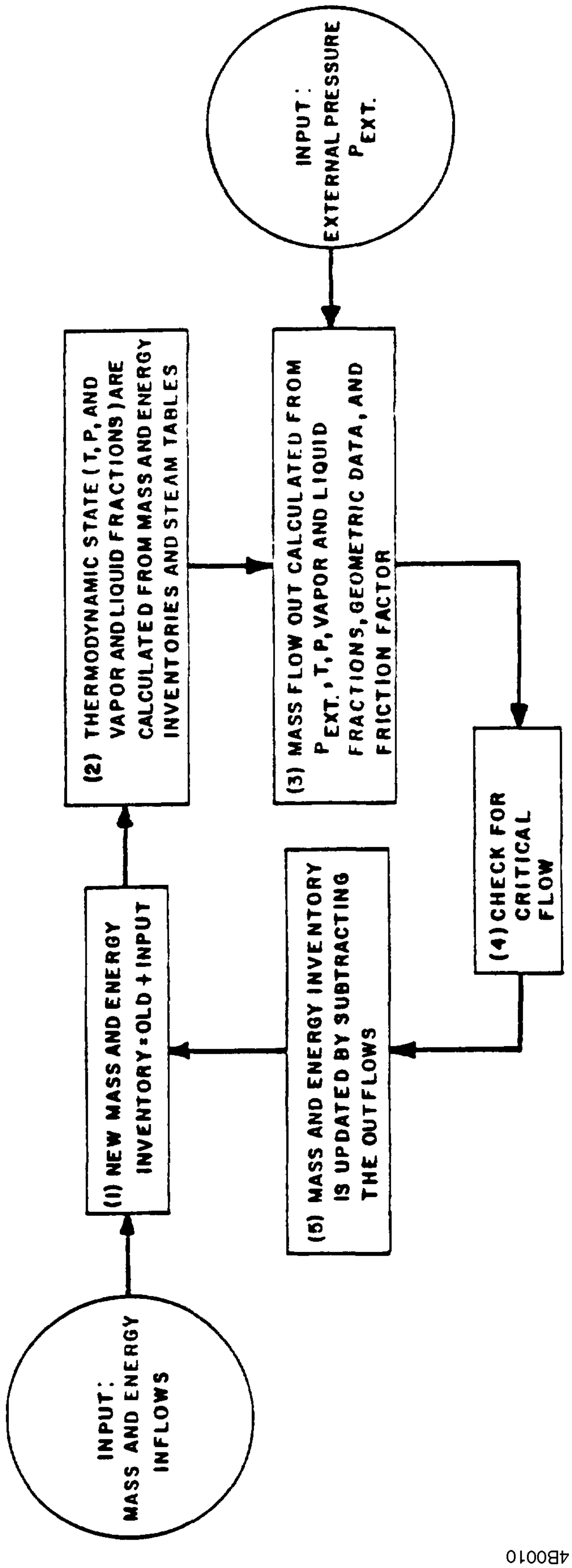
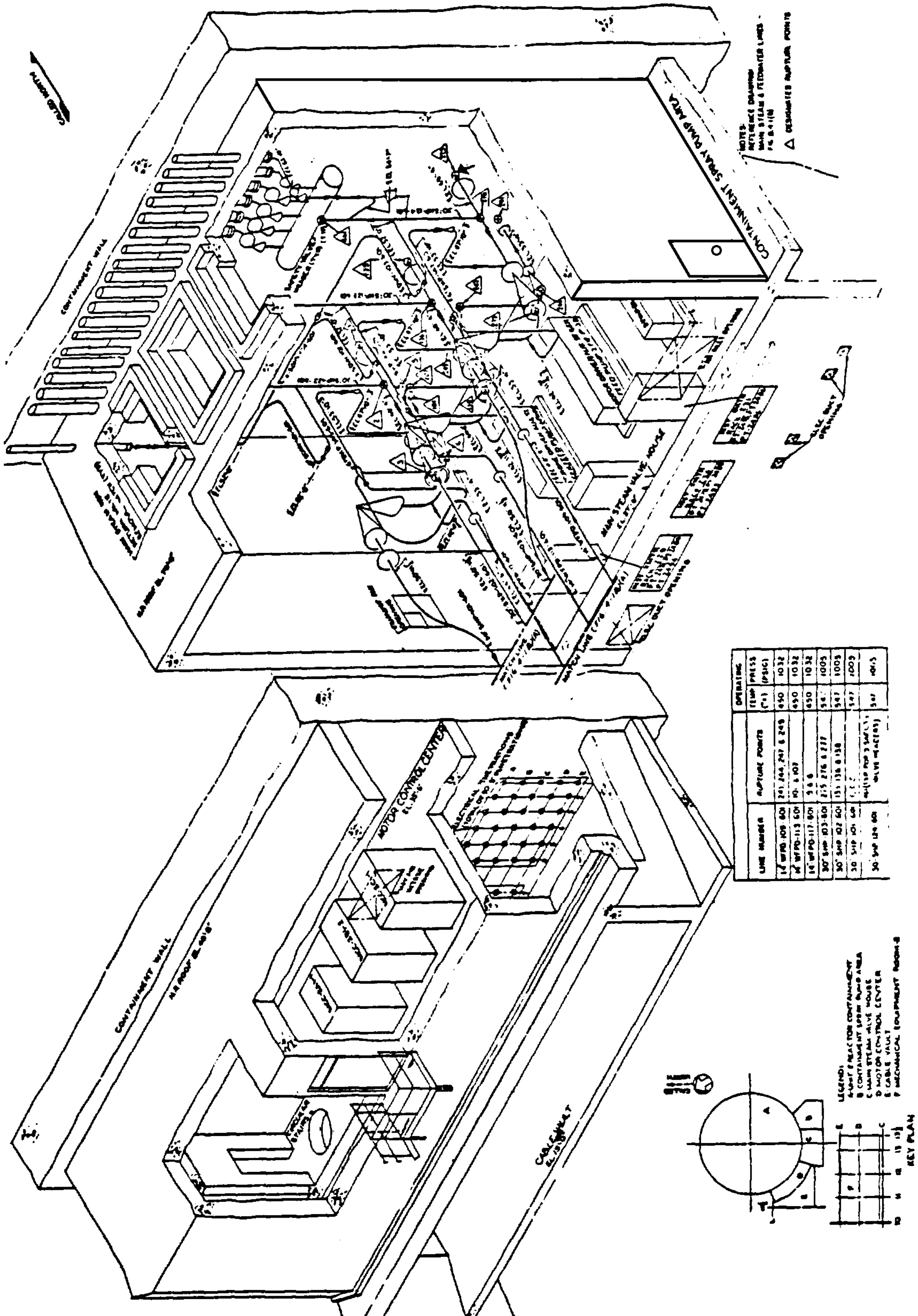


Figure 14B-9
CUPAT LOGIC DIAGRAM

Figure 14B-10 (SHEET 1 OF 2)
MAIN STEAM AND FEEDWATER LINES-UNIT 2



S14B0011

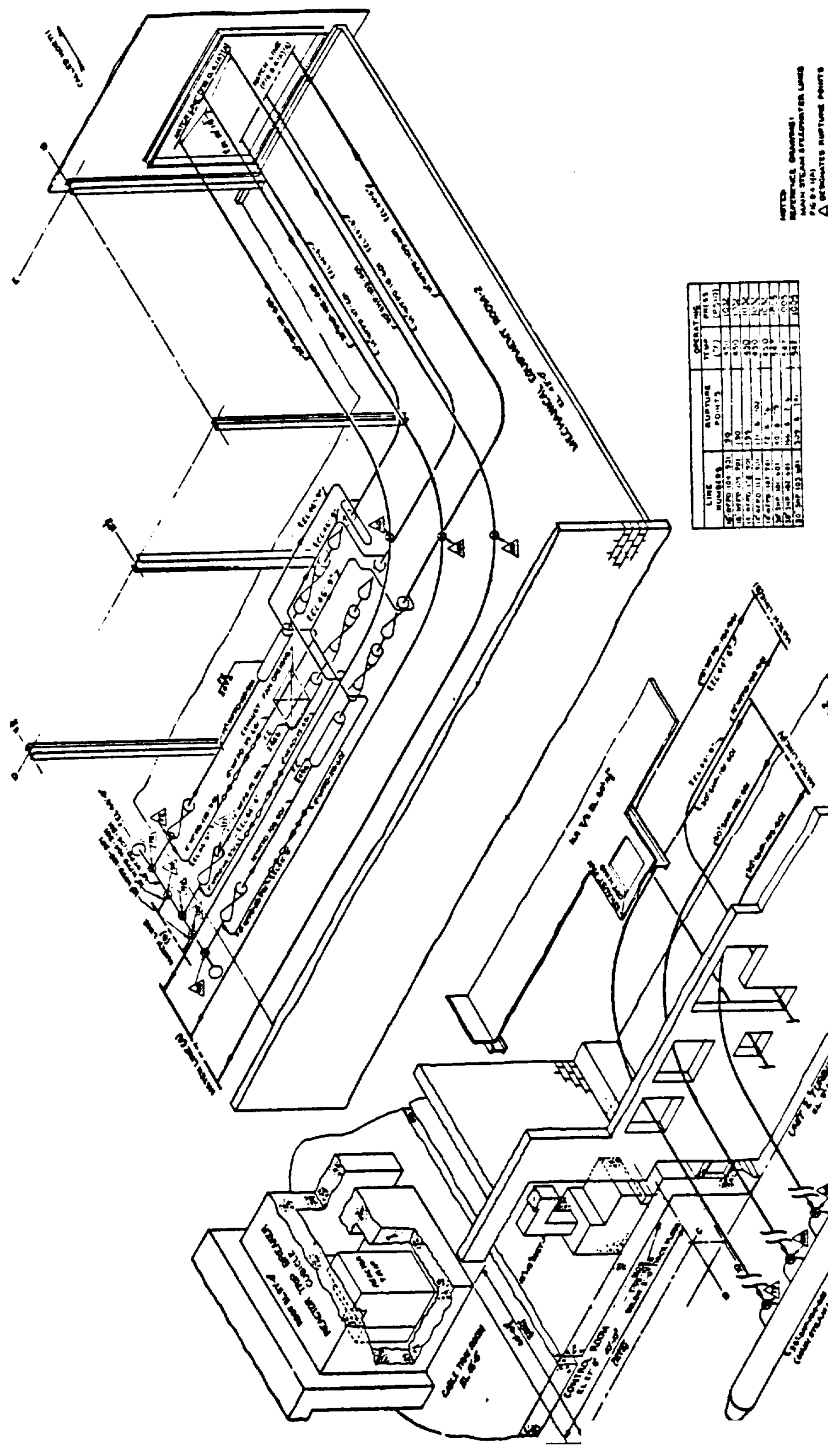


Figure 14B-10 (SHEET 2 OF 2)
MAIN STEAM AND FEEDWATER LINES-UNIT 2

S14B0012

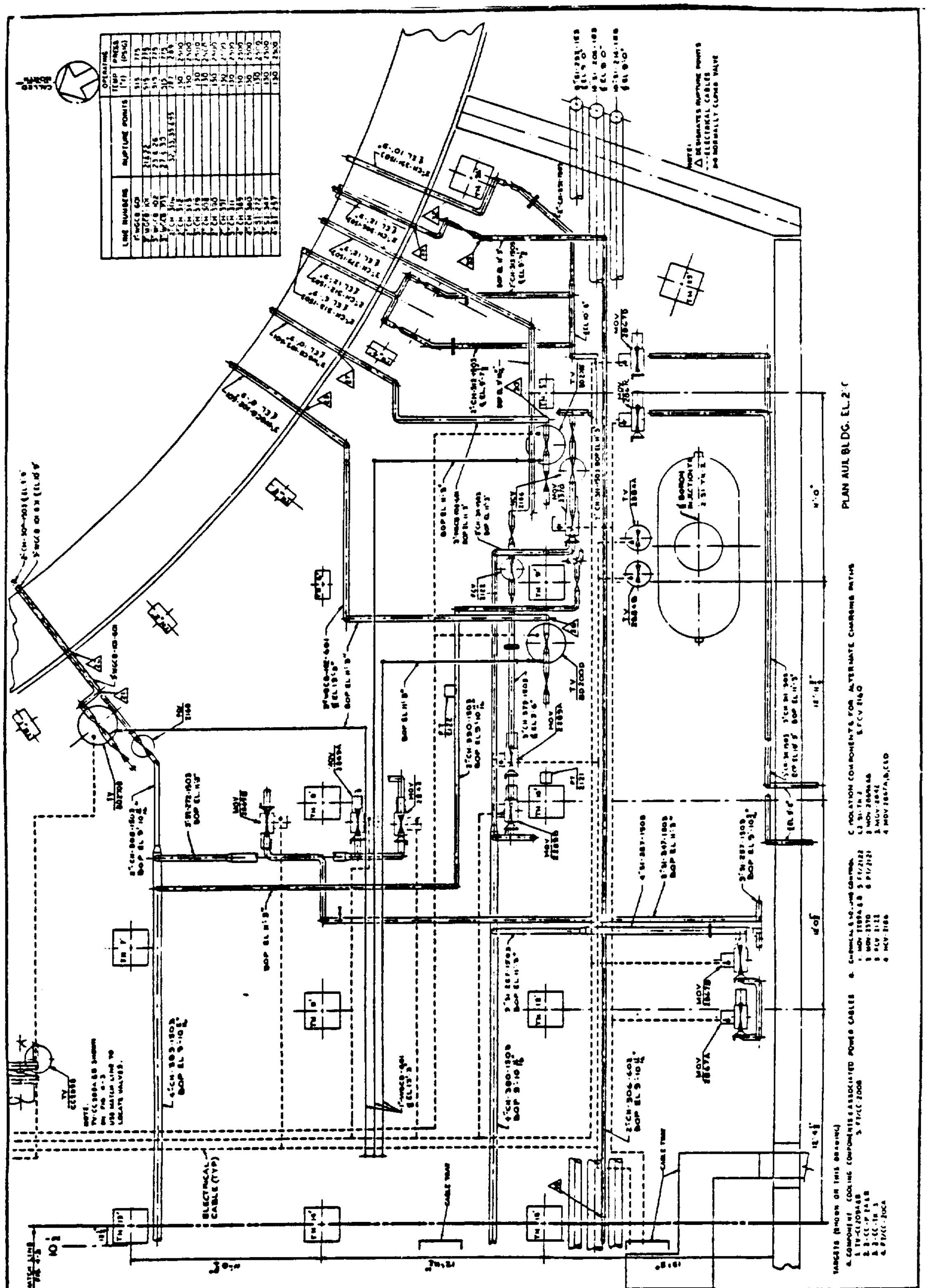
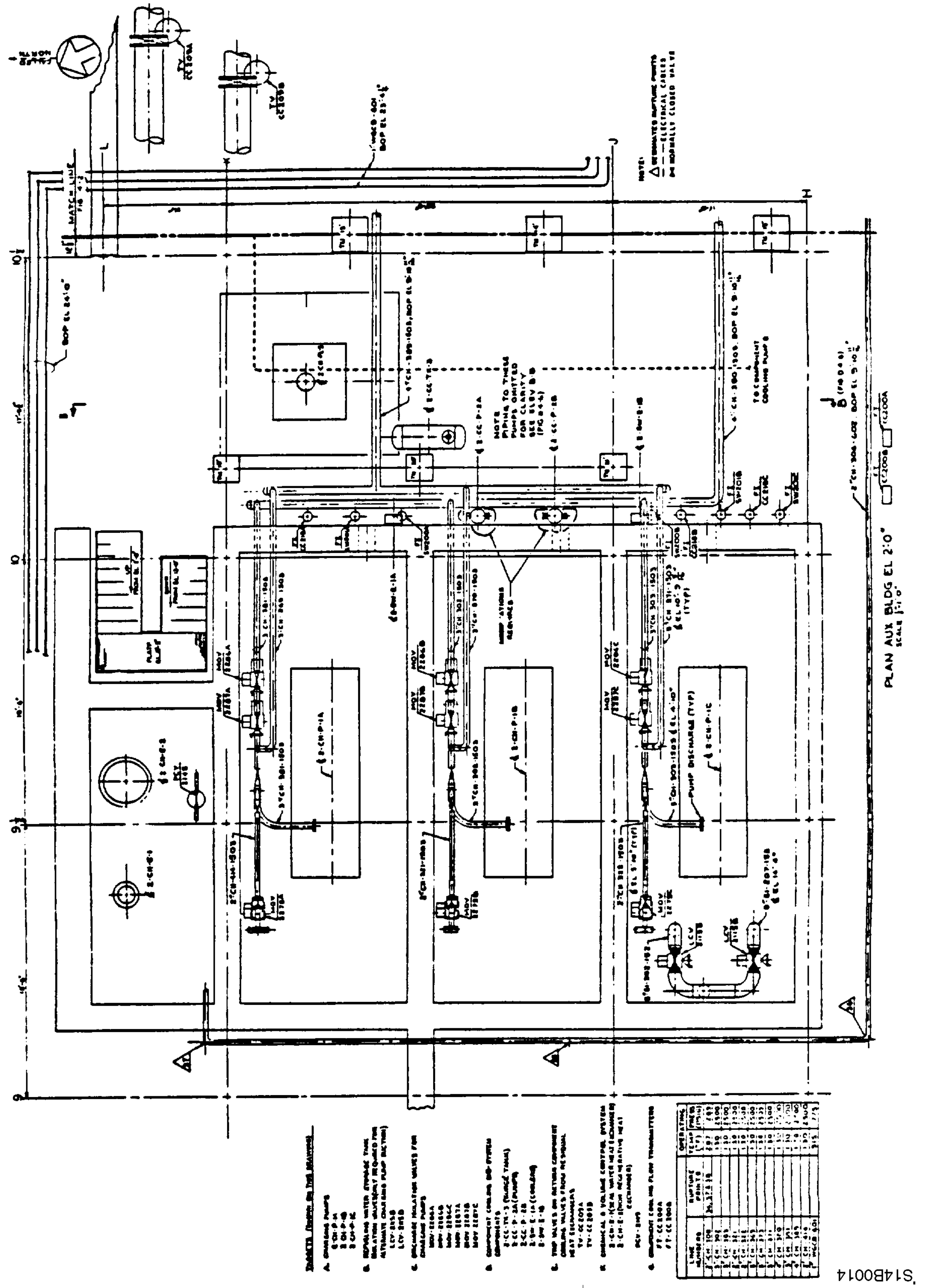


Figure 14B-11
AUXILIARY BUILDING SOURCES AND TARGETS-UNIT 2 SHEET 1

Figure 14B-12
AUXILIARY BUILDING SOURCES AND TARGETS SHEET 2



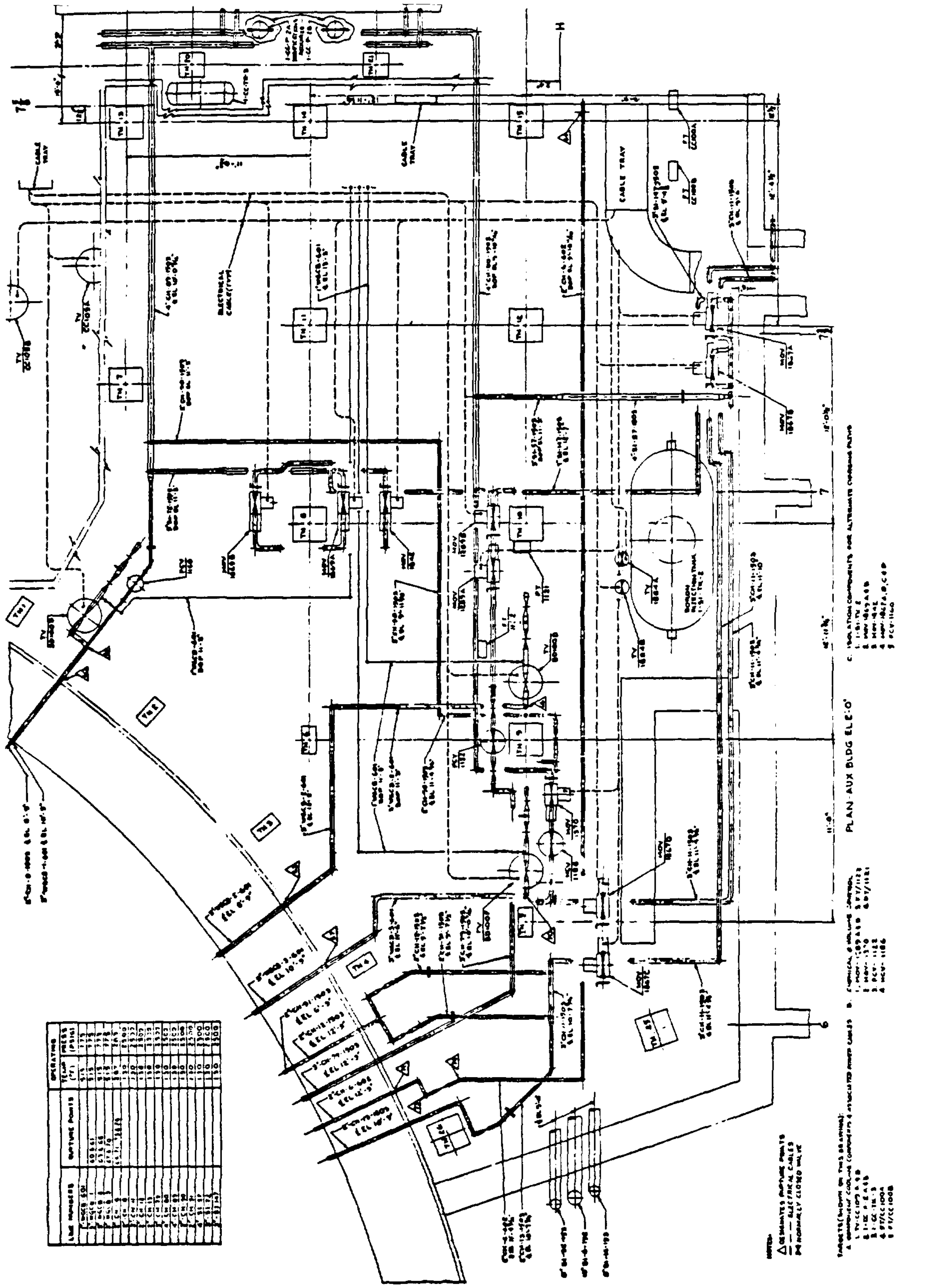


Figure 14B-13
AUXILIARY BUILDING SOURCES AND TARGETS-UNIT 1 SHEET 3

S14B0015

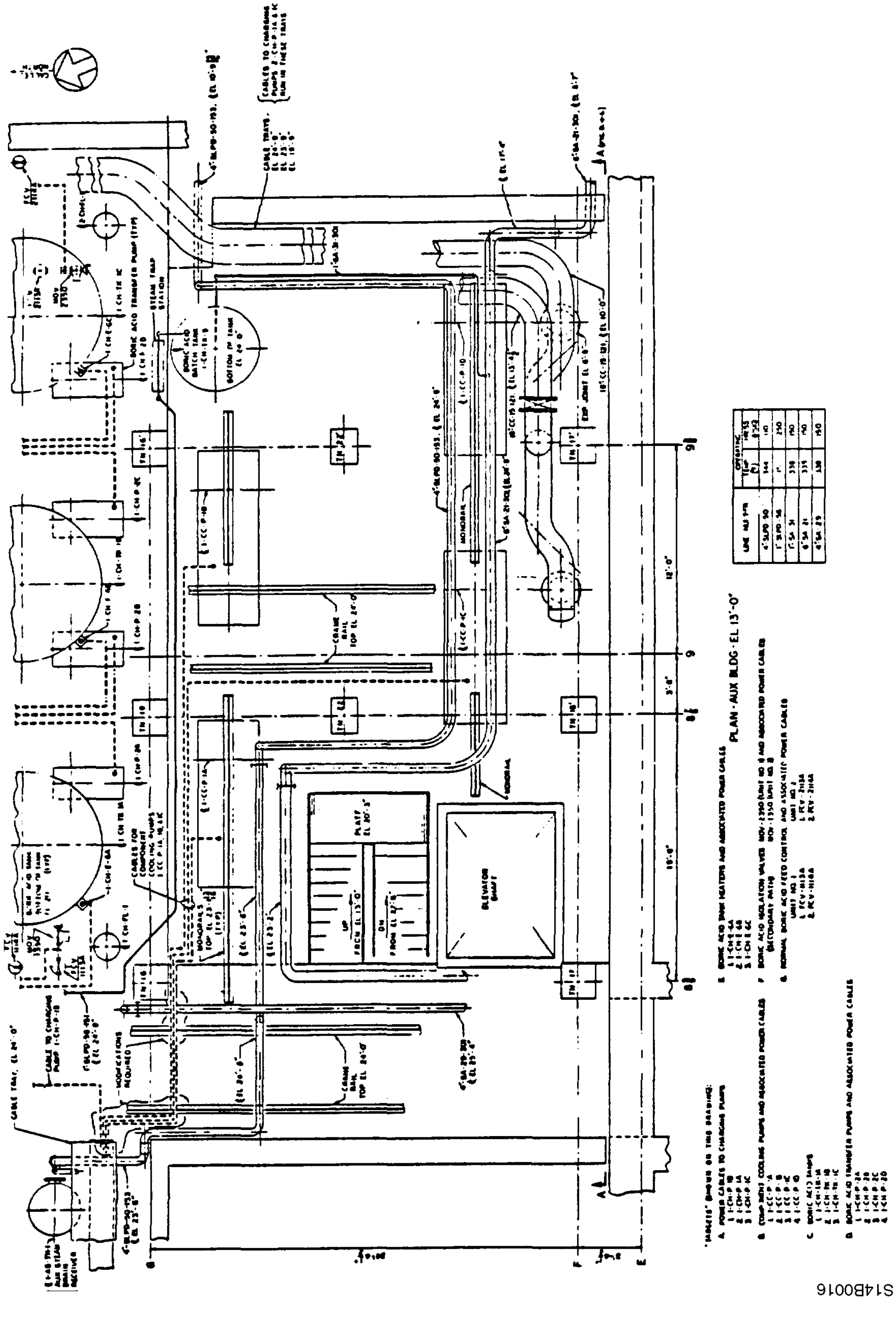


Figure 14B-14
AUXILIARY BUILDING SOURCES AND TARGETS SHEET 4

Figure 14B-15
AUXILIARY BUILDING SOURCES AND TARGETS SHEET 5A

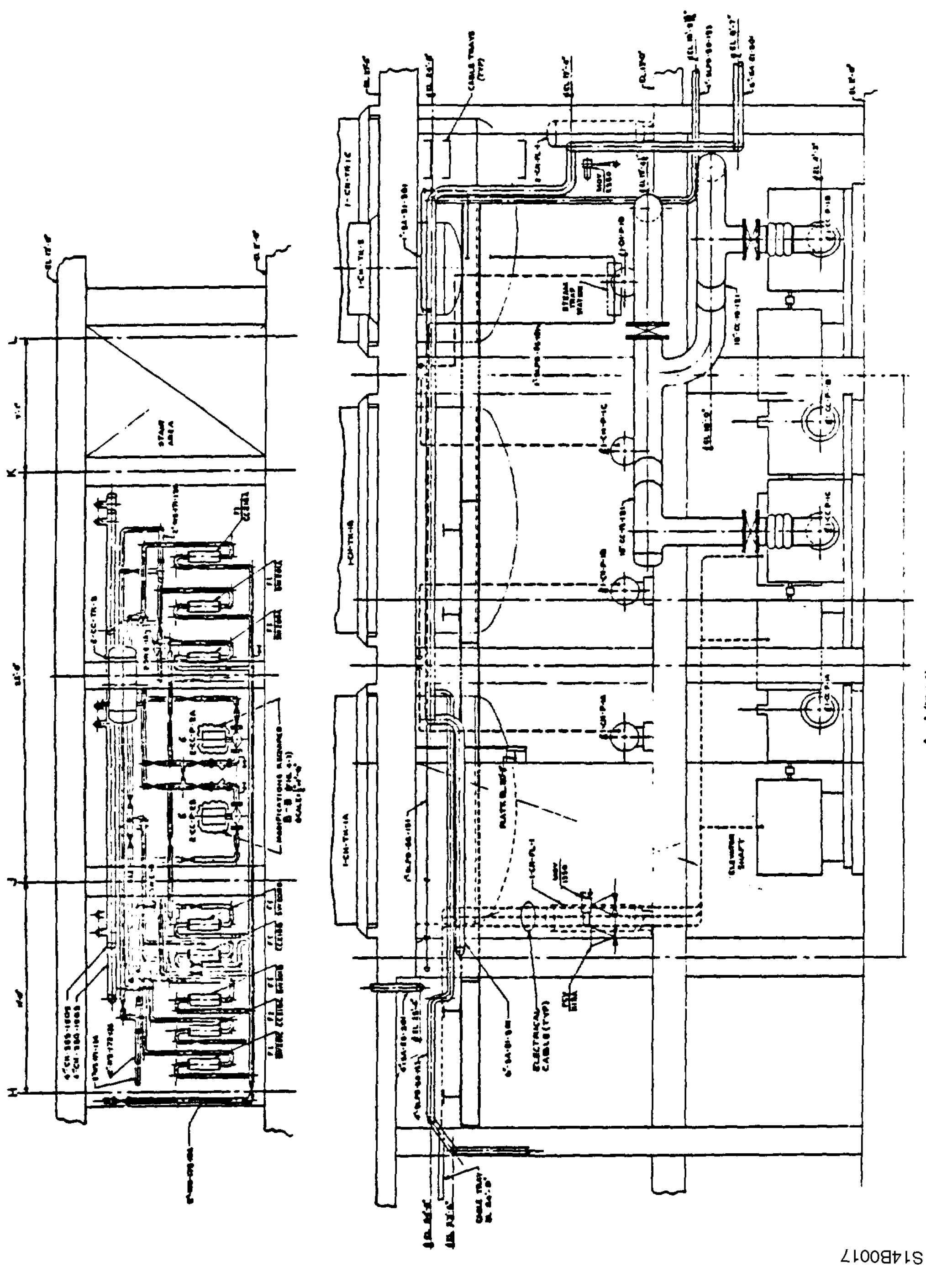
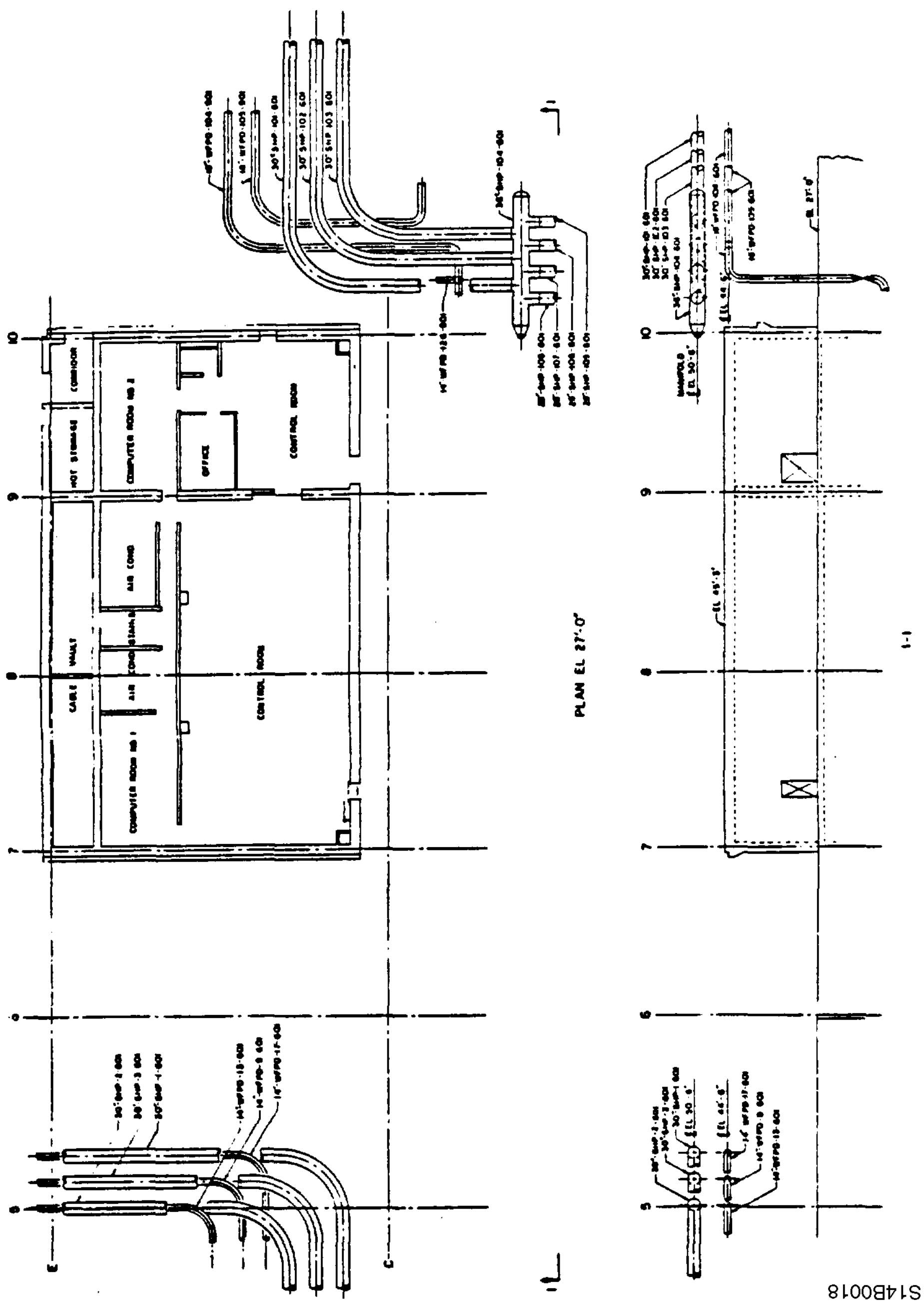


Figure 14B-16
CONTROL ROOM IN RELATION MAIN STEAM AND FEEDWATER LINE



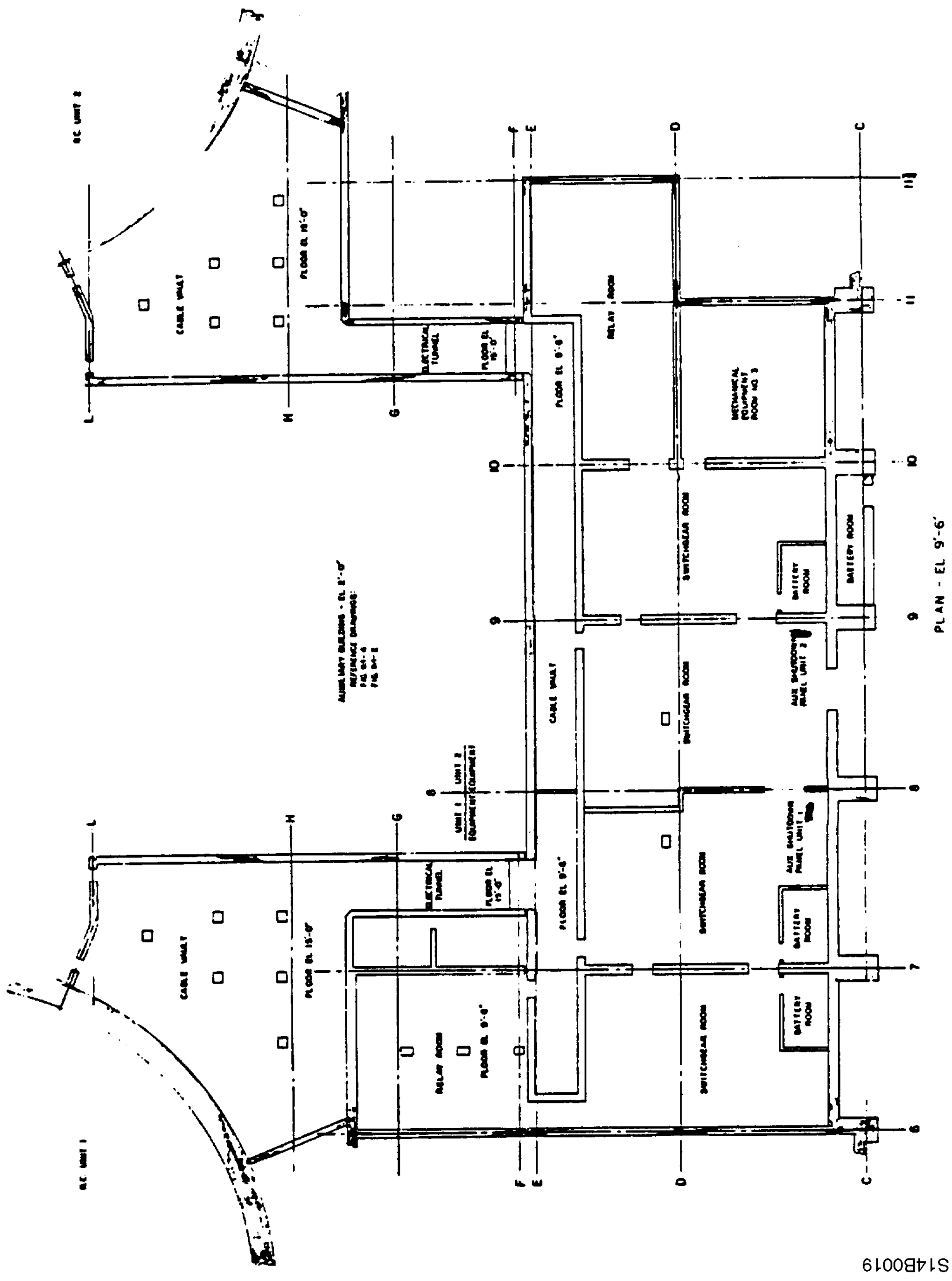


Figure 14B-17
EXCLUDED AREAS

Figure 14B-18
PRESSURE BUILDUP IN MAIN STEAM VALVE HOUSE

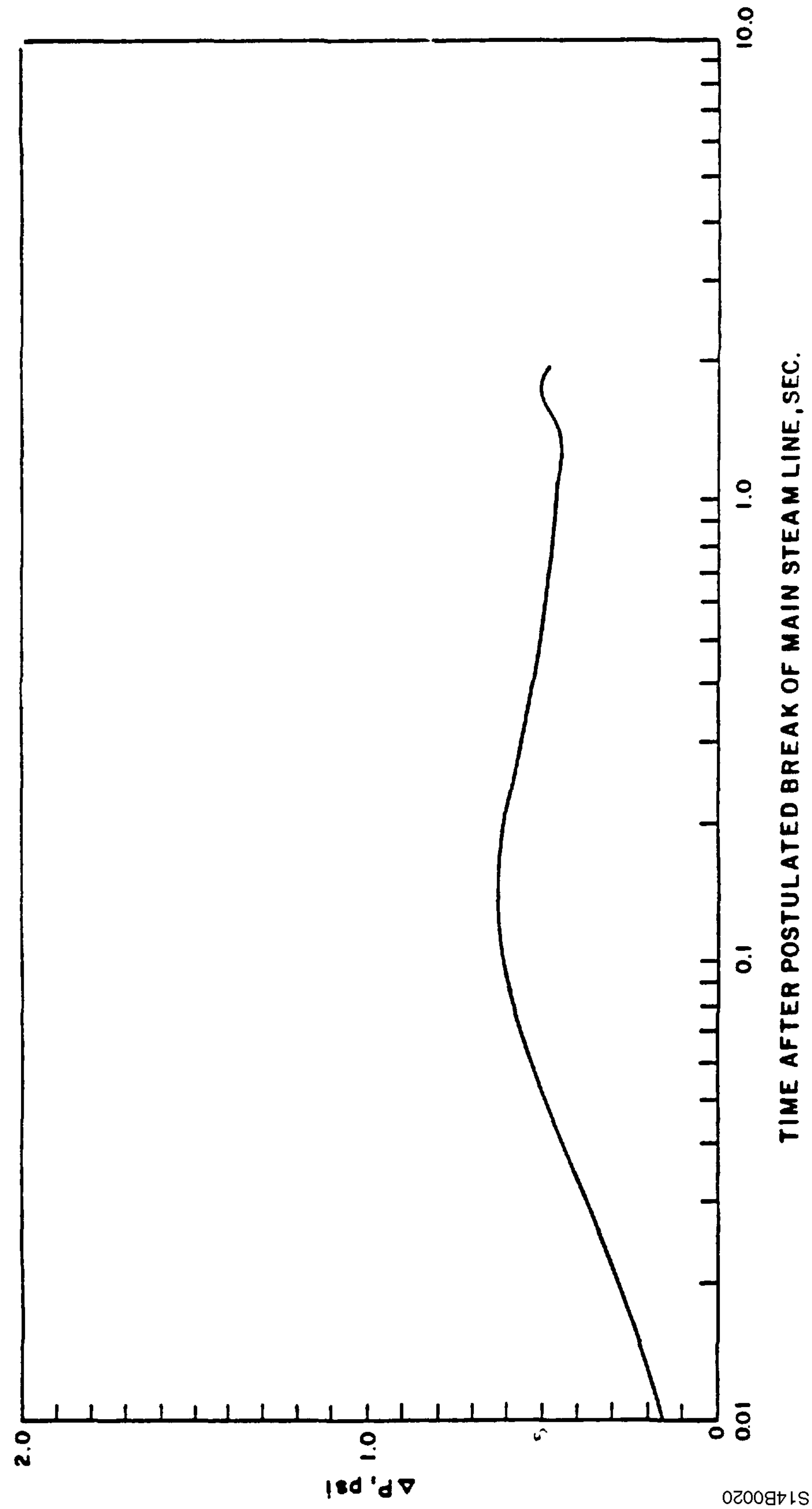
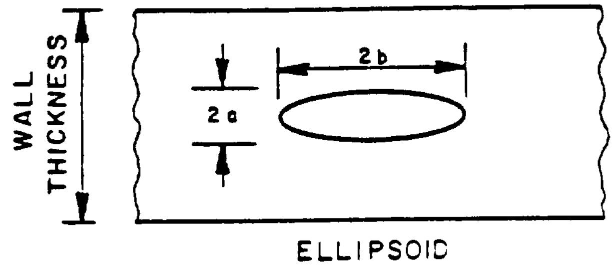


Figure 14B-19
CRACK AND FLAW GEOMETRIES

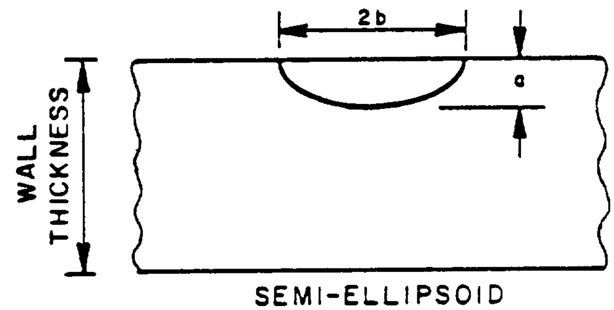
PART A

INTERNAL FLAWS



PART B

SURFACE FLAWS



PART C

AXIAL THROUGH WALL CRACKS

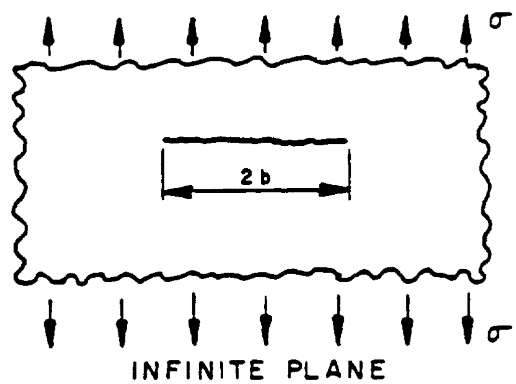


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Figure 14B-20
FLOW DIAGRAM CROSS-CONNECTS FOR AUXILIARY FEED

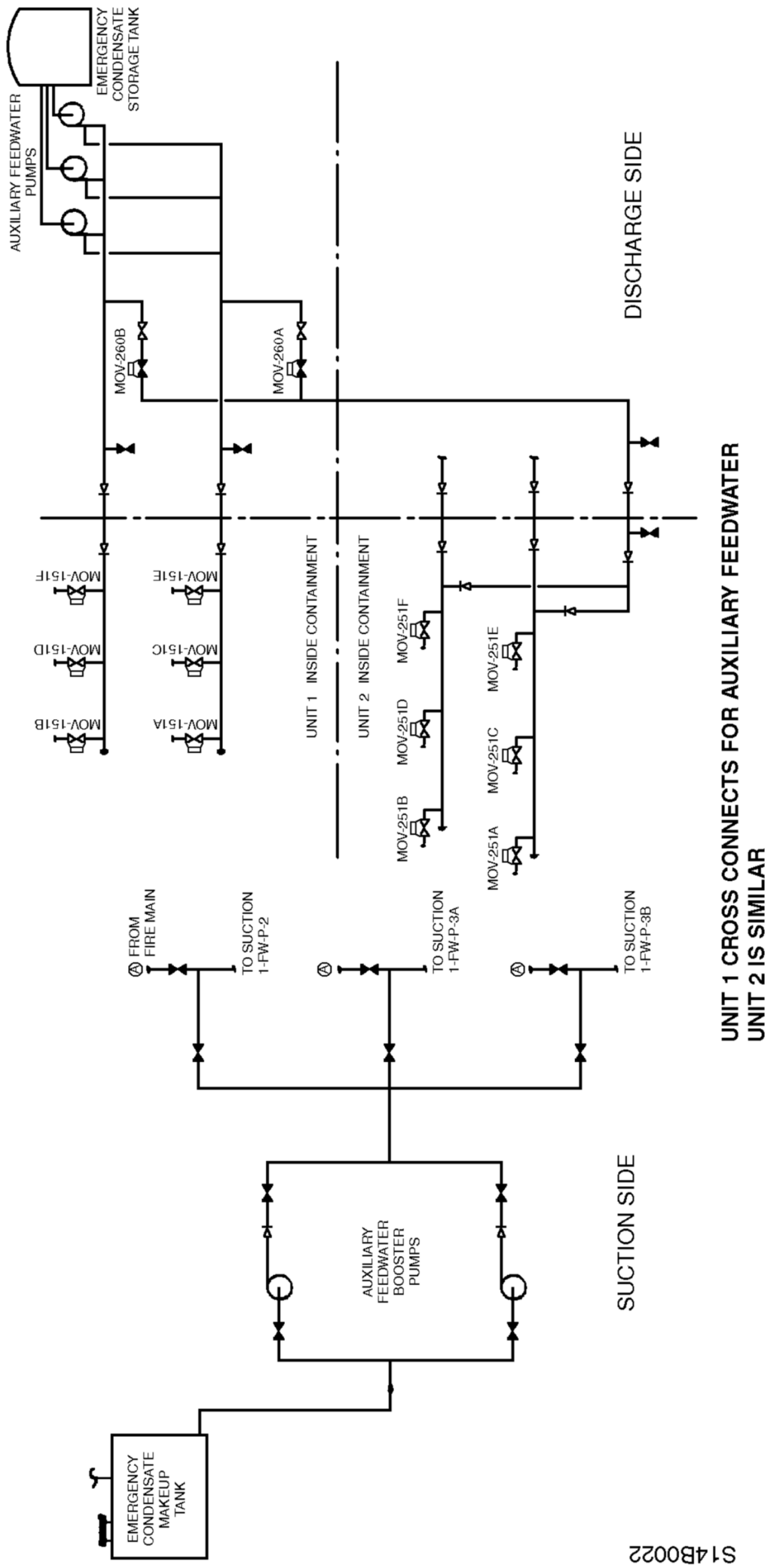
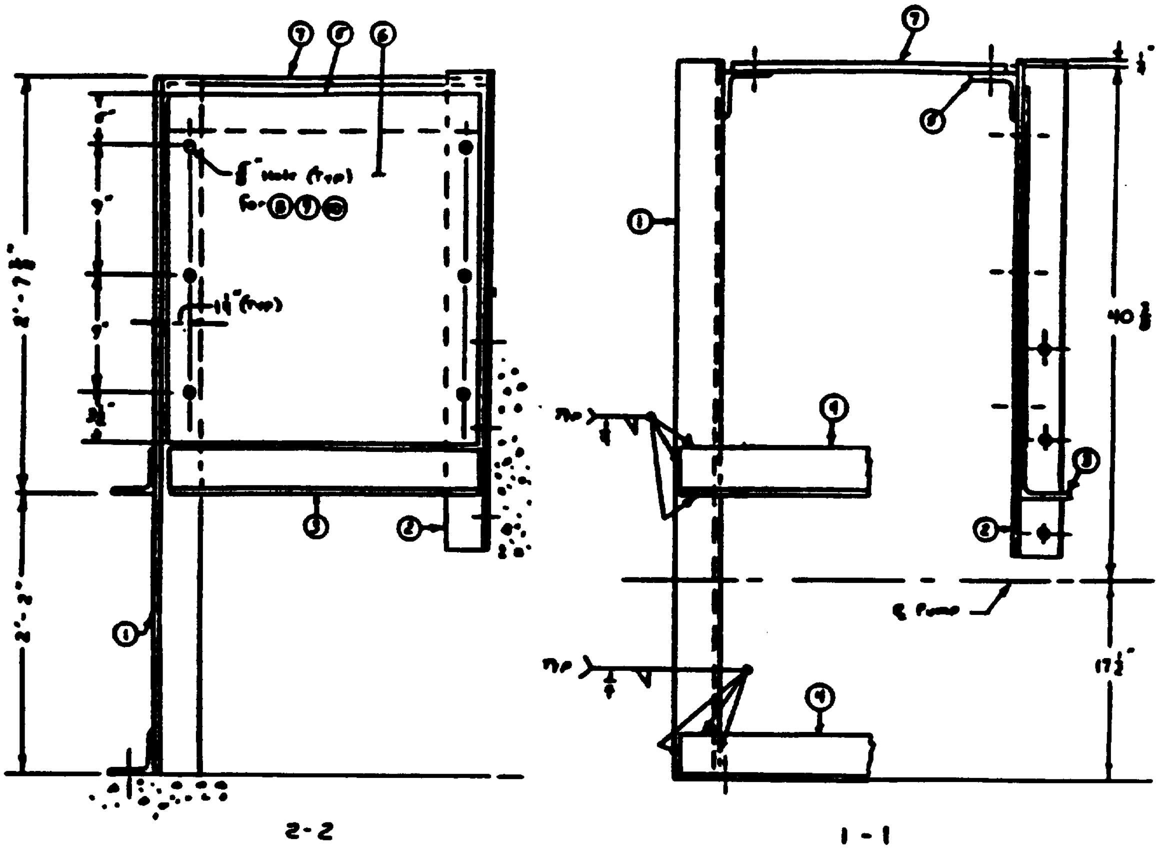
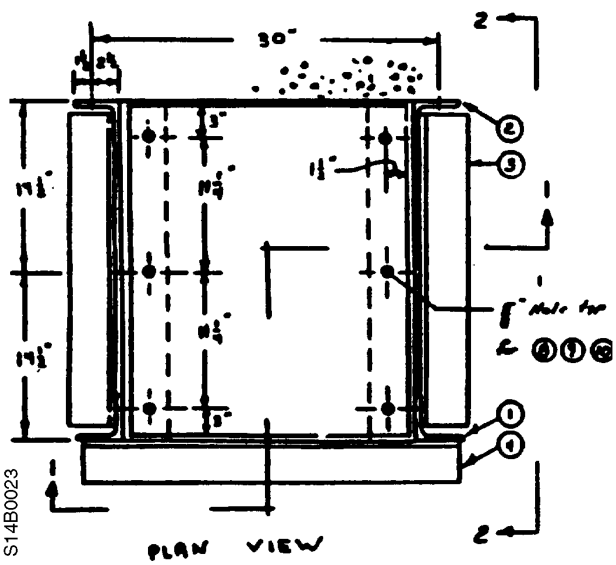


Figure 14B-21
BILL OF MATERIALS; JET IMPINGEMENT SHIELD





BILL OF MATERIALS

JET IMPINGEMENT SHIELD

Item	Quanti	Description	Material Spec.
1	2	L 4 x 4 x 3/8 4'-9 7/8" LG	A36
2	2	L 4 x 4 x 3/8 3'-1 5/8" LG	A36
3	2	L 4 x 4 x 3/8 2'-2 1/2" LG	A36
4	2	L 4 x 4 x 3/8 2'-7" LG	A36
5	2	L 4 x 4 x 3/8 2'-3" LG	A36
6	2	PL 3/8 x 2'-2 1/2" x 2'-2 1/2"	A36
7	1	PL 3/8 x 1'-11" x 2'-4 1/2"	A36
8	18	1/2" DIA BOLTS 2" LG	A325
9	18	1/2" DIA NUTS	A194
10	18	1/2" DIA WASHERS	

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Chapter 15

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CHAPTER 15 STRUCTURES AND CONSTRUCTION

15.1 STRUCTURES AND MACHINERY ARRANGEMENT

The site arrangement, plot plan, and the general arrangement of equipment within the principal Class I structures are shown on the Figures and Reference Drawings listed in the following tabulation:

Item	Reference Drawing
Site Plan	Figure 15.1-1
Plot Plan	Figure 15.1-2 and Reference Drawing 1
Containment Structure and Containment Auxiliary Structures	Reference Drawings 2 through 8
Auxiliary Building	Reference Drawings 9, 10, 11, & 12
Fuel Building	Reference Drawings 13 & 14
Control Area	Reference Drawing 15

15.1 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11448-FY-1D	Plot Plan
2.	11448-FM-1A	Machine Location: Reactor Containment, Elevation 47'- 4"
3.	11448-FM-1B	Machine Location: Reactor Containment, Elevation 18'- 4"
4.	11448-FM-1C	Machine Location: Reactor Containment, Elevation 3'- 6"
5.	11448-FM-1D	Machine Location: Reactor Containment, Elevation 27'- 7"
6.	11448-FM-1E	Machine Location: Reactor Containment; Sections “A-A”, “E-E”, & “Z-Z”
7.	11448-FM-1F	Machine Location: Reactor Containment; Sections “B-B”, “X-X”, & “Y-Y”
8.	11448-FM-1G	Machine Location: Reactor Containment, Sections “C-C” & “D-D”
9.	11448-FM-5A	Arrangement: Auxiliary Building
10.	11448-FM-5B	Arrangement: Auxiliary Building, Unit 1
11.	11448-FM-5C	Arrangement: Auxiliary Building
12.	11448-FM-5D	Arrangement: Auxiliary Building
13.	11448-FM-9A	Arrangement: Fuel Building, Sheet 1
14.	11448-FM-9B	Arrangement: Fuel Building, Sheet 2, Unit 1
15.	11448-FA-1E	Control and Relay Room Service Building

Figure 15.1-1
SITE PLAN

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

15.2 STRUCTURAL DESIGN CRITERIA

15.2.1 General

The structures, systems, and components of the Surry Power Station, Units No. 1 and No. 2, are classified into groupings requiring seismic, tornado or conventional design. The effects of the Power Uprating to a core power of 2546 MWt on pipe stress and supports were reviewed for the systems listed below. The review determined that the existing piping and support configuration is adequate to withstand the increase in pressure and temperature associated with the Power Uprating.

Systems: Main Steam

Condensate

Extraction System

H. P. Heater Drain

L. P. Heater Drains

Reactor Coolant

Class I design encompasses those structures, systems or components of reactor facilities that are essential to the prevention of accidents that could affect the public health and safety, or to the mitigation of their consequences.

Structures, systems, and components are designed, fabricated, and constructed to performance standards that will enable the facility to withstand, without loss of capability to protect the public, the additional forces that might be imposed by:

1. The operating-basis earthquake and the design-basis earthquake.
2. Tornados and other local site effects including flooding conditions, winds, and ice. Radiation levels that constitute a hazard to the public are defined in 10 CFR 50.67.

A Class I structure is designed for resistance to seismic loadings in accordance with Section 15.2.4 and for tornados, where applicable, in accordance with Section 15.2.3. There are some structures, systems, or components whose loss or failure by earthquake will not affect the public health or safety and will permit safe station shutdown, although their loss could interrupt power generation. These structures, systems, or components are not designed for specific seismic or tornado loadings.

Structures not designed for seismic or tornado loadings are designed according to *Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings* (AISC-1963), and *Building Code Requirements for Reinforced Concrete* (ACI 318-63, Part IVA - *Working Stress Design*).

These structures are designed for dead, live, and normal wind loads using allowable stress levels given in the above codes.

Some structures, systems and components of the station are necessary for a safe and orderly shutdown during a tornado. These structures are designed for tornado loadings, and systems and components are protected by tornado-resistant structures.

A list of the structures, systems, and components designed to satisfy seismic and/or tornado criteria is given in Table 15.2-1.

15.2.2 Normal Wind Loading

All structures were designed to withstand the following wind loads applied to the projected area of all surfaces:

Elevation 26 ft. 6 in. to Elevation 56 ft. 6 in., 30 lb/ft^2

Elevation 57 ft. 6 in. to Elevation 75 ft. 6 in., 35 lb/ft^2

Elevation 75 ft. 6 in. to Elevation 130 ft. 0 in., 45 lb/ft^2

Elevation 131 ft. 0 in. and above, 55 lb/ft^2

Roofs were designed for uplift using 1.25 times the wind load taken at the corresponding elevation of the roof.

Members subject to stresses produced by this wind load combined with live and dead loads were proportioned for stresses 33-1/3% greater than conventional working stresses, provided that the section thus required is not less than that required for the combination of dead and live loads computed without the one-third increase.

15.2.3 Tornado Criteria

Section 2.2 outlines the probability of a tornado occurring at the site. Although no structural damage is known to have resulted to a reinforced concrete building in a tornado (Reference 1), the structures and systems so indicated in Table 15.2-1 are designed to ensure safe shutdown of the reactor when subjected to tornado loadings.

The tornado model used for design has the following characteristics:

Rotational velocity	300 mph
Translation velocity	60 mph
Pressure drop	3 psi in 3 sec
Overall diameter	1200 ft
Radius of maximum winds	200 ft

Applicable structures are designed to resist a maximum wind velocity associated with a tornado of 360 mph, which is obtained by adding the rotational and translational velocities. Structures and systems are checked for tornado pressure loading, vacuum loading, and the combination of these two.

The tornado wind velocity is converted to an equivalent pressure, which is applied to the structures uniformly using the formula:

$P = 0.00256\ V^2$

where:

P = equivalent pressure, lb/ft^2

V = wind velocity, mph

This pressure is multiplied by applicable shape factors and drag coefficients as given in ASCE Paper No. 3269 by Thomas W. Singell (Reference 2), and applied to the silhouette of the structure.

A reduction of the full negative pressure differential is made when venting of the structures is provided. The amount of the reduction is a function of the venting area provided.

Tornado wind loads are combined with other loads as described in Section 15.5.1.2. Tornado and earthquake loads are not considered to act simultaneously. A uniform wind velocity and a nonuniform atmospheric pressure gradient is incorporated in the design of the containment structure.

Structural design criteria for tornado loading for the containment structure are given in Table 15.5-1 and Section 15.5.1.5.

It is assumed that a tornado could generate either of the following potential missiles:

1. Missile equivalent to a wooden utility pole 40 feet long, 12-inch diameter, weighing 50 lb/ft^3 and traveling in a vertical or horizontal direction at 150 mph.
2. Missile equivalent to a 1-ton automobile traveling at 150 mph.

The amount of penetration of the missile is determined from the Ballistic Research Laboratories formula (Reference 3).

The design assumes maximum wind forces and partial vacuum to occur simultaneously with the impact of either of the missiles singly. Allowable stresses do not exceed 90% of the certified minimum yield strength of the steel, the capacity reduction factor given in Section 15.5.1.2 times the certified minimum yield strength of the reinforcing steel, and 75% of the ultimate strength of the concrete.

15.2.4 Seismic Design

Class I structures, systems, and components designed to resist seismic forces are listed in Table 15.2-1. The design is based on two separate seismic criteria: the operating-basis earthquake (OBE) and the design-basis earthquake (DBE), as described in Section 2.5.

The seismic analysis of Class I structures, such as the containment structure, auxiliary building, fuel building, service building (including the control room), and safeguard areas, was based on the modal analysis response spectra technique. Major equipment-supporting structures, such as steam generator supports, reactor coolant pump supports, and pressurizer supports, were treated in an identical manner. Acceleration response spectra for the OBE and DBE are given on Figures 2.5-5 and 2.5-6.

Seismic loading includes the horizontal or vertical responses acceleration or combinations of both where the effects, as measured by the separate acceleration components, of horizontal and vertical accelerations are combined to produce maximum stress intensities, taking into account any potential adverse effect due to phase of the separate accelerations.

Damping factors for the structures, systems, and components are given in Table 15.2-2.

The design of the containment structures is based on ultimate strength design and loading factors as described in Section 15.5.1.2. Maximum allowable stress levels for both the operating-basis earthquake and the design-basis earthquake are based on proportions of the minimum yield strength.

For other Class I structures, the operating-basis earthquake loading is combined with dead, live, and other static loads. Normal wind or tornado loadings are not assumed to occur simultaneously with the earthquake loading. Members are proportioned for stresses 33-1/3% greater than conventional working stresses, provided that the section thus required is not less than that required for the combination of dead and live loads computed without the one-third increase. Allowable soil-bearing values are increased one-third.

For Class I structures other than the containment structure, the design basis earthquake is combined with static loads using loading combinations given in Table 15.5-1. For these structures under the design-basis earthquake loading, the allowable stresses do not exceed 90% of the certified minimum yield strength for structural steel, the capacity reduction factor, given in Section 15.5.1.2, times the certified minimum yield strength for reinforcing steel, and the capacity reduction factor times the specified strength for concrete. Allowable soil bearing values are increased by one-half.

To allow for unimpeded relative motions between structures, a rattlespace is provided between the:

In general, the periphery of the rattlespace between buildings is arranged to prevent material entering the space, with the inner areas left as a void.

Maximum relative motions between adjoining structures are included in the stress analyses of all piping that extends from one building to another.

Type “A” sand, as described in Section 2.4.3.3, was removed from under the fuel building, auxiliary building, and control area and replaced by a dense graded granular fill material as described in Section 2.4.5.1.

The analytical procedure used for the nuclear steam supply system is described by Section 15A.3 of Appendix 15A.

The reactor protection system, engineered safety feature (ESF) circuits, and the emergency power system are designed so that they will not lose the capability to shut the plant down and maintain it in a safe shutdown condition under operating-basis earthquake or design-basis earthquake conditions. For the design-basis earthquake, permanent deformation of the equipment is allowable, provided that the capability to perform its function is maintained.

Typical protection system equipment is subjected to type tests under simulated seismic accelerations to demonstrate its ability to perform its functions. Type testing was performed using conservatively large accelerations and applicable frequencies. Analyses were done for structures that were not done for the reactor protection system equipment. However, the peak accelerations and frequencies were checked against those derived by structural analyses of operational and design-basis earthquake loadings.

A Westinghouse topical report, WCAP-7397-L, provides the seismic evaluation of safety-related equipment. The type tests covered by this report are applicable to the Surry Power Station, with the exception of the process control equipment, which is covered in a supplement to WCAP-7397-L.

The emergency switchgear has been tested under seismic conditions, and manufacturers’ test data are available. The emergency generator and control panels are identical with those used in locomotives, and have been tested under severe conditions, but no seismic tests have been made.

The control board was designed to withstand earthquake conditions, and an analysis was performed to verify the adequacy of the seismic design, but tests were not performed.

The emergency batteries are supported on rigid reinforced concrete pedestals firmly anchored to the floor. Steel retaining members integral with the pedestals prevent the batteries from being dislodged under seismic excitation.

15.2.5 Hydrostatic Loadings

Finish ground grade at the station is at Elevation 26 ft. 6 in. Natural ground water level is at approximately Elevation 4 ft. 0 in.

The exterior wall of the containment structure extends approximately 66 feet below the finished ground level. Water-resistant membrane protection for this structure is defined in Sections 15.5.1.9 and 15.5.1.10. External pumps for reducing the hydrostatic head on the containment structure are described in Section 15.5.1.3. This latter section also discusses the effect of the buoyant pressure on the containment structure.

Exterior surfaces of walls of other Class I structures with floor levels below Elevation 26 ft. 6 in. are covered with a mopped-on bitumastic coating to establish a water-resistant membrane.

15.2 REFERENCES

1. V. C. Gilberton and E. E. Mageanu, *Tornadoes*, *AIA Technical Reference Guide*, TRG 13-2, U. S. Weather Bureau.
2. T. W. Singell, *Wind Forces on Structures: Forces on Enclosed Structures*, *Journal of the Structural Division of the ASCE*, July 1958.
3. C. R. Russell, *Reactor Safeguards*, MacMillan, 1962.
4. Letter, NRC to Vepco, Serial #85-885, dated December 4, 1985
5. ASME Boiler and Pressure Vessel Code, Section III, Division I, Code Case N-411, *Alternative Damping Values for Seismic Analysis of Piping Sections,* American Society of Mechanical Engineers, 345 E. 47th Street, New York, NY 10017, dated September 17, 1984.
6. NRC Bulletin No. 88-11: *Pressurizer Surge Line Thermal Stratification,* USNRC, December 20, 1988.
7. Virginia Power Letters Serial Nos. 89-006A dated May 3, 1989 and 89-006B dated November 13, 1989 to United States Nuclear Regulatory Commission.
8. *Revised report on the Reanalysis of Safety-Related Piping Systems - Surry Power Station, Unit 1*, August 1979, by Stone & Webster Engineering Corporation.
9. *Report on the I.E. Bulletin 79-14, Analysis for As-Built Safety-Related Piping Systems - Surry Power Station - Unit 2,* July 1981, by Ebasco Services, Inc.
10. *Report on the Reanalysis of Safety-Related Piping Systems - Surry Power Station - Unit 2,* Rev. 1, April 1980, by Ebasco Services, Inc.
11. Mitsubishi Heavy Industries, LTD., Design Report DG KCS-03-0008, *Dominion Generation, Surry Power Station Unit 2, Control Rod Drive Mechanism Design Report,* Rev. 3.

Table 15.2-1

STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Structures				
Reactor containment			SW	
Reinforced-concrete substructure	I	P		
Reinforced-concrete superstructure	I	T		
Reinforced concrete interior shields and walls	I	NA		
Steel plate liner	I	P		P for containment integrity
Piping, duct, and electrical penetrations and shield wall	I	P		P for shield wall and critical system penetrations only
Personnel access hatch	I	P		P for containment integrity
Equipment access hatch	I	P		P for containment integrity
Cable vault and cable tunnel	I	T	SW	
Pipe tunnel to containment from auxiliary building	I	T	SW	
Auxiliary steam generator feed pump cubicle	I	T	SW	
Cubicle for main steam and feedwater isolation valves	I	T	SW	
Recirculation spray and low-head safety injection pump cubicle and pipe tunnel	I	P	SW	
Safeguards ventilation room	I	NA	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Structures (continued)				
Auxiliary building			SW	
Reinforced-concrete structure	I	T		
Steel superstructure	I	NA		
Vacuum equipment area	I	NA		
Fuel building			SW	
Reinforced-concrete structure	I	T		T for horizontal missile only
Steel superstructure	I	T		T for tornado winds only
Spent-fuel storage rack	I	P		P for horizontal missile only
Fuel-handling trolley support structure	I	T		T for tornado winds only
Control room	I	T	SW	
Emergency switchgear and relay room	I	T	SW	
Battery rooms	I	T	SW	
Air-conditioning equipment room	I	T	SW	For control room and relay room only
Reactor trip breaker cubicle	I	T	SW	
Emergency diesel-generator rooms			SW	
Reinforced-concrete floor	I	T		
Walls, excluding louvers	I	T		
Roof slab	I	T		

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Structures (continued)				
Turbine building	NA	NA	SW	By design, building collapse will not damage any Class I structures and components during earthquake, or tornado-resistant structures and components during tornado
Mechanical Equipment Room-5	I	T		
Circulating water pump intake structure	I	T	SW	T for emergency service water pump cubicle only
High-level intake structures	I	T	SW	T, no missile protection required
Seal pits	I	T	SW	T, no missile protection required
High-level intake canal	I	NA	SW	
Fire-pump house	I	T	SW	Engine-driven pump only
Fuel-oil transfer pump vault	I	T	SW	
Boron recovery tank dikes	I	T	SW	
Systems				
Reactor coolant system				
Steam generators	I	P	W	
Steam generator supports	I	P	SW	
Reactor coolant pumps[b]	I	P	W	

a. HISTORICAL information, see Note 2.
b. All references to “pumps” include drivers.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Reactor coolant system (continued)				
Reactor coolant pump supports	I	P	SW	
Pressurizer and pressurizer heaters	I	P	W	
Pressurizer support	I	P	SW	
Pressurizer relief tank	I	P	W	
Reactor vessel				
Reactor core support structure	I	P	W	
Reactor control rod guide structure	I	P	W	
Fuel assemblies	I	P	W	
Control rod and drive shaft assemblies	I	P	W	
Incore instrumentation thimbles	I	P	W	
Reactor vessel supports and neutron shield tank	I	P	SW	
Control rod drive mechanisms	I	P	W	
Reactor coolant piping, valves, and supports [b]	I	P	W	
Reactor coolant bypass piping, valves, and supports	I	P	W	
Pressurizer surge line	I	P	W, SW [c]	
Pressurizer spray lines, valves, and supports	I	P	SW	

a. HISTORICAL information, see Note 2.

b. All references to "piping and valves" include root valves connecting to non-Class I systems, and valve operators.

c. Pressurizer surge line was reanalyzed per NRC Bulletin 88-11, dated December 20, 1988.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Reactor coolant system (continued)				
Pressurizer safety and relief valves	I	P	W	
Safety injection system				
Accumulators and supports	I	NA	W	
Low-head safety injection pumps and piping	I	P	W	P for containment integrity
Boric acid injection tanks and piping	I	P	W	
Piping, valves, and supports	I	NA	SW	Except drain/sample lines
Containment spray system				
Refueling water storage tank	I	NA	SW	
Containment spray pumps	I	NA	SW	
Piping, valves, and supports	I	NA	SW	Except recirculation lines
Refueling water chemical addition tank	I	NA	SW	
Recirculation spray systems				
Recirculation spray pumps and piping	I	P	SW	P for containment integrity
Recirculation spray heat exchangers	I	NA	SW	
Reactor containment sump and screens	I	NA	SW	
Piping, valves, and supports	I	NA	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Containment vacuum system				
Process vent	I	NA	SW	
Vacuum pump piping, valves, and supports	I	NA	SW	
Chemical and volume control system				
Boric acid storage tanks	I	NA	W	
Boric acid transfer pumps	I	P	W	
Boric acid blender	I	P	W	
Charging/safety injection pumps	I	P	W	
Regenerative heat exchanger	I	P	W	
Nonregenerative heat exchanger	I	P	W	
Mixed-bed demineralizers	I	P	W	
Reactor coolant filter	I	P	W	
Volume control tank	I	P	W	
Seal-water heat exchanger	I	P	W	
Seal-water filter	I	P	W	
Excess letdown heat exchanger	I	P	W	
Piping, valves, and supports				
Boric acid piping	I	P	SW	
Feed and bleed piping	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Chemical and volume control system (continued)				
Piping, valves, and supports (continued)				
Hydrogen, nitrogen, and vent piping for volume control tank	I	P	SW	
Residual heat removal system				
Residual heat removal pumps	I	P	W	
Residual heat exchangers	I	P	W	
Piping, valves, and supports	I	P	SW	
Boron recovery system				
Gas stripper	I	P	SW	
Gas stripper overhead condenser	I	P	SW	
Primary drain tank	I	P	SW	
Primary drain tank vent chiller condenser	I	P	SW	
Overhead gas compressors	I	P	SW	
Gas stripper surge tank	I	P	SW	
Component cooling system				
Component cooling pumps	I	P	SW	
Component cooling water heat exchangers	I	P	SW	
Component cooling surge tank	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Component cooling system (continued)				
Piping, valves, and supports				
For residual heat exchangers	I	P	SW	
For fuel pool coolers	I	P	SW	P for horizontal missile only
RCP thermal barrier supply piping from the upstream check valves to the RCPs	I	P	SW	
Charging pump component cooling water system				
Charging pump component cooling water pumps	I	P	SW	
Charging pump mechanical seal coolers	I	P	SW	
Charging pump seal cooling surge tank	I	P	SW	
Charging pump intermediate seal coolers	I	P	SW	
Piping, valves, and supports	I	P	SW	
Fuel pool cooling system				
Fuel pool pumps	I	P	SW	P for horizontal missile only
Fuel pool coolers	I	P	SW	P for horizontal missile only
Piping, valves, and supports connecting above equipment to spent-fuel pool	I	P	SW	P for horizontal missile only

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Compressed air system				
Critical system interface components - local instrument air accumulators/bottled sources and associated piping, valves, and supports				
Interfacing systems:				
Reactor coolant system - Pressurizer PORVs	I	P		
Component cooling system - Containment isolation trip valves from the RHR HXs	I	P		
Main steam/feedwater system - Steam supply admission valves to the turbine driven auxiliary feedwater pump	I	P		
Ventilation vent system - Dampers - Fuel building filtered exhaust, containment air compressor cubicle exhaust, safeguards area normal exhaust, charging pump normal and filtered ventilation, and containment exhaust isolation	I	N/A		
Service water system				
Engine-driven emergency service water pumps	I	P	SW	
Charging pump service water pumps	I	P	SW	
Charging pump lubricating oil coolers	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Service water system (continued)				
Service water piping, valves, and pipe supports for:				
Recirculation spray heat exchangers	I	NA	SW	
Component cooling heat exchangers	I	P	SW	
Engine-driven emergency service water pump	I	P	SW	
Emergency diesel-generator cooling	I	P	SW	
Control room air-conditioning equipment condensers	I	P	SW	
Charging pump lube-oil coolers	I	P	SW	
Diesel-oil tank for emergency service water pump	I	P	SW	
Fire protection system				
Engine-driven fire pump	I	P	SW	
Diesel-oil tank (300 gallons)	I	P	SW	
Yard hydrant piping system	I	NA	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Fuel handling system				
Manipulator crane in containment	I	P	W	Crane will be parked and secured so no damage to reactor control rod drive mechanisms can occur during earthquake
Movable platform with hoist in fuel building	I	NA	SW	Platform will be parked and secured so no damage to fuel can occur during earthquake or tornado
Fuel-handling trolley in fuel building	I	NA	SW	Trolley will be parked and secured during tornado warning periods so no damage to spent fuel can occur
Fuel transfer tube with blind flange	I	P	SW	P for containment isolation
Fuel elevator in fuel building	I	NA	SW	
Ventilation system				
Ventilation equipment for safeguards ventilation room	I	NA	SW	
Ductwork from safeguards ventilation room to ventilation vent no. 2	I	NA	SW	
Ductwork for containment purge system penetrating containment between and including isolation butterfly valves	I	P	SW	P for containment isolation

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Ventilation system (continued)				
Air-conditioning equipment for main control room and relay room	I	P	SW	
Emergency main control and relay room ventilation	I	P	SW	
Ventilation vent no. 2	I	NA	SW	
Control rod drive ventilation fans	NA	P	SW	
Main steam system				
Steam piping from main steam lines to auxiliary steam generator feed pump turbine	I	P	SW	
Main steam piping from steam generators to and including main steam trip valve	I	P	SW	
Main steam piping, valves, and supports from trip valves to and including turbine stop valves	NA	NA	SW	Check was made that design-basis earthquake would not cause failure to function
Turbine steam bypass piping, valves, and supports to condenser	NA	NA	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Circulating water system				
Condenser	NA	NA	SW	Check was made that condenser water boxes will not fail during earthquake
Circulating water piping, valves, and supports from high-level intake canal to circulating water discharge tunnel to and including condenser intake butterfly valve; condenser discharge butterfly valve	I	P	SW	P, underground
Circulating water discharge tunnel	I	P	SW	
Circulating water pump vacuum breaker	NA	NA	SW	No credible failure mode for passive vacuum breaker
Condensate and feedwater system				
Emergency condensate storage tank	I	P	SW	
Emergency condensate makeup tank	I	P	SW	
Auxiliary steam generator feed pumps	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Condensate and feedwater system (continued)				
Piping, valves, and supports				
Emergency condensate storage tank to auxiliary steam generator feed pump	I	P	SW	
From auxiliary steam generator feed pumps to steam generator feed lines	I	P	SW	
Steam generator feed lines inside containment to and including first isolation check valve outside containment	I	P	SW	
Primary vent and drain system				
Primary drain cooler	I	P	W	
Piping, valves, and supports	I	P	SW	
Primary drain transfer tank	I	P	SW	
Secondary vent and drain system				
Steam generator blowdown piping, valves, and supports inside containment to and including first isolation trip valves	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Gaseous waste disposal system				
Waste gas decay tanks	I	P	SW	
Waste gas recombiner system	I	NA	SW	
Waste gas compressors	I	NA	SW	
Waste gas filter	I	NA	SW	
Process vent blowers	I	NA	SW	
Waste gas piping, valves, and supports from stripper to process vent	I	NA	SW	
Process radiation monitoring system				
Process vent particulate monitor	I	NA	SW	
Process vent gas monitor	I	NA	SW	
Spray recirculation heat exchanger service water monitors	I	NA	SW	
Area radiation monitoring system				
Main control room monitor	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Instrumentation and control				
All instrumentation and control to operate and monitor operation of critical system component shown above during MCA or controlled shutdown				
Systems include:				
Reactor protection (in part)	I	P	W	
Safeguards initiation	I	N/A	W/SW	
Containment isolation	I	P	W/SW	
Reactor control (in part)	I	P	W	Includes trip breakers
Steam generator water level control system	I	P	W	
Reactor makeup control	I	P	W	
Nuclear instrumentation (in part)	I	P	W	
Non-nuclear process instrumentation	I	P	W/SW	
Circulating water intake canal low level isolation system	I	P	N/A	
Electrical systems				
Emergency diesel-generators	I	P	SW	
Fuel-oil day tanks	I	P	SW	
Fuel-oil transfer pumps	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note
Systems (continued)				
Electrical systems (continued)				
Underground fuel-oil storage tanks	I	P	SW	
Fuel-oil piping, valves, and supports, emergency diesel-generators	I	P	SW	P for piping to protected generators
Station service batteries and chargers	I	P	SW	
Vital bus and inverters	I	P	SW	
Emergency station service transformers	I	P	SW	
Emergency station service switchgear	I	P	SW	
Control panel boards	I	P	SW	
Pressurizer heater control group only	I	P	SW	
Cable to critical components, instruments, and controls as shown above	I	P	SW	Cable passing through unprotected areas will be in rigid conduit
Miscellaneous				
Reactor containment crane	I	P	SW	
Piping	I	P	SW	

a. HISTORICAL information, see Note 2.

Table 15.2-1 (CONTINUED)
STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED FOR SEISMIC AND TORNADO CRITERIA

(Refer to the equipment classification list (Q-list) for a more comprehensive list of components. See Note 1.)

Item	Earthquake Criterion	Tornado Criterion	Sponsor[a]	Note

Legend

W - Westinghouse Electric Corporation.

SW - Stone & Webster Engineering Corporation.

I - Refers to Seismic Class I criteria. All Class I components and structures are designed to resist the operating-basis earthquake within allowable working stresses. A check has been made to determine that failure to function will not occur with a design-basis earthquake.

T - Refers to structures that will not fail during the design tornado.

P - Refers to systems and components that will not fail during the design tornado since they are protected by tornado resistant structures or by being buried underground.

NA - Not applicable.

NOTES:

1. The list of structures, systems, and components in Table 15.2-1 was provided as part of the original FSAR. Since licensing, an equipment classification listing (Q-list) has been developed which provides a more comprehensive and current list of individual components and their classifications than does this table, which only provides a general list of systems and components. Table 15.2-1 has been maintained as a description of the original licensing basis for the structures, systems, and components listed, but in general it has not been updated as systems have been modified or individual components have been added or reclassified. The classifications of individual components are maintained up-to-date in the Q-list.
2. The information in the sponsor column designates the division of responsibility between Westinghouse and Stone & Webster for the original design of listed structures, systems, and components. These designations are considered HISTORICAL and are not intended or expected to be updated for the life of the plant.

Table 15.2-2
DAMPING FACTORS FOR CLASS I STRUCTURES

Component	Percent of Critical Damping
1. Reactor vessel internals and control rod assembly drives	
a. Welded assemblies	1.0
b. Bolted assemblies	2.0
c. Control rod assembly drives	5.0[a]
2. Reinforced concrete reactor support structure, including the reactor vessel	5.0
3. Vital piping systems	
a. Carbon steel	0.5 OBE, 1.0 DBE[b]
b. Stainless steel	0.5 OBE, 1.0 DBE[b]
4. Containment structure and foundation	5.0
5. Steel framed structures, including supporting structures and foundations	
a. Bolted	2.5
b. Welded	1.0
6. Concrete structures aboveground	
a. Shear-wall type	5.0
b. Rigid-frame type	5.0
7. Mechanical equipment, including pumps, fans, and similar items	2.0

a. For Surry Unit 2 control rod assembly drives, the "Percent of Critical Damping" used is 5% as justified in Mitsubishi Heavy Industries Design Report, Reference 11.

b. In accordance with reference 4, the damping values of ASME Code Case N-411 (reference 5) have been approved for use at Surry as an alternate, for both the operating-basis earthquake and the design-basis earthquake.

The values specifically are: five percent below frequency of 10Hz; linear reduction from five percent to two percent between 10Hz and 20Hz and two percent above 20Hz. These damping values are used in the following situations and with the following additional considerations:

a. For seismic analyses in cases where new piping is added, existing systems are modified, existing systems are re-evaluated for support optimization.

b. For seismic analyses using response spectrum methods and not for seismic analyses using time-history analyses methods.

c. When these damping values are used, the ±15% peak broadening criteria of Regulatory Guide 1.122, *Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components*, will be used.

d. When these alternate damping values are used, they are used in a given analyses in their entirety.

e. When these damping values are used together with changes in the support arrangement that increases the flexibility of piping systems, the predicted maximum displacements are reviewed to ensure that such displacements do not cause adverse interaction with adjacent structures, components or equipments.

15.3 MATERIAL

15.3.1 Concrete

See Section 15.5.2.4 for the description of the concrete used for the Reactor Pressure Vessel Head Replacement Project.

15.3.1.1 **Cement**

All cement used was an approved American brand conforming to the specification for Portland cement, ASTM Designation C150, Type II, low alkali. It is suitable for Class I structures because of its lower heat of hydration and improved resistance to sulphate attack. A low-content alkali was specified to minimize the possibility of reaction with aggregates. Certified copies of mill tests, showing that the cement meets or exceeds the ASTM requirements for Portland cement, were furnished by the manufacturer. An independent testing laboratory performed tests on the cement for compliance with the specifications.

15.3.1.2 **Admixtures**

An air-entraining agent was used in the concrete in an amount sufficient to entrain from 3 to 5% air by volume of the concrete. This agent conformed to the requirements of Standard Specification for Air-Entraining Admixtures for Concrete, ASTM C260, when tested in accordance with Standard Method of Testing Air-Entraining Admixtures for Concrete, ASTM C233.

The air-entraining agent was added separately to the batch in solution in a portion of the mixing water. The solution was batched by means of a mechanical dispenser capable of accurate measurement, and in a manner that ensured uniform distribution of the agent throughout the batch during the specified mixing period.

Water-reducing agents were used when their use was approved in writing. Water-reducing agents were Master Builders NB-100, type R or N, manufactured by Master Builders of Cleveland, Ohio. Type N is normal NB-100 and is used when a normal rate of hardening is required. Type R contains a retarder and is used in warm weather to reduce the rate of hardening and to avoid cold joints.

Calcium chloride was not used under any circumstances.

15.3.1.3 **Water**

Mixing water was obtained from a deep well and was kept clean and free from injurious amounts of oils, acids, alkalies, salts, organic materials, or other substances deleterious to concrete or steel. The quality of the water was the equivalent of that suitable for drinking. The water was continuously checked and tested for compliance with the above requirements by an independent testing laboratory.

15.3.1.4 **Aggregates**

Fine and coarse aggregates conformed to the requirements of the Standard Specifications for Concrete Aggregates ASTM C33. Aggregates were evaluated for potential chemical alkali reactivity. Aggregates were free from any materials that could have been deleteriously reactive in any amount sufficient to have caused excessive expansion of mortar or concrete. All aggregates were tested for compliance with the above requirements by an independent testing laboratory.

15.3.1.5 **Proportioning**

Proportioning of structural concrete conformed to ACI 301, Chapter 3. Working-stress-type concrete and ultimate-strength-type concrete conformed to the requirements of ACI 301, Paragraph 302. Ultimate-strength-type concrete was used in the construction of the foundation mat, exterior wall, and dome of the reactor containment. In general, working-stress-type concrete was used for other areas. Concrete mixes had a 28-day specified strength of 3000 psi, except as otherwise noted on the engineering drawings.

Proportions of ingredients were determined and tests conducted by an independent laboratory in accordance with the method detailed in ACI 301, Paragraph 308, for combinations of materials established by trial mixes.

The maximum slump of mass concrete, as defined in ACI 301, Chapter 14, in general did not exceed 3 inches. Slump of other concrete conformed to ACI 301, Paragraph 305. The samples for the slump tests were taken at the end of the last conveyor, chute, or pipeline before the concrete was placed in the forms.

The close and complex spacing of reinforcing steel in the heavily reinforced sections surrounding the equipment and personnel hatches results in the use of concrete with a maximum slump of 5 inches. The results of strength tests indicate that the 5-inch slump concrete will have a minimum compressive strength of approximately 4000 psi at 28 days. This is considerably higher than the nominal stipulated value of 3000 psi at 28 days used for design purposes, and demonstrates that the structural strength of the containment would not be jeopardized by the use of concrete with a slump of 5 inches.

15.3.2 Reinforcing Steel

Except for the No. 14 and No. 18 reinforcing bars for the foundation mat, exterior wall, and dome of the containment structure, all reinforcing conforms to Grade 40 (or higher strength steel) of the Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement ASTM A615.

For No. 14 and No. 18 reinforcing bars and splices for the foundation mat, exterior wall, and dome of the containment structure, see Section 15.5.1.9. See Section 15.5.2.3 for the description of the reinforcing steel used for the Reactor Pressure Vessel Head Replacement Project.

Mill Test Reports showing chemical and physical properties were obtained and evaluated for each heat of steel used in making all reinforcing steel furnished.

15.4 CONSTRUCTION PROCEDURES AND PRACTICES

See Section 15.5.2 for the description of the restoration of the construction opening used for the Reactor Vessel Head Replacement Project.

15.4.1 Codes of Practice

Materials and workmanship conformed to the following codes and specifications:

ACI 301-66	*Structural Concrete for Buildings* and all specifications of the American Society for Testing and Materials referred to in Section 105 and declared to be a part of ACI 301-66 as is fully set forth therein.
ACI 304	Recommended Practice for Measuring, Mixing, and Placing Concrete.
ACI 305	Recommended Practice for Hot Weather Concreting.
ACI 306	Recommended Practice for Cold Weather Concreting.
ACI 318-63	Building Code Requirements for Reinforced Concrete.
ACI 347	Recommended Practice for Concrete Formwork.

See Section 15.5.2.1 for the description of the codes and specifications used for the restoration of the construction opening used for the Reactor Pressure Vessel Head Project.

Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels was used as a guide in the selection of materials, design stresses, and fabrication of the steel containment liner.

ACI 301-66, *Specifications for Structural Concrete for Buildings,* together with ACI 347-63, *Recommended Practice for Concrete Formwork*, and ACI 318-63, *Building Code Requirements for Reinforced Concrete*, formed the basis for the concrete specifications.

ACI 301-66 was supplemented as necessary with mandatory requirements relating to types and strengths of concrete, including minimum concrete densities, proportioning of ingredients, reinforcing steel requirements, joint treatments, and testing agency requirements.

Admixtures, types of cement, bonding of joints, embedded items, concrete curing, additional test specimens, additional testing services, cement and reinforcing steel mill test report requirements, and additional concrete test requirements were specified in detail.

Concrete protection for reinforcement, preparation, and cleaning of construction joints, concrete mixing, delivering, placing, and curing, with the following exceptions, equaled or exceeded the requirements of ACI 301:

Section 1404 (a) - Maximum slump was generally restricted to 3 inches to permit placing concrete in the heavily reinforced containment structures. The slump was increased to 5 inches in the areas of the containment wall adjacent to the equipment and personnel hatches where the large

steel inserts and additional reinforcing steel required a more plastic mix for adequate concrete placement. All concrete mixes were designed and tested before use. All concrete mixes used in the work were fully documented.

Section 1404 (b) - Maximum placing temperature of the concrete when deposited conformed to the requirements of ACI 305-59, *Recommended Practice for Hot Weather Concreting*.

Section 1404 (c) - Minimum placing temperature of the concrete when deposited conformed to the requirements of ACI 306-66, *Recommended Practice for Cold Weather Concreting*.

15.4.2 Concrete

Concrete ingredients were batched in a batch plant and transferred to transit mix trucks for mixing, agitating, and delivering to the point of placement. Water was added to the mix with the other ingredients before the truck left the batch plant area. Batching and mixing otherwise conformed to ACI 301, Chapter 7.

Placing of concrete was by bottom-dump buckets, concrete pumps, or by conveyor belt. Bottom-dump buckets did not exceed 4 yd^3 in size. The discharge of concrete was controlled so that concrete could be effectively compacted around embedded items and near the forms.

For placing of concrete for the wall and dome of the containment structure, see Section 15.5.1.10. See Section 15.5.2.4 for the description of the concrete used for the Reactor Pressure Vessel Head Replacement Project.

Vertical drops greater than 6 feet for any concrete were not permitted, except where suitable equipment was provided to prevent segregation. All concrete placing equipment and methods were subjected to the approval of the structural engineer.

The surfaces of all construction joints were thoroughly treated to remove all laitance and to expose clean, sound aggregate. Surfaces of fresh concrete were roughened by cutting with an air-water jet after the initial concrete set had occurred, but before the concrete had reached its final set. After cutting, the surface was washed and rinsed. Where the use of an air-water jet was not advisable in any specific instance, then that surface was roughened by hacking with hand tools or other satisfactory means to produce the requisite clean surface.

Before placing subsequent concrete lifts, the surfaces of all construction joints were thoroughly cleaned and wetted, and all excess water that was not absorbed by the concrete was removed. Horizontal construction joints were then covered by a 0.50-inch-thick layer of sand/cement grout of the same sand/cement ratio as the concrete, and new concrete was then placed immediately against the fresh grout.

Curing and protection of freshly deposited concrete conformed to ACI 301, Chapter 12, using curing compounds conforming to ASTM C309.

For curing of the top surface of the containment foundation mat, see Section 15.5.1.10.

Concrete strength tests were performed in accordance with ACI 301, Chapter 16, Section 1602 (a), Paragraph 4, supplemented as follows.

No fewer than two sets of compression test specimens for each mix design of concrete placed were taken during the first two days of placing concrete, or at least one set of test specimens for each 250 yd^3 placed. Thereafter, one set of test specimens was taken for each 250 yd^3, or fraction thereof, for each mix design of concrete placed in any one day. In addition, one set of specimens was taken whenever, for any reason, the materials, methods of concreting, or proportioning were changed.

The test specimens for compressive strength were cylinders 6 inches in diameter and 12 inches long. Each set consisted of five specimens, at least one of which was tested at 7 days and three at 28 days age. The remaining cylinder was retained at the laboratory for further tests at 60 days age if the result of the previous tests made such a test desirable.

Concrete strength tests were evaluated by the engineers in accordance with ACI 214-65, *Recommended Practice for Evaluation of Compression Test Results of Field Concrete*, and ACI 301-66, Chapter 17.

Strengths of working-stress-type concrete were considered satisfactory if the average of any five consecutive strength tests of the laboratory-cured specimens at 28-days age was equal to or greater than the specified compressive strength, f'c, of the concrete.

Strengths of ultimate-strength-type concrete were considered satisfactory if the average of any three consecutive strength tests of the laboratory cured specimens at 28-days age was equal to or greater than the specified compressive strength, f'c, of the concrete.

If any tests for individual cylinders or group of cylinders failed to reach the specified compressive strength, f'c, of the concrete, the responsible engineers were immediately notified to determine if further action would be required.

The field tests for slump of Portland cement concrete were in accordance with ASTM C143. Any batch not meeting specified requirements was rejected.

Slump tests were made frequently during concrete placement and each time concrete test specimens were made.

Statistical quality control of the concrete was maintained by a computer program. This program analyzed compression test results reported by the testing laboratory in accordance with methods recommended by ACI 214, *Recommended Practice for Evaluation of Compression Test Results of Concrete*.

15.4.3 Reinforcing Steel

Placing of reinforcing steel conformed to the requirements of Chapter 5 of ACI 301, *Structural Concrete for Buildings*, and Chapter 8 of ACI 318, *Building Code Requirements for Reinforced Concrete*. See Section 15.5.2.3 for the description of the placement of the reinforcing steel used for the Reactor Pressure Vessel Head Replacement Project.

All Cadweld splices were made in accordance with the instructions issued by the manufacturer, Erico Products, Inc., Cleveland, Ohio.

In order to qualify operators for making Cadweld process joints, each operator was required to demonstrate to the Senior Quality Control Engineer his ability to make an acceptable fixed joint using the Cadweld process. Cadwelders were requalified after every 200 Cadwelds. Testing was by tensile testing a Cadweld made under simulated field conditions.

The ends of the reinforcing steel bars to be joined by the Cadweld process were square cut by the fabricator. Ends of the bars were then thoroughly cleaned of all rust, scale, grease, oil, water, or other foreign matter before the joints were made.

Welding was performed using the “Metallic Arc Welding Process” with coated electrodes, or the “Metallic Inert Gas Shielding Welding Process” (MIG) using bare wire. The filler metal for the Metallic Arc Welding Process conformed to ASTM A316, *Coated Arc Welding Electrodes*, Classification E-10016-D2 or E-10018-D2.

The filler metal for the MIG welding process was a spooled bare wire 0.30 inch or 0.35 inch in diameter, Linde or Arcos Type 515. The shielding gas used for the MIG welding process was Linde C-25, a mixture of 75% argon and 25% carbon dioxide.

The ends of the bars to be joined by butt welding were prepared by sawing or flame cutting, and dressing by grinding, where necessary, to form a single vee butt joint.

Mill test reports of the heats of steel used for making the rebars were obtained by the Senior Quality Control Engineer to confirm the grade of steel welded. Where preheating was required, temperatures were checked with Tempilstiks.

In order to qualify welders for work on the reinforcing steel bars, each welder made a reinforcing bar test weld in the horizontal fixed position, welding vertically up. Each test weld was sectioned through the center of the weld by power sawing and machining. The cross-sectioned surface was etched with a 10% solution of nitric acid and water. The etched surface was examined by the field welding supervisor, who determined the qualification of the welding operator.

Tack welding of rebar was not permitted.

Special criteria for placing reinforcement for the containment structure are provided in Section 15.5.1.6.

15.4.4 Construction Procedures

The portion of the site to be covered by structures was cleared, and general excavation performed to the underside of the foundations for the various buildings. In general, this excavation was from elevation +34 to +10, with some building foundations slightly higher or lower. The major Class I structures (except the Fuel Building and main steam valve enclosure structures) are supported on mat foundations; the Fuel Building and the main steam valve enclosure structures are supported on pile foundation. For additional construction procedures for other Class I structures, see Section 15.6.1.

15.4.5 Construction Practice

Vepco maintained quality control personnel on the site at all times to serve as qualified inspectors in all phases of work, so as to ensure and document that all construction operations met the rigid requirements of the specifications as outlined in the quality assurance report. The qualification of welding procedures and welders was performed in accordance with Part A of Section IX of the ASME Boiler and Pressure Vessel Code or, for structural steel, in accordance with American Welding Society requirements.

Concrete was sampled and tested during construction, in accordance with ACI 318, to ensure compliance with the specifications. A competent independent testing laboratory was retained to design the concrete mixes, take samples, perform all tests of aggregates and concrete cylinders, and report to Vepco for approval.

Special practices to be followed for the containment liner are contained in Section 15.5.1.8.

15.4.6 Quality Assurance Program (Construction Phase)

The descriptions of the quality assurance program during the construction phase have been deleted. These activities have been completed and the descriptions are no longer needed for the operational phase. The NRC-approved Operational Quality Assurance Program is described in Chapter 17.

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15.5 SPECIFIC CONTAINMENT STRUCTURAL DESIGNS

15.5.1 Containment Structure

15.5.1.1 General

For arrangement of the containment structure, see Reference Drawings 1 through 7.

Each of the reactor containment structures is similar in design and construction to that of the Connecticut Yankee Atomic Power Plant at Haddam, Connecticut. Each is a steel-lined, heavily reinforced concrete structure with vertical cylindrical wall and hemispherical dome supported on a flat base mat. Below grade, the containment structures are constructed inside a cofferdam of steel sheet piling. The structures are soil-supported. The base of the foundation mats is located approximately 66 feet below finished ground grade.

Each containment structure has an inside diameter of 126 ft. 0 in. The spring line of the dome is 122 ft. 1 in. above the top of the foundation mat. The inside radius of the dome is 63 ft. 0 in. The interior vertical height is 185 ft. 1 in., and the base mat is 10 ft. 0 in. thick. The steel liner for the wall is 3/8-inch thick, except over the base mat, where 0.25-inch and 0.75-inch plate is used. The steel liner for the dome is 0.50-inch thick. A waterproof membrane, as shown in Figure 15.5-1, is placed below the containment structural mat and carried up the containment wall to ground level. Attached to and entirely enveloping the part of the structure below grade, the membrane protects the structure from the effects of ground water and the steel liner from external hydrostatic pressure. Ground water immediately adjacent to the containment structure is kept below the top surface of the foundation mat by pumping as required.

Access to the containment structure is provided by a 7 ft. 0 in. i.d. personnel hatch penetration, and a 14 ft. 6 in. i.d. equipment hatch penetration. Other smaller containment structure penetrations include hot and cold pipes, main steam and feedwater pipes, fuel transfer tube, and electrical conductors.

The reinforced concrete structure has been designed to withstand all loadings and stresses anticipated during the operation and life of the unit. The steel lining is attached to and supported by the concrete. The liner functions primarily as a gastight membrane, and transmits incident loads to the concrete. The containment structure does not require the participation of the liner as a structural component. No credit has been taken for the presence of the steel liner in designing the containment structure to resist seismic force or other design loads.

The steel wall and dome liners are protected from potential interior missiles by interior concrete shield walls. CRDM missile protection is provided by a concrete shield on Unit 1 and a steel shield on Unit 2. The base mat liner is protected by a 1.50 to 2-foot thick concrete cover, except where a 0.75-inch-thick liner plate was used beneath the reactor vessel incore instrumentation, and at a drainage trench where floor grating provides additional protection.

As an added precaution against water seepage that might penetrate the waterproofing membrane in small quantities, pipe sumps are provided in each of the instrument observation pits located outside the cylindrical wall of the containment but within the waterproofing membrane. The sumps penetrate the base mat and terminate in the porous concrete immediately below the mat.

Pumps are provided to remove ground water outside the waterproofing membrane, as described in Section 15.5.1.3.

15.5.1.2 **Design Criteria**

The design of the containment structures is based on:

1. Biological shielding requirements.
2. The temperature and pressure generated by the design-basis accident (DBA), Section 14.5.2.
3. The operating and design-basis earthquakes discussed in Section 2.5
4. Severe weather phenomena.
5. The maximum calculated power level of 2597 MWt.

The design-basis accident was selected as the design basis for the containment structure because all other bases would result in lower temperatures and pressures. The containment structure is also designed for the normal subatmospheric operating conditions. Further, the containment structure is designated for a leakage rate not to exceed 0.1% of the contained volume per day at 45 psig.

The minimum operating pressure for the containment is 10.1 psia with about 1.0 psia additional partial water vapor pressure. The resulting total containment pressure is approximately 11.1 ± 0.5 psia. The temperature of the containment air fluctuates between a maximum temperature of 125°F and a minimum of 75°F during normal operation, and 60°F during shutdown, depending upon the ambient temperature of available service water. The normal operating pressure allows accessibility for inspection and minor maintenance during operation without requiring containment pressurization or the use of supplementary breathing equipment for personnel.

The containment structure is designed by ultimate strength methods conforming to ACI 318-63, Part IV-B. Design load criteria based on ACI requirements and others given below conform to current containment design.

The ultimate load capacity of the containment structure as modified by the safety provisions of ACI 318-63, Section 1504, is not less than that required to meet the containment structural loading criteria.

Loads imposed on the containment shell design include:

1. Dead load.
2. DBA pressure.
3. Temperature rise in liner associated with DBA.
4. Normal operating temperature gradients.
5. Earthquake.
6. Wind loads, including tornado winds.

Loads imposed on the containment mat design include:

1. Mat and interior structures during construction.
2. Dead load for complete structure and contents.
3. Dead load and DBA pressure and liner loading.
4. Dead load, DBA pressure, liner loading, and earthquake.
5. Dead load and earthquake.

The ultimate load capacity of the containment structure, as modified by the safety provisions of ACI 318-63, Section 1504, is not less than that required to satisfy the following structural loading criteria, tabulated in Table 15.5-1.

The seismic design coefficients and critical damping factors used in the design of the reactor containment structure are given in Section 15.5.1.4. The average acceleration spectra curves are included in Section 2.5. The earthquake loads include the horizontal or vertical acceleration, or a combination of both where the effects, as measured by the stresses resulting from the separate acceleration components, of horizontal and vertical ground accelerations are combined algebraically.

The load capacity of the tension members is based on the guaranteed minimum yield strength of the reinforcing steel. Load capacities of flexural and compression members are determined in accordance with the *Building Code Requirements for Reinforced Concrete,* ACI 318. The load capacity so determined is decreased by a reduction factor multiplier “ϕ”, to compensate for small adverse variations in material and workmanship. The reduction factors are listed in Table 15.5-2.

The load capacity reduction factor for stresses in concrete produced by tornado-carried missiles, in combination with other tornado-produced stresses as given in Loading Criteria 5, is 0.75.

The dominant design load is the 45-psig containment design pressure, which creates major tensile membrane stresses in the reinforcing steel, coincident with moments at the junction of the containment wall and mat.

The design tornado wind loading and pressure drop criteria are stated in Sections 2.2 and 15.2.3.

Since the DBA pressure load is greater than the negative pressure load of tornadoes, the containment structure is able to maintain its integrity and permit an orderly shutdown on the reactor unit should a tornado strike the structure.

15.5.1.3 **Buoyant Loads**

15.5.1.4 **Dynamic Analysis**

Analyses were conducted to determine response stresses in the containment structure due to the application of seismic loading. Earthquake ground motion values were applied simultaneously in the horizontal and vertical directions. Vertical ground motions were assigned a magnitude equal to two-thirds of the horizontal motions. The magnitudes of the operating-basis earthquake and the design-basis earthquake are derived and assigned as described in Section 2.5. Design loading

conditions combined with seismic loading and allowable stress levels are stated in Section 15.5.1.2.

The earthquake loading was analyzed using a Stone & Webster program, *Container Vessel Seismic Analysis,* based upon the dynamic analysis of a containment structure by Messrs. Hansen, Holley, and Biggs of MIT.

The general analytical model of the containment structure responding to horizontal earthquake forces is a coupled two-mass system in which the wall and dome comprise one mass and the base slab and internals comprise the second mass. This model responds to three degrees of freedom: flexure in the wall and dome, translation, and rocking of the structure as a unit. The model includes the first three modes of vibration.

The stiffness of the wall and dome was obtained through formulas recommended by Professor R. V. Whitman of MIT, based on work by G. N. Bycroft.

The output of the computer program was spot-checked by manual analysis, which confirmed the program basis.

Another independent manual analysis that considered the internals as a third coupled mass resulted in loading values that were not greater than those obtained from the analysis of the two-mass system.

A preliminary analysis of response to vertical earthquake forces using a single-mass system showed that these forces are not controlling factors in the design.

When computing the response of the reinforced concrete containment structure to earthquake forces, the value of 5% of critical damping was used with the design earthquake acceleration of 0.07g. This is an overall value that includes the damping in both the reinforced concrete structure and the soil. The magnitudes of earthquake forces applied to the structure were obtained from the response spectrum for 0.07g at zero period and 5% critical damping at the calculated frequency of the structure, and then distributed over the structure in accordance with the relative motions of the structure as determined by dynamic analysis.

The force derived by use of this damping factor was used for the entire reinforced concrete containment. The value of 5% of critical damping, together with the damping factors for other systems, structures, and equipment, is listed in Table 15.2-2.

The value of 10% of critical damping was used with the design-basis earthquake of 0.15g on the basis of increased cracking in the concrete and increased movement in the concrete and soil.

To verify the damping used for design, an analysis of the soil structure interaction damping was made in accordance with the procedures suggested in *Analysis of Foundation Vibrations,* by

Robert V. Whitman, Proceedings of a Symposium organized by the British National Section of the International Association for Earthquake Emergency.

Damping factors for soil were calculated for the rigid body translation and rocking. Flexure damping was assessed as suggested by Newmark.

For each of the four modes of vibration, energy losses in structural flexure, sliding, and rocking were calculated and proportioned to determine the total system energy loss, thereby defining the damping to be used in spectrum response.

This analysis demonstrated that the damping factors used for design and the resulting seismic response characteristics are conservative.

Earthquake load criteria are included in the loading criteria described in Section 15.5.1.2. Operating and design-basis earthquake factors are each combined with other loads, including the design-basis accident pressure. Resulting shears are computed by the computer program.

Lateral earth pressure under seismic loadings on the containment mat was determined by computing the lateral resistance developed in the soil as the structure responds in flexure, translation, and rocking. In this analysis, the translational restraining force has two components, a shear across the base of the structure and lateral soil pressures on the side wall of the containment structure developed by its displacement relative to its static position.

The “spring constant,” that is, force per unit of lateral displacement by shear, for a circular rigid base on an elastic half space is given by Bycroft (Reference 1) as:

$$k_x = \frac{32\ (1 - u)\ Gr_o}{7 - 8u}$$

where:

G = shear modulus

r_o = radius of base

u = Poisson’s Ratio

Note: Values in consistent units

For usual values of u this reduces approximately to:

$k_x = 5Gr_o$

The horizontal pressure on the side wall of the containment structure can be evaluated from the theories of horizontal subgrade reaction. From Terzaghi (Reference 2) the relation between horizontal deflection and pressure at any point is given by:

$$k_h = \frac{P}{y_h}$$

where:

P = horizontal pressure at soil structure interface

y = horizontal deflection of soil at interface

k_h = coefficient of horizontal subgrade reaction

further:

$$k_h = \frac{n_h z}{B}$$

where:

n_h = coefficient dependent upon physical properties of the soil

z = depth below free surface of soil

B = width of loaded area, which may be taken as diameter of containment structure

For purposes of this analysis, a value of n = 40 tons/ft^3 was selected from tables presented by Terzaghi. This value is appropriate to dense sand above the ground water table. It is a conservative value, since the higher the coefficient, the stiffer the soil, and the greater the loads imposed upon the side walls of the structure.

The rotational, translational, and flexural deflections of the structure were determined from response analysis and added so as to obtain maximum deflections. The lateral soil pressures on the side wall of the structure were then computed for these total deflections using the theory of horizontal subgrade reaction.

In determining these pressures, the side wall of the structure was assumed to be rigid radially, since radial deflection of the side wall would reduce relative soil-structure deflections, and thus the soil forces acting upon the structure.

The analysis was performed for both the operating-basis earthquake of 0.07g and the design-basis earthquake of 0.15g. The analysis for a 0.15-g earthquake indicates a lateral force of 300 lb/ft^2 at Elevation -8 ft. 6 in. which defines approximately the magnitude of this component.

It should be noted that these forces, if included in the seismic loadings on the structure, would reduce the base shear and vertical bending stresses in the shell. Accordingly, they are not

included when computing such stresses in the shell and thereby contribute to the conservation of the design.

Rocking motion of the containment structure was considered in the determination of the natural frequency, the distribution of inertia forces, and in the amplitudes of motions.

The containment wales supporting the cofferdam structure do not affect consideration of horizontal pressure under seismic loading on the containment wall.

Four circular concrete wales originally supported the sheet steel cofferdam in which the containment structure is founded. The top wale, Wale A, has been partially removed at several points to permit completion of adjacent structures; in this condition it does not impose any restraint on the containment structure. The bottom wale, Wale D, is in the lower plane of the containment mat and below the plane of the wall, and offers no restraint. Wale C extends from a height of 4 ft. to 8 ft. above the base of the wall. Wale B extends from a height of 17 ft. 6 in. to 21 ft. 6 in. above the base of the wall. These two wales are approximately 3 ft. 9 in. from the containment wall, and the space between the wales and wall is backfilled with pervious fill. Under seismic loading, the distribution of the lateral earth pressure through the cofferdam wales would not have any different effect than if these pressures were applied directly to the structure.

15.5.1.5 **Static Analysis**

The containment structure was analyzed and designed for all loading conditions combined with load factors as outlined in Section 15.5.1.2. The forces, shears, and moments in the structural shell were obtained from a computer program based on *Numerical Analysis of Unsymmetrical Bending of Shells of Revolution*, by B. Budiansky and P. P. Radkowski, published in the American Institute of Aeronautics and Astronautics Journal, dated August 1963.

Forces, moments, and shears in the base slab were obtained from a Stone & Webster computer program, *Flat Circular Mat Foundations for Nuclear Secondary Containment Structures.* The program analyzes a flat circular plate supported on an elastic foundation and computes the discontinuity stresses at the junction of the mat and cylinder, and the soil bearings pressure.

Discontinuity stresses, shears, and moments at the junction of the cylinder and mat were determined using an analogy to the Hardy Cross method for distributing fixed-end moments in continuous frames. The theoretical fixed-end moments obtained from the shell and mat computer analysis were balanced in proportion to the relative stiffness of the mat and cylinder.

An independent, manual computation, based on *Theory of Plates and Shells*, by S. Timoshenko, at a few selected points produces forces, shears, and moments substantially the same as those produced by the computer programs for the shell and the mat.

The containment shell program used to derive stresses in the shell assumes an isotropic material. The program does not include considerations of temperature gradients due to the thermal loadings across the containment wall.

To compute maximum stresses due to the thermal load, six general strain equations were derived, one equation for each of the four principal areas of reinforcing steel and one for each major axis of the steel liner. These equations relate strain to position, temperature, and incident stress for each item considered. To solve these general strain equations, six additional equations were used: four equations for strain compatibility, which equate radial and longitudinal strains, and two equations for load equilibrium. The solution of these equations for incident conditions gives the stress in each of the principal areas of reinforcing steel and the stress on the steel liner.

These equations permit the thermal stresses to be considered separately without modification of the major shell program.

The thermal operating load in the containment concrete wall, combined with incident condition loadings, produces a stress difference of approximately 6000 psi between the reinforcing steel adjacent to the inside face of the wall and the reinforcing steel adjacent to the outside face of the wall. This difference exists in both the longitudinal steel and the hoop reinforcing steel.

To permit the addition of these stresses to those obtained from the containment shell program, without exceeding the maximum, the containment shell program stresses are limited to 3000 psi below the maximum allowable design stress. This approach is considered extremely conservative since it limits the design stress in the interior layers of reinforcing steel to approximately 6000 psi, less than the maximum allowable design stress permitted on the exterior layers of reinforcing steel.

Structural failure cannot occur, however, until the interior reinforcing steel exceeds yield. Up to that point plastic yielding of the outside reinforcing would be controlled by the elastic behavior of the interior steel.

In the solution of the general strain equations, the effect of the concrete has been ignored, since it is assumed to be cracked and incapable of carrying any of the tensile loads considered. The dead load of the concrete is also ignored, as this was found to have little effect on the hoop stresses. This assumption also provides a more conservative result.

The loads exerted on the concrete shell by the thermal effects of the exposed steel liner were obtained from the calculations discussed above. The equivalent pressure, p, equals the hoop stress, f, in the steel liner multiplied by the liner thickness, t, and divided by the radius of the liner, r. The computed equivalent pressure associated with 1.5 times incident pressure equals 5.45 psi.

Stiffness factors were used to distribute computed fixed-end moments derived from an analysis of the containment cylindrical wall, considered as a shell with a fixed-end moment, and

from an analysis of the containment mat, considered as a flat circular plate with uniform fixed-edge moment.

Stiffness factors for the cylinders were computed from formulas given in Raymond J. Roark's book, *Formulas for Stress and Strain*, for long, thin-walled cylinders. Stiffener factors for the mat were computed from formulas for circular flat plates with uniform edge moment, from the same source.

Variation of the modulus of elasticity of the concrete to differentiate between uncracked and cracked concrete was not considered in determining the stiffness factors chosen.

Use of such a variable would modify the distribution of the moments and shear forces to some degree, but it is not believed that this would significantly affect the accuracy of the results. The safety factor inherent in the present design would accommodate such small variations.

The actual distribution of the moments and forces at the junction of the wall and mat are a function of the relative stiffness of each member. This is determined by the design approach used. Provided the total forces are distributed between the two areas under consideration, differences of distribution due to theoretical variations of the theoretical value of Young's Modulus for concrete are not considered likely to improve the results beyond the accuracy obtained with the assumptions already used.

The methods for computing soil pressures under the mat were based upon an analogy to E. P. Popov's *Method of Successive Approximations for Beams on an Elastic Foundation,* published in the Proceedings of the ASCE, Separate No. 18, dated May 1950. The program computes the deflection at the center of the mat relative to a point on the mat at the intersection of the center line of the containment shell walls. The elastic curve of the mat deflection is assumed to be parabolic between these two points. Multiplying the deflection by the subgrade spring constant, the program then provides a parabolic soil pressure curve, which is combined with the rectangular soil pressure curves to provide final soil pressures under the mat. The subgrade spring constant is derived from Professor R. V. Whitman's formula:

$$k = \frac{4G}{\pi(1 - U)R}$$

where:

k = Spring constant

G = Shear modulus of subgrade material

R = Radius of mat

U = Poisson's ratio of subgrade material

While the subgrade reaction varies with depth, a single typical value for the reaction was used which is representative of the zone at the level being considered. The shear modulus was computed using a formula developed by Hardin and Black (Reference 3) from observed soil samples, and substantiated by dynamic triaxial tests of the soil. The stiffness of the soil was also based on work by G. N. Bycroft which is referred to in the Section 15.5.1.4.

A variable soil pressure conforming to the deformation of the mat was used in determining the stresses in the structure.

Maximum wind velocity associated with a tornado is given as 360 mph. This velocity was converted to an equivalent pressure using the formula $P = .00256V^2$, where P = equivalent pressure, lb/ft^2 and V = wind velocity, mph. Wind pressure was distributed over the containment dome in accordance with the methods given in *Wind Stresses in Domes,* by P. Gondikas and M. G. Salvadori, published in ASCE proceedings No. 2616, dated October 1960.

Wind pressure was distributed over the containment cylindrical shell in accordance with the methods given in *Wind Forces on Structures*, by T. W. Singell, published in ASCE proceedings No. 1710, dated July 1958.

Tornado wind loads were combined with other loads as described in Section 15.5.1.2.

An analysis of the containment structure indicated that resulting membrane stresses due to tornado wind loading in the dome reinforcing are less than 5000 psi, and that discontinuity stresses at the junction of the dome and cylinder are somewhat less.

The wind loading on the cylindrical shell creates bending, direct and shear stresses. The bending and direct stresses in the horizontal reinforcing equal 16,000 psi.

An investigation of overturning due to wind shows that the resultant (DL + wind) falls within the Kern point radius of the cylinder, indicating that the vertical reinforcing will not be subject to tensile forces from this load.

Containment torsional loadings from wind were considered negligible, in view of the ideal shape of the containment when considered as a torsion resistant shell supplemented by the diagonal reinforcing throughout the walls provided to resist earthquake loads.

The Stone & Webster computer programs for the reactor containment base slab, cylindrical wall, and dome use a constant Young's Modulus of Elasticity and Poisson's Ratio. No attempt was made to assign varying numerical values to these factors to differentiate between the relative amount of cracking in different parts of the structure.

The output of the mat program furnishes the following information:

1. Radial and tangential bending moments and vertical shear at five-foot intervals along horizontal radii from the center of the mat, spaced at 30-degree intervals.

2. Discontinuity stresses at the junction of the mat and cylinder.
3. Soil pressure.

The output of the shell program furnished forces, shears, and moments at 1-foot intervals in the height of the cylindrical wall, and at one-degree intervals in the height of the dome. Similar information is furnished at each of 16 equidistant points on the circumference of the vessel at each level considered.

Scaled load plots obtained from the computer programs for moment, shear, deflection, longitudinal force, and hoop tension are shown in Figure 15.5-2 for each of three design load conditions. The fourth design load condition did not govern design and is not represented.

The following assumptions were made:

a. The dead and live structural loads are included in all three of the design load cases.
b. Pressure load, factored and unfactored, is the dominant load condition.
c. Wind loading replaces earthquake loads where wind loads exceed earthquake loads.
d. Tornado loads are included under the general category of wind loads discussed above.
e. Buoyant water loads as discussed in Section 15.5.1.3 are substantially less than dead loads.
f. Earthquake loads, both for the operating-basis earthquake and the design-basis earthquake, are included in the analysis.
g. Thermal load from the liner is converted into an equivalent pressure and added to the incident pressure load when computing moments, shear, and tension associated with the design-basis accident.
h. Thermal load from the concrete is discussed in Section 15.5.1.5. Stresses resulting from this load are combined with incident pressure load stresses.

15.5.1.6 **Reinforcing Steel Arrangement**

The foundation mat of the containment structure is reinforced with both top and bottom layers of reinforcing. Bottom mat reinforcing is placed in a rectangular grid pattern with layers at 90 degrees to each other. Reinforcing for the top of the mat consists of concentric circular bars combined with radial bars. The reinforcement pattern for the top of the mat is arranged to permit maintaining a uniform spacing of the vertical wall rebars that extend into the mat. Splices in adjacent parallel rebar in the mat are in general not less than 4 feet apart.

Hoop tension in the cylinder wall is resisted by horizontal bars located near both the outer and inner surfaces of the wall. All horizontal circumferential bars, including those in the dome, have their joints staggered at a minimum of 3 feet apart.

Longitudinal tension in the cylinder wall is resisted by two rows of vertical bars, one near the interior face and the other near the exterior face of the wall. Vertical bars are placed in groups of 20 bars of equal length. These are arranged so that no adjacent group in the same or opposite face of the wall has splices closer than 6 feet vertically.

See Section 15.5.2.3 for the description of the splicing scheme used for the Reactor Pressure Vessel Head Replacement Project.

The dome reinforcing consists of layers of rebar placed radially extending from the vertical reinforcing of the cylindrical wall and horizontal layers of circumferential hoop bars. Layers are located near both the inner and outer faces of the concrete. The radial pattern of the reinforcing steel terminating in the containment dome results in a high degree of redundancy of reinforcing steel in the dome. Bars are terminated beyond a point where there is more than twice the amount of steel required for design purposes. The rate of convergence of these bars, and low-stress requirements dictated by the arrangement, produces a low bond stress. In a limited number of cases where bars are terminated close to the center of the dome, anchorage stresses are more critical, and bars are hooked to provide the required anchorage. Near the crown, the rebars are welded to a concentric ring cast in the concrete.

Radial shear loads generated by internal pressure resulting from the design-basis accident are resisted by rebars inclined at 45 degrees with the horizontal and extending between the surfaces of both the vertical reinforcing closest to the interior and exterior faces of cylinder wall. This radial shear will vary from a maximum at the base of the wall where the foundation mat restrains the independent movement of the wall to zero at some level above the mat. Anchorage bond stresses in these shear bars is kept below allowable stress levels to minimize potential cracking of the concrete. In addition, sufficient longitudinal and circumferential reinforcing is carried to the base of the wall to carry all potential loads without assistance from the radial shear reinforcing.

The tangential shears resulting from the earthquake loading are resisted by rebars inclined at approximately 45 degrees in each direction, in the plane of the wall parallel to the main reinforcing steel.

Minimum concrete cover for all principal reinforcing steel of the containment structure exceeds the requirements of ACI 318, Paragraph 808(d), which states, “Concrete protection for reinforcement shall in all cases be at least equal to the diameter of the bars.” The largest and principal reinforcing bar is No. 18, which requires a minimum cover of only 2-3/8 inches by the code.

15.5.1.7 **Penetration Design**

Penetration through the containment structure is divided into one of the following three categories:

1. Pipe penetrations nine inches in diameter or less.

   No special structural reinforcing is provided for penetrations nine inches in diameter or less. Penetrations in this category are located to avoid interference with the reinforcing steel.

2. Pipe penetrations greater than nine inches and up to 3 ft. 6 in. in diameter.

   For penetrations greater than nine inches, and up to and including 3 ft. 6 in. diameter, supplementary reinforcement is provided in amount and distribution such that area requirements for reinforcement are adequately satisfied.

   At all these size penetrations, reinforcing steel interrupted by the openings is terminated at each side of the opening. Supplementary reinforcing was placed parallel to the interrupted bars to provide bar continuity. Horizontal, diagonal, and vertical bars were used to effectively frame the opening. The total area of reinforcement provided in any plane is not less than twice the area of steel interrupted or cut by the opening, with half of this placed on each side of the opening.

   Additional reinforcing around these openings is not less than 20 feet in length, and of sufficient length to develop the full ultimate strength of the bar in ultimate bond stress to conform to the requirements of ACI 318, Section 1801(C 2). Horizontal bars are considered as top bars for this purpose.

3. Openings larger than 3 ft. 6 in. in diameter.

   The two openings in this category are the 7 ft. 0 in.-diameter personnel access hatch and the 14 ft. 6 in.-diameter equipment access hatch. Details of the additional reinforcement provided around the equipment access hatch and personnel access hatch are shown in Figures 15.5-3 through 15.5-6, inclusive.

These penetrations are analyzed by means of a computer program (Reference 4). This program analyzes a ring beam based on the method of virtual work. The program assumes the ring beam to be isolated from the containment shell and loaded in two planes. The analysis includes the effect of the stiffened ring and the moments introduced by transferring external loads from the shell at the perimeter of the ring to the center line of the beam.

The ring beams are designed to resist biaxial bending moments, axial tension, torsion, and biaxial shear resulting from loading criteria listed in Section 15.5.1.2. The biaxial bending moments and axial tension are assumed to be resisted by the reinforcing bars only, the concrete being neglected. The torsional and biaxial shear stresses are assumed to be resisted entirely by binders placed radially around the penetrations. Torsion is computed by the formulas for torsion in a rectangular beam. The principal circumferential and meridional reinforcing is extended to the

inner face of the ring beam and bent at right angles, hereby providing additional shear resistance, the availability of which is considered in the design.

The normal pattern of membrane stress in the cylinder wall is interrupted in the area adjacent to the stiffened openings. This redistribution of stress was investigated by means of a computer program, based upon a paper by B. Budiansky and P. Radkowski (Reference 5). For this investigation, a flat circular plate with a radius equal to three times the distance from the center of the opening to the outside face of the stiffening ring beam was used to establish the stress pattern. The movement of both the stiffened ring and the adjacent shell was compared to determine if significant discontinuity stresses were present. Extra reinforcement was added to regions of marked deviation from the normal pattern to keep the discontinuity effects to the level at which they can be considered negligible. The gross concrete area of the ring section was used to determine the section stiffness and rigidity.

15.5.1.8 **Steel Liner and Penetrations**

The containment structure has an inside diameter of 126 ft. 0 in., and an interior vertical height of 185 ft. 1 in., measured from the top of the foundation mat to the center of the dome. The cylindrical steel wall liner is 3/8 inch thick, the hemispherical dome liner plate is 0.50 inch thick, and the flat base liner is 0.25 inch and 0.75 inch thick.

See Section 15.5.2.2 for the description of the restoration of the steel liner for the Reactor Pressure Vessel Head Replacement Project.

The top of the containment dome at Surry has a 39-inch penetration that was used during construction. This penetration is sealed by a welded plug on the liner side and a bolted plate on the outer end, and is filled with sandbags.

The steel lining is attached to and supported by the concrete; the liner functions primarily as a gastight membrane. The steel wall and dome liner are protected from potential interior missiles by interior concrete shield walls. CRDM missile protection is provided by a concrete shield on Unit 1 and a steel shield on Unit 2. The base liner is protected by a 1.50- to 2-foot-thick concrete mat, except in two areas where 0.75-inch-thick liner plate is used beneath the reactor vessel incore instrumentation, and at a drainage trench where floor grating provides additional protection.

The steel liner is designed to withstand the effects of all temperature, earthquake, and pressure loads, including the effect of the subatmospheric operating pressure.

The liner stress limits and their associated strains are limited to the stress criteria given in Paragraph N-1314 of Section III of the ASME Boiler and Pressure Vessel Code for nuclear vessels, and to basic primary stress levels taken from Table N-421 of that Code. The liner material SA-442-GR60 has a specified NDT that is at least 80°F below the minimum liner operating temperature, and considerably more than 120°F below the design-basis accident temperature. Under either of these conditions, the liner material is able to accommodate at least 60% plastic

strain without cracking. A strain of this magnitude is at least 80 times greater than the maximum strain that will be imposed on the liner.

Reference to a generalized fracture analysis diagram shows that the "Crack Arrest Temperature" (CAT) curve crosses the NDT +80°F line at approximately 60% of the span between the "Fracture Temperature Elastic" (FTE) and the "Fracture Temperature Plastic" (FTP) ordinates, indicating that the steel can be strained to 60% of the required strain to fracture without cracking, even in the presence of large flaws.

To demonstrate that this plate material can accommodate plastic strains of this magnitude when biaxially stressed, tests were conducted on samples of 3/8-inch-thick plate of identical specification to the steel to be used in this containment liner, at a temperature of 90°F above the NDT of the steel. In these tests, the plate samples were each laid across a 23-inch-diameter ring, and a 3-5/8-inch-diameter mandrel was forced into the plate at the center line of the ring. In all cases, the mandrel deformed the plate by an amount in excess of 4 inches before shearing through.

The design-basis earthquake can be expected to produce tremors to the extent of not more than 8 to 10 cycles, and the operating-basis earthquake not more than 4 to 5 cycles.

Operating pressure variations from 9.5 psia to 14.7 psia can be expected to occur not more than 150 times during the lifetime of the unit, since personnel access is permitted under subatmospheric conditions. Temperature variations from 70°F to 105°F resulting from seasonal swings and shutdowns of the unit can be expected to occur not more than 600 times during the lifetime of the unit.

The containment liner is designed for 1500 cycles of operating pressure variations, 6000 cycles of temperature variation, and 20 cycles of design-basis earthquake, all simultaneously applied.

The containment liner is also designed for one cycle of design-basis accident pressure, one cycle of design-basis accident temperature, and ten cycles of design-basis earthquake, all considered simultaneously applied.

The containment liner is designed, within allowable working stresses, to withstand a vacuum increase of not less than 1.5 psi. The shell and dome plate liner is capable of withstanding an internal pressure of 3 psia, and the bottom mat liner is capable of withstanding an internal pressure of 8 psia, with reference to standard atmospheric conditions outside the containment.

The change in barometric pressure due to tornadoes is not expected to exceed 3 psig, and the change due to maximum hurricane will be approximately 1.1 psig. These pressure changes will result in a decrease in the atmospheric pressure, which will decrease the differential between atmospheric pressure and the containment structure ambient pressure, thereby decreasing the potential for stresses in the containment.

The accumulated effects of the above are evaluated in accordance with Paragraph N-415.1 of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The steel containment liner is securely anchored to the concrete wall and dome with Nelson stud-type concrete anchors. Failure could occur by stud failure in shear or tension, by studs pulling out from the concrete, or by studs tearing off from the liner plate. Tests conducted by Northeastern University, Boston, Massachusetts, using 1/2-inch-diameter studs and 3/8-inch-thick plate, show that shear failure occurs in the stud adjacent to the weld connecting the stud to the plate; in no instance was the plate damaged. Tests conducted for the stud manufacturer under the direction of Dr. I. M. Viest indicate that, with the manufacturer's recommended depth of embedment of the stud in concrete, the ultimate strength of the stud material can be developed in direct tension.

The principal design load imposed on the studs is due to the subatmospheric pressure operating condition, with the anchor lattice spacing based on considerations of plate buckling. A safety factor greater than 10 is provided against stud failure in tension.

Shear due to design-basis accident conditions and earthquake will result in stresses less than the allowable working stresses.

In addition to the concrete stud anchors, the wall and base mat sections are anchored and joined at the intersection of the vertical wall and the base mat with a continuous steel skirt embedded and anchored in the concrete.

All anchors are designed so that failure occurs in the anchor, thereby assuring that the leaktightness of the containment liner will be maintained during and after anchor failure.

Probable mode of failure will be one of random stud failure due to poor workmanship during stud attachment. This type of failure will result in separation of the stud from the liner without impairment of the liner ductility or integrity.

Loss of random anchor points will not trigger a chain reaction, since the design load on each stud is low compared with the stud load capability. Design spacing of these studs is such that a group of at least 10 adjacent studs would have to fail to cause a liner plate to reach its yield stress under design operating conditions. Even with this unlikely condition, the loads on the studs adjacent to this area would remain within their safe load capability.

As shown in Figure 15.5-7, the liner was welded to a skirt ring which in turn is embedded and anchored into the concrete mat. The skirt-to-liner juncture and the skirt-to-mat anchorage were proportioned to develop the full strength of the liner. Under DBA conditions, the liner at the base juncture will be under a state of biaxial compressive strain, due primarily to thermal effects.

All thermally hot pipes penetrating the reinforced concrete containment wall pass through individual sleeves that are approximately 1 foot in diameter larger than the pipe, and project inward a distance of approximately 2 feet from the liner. A typical application is shown in

Figure 15.5-10. The pipe is welded to a thick cap that is an integral part of the end of the penetration sleeve.

Each penetration sleeve with a thermally hot pipe penetration is equipped with two water-cooled heat exchangers to limit the temperature of the liner and the concrete in contact with the sleeve. One heat exchanger is located inside the sleeve encompassing its length (inner unit); the other is located outside the penetration sleeve in proximity to the liner. Either of the heat exchangers will provide adequate cooling for the penetration if the other is out of service. The associated component cooling water system has two independent lines. One line circulates water through the outer unit; the other circulates water through the inner unit. The inner unit limits the radial heat flow resulting from convection and thermal radiation from the thermally hot pipe penetration, to keep the temperature of the concrete in contact with the sleeve within allowable limits. In addition, the inner unit controls the longitudinal heat flow resulting from conduction from the same heat source, thus limiting the temperature of the liner and temperature gradient along the sleeve to keep the resulting thermal stresses in the liner and sleeve within the limits set forth in Section III of the ASME Pressure Vessel Code. The outer unit also limits the longitudinal heat flow, providing independent thermal protection of the penetration sleeve and liner.

The circumferential groove in the attachment plate, between the sleeve and penetration with its outside threaded connection, serves as a test chamber for the testing of the welds joining the attachment plate and penetration.

All penetrations are anchored in the reinforced concrete containment wall. The anchor strength is equal to the full yield strength of the pipe with regard to torsion, bending, and shear, and to the maximum possible pipe jet reaction. All stresses induced in the liner by these combinations of loadings are only those reflected by the resulting distortions in the reinforced concrete containment wall, and are minor in intensity. So, loads will not be imposed on the liner, thereby preserving its integrity.

All highly stressed insert plates at penetrations and equipment supports that are welded into the liner to transfer loads into the concrete have been ultrasonically tested to check for possible laminations. Tests were conducted on all plates where analysis showed a higher than average stress field, although all such plates are stressed well below the allowable limits for the materials. These tests show that no faults exist in the insert plates.

The pipes anchored to the containment penetrations between containment isolation valves constitute an extension of the containment, and are designed in accordance with the *USA Standard Code for Pressure Piping - Power Piping,* USAS B31.1.0-1967, with respect to materials and allowable stress. Analyses of stresses due to thermal expansion and shock loadings from earthquake, pipe jet reaction, and other causes were made using established digital computer calculation techniques.

In order to determine the loading combinations that act on a penetration, the pipe line passing through the penetration sleeve was assumed to have failed transversely at several

locations along its run. The location at which the reaction of the ensuing jet of fluid flowing from the broken end first causes the pipe to completely yield, in either bending or torsion, was taken as the design case from which all resultant combinations of penetration loading were determined for that particular pipe line. The maximum stress allowed on any individual element of the penetration is 90% of the minimum yield point.

The intent of this criterion is to keep the material assembly components within the elastic range of the material. Under operating conditions of pressure, temperature, and external loads, the stresses in the assembly will be within the limits established in Section III of the ASME Pressure Vessel Code.

As a part of the issues identified in NRC GL 96-06, isolated containment penetration piping with confined fluid was reviewed for susceptibility to thermal over-pressurization following a DBA. The linear elastic analysis criteria stipulated in the 1989 version of the ASME Boiler and Pressure Vessel Code Section III, Appendix F was used for structural integrity evaluation. The internal pressure in piping penetrations during a design basis accident (LOCA or MSLB) was calculated by taking into account the difference in the expansion of the fluid and the pipe, the temperature increase immediately following the DBA and credit for a limited amount of circumferential strain in the pipe. The analysis established that thermally induced over-pressurization of isolated water-filled piping sections in the containment boundary could not jeopardize the ability of the accident mitigating systems to perform their safety functions and could not lead to a breach of containment integrity (Reference 10).

All liner seams were strength-welded. Small steel channels welded continuously along the edges of their flanges to the liner plate cover the plate weld seams, in a manner similar to those installed at the Connecticut Yankee Station. These channels are zoned into test areas by dams welded to the ends of the sections of the channels. Fittings are provided in the channels for periodic testing of the weld seams for leaktightness under pressure. Typical liner details are shown in Figure 15.5-12. Testing of the liner is described in Section 5.5.

To transfer the stress adequately around penetration openings, or to transfer the pipeyield load adequately to the concrete within the limits of this material, whichever is larger, the liner is reinforced in accordance with the rules set forth in the ASME Boiler and Pressure Vessel Code, 1968, Section III, Nuclear Vessels.

All major equipment and pipe loads are carried on the interior concrete structure or by the neutron shield tank. A 1.50- to 2-foot-thick concrete slab placed over the bottom mat steel liner provides anchorage and support for other equipment located in the base of the containment structure. The neutron shield tank skirt is attached to the containment mat by 1.50-inch-diameter anchor bolts. The skirt support was welded to the liner, and the entire weld, including the anchor bolts, covered by test channels. The internal concrete structure is attached to the containment mat by lengths of 3-inch by 6-inch steel bars which, placed horizontally, intersect the steel plate liner as shown in Figure 15.5-8. The main vertical reinforcing steel bars were welded to the top and

bottom faces of these bars, thus providing bar continuity without creating multiple penetrations through the liner.

The 1.50-foot-thick concrete slab is anchored through the steel liner plate in a similar manner using 7-inch by 0.50-inch bars, as shown in Figure 15.5-8. These bars, termed bridging bars, form an integral part of the steel liner, and conform to the material and workmanship specifications of the steel liner. All welded joints are covered by test channels and tested as all other liner plate joints.

Access to the containment structure is provided by a 7 ft. 0 in. i.d. personnel hatch and a 14 ft. 6 in. i.d. equipment hatch. Other smaller containment structure penetrations include hot and cold pipes, main steam and feedwater pipes, fuel transfer tube, and electrical conductors.

Electrical conductors penetrating the containment structure range in size from No. 16 AWG thermocouple leads to 1-inch-diameter solid copper rods for 4160V power circuits. Each penetration group passes through 8-inch-diameter steel sleeves. The sleeves were welded into the containment liner with a test channel around the weld for periodic leak testing, as shown in Figure 15.5-9 (Amphenol electrical penetration depicted).

The basic Amphenol electrical penetration consists of an eight-inch steel tube with bolted-on flanges, through which pass the sealed conductors. The hermetically sealed connectors, as shown in Figure 15.5-9, were bench-tested for leaktightness.

Each flange is held tightly in place with eight bolts that draw the flange against a high temperature sealing ring and a backing plate welded to the sleeve. Each flange is tapped for leak testing. A make-up method is used to determine the penetration leakage by applying a test pressure equal to greater than containment design pressure (45 psig) between the o-ring seals. An electrical connector may be replaced, if necessary, without welding or cutting the containment liner or sleeve.

The design and qualifications of the Amphenol electrical mating connectors are based upon the requirements of military specification number MIL-C-5015. Connector design is such that silastic components are provided in the connector to feed through the interface. This type of interface has been proven adequate to meet the environmental requirements of MIL-C-5015. Additional capability to withstand elevated temperatures is provided in the material used for the sealing members.

The original tests conducted at the Amphenol's shop consisted of the following:

> Connectors installed in the flanges normally operate at ambient conditions of 105°F and 9.75 psia, and were tested for leak rate and tagged for integrity before shipment to the job. A test facility was set up by the manufacturer suitable for 50 psig, with provisions for thermocycling from 32° to 300°F. A thermocycle run of at least three cycles was made on one of each type flange. A time interval of 30 minutes was

> allowed between the thermocycles. The leak rate test after thermocycling was made at 50 psig and 300°F. Each completed flange had a leak rate of less than 1×10^{-6} cc/sec per assembled flange. All flanges were leak tested at 50 psig and 300°F. Helium gas was used in the test facility.

For the Amphenol triaxial cable penetrations a more detailed procedure for the thermocycle test was followed in shop test:

> The type sample consisted of a containment side flange disk with hermetic assemblies welded in place. A thermocouple was installed to monitor disk temperature. The disk was stabilized at 32°F and then placed in an oven heated previously to 280°F. On entrance of the disk, the oven temperature was reduced, straight line, to 150°F over a 60-minute period. The disk was removed and cooled to 100°F, while the oven was reheated to 280°F. The disk was then returned to the oven and the oven temperature reduced to 150°F as before. The highest metal temperature reached during this cycle was recorded and a 50-psig helium leak test was conducted at this metal temperature for all discs of this type.

Each Amphenol penetration assembly, without external cable mating connectors, was tested in the factory to demonstrate insulation resistance of at least 1000 megohms at 1000V dc. In addition, each penetration has passed an overpotential test. After initial installation, each penetration with external cables connected was tested at 1000V dc for 5 minutes.

Containment electrical penetrations now in use at the Surry Power Station were manufactured by either Amphenol Space and Missile Systems, Conax Corporation, or Westinghouse Electric Corporation. Nitrogen pressure is not required for penetration functional capability; however, each penetration is capable of being pressurized with nitrogen for leak detection purposes. RTV-8112, THIOKOL, or POLYSYLFONE are used to provide a tight seal around conductors.

Amphenol penetration electrical connectors were tested by D. G. O'Brien, Inc. in 1972. The purpose of this test was to demonstrate operability during simulated LOCA conditions. The connectors passed the test with no less than 34 megohms internal resistance while retaining complete electrical continuity. The test had no observable physical effect on the connector assembly or cable. No connectors are associated with the CONAX or Westinghouse type penetrations; however, both manufacturer's have provided data regarding the performance of the materials used in their penetrations. This information includes thermal performance, radiation resistance, and chemical resistance tests. All data indicate excellent performance characteristics for a LOCA environment.

All containment structure piping penetrations consist of a basic containment insert, plus additional items, as required for the individual services. Two basic types of penetrations are used for piping systems:

1. Unsleeved - These penetrations consist of piping installed through the containment wall without a sleeve around the outside of the piping. Unsleeved penetrations are used for cold piping systems (temperature of the fluid in the piping is less than 150°F) when only one pipe passes through the penetration.
2. Sleeved - These penetrations have a sleeve around the outside of the piping. Sleeved penetrations are used for all multiple piping systems passing through one penetration and for all thermally hot (over 150°F) piping systems, both single and multiple. Typical piping penetrations are shown in Figure 15.5-10.

The main steam and feedwater penetrations are provided with adequate space between the piping and the sleeve for the necessary pipe insulation, and for a pipe coil outside the insulation through which component cooling water is circulated. This cooling coil reduces the temperature of the sleeve and prevents any excessive heating of the concrete in contact with the sleeve. All welded seams subjected to containment pressure are leaktested by introducing air through each test boss. In addition, the sleeve end is drilled and tapped, as shown in the details, so that any leakage between pipe wall and sleeve end can be detected during periodic containment leakage testing.

The liquid and gas pipe penetration assemblies, in nearly all instances, consist of more than one pipe inside the penetration sleeve. The diameter of the sleeve depends on the number and size of the pipes installed in a given penetration. Each of these penetrations was tested with air using the same procedure as that used for the steam and feedwater penetrations.

A 20-inch o.d. fuel transfer tube penetration is provided for fuel transfer between the refueling canal in the containment structure and the spent-fuel pool in the fuel building. The penetration consists of a 20-inch stainless steel pipe installed inside a 26-inch pipe, as shown in detail in Figure 15.5-10, Sheet 1. The inner pipe acts as the transfer tube, and connects the containment refueling canal with the spent-fuel pool. The outer pipe is welded to the containment liner, and provision is made, by use of a special seal ring, for air leak testing of all welds essential to the integrity of the penetration. Bellows expansion joints are provided on the outer pipe to compensate for any differential movement between the two pipes.

The equipment hatch is a 14-ft. 6-in. single closure penetration. The equipment hatch cover is mounted inside the containment structure and is double gasketed with a leakage test tap between the o-rings. The equipment hatch cover is provided with a hoist with two point suspension and a sliding rail for storage. A positive locking device is furnished to prevent circular swing. The equipment hatch was designed, fabricated, and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class B. A removable concrete tornado missile shield protects the equipment hatch and acts as equivalent shielding.

The equipment hatch opening is analyzed in two basic steps, using the actual space curve shape. In the first step, it is assumed that the pattern of stress concentration at the junction of the cylinder and the ring beam is the same as if the cylinder were of infinite radius and the ring and the cylinder were in the same plane; that is, the cylinder wall is assumed to be a flat plate. The forces are imposed on this flat plate at such a distance from the opening that they are not influenced by the opening. These forces are membrane meridional (vertical) loads, circumferential (hoop) loads, and tangential shear loads. An outward longitudinal force is delivered to the inside face of the ring beam by a bearing plate which is welded to the liner plate. This force is caused by incident pressure acting on the hatch.

The first step of analysis is performed by using the Stone & Webster Shell I computer program, which is based on general first-order linear plate theory modified by Sanders. This program is developed from a numerical analysis of shells of revolution as published by Bernard Budiansky and Peter Radkowski in the AIAA Journal, Vol. 1, No. 8, August 1963.

In the second step, the ring is isolated and analyzed, taking into account its actual geometry (curved in two planes). The loads imposed on the ring beam have been obtained at the junction of the ring and normal shell in the first step, plus the incident pressure loads and temperature effects on the ring surface.

The second step of the analysis is performed by means of the Stone & Webster computer program, *Reinforcing Opening in a Cylindrical Structure.* This program analyses structures curved in space about two axes. The loads imposed on the ring are the membrane forces at the juncture of the thickened ring and the normal shell, as obtained from the first step, and a modified incident pressure acting on the ring beam and hatch cover.

The analysis of the isolated ring is based on the theory of curved beams, as demonstrated in Seely and Smith, *Advanced Mechanics of Materials.* Although the theory in this textbook is confined to the one-dimensional, curved beam, the assumptions set forth are extended to the two-dimensional case. These assumptions permit a simplified calculation of the stresses and deformations.

The ring, loaded in two planes, is statically indeterminate to the sixth degree. The analysis for the ring in space consists of cutting the ring, imposing six unknown loads at the cut section, and solving for the six unknown forces by equating differential deflections and rotations on either side of the cut to zero. The six unknown forces are: direct force, a torsional moment, a transverse shear, a radial shear, and bending moments about two axes. The curvature of the ring is considered in obtaining the bending moment strains and effects on total strain energy.

An emergency airlock is provided through the equipment hatch for emergency access to the containment. The airlock is flanged to the outside of the equipment hatch cover utilizing a double o-ring seal, and has an outside diameter of 6 ft 0 in., and a length of 12 ft 8.50 in. A 30-inch-diameter door is located at each end of the air lock. The air lock doors, which swing toward the center of the containment, are interlocked so that one door cannot be operated unless

the other is closed. These are mechanical interlocks, and provisions have been made for deliberate violation of the interlock by use of a special tool. This tool shall be kept under administrative control.

Each door is equipped with a valve for equalizing the pressure across the door. At no time can the equalizing valves on both doors be open simultaneously, and in no case can an equalizing valve be open on one door while the other is operating.

The operations required at each station for engagement or release of the interlocks, for operation of the equalizing valve, and for opening or closing the door, are accomplished by rotation of a single handwheel. Provisions have been made to allow operation of the outer door from inside the containment and the inner door from outside the containment, in addition to local operation.

Both doors are designed to withstand the containment test pressure of 52 psig. Each door is also designed to withstand 8.0 psia pressure in the containment structure with full atmospheric pressure outside. The interior door is provided with an additional securing device to facilitate testing the air lock to the maximum test pressure when the containment structure is at 8.0 psia.

All shafts penetrating the door or bulkhead have double packing. A blind flanged emergency air port is provided on the air lock outside containment. A light is provided inside the air lock and is powered near the air lock for communication.

A track is provided for emergency air lock removal via a cart. The track consists of two continuously supported rails that extend through the equipment hatch barrel onto the platform. Chicago Bridge and Iron Company designed and installed the track inside the equipment hatch barrel, and Stone & Webster designed the identical mating rails on the platform.

The tornado missile shield outside the containment equipment hatch has been modified to provide a labyrinth passage to the air lock. The missile shield slabs are fastened to the equipment hatch platform, which consequently has been modified. The equipment hatch platform has sufficient structural steel to withstand tornado wind loads on the attached missile shields.

The design, fabrication, and testing of the emergency air lock was performed by Chicago Bridge and Iron Company according to the ASME Code Section III, Subsection NE, 1971 Edition through the Winter 1972 Addenda. Welder procedures and performance qualifications were controlled under ASME Code Section IX.

The personnel hatch is a 7 ft. 0 in. i.d. double closure penetration as shown in Figure 15.5-11. Each closure head is hinged, double gasketed with a leakage test tap between the o-rings. Both doors are interlocked so that in the event one door is open, the other cannot be actuated. Both doors are furnished with a pressure equalizing connection. The equalizing valves are manually operated by persons entering or leaving the personnel hatch. The personnel hatch was designed, fabricated and stamped in accordance with the ASME Boiler and Pressure Vessel

Code, Section III, Class B. The personnel hatch is externally protected from tornado missiles by concrete shield walls and roof.

An 18-inch-diameter manway on the inner door of the personnel airlock is also provided for emergency egress from the containment. A positive locking device prevents inadvertent opening of the emergency manway. Manway position indication is provided in the control room. Alarm indication is also provided in the control room, and on the control panels on either side of the personnel airlock inner door, to indicate whenever the manway locking bar is not in the proper position to prevent inadvertent opening of the manway.

Material for the liner and penetrations is carbon steel plates conforming to ASTM A442, Grade 60, which has a specified minimum tensile strength of 60,000 psi, a minimum guaranteed yield strength of 32,000 psi, and a guaranteed minimum elongation of 25% in a standard 2-in. specimen. The liner has sufficient ductility to tolerate local deformation without rupture. This material has a nil ductility transition temperature of -20°F, which is 80°F below the normal minimum shutdown temperature given in Section 5.4.1.

Steel items, except backing plates and anchors, gas testing channels, equipment hatch bolts, and equipment hatch nuts are made to fine grain practice and normalized. In addition, steel items other than the above have passed NDT tests performed in accordance with the following:

1. Material 5/8 inch and thicker was tested by the Drop Weight Test method in accordance with ASTM E 208.
2. Material less than 5/8 inch thick was tested by the Drop Weight Tear Test method as developed by the U.S. Naval Research Laboratory (NRL Report 6300).
3. Material 5/8 inch and thicker has an NDT no higher than -20°F.

The liner plates were ordered to conform to standard mill practice with regard to thickness tolerances. Therefore, the 3/8-inch-thick cylindrical shell liner plate ranges in thickness from 0.365 inches to 0.406 inches. The 0.50-inch-thick hemispherical dome liner plate ranges in thickness from 0.490 inches to 0.535 inches, and the 0.25-inch-thick flat base liner plate ranges in thickness from 0.240 inches to 0.285 inches.

Physical and chemical properties of materials used in the construction of the containment liner, weldability tests, and liner thickness were checked by the Stone & Webster Field Quality Control Organization on a random sampling basis.

All welding procedures and tests required in Section IX of the ASME Boiler and Pressure Vessel Code for Welding Qualifications were adhered to in the selection of weld rod material, weld rod flux, heat treatment, and qualification of the welding procedures and the performance of welding machines and welding operators engaged in the construction of the containment liner. The welding qualification included 180-degree bend tests of weld material. These procedures ensure that the ductility of welded seams was comparable to the ductility of the containment liner plate material.

Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels was used as a guide in the selection of materials.

Erection of the steel liner followed completion of the concrete mat. The 3/8-inch-thick steel wall liner was erected to approximately Elevation +60 ft. The 0.25-inch-thick mat liner plate was installed on top of the concrete foundation mat during this period. On completion of the wall liner to Elevation +60 ft. and completion of the mat liner, all welds were checked for compliance with the approved weld inspection and gas test requirements. Work on the liner was then stopped until the containment interior concrete structure was completed, the polar crane was erected, and the concrete containment wall was completed to ground grade (Elevation 26 ft. 6 in.).

The 3/8-inch-thick steel wall liner was erected from Elevation +60 ft. to Elevation +92 ft. 6 in., and the containment liner completed with the construction of the 0.50-inch-thick steel dome liner. A-1 welds were inspected and gas-tested for compliance with the weld requirements.

The reinforced concrete wall, above ground grade, was completed, following as closely as practical the construction of the wall liner.

The reinforced concrete dome was constructed upon completion of the dome liner.

The steel wall liner was braced internally and locally with temporary bracing to prevent distortion during concrete placement. The exterior concrete forms were supported from the placed concrete and tied to form a tension ring.

Cantilevered steel strongbacks were used in the construction of the concrete dome to support the steel dome liner against deformation due to the weight of reinforcing steel formwork and wet concrete. Strongbacks were cantilevered from the completed concrete of the dome.

The containment liner is not a coded pressure vessel, so there was no section of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels directly applicable to its design and construction. However, to ensure that good engineering practices were followed, certain portions of Section III of the Code were reviewed for suggested guidance as to design and construction practices that should be incorporated in the liner specifications. Those sections reviewed for information were:

- N-511 Certification of Materials by Vessel Manufacturer
- N-512 Material Identification
- N-513 Examination During Fabrication
- N-514 Repair of Material by Welding
- N-515 Forming Shell Sections and Heads
- N-518 Attachments

- N-519 Cutting Plates and Other Products
- N-521 Welding Processes
- N-522 Welding Qualifications and Weld Records
- N-523 Precautions for Welding
- N-524 Assembly
- N-526 Finished Longitudinal and Circumferential Joints
- N-527 Miscellaneous Welding Requirements
- N-528 Repair of Weld Defects
- N-531 Preheating
- N-541 Modification of Section IX - Welding Procedure Qualification Requirements
- N-611 Inspection, General
- N-612 Qualification of Inspectors, Engineering Specialists, and Inspection Agencies
- N-613 Access for Inspector
- N-614 Inspection of Materials
- N-615 Marking on Plates and Other Material
- N-616 Final Inspection
- N-620 Inspection of Welding
- N-622 Check of Welder and Welding Operator Performance Qualifications
- N-623 Check of Nondestructive Examination Methods
- N-625 Ultrasonic Examination of Welded Joints
- N-626 Magnetic Particle Examination
- N-627 Liquid Penetrant Examination
- N-713 Pneumatic Test
- N-714 Pressure Test Gauges

The liner attachments are Nelson concrete anchors, welded on a triangular pattern to the wall and dome liner, and cast in the containment concrete as the concrete was poured against the liner. The attachment spacing was determined by the procedure (Reference 6) set forth for buckling of a cylindrical shell under combined axial and uniform lateral pressure where each attachment constitutes a buckling wave nodal point and was so spaced that the critical buckling stress will take place in plastic range of the liner material. The liner dome was treated in a similar manner. Maximum variation from the correct stud location, where relocation was necessary to avoid an obstruction, did not exceed 1.50 inches. The bottom mat liner was covered with 1.50- to 2-foot-thick reinforced concrete slab to protect it from both pressure and temperature loadings, so that it will remain virtually unstressed.

All penetrations are anchored into the concrete containment structure wall with a loading resistance level greater than the plastic strength of the penetration pipe. Openings in the liner plate are reinforced with reinforcing plate, and/or collar, sized to develop the full relief of the liner plate. The stress around each reinforced opening was analyzed in accordance with the appropriate procedure (Reference 7).

Departure from the original specified out-of-roundness tolerance of the reactor containment liners was necessary due to erection difficulties. Attempts were made to obtain the specified tolerance by means of an adjustable ring girder and supplementary anchorage to the cofferdam. As work progressed above the cofferdam level, it was found that it was impractical to obtain the specified liner tolerance.

A thorough review was made of the necessity for this close tolerance, and it was found unnecessarily restrictive.

The liner shell and dome are studded to the concrete and the plate is essentially plane within an equilateral triangle, 12 inches at the base and bounded by studs at the apexes of the triangle. The response of each individual triangular element to its own particular loading system establishes the adequacy of the structure as a whole. Therefore, actual roundness of the shell has no effect on liner performance.

The following revised out-of-roundness tolerances were adopted after a thorough review of the problem.

1. The out-of-roundness tolerance shall not exceed plus or minus 3 inches from the true radius.
2. The maximum plus or minus deviation from a true circular form shall not deviate more than 0.25 inches from a straight line in any 14-inch space in any plane in any location on the liner.

The revised out-of-roundness tolerances have no adverse effect on the buckling strength of the liner, and ensure that plate buckling between studs will not occur in the elastic range.

The adjustable ring girder was found to be of limited value during the erection of the liner, due to the many liner penetrations and the stiffness of the liner shell. Therefore, the ring girder

was used for rounding the shell only in areas where its application was found advantageous by the liner fabricator.

15.5.1.9 **Materials**

See Sections 15.5.2.2, 15.5.2.3, and 15.5.2.4 for descriptions of the construction materials used for the Reactor Pressure Vessel Head Replacement Project.

15.5.1.9.1 Concrete

The description of concrete materials is given in Section 15.3.1.

See Section 15.5.2.4 for the description of the concrete used for the Reactor Pressure Vessel Head Replacement Project.

15.5.1.9.2 Porous Concrete

Porous concrete is used under the base mat to provide drainage for the containment structure. The type of concrete is formed by the omission of the fine aggregate from a standard structural concrete mix. The mix was designed to have a 28-day compressive strength greater than 1000 psi.

Water porosity tests were performed earlier in an independent laboratory for porous concrete, using 6-inch by 12-inch cylinders prepared in the laboratory by compacting the material in three layers with standard tamping rods. A varying number of strokes, ranging from 10 to 40 for each layer, were used for different cylinders. After the concrete test cylinders had been properly cured, the amount of water that would flow through the 12-inch length of specimen during a three-minute period with a constant head of 4 inches of water above the top of each cylinder was determined. Results indicated water porosities of from 28 to 47 gpm/ft^2, depending upon the amount of compaction and resulting density of the cylinders.

The porosity determined by the laboratory tests indicated that the four-inch porous concrete layer under the base mat provides adequate drainage, since the leakage through the membrane waterproofing of the container would be minor. This layer serves as the collection means for the seepage removal system in the mat, described in Section 15.5.1.3.

15.5.1.9.3 Reinforcing Steel

Special large-size reinforcing bars, No. 14 and No. 18, used in the construction of the reactor containment structure, are steel of 50,000 psi minimum yield point, conforming to Grade 40 of the *Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement,* ASTM A615 as modified to meet the following chemical and physical requirements:

- Carbon 0.35% maximum
- Manganese 1.25% maximum

- Silicon 0.15 to 0.25%
- Phosphorus 0.05% maximum
- Sulphur 0.05% maximum
- Minimum yield strength 50,000 psi.
- Elongation 16% minimum in a 2-inch test sample
- Tensile strength 70,000 to 90,000 psi

For these special chemistry bars, all ingots were identified and all billets were stamped with identifying heat numbers. All bundles of bars were tagged with the heat number as they came off the rolling mill. A special stamp marking was rolled into all bars conforming to this special chemistry, to identify them as processing the chemical and mechanical qualities specified.

See Section 15.5.2.3 for the description of the reinforcing steel used for the Reactor Pressure Vessel Head Replacement Project.

The engineers' quality assurance inspectors witnessed, on a random basis, the pouring of the heats and the physical and chemical tests performed by the fabricator. Bars containing inclusions, or failing to conform to the required chemical and physical requirements, were rejected.

One 12-inch-long test sample was furnished to the engineers from a finished bar from each heat of the special chemistry rebars, to permit independent verification of physical and chemical analysis tests by the engineers.

Test specimens for the special chemistry rebars conformed to Section 10.1.1 of ASTM A615 and were Standard 0.505-inch-diameter specimens with 2-inch gauge length. Rate of loadings was such that the tension-tested sample was brought to the yield point in not less than 2 minutes.

For containment structure, reinforcing steel, consisting of No. 11 bars and smaller, is of 40,000 psi minimum yield point, conforming to Grade 40 of the *Standard Specification for Deformed Billet-Steel for Concrete Reinforcement,* ASTM A615.

The reinforcing steel for structures other than the containment structures is described in Section 15.4.3.

15.5.1.9.4 Cadweld Splices

Cadweld reinforcing steel splices, Type "T" full tension splices, as manufactured by Erico Products, Inc., Cleveland, Ohio, were used to splice 50,000 psi minimum yield point reinforcing bar sizes No. 14 and No. 18. These splices, including the sleeves, develop tensile strengths not less than 90% of the minimum ultimate strength of the reinforcing bar. The average value of two

or more successive splices develop at least the minimum ultimate strength of the rebar. Information for splices other than No. 14 and No. 18 reinforcing bars is given in Section 15.4.3.

See Section 15.5.2.3 for the description of Cadwelds, including operator qualification and tensile testing, used for the Reactor Pressure Vessel Head Replacement Project.

15.5.1.9.5 Waterproofing Membrane

The waterproofing membrane is a flexible polyvinyl chloride sheet having a minimum thickness of 40 mils. Associated adhesives and tapes consist of the membrane manufacturer's recommended material for the application conditions.

15.5.1.10 **Construction Procedures and Practices**

After performing the general excavation described in Section 15.4.4, two 149-ft. 5.25-in.-diameter cofferdams were constructed, one for each reactor. The cofferdams consist of interlocking steel sheet piles supported by a system of heavily reinforced concrete internal ring wales. The top of the sheet piles is at Elevation +10 ft. and tip grade is at Elevation -48 ft. The interior of the cofferdams was excavated to approximately Elevation -41 ft. Seepage drains were then driven through a 12-inch layer of crushed stone placed in the bottom of the excavation, as described in Section 15.5.1.3.

A 2-inch-thick concrete leveling slab was placed over the crushed stone and 40-mil-thick vinyl waterproof membrane placed over this concrete. A 4-inch layer of porous concrete was then placed over the membrane to protect the membrane and to serve as an internal drainage system, as described in Section 15.5.1.12.

Porous concrete was also placed around the sides of the cofferdam to fill the space between the cofferdam and the edge of the concrete mat, and to provide a form for the mat concrete. The waterproof membrane was extended vertically in this area, and protected by concrete block.

The reinforcing steel, steel bridging bars as described in Section 15.5.1.8, and other miscellaneous steel inserts required in the containment mat were placed, and the concrete poured. The mat was constructed in six sections.

The 3/8-inch-thick steel wall liner was then erected to Elevation +60 ft on the containment wall. The steel mat liner plates were installed on top of the concrete mat. All welds were checked for compliance with the approved weld inspection and gas test requirements. The containment interior concrete structure was then built on the mat liner. On completion of the interior concrete structure, the polar crane was erected.

The exterior containment concrete wall was constructed to approximately Elevation 24 ft. 6 in. during the construction of the interior concrete. On completion of the concrete substructure, a vinyl waterproof membrane was attached to the exterior concrete surface with adhesives. The membrane completely encloses the containment structure below grade.

The space between the cofferdam and the containment structure was then backfilled with crushed stone compacted in 6-inch layers. A 2-foot-thick layer of compacted impervious fill was placed at Elevation -4.0 ft. to seal the area and to minimize the amount of ground water seeping into the area.

The liner was then completed and finished with the construction of the 0.50-inch-thick steel dome, with all welds inspected and gas-tested. The steel dome liner was supported during erection with open web steel trusses.

See Section 15.5.2.2 for the description of the restoration of the steel liner during the Reactor Pressure Vessel Head Replacement Project.

The reinforced concrete wall above ground grade was completed, following as closely as practical the construction of the wall liner.

The completed steel wall liner was braced internally and locally with temporary bracing to prevent distortion during concrete placement. The exterior concrete forms were supported from the preceding concrete.

Cantilevered steel strongbacks were used in the construction of the concrete dome to support the steel dome liner, reinforcing steel, formwork, and wet concrete against deformation. Strongbacks were cantilevered from the completed concrete of the wall or the dome.

Careful inspection of the dome was maintained during concrete placing and until the concrete had definitely taken initial set. Concrete buckets used during the first two lifts of the dome were limited to 2 yd^3 in size. Bucket sizes were increased after the second lift had set, when placing results of these lifts were satisfactory and warranted such a move.

Concrete in the wall and dome of the containment structure was poured in uniform 6-foot lifts around the entire circumference. Each lift was constructed in approximately 18-inch layers.

See Section 15.5.2.4 for the description of the concrete used for the Reactor Pressure Vessel Head Replacement Project.

Concrete forms were used on the exterior of the concrete dome to a line 50 degrees above the horizontal. The permanent steel liner served as the inner form for pouring concrete. For the area where exterior forms were used, the concrete points were in horizontal planes. Above the 50 degree line, the remainder of the dome concrete was poured as one lift.

Particular care was taken to check the special markings of the No. 14 and No. 18, 50,000-psi minimum-yield rebars for the containment structure.

See Section 15.5.2.3 for the description of welded splices and Cadwelds, including operator qualification and tensile testing, used for the Reactor Pressure Vessel Head Replacement Project.

Welded splices conform to *Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections, in Reinforced Concrete Construction,* AWS D12.1. Bars spliced by metallic arc welding develop not less than 90% of the minimum ultimate strength of the reinforcing bar, and the average of two or more successive splices develop at least the minimum ultimate strength of the bar.

Structural ductility was maintained by staggering critical splices where possible. Full scale pressure tests conducted in May 1967, on a recently completed concrete containment structure (Reference 8) in which similar Cadweld splices and welded splices were used, showed no stress concentrations or lack of structural ductility. Locations of splice groups were not discernible from inspection of the test crack patterns.

All Cadweld Process Type "T" joints were visually inspected. The visual inspection included inspection of the ends of the bars for dryness and cleanliness prior to fitting the sleeve over the ends. It also included inspection of the completed splice for properly filled joints to ensure that filler metal was visible at both ends of the sleeve and at the top hole in the center of the sleeve. Randomly selected splices were removed from the structure and strength-tested for compliance with the specification. Joints that did not meet all these inspection criteria were replaced.

Randomly selected Cadweld Type "T" splices were removed from the containment structure and tensile-tested for compliance with the specifications, in accordance with the following schedule:

1. One out of first 10 splices.
2. Three of next 100 splices.
3. One out of each subsequent unit of 100 splices.

Welding inspection of reinforcing bars was by quality control inspectors. Radiographic inspection, dye-penetrant inspection, magnetic-particle inspection, or other nondestructive inspection methods for welded joints was performed on a random basis under the direction of the Senior Quality Control Engineer.

All welds were visually inspected. Any cracks, porosity, or other defects were removed by chipping or grinding until sound metal was reached, and then repaired by welding. Peening was not permitted.

Completed welded splices were selected on a random basis and removed from the structure with suitable lengths of adjacent bars. These removed splices were tensile-tested for compliance with the specifications in accordance with the same schedule followed for the Cadweld Type "T" splices.

Tack welding of special chemistry rebar was not permitted.

15.5.1.11 **Missiles and Piping Rupture**

15.5.1.11.1 Interior Missiles

> Missiles could be either concrete or steel. Because of lower density and lower strength, a concrete missile would have to be an order of magnitude heavier than a steel missile of comparable diameter and velocity for it to cause the same damage on impact with a steel shell. Also, in the context of the design-basis accident, there are more potential steel missiles and these have been studied in detail.

The most penetrating steel missile for a given mass and velocity would be rod-shaped, impacting end-on; therefore, rods of various diameters and weights have been investigated.

Missile velocities of 100 fps might be generated by rupture of a reactor coolant loop[1], and this value has been used with penetration equations developed by D. A. Davenport (Reference 9) to estimate their penetrating capability.

Table 15.5-3 summarizes the results of the analysis. Inspection of these results indicates that, except for the containment liner, at 100 fps, the required weight and dimensions for penetration of the metal thicknesses of interest are not credible for missile sizes that can be postulated within the reactor containment. The metal thicknesses shown in the table bracket the thicknesses of interest for the containment liner and piping systems. The analysis for the containment liner does not consider the added resistance to penetration afforded by the interaction between the concrete containment structure and the containment liner. This added resistance will not permit penetration by missiles of credible weight and size. Major components, such as the steam generator, have greater shell thicknesses than the values in the table, and therefore will not be penetrated by the postulated missiles.

All potential missiles were evaluated, and those that constitute a hazard to either the liner or adjacent equipment, by virtue of their velocity and/or size, are restrained by local barriers or other mechanical means.

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1. As discussed in Section 15.6.2, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop piping. However, pipe ruptures of other high-energy lines are still postulated.

15.5.1.11.2 Exterior Missiles

15.5.1.11.3 Pipe Rupture Incident

The containment internal structure is designed to accommodate the loading due to rupture of the reactor coolant and connecting piping[1], or main steam or feedwater piping. Incident rupture was considered in only one line at a time. The support system was designed to preclude damage to or rupture of any of the lines as a result of the incident. The snubber and key systems are designed to transmit rupture thrusts from a steam generator into the containment internal structures. In determining the steam generator support reactions, the system was reduced to a dynamic model consisting of a suitable number of masses and resistance elements under impulse loading. The structural support system resilience and mass was included in the model. The dynamic problem was solved by numerical methods, using a thrust-time history as loading. Resistance, dynamic amplification of the thrust, and rebound forces were calculated versus time. Design of the support system was based upon stress levels defined in Section 15.5.1.8. The reactor vessel and support system were similarly treated.

The steam lines are strapped to the crane wall at intervals selected to prevent a whipping pipe from contacting the liner. The straps are designed so that no interference with the normal thermal expansion modes of the steam lines results.

1. As discussed in Section 15.6.2, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop piping. However, pipe ruptures of other high-energy lines are still postulated.

15.5.1.12 **Ground Water Protection and Corrosion**

The ground-water level external to the membrane protection of the exterior surfaces of the containment structure will be kept below the top surface of the foundation mat by pumps, as described in Section 15.5.1.3.

If water penetrates or otherwise circumvents the membrane, it drains to a layer of porous concrete directly below the mat and above the membrane. This 4-inch-thick layer of porous concrete serves as a horizontal drain under the entire structure. The porous layer is vented by two 4-inch-diameter pipes that extend from the underside of the mat into a subsurface cubicle adjacent to the outside of the containment structure. This cubicle is inside the waterproof membrane. Access is provided by a concrete shaft from ground level. The 4-inch-diameter vent pipes are installed to discharge water to the floor of the cubicle at a level three feet below the mat liner; thus, flooding of the cubicle would have to occur before any hydrostatic head would be applied to the steel liner. A water level alarm is installed in the cubicle, and pumps are used as necessary to remove the water. Vertical drainage to the base of the mat is aided by three vertical inspection shafts, and various tunnels and cubicles located adjacent to the exposed exterior face of the concrete containment wall, in which the concrete is exposed.

Cathodic protection is not provided, since adequate corrosion protection of the embedded reinforcing is otherwise provided. Research by the National Bureau of Standards and other references indicates that cathodic currents damage the bond between the reinforcing steel and concrete. This bond softening is due to the gradual concentration of sodium and potassium ions. In time, the alkali concentration becomes strong enough to attack the steel.

The surface of the steel liner in contact with concrete is not subject to corrosion because of the alkaline nature of the concrete. The interior exposed surface of the liner is protected by one coat of inorganic zinc silicate primer with one top coat of epoxy enamel. These materials were used during construction. The repair coatings currently used are selected in accordance with administrative controls. No other protective coatings or insulation are considered necessary.

15.5.1.13 **Testing and Inservice Surveillance**

15.5.1.13.1 General

The completed containment structure was tested for structural integrity by subjecting the structure to an air pressure test equal to 115% of the design pressure. The structure was first carefully surveyed, measured, and inspected for cracks prior to the test, and all measurements recorded. All measurements were related to an independent datum. The pressure was then raised in 10-psi increments to the 115% test pressure (52 psig) and held at that pressure for one hour. Pressure was then reduced to complete the containment liner leak rate test described in Section 5.3.

During the 48-hour period, visual examination was made of the containment exterior surface for cracks and crack patterns as well as distortion.

Visual and instrumented observations at each pressure increment were made of the containment response during the test. Crack patterns were observed and their development noted.

Temperature, barometric pressure, and weather conditions were recorded hourly during the test period.

A further detailed dimensional survey was made of the structure on completion of the tests to record recovery of the structure.

15.5.1.13.2 Test Instrumentation

Instrumentation was designed to provide control and information on containment response during the air pressure test. Measurements were made of the radial deflection of the containment wall at selected locations from the top of the mat to the spring line of the dome. Vertical deflections were measured at the top of the mat and at the top of the dome. Additional measurements were made around the equipment access hatch and in other areas where stresses were critical.

Strain gauges were attached to the steel liner to record strains at the junction with the mat liner, at mid-height, and at the spring line of the dome. Additional strain gauges were attached to the liner around the equipment access and personnel hatches.

Exterior visual observations, above grade, were obtained using engineer’s scales attached to the structure and read by transits placed nearby. Transit measurements were calibrated with independent datum points. Readings obtained by this method were considered accurate to within 0.10 inch.

Exterior deformations below grade were measured by linear variable differential transducers (LVDTs) mounted in the two pits provided for this purpose. Linear variable differential transducers recorded displacements in mils, which is an accuracy in excess of that required. Linear variable differential transducers were also used to measure displacements of the concrete rings surrounding the equipment access and personnel access hatches.

Electrical strain gauge rosettes and conventional strain gauges, reading in microinches per inch, were used to monitor strains in the liner. Since major inaccuracies with this type of gauge have resulted from inadequate installation techniques, particular attention was given to the technique used.

Redundancy of instrumentation was obtained by multiplicity of points at which measurements were made, so that loss or damage to any one station was not critical.

The range of strains and deformations expected at the 45-psig test pressure was as follows:

1. Maximum vertical elongation of the structure, not more than 1.5 inches.
2. Increase in containment diameter, not more than 1.4 inches.

3. Maximum width of new cracks or increase in existing cracks, not more than 0.03 inch per crack.
4. After containment pressure was reduced to atmospheric, the residual width of new cracks or the increased width of existing cracks, not more than 0.01 inch.
5. There was no visual distortion of the liner plate.

The containment structure remained in the elastic range during the pressure test, and there was permanent distortion in the liner or in the concrete once the pressure was reduced to atmospheric or below. However, it was fully expected that there would be small residual cracks in the concrete as a result of shrinkage in the concrete.

Under the test program outlined herein, all instruments and measuring devices were installed just prior to the test, and normal care and protection was adequate. Items damaged for any reason were readily replaced at the initiation of the test.

15.5.1.13.3 Comparison of Test Results

The selection of a test pressure, which was 115% of design pressure, was based primarily on the fact that a similar reinforced concrete containment structure for Connecticut Yankee Atomic Power Plant has been tested and accepted at 115% of its design pressure; thus, a comparative case history of structural response has been created that permits valid comparison of similar design.

The selection of 115% test pressure also conforms to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection B, Requirements for Class B Vessels, paragraph N-1312(d). This relates to metal vessels that perform the same function as a reinforced concrete containment structure.

A comparison of stresses under 115% test pressure with those in the structure under incident conditions is given in Table 15.5-4. As a sensitivity analysis, the stresses associated with 125% pressure are also included. Incident stresses shown result from incident pressure, dead load, loads due to temperature effects on the steel liner, and temperature gradients through the concrete. Stresses resulting from earthquake combined with incident loads are shown separately.

Scaled load plots comparing moments, shears, tension, and deflections resulting from the structural proof test pressure with moments, shears, tension, and deflections due to the unfactored design incident conditions are shown in Figure 15.5-13. A comparison of the test load with the hypothetical incident load conditions should include a review of the load plots in Figure 15.5-2 Sheet 2. This shows the increase in moments, radial shear, hoop tension, vertical tension, and radial deflection (deformation) imposed on the structure by the incident and test load. It can be seen that the test load conditions exceed incident conditions in all cases, except that of radial deflection.

The distribution of stress varies between the structural elements under apparently similar load conditions because of the contradictory action of the containment steel liner. Under test

conditions, the steel liner was in a state of biaxial tension and gave considerable assistance to the reinforcing steel, particularly to the longitudinal reinforcing. Under incident conditions, the steel liner is subjected to a point where it is restrained in compression by the reinforcing steel. This effect is greater in the dome than in the cylindrical wall, due to the increased thickness of the dome liner and the shape factor of the dome.

This also provides an additional factor of safety against ultimate failure of the structure. In the event of excessive yield in the reinforcing steel, the liner will act as a tensile membrane that would assist the reinforcing steel. This assistance would be significant, since the liner will bring a considerable reserve of energy to bear, for which design credit has not otherwise been claimed.

The test pressure of 52 psig, based on 115% design pressure, created stresses equal to or greater than the incident stresses in the following critical areas:

1. Foundation mat, where test stresses are 30 to 40% above incident conditions.
2. Large access openings, such as equipment and personnel hatches, where test stresses are comparable with incident stresses.

It is recognized that the average stress levels attained under the test conditions in the principal longitudinal and circumferential steel are below these stresses resulting from incident conditions. This is considered acceptable when the test is associated with dimensional strain measurements, when such a test provides confirmation of structural continuity and structural ductility with the concrete cracked, and when the steel is shown to carry the load in tension according to design assumptions.

An analysis of the crack pattern of the concrete under test conditions confirms stress distribution in the structure, and also reveals areas of stress concentrations. In fact, a pattern of severe local cracking would indicate structural weakness more effectively than considerations of average stress levels.

Measured response of the structure, as indicated by increase in height, diameter, and degree of recovery, together with measurements of local deformations, is extremely important in predicting structural response to incident conditions. The structural response to the test pressure is of sufficient magnitude to allow simple direct measurements of deformations without the need for high-precision methods of measurement.

In summary, it is not possible to exactly duplicate incident stress conditions with a pressure test. An increase in the test pressure above 115% would only preserve and amplify the present stress anomalies, without furnishing more meaningful data. In addition, such a test would endanger or damage the container by seriously overstressing critical areas, or it would require a container design modification directed specifically to withstand the higher pressure test without proportionate improvement in withstanding the incident condition. Modification of the containment design to obtain closer test verification of structural integrity under the test pressure would require specific redesigning for test conditions of the critical areas in the foundation mat

and at the large openings. Such redesigning would not improve the capability of the containment structure to meet the incident load conditions. A design meeting both incident and test conditions is not considered practical in this type of containment design.

The 115% pressure test provided a valid test of all criteria areas with stresses equal to or greater than incident conditions; in less critical areas, the pressure test provided sufficient information to permit a reliable evaluation of the complete structural response under incident conditions.

The average anticipated crack width at the 45-psig test pressure was 0.015 inch.

A rectangular crack pattern was anticipated, with vertical cracks spaced 12 to 15 inches on centers, and horizontal cracks spaced approximately 2 feet on centers. Horizontal crack spacing was primarily controlled by the horizontal construction joints.

The average crack width was related to the anticipated increase in containment diameter, the anticipated vertical elongation of the structure, and the crack spacing. It was assumed that the total containment extension was equal to the sum of the number of cracks multiplied by the average crack width in each direction.

Maximum summer temperature and minimum winter temperature difference is approximately 95°F. Annual average temperature variation is 40°F at the station site.

During unit operation, the annual maximum thermal cycling temperature variation is approximately 45°F.

Ambient temperature variations of this magnitude, +20°F, or even the extreme +45°F, will not reopen by any significant amount the crack pattern created in the structure by the test pressure of 45 psig.

The width of thermal cycling cracks was significantly less than the 0.010 allowed for exterior members by ACI 318.

The stresses given in Table 15.5-4 are the results obtained from computer programs referred to in the following sections:

- Section 15.5.1.5 —*Numerical Analysis of Unsymmetrical Bending of Shells of Revolution*
- Section 15.5.1.4 —*Container Vessel Seismic Analysis*
- Section 15.5.1.5 —*Flat Circular Mat Foundations for Nuclear Secondary Containment Structures*
- Section 15.5.1.7 —*Nuclear Containment Structure Access Opening* - Stone & Webster Computer Program

At large openings, the stresses due to thermal load were obtained by converting the thermal effect to a pressure equivalent, as described in Section 15.5.1.5.

Since all of the shears in the wall and dome were taken by the reinforcing, the effects of shrinkage and creep are not included.

15.5.1.13.4 Inservice Surveillance Tests

Periodic structural testing of the containment structure is not planned, since it would provide no more information on the containment structure capability than that obtained from the initial test. In fact, periodic testing would cumulatively damage the concrete in the structure to the point where the test itself would be the major cause of structural deterioration.

The inservice stress and environmental conditions are not of a nature or magnitude such that any significant deterioration of the reinforcing steel or concrete could reasonably be expected, and periodic testing for structural purposes could be duplicated if at any time further tests were required. The minimum test level required to verify continued structural integrity would be no less than the 115%, or 52-psig initial test pressure.

Periodic inspection of the steel liner is accomplished by a type A leak rate test in accordance with 10 CFR 50 Appendix J. All welded joints and all penetrations of the liner are designed for periodic halogen gas testing.

In summary, no basis exists for attempting to develop structural performance information from leak rate tests conducted at moderate pressures.

15.5.2 Reactor Pressure Vessel Head Replacement Project (Applicable to Unit 1 and Unit 2)

The Reactor Pressure Vessel (RPV) Head Replacement Project created and restored a construction opening in the reactor containment structure in accordance with administrative procedures and the design control program. The opening was used to facilitate the movement of original and replacement RPV heads in and out of the reactor containment structure. The opening was restored to meet the original design bases of the containment structure.

15.5.2.1 **Codes and Specifications**

ACI 318-63 is the design code for the restored containment structure. The restored structure meets all applicable design loads and load combinations required by ACI 318-63.

Concrete placement, curing, and repair were in accordance with ACI 301-99 with the incorporation of Hot Weather Concreting per ACI 305R-99, as appropriate, or with the incorporation of Cold Weather Concreting per ACI 306.1/ACI 306R, as appropriate. The use of ACI 301-99 is in accordance with Section 2.2 of ANSI N45.2.5-74.

Concrete mix proportioning was per ACI 211.1-91 (reapproved 1997) in accordance with Table A of ANSI N45.2.5-74.

Bechtel specifications (References 11-18) address:

- reinforcing steel procurement, testing, and placement
- Cadweld reinforcing steel splices procurement, testing, and installation
- concrete mix design, testing and placement
- structural steel and materials procurement

15.5.2.2 **Liner Restoration**

The cut section of the containment liner plate was rewelded to the liner plate with a full penetration weld. The weld was tested to ensure no leakage. In addition, the full penetration weld was covered by a seal welded leak chase channel to facilitate testing.

Replacement material was purchased for the liner plate, Nelson studs, and leak chase channels. The Nelson studs, and leak chase channels were used for the reinstallation of the plate and the leak chase channel system. Reference 18 requires the liner plate material to be ASTM A-516-Grade 60 (or better), fine-grained and normalized.

15.5.2.3 **Reinforcing Steel Restoration**

The reinforcing steel bars cut during the creation of the opening were re-installed using Cadweld splices or welding, as required, in accordance with References 14, 15, and 19. Reinforcing steel bars that were damaged during the creation of the opening were repaired in accordance with References 13 and 19 or were replaced with reinforcing steel procured in accordance with Reference 12. New N18 reinforcing steel used for containment wall restoration conforms to either ASTM A615 Grade 60 and/or ASTM A706 Grade 60, and meets or exceeds the additional elongation and chemical composition requirements described in Section 15.5.1.9.3 for the containment structure existing reinforcing steel.

In lieu of the Cadweld testing protocol, described in Section 15.5.1.10, which was used during original construction, Cadweld testing was performed in accordance with Dominion's Operational Quality Assurance Program Topical Report which includes:

- In-process testing of Cadweld splices in accordance with Subsubparagraph CC-4333.5.2 of ASME B&PVC Section III Division 2 (1995 Edition, 1996 Addenda).
- Cadweld Splice System Qualification in accordance with Subparagraph CC-4333.2 of ASME B&PVC Section III Division 2 (1995 Edition, 1996 Addenda).
- Cadweld Operator Qualification in accordance with Subparagraph CC-4333.4 of ASME B&PVC Section III Division 2 (1995 Edition, 1996 Addenda).

- Cadweld Testing Frequency in accordance with Subsubparagraph CC-4333.5.3 of ASME B&PVC Section III Division 2 (1995 Edition, 1996 Addenda).

To minimize the size of the construction opening, the Cadweld splice locations were not staggered as described in Section 15.5.1.10. Section 805 of ACI 318-63 does not require staggered Cadweld splices if the splice can develop in tension at least 125 percent of the specified yield strength of the reinforcing steel bar. The minimum acceptance criteria for the Cadweld splice testing in Reference 15 is that the minimum tensile strength of each sample tested shall be equal to or exceed 125 percent of the yield strength of the reinforcing steel bar. Also, the splicing scheme for the RPVH Replacement Project construction opening is similar to that used during the closure of the original construction opening.

15.5.2.4 **Concrete Restoration**

As discussed in Dominion's Operational Quality Assurance Program Topical Report commits to ANSI N45.2.5-74 (with clarifications) for satisfying the quality assurance requirements for installation, inspection, and testing of structural concrete during the operational phase of Surry Power Station. Section 2.2 of ANSI N45.2.5-74 requires that the installation, inspection, and test activities be performed in accordance with the latest codes. Tables A and B of ANSI N45.2.5-74 provide the requirements for the qualification and in-process testing of the concrete ingredients and concrete mix.

The concrete was replaced and the restored structure tested in accordance with ASME B&PVC Section XI, Articles IWL 4000 and IWL 5000, respectively. In accordance with the guidance of Table A of ANSI N45.2.5-74 concrete mix design is based on ACI 211.1-91 (reapproved 1997). The activities associated with placement of concrete were performed in accordance with References 11 and 17, which meet the requirements of ACI 301 and ANSI N45.2.5-74. In-process sampling, testing, and acceptance requirements for all repair material were in accordance with Table B of ANSI N45.2.5-74. Reference 11 provides the testing frequencies, sampling and testing standards, and acceptance criteria for concrete ingredients and concrete mix. The concrete had a minimum 5-day strength of 3000 psi.

The water used for the concrete mix was evaluated in accordance with the requirements of AASHTO T-26, as specified in Table A of ANSI N45.2.5-74. The water testing and acceptance criteria included in Reference 11 required that the water used during the restoration was free of harmful levels of contaminants.

The cement used in the new concrete was Type II Low Alkali (as defined in ASTM C 150).

For RPV Head Replacement Project, the restoration of the containment wall used size 57 (25 mm to 4.75 mm) coarse aggregate due to the limited size of the opening and the use of pour ports/bird mouths for concrete placement. Both fine and coarse aggregates were tested in accordance with the requirements of ANSI N45.2.5-74 to ensure acceptable physical

characteristics and that they were free of harmful levels of alkali reactivity and deleterious substances (acceptance criteria are defined in ASTM C 33).

Admixtures used to modify the concrete mix properties met the requirements of ASTM standards and were used in accordance with the manufacturer's written procedures and applicable ACI standards (primarily ACI 211.1-91 (reapproved 1997) for mixing and ACI 301-99 for placement). Reference 16 prohibited the use of admixtures with chlorides. Uniformity of admixture lots was verified with Infrared Spectrophotometry in accordance with Table B of ANSI N45.2.5-74.

In its ready mix state, the new concrete had an air content of 4.5% (± 1.5%) at the point of placement. This is consistent with Table 6.3.3 of ACI 211.1-91 (reapproved 1997) for the maximum aggregate size being used in the concrete mix (1" for Size No. 57 coarse aggregate) and air-entrained concrete.

A water-reducing admixture was utilized in the concrete resulting in a maximum slump of 8 1/2 inches at point of placement based on the footnote to Table 6.3.1 of ACI 211.1-91 (reapproved 1997), which approves higher concrete slump (than the recommended 1 inch to 4 inch slump) when chemical admixtures are used provided that there are no signs of segregation or excessive bleeding.

15.5.2.5 **Post Modification Testing**

The nondestructive examination of the containment liner was in accordance with Safety Guide 19, Nondestructive Examination of Primary Containment Liners with the following changes: after vacuum box testing of the liner seam weld and installation of the channel, the channel to liner weld was tested by a static pressure test (decay test) with an acceptance criteria of zero leakage. Soap bubble testing was used to identify leakage. Leaking areas of the joint were repaired and retested. In addition, following the containment building pressure test, the channel was re-pressurized and an "as-found" LLRT, meeting ANS 56.8-1994 requirements, was performed.

Prior to placing the containment structure in-service, a containment pressure test that bounds the calculated peak containment internal pressure was performed in accordance with IWL Article 5000 of the ASME B&PVC Section XI. The surface of the replacement concrete at the temporary construction opening was examined in accordance with IWL-5250 prior to pressurization, at test pressure, and following completion of pressurization.

15.5 REFERENCES

1. G. N. Bycroft, *Forced Vibrations of a Rigid Circular Plate on a Semi-Infinite Elastic Space and on an Elastic Stratum, Philosophical Transactions*, Royal Society, London, Series A, Vol. 248, pp. 327-368.
2. Karl Terzaghi, *Evaluation of Coefficients of Subgrade Reaction, Geotechnique,* Vol. 5, pp. 297-326, 1955.
3. B. O. Hardin and W. L. Black, *Vibration Modulus of Normally Consolidated Clay, Symposium on Wave Propagation and Dynamic Properties of Soils*, University of New Mexico, 1967.
4. Stone & Webster Engineering Corporation, *Nuclear Containment Structure Access Opening*.
5. B. Budiansky and P. Radkowski, *Numerical Analysis of Unsymmetrical Bending of Shells of Revolution, AIAA Journal*, August 1963.
6. S. Gere and S. Timoshenko, *Theory of Elastic Stability*, second edition.
7. J. N. Goodier and S. Timoshenko, *Theory of Elasticity*, second edition.
8. Stone & Webster Engineering Corporation, *Report on Pressure Testing of Reactor Containment for Connecticut Yankee Atomic Power Plant, Connecticut Yankee Atomic Power Company, Haddam, Connecticut*, 1967.
9. D. A. Davenport, *Penetration of Reactor Containment Shells, Nuclear Safety*, Vol. 2, No. 2, December 1960.
10. Letter dated March 30, 1999, Serial No. 99-134, From Virginia Power to the NRC, *Supplemental Response to Generic Letter 96-06.*
11. Bechtel Specification 24841-120-C-101, Revision 6, *Technical Specification for Material Testing Services*, April 25, 2003. (Unit 1)

    Bechtel Specification 24841-120-C-101, Revision 7, *Technical Specification for Material Testing Services,* December 3, 2003. (Unit 2)
12. Bechtel Specification 24841-120-C-303, Revision 2, *Technical Specification for Purchase of Reinforcing Steel*, February 28, 2003.
13. Bechtel Specification 24841-120-C-304, Revision 2, *Technical Specification for Installation of Reinforcing Steel (Rebars)*, February 28, 2003.
14. Bechtel Specification 24841-120-C-309, Revision 1, *Technical Specification for Purchase of Cadweld Rebar Splices*, February 28, 2003.
15. Bechtel Specification 24841-120-C-310, Revision 2, *Technical Specification for Installation of Cadweld Rebar Splices*, February 28, 2003.

16. Bechtel Specification 24841-120-C-321, Revision 9, *Technical Specification for Purchase of Ready Mix Concrete Qualified as Safety-Related*, May 20, 2003.
17. Bechtel Specification 24841-120-C-322, Revision 3, *Technical Specification for Placement of Ready Mix Concrete Qualified as Safety-Related*, February 28, 2003.
18. Bechtel Specification 24841-120-C-502, Revision 3, *Technical Specification for Purchase of Non-Safety Related and Safety Related Structural Steel and Materials*, February 28, 2003.
19. *Special Processes Manual for Surry Nuclear Power Station RPV Head Replacement Project,* Revision 3, Bechtel Job 24841, May 12, 2003. (Unit 1)
20. *Special Processes Manual for Surry Nuclear Power Station RPV Head Replacement Project*, Revision 4, Bechtel Job 24841, September 25, 2003. (Unit 2)

15.5 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11448-FM-1A	Machine Location: Reactor Containment, Elevation 47'- 4"
2.	11448-FM-1B	Machine Location: Reactor Containment, Elevation 18'- 4"
3.	11448-FM-1C	Machine Location: Reactor Containment, Elevation 3'- 6"
4.	11448-FM-1D	Machine Location: Reactor Containment, Elevation 27'- 7"
5.	11448-FM-1E	Machine Location: Reactor Containment; Sections “A-A”, “E-E”, & “Z-Z”
6.	11448-FM-1F	Machine Location: Reactor Containment; Sections “B-B”, “X-X”, & “Y-Y”
7.	11448-FM-5A	Arrangement: Auxiliary Building

Table 15.5-1
CONTAINMENT STRUCTURAL LOADING CRITERIA

Case	Loading Combination	Required Load Capacity of Structure
1	Operating plus DBA	= (1.0 ± 0.05)D + 1.5P + 1.0 (T + TL)
2	Operating plus DBA plus operating-basis earthquake	= (1.0 ± 0.05)D + 1.0P + 1.0 (T + TL) + 1.5E
3	Operating plus DBA plus design-basis earthquake	= (1.0 ± 0.05)D + (1.25P) + (T' + TL') + 1.0HE
4	Operating plus 1.25 DBA and 1.25 operating basis earthquake	= (1.0 ± 0.05)D + (1.25P) + (T' + TL') + 1.25E
5	Operating plus tornado loading	= (1.0 ± 0.05)D + 1.0T' + 1.0C

Legend

DBA - Design-basis accident.

C - Load due to negative pressure and horizontal wind velocity resulting from tornado and missiles. For description of tornado, refer to Section 15.2.3.

D - Dead load of structure and contents including effect of earth and hydrostatic pressures, buoyancy, ice and snow loads. To provide for variations in the assumed dead load, the coefficient for the dead load components is adjusted by ±5% as indicated in the above formulas to provide the maximum stress levels.

P - Pressure load from DBA. Pressure for containment design is 45 psig.

T - Load due to maximum temperature gradient through the concrete shell and mat based on temperature associated with 1.5 DBA pressure.

T' - Load due to maximum temperature gradient through the concrete shell and mat based on normal operating temperature.

TL - Load exerted by the exposed liner based upon temperature associated with 1.5 times DBA pressure.

TL'- Load exerted by the exposed liner based upon temperatures associated with 1.25 times DBA pressure.

T - Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associates with 1.0 times DBA pressure.

TL - Load exerted by the exposed liner based upon temperature associated with 1.0 times DBA pressure.

E - Operating-basis earthquake loading. Based on a ground acceleration of 0.07g horizontally at zero period and a damping factor of 5%. For description of the operating-basis earthquake, refer to Section 2.5.

HE - Design-basis earthquake loading. Based on a ground acceleration of 0.15g horizontally at zero period and a damping factor of 10%. For description of the design-basis earthquake, refer to Section 2.5.

Note: Normal wind loadings replace earthquake loads where they exceed earthquake loadings. Normal wind or tornado loads are not considered coincident with earthquake loads.

Table 15.5-2
CAPACITY REDUCTION FACTOR FOR CONCRETE

Member	Reduction Factor
Tension and flexure	0.90
Diagonal tension, bond and anchorage	0.85

Table 15.5-3
MISSILE DIMENSIONS AND WEIGHTS REQUIRED TO PENETRATE PLATE OF VARYING THICKNESSES

Material	Missile Diameter, in.				
	1	2	3	4	5

Table 15.5-4
COMPARISON OF STRESSES UNDER TEST PRESSURE WITH STRESSES UNDER INCIDENT CONDITIONS AND EARTHQUAKE PLUS INCIDENT CONDITIONS

	Incident Stress, psi	Earthquake Plus Incident Stress, psi	115% Test Stress psi (52 psig)	125% Test Stress psi (57 psig)
Top bars at cylinder wall	25,700	26,500	33,500	38,700
Bottom bars near mat center	26,300	30,500	29,000	38,700
Cylinder Wall (approximately mid-height)				
Circumferential reinforcing				
Inner	22,300	22,300	21,000	22,800
Outer	28,600	28,600	20,000	21,700
Longitudinal reinforcing				
Inner	19,800	21,600	9200	10,000
At base of wall	12,900	19,300	16,400	19,300
Outer	27,900	29,650	9200	10,000
Diagonal reinforcing	28,300	33,600	15,500	16,900
Diagonal (radial) shear reinforcing at base of wall	15,700	16,500	20,200	23,200
Dome				
Radial reinforcing				
Inner	28,900	28,900	14,200	15,500
Outer	34,500	34,500	13,800	15,000
Circumferential reinforcing				
Inner	28,900	28,900	14,200	15,500
Outer	34,500	34,500	13,800	15,000
Large Openings				
Equipment access hatch	32,000	33,500	31,300	33,700
Personnel access hatch	30,200	31,700	30,700	34,300

Figure 15.5-1
REACTOR CONTAINMENT WATERPROOFING

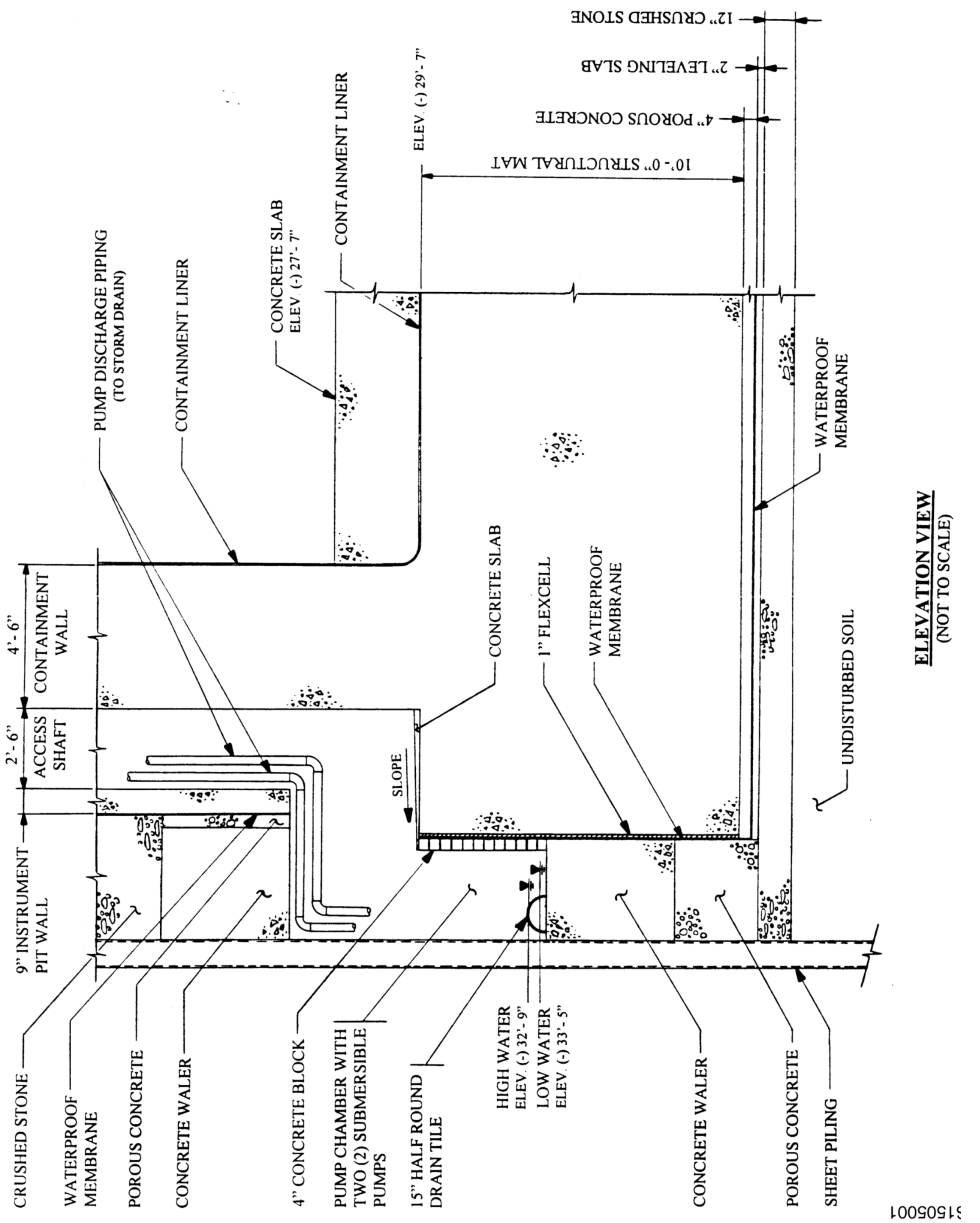


Figure 15.5-2 (SHEET 1 OF 3)
CONTAINMENT LOADING PLOT

LOADING: (1.0 ± 0.05) D + 1.0 P + 1.0 (T + %L) + 1.5 E. EQ. 2

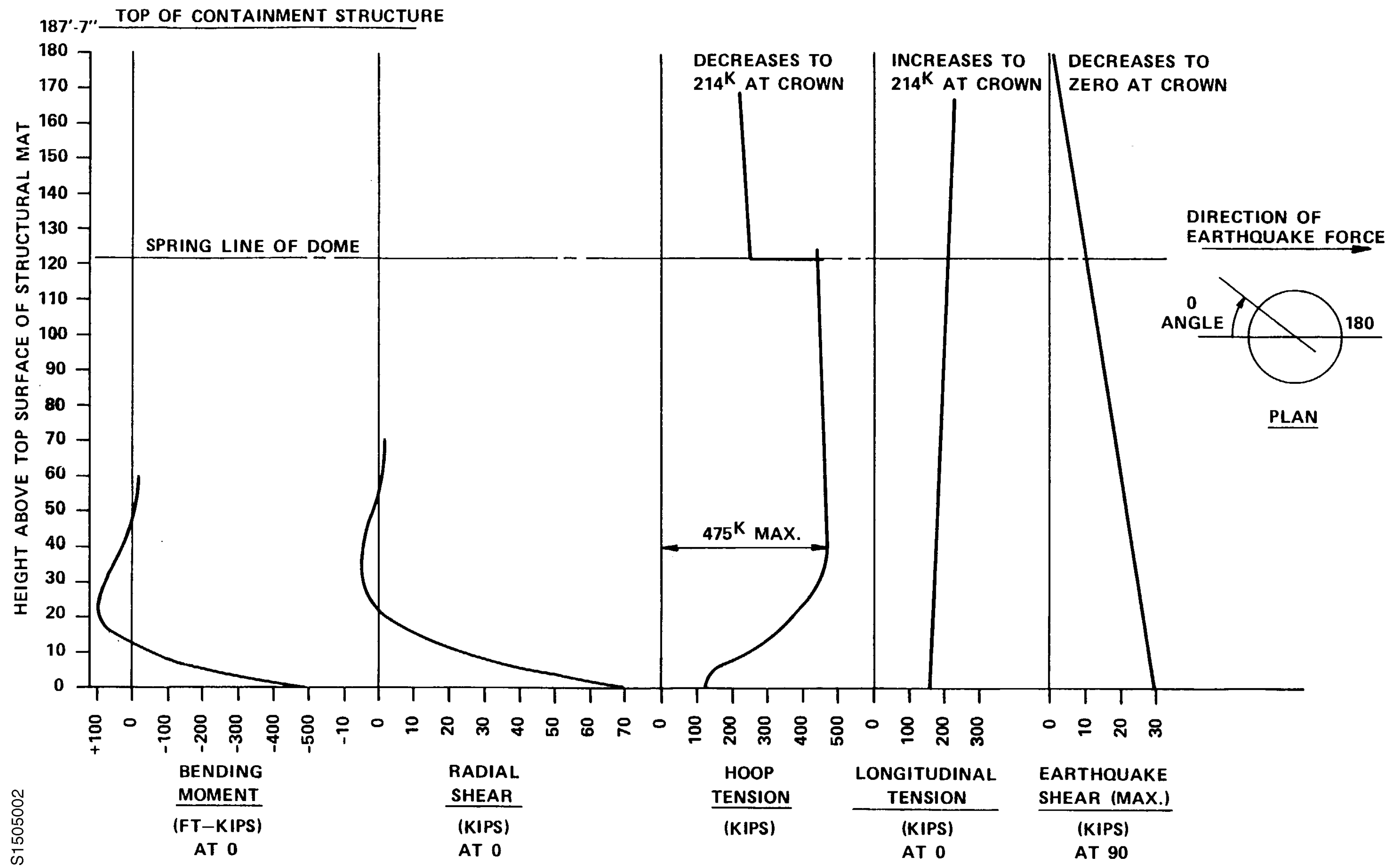


Figure 15.5-2 (SHEET 2 OF 3)
CONTAINMENT LOADING PLOT

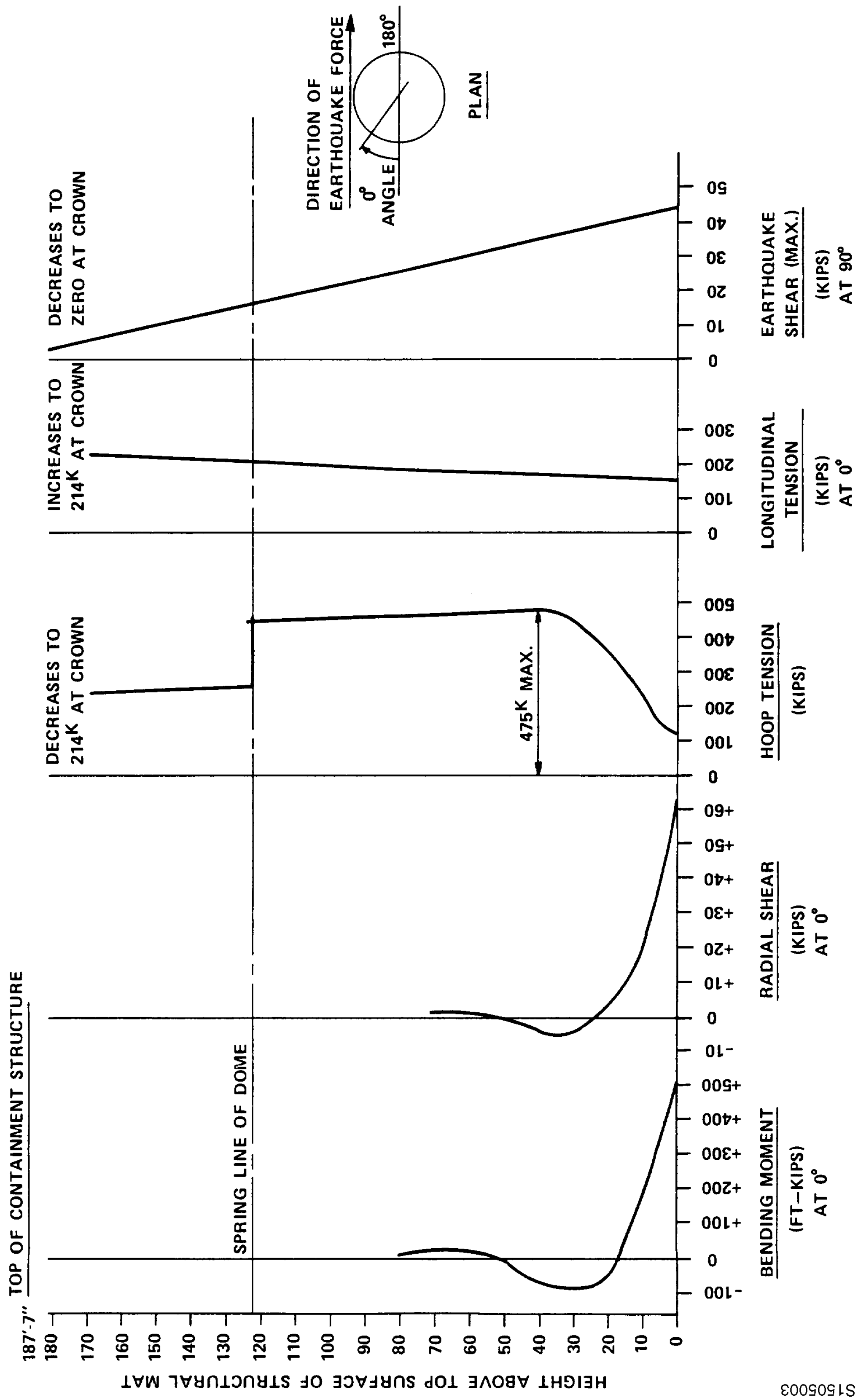


Figure 15.5-2 (SHEET 3 OF 3)
CONTAINMENT LOADING PLOT

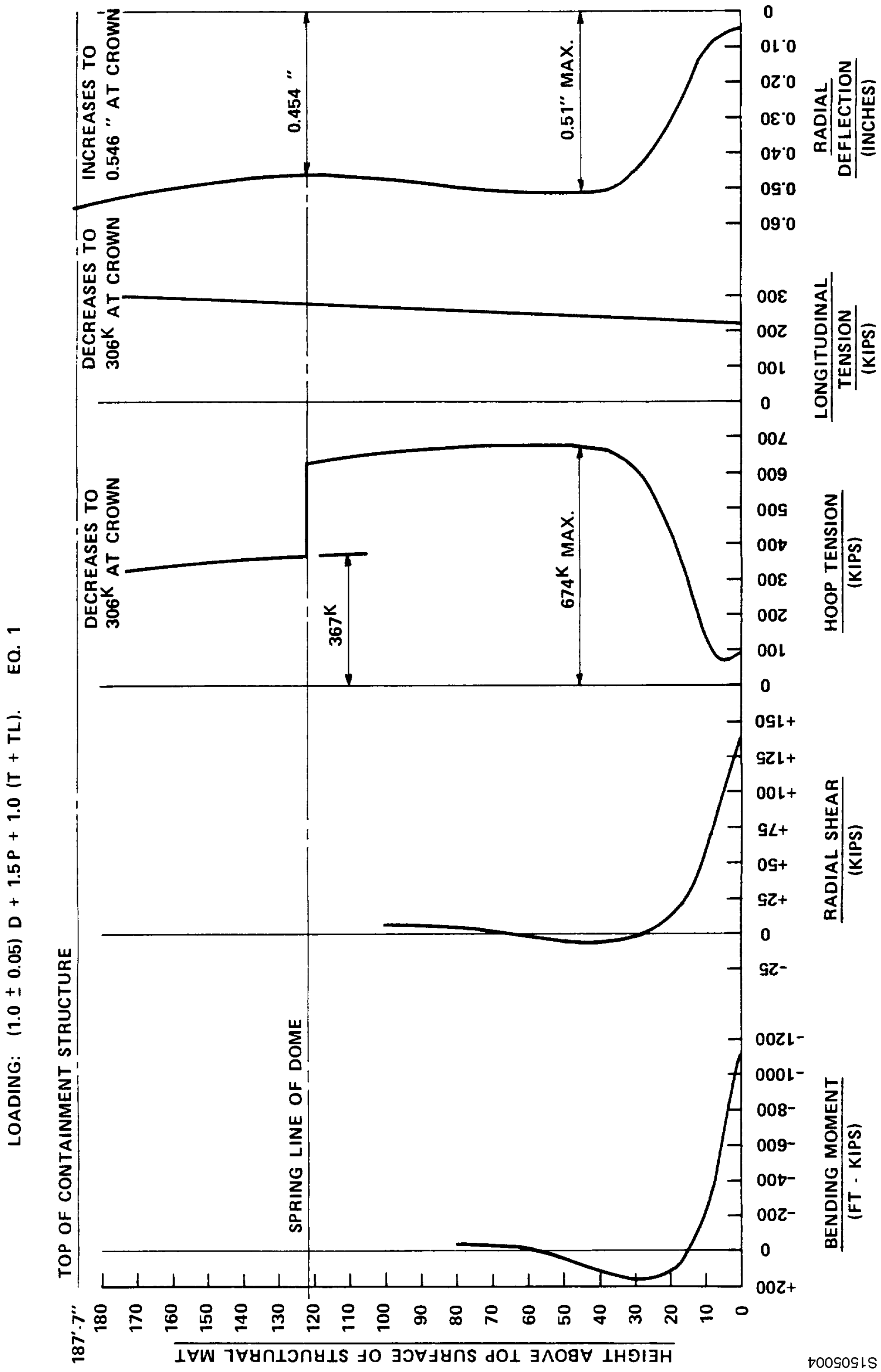


Figure 15.5-3
REINFORCING DETAILS EQUIPMENT ACCESS HATCH OPENING

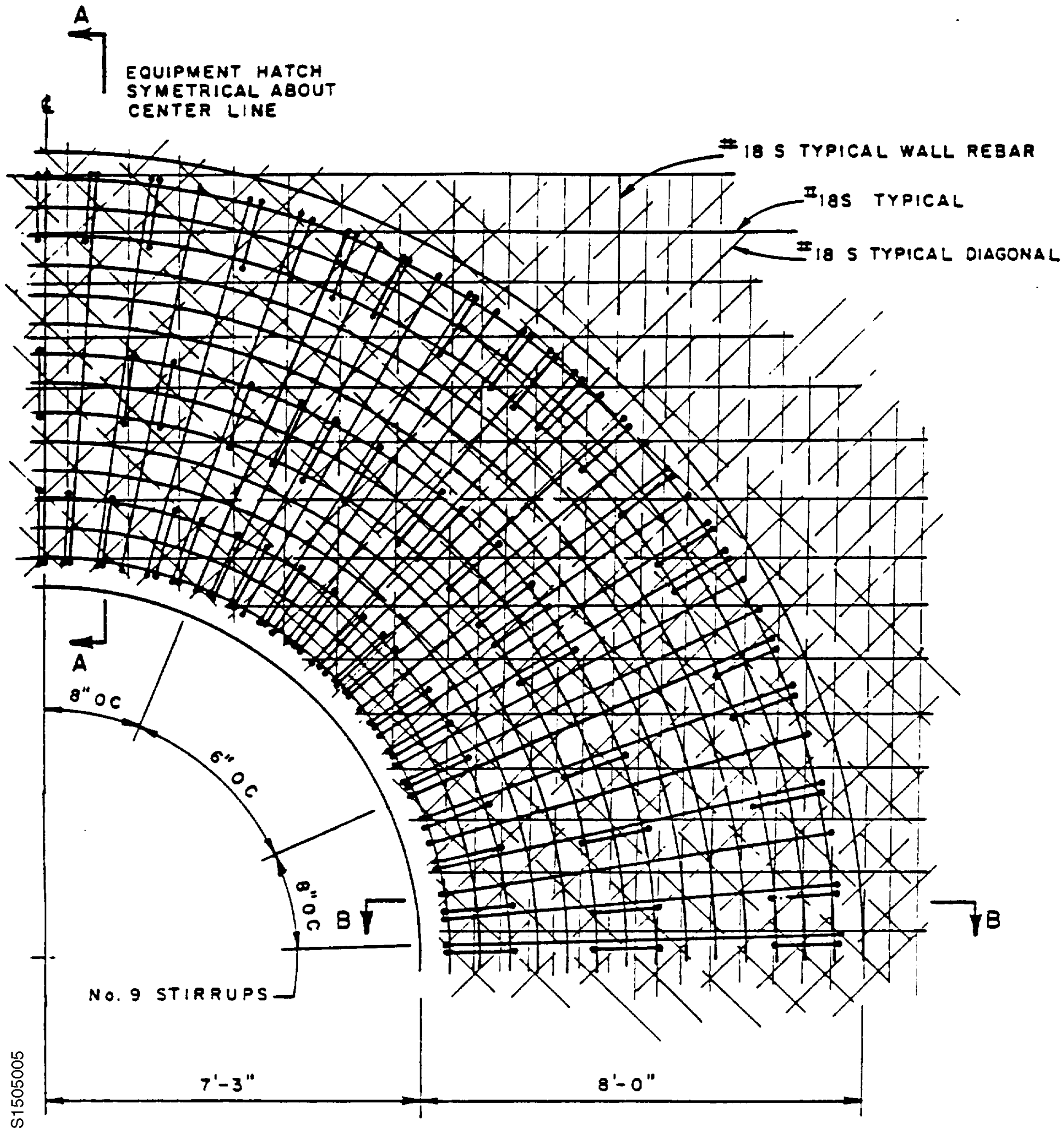


Figure 15.5-4
REINFORCING DETAILS SECTIONS THROUGH RING BEAM
EQUIPMENT ACCESS HATCH

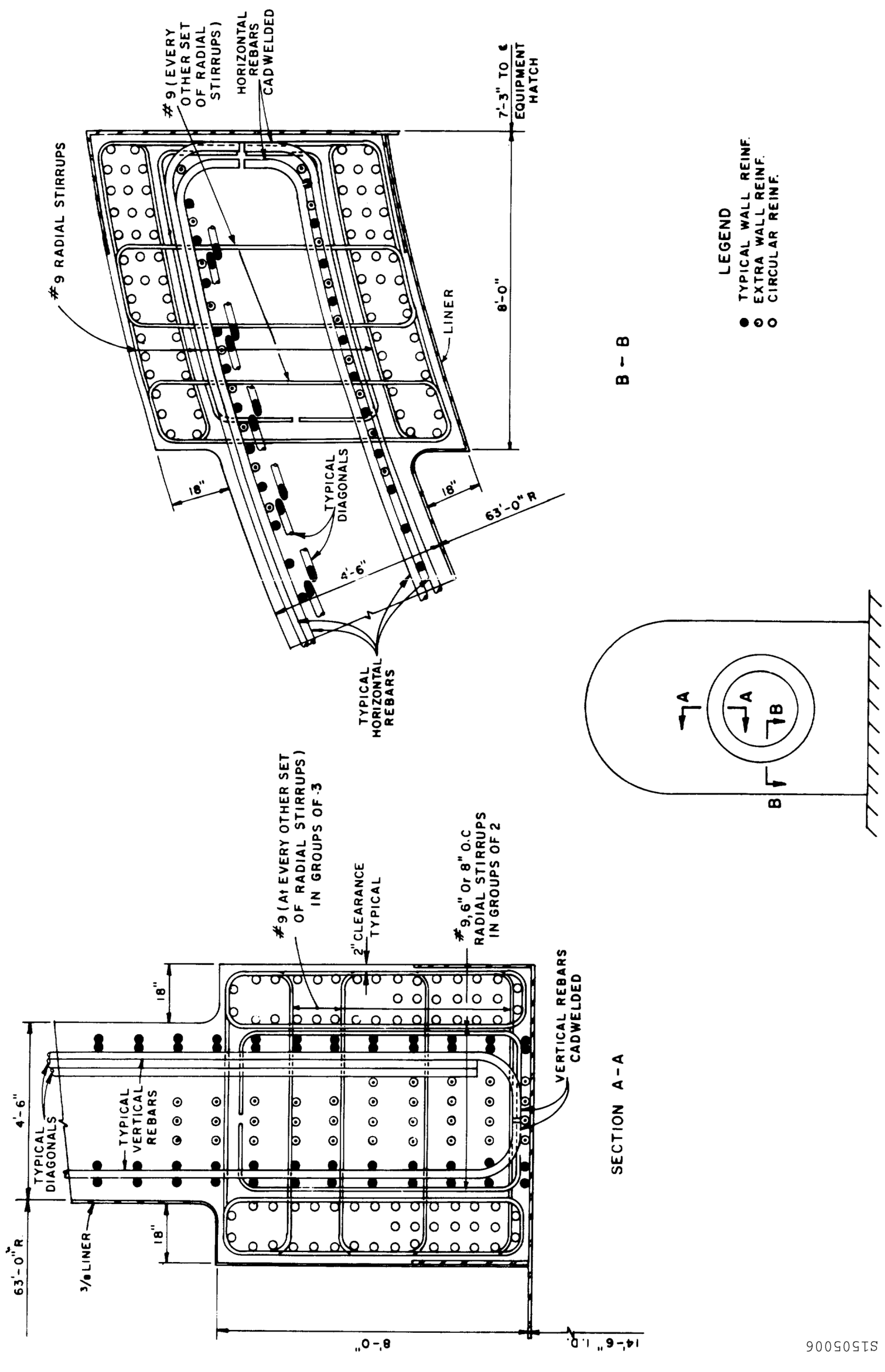
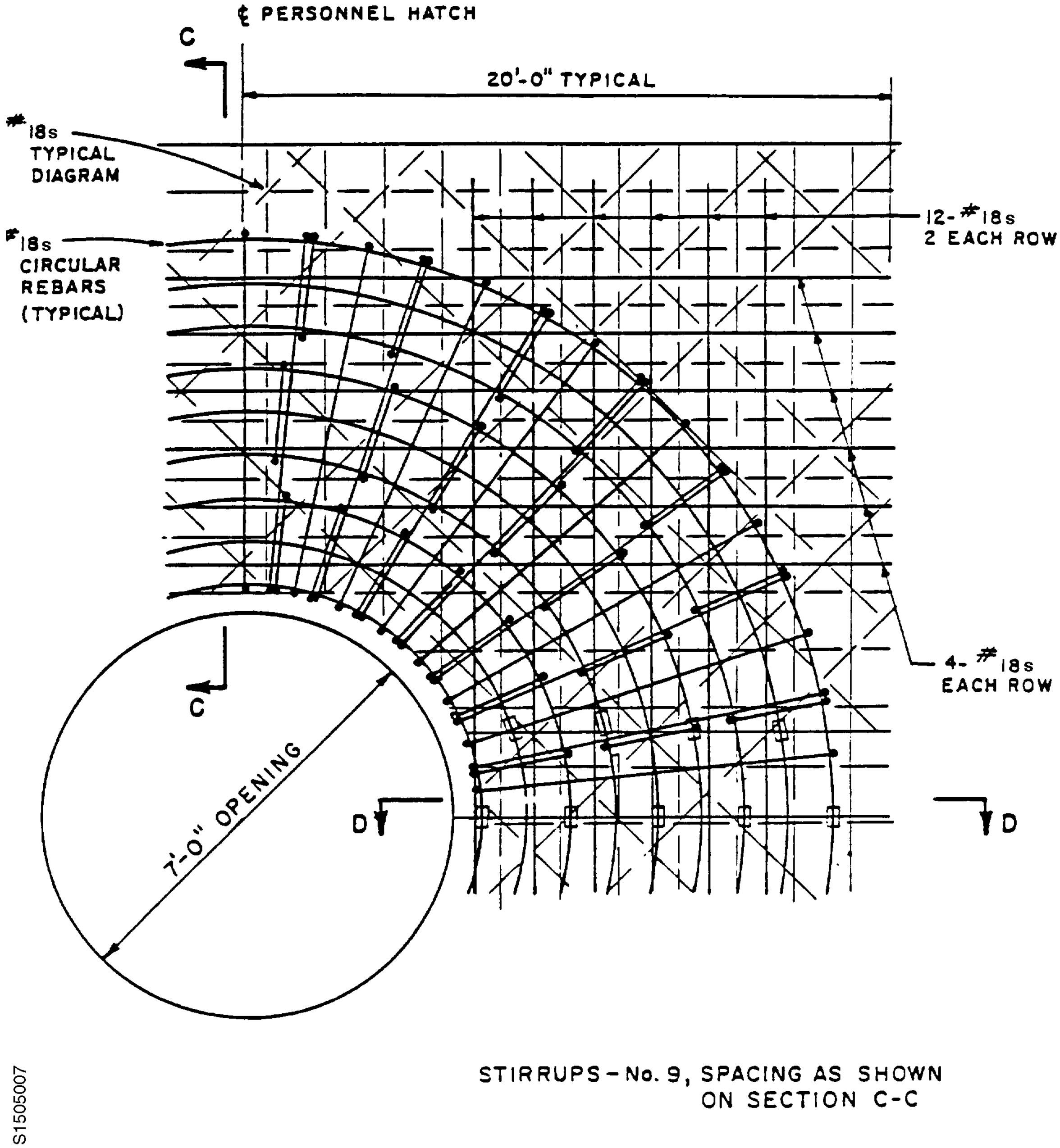


Figure 15.5-5
REINFORCING DETAILS PERSONNEL HATCH OPENING



(Figure 15.5-6)

Figure 15.5-6
REINFORCING DETAILS SECTIONS THROUGH RING BEAM PERSONNEL HATCH

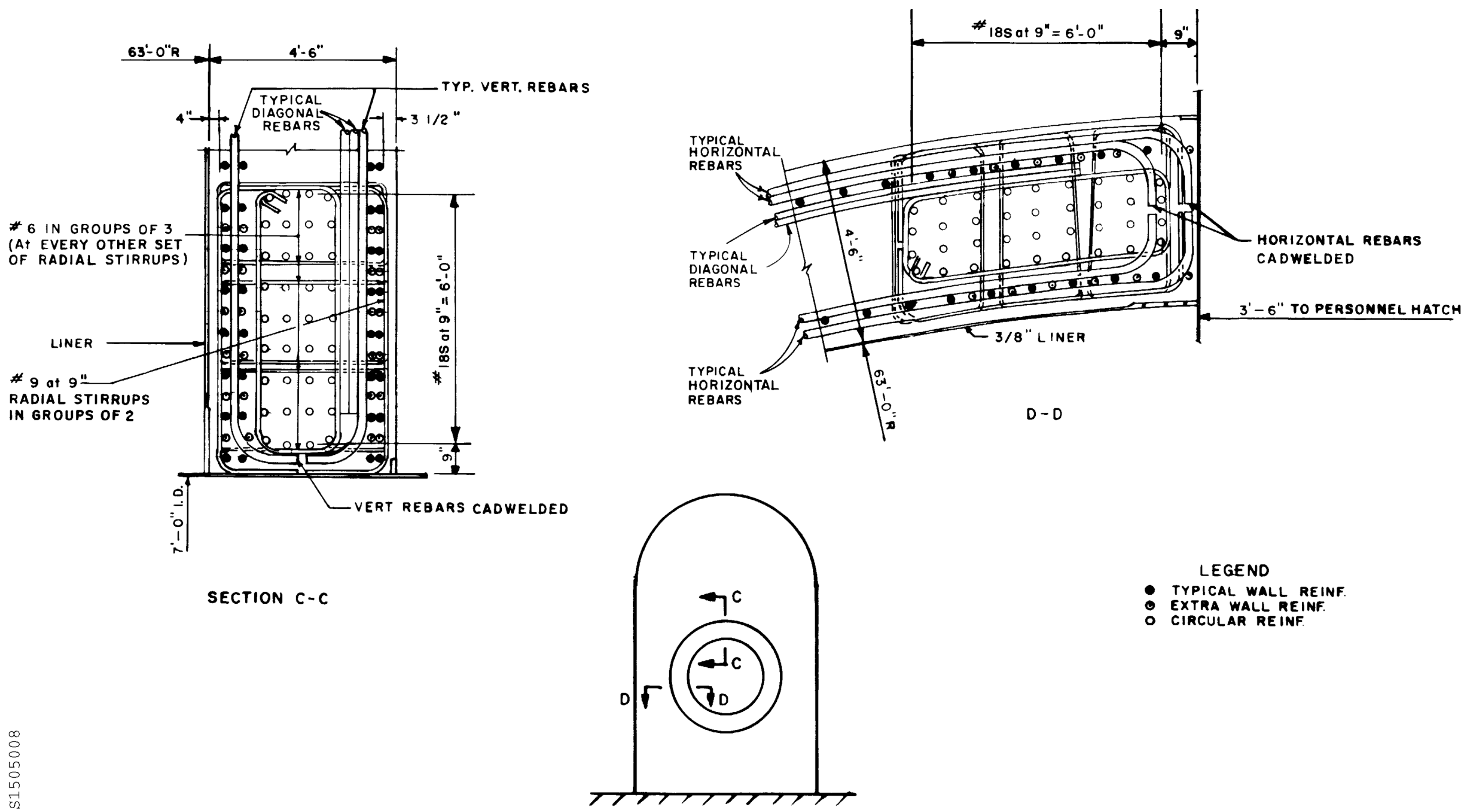


Figure 15.5-7
WALL AND MAT JOINT

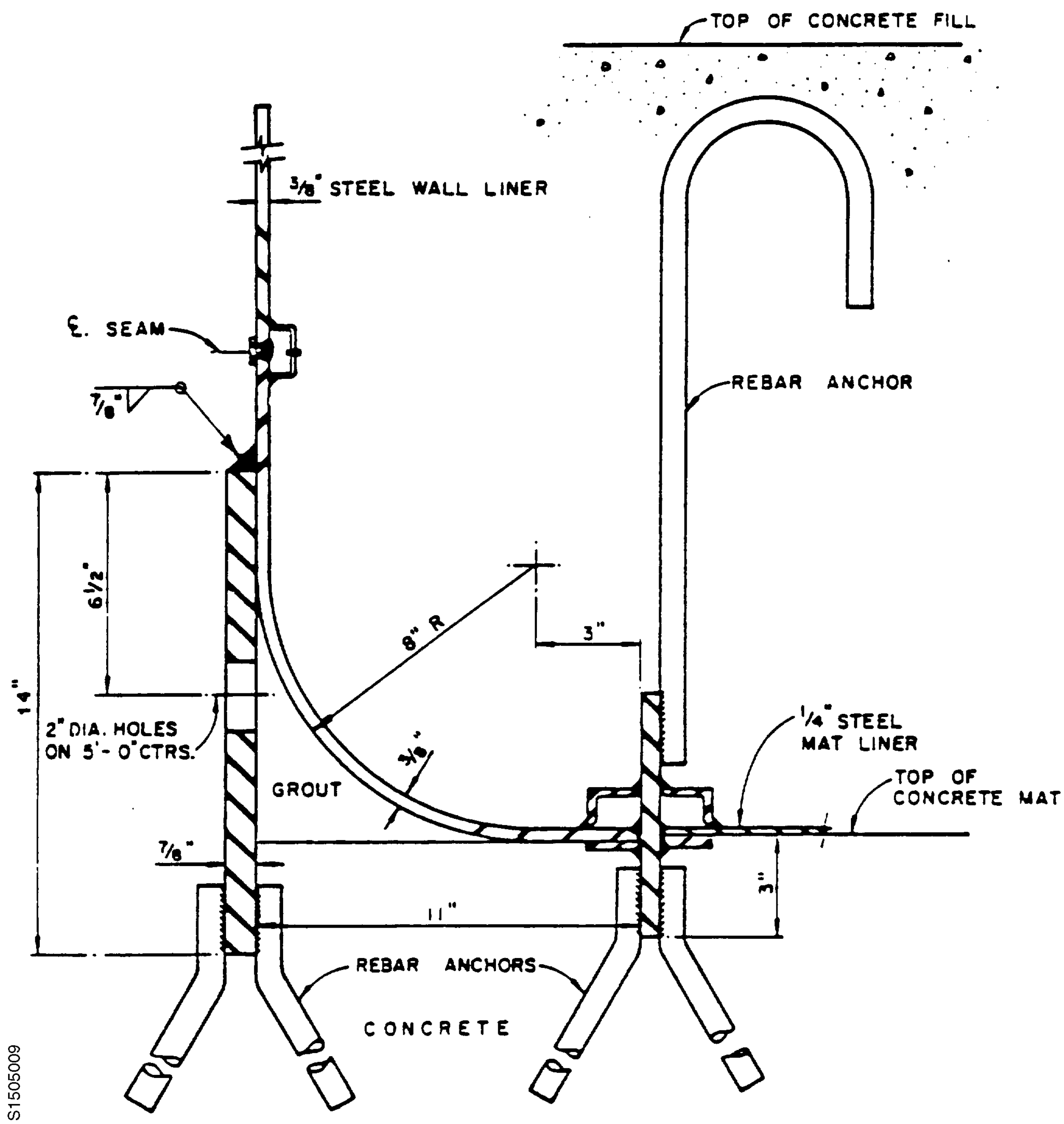
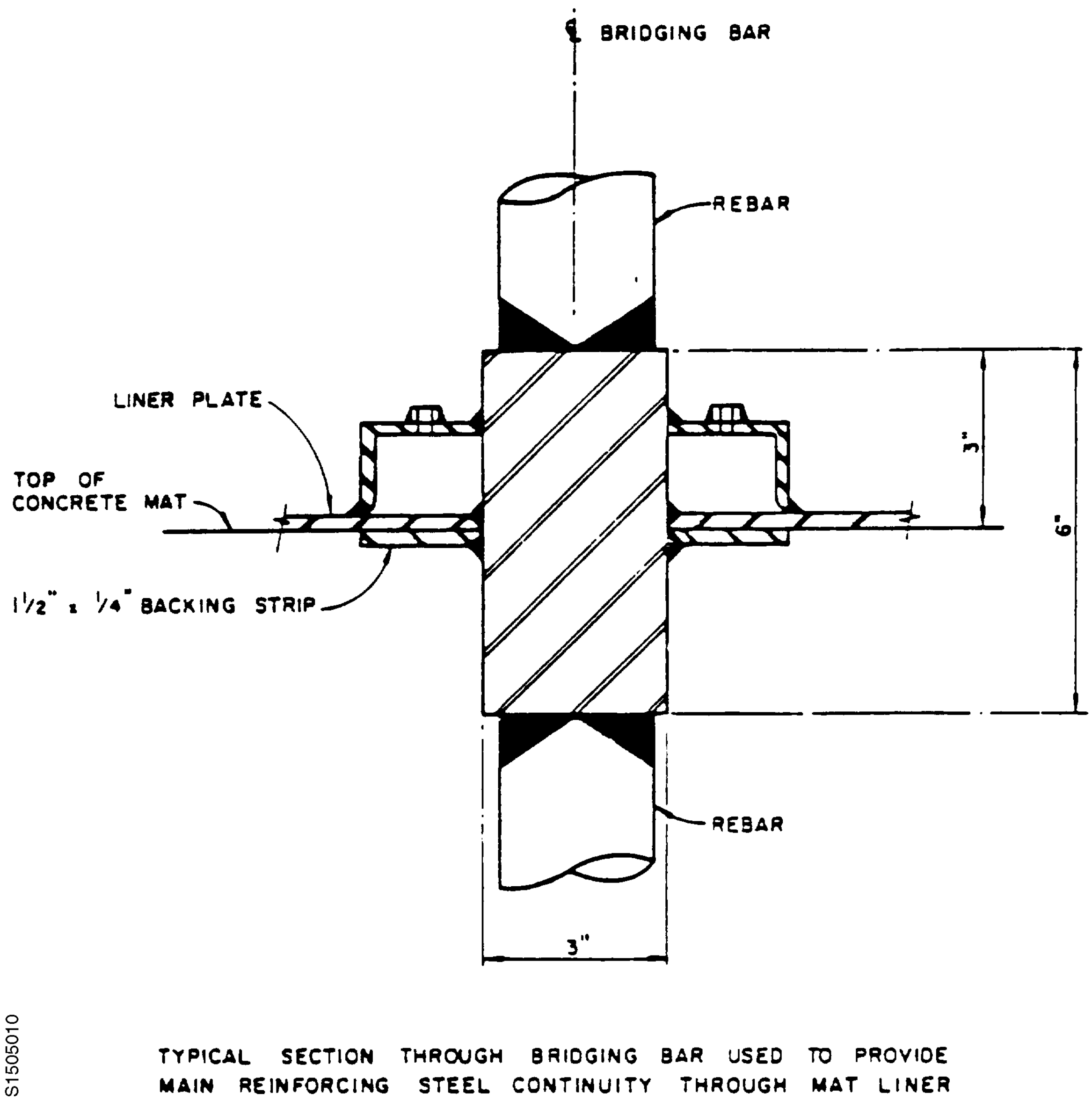
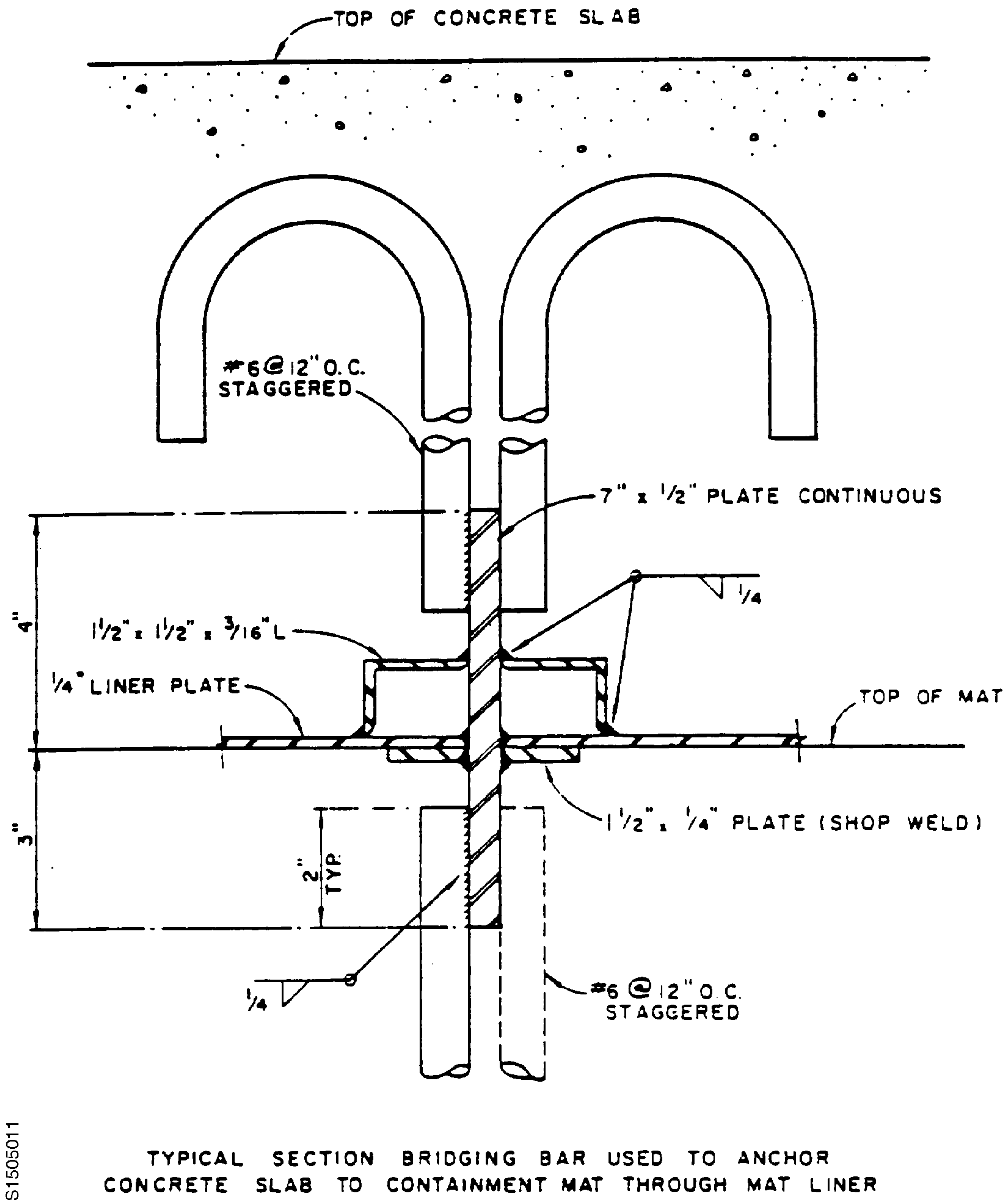


Figure 15.5-8 (SHEET 1 OF 2)
SECTION-TYPICAL BRIDGING BAR



TYPICAL SECTION THROUGH BRIDGING BAR USED TO PROVIDE
MAIN REINFORCING STEEL CONTINUITY THROUGH MAT LINER

Figure 15.5-8 (SHEET 2 OF 2)
SECTION-TYPICAL BRIDGING BAR



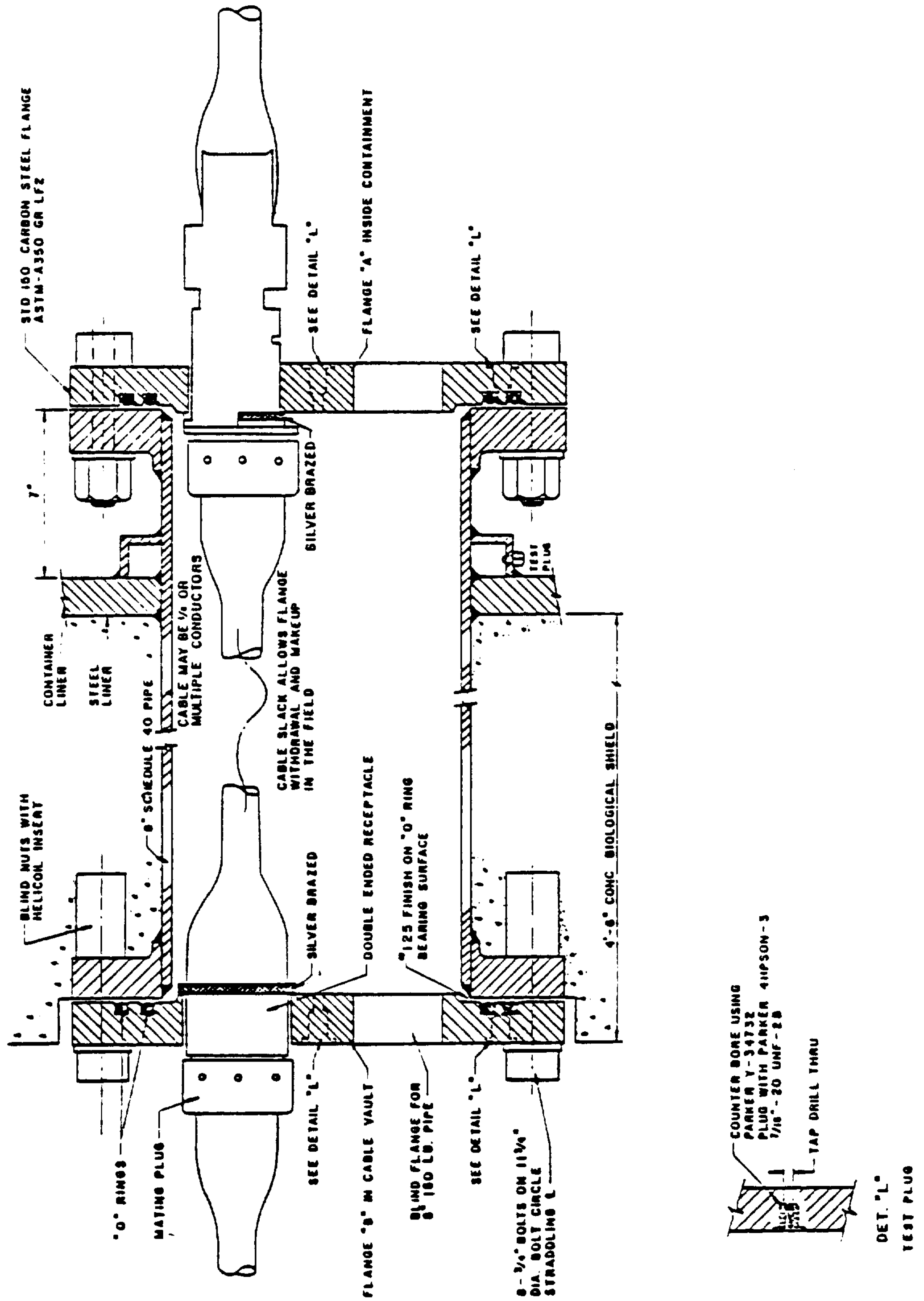
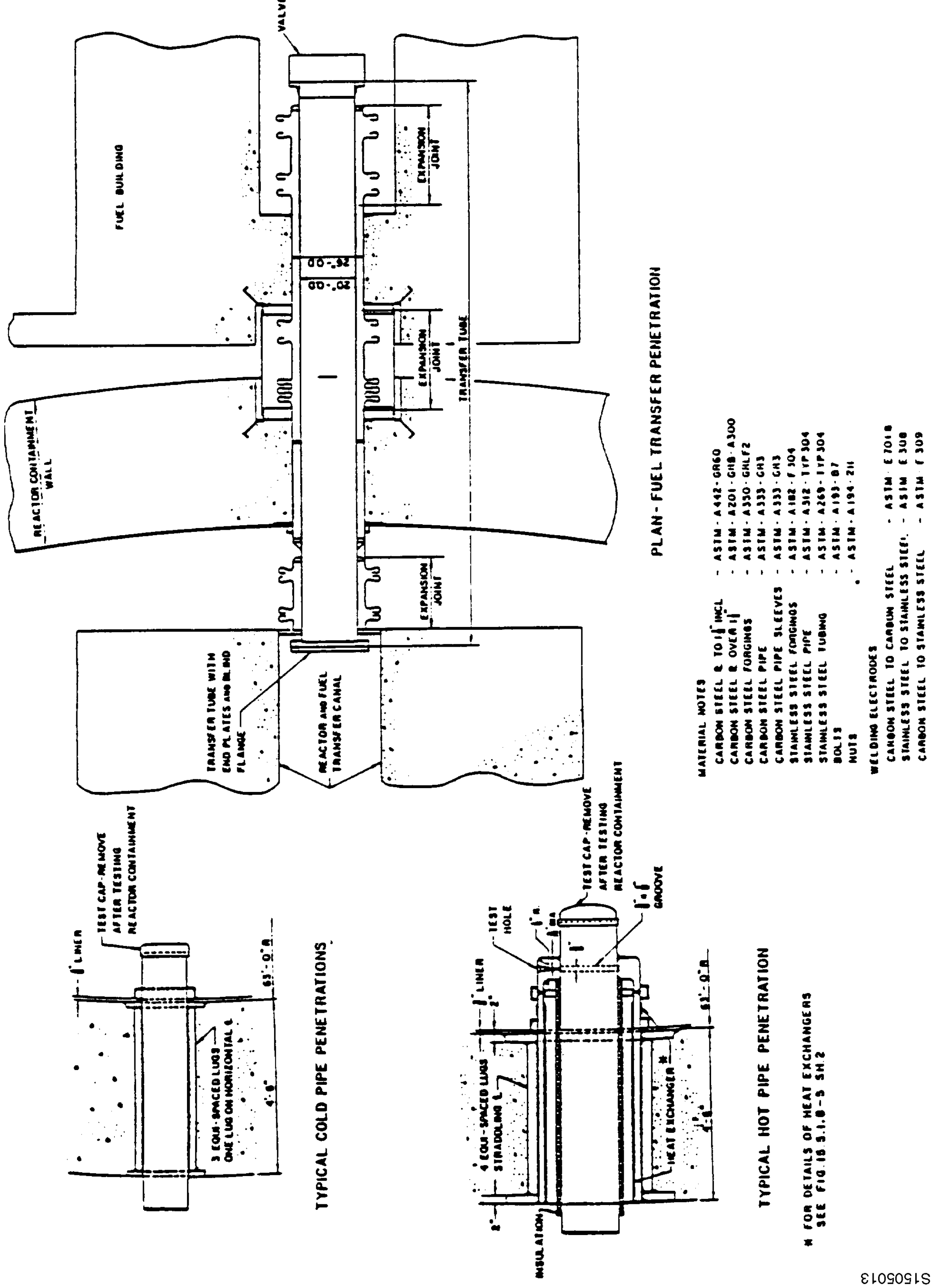


Figure 15.5-9
TYPICAL ELECTRICAL PENETRATION SLEEVE WITH FLANGES

S1505012

Figure 15.5-10 (SHEET 1 OF 2)
TYPICAL PIPING PENETRATIONS



S1505013

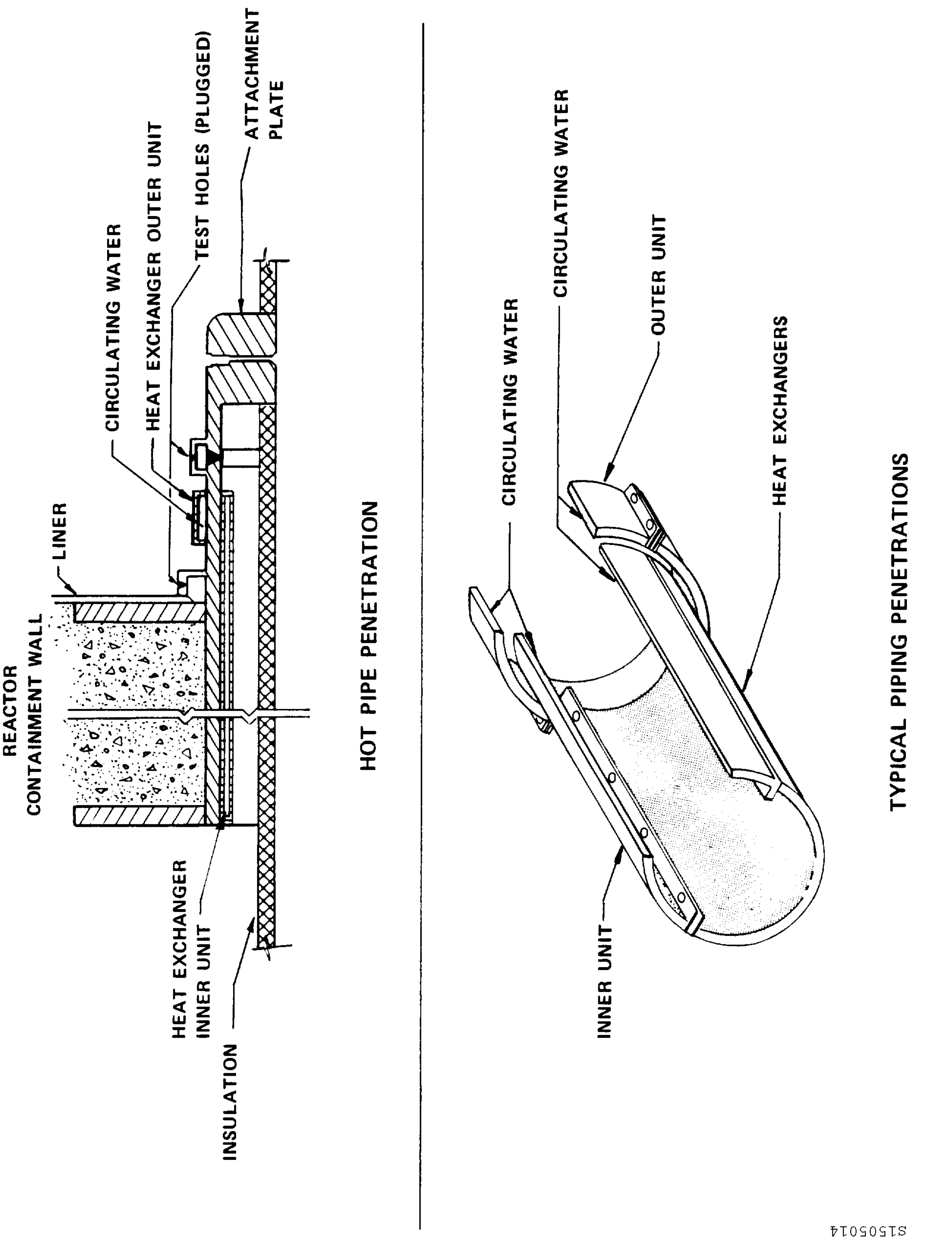


Figure 15.5-10 (SHEET 2 OF 2)
TYPICAL PIPING PENETRATIONS

Figure 15.5-11
PERSONNEL HATCH ASSEMBLY

Figure 15.5-12
TYPICAL LINER DETAILS

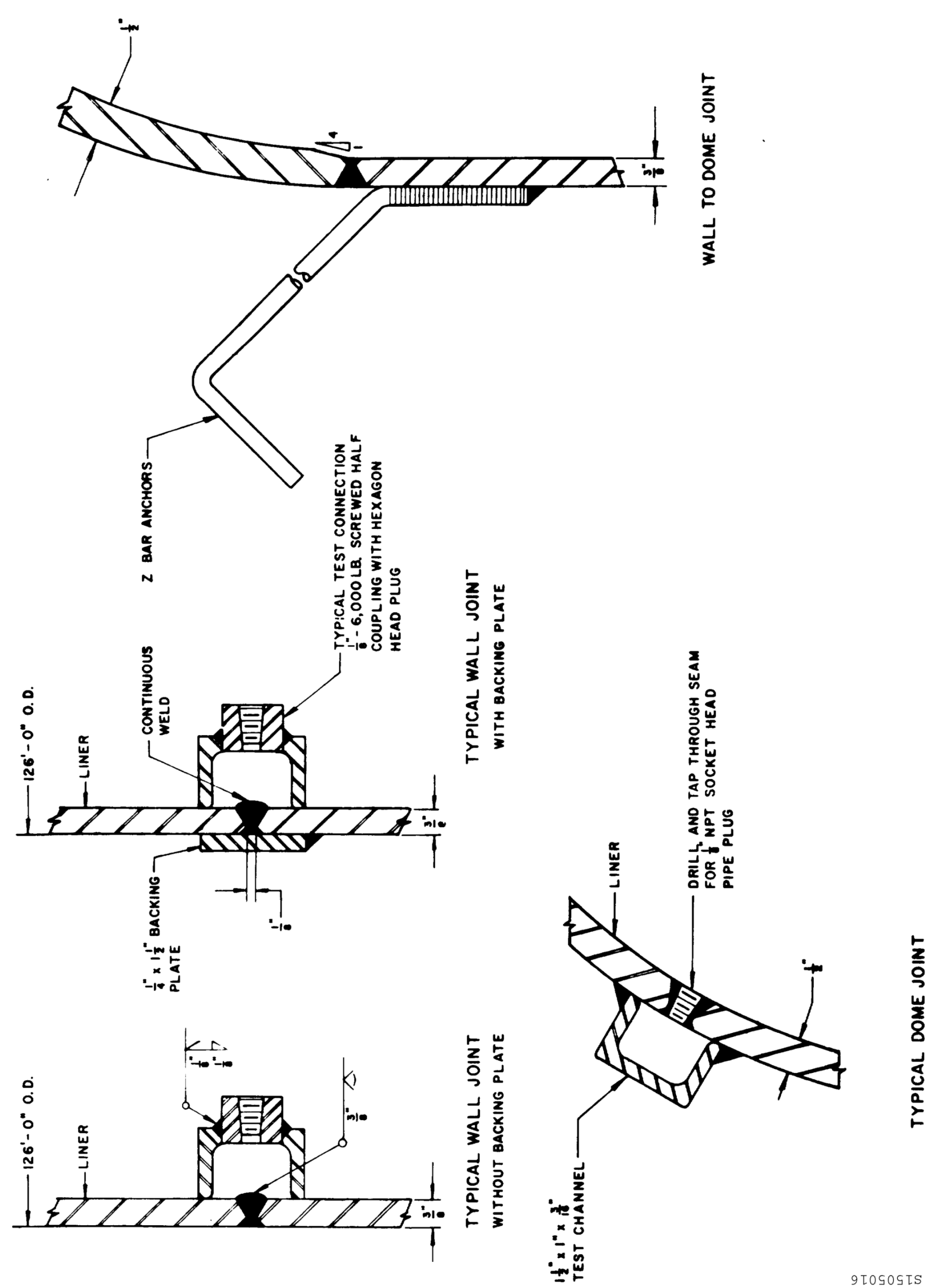


Figure 15.5-13
CONTAINMENT LOADING PLOT

Figure 15.5-14
MACHINE SHOP REPLACEMENT FAC.; SOUTH ELEVATION

Figure 15.5-15
MACHINE SHOP REPLACEMENT FAC.; SITE PLAN

Figure 15.5-16
MACHINE SHOP REPLACEMENT FAC.; EAST/WEST ELEVS

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15.6 OTHER CLASS I STRUCTURES

Class I structures other than the reactor containment structure are listed in Table 15.2-1. The major structures include the auxiliary building, control room area, including switchgear and relay rooms; fuel building; emergency diesel-generator rooms; containment auxiliary structures that contain main steam and feedwater isolation valves, recirculation spray and low-head safety injection pump cubicles, auxiliary steam generator feed pump cubicle, and safeguards ventilation room; and circulating water intake structures, including the high-level canal.

The fuel building, the main steam and feedwater isolation valve section of the containment auxiliary structures, and the refueling water storage tanks are supported on reinforced-concrete mats on concrete-filled steel pipe piles. All other structures are soil-supported on reinforced-concrete mats or spread footings. All Class I structures designed to meet tornado missile criteria, as listed in Table 15.2-1, are enclosed with heavily reinforced, 2-foot-thick concrete walls and roof slabs with all openings shielded against missiles.

Class I structures are designed to resist the operating-basis earthquake without exceeding allowable working stresses, where allowable stresses are one-third above the normal applicable code normal working stress. For concrete structures, a 5% critical damping function is assumed, and the accelerations selected from the acceleration response spectrum curves are considered in conjunction with the natural frequency of each structure. A check has been made to ensure that collapse-type failures will not occur under the design-basis earthquake. For this condition, a 10% damping factor is assumed for concrete structures, and stresses are limited to not more than 120% of the minimum yield point stress. In practice, the controlling feature of the design of these structures was to the satisfaction of the operating-basis earthquake requirements with the limited 5% damping factor.

The high-level intake canal is formed by excavating to Elevation +5 ft. from an average grade of approximately 35 feet. Earth fill dikes constructed on either side of the canal bring the finished height to Elevation +36 ft. throughout the length of the canal. The interior surfaces of the canal are lined with a 4.5-inch-thick reinforced-concrete slab. Under drains and pressure relief valves are provided to prevent uplift of the concrete liner by unbalanced hydrostatic pressure.

15.6.1 Other Structures

All other structures are designed to adequately support all dead, live, and wind loads. Where necessary, subsurface walls and slabs are designed to resist the horizontal component of the soil with applicable surcharge and hydrostatic pressures.

Structural steel design conforms to the 1963 issue of the *Specification for the Design, Fabrication and Erection of Structural Steel for Buildings* of the American Institute of Steel Construction. All concrete work is designed in accordance with the *Building Code Requirements for Reinforced Concrete*, serial designation 318-63 of the American Concrete Institute. Access and egress requirements, as well as fire ratings of walls and floor systems, satisfy the

requirements of the Basic Building Code of the Building Officials Conference of America, 1966 issue.

Under the design-basis accident loading, the allowable stresses do not exceed 90% of the minimum yield strength of the structural steel. From mill test reports, the yield strength of structural steel is 42,000 psi, with an ultimate strength of 63,000 psi. Using a minimum yield of 36,000 psi for A36 steel, the design-allowable stress is 90% of 36,000 = 32,400 psi.

Design-allowable stress for structural steel is $\frac{32,400}{63,000}$ = 51.5% of the ultimate strength.

Tests on special reinforcing steel with a minimum yield of 50,000 psi have resulted in yield strength of 55,500 psi, with an ultimate strength of 90,000 psi; with a design-allowable stress of 90% of minimum yield, the design-allowable stress is $0.9 \times 50,000 = 45,000$ psi.

Design-allowable stress on reinforcing is $\frac{45,000}{90,000}$ = 50% of ultimate strength.

Concrete continues to increase in strength beyond the 28-day strength of 3000 psi. The Bureau of Reclamation Concrete Manual indicates that Type II cement concrete can be expected to increase in strength approximately 30% in 1 year from the 28-day strength.

Approximate 28-day strength for 3000-psi concrete from test reports = 3800 psi

Design allowable 85% of 3000 psi = 2500 psi

Ultimate strength in 1 year = $1.3 \times 3800 = 4950$ psi

Design allowable is $\frac{2500}{4950}$ = 51% of ultimate strength in 1 year.

The above figures show that, for structures designed for the design-basis accident loading, structural steel and reinforced concrete are designed at approximately 50% of their ultimate strength. In the design of concrete structural members under design-basis accident conditions, concrete strength is not the controlling factor.

Allowable soil bearing values for foundations are determined from the soil boring logs and the results of triaxial shear tests of the soil. Applicable factors of safety are applied to the test results.

15.6.2 Reactor Coolant System Supports

The reactor coolant system includes the reactor vessel, three steam generators, three reactor coolant pumps, and a pressurizer for each unit. Structures are provided to support these heavy vessels and equipment, and to ensure system integrity during normal operation and design-basis accident conditions.

The primary equipment supports of the reactor coolant system are designed to withstand the design-basis earthquake acting simultaneously with an instantaneously applied pipe break. The original configuration of the Reactor Coolant System (RCS) equipment supports included ten

large-bore (12-inch diameter) hydraulic snubbers per loop to carry the loads from postulated reactor coolant system, main steam line and feedwater pipe ruptures.

Subsequently, studies to address Unresolved Safety Issue A-2 (effects of asymmetric pressure loads resulting from PWR primary loop ruptures) concluded that the probability of rupture of the primary coolant loop is extremely small, and that the presence of a pipe crack could be detected by leakage well before the crack grew to critical size which would cause rupture. These “leak-before-break” analyses, documented in References 1 and 2, were submitted to the NRC on behalf of the Westinghouse Owners Group, which included Surry Power Station.

NRC Generic Letter 84-04, *Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Loops*, provided the NRC staff safety evaluation concluding that, provided certain specific conditions are met, an acceptable technical basis exists so that asymmetric pressure loads resulting from pipe breaks in the reactor coolant system primary loop need not be considered as a design basis for the reviewed plants. The plant-specific conditions were a limitation on the maximum bending moments in the primary loop piping for normal operating and seismic loadings, and the existence of an adequate reactor coolant leakage detection system. The affected plants were encouraged to submit requests for partial exemptions from General Design Criterion 4 (GDC-4), to permit elimination of pipe rupture restraints required to protect against these previously postulated breaks.

Because a number of the large-bore snubbers served primarily as pipe rupture restraints, Surry proceeded with an exemption request to allow elimination of 6 of the 10 snubbers per loop, based on application of leak-before-break in accordance with Generic Letter 84-04. (The other 4 large-bore snubbers on the upper steam generator support were required for lateral loads due to a postulated longitudinal split of the main steam line.) The reactor coolant loop system was re-evaluated with all snubbers on the steam generator lower support and the reactor coolant pump supports eliminated, to assure that the conditions of pressure, deadweight, thermal, seismic, and remaining pipe rupture effects, would not result in unacceptable stress levels or factors of safety. Largely independent analyses were performed by Westinghouse and Stone & Webster in accordance with the original division of design responsibilities. Interface force allowable limits at NSSS boundaries were assured and support design load interfaces were reviewed for acceptance.

The exemption request to allow elimination of 18 snubbers per unit was filed with the NRC on November 5, 1985 (Reference 3). The detailed technical basis (Reference 4) provided separate attachments addressing load evaluation, leakage detection, and net safety balance. The proposed design changes were discussed with the NRC and resulting NRC concerns were addressed (References 5 & 6). The GDC-4 “limited scope” revision (Reference 7) was subsequently published (effective May 12, 1986) permitting the use of leak-before-break technology to justify elimination of the dynamic effects of primary loop breaks from the design basis of PWRs. With the publication of the notice of revision to GDC-4, a license amendment to the plant design basis was requested to implement the snubber reductions under the new rule (Reference 8). By letter dated June 16, 1986, the NRC approved the license amendment (#108) for both units

(Reference 9). The snubber eliminations were implemented by Design Changes 85-04-1 and 86-12-2.

Subsequently, the NRC issued Generic Letter 87-11, *Relaxation of Arbitrary Intermediate Breaks* which provided the revised Mechanical Engineering Branch position eliminating the need to postulate Arbitrary Intermediate Breaks. As stated in the generic letter, the elimination of Arbitrary Intermediate Breaks allows elimination of the associated pipe whip restraints and jet impingement shields. Because the stresses in the main steam lines inside containment are well below the stress criteria for required mandatory intermediate breaks, the only breaks which need to be postulated are terminal end breaks which do not apply lateral loads to the steam generator. Therefore, the governing lateral loads on the steam generator become those imposed by the main feedwater break, which are low enough that only a single snubber in each pair will be required to carry the load. Analyses were performed by Westinghouse and Stone & Webster similar to those performed for the earlier large-bore snubber reduction to ensure piping stress levels and component factors of safety were acceptable. In addition, it was necessary to verify that the basis for the previous license amendment based upon leak-before-break of the primary loop remained valid. Comparison of results with the lower large-bore snubber of each pair removed as a result of eliminating the main steam line split vs. the results in the previous leak-before-break submittals, confirmed that there were no significant reduction in margins of safety. Therefore, elimination of the lower large-bore snubber of each pair does not compromise the bases for the previous leak-before-break analysis, namely:

1. The loading on the primary loop piping is still enveloped by the generic analyses submitted on behalf of the A-2 Owners Group, and accepted by the NRC staff in Generic Letter 84-04, and specifically for Surry by NRC letter dated June 16, 1986; and
2. The reactor coolant system equipment, piping, and supports continue to have acceptable margins of safety under licensed loading conditions other than the now-eliminated ruptures of the primary loop piping and Arbitrary Intermediate Break of the main steam lines.

15.6.2.1 **Design Basis**

All supports in the reactor coolant system are designed to withstand the design-basis earthquake acting simultaneously with an instantaneously applied pipe break. As discussed above, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop. However, single ruptures are postulated to occur in either the pressurizer surge or other reactor coolant branch lines, the main steam piping, or the main feedwater piping. In general, two types of piping failures are considered: a double-ended rupture, or a longitudinal rupture on either the horizontal or vertical axis of the pipe. The longitudinal rupture area was taken to be equal to the area of the double-ended rupture for these piping failures. Stresses in the main steam piping inside containment have been reviewed and, in accordance with NRC Generic Letter 87-11, are sufficiently low that no intermediate break need be postulated. Only terminal end breaks need be postulated; in accordance with Generic Letter 87-11, it is not necessary to postulate longitudinal rupture at terminal end breaks.

Therefore, only vertical loads are considered to be applied to the steam generators due to a postulated main steam line break. For all postulated breaks, the pipe rupture loads are combined with design-basis seismic loads by the square-root-sum-of-squares (SRSS) method.

The peak value of the pipe thrust for any of the main steam piping breaks considered is approximately 620,000 lb; and the peak value for the pipe thrust for the pressurizer surge pipe break is approximately 195,000 lb. These thrust values are equal to $P \times A$, the system pressure times rupture area.

For the pressurizer surge line break and other branch line breaks, the load versus time transients of these breaks are provided by a computer program that analyzes the shock wave initiated at the break as it passes through the complete piping loop. Results from this program are used as forcing functions in a structural dynamic program that results in the dynamic loadings of the supports. For the main steam line and feedwater line breaks, the dynamic forces were applied only at the steam generator nozzles, because the primary reactor coolant loop piping remains intact. The peak values of the pipe thrust for the postulated piping breaks were computed as $C_r \times P \times A$, the thrust coefficient times the system pressure times the rupture area. These values were used as the basis for developing conservative time-history forcing functions of the postulated breaks. In addition to the thrust loading, jet impingement effects were included as appropriate. For the main steam line vertical break, credit was taken in calculating the thrust loading for the flow area reduction of the flow restrictors installed at the nozzle during the steam generator replacement project; the jet impingement loading is not reduced by the flow reducers. The time-history forcing functions are input into the structural dynamic analysis program that calculates maximum loadings on the supports. For each analyzed break, the maximum support loads are determined and then combined by SRSS summation with design-basis earthquake loads, and then added directly to the loadings due to normal operation. Combined stresses are maintained within 90% of the minimum yield point of the structural material used. For the RPV sliding foot supports, LOCA loads are developed using the AREVA methodology described in Chapter 14 Section 14.5.3.4.1.

All welding is in accordance with Section IX of the ASME Code, and all welds are examined by either radiographic, sonic, dye penetrant, or magnetic particle techniques, depending on the material and the state of stress at the weld.

The seismic restraints (snubbers) that are installed on the piping systems throughout the plant and that are required to protect the reactor coolant system or any other safety-related system are subject to operability and surveillance requirements contained within the Technical Specifications. Vepco has established a program and procedures for inspecting, testing, and maintaining snubbers in compliance with the Technical Specification requirements. A listing of all safety-related hydraulic and mechanical snubbers is maintained by Surry Power Station.

15.6.2.2 **Description**

15.6.2.2.1 Reactor Vessel Support

The reactor vessel is supported by six sliding foot assemblies mounted on the neutron shield tank. The support feet are designed to restrain lateral and rotational movement of the reactor vessel while allowing thermal expansion. The neutron shield tank is a double-walled cylindrical structure that transfers the loadings to the heavy reinforced-concrete mat of the containment structure. The tank also serves to minimize gamma and neutron heating of the primary concrete shield, and to attenuate neutron radiation outside of the primary shield to acceptable limits (Section 11.3.2.1). The neutron shield tank assembly and material listing are shown on Figure 15.6-1.

Sliding support blocks mounted on top of the shield tank support the reactor vessel. These sliding support blocks permit radial thermal expansion of the reactor vessel, while preventing translation, rotation, or uplifting[1]. The support blocks are also designed to adjust to the correct height for plumbing the reactor vessel and for distributing the load properly among the six supports.

The loading conditions used in the analysis of the neutron shield tank were the simultaneous accelerations of a design-basis earthquake, the thrust forces exerted by the reactor vessel due to a double-ended reactor coolant pipe rupture,[2] and the dead weight of the system and the tank itself, with stresses not to exceed allowable working stresses, and with no loss of integrity or function.

The neutron shield tank has been designed using the theory of a shell structure, dynamically loaded in both horizontal and axial planes, which results in meridional and circumferential stresses at all points along its length. The stresses in the tank were determined by using the methods and theories of Timoshenko for plates and shells, elasticity, and elastic stability. All membrane stress levels were held to the limits stated in Section VIII of the ASME Code; all membrane-plus-bending stresses were held to 90% of yield point. Section VIII of the ASME Boiler and Pressure Vessel Code was used as a guide in the fabrication and welding of the tank. A code stamp was not required, since this is not a code pressure vessel but a supporting structure for the reactor pressure vessel.

All material employed in the fabrication of the tank was new and conformed to ASTM Standards. The tank shell was constructed from ASTM A516, Grade 60, and the six sliding support blocks were made of maraged steel. The material has an NDT of -20° to -40°F. Drop weight tests were performed to determine the nil ductility transition temperature of the deposited weld metal in welding the ASTM A516, Grade 60 material with an NDT of -40°F. The maraged

1. It is no longer necessary for the RPV sliding foot supports to restrain uplift based on new LOCA loads developed as discussed in Section 14.5.3.4.1. The combined seismic and LOCA loads when added to dead weight are not sufficient to create uplift in the supports.
2. As discussed previously in this section, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop piping. However, pipe ruptures of reactor coolant branch lines must still be considered.

steel was ultrasonically tested for flaws to the quality level of MIL-1-8950. Flaws detected ultrasonically were verified by X-ray. Maraged steel pieces with verified flaws larger than 1/32 in. were rejected. The maraged steel had a maximum hardness of 35 Rockwell C and a minimum grain size of 6, in accordance with ASTM E-112 and MIL-Std-430, *Macrograph Standards for Steel Bars, Billets, and Blooms.* The nonmetallic inclusion content for each billet was determined in accordance with ASTM E45. Fracture toughness tests were performed on maraged steel in accordance with ASTM Specification, *Proposed Recommended Practice for Plane-Strain Fracture Toughness Testing of High-Strength Metallic Materials Using a Fatigue-Cracked Bend Specimen,* Part 31, ASTM Standards.

All welds, where possible, were 100% radiographed in accordance with Paragraph UW-51 of the ASME Boiler and Pressure Vessel Code Section VIII, Division 1. Other welds that could not be radiographed were dye-penetrant checked at root pass, intermediate depths at half-inch increments, and the final pass, or magnetic-particle tested, in accordance with Appendices VI and VIII, Section VIII, ASME Boiler and Pressure Vessel Code. The surfaces of welds were ground to a surface condition suitable for the inspection procedure employed. Defects in welds were removed by chipping, grinding, or arc gouging until sound metal was reached. The resulting cavity was rewelded, employing an approved procedure.

After shop fabrication, the completed tank was subjected to a hydrostatic test of 15 psi, measured at the top of the tank. No water loss was observed for a 24-hour period. The tank was then leak tested with air at 5 psi gauge, applying soapsuds to all welds accessible from outside the tank. Leaks were repaired and the tank retested until no leakage was detected. All tests were recorded and certified. After installation of the neutron shield tank at the job site, the tank was hydrostatically retested.

15.6.2.2.2 Steam Generator Support

The steam generator support consists of two (upper and lower) cast rings and associated suspension rods, lateral restraints and snubbers. The lower ring, which carries the steam generator weight, is suspended by means of three pipe columns. Hydraulic snubber cylinders and rigid lateral guides connect the upper casting to the steam generator cubicle structure to allow guided thermal expansion of the steam generator outward from the reactor during normal operation, while resisting movement during seismic and pipe break conditions. Due to the design of this support system (i.e., pin-ended connections at all member joints) lamellar tearing of the supports could not occur. The steam generator support assembly and material listing is shown on Figure 15.6-2. The supports do not have any heavy section intersecting member weldments.

The major materials used in the construction of the steam generator supports are listed in Table 15.6-1. The difference between the operating temperature and the NDT of the material ensures toughness and ductility of the steam generator supports under all operating conditions. Welding associated with the supports was conducted in accordance with Section IX of the ASME code.

In addition to numerous inspections and tests carried out by the material suppliers and fabricators, all of the components for these supports were subject to inspection during fabrication and installation by Stone & Webster Engineering Corporation. All welds were subjected to examination by either the magnetic particle, liquid penetrant, or radiographic methods. The Vascomax 350 CVM and 300 CVM materials, and the A-352 grade LC3 casting materials were subjected to examination by the magnetic particle and ultrasonic methods. A visual inspection of a portion of these supports is required by the Technical Specifications.

The upper restraining ring is composed of two girth straps coupled together by studs to form a continuous ring. The studs are 1.25-inch 12UNF-2A Vascomax 18% Ni, maraging grade 350, with nickel cadmium coating. The nuts are 1.25-inch 12UNF-3B Vascomax 18% Ni, maraging grade 250, with Helicoil inserts. The studs and nuts are designed to minimize stress concentration during manufacture, and the studs are coated for environmental protection. The studs are pretensioned across a joint flange spacer block, which serves to reduce bending stresses in the studs.

A total of nine machined shoe openings are welded to each vessel girth strap. These shoe openings accommodate nine keys which themselves are fastened by dowels to the large upper restraint casting which is shown on Figure 15.6-2. These key and shoe openings function to allow vertical thermal expansion of the steam generator within the upper restraint casting, but will restrict lateral movement resulting from forces generated during a seismic event and/or a major pipe break applying lateral loads to the steam generator.

In a seismic event and/or a major pipe break applying lateral loads to the steam generator, the shoe openings in the vessel girth straps act against the keys, which results in a tangential load on the girth straps. The subject studs are designed to accommodate the maximum tangential load resulting from this accident condition. Existing space restrictions and restraint design required a limitation on stud size and quantity which necessitates the use of an ultra-high strength bolting material. The studs have a minimum yield strength of 326,000 psi. The nuts have a minimum yield strength of 150,000 psi.

The upper restraint casting is anchored to the containment structure (approximate 47-foot level) through rigid lateral guides oriented in the direction perpendicular to the outward thermal movement of the steam generator, and by two horizontal large-bore hydraulic snubbers, which permit the thermal movement of the steam generators outward from the reactor. The large-bore snubbers are 12-inch-diameter Pathon snubbers which have been refurbished and upgraded for increased reliability by Paul-Monroe/Remco Hydraulics under Design Change 85-05. The modifications included chroming the cylinder inner diameters; replacement of all non metallic seals with extended-service-life Tefzel seals or metallic seals; installation of poppet-type self-cleaning control valves for improved performance; conversion to the standard snubber hydraulic fluid (General Electric SF-1154); and incorporation of test-in-place features. In addition, the original common reservoir serving a number of snubbers has been replaced by individual pressurized reservoirs installed in more readily accessible locations in lower radiation

areas outside the biological shield walls. The reservoirs are plated and the tubing and fittings are stainless steel for corrosion resistance.

The lower restraining ring is also connected to the steam generator cubicle concrete structure by couplings consisting of two end plates installed perpendicular to one another. The end plates have machined dovetails and are joined by a connector plate with mating dovetails. Although the couplings are resistant to corrosion cracking, additional protection is provided by enclosing each coupling in a rubber boot filled with silicone lubricant. The boot and lubricant are compatible with the coupling and are also radiation resistant.

15.6.2.2.3 Reactor Coolant Pump Support

The reactor coolant pumps are supported by a four-legged suspended structure. (Four small-bore snubbers originally installed in the upper frame structure of the pump support were eliminated under Design Changes 85-04-1 and 86-12-2. The dynamic characteristics of the reactor coolant pump were not significantly affected by removal of these snubbers as discussed in Reference 6.) The reactor coolant pump support assembly and material listing are shown on Figure 15.6-3. The supports do not have any heavy section intersecting member weldments.

The major materials used in the construction of the reactor coolant pump supports are listed in Table 15.6-2. Welding associated with the supports was conducted in accordance with Section IX of the ASME Code.

In addition to numerous inspections and tests carried out by the material suppliers and fabricators, all of the components for these supports were subject to inspection during fabrication and installation by Stone & Webster Engineering Corporation. All welds were subjected to examination by either the magnetic particle, liquid penetrant, or radiographic methods. The Vascomax 350 CVM and 300 CVM materials, and the A-352 grade LC3 casting materials were subjected to examination by the magnetic particle and ultrasonic methods. A visual inspection of a portion of these supports is required by the Technical Specifications; however, there is no formal inspection program for all components of the supports during the life of the facility. Major inspections potentially involving disassembly can be conducted on an as-needed basis.

During normal operation the loads and stresses for piping, component connections, and other remaining component supports are not sufficient to cause the failure of the reactor coolant system piping, should there be a complete failure of the reactor coolant pump supports. The maximum stresses that can be expected in the reactor coolant piping as a result of failure of the reactor coolant pump supports during normal operation are summarized in Table 15.6-3. These loads are within the allowable nozzle loads for both the steam generator nozzle and the reactor pressure vessel nozzles. The allowable stresses for the reactor coolant pipe material (A376 Tp 316) are also summarized below. While several of these values are above yield at 650°F, they are all less than 50% of the material's ultimate strength at that temperature. The reactor coolant pipe material has an S_n (code-allowable for normal operation) of 16 ksi at 650°F, and the faulted allowable stress would be 1.85 S_n or 28.8 ksi. All of the loads summarized below are within this

faulted allowable, with the exception of the pressurizer inlet. However, thermal stresses have been conservatively included, and their deletion brings the stress levels well within the allowables.

During any postulated accident condition, i.e. seismic and/or pipe breaks, a concurrent complete failure of the reactor coolant pump support would result in unacceptable consequences throughout the reactor coolant loop piping, in terms of loads and stresses. If these supports were to fail during operation, it would be detected, as there is vibration monitoring instrumentation on both the shaft and the frame of the reactor coolant pumps. The amount of vibration is indicated in the control room, and any excessive vibration would cause an annunciator alarm to sound in the control room. To trigger the annunciator alarms, vibration greater than 3 mils on the frame and greater than 15 mils on the pump shaft must occur.

15.6.2.2.4 Pressurizer Support

The pressurizer vessel is mounted in a rigid support ring girder suspended by three hanger columns from above. Antisway brackets are welded to the shell of the pressurizer to accommodate shear blocks on the ring; the ring girder is laterally supported by a reinforcement plate attached to embedments in the concrete structure. In addition, lateral support for dynamic loads is provided near the vessel's center of gravity by four gapped restraints[1] at lugs on the pressurizer which transmit the loads into baseplates on the concrete floor. The lateral gapped restraints and hanger shear blocks and reinforcement plate are able to take all incident loads while allowing the pressurizer vessel to expand radially and vertically. The pressurizer support assembly and material listing are shown on Figure 15.6-4.

15.6.2.2.5 Evaluation

The dynamic analysis program accounts for dynamic amplification of the support forces. These forces are then combined with the design-basis seismic loadings by SRSS summation to ensure that the supports are conservatively designed to withstand the condition of a pipe break occurring as a result of an earthquake. Rigid quality assurance criteria during fabrication ensure conformance with the conservative design.

15.6.3 Containment Internal Structure

The reactor containment internal structure is a reinforced concrete structure that furnishes:

1. Supports and restraints for all internal equipment and piping including the polar crane and jib crane.
2. Missile shielding for the containment steel liner and main steam lines against internally generated missiles.
3. Biological shielding for station operators inside the containment structure under all phases of reactor operation.

1. The four lateral gapped restraints were installed by Design Changes 85-04-1 and 85-05-2 to replace the four snubbers in the original support configuration.

The structure is designed to withstand the design-basis earthquake together with the simultaneous loss-of-coolant accident (LOCA)[1], without loss of function. Clearance is maintained between all internal structures and the steel liner of the reactor containment shell to permit differential earthquake motion. The steam generator cubicles and the pressurizer cubicle are designed to withstand an internal differential pressure load of 35 psi resulting from the postulated double-ended primary coolant pipe break. The primary shield is designed to withstand an internal pressure of 100 psig resulting from a hypothetical reactor coolant pipe break within the primary shield.

The differential pressure rise within the cubicles is controlled by open and shielded vent spaces in each cubicle, which permit rapid pressure equalization within the containment structure.

This transient pressure condition has been calculated by Stone & Webster's CUPAT Program, using input from the LOCTIC Program.

Temperature differentials between cubicles are considered coincident with the pressure differentials. The short duration of the transient accident relative to the low thermal conductivity of the concrete is such that no significant temperature gradient occurs across the walls. Also, the transient accident is not considered to add to the differential cubicle wall loadings.

Structural concrete design conforms to the requirements of ACI 318, Part IV-B, Ultimate Strength design. Maximum stresses are limited to 90% of the minimum yield point in bending, or 85% of the minimum yield point in diagonal tension, bond, and anchorage.

Special large-size reinforcing steel bars No. 14 and No. 18 are controlled chemistry steel of 50,000 psi yield point, otherwise conforming to the requirements of ASTM A408. All other reinforcing steel is steel of 40,000 psi yield point conforming to ASTM A-15 and ASTM A305.

A stainless-steel-lined fuel transfer canal and reactor refueling cavity is incorporated in the concrete structure above the reactor vessel. A 0.25-inch-thick stainless-steel plate is used to prevent leakage of water from these areas into the containment structure.

Portions of the biological shield walls in the steam generator cubicles are composed of removable precast, reinforced-concrete sections. The wall sections are designed for nondestructive removal to assist future servicing of the steam generators.

Structural steel framing is used as bracing along the top, corners, and ends of the removable shield wall sections. The bracing components conform to ASTM A-36 specifications for structural steel. The precast concrete wall sections have an ultimate compressive strength of

1. As discussed in Section 15.6.2, "leak-before-break" analyses have demonstrated that the probability of a rupture of the primary reactor coolant loop piping is extremely low, and it is no longer necessary to consider the dynamic effects of such a break. However, the requirements for design of containment and compartments under pressures associated with a postulated primary reactor coolant loop LOCA remain unchanged.

3000 psi at 28 days, with steel reinforcement conforming to ASTM specifications for A615, grade 40.

15.6 REFERENCES

1. Westinghouse Topical Report WCAP 9558, Revision 2, *Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack,* May 1981.
2. Westinghouse Topical Report WCAP 9787, *Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation,* May 1981.
3. Letter from Vepco to NRC, Subject: Request for Partial Exemption from General Design Criterion 4, dated November 5, 1985 (Serial No. 85-136).
4. Letter from Vepco to NRC, Subject: Request for Partial Exemption from General Design Criterion 4 - Supplement, dated December 3, 1985 (Serial No. 85-136A).
5. Letter from Vepco to NRC, Subject: Partial Exemption from General Design Criterion 4 - Request for Additional Information, dated December 27, 1985 (Serial No. 85-136B).
6. Letter from Vepco to NRC, Subject: Partial Exemption from General Design Criterion 4 - Request for Additional Information, dated January 14, 1986 (Serial No. 85-136C).
7. *Amendment to General Design Criterion 4 (GDC-4), 10 CFR Part 50, Appendix A,* published in Federal Register 51 FR 12502, effective May 12, 1986.
8. Letter from Vepco to NRC, Subject: Proposed License Amendment - GDC 4, dated April 30, 1986 (Serial No. 86-245).
9. Letter from NRC to Vepco transmitting Surry Unit 1 and 2 License Amendments No. 108 and related safety evaluations, dated June 16, 1986.
10. Letter from Vepco to NRC, Subject: Generic Letter 87-11, dated September 12, 1988 (Serial No. 88-371).
11. Manual of Steel Construction, 7th Edition, American Institute of Steel Construction.

Table 15.6-1
STEAM GENERATOR SUPPORT MATERIALS

Material Specifications	Product Form	Supplemental Requirements	Toughness Tests	Mill Test Reports	NDE
A-352grLC3 mod.	casting	yes	yes	yes	yes
Vascomax CVM 350 & 300	forging	yes	yes	yes	yes
A-106grB	14 in. pipe	none	none	yes	yes

Table 15.6-2
REACTOR COOLANT PUMP SUPPORT MATERIALS

Material Specifications	Product Form	Supplemental Requirements	Toughness Tests	Mill Test Reports	NDE
A-106grB	14 in., 6 in. pipe	none	none	yes	yes
A-105 GR 11	forging	none	none	yes	yes
AlSl-4340	forging, plate	none	none	yes	no
Vascomax CVM 350 & 300	forgings	yes	yes	yes	yes
A-285grC	plate	none	none	yes	no
A-193grB7	bar	none	none	yes	no

Table 15.6-3
SUMMARY OF STRESS FOR FAILURE OF REACTOR COOLANT PUMP SUPPORT DURING NORMAL OPERATION

Location	Loop	Stress (psi)
Steam Generator Outlet	A	20,291
	B	17,388
	C	20,743
Reactor Vessel Inlet	A	16,943
	B	18,337
	C	21,060
Crossover Leg	A	21,940
	B	13,292
	C	20,196
Pressurizer Inlet	C	32,728

Material Properties (A376 Tp 316)

°F	S_{yield}	S_{ult}
100	30 ksi	75 ksi
600	18.8 ksi	71.8 ksi
650	18.5 ksi	71.8 ksi

Figure 15.6-1
REACTOR NEUTRON SHIELD TANK ASSEMBLY

Figure 15.6-2
STEAM GENERATOR SUPPORT ASSEMBLY

Figure 15.6-3
REACTOR COOLANT PUMP SUPPORTS GENERAL ARRANGEMENT

Figure 15.6-4 (SHEET 1 OF 2)
PRESSURIZER SUPPORT

Figure 15.6-4 (SHEET 2 OF 2)
PRESSURIZER SUPPORT

15.7 MASONRY WALLS

Concrete masonry walls are used throughout the plant to provide barriers for radiation shielding, fire protection and personnel separation. Those walls utilized in the construction of Seismic Class I structures are not designed or intended to act as bearing or for transmitting building shear forces. These walls are not used as major load-bearing walls and are not included as part of the overall building shear wall system.

All of the concrete masonry walls that are in proximity to or have attachments for safety-related or equipment such that wall failure could affect a safety-related system were identified and analyzed in accordance with the requirements of IE Bulletin 80-11 (Reference 1). The reevaluation of the masonry walls was performed based upon criteria for reevaluating concrete masonry walls submitted in Reference 2, which uses as its acceptance criteria the allowable stresses specified in ACI-531-79, *Building Code Requirements for Concrete Masonry Structures*. Review of test data and the literature substantiates the use of these allowable stresses. The review also included research of acceptable damping percentages, analysis techniques, in-plane effects, arch action and local stress valves.

The reevaluation criteria considered loads from both safety- and non-safety-related attachments as well as relative interstory displacements between building elevations where applicable. All applicable loads and load combinations specified in the Surry FSAR for concrete design were included in the reevaluation. A review of the walls determined that the walls are not subjected to tornado missiles or depressurization, pipe whip or jet impingement loads. The global review of the walls included seismic inertia loads, interstory displacement loads for both in-plane and out-of-plane effects, equipment loads, and wind loads where applicable.

The local review included discontinuities such as openings and the mechanism for local load transfer into the walls. This included a review of potential local block pull out as well as possible overstress within individual blocks due to attached equipment. Multiwythe walls were also reviewed to ensure the integrity of the collar joint. Calculated shear and tension stresses across the collar joints were compared against allowable values that were conservatively chosen to account for potential small areas of voids or other discontinuities.

At the completion of the response to IE Bulletin 80-11, all identified masonry block walls were evaluated and, modified, as required, to meet the acceptance criteria. The results of this reevaluation program were transmitted to the Nuclear Regulatory Commission. The analysis results of the masonry wall reevaluation program indicated that of the walls requiring reanalysis, 79 walls were acceptable, two walls were acceptable after equipment was removed from the wall, and 31 walls were modified to meet acceptance criteria. Seven safety-related masonry walls in the fuel building were not acceptable under extreme loading conditions and were replaced with blow-off siding. An additional 217 non-safety related walls were also reviewed to ensure that they did not endanger safety-related equipment. Following the approval of responses to IE Bulletin 80-11 by the Nuclear Regulatory Commission, all subsequent modifications involving

masonry block walls are evaluated under the Nuclear Design Control Program, which continues to invoke the technical requirements of IE Bulletin 80-11 (References 3 & 4).

15.7 REFERENCES

1. IE Bulletin 80-11, *Masonry Wall Design*, Nuclear Regulatory Commission, Office of Inspection and Enforcement, May 8, 1980.
2. Letter from B.R. Sylvia, VEPCO, to James P. O'Reilly, NRC. November 3, 1980, *IE Bulletin No. 80-11 Interim Report, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 & 2*, Serial No. 878.
3. Letter from Chandu P. Patel, NRC, Office of Nuclear Reactor Regulation, *Masonry Wall Design, IE Bulletin 80-11 Surry Power Station Unit Nos. 1 and 3 (TAC Nos. 42867 and 42868)*, August 11, 1988, Serial No. 88-551.
4. Letter from Bart C. Buckley, NRC, Office of Nuclear Reactor Regulation, *Surry Units 1 and 2 - Safety Evaluation Masonry Wall Design, IE Bulletin 80-11, (TAC Nos. 42867 and 42868)*, October 2, 1989.

Appendix 15A

Seismic Design for the Nuclear Steam Supply System and Miscellaneous Components

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APPENDIX 15A SEISMIC DESIGN FOR THE NUCLEAR STEAM SUPPLY SYSTEM AND MISCELLANEOUS COMPONENTS

15A.1 GENERAL SEISMIC DESIGN CRITERIA FOR THE NUCLEAR STEAM SUPPLY SYSTEM

This Appendix is based on Appendix B to the initial FSAR.

All Class I components of the nuclear steam supply system are designed in accordance with the following criteria:

1. Primary operating stresses, when combined with the operating-basis earthquake seismic stresses resulting from a dynamic analysis using a response spectrum normalized to a maximum horizontal ground acceleration of 0.07g and a simultaneous vertical ground acceleration of two-thirds the horizontal, are maintained within the allowable stress limits in Table 15A-1.

2. Primary operating stresses when combined with the design-basis earthquake seismic stresses resulting from a dynamic analysis using a response spectrum normalized to a maximum horizontal ground acceleration of 0.15g and a simultaneous vertical ground acceleration of two-thirds of the horizontal, are limited so that the function of the component or system shall not be impaired, preventing a safe and orderly shutdown of the unit. Further, the primary operating stresses are maintained within the allowed stress limits in Table 15A-1.

   "No loss of function" requires that rotating equipment will not seize, pressure vessels will not rupture, and supports will not collapse or deform to such a degree as to cause failure of the supported equipment. In addition, systems required to be leaktight will remain leaktight, and engineered safeguards will perform intended functions.

15A.2 SEISMIC DESIGN CRITERIA FOR PIPING, VESSELS, SUPPORTS AND REACTOR VESSEL INTERNALS

Following discussions with the Staff of the Atomic Energy Commission (AEC) Division of Reactor Licensing during the Construction Permit Application Review for Diablo Canyon Unit 1, the criteria presented in WCAP-5890, Revision 1 (Reference 1), for the generation of limit curves were modified. Details of the manner in which this modification was developed are given in Section 15A.5.1.

The loading conditions employed in the design of Class I piping, vessels, and supports are enumerated and defined in Section 15A.2.1. The allowable stress limits associated with the various loading conditions are shown in Table 15A-1. Since the reactor vessel internals must also satisfy deformation limits to be able to perform their function, i.e., allow core shutdown and cooling, the vessel internals are discussed separately in Section 15A.2.3.

15A.2.1 Loading Condition Definitions

The loading condition definitions given below are taken from Section III of the ASME Boiler and Pressure Vessel Code, Summer 1968 Addenda.

15A.2.1.1 **Normal Conditions**

Any condition in the course of system start-up, operation in the design power range, and system shutdown, in the absence of upset, emergency or faulted conditions.

15A.2.1.2 **Upset Conditions**

Upset conditions are any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, transients due to loss of load or power, and any system upset not resulting in a forced outage. The estimated duration of an upset condition shall be included in the design specifications. The upset conditions include the effect of the operating-basis earthquake for which the system must remain operational or must regain its operational status.

15A.2.1.3 **Emergency Conditions**

Any deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system.

15A.2.1.4 **Faulted Conditions**

Those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent where considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities. Among the faulted conditions may be a specified design-basis earthquake for which safe shutdown is required.

15A.2.2 Piping, Vessels, and Supports

The reasons for selection of the above-mentioned loading conditions and allowable stress limits are as follows:

1. When subjected to the operating-basis earthquake, the nuclear steam supply system is designed to be capable of continued safe operation. Equipment and supports needed for this purpose are required to operate within normal design limits, as shown in Table 15A-1. The load combination corresponding to the upset loading condition is the normal load, plus the operating-basis earthquake load.

2. In the case of the design-basis earthquake, it is necessary to ensure that components required to shut the unit down and maintain it in a safe shutdown condition do not lose their capability to perform their safety function. This capability is ensured by maintaining the emergency stress limits as shown in Table 15A-1.

3. For the highly unlikely but postulated case of pipe rupture, a reactor coolant branch or other potentially governing break, the effects of the pipe rupture will not cause failure propagation to the reactor coolant piping. The load combination corresponding to the faulted loading condition is the design-basis earthquake and/or design-basis accident load.

4. For the extremely remote event of simultaneous occurrence of a design-basis earthquake and postulated pipe rupture of a reactor coolant system branch line or other potentially controlling break, the Class I piping and component are checked for no loss of function, i.e., the capability to contain fluid, allow fluid flow, and perform vital engineered safeguards functions. This is ensured by limiting the various stress combinations within the faulted condition design limits shown in Table 15A-1.

The minimum margin of safety between the design limit stress and the expected collapse condition is for the case of pure tension, and is defined as:

$$\frac{S_{ultimate} - S_{design}}{S_{design}}$$

Under more realistic operating conditions, piping and vessels will always experience some combination of tension and bending. For these combined load cases, the margin of safety is greater than that for pure tension, as shown by the limit curves contained in WCAP-5890, Rev. 1, and shown in Figures 15A-4 and 15A-5. Therefore, it is conservative to base the margin of safety on pure tension. Table 15A-2 illustrates the margin of safety between the stress limits for various load conditions and the expected failure or collapse condition for typical materials.

Plastic or limit analyses conducted within the limits of the faulted condition were performed considering plastic material behavior, including, as required, modifications of material stiffness characteristics, formation of plastic hinges and other non-linear effects, as determined in detailed structural analysis, and provided in standard stress reports.

15A.2.3 Reactor Vessel Internals

15A.2.3.1 Design Criteria for Normal Operation

The internals and core are designed for normal operating conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The response of the structure under the operating-basis earthquake is included in this category as well as operational transients (upset conditions).

The stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4, have been adopted as a guide for the design of the internals and core, with the exception of those fabrication techniques and materials not covered by the Code, such as the fuel rod cladding. Seismic stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basic principles of: (1) maintaining deflections within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) preventing fatigue failures.

15A.2.3.2 Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a pipe break[1] combined (by SRSS combination) in the most unfavorable manner with the effects associated with the design-basis earthquake.

For this condition, the criteria for acceptability are that the reactor is capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. Additional stress criteria for critical structures under normal operation, plus the design-basis earthquake and blowdown excitation, ensure that the structural integrity of the components is maintained.

15A.3 GENERAL ANALYTICAL PROCEDURE FOR SEISMIC DESIGN

The design and analysis of Class I components of the nuclear steam supply system utilizes the “response spectrum” approach.

1. As discussed in Section 15.6.2, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop piping. However, pipe ruptures of reactor coolant branchlines are still postulated.

The dynamic analysis is performed using the normal mode methods. The inertial properties of the models are characterized by the mass and mass moment of inertia of each mass point. The stiffness properties are characterized by the moment of inertia, area, shear shape factor, torsion constant, Young's modulus, and shear modulus.

Table 15.2-2 gives the damping ratios to be used in the dynamic analysis of components.

15A.3.1 Mechanical Equipment

The Westinghouse-supplied Class I mechanical components for Surry that require a seismic analysis were determined and checked for seismic adequacy by employing the following procedure:

1. The manufacturer's drawings were reviewed to classify the component.
    a. If the component fell within a category that was previously analyzed using a multi-degree-of-freedom model and shown to be relatively rigid, then a static seismic analysis was performed to check equipment seismic adequacy.
    b. If the component could not be categorized as similar to previously analyzed components, then a seismic modal analysis was performed, using multi-degree-of-freedom dynamic models.
2. Stresses and deflections were checked to ensure that they were within allowable limits and did not result in loss of function.

Typical Class I mechanical equipment of the engineered safety feature (ESF) systems supplied by Westinghouse was originally analyzed on a worstplant basis using a multi-degree-of-freedom modal analysis. The term "worstplant basis" is defined, for the particular component in question, as the most severe seismic response spectra applicable to any Westinghouse plant employing that particular piece of equipment. All contributing modes were considered. A sufficient number of masses was included in the mathematical models to ensure that coupling effects of members within the component were properly considered. The results of these analysis indicated that the models contained more masses than necessary, and that future analyses of comparable equipment could be considerably simplified by considering fewer masses, or merely performing a simple static analysis.

The method of dynamic analysis used a proprietary computer code called WESTDYN. This code uses inertia values, member sectional properties, elastic characteristics, support and restrain data characteristics, and the appropriate seismic response spectrum as input. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously. The modal participation factors are combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode. The internal forces and moments are computed for each mode from which the modal stresses are determined. The stresses are then summed using the square-root-sum-of-squares method, except for the major components in the

reactor coolant loop. As discussed in Section 15A.3.3, for the reactor coolant loop analysis, the combination of modal responses for closely spaced modes is performed using the grouping methodology in Regulatory Guide 1.92.

In analyzing equipment to resist seismic loads, the vertical seismic spectrum, equal to two-thirds of the horizontal response spectrum, is used to determine the acceleration appropriate to the vertical frequency. An idealized umbrella spectrum was used in the analyses. The floor response spectra at the Surry site are encompassed by the umbrella spectrum used in the dynamic analysis of Westinghouse-supplied equipment.

Typical Class I engineered safeguards tanks supplied by Westinghouse, e.g., for boric acid, accumulator, and boron injection, were analyzed using the method above, with the combined horizontal and vertical seismic excitation occurring simultaneously in conjunction with normal loads. Hydrodynamic analyses of these tanks have been performed using the methods described in TID 7024 (Reference 2).

Selected critical Class I ESF valves supplied by Westinghouse have been analyzed using the above method, and the results indicate that their fundamental natural frequency is sufficiently separated from the building frequency. The results indicate that the total stress, considering all modes, is far below the allowable stress limit. Motors attached to motor-operated valves have been included in the mathematical models.

The deflections and stresses obtained from the seismic analysis are added to the deflections and stresses associated with the operational mode of the mechanical equipment to verify that clearances are not exceeded and the stresses are within allowable limits. This criterion ensures that this equipment will perform the intended function under seismic conditions.

All mechanical equipment analyzed had fundamental modes in the rigid range, with the exception of loop stop valves and some vertical tanks.

The fundamental frequency of vertical tanks and the loop stop valves was greater than 9 Hz, which is outside the resonance range of the structures in which they are housed. The component supports were modified to remove the fundamental frequency of the item from the resonance range of the structure, by providing stiffer supports to increase the fundamental frequency of the component. The selection of the type of restraint used was dependent upon the component analyzed and the structure surrounding the component.

Restraints, snubbers, or other devices are not used to preclude resonance of the electrical and control systems equipment for seismic loading. Protection system equipment that is typical of the design has been subject to tests under simulated seismic accelerations to demonstrate the ability to perform and complete its function. These data are contained in WCAP-7397-L (Reference 3).

The seismic loads for the design and analysis of Class I mechanical components, including pumps, valves, heat exchangers, and tanks within Stone & Webster's scope of responsibility, were based on the ground response spectra (GRS) shown in Figures 2.5-5 and 2.5-6 or amplified floor response spectra (ARS), depending on their location. Seismic loads were developed for the operating-basis earthquake (OBE) and the design-basis earthquake (DBE). The spectra used in the evaluation of Class I mechanical components were based on damping values consistent with those indicated in Table 15.2-2. Where applicable, seismic loads were combined with the results from other load cases such as thermal and dead load. Constraints such as snubbers or other appropriate devices are utilized wherever necessary to meet design requirements.

All Class I mechanical components are designed to withstand the operating-basis earthquake and to function through the design-basis earthquake to safe shutdown. Vendors supplying the components are required by specification to design the components to function, as outlined above, under the seismic loadings. The vendor is required to validate component integrity under the specified seismic conditions.

15A.3.2 Earthquake Experience-Based Method Developed for Unresolved Safety Issue (USI) A-46 for Seismic Verification of Equipment

In response to U.S. Nuclear Regulatory Commission Generic Letter 87-02 on USI A-46, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors,* a Generic Implementation Procedure (GIP) was developed by the Seismic Qualification Utilities Group (SQUG). The criteria and methodology in Revision 3 of the GIP (Reference 40), as modified and supplemented by the NRC Supplemental Safety Evaluation Reports (SSERs) 2 and 3 (References 44 & 45) may be used, with certain additional considerations, as an alternative to other licensing basis methods for seismic design and verification of existing, modified, new and replacement equipment classified as safety-related, NSQ or seismic category 1. Considerations that are additional to the GIP pertain to the following issues:

- Use of GIP Method A for estimating seismic demand.
- Additional criteria applicable for the design and analysis of new flat bottom vertical tanks.
- Applicability of Part II, Section 5 of the GIP for conduit and cable tray raceways.
- Use of criteria associated with damping values, static coefficient and expansion anchor safety factors for equipment anchorage evaluations conforming to the current, conservative, licensing basis commitments.
- Documentation of the results of the Screening Verification and Walkdown in Section 4.6 of the GIP may be limited to the use of walkdown checklists. It is not necessary to complete the Screening Verification Data Sheets (SVDS).

- It is not necessary to identify "essential relays" and perform functionality screening as defined in Section 6 of the GIP. Relays designated as Class 1E are evaluated by comparing seismic capacity to seismic demand.
- The GIP method is generally applicable only for equipment located in mild environment. However, with case-by-case justification, it may be used for equipment in harsh environment.

Guidance for the use of the GIP for the seismic design and verification of mechanical and electrical equipment, including a discussion of the above exceptions, is provided in an engineering procedure (Reference 62).

15A.3.3 Reactor Coolant Loops and Supports

The original configuration of the reactor coolant system equipment supports included ten large-bore hydraulic snubbers per loop to carry the loads from postulated pipe ruptures of the reactor coolant system, main steam line, and feedwater line. Subsequent fracture mechanics analyses, submitted to the NRC on behalf of the Westinghouse Owners Group, demonstrated that the probability of rupture of the primary coolant loop is extremely small, and that the presence of a pipe crack could be detected by leakage well before the crack grew to a critical size which would cause rupture. NRC Generic Letter 84-04 (Reference 4) provided the NRC staff safety evaluation of these "leak-before-break" analyses, concluding that, provided certain specific conditions are met, the dynamic effects of a postulated pipe break in the reactor coolant system primary loop need not be considered as a design basis for the reviewed plants, including Surry Units 1 and 2. Subsequently, Generic Letter 87-11 (Reference 5) eliminated the need to postulate Arbitrary Intermediate Breaks and allowed removal of the associated pipe whip restraints and jet impingement shields. Because the stresses in the main steam lines inside containment are well below the stress criteria for required mandatory intermediate breaks, the only breaks which need be postulated are terminal end breaks which do not apply lateral loads to the steam generator. Based on the relief provided by these two relaxations of criteria, the reactor coolant system supports have been modified to eliminate eight of the ten large-bore snubbers per loop of the reactor coolant system. These efforts are discussed in Section 15.6.2; additional information is contained in References 6-15.

The reactor coolant loop system was re-evaluated with the snubbers eliminated to assure that the conditions of pressure, deadweight, thermal, seismic, and remaining pipe rupture effects, would not result in unacceptable stress levels or factors of safety. Two essentially independent analyses of a representative single primary loop were performed by Westinghouse Electric Corporation and Stone & Webster Engineering Corporation, in accordance with the original division of design responsibilities. Westinghouse performed deadweight, pressure, thermal and seismic analyses using a simplified model as the run of record to obtain piping stresses. Stone & Webster performed analyses using a model incorporating a detailed representation of the support

members, principally to obtain component support loads under normal and accident loadings. Both analytical and models were revisions to existing models and incorporated changes due to the steam generator replacements as well as the snubber elimination efforts.

These analyses incorporated the loads from deadweight, internal pressure, thermal expansion, seismic events (OBE and DBE), and the dynamic effects of pipe ruptures of other systems (controlling breaks, for example), in the main steam line, pressurizer surge line, main feedwater line, etc.) No other hydraulic transient loading was considered as significant.

For seismic analysis, the soil structure interaction amplified response spectra for 0.5 percent critical equipment damping (OBE) and 1 percent equipment damping (DBE) were used with appropriate “bump” factors as discussed in Section 15A.3.5.3. These damping values are lower than those in Regulatory Guide 1.61, *Damping Values for Seismic Design of Nuclear Power Plants* and in ASME Code Case N-411, *Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping.* The vertical and horizontal earthquake responses were combined for piping analysis as described in Section 15A.3.3. For component support analysis, the responses to the three directions of earthquake loading were combined by SRSS. The combination of modal responses for closely spaced modes was performed using the grouping methodology in Regulatory Guide 1.92.

The Westinghouse analysis used the WESTDYN code and a simplified representation of the component supports as stiffness matrices. The component support stiffness matrices were supplied by Stone & Webster. The WESTDYN computer code has been utilized on numerous Westinghouse plants, and was reviewed and found acceptable by the NRC for the Surry units in 1974. A detailed description of the WESTDYN method of analysis is given below.

The code uses as input system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate Surry seismic response spectrum of 0.5% critical damping. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously.

With these input data, the overall stiffness matrix [K] of the three-dimensional piping system is generated (including translational and rotational stiffness). Restraints are deleted, and the stiffness matrix is inverted to give the flexibility matrix [F] of the system.

$$[F] = [K]^{-1}$$

A product matrix is formed by the multiplication of the flexibility and mass matrices. This product matrix forms the dynamic matrix [D] from which the modal matrix is computed.

$$[D] = [F]\,[M]$$

The modal spectral matrices are generated using a modified Jacobi method.

$$(\omega^2\,[M] - [K])\,X = O$$

From this information, the modal participation factor is combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode.

The Stone & Webster analysis used the STARDYNE computer code and a model incorporating a detailed representation of the support members. STARDYNE is a public domain computer program and is recognized as a Category 1 computer program suitable for nuclear safety-related work. The program used is maintained under Stone & Webster's Quality Control procedures. The following modules of STARDYNE, Version 3, Level H, were used:

- STAR (Static and Modal Extraction Analysis)
- DYNRE4 (Seismic Response Spectrum Analysis)
- DYNRE6 (Time History Transient Analysis)—used only for evaluating pipe rupture loadings

Dynamic analyses were performed of the controlling pipe ruptures in the pressurizer surge line, main steam line, and main feedwater lines. The originally-postulated terminal and intermediate breaks were reviewed by Stone & Webster to determine those breaks which would cause the most severe loadings on the revised support configuration with snubbers removed. Time-history forcing functions were applied to the detailed model representing these potentially limiting breaks, to obtain maximum member loads with the revised support configuration. These loads were combined by SRSS with the seismic DBE loads and then summed with deadweight and pressure loads.

The results of the two independent analyses with revised support configuration established that the frequencies of most vibrational modes are virtually unchanged by the snubber eliminations. Comparison of the interface loads calculated by the two models was performed to ensure that the results of the two models were consistent; the significant interface loads were found to be within 15%, which is considered to be good agreement. The analyses demonstrated that the piping components and supports are stressed within acceptable limits, and adequate safety margins exist in a seismic event. The maximum level of stress (percentage) compared to the Code allowable at the highest stress point in each leg of the reactor coolant loop for thermal, deadweight, and seismic conditions are given in Table 15A-5.

In addition, the maximum resultant bending moment in the primary coolant loop piping under combined deadweight, pressure, thermal, and design basis seismic loadings is 28,860 inch-kips. This value is less than the enveloped value in the Westinghouse topical report, WCAP-9558, Revision 2 (Reference 6), and also less than 42,000 inch-kips which was identified in NRC Generic Letter 84-04 as the maximum allowable moment for the Westinghouse Owner's Group plants for justifying that pipe rupture need not be postulated in the primary reactor coolant loop piping.

For combined normal operating and seismic loads,

Stone & Webster evaluated the calculated maximum loadings on the supports of the modified support configuration with eight of the large-bore snubbers eliminated. The factors of safety (allowable load/combined load) for the combined deadweight, pressure, thermal, and design basis seismic loads are given in Table 15A-6. Similar factors of safety under the combined deadweight, pressure, thermal, and SRSS combination of design basis seismic loads plus maximum pipe rupture loads, are given in Table 15A-7. These tables demonstrate that adequate factors of safety exist under all loading conditions.

The results of these analyses confirm that the large-bore snubber eliminations do not compromise the bases for the previous leak-before-break analysis, namely:

1. The loading on the primary loop piping is still enveloped by the generic analyses submitted on behalf of the A-2 Owners Group and accepted by the NRC staff in Generic Letter 84-04, and specifically for Surry by NRC letter dated June 16, 1986; and

2. The reactor coolant system equipment, piping, and supports continue to have acceptable margins of safety under licenced loading conditions other than the now-eliminated ruptures of the primary loop piping and Arbitrary Intermediate Break of the main steam lines.

The inertial forces and moments are computed for each mode from which the modal stresses are determined. The stresses are then summed using the square-root-sum-of-squares method.

The maximum stresses in the reactor coolant loop piping imposed by the normal loads plus loads associated with the design-basis earthquake are below the allowable stress limit. The stress levels in the reactor coolant loop piping are provided in Table 15A-7 (References 9 and 11).

15A.3.4 Anchor Bolts

The majority of anchor bolts originally installed at the Surry Power Station were shell-type Phillips self-drilling anchors. A minimum safety factor of four was used. Cyclic loads and the effect of baseplate flexibility were not specifically considered; however, supports, baseplates, and anchor bolts were designed to withstand the maximum force exerted by seismic and thermal conditions.

In response to IE Bulletin 79-02 (Reference 16), all pipe support baseplates were analyzed considering baseplate flexibility, and modifications were made when baseplates and/or anchor bolts were found inadequate. Wedge-type Hilti bolts were installed in accordance with manufacturer's requirements based on onsite testing conducted by Hilti, Inc.

A finite element analysis was performed using the ANSYS computer program as provided by an owner's group organized by Teledyne Engineering Services. A description of this program was submitted to the NRC as Technical Report TR-3501-1, Revision 1 (Reference 17). In some cases where the support plate could not be modeled in the computer program, hand calculations were performed, allowing sufficient margin for baseplate flexibility and prying action.

The factors of safety of four for wedge-type anchors and five for shelltype anchors were used to determine the anchor bolt allowable loads for the reanalysis. All baseplates were reanalyzed to ensure that these factors of safety were met. Where the factors of safety were not met by analysis, modifications were provided to ensure the appropriate factor of safety. The original design for anchor bolts at Surry was based on a factor of safety of four for all anchor bolts, based on a design concrete strength of 3000 psi. In conjunction with IE Bulletin 79-02, a concrete inspection program was performed to demonstrate a concrete strength of 4000 psi, which would provide the factor of safety of five required by the Bulletin. Thirty-two Windsor probe tests were performed at various locations throughout the plant (Surry Units 1 and 2) to provide data for the evaluation. The results of this program show a 95% confidence level of at least 4000 psi concrete. Therefore, the analysis was based on 4000 psi concrete with the factors of safety of four and five required by Bulletin 79-02.

No special design requirements for the anchor bolts to withstand cyclic loads were applied. Testing performed for the owner's group, the results of which are presented in Technical Report TR-3501-1 indicates that cyclic loading on the anchors does not result in a general reduction of the ultimate capacity of the anchor. Bolts for shell-type anchors (Phillips Red-Head self-drilling anchors) were tightened snugly, but were not preloaded. Wedge anchors (Hilti bolts) were preloaded to the design allowable load.

To ensure that the design requirements have been met for the installed anchor bolts, an inspection and testing program was conducted. Under this program, one anchor bolt per accessible base plate was inspected and tension tested to at least the anchor bolt design load. Anchor bolt installations which were suspect based on the visual inspection were tension tested to at least the anchor bolt design allowable load (20% of the manufacturer's ultimate) and evaluated for a factor of safety of five by tension testing to five times the design load, or determining the anchor capacity based on the results of the visual inspection. When the anchor was found to be inadequate as a result of the evaluation, or of slippage greater than 1/16 inch under the tension test, the baseplate was reanalyzed with that bolt missing. The remaining bolts on the baseplate were inspected and tested for adequacy under the higher redistributed load when the reanalysis was acceptable, or for the original design load when the loads could not be redistributed. Inspection and test results showed that 97% of the baseplates were acceptable. All anchor bolts that were inadequate or damaged were replaced to ensure adequacy of the anchorage system.

In order to evaluate operability of each Seismic Category I piping system, the anchor bolt inspection and test results were recorded on a system basis. The system designations shown in Table 15.2-1 were used in conjunction with a QA Category I piping line table to determine

systems for Bulletin 79-02 purposes. Of the 14 systems for which anchor bolt inspections were performed, 12 of the systems had acceptance percentages greater than 95%. The acceptance percentages for the other two systems, the reactor coolant system and the residual heat removal system, were 94.7% and 92.1%, respectively. All systems with baseplates inaccessible for bolt inspection and testing had remote visual inspections performed to ensure that all anchor bolts were present and no gross deficiencies existed. For the two systems with less than 95% acceptance, the inaccessible baseplates were further evaluated to ensure high design factors of safety greater than those required by the Bulletin. Review of the baseplates where anchor bolts were found to be inadequate did not indicate any common characteristic (i.e., floor plates, wall plates, or ceiling plates) which would necessitate further inspection and testing of the inaccessible baseplates with a particular characteristic.

Piping systems 2 in. in diameter or less were originally designed by a chart analysis method. To ensure adequacy of the baseplates and anchor bolts in justifying operability of the small-bore piping, a sampling program was initiated. Five 2-inch lines were selected as representative of the small bore piping. Three safety injection lines and two chemical volume and control lines, which have a total of 22 supports with 43 baseplates, were analyzed in this effort. Baseplate analysis efforts show anchor bolt factors of safety ranging from 5.2 to 638, with the majority of anchor bolts having design factors of safety above 60.

Seventy-three anchor bolts on 12 of the small-bore baseplates were inspected and tension tested. Sixty-eight of these anchor bolts (93%) were accepted. The baseplates for the five rejected anchors were reanalyzed with the discrepant bolts missing and all were found acceptable and within the allowable limits. All anchor bolts which were inadequate or damaged were replaced to ensure adequacy of the anchorage system.

The small-bore piping baseplates and anchor bolts were designed and installed by the same architect-engineer and contractor that performed the work on the large-bore piping. Therefore, based on the above results, which are consistent with the large-bore anchor bolt program, a sufficient degree of conservatism exists in the baseplates and anchor bolts of the small-bore piping to justify acceptance of this piping.

All Seismic Category I pipe supports on masonry walls were resupported without attachment to the masonry walls.

IE Bulletin 79-02 inspection details were provided to the NRC by References 18, 19, and 20.

15A.3.5 Piping Systems

The Stone & Webster PSTRESS/SHOCK 2 computer code was used in the seismic analyses of certain systems at Surry Units 1 and 2. This code summed earthquake loadings algebraically, which is unacceptable for reasons set forth by the NRC in IE Bulletin 79-14 (Reference 21) and in a March 13, 1979 Order to Show Cause (Reference 22). As a result, Vepco reanalyzed

safety-related systems originally analyzed by SHOCK 2, modified those systems as necessary, depending on the results of the reanalyses, and provided support for the acceptability of the analysis methods used on the remaining Seismic Class I systems.

Portions of the following systems were identified as having been analyzed with SHOCK 2:

- Pressurizer spray and relief.
- Low-head safety injection.
- High-head safety injection.
- Containment and recirculation spray.
- Residual heat removal.
- Component cooling water.
- Service water.
- Main steam.
- High-pressure steam.
- Feedwater.
- Auxiliary feedwater.
- Containment vacuum.
- Fire protection (Unit 1 only).
- Diesel muffler exhaust (Unit 1 only).

Vepco has reanalyzed all pipe stress problems originally analyzed by SHOCK 2. All supports were reanalyzed and modifications completed as necessary.

Reanalysis and safety evaluation details are given in References 23 through 26.

15A.3.5.1 **Reanalysis Methods and Results**

As the original analysis used an algebraic intramodal summation technique, the safety-related piping system supports and attached equipment were reanalyzed with acceptable methods. The reevaluation included a dynamic computer analysis using NUPIPE programs, which incorporated a lumped mass response spectra modal analysis technique.

The floor response spectra used in the reanalysis included the original amplified response spectra specified in this appendix. In some cases, piping was reanalyzed utilizing ARS that were developed using SSI techniques. The peaks in the amplified floor response spectra were broadened by ±15% in accordance with Regulatory Guide 1.122 to account for variation in material properties and approximations in modeling.

The piping systems were modeled as three-dimensional lumped mass systems which included considerations of eccentric masses at valves and appropriate flexibility and stress intensification factors. The dynamic analysis procedures meet the criteria specified in this appendix, and are acceptable. The resultant stresses and loads from the reanalysis were used to evaluate piping, supports, nozzles, and penetrations.

Based on NRC review of the computer codes used for reanalysis, independent check analyses, and a review of modeling methods used by the Licensee, the NRC found the procedures and methods used in reanalyzing these problems acceptable.

The reanalysis included problems involving the reactor coolant system boundary and the supports associated with those problems. Since the reactor coolant system boundary is inside containment and all of the supports have been modified as necessary, there is no potential for a loss-of-coolant accident in the event of a design-basis earthquake.

At the request of the NRC, its consultant, EG&G, performed audit pipe stress calculations of five Surry 1 problems using the NUPIPE computer code. The results of the EG&G audit compared favorably with Vepco’s results.

The piping support designs for affected system piping were inspected by Vepco to verify the location, orientation, support clearances, and support type. Any deviations were incorporated into piping reanalyses. These piping systems were also verified by the NRC Office of Inspection and Enforcement.

The pipe supports were reevaluated in cases where the original support design loading was exceeded as a result of piping reanalysis. In cases where the original support capacity was exceeded, the support reevaluation included the consideration of baseplate flexibility and a verification of actual field construction of the support. Where concrete expansion anchor bolts were used, their capacities, without compromising the originally committed safety margin, were also included in the reevaluation.

The pipe break criteria of this appendix were reviewed in connection with the possible effect of changes of the high-stress point resulting from the reanalyses. Results of the evaluation of the effect the reanalysis has on the pipe break criteria show that no new whip restraints are required. Therefore, the reanalysis has not changed the pipe break protection.

The design and analysis of the supports and attached equipment are in accordance with the criteria specified in this appendix. The piping systems and supports were designed to the allowable limits of ANSI B31.1 for the gross properties, and to the limits of ANSI B31.7 Appendix F, for local stress considerations as per the criteria of this appendix. A reanalysis of the pressurizer surge line to account for the effect of thermal stratification and striping was performed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III 1986 with addenda through 1987 incorporating high cycle fatigue as required by NRC Bulletin 88-11, dated December 20, 1988.

15A.3.5.2 **Verification of Analysis Methods**

The following computer codes and analysis methods have been identified as the current basis for the facility piping design:

1. NUPIPE/S&W
2. NUPIPE
3. Static analysis methods
4. PIPESTRESS

15A.3.5.2.1 NUPIPE/S&W

Stone & Webster has submitted documentation on the NUPIPE/S&W code to the NRC. This code calculates intramodal and intermodal responses according to the provision in Regulatory Guide 1.92 (Reference 27). A review of the code listing by the NRC staff has confirmed this statement. The option used by Vepco specifies an intramodal combination consisting of the addition of the absolute value of the responses due to the vertical earthquake component and the root-mean-square combination of the responses due to the two horizontal earthquake components. Additional documentation has also been submitted by the originators of this code (Quadrex), providing detailed information on the methods of modal combination.

Vepco solved three NRC benchmark piping problems and the solutions showed acceptable agreement with the benchmark solutions. In addition, a confirmatory problem (No. 323A) was provided to Brookhaven National Laboratory for confirmatory solution. A comparison of the solutions demonstrated good agreement (within about 10%).

15A.3.5.2.2 NUPIPE

Ebasco Services, Inc., has submitted documentation to the NRC on the NUPIPE computer code, which was used in the piping reanalysis of Unit 2. This code is considered acceptable for analyses for both units.

This code has previously been reviewed and has been found to satisfy the requirements of Regulatory Guide 1.92. Ebasco solved three of the NRC benchmark piping problems, and its solutions were found to agree closely with the benchmark solutions. They also provided a confirmatory problem (2508A), which was solved by the Brookhaven National Laboratory. Comparison of the solutions showed good agreement.

15A.3.5.2.3 Static Analysis Methods

Static analysis methods, which were used in cases not subjected to computerized seismic analysis, are based on simple beam formulations, wherein seismic stress levels are controlled through use of pre-established seismic spans. These simple beam formulations were utilized to calculate maximum allowable spans based upon an assumed acceleration factor of 1.5 times the peak acceleration obtained from the response spectra. In calculating the maximum span lengths, it was conservatively assumed that a longitudinal pressure stress of 4000 psi and a maximum deadweight stress of 1500 psi were present in the pipe. This combined value of 5500 psi was subtracted from the allowable stress (1.8 S_h for pressure and deadweight and seismic) to obtain a seismic allowable stress.

Calculating maximum spans by this procedure results in maximum allowable spans greater than the deadweight spans recommended in ANSI B31.1. Thus, dead weight governs and provides a greater number of supports resulting in closely spaced restraints. To minimize effects of concentrated weights, restraints were placed as required at valves and other concentrated masses.

For Surry, piping 6 inches in diameter and smaller was generally analyzed using the simplified static method, with the option of utilizing more rigorous methods available to the analyst. Piping 2 inches and below was shown on the piping drawings diagrammatically (i.e., without detailed dimensions). The stress engineers located supports during the installation process working at the site with erection isometric sketches.

The stress analysis was performed by assuming many simple supported straight beams, the spans of which are governed by deadload spacing requirements of ANSI B31.1. The piping fundamental frequencies associated with these maximum allowable spans (9.7 to 13.6 cycles per sec) are not in resonance with the building in which they are located (2 to 8 cycles per sec). The method of equivalent static analysis outlined in this procedure was compared with the NRC's Standard Review Plan 3.7.2 (Reference 28) and found to be acceptable.

15A.3.5.2.4 PIPESTRESS

The Unit 2 RVLIS piping and Head Vent piping were reconfigured to accommodate the head assembly upgrade package. Reanalysis of the piping was performed using PIPESTRESS (Reference 68).

The PIPESTRESS program is a finite element computer program which performs linear elastic analysis of piping systems using the stiffness method of finite element analysis; the displacements of the joints of a given structure are considered basic unknowns. The dynamic analysis by the modal synthesis method utilizes known maximum accelerations produced in a single degree of freedom model of a certain frequency. The principal program assumptions are as follows:

- It is a linearly elastic structure.
- Simultaneous displacement of all supports is described by a single time-dependent function.
- Lumped mass model satisfactorily replaces the continuous structure.
- Modal synthesis is applicable.
- Rotational inertia of the masses has negligible effect.

The results obtained from the pipe stress program PIPESTRESS have been compared with the following:

- ASME Benchmark problem results, Pressure Vessel and Piping 1972 computer programs verification, American Society of Mechanical Engineers.
- Longhand calculations - PIPESTRESS is compatible with NRC Regulatory Guide 1.92. A synthesis of closely spaced modes is provided based on equation 4 of Regulatory Guide 1.92.
- Benchmark confirmatory piping analysis problems were reviewed by the NRC and Brookhaven National Laboratory.

The PIPESTRESS program is used to determine stresses and loads in the piping systems due to restrained thermal expansion, deadweight, seismic inertia and anchor movements, externally applied loads such as jet-loads, and transient forcing functions such as created by fast relief valve opening and closing, fast check valve closure after pipe breaks in main feedwater line, fast valve closure in main steam line, etc. PIPESTRESS analyzes piping systems in accordance with ANSI and ASME codes.

15A.3.5.3 **Soil Structure Interaction**

Soil structure interaction amplified response spectra (SSI-ARS) were used in reanalyzing the piping systems for those cases where the original amplified response spectra did not give satisfactory results. Based upon review of Vepco's information submitted by References 29 and 30, the NRC informed Vepco by Reference 31 and 32 that SSI-ARS was acceptable.

The amplified floor response spectra (ARS) for three levels in the containment, base mat, operating floor and spring line were computed using the multi-layered elastic half-space method and the finite element methods. The results of these analyses were compared for frequency and acceleration of the floor response spectra. The elastic half-space method gave acceleration values that were larger than the finite element method for the operating floor and the spring line. The finite element method gave accelerations slightly higher than the elastic half-space method for the containment base mat. Since no piping systems are located at, and would not use, the base mat spectra for analysis, it was concluded the elastic half-space method would be used for the reevaluation because that would be conservative. The time history used for this comparison was the original design time history used in the original design of the plant, along with the original damping values.

The same floor response spectra were generated for the Regulatory Guide 1.60 (Reference 33) requirements anchored at 0.15g, along with the Regulatory Guide 1.61 (Reference 34) damping values for comparison with the original earthquake input requirements. The time history and the damping values are considered as a consistent set of design parameters. The comparison of the original FSAR design requirements and the Regulatory Guide 1.60 and 1.61 set of values shows that the responses are very consistent, and that the original design requirements are adequate.

The ground-response spectra at the base of the reactor containment structure were calculated and plotted using SHAKE. The response spectra were calculated for three soil profiles, represented by the average low-strain shear modulus, G_{max}, calculated from seismic cross-hole surveys, G_{max} plus 50%, and G_{max} minus 50%. The spectra for each soil profile are plotted on Figures 15A-1, 15A-2, and 15A-3, respectively. Also plotted on these figures is the envelope for 0.5% damping, as presented on Figure 2.5-6.

A study of the effects of the variation of the soil properties was undertaken. The response spectra for the three locations in the containment building were computed for five variations of the soil properties. Variation one considered the computed strain dependent properties using the best estimate of the in-situ properties as input to computer code SHAKE; variation two used the in-situ properties plus 50% as input to the computer code SHAKE; variation three used the in-situ properties minus 50% as input to the computer code SHAKE; variation four considered the first iteration value of the computer code SHAKE using the in-situ properties as input; and variation five used the measured values (low strain) of the soil properties. This study indicated that the response of the structure to the variations in the soil properties is essentially limited to the amplitude of the floor response spectra. It was determined that an increase of the values of the response spectra already used in piping stress calculations by a factor of 1.50 would be acceptable. This increase in the acceleration value for the floor response spectra results in a conservative reanalysis.

To further verify that this increase (1.5) is conservative, the NRC staff conducted an independent study of the variation of soil properties used in the dynamic analyses. First the staff confirmed the adequacy of the average soil properties selected by Vepco and then considered parametric studies of these properties. The results of this effort indicated that a variation of ±25% for the input shear modulus (G_{max}) would accommodate uncertainties in the in-situ soil properties. The results of this variation appear to bound the possible range in soil properties based on staff experience with other site studies. Therefore, Vepco's studies for ±50% and the increase (1.5 factor) in the response spectra are conservative.

Because the soil shear moduli used in the generation of amplified floor response spectra depend upon the level of strain induced by earthquake motion, the amplified floor response spectra are not in direct proportion to the maximum ground acceleration. Therefore, an investigation of the effects of earthquakes smaller than the design-basis earthquake was also undertaken. For the purpose of this study, amplified floor response spectra were computed for various average strain compatible shear moduli, each due to a peak horizontal ground acceleration ranging from 0.15 to 0.05g. Vepco has provided the resulting family of amplified floor response spectra at the operating floor, which show the design-basis earthquake spectrum to envelope the other spectra due to smaller earthquakes. This demonstrated that the effects of design-basis earthquake are not exceeded by those of smaller earthquakes.

The computer codes used in the reanalysis for the soil structure interaction were:

1. SHAKE
2. PLAXLY
3. REFUND
4. KINACT
5. FRIDAY

The computer code SHAKE is a public domain program and was used to compute only the strain-dependent properties of the supporting soil under the structures. Because this code was only used to compute soil properties, no further verification was necessary.

The computer code PLAXLY is a proprietary code and was qualified by comparison to the existing public domain computer code FLUSH. Amplified response spectra for the containment operating floor computed by both codes were compared.

The computer code REFUND computes the frequency dependent compliance functions for a multi-layered elastic half-space. This code is a proprietary code and was qualified by comparing the results of a sample problem with the results published in the literature.

The computer code KINACT is a proprietary code and is used to compute the translation and rotation time history at the base of the structure from the design time history applied at the free ground surface. This code was qualified by comparing the results of a sample problem to the results of the computer code PLAXLY.

The computer code FRIDAY uses the results of REFUND and KINACT to compute the floor response spectra for each mass point in the mathematical model of the structure. The code is a proprietary program and was qualified by comparing the results of a sample problem with the results of the public domain program STARDYNE.

Additional soil-structure interaction analysis was performed for the Service Building, Safeguards Building, emergency diesel generator rooms, and Containment Spray Pump House to develop in-structure response spectra (ISRS) at certain spectral damping values that were not originally developed for these buildings. A discussion of the methods and criteria used in developing these ISRS was provided to the NRC in Reference 42, as part of the resolution to Generic Letter 87-02 (Reference 39) and Unresolved Safety Issue A-46. These methods and criteria were found adequate and acceptable in the NRC's Safety Evaluation, as indicated in Reference 43. This analysis utilized synthetic time histories as free-field motions. Three synthetic time-histories were developed such that their spectra at 5% damping closely matched the corresponding UFSAR horizontal and vertical ground spectra for Surry, consistent with Figures 2.5-5 and 2.5-6 for OBE and DBE respectively. A very close fit was reached to ensure that no lack of energy occurs at any frequency of interest. The three time histories were statistically independent. Soil and structures were modeled in detail, as discussed below.

The low strain soil properties were obtained from Section 2.4. Industry Standard Code SHAKE was used to perform dynamic analyses of the soil profile to generate the strain compatible soil properties. To account for uncertainties in the soil properties, three low strain soil properties were considered for each seismic input, in accordance with the recommendations in the Standard Review Plan (SRP - NUREG 0800). They are: best estimate, lower bound (shear modulus equal to half the best estimate shear modulus), and upper bound (shear modulus equal to twice the best estimate shear modulus). Thus, three strain compatible soil profiles were developed consistent with soil strains induced by the Housner input.

The structures were modeled as three-dimensional "stick" models with lumped mass and with six degrees of freedom at each node. Eccentricities were explicitly considered at each modeled elevation to account for the effects of torsion and rocking. The damping values were in accordance with Table 15.2-2. For the cases where different portions of the structures were assigned different damping, composite modal damping values were generated based on the strain energy weighted approach.

For each structure, the proper foundation embedment was considered and frequency dependent impedance and scattering functions were calculated for each strain compatible soil case. In addition, the deconvolved time-histories at the foundation levels were verified according

to the recommendation of the SRP such that their response spectra (envelop of three soil cases) are not less than sixty percent of the surface spectra. The building models were used together with the proper impedance and scattering functions and for each strain compatible soil case, the three orthogonal time-histories were applied individually. The In Structure Response Spectra (ISRS) in this effort were developed at each elevation for each structure, and peak broadened +15% and -15% to account for uncertainties and variabilities in the structural frequencies, in accordance with Regulatory Guide 1.122.

Additional information on soil-structure interaction can be found in Chapter 2.

15A.4 MOVEMENT OF REACTOR COOLANT SYSTEM COMPONENTS

The criterion for movement of the reactor pressure vessel, under the worst combination of loads, i.e., normal plus the design-basis earthquake plus reactor coolant pipe rupture loads[1], is that the movement of the reactor vessel will not exceed the clearance between a reactor coolant pipe and the surrounding concrete.

The relative motions between reactor coolant system components will be controlled by the structures that are used to support the reactor pressure vessel, steam generators, pressurizer and reactor coolant pumps.

Piping runs that are external to the plant or between buildings and that would affect the health and safety of the public are dynamically stress analyzed. Necessary earthquake stops, constraints, or anchors are judiciously located to withstand motion, but allow for thermal movement.

The dynamic seismic stresses are calculated using the appropriate operational-basis earthquake and design-basis earthquake response spectrum. The design criteria for the analysis of Class I piping systems are in accordance with the code requirements of ANSI B31.1. Pressure, deadload, thermal, and seismic pipe stresses are combined in accordance with the code.

The structures to which the piping is attached are also designed to withstand these loads. Included in the pipe stress analysis are horizontal and vertical differential motion caused by rotation, translation, and flexure of the respective structures assumed to be out of phase with each other, plus the relative motion from earthquake orbital displacement of the founding soil.

In most cases, piping runs are stress-analyzed by system, thus determining the effect of branch lines joining the main run or other piping.

1. As discussed in Section 15.6.2, it is no longer necessary to consider the dynamic effects of a postulated rupture of the primary reactor coolant loop piping. However, pipe ruptures of reactor coolant branch lines, the main steam lines are still postulated.

In the analysis of piping running between structures, and to different elevations within the same structure, consideration has been given to differential motion of the piping supports and anchors.

The building displacements caused by an earthquake, which include rocking and translation of the structures, as well as the relative motion from orbital displacement of the foundation soil, are evaluated at the elevation or elevations at which the piping is supported. These displacements are then applied to the supports and anchors as external movements to the piping system, and stresses are calculated.

For the analysis of piping supported by different structures, an out-of-phase condition is always assumed, and the direction of the displacements applied to the piping supports and anchors are chosen to reflect the out-of-phase condition to yield the most conservative results.

15A.5 TESTS TO DEMONSTRATE THE CONSERVATISM OF THE LIMIT CURVES

Tests performed at Westinghouse Material Testing Laboratory in Pittsburgh demonstrate the conservatism of the limit curves presented in WCAP-5890, Revision 1 (Reference 35). Carbon steel and stainless steel pipes have been tested under various combinations of axial and transverse loads to determine failure loads. Specimens about 1.5 foot long have been cut from 1.5-inch nominal diameter Schedule 160 pipes. The materials employed were SA 106B carbon steel and Type 304 stainless steel. These specimens were kept internally pressurized to 3000 psia for the entire duration of the tests. Tables 15A-3 and 15A-4 summarize the tests that have undergone evaluation and the results of this evaluation.

Standard ASTM tensile specimens have been modeled from pieces of the test pipes and stress-strain curves determined. These curves have been conservatively approximated with trapezoidal stress-strain curves as indicated in WCAP-5890, Revision 1. The limit curves for both SA 106B carbon steel and Type 304 stainless steel for the test conditions have been calculated and are reported in Figures 15A-4 and 15A-5, respectively. The experimental points, i.e., stress intensities versus axial stress as listed in Tables 15A-3 and 15A-4 are shown in Figures 15A-4 and 15A-5. Also shown in these figures are the limit curves as calculated by the use of the trapezoidal stress-strain curves up to the ultimate stress. Comparisons between the experimental points and the design limit curves show the conservatism of the latter.

15A.5.1 Westinghouse Topical Reports

WCAP-5890, Rev. 1, has been replaced by WCAP-7287 (Reference 36). The revisions affected limits for the combination of normal loads plus design-basis earthquake loads plus pipe rupture loads associated with a loss-of-coolant accident. The changes reflected agreement with the staff of the Atomic Energy Commission (AEC) Division of Reactor Licensing on the stress

limits for the above-mentioned load combinations. Details of the manner in which the revisions were developed are as follows:

1. Material data used to develop stress-strain curves.

   Typical stress-strain curves of type 304 stainless steel (Figure 15A-6), Inconel 600 (Figure 15A-7) and SA 302B low alloy steel (Figure 15A-8) at 600°F were generated from tests using graphs of applied load versus cross-head displacement as automatically plotted by the recorder of the tensile test apparatus. The scale and sensitivity of the test apparatus recorder ensured accurate measurement of the uniform strain.

   For materials other than these three, stress-strain curves were developed by conservative use of pertinent available material data (i.e., lowest values of uniform strain and initial strain hardening). Where the available data were not sufficient to develop a reliable stress-strain curve, three standard ASTM tensile tests of the material in question were performed at design temperature. These data were conservatively applied in developing a stress-strain curve as described above.

2. The ordinate (stress) of the stress-strain curves was normalized to the measured yield strength.

3. Twenty percent of uniform strain as defined on the curve developed under Item 1 was used as the allowed membrane strain.

4. The normalized stress ratio was established at 20% of uniform strain on the normalized stress-strain curves developed under item 2.

5. The value of the membrane stress limit was established.

6. The normalized stress ratio in item 4 was multiplied by the applicable code yield strength at the design temperature to get the membrane stress limit. The actual physical properties as determined from standard ASTM tensile tests on specimens from the same heats was allowed as an alternate method of determining the membrane stress limit. Sufficient documentation was provided to support the actual material properties used.

7. Limit curves for the combination of local membrane and bending stresses were developed.

   The limit curves were developed by using the analytical approach presented in WCAP-5890, Revision 1, and the stress-strain curve up to the membrane stress limit as developed under item 5. These limit curves were within the limit curves discussed with the staff of the AEC Division of Reactor Licensing during meetings on November 30 and December 1, 1967, for the same materials.

15A.5.2 Framatome Computer Programs (Unit 1 only)

This section describes computer programs that were used by Framatome ANP for the dynamic and static analysis of Class 1 equipment and components during the process of qualifying the Unit 1 replacement reactor vessel closure head to ASME Section III requirements.

These computer programs meet the requirements of the Dominion and Framatome ANP software validation programs. The validation program meets the requirements of 10 CFR 50 Appendix B, ASME NQA 1, and ANSI N45.2. The software validation compliance was verified during an onsite quality audit of the replacement closure head vendor. Audit results and objective evidence of the software validation are available in the Framatome ANP audit file. These programs provide results that are essentially the same or more conservative than the analyses of record.

15A.5.2.1 **BWSPAN**

BWSPAN (Reference 63) is designed to perform analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components and the ANSI B31.1 Power Piping Code. This code has been specifically used for evaluating the configuration of the RVLIS piping routed from the closure head up to and including “RX Vessel Vent Line to RVLIS Isolation Valve,” 1-RC-603, “Rx Vessel Vent Line to RVLIS Isolation Vent Valve,” 1-RC-36 (including associated drain valve 1-RC-186) and to a location in the run of the pipe just upstream of valve 1-RC-185.

15A.5.2.2 **BIJLAARD**

BIJLAARD (Reference 64) is designed to calculate local stresses in a cylindrical or spherical shell induced by a nozzle or support.

15A.5.2.3 **FERMETURE**

FERMETURE (Reference 65) is designed to calculate the loadings used for the closure analysis. FERMETURE calculates the stud load components for a given set of temperature and pressure values. Additionally, FERMETURE verifies the leak tightness of the vessel closure.

15A.5.2.4 **SYSTUS**

SYSTUS (Reference 66) is designed to analyze the thermal-mechanical behavior of beams and solid structures in two or three dimensions.

15A.5.2.5 **RCCM-ASME**

RCCM-ASME Program (Reference 67) is a special postprocessor of SYSTUS that allows manipulation of SYSTUS results for stress analyses in accordance with the rules defined by the ASME Code Section III including stresses linearization, usage factor calculation and thermal ratchet analysis.

15A.6 REACTOR COOLANT LOOP (RCL) PIPING REANALYSIS SUBSEQUENT TO LEAK BEFORE BREAK AND SNUBBER ELIMINATION

Table 15A-5 identifies the maximum level of stress as a percentage of the Code allowable stress for the analysis that was performed for implementation of Leak Before Break (LBB), which allowed for the removal of large snubbers connected to the primary loop piping. Reanalysis of the Reactor Coolant Loop piping and supports was performed subsequent to the implementation of LBB to support implementation of several plant changes and refinement of analytical modeling, which include:

- Measurement Uncertainty Recapture (MUR) power uprate
- Implementation of the 15 x 15 Upgrade Fuel Design
- Nuclear Safety Advisory Letter NSAL-11-2 (Reference 71) concerning the assumed stiffness of the Reactor Vessel Lower Radial Keys.

The resultant updated stresses as a percentage of the Code allowable stress and factors of safety for supports are shown in Table 15A-8 and Table 15A-9 respectively.

In the stress reanalyses performed by Westinghouse, and documented in References 69 and 70, two additional RCL branch line pipe break cases, consisting of RHR Suction and Accumulator Injection line breaks, were added, which were not analyzed previously. Addition of these postulated breaks increased the faulted stresses. However, the recalculated stresses still remain below the code allowable stress of 2.4 S_h. The recalculated stresses for analyzed loading conditions are shown in Table 15A-8. The change only affects the faulted condition evaluation. All the other stresses (Pressure, Deadweight, Thermal, Seismic OBE and DBE) are unchanged. The recalculated margins for pipe supports based upon Shaw calculation 13019801-P-0001 (Reference 71) are shown in Table 15A-9. The stated change affects only the faulted loads due to the two additional pipe break cases analyzed. There is no change in other loads, including seismic OBE and DBE loads.

15A REFERENCES

1. R. A. Wiesemann, R. E. Tome, and R. Salvatore, *Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels under Earthquake Loadings*, WCAP-5890, Revision 1.
2. TID-7024, Chapter 6.
3. *Seismic Testing of Electrical and Control System Equipment*, WCAP-7397-L.
4. NRC Generic Letter 84-04, *Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops*, dated February 1, 1984.

5. NRC Generic Letter 87-11, *Relaxation In Arbitrary Intermediate Pipe Rupture Requirements,* dated June 19, 1987.

6. Westinghouse Topical Report WCAP 9558, Revision 2, *Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack*, May 1981.

7. Westinghouse Topical Report WCAP 9787, *Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation*, in May 1981.

8. Letter from Vepco to NRC, Subject: *Request for Partial Exemption from General Design Criterion 4,* dated November 5, 1985 (Serial No. 85136).

9. Letter from Vepco to NRC, Subject: *Request for Partial Exemption from General Design Criterion 4 - Supplement,* dated December 3, 1985 (Serial No. 85-136A).

10. Letter from Vepco to NRC, Subject: *Partial Exemption from General Design Criterion 4 - Request for Additional Information,* dated December 27, 1985 (Serial No. 85-136B).

11. Letter from Vepco to NRC, Subject: *Partial Exemption from General Design Criterion 4 - Request for Additional Information,* dated January 14, 1986 (Serial No. 85-136C).

12. *Amendment to General Design Criterion 4 (GDC-4), 10 CFR Part 50, Appendix A,* published in Federal Register 51 FR 12502, effective May 12, 1986.

13. Letter from Vepco to NRC, Subject: *Proposed License Amendment - GDC 4,* dated April 30, 1986 (Serial No. 86-245).

14. Letter from NRC to Vepco transmitting Surry Unit 1 and 2 License Amendments No. 108 and related safety evaluations, dated June 16, 1986.

15. Letter from Vepco to NRC, Subject: Generic Letter 87-11, dated September 12, 1988 (Serial No. 88-371).

16. U.S. Nuclear Regulatory Commission, *Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts*, IE Bulletin 79-02, March 8, 1979 (Revision 1), August 20, 1979 (Revision 2).

17. Teledyne Engineering Services, *Summary Report. Generic Response to USNRC I&E Bulletin No. 79-02*, Base Plate/Concrete Expansion Anchor Bolts, Technical Report TR-3501-1, Revision 1.

18. Letter from Vepco to NRC, Subject: IE Bulletin 79-02, dated December 7, 1979.

19. Letter from W. C. Spencer, Vepco to J. P. O'Reilly, NRC, Subject: *Supplemental Response to IE Bulletin 79-02, Revision 2, Surry Power Station Units 1 and 2,* dated January 30, 1980.

20. Letter from W. C. Spencer, Vepco, to J. P. O'Reilly, NRC, Subject: *Final Report on IE Bulletin 79-02, Revision 2, Surry Power Station Unit 2,* dated August 4, 1980.

21. U.S. Nuclear Regulatory Commission, *Seismic Analyses for As-Built Safety-Related Piping Systems,* IE Bulletin 79-14, July 2, 1979.

22. Letter from H. R. Denton, NRC, to W. L. Proffit, Vepco, Subject: *Show Cause Orders for Surry Power Station, Units 1 and 2,* dated March 13, 1979.

23. Letter from W. C. Spencer, Vepco, to H. R. Denton, NRC, transmitting *Report on the Reanalysis of Safety-Related Piping Systems, Surry Power Station Unit 1*, dated June 5, 1979.

24. Letter from H. R. Denton, NRC, to W. L. Proffit, Vepco, Subject: *Order Lifting the Suspension of Facility Operation Required by the Order to Show Cause Dated March 13, 1979, for the Surry Power Station, Unit 1,* dated August 22, 1979.

25. Letter from W. C. Spencer, Vepco, to H. R. Denton, NRC, transmitting *Report on the Reanalysis of Safety-Related Piping Systems, Surry Power Station Unit 2,* dated February 22, 1980.

26. Letter from H. R. Denton, NRC, to W. L. Proffit, Vepco, Subject: *Order Lifting the Suspension of Facility Operation Required by the Order to Show Cause Dated March 13, 1979, for the Surry Power Station,* Unit 2, March 26, 1980.

27. U.S. Nuclear Regulatory Commission, *Combining Modul Responses and Spatial Components in Seismic Response Analysis,* Regulatory Guide 1.92, February 1976.

28. *Seismic System Analysis*, Standard Review Plan 3.7.2, November 24, 1975.

29. Letter from W. C. Spencer, Vepco, to V. Stello, NRC, Subject: *Response to NRC Information Request on Soil-Structure Interaction,* dated May 2, 1979.

30. Letter from W. C. Spencer, Vepco, to D. G. Eisenhut, NRC, Subject: *Operating-Basis Earthquake Reanalysis of Piping Systems, Surry Power Station Units 1 and 2,* dated September 13, 1979.

31. Letter from NRC to Vepco, Subject: *Soil-Structure Interaction Position,* dated March 25, 1979.

32. Letter from D. G. Eisenhut, NRC, to W. L. Proffit, Vepco, Subject: *Soil-Structure Interaction Conclusions for Surry Power Station Units 1 and 2,* dated November 15, 1979.

33. U.S. Nuclear Regulatory Commission, *Design Response Spectra for Seismic Design of Nuclear Power Plants*, Regulatory Guide 1.60, December 1973.

34. U.S. Nuclear Regulatory Commission, *Damping Values for Seismic Design of Nuclear Power Plants*, Regulatory Guide 1.61, October 1973.

35. WCAP-5890, Revision 1.

36. WCAP-7287.

37. U.S. NRC Bulletin No. 88-11: *Pressurizer Surge Line Thermal Stratification,* USNRC, December 20, 1988.

38. Virginia Power Letters Serial Nos. 89-006A dated May 3, 1989 and 89-006B dated November 13, 1989 to United States Nuclear Regulatory Commission.

39. NRC Generic Letter 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46,* February 19, 1987.

40. *Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3, Updated 05/16/97,* prepared by Seismic Qualification Utility Group (SQUG) and sent to the NRC by letter dated May 16, 1997.

41. Letter from NRC to Vepco, Subject: *Computer Program for Pipe Stress Analysis of ASME Class 2 and 3 Piping,* dated June 16, 1993. (Serial No. 93-395)

42. Letter from W. L. Stewart, Virginia Power, to U.S. Nuclear Regulatory Commission, September 18, 1992, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Response to Supplement 1 of Generic Letter 87-02 SQUG Resolution of Unresolved Safety Issue A-46,* Serial Number 92-384.

43. NRC letter to Virginia Power, Serial No. 92-763, *Safety Evaluation of NAPS Units 1 and 2 and SPS Units 1 and 2, 120 day response to Supplement No. 1 to Generic Letter 87-02,* November 20, 1992.

44. *Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on Seismic Qualification Utilities Group's Generic Implementation Procedure, Revision 2, Corrected February 14, 1992, for Implementation of GL 87-02 (USI A-46), Verification of Seismic Adequacy of Equipment in Older Operating Nuclear Plants,* U.S. Nuclear Regulatory Commission, May 22, 1992.

45. *Supplemental Safety Evaluation Report No. 3 (SSER No. 3) on the Review of Revision 3 to the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment, Updated May 16, 1997,* U.S. Nuclear Regulatory Commission, December 4, 1997.

46. Virginia Power Letter to Document Control Desk (NRC), *Surry Power Station, Units 1 and 2, Summary Report for Resolution of Unresolved Safety Issue (USI) A-46,* November 26, 1997.

47. NRC letter to Virginia Power, *Surry Power Station, Units 1 and 2 - Request for Additional Information Regarding Seismic Qualification of Mechanical and Electrical Equipment (GL 87-02),* February 23, 1998.

48. Virginia Power letter to NRC, *Surry Power Station, Units 1 and 2 - Response to Request dated February 23, 1998 for Additional Information on USI A-46 Summary Report (GL 87-02),* April 22, 1998.

49. NRC letter to Virginia Power, *Surry Power Station, Units 1 and 2 - Request for Additional Information Regarding Seismic Qualification of Mechanical and Electrical Equipment (GL 87-02),* September 11, 1998.

50. Virginia Power letter to NRC, *Surry Power Station, Units 1 and 2 - GL 87-02 - Response to Request for Additional Information dated September 11, 1998 on USI A-46 Summary Report,* January 5, 1999.

51. Virginia Power Letter to Document Control Desk (NRC), *Surry Power Station, Units 1 and 2, Completion of Outlier Resolution - USI A-46 Program, GL 87-02 - Verification of Seismic Adequacy of Mechanical and Electrical Equipment,* February 4, 2000.

52. NRC Letter to Virginia Power, Serial Number 00-398, transmitting Safety Evaluation by the Office of Nuclear Reactor Regulation Evaluation of Virginia Electric and Power Company Response to Supplement No. 1 to Generic Letter 87-02, Surry Units 1 and 2, July 25, 2000.

53. NRC (Brian Sheron) letter to SQUG (Neil Smith), *Clarification of the Staff's Position Regarding Incorporation of the GIP Method as a Revision to the Plant Licensing Basis,* June 19, 1998.

54. NRC letter to SQUG, *Review of Seismic Qualification Utility Group's Report on the Use of the Generic Implementation Procedure for New and Replacement Equipment and Parts,* June 23, 1999.

55. *Implementation Guidelines for Seismic Qualification of New and Replacement Equipment/Parts (NARE) Using the Generic Implementation Procedure (GIP), Revision 4,* Prepared for EPRI and SQUG by MPR Associates, July 2000.

56. *Guideline for the Seismic Technical Evaluation of Replacement Items (STERI) for Nuclear Power Plants*, EPRI Report NP-7484, February 1993.

57. *Generic Seismic Technical Evaluations of Replacement Items (GSTERI) for Nuclear Power Plants*, EPRI Report TR-104871, May 1995.

58. *Generic Seismic Technical Evaluations of Replacement Items for Nuclear Power Plants - Item Specific Evaluations, including Supplement 1,* EPRI Report SU-105849, September 1997.

59. *Critical Characteristics for Acceptance of Seismically Sensitive Items (CCASSI),* EPRI Report TR-112579, September 2000.

60. EPRI NP-6041-SL, Revision 1, *A Methodology for Assessment of Nuclear Plant Seismic Margin",* August 1991.

61. BNL 52361, *Seismic Design and Evaluation Guidelines for the Department of Energy High Level Waste Storage Tanks and Appurtenances,* Brookhaven National Laboratory, Upton, New York, October 1995.

62. Dominion General Engineering Nuclear Standard STD-GN-0038, *Seismic Qualification of Equipment*.

63. *BWSPAN - Pipe Stress Analysis,* Version 10.2 (2A-4 Rev. T), Users Manual, Developed by Framatome ANP.

64. *BIJLAARD - Local Shell Stress Calculation,* Version 1.0, Developed by Framatome ANP.

65. *FERMETURE - Closure Analysis,* Version AXP 2.2, Developed by Framatome ANP.

66. *SYSTUS - General Finite Element System,* Developed by Framatome ANP Module SYSNUKE Version 233 AXP 3 for Closure Head Stress and Fatigue Analysis Module NUKE Version 1.03 SGI for Adapter Tube and Vent Pipe Stress and Fatigue Analysis.

67. *RCCM-ASME - Pressure Retaining Boundary Analysis,* according to ASME Code Developed by Framatome ANP, Version AXP 2.3.12 for Closure Head Stress and Fatigue Analysis Version 2.3.13 SGI and 2.3.14 SGI for Adapter Tube and Vent Pipe Stress and Fatigue Analysis.

68. *PIPESTRESS Computer Software,* DST Computer Services S.A.

69. *Surry 1 and 2 RCL Piping Stress Analysis and Surge Stratification Analysis for RTSR*, Westinghouse Calculation No. CN-PAFM-10-36, Rev. 1, January 2013.

70. *Factor of Safety for Primary Components and Supports with Reduced Snubber Configuration, Calculations No. 14937.24-NMB-333-GA, Rev. 0, Addendum 00B, August 2013*

71. *Impact of Change in Lower Radial Key Stiffness Value, Westinghouse Nuclear Safety Advisory Letter NSAL-11-2, June 2011.*

Table 15A-1
LOADING CONDITIONS AND STRESS LIMITS

Pressure Vessels

Loading Conditions	Stress Limits	
1. Normal conditions	a. $P_m \leq S_m$	
	b. P_m(or P_L)+$P_B \leq 1.5S_m$	(Note 1)
	c. P_m(or P_L)+P_B+$Q \leq 3.0S_m$	(Note 2)
2. Upset conditions	a. $P_m \leq S_m$	
	b. P_m (or P_L)+$P_B \leq 1.5S_m$	(Note 1)
	c. P_m(or P_L)+P_B+$Q \leq 3.0S_m$	(Note 2)
3. Emergency conditions	a. $P_m \leq 1.2S_m$, or $P_m \leq S_y$, whichever is larger	
	b. P_m (or P_L)+$P_B \leq 1.5(1.2S_m)$, or P_m (or P_L)+$P_B \leq 1.5$ (S_y) whichever is larger	(Note 3) (Note 3)
4. Faulted conditions	Design limit curves of WCAP-5890, Rev. 1, as modified by Section 15A.5.1	(Note 4)

where:

P_m = primary general membrane stress intensity
P_L = primary local membrane stress intensity
P_B = primary bending stress intensity
Q = secondary stress intensity
S_m = stress intensity value from ASME B&PV Code, Section III, Nuclear Vessels
S_y = minimum specified material yield

Pressure Piping (Note 6)

Loading Conditions	Stress Limits	
1. Normal conditions	$P_m \leq S$	
2. Upset conditions	$P_m \leq 1.2S$	
3. Emergency conditions	$P_m \leq 1.8S$	
4. Faulted conditions	Design limit curves of WCAP-5890, Revision 1, as modified by Section 15A.5.1	(Note 4)

where
P_m = principal stress
S = Allowable stress from USAS B31.1, Code for Power Piping

7	Equipment Supports
1. Normal conditions	Within working limits
2. Upset conditions	Within working limits

Table 15A-1 (CONTINUED)
LOADING CONDITIONS AND STRESS LIMITS

	Equipment Supports (continued)	
3. Emergency conditions	Within material yield strength after load redistribution	(Note 5)
4. Faulted conditions	Within material yield strength after load redistribution	(Note 5)

Note 1: The limits on local membrane stress intensity ($P_L \leq 1.5Sm$) and primary membrane plus primary bending stress intensity (P_M (or P_L) + $P_B \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 2/3 of the lower bound collapse load as per paragraph N-417.6(b) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L \leq 1.5Sm$) or the primary plus secondary stress intensity ($P_L + P_B + Q \leq 3S_m$) at a specific location, the structural action may be calculated on a plastic basis and the design will be considered to be acceptable if shakedown occurs, as opposed to continuing deformation, and if the deformation which occur prior to shakedown do not exceed specified limits, as per paragraph N-417.6(a)(2) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 3: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by test that the specified loadings do not exceed 120% of 2/3 of the lower bound collapse load as per paragraph N-417.10(c) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 4: As an alternate to the design limit curves that represent a pseudo plastic instability analysis, a plastic instability analysis may be performed in some specific cases considering the actual strainhardening characteristics of the material, but with the yield strength adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11c of the ASME B&PV Code, Section III, Nuclear Vessels. These specific cases will be justified on an individual basis.

Note 5: Higher stress values can be adopted if a valid limit or plastic instability analysis of the support and supported component/system is performed.

Note 6: As required by NRC Bulletin 88-11, pressurizer surge line is re-evaluated in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1986 with addenda through 1987 incorporating high cycle fatigue.

Table 15A-2
MINIMUM MARGINS OF SAFETY

	Loading Conditions		
Material	Upset Conditions	Emergency Conditions	Faulted Conditions (Note 1)
SA302 Grade B	200%	150%	27%
Inconel 600	228%	172%	43%
316 SST	222%	169%	60%
A212 Grade B	346%	272%	55%

Note: Based upon the limit curves computed using Section 15A.5.1.

Table 15A-3
TESTS AND TEST RESULTS ON SA 106B CARBON STEEL PIPE SPECIMENS
(INTERNAL PRESSURIZATION = 3000 PSIA)

	Pseudo-elastic Axial Stress Normalized to the Yield Stress	Pseudo-elastic Bending Stress Normalized to the Yield Stress	Pseudo-elastic Stress Intensity Normalized to the Yield Stress	Strain Percent (gauge length)
Pure tension (no weld)	1.736	0.0	1.770	22.61% (12 in.)
	1.840	0.0	1.847	22.32% (12 in.)
Pure bending [a]	0.10	>2.348	2.382	>35.4X (1 in.)
Tension + bending (no weld)	>1.375	>1.030	2.440	>7.75% (6 in.)
	>1.585	>0.565	2.180	--
	>1.845	>0.266	2.145	--
Compression + bending (no weld)	>1.130	1.08	2.410	--
Pure tension (circumf. weld)	1.852	0.0	1.886	20.05% (12 in.)
Pure bending (circumf. weld)	0.10	>2.580	2.614	>30.19X (1.5 in.)
Pure bending [b] (rejected circumf. weld)	0.10	2.460	2.494	14.51% (2 in.)

a. The limit capability of the test apparatus has been reached before failure of these specimens was approached.

b. This test was performed on a welded pipe specimen that has been rejected by the inspector prior to the test for gross lack of penetration in weld. This was the only test in which the weld failed. The specimen exhibited substantial ductility prior to failure.

Table 15A-4

TESTS AND TEST RESULTS ON 304 STAINLESS STEEL SPECIMENS
(INTERNAL PRESSURIZATION = 3000 PSIA)

	Pseudo-elastic Axial Stress Normalized to the Yield Stress	Pseudo-elastic Bending Stress Normalized to the Yield Stress	Pseudo-elastic Stress Intensity Normalized to the Yield Stress	Strain Percent (gauge length)
Pure tension (no weld)	2.495	0.0	2.53	52. 12 (12 in.)
Pure bending [a]	+0.10	>2.91	>2.945	>30.0% (1 in.)
Tension + bending (no weld)	>+2.09	>0.42	>2.55	--
Pure bending (circumf. weld)	+0.10	>3 27	>3.30	>25.0% (1.5 in.)
Pure tension (circumf. weld)	+2.46	0.0	2.49	44.6% (12 in.)

a. The limit capability of the test apparatus has been reached before failure of these specimens was approached.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 15A-5

LEVEL OF STRESS AS A PERCENTAGE OF CODE ALLOWABLE STRESS

Loading	Hot Leg	Crossover Leg	Cold Leg	Code Allowable Stress
Thermal	38.6%	15.6%	7.4%	S_A
Pressure + Deadweight	68.0%	48.7%	53.3%	$1.\ 0S_h$[a]
Pressure + Deadweight + OBE	65.5%	65.0%	60. 0%	$1.\ 2S_h$[a]
Pressure + Deadweight + DBE	49.6%	51.1%	48.9%	$1.\ 8S_h$[a]

In addition, the pipe Stresses in the primary reactor coolant loop piping under combined accident loadings from pressure, deadweight, plus SRSS of design-basis earthquake and controlling pipe ruptures are less than $1.8S_h$, which is conservative both respect to the allowables per Section 15A.5 and to $2.\ 4S_h$ which is permitted by the current ASME Code.

a. S_h =15 Ksi

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 15A-6

FACTORS OF SAFETY FOR COMPONENT SUPPORTS UNDER DESIGN-BASIS SEISMIC AND NORMAL OPERATING LOADS SURRY UNITS 1 AND 2

Component	Factor of Safety[a]		
	Original Design[b]	Modified Design[c]	Final Design[d]
Steam Generator Shell	>20.0	>20.0	>20.0
Steam Generator Upper Support			
Component	19.2	15.9	13.7
Upper Guide	7.3	4.6	3.3
Snubber	14.2	14.2	6.3
Steam Generator Lower Support			
Hanger Rod	1.8	1.8	1.8
Swivel End Coupling	16.3	13.4	12.2
Steam Generator Foot			
Vertical Force	2.7	2.8	2.8
Tangential Force	16.0	12.4	12.4
Reactor Coolant Pump Foot			
Vertical Force	5.5	5.2	5.3
Tangential Force	15.4	15.8	15.9
Radial Force	15.4	15.0	12.8
Reactor Coolant Pump Support			
Upper Vertical	5.5	5.0	5.0
Upper Horizontal	5.0	5.0	5.0
Lower Vertical	3.6	3.6	3.6
Lower Diagonal	4.6	4.6	4.6

a. Factor of Safety - (Allowable Load)/(TotaL Load of Deadweight, Pressure, Thermal and DBE)

b. Original design incorporated ten large-bore snubber per loop for primary reactor coolant system pipe rupture loads.

c. Modified design implemented elimination of six large-bore snubbers based on “leak-before-break”; n the four remaining large-bore snubbers were required to carry loads of main steam line rupture.

d. Final design incorporates only two large-bore snubbers on Steam Generator, following elimination of lateral loads due to postulated AIB of Main Steam Line.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 15A-7

FACTORS OF SAFETY FOR COMPONENT SUPPORTS UNDER COMBINED ACCIDENT LOADS SURRY UNITS 1 AND 2

Component	Factor of Safety[a]		
	Original Design[c]	Modified Design[d]	Final Design[e]
Steam Generator Shell	2.5	2.4	8.3
Steam Generator Upper Support			
Component	2.8	2.8	7.5
Upper Guide	1.1	1.1	2.6
Snubber	1.3	1.4	1.9
Steam Generator Lower Support			
Hanger Rod	1.7	1.7	1.7
Swivel End Coupling	2.3	2.3	2.3
Steam Generator Foot[b]			
Vertical Force	4.0	4.2	4.3
Tangential Force	7.4	7.2	7.3
Reactor Coolant Pump Foot[b]			
Vertical Force	11.9	11.3	10.9
Tangential Force	>20.0	>20.0	>20.0
Radial Force	>20.0	>20.0	>20.0
Reactor Coolant Pump Support			
Upper Vertical	4.5	4.3	4.2
Upper Horizontal	5.3	4.8	4.6
Lower Vertical	3.8	3.5	3.5
Lower Diagonal	3.0	3.0	3.0

a. Factor of Safety - (Allowable Load)/[TotaL Load of Deadweight, Pressure, Thermal and SRSS (DBE+Pipe Rupture)].

b. Allowable loads from Westinghouse specification are higher for pipe rupture case; this results in higher normal factors of safety for some components for this case compared with factors of safety for design basis seismic loads only (Table 15A-4).

c. Original design incorporated ten large-bore snubber per loop for primary reactor coolant system pipe rupture loads.

d. Modified design implemented elimination of six large-bore snubbers based on "leak-before-break"; n the four remaining large-bore snubbers were required to carry loads of main steam line rupture.

e. Final design incorporates only two large-bore snubbers on Steam Generator, following elimination of lateral loads due to postulated AIB of Main Steam Line.

Table 15A-8
CALCULATED STRESS AS PERCENTAGE OF CODE ALLOWABLE STRESS
(REFERENCE 69)

Loading	Hot Leg	Crossover Leg	Cold Leg	Code Allowable Stress
Thermal	38.6% (unchanged)	15.6% (unchanged)	7.4% (unchanged)	S_A
Pressure + Deadweight	68.0% (unchanged)	48.7% (unchanged)	53.3% (unchanged)	1.1 S_h[a]
Pressure + Deadweight + OBE	65.5% (unchanged)	65.0% (unchanged)	60.0% (unchanged)	1.2 S_h[a]
Pressure + Deadweight + DBE	49.6% (unchanged)	51.1% (unchanged)	48.9% (unchanged)	1.8 S_h[a]
Pressure + Deadweight + $\{(DBE)^2 + (LOCA/Pipe Rupture)^2\}^{1/2}$	50.8%	53.9%	98.1%	2.4 S_h[b]

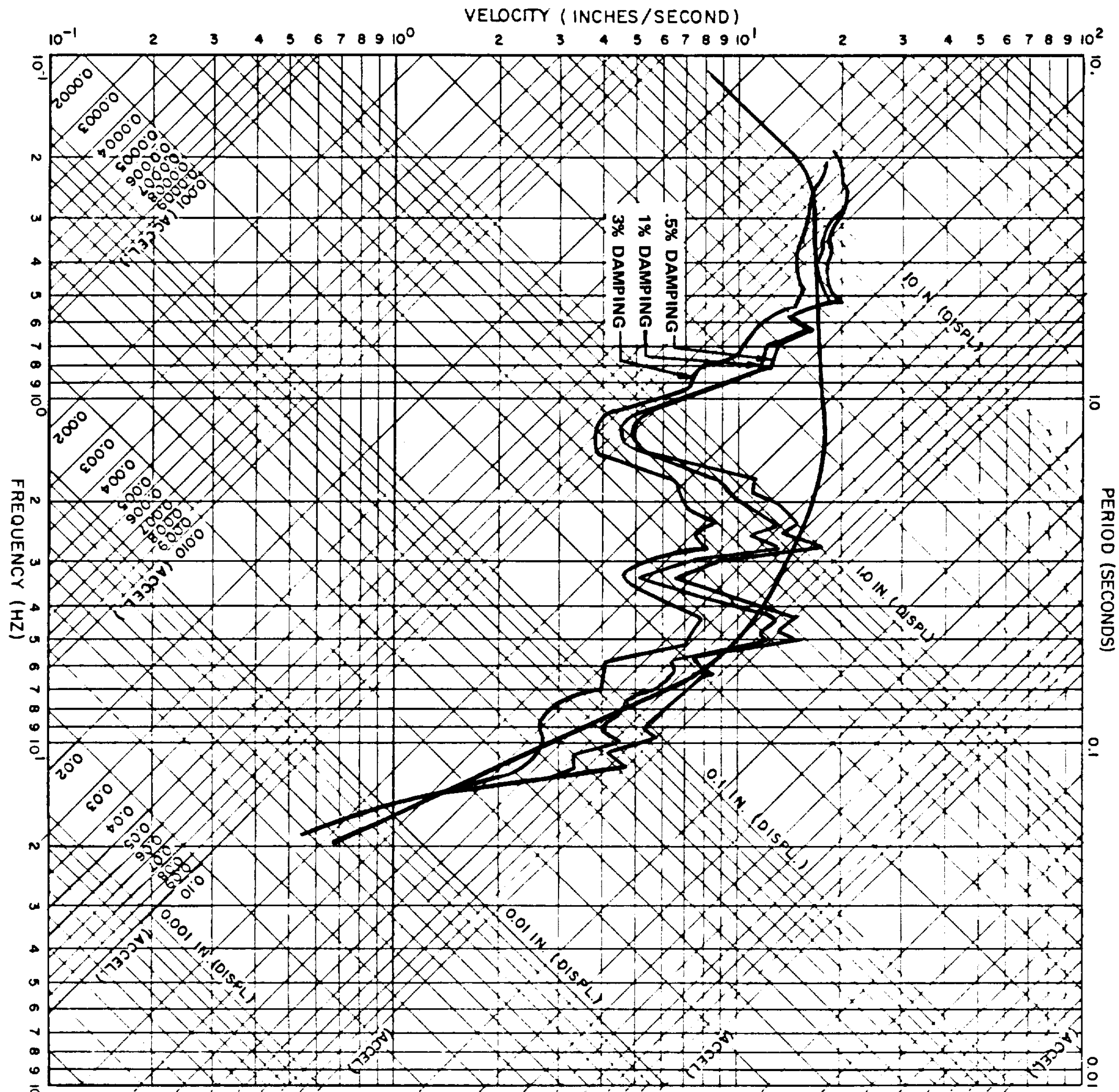
a. S_h = 15 ksi

b. The faulted case maximum calculated stress which includes maximum stress for worst case LOCA/pipe break is compared against ASME code allowable stress of 2.4 S_h. This is acceptable under current industry practice and widely used by Westinghouse in the reanalysis of RCL piping for their plants. Based upon the material test data performed by Westinghouse, documented in WCAP-5890, Rev. 1 which was later replaced by WCAP-7287 and shown summarily in Tables 15A-3 plus 15A-4 and Figures 15A-4 plus 15A-5, the design limit curves are conservative compared to test results shown.

Table 15A-9
FACTORS OF SAFETY FOR STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (REFERENCE 70)

Component	Factor of Safety
Steam Generator Upper Support:	
Component	7.5
Upper Guide	2.5
Snubber	1.5
Steam Generator Lower Support:	
Hanger Rod	1.2
Swivel End Coupling	1.7
Reactor Coolant Pump Support:	
Upper Vertical	3.1
Upper Horizontal	3.4
Lower Vertical	2.6
Lower Diagonal	2.2

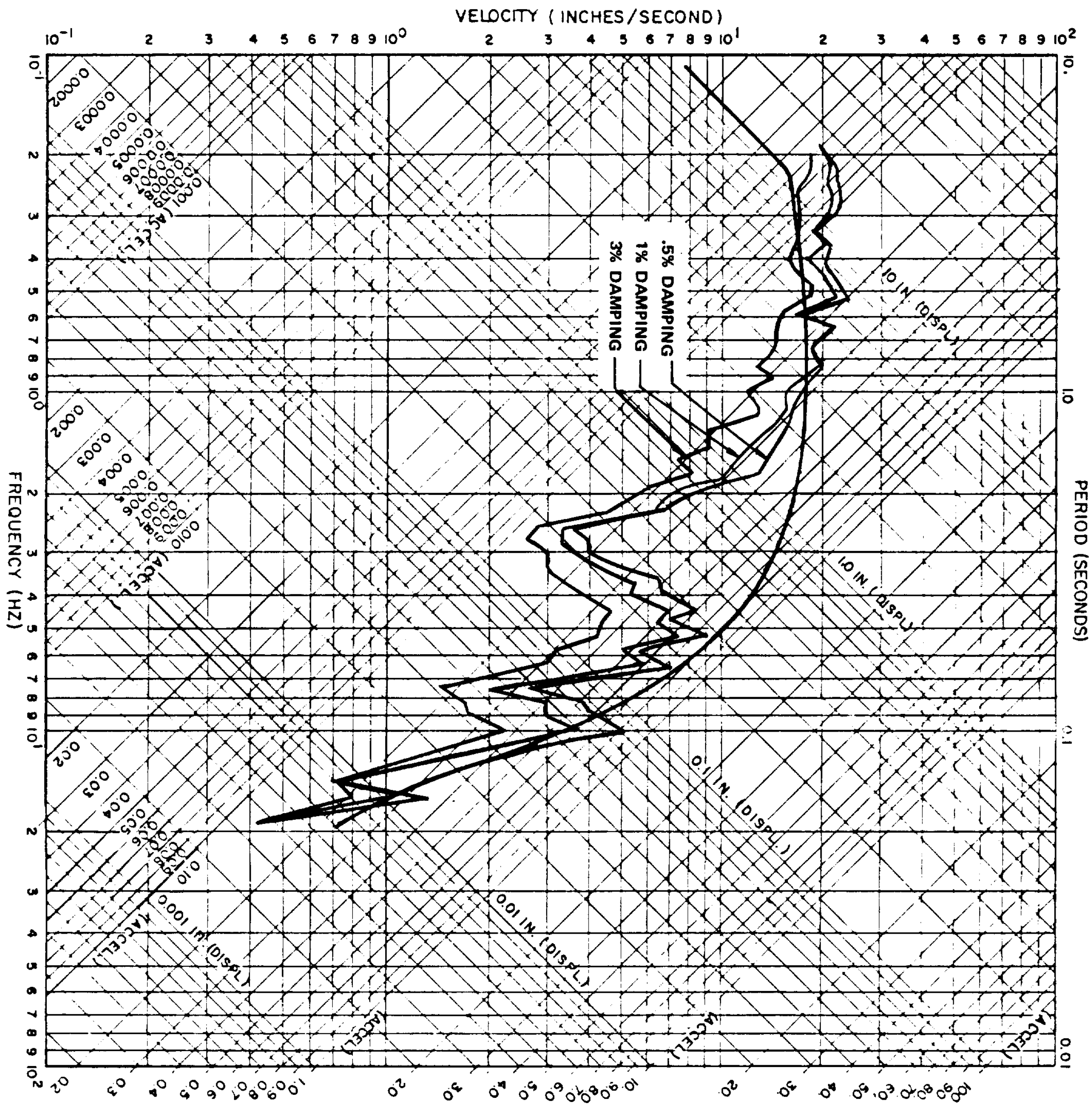
Figure 15A-1
ENVELOPE FOR 0.5% DAMPING GROUND RESPONSE SPECTRA-AVERAGE G_{max}



ARTIFICIAL EQ. (DBE) FREE FIELD	0.005
FREE FIELD DECONVOLUTE (—50% 0 MAX)	0.010
SPECTRA FOR TOP OF LAYER 4	0.030

S15A0001

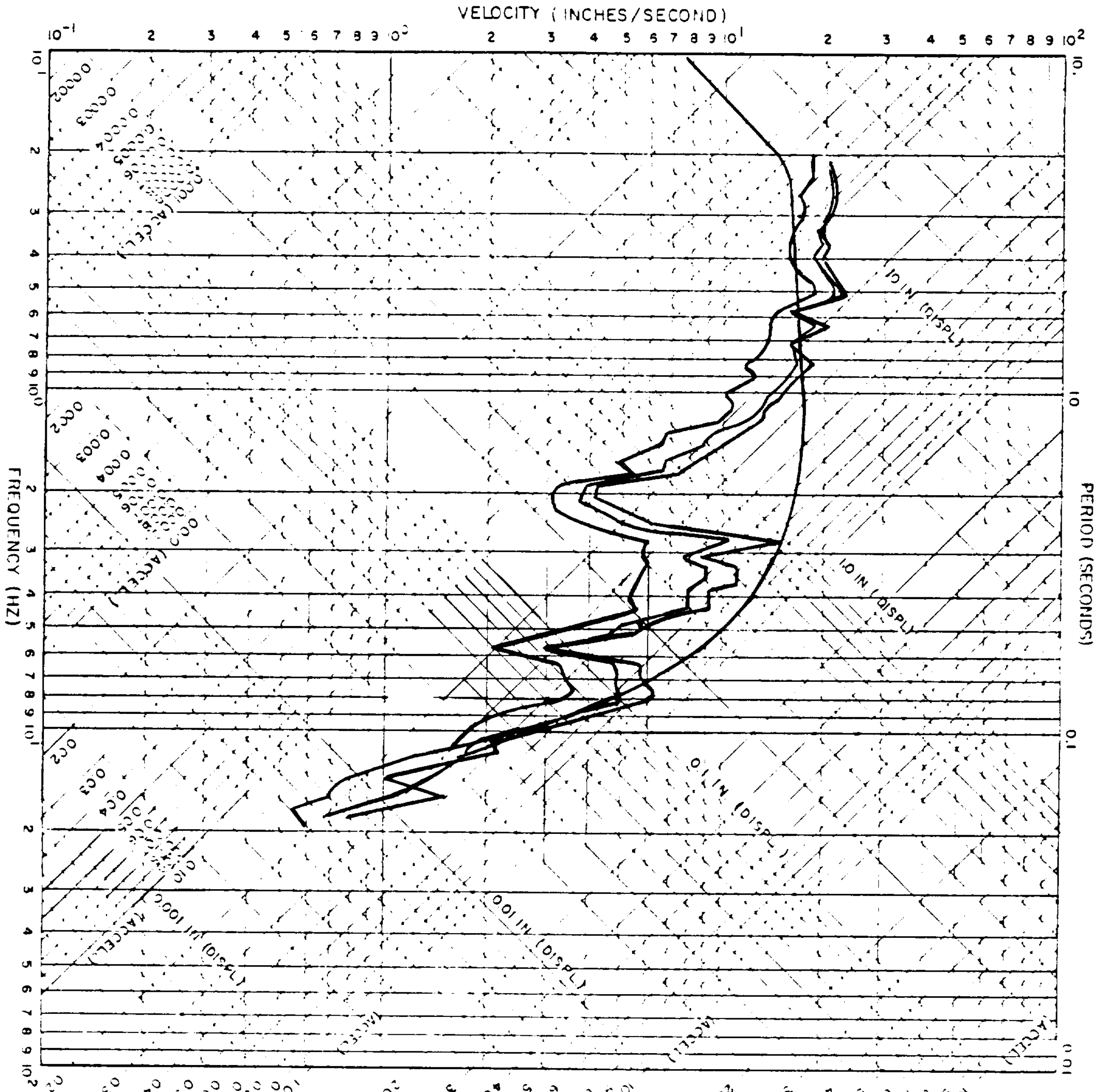
Figure 15A-2
ENVELOPE FOR 0.5% DAMPING
GROUND RESPONSE SPECTRA-AVERAGE G_{max} + 50%



ARTIFICIAL EQ. (DBE) FREE FIELD	0.005
FREE FIELD DECONVOLUTE (50% 0 MAX)	0.010
SPECTRA FOR TOP OF LAYER 4	0.030

S15A0002

Figure 15A-3
ENVELOPE FOR 0.5% DAMPING
GROUND RESPONSE SPECTRA-AVERAGE G_{max} -50%



ARTIFICIAL EQ. (DBE) FREE FIELD	0.006
FREE FIELD DECONVOLUTE (0 MAX)	0.010
SPECTRA FOR TOP OF LAYER 4	0.030

Figure 15A-4
DESIGN LIMITS COMPARED TO EXPERIMENTAL POINTS,
SA 106B CARBON STEEL

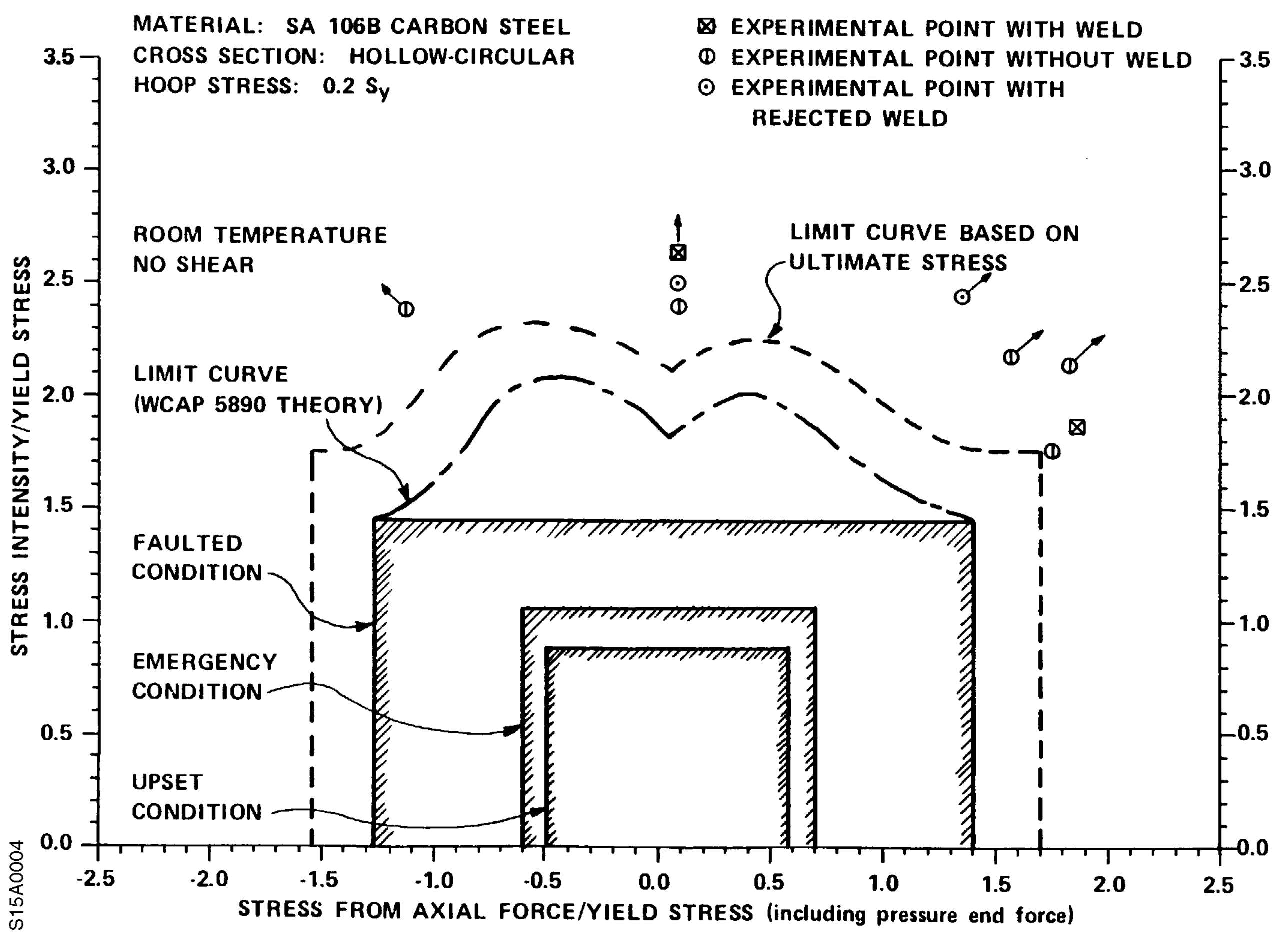


Figure 15A-5

DESIGN LIMITS COMPARED TO EXPERIMENTAL POINTS, 304 STAINLESS STEEL

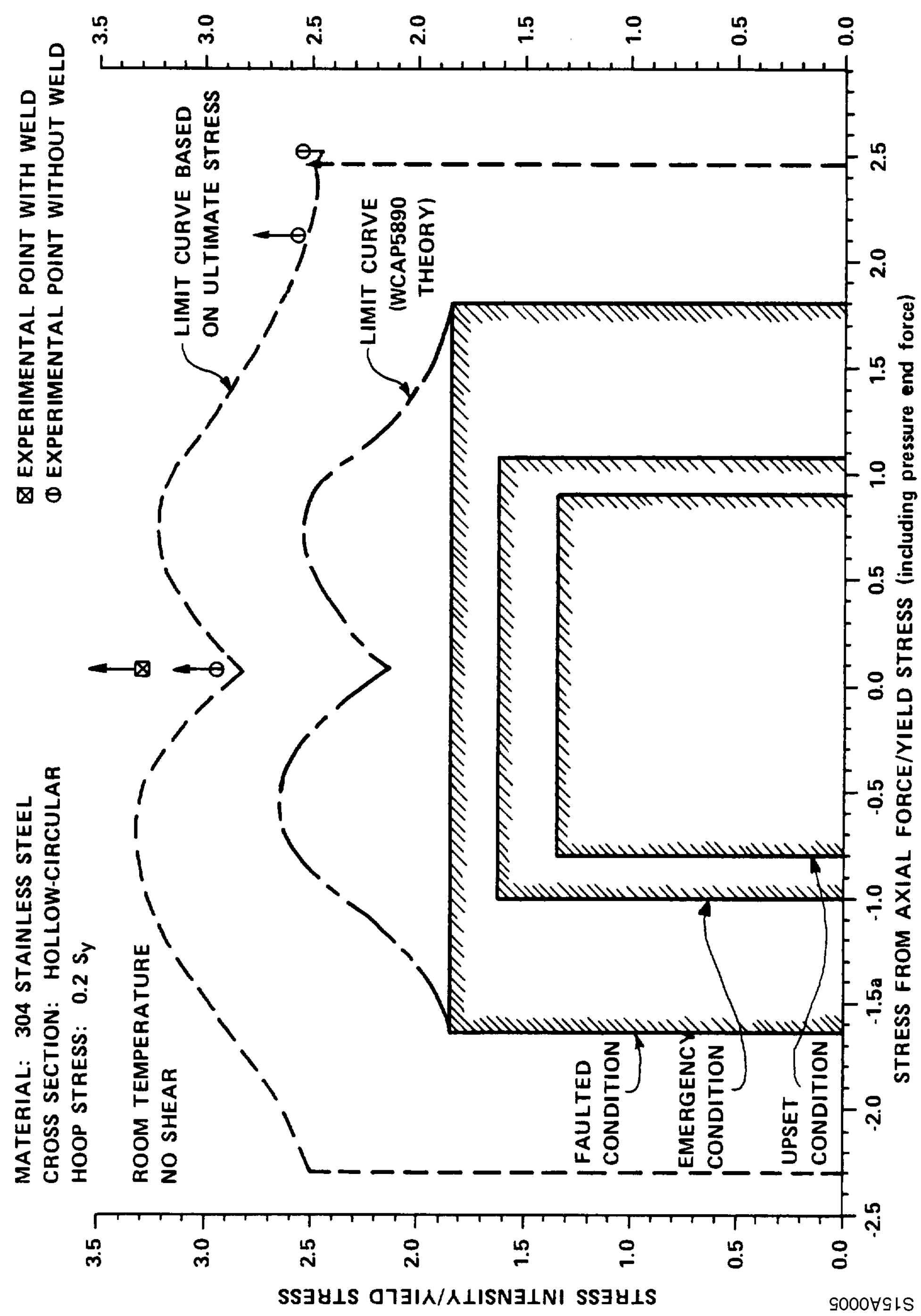


Figure 15A-6

TYPICAL STRESS-STRAIN CURVE, 304 STAINLESS STEEL

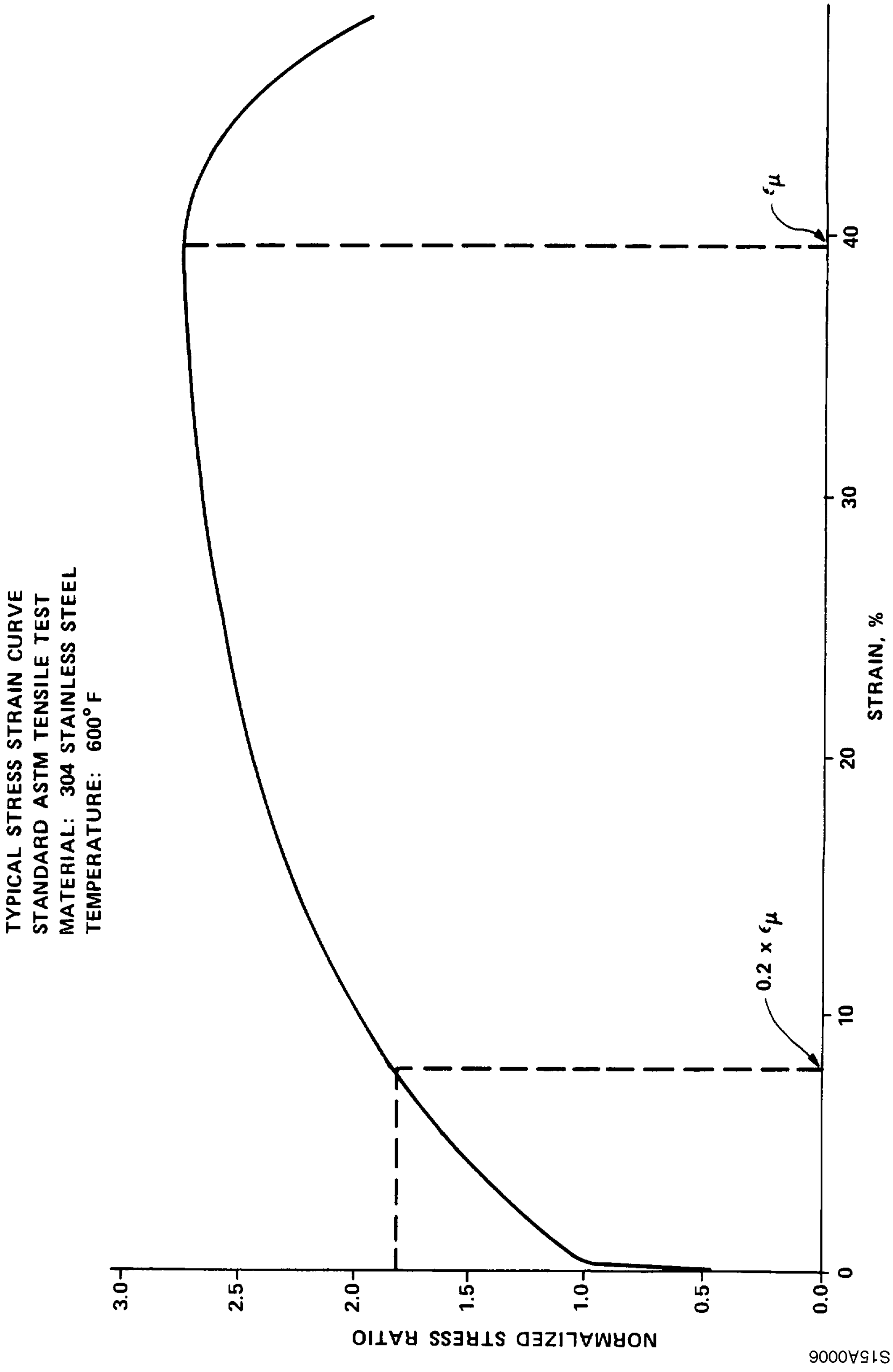


Figure 15A-7
TYPICAL STRESS-STRAIN CURVE, INCONEL 600

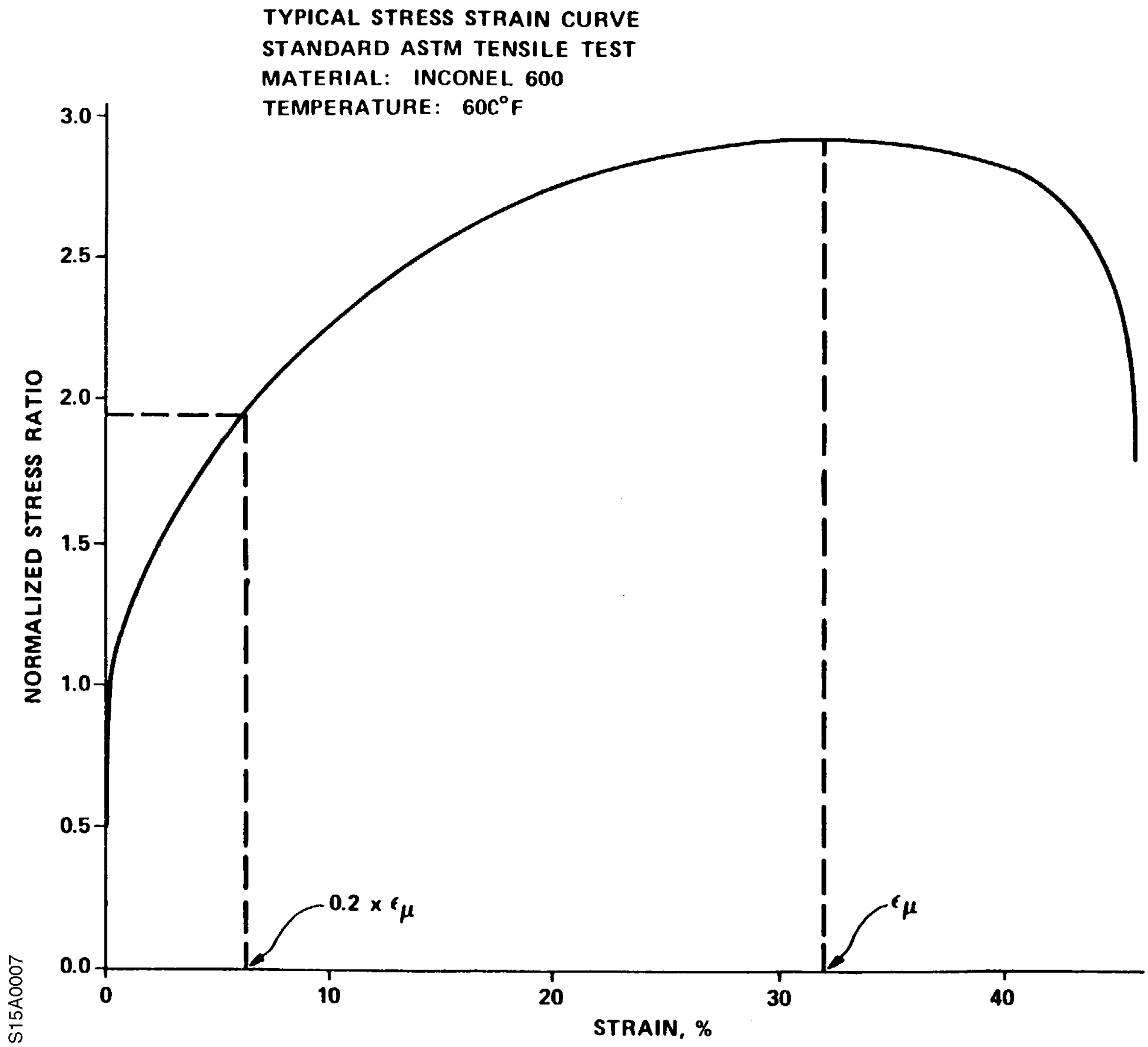
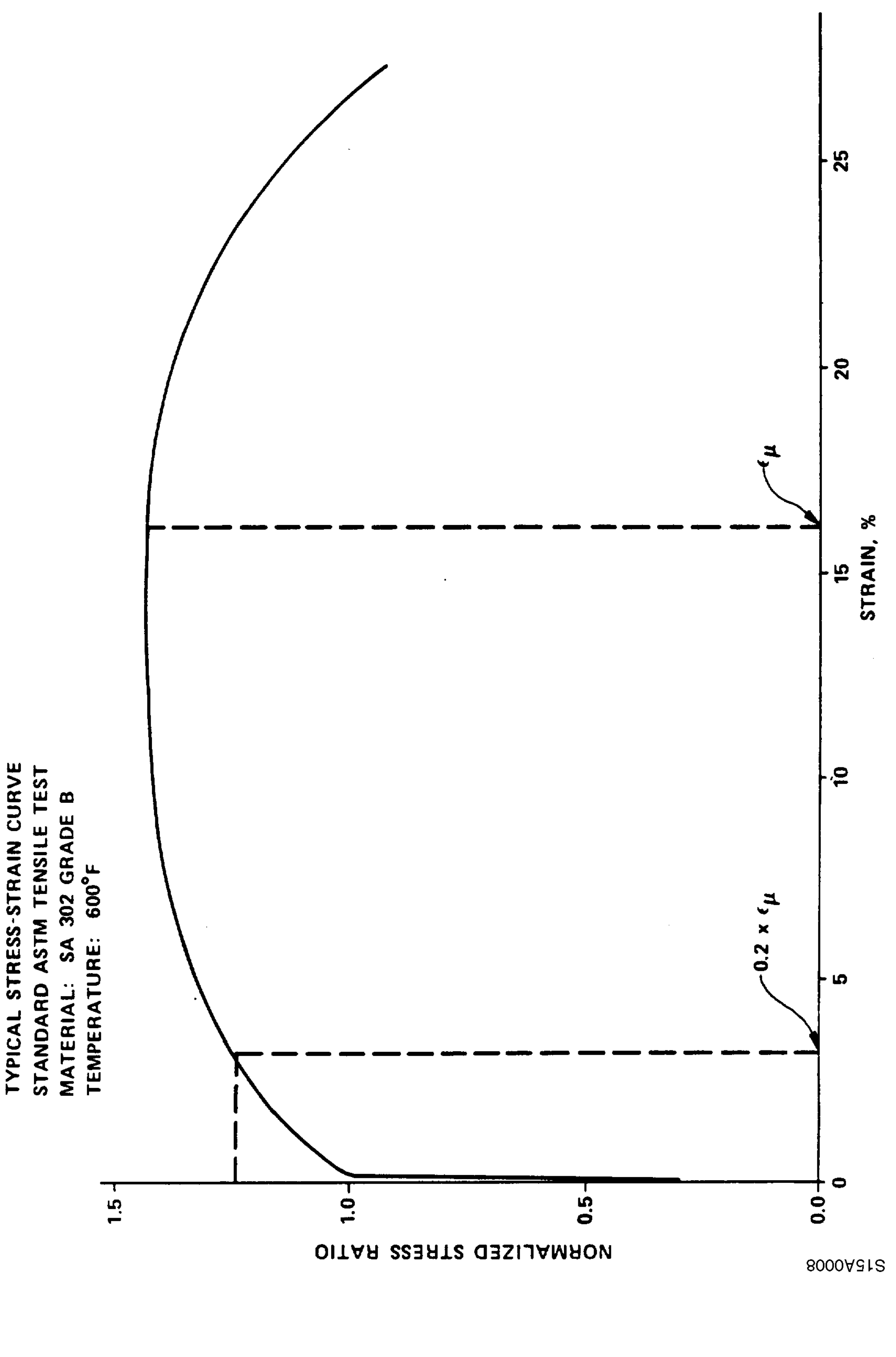


Figure 15A-8
TYPICAL STRESS-STRAIN CURVE, SA 302 GRADE B



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Chapter 16

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Chapter 16: Technical Specifications and Technical Requirements
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CHAPTER 16 TECHNICAL SPECIFICATIONS AND TECHNICAL REQUIREMENTS

16.1 TECHNICAL SPECIFICATIONS

Technical Specifications were proposed in accordance with 10 CFR 50.36 during the initial licensing of the plant. The Technical Specifications now reside in Appendix A of the Operating License for each unit.

Technical Specifications define plant variables, operating conditions, surveillance requirements, and administrative controls that are considered necessary to ensure the health and safety of the public.

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16.2 TECHNICAL REQUIREMENTS

The Technical Requirements Manual (TRM) contains requirements for plant operation and surveillance of systems formerly contained in the Technical Specifications, along with other selected items. Some requirements were removed from the Technical Specifications as part of NRC and industry efforts to simplify Technical Specifications.

The TRM is controlled by station procedure, and a 10 CFR 50.59 review is required to change the TRM. Changes to the TRM may be made without prior NRC approval, provided that the changes do not involve a license amendment as defined in 10 CFR 50.59. Changes to the TRM that are implemented without prior NRC approval are reported to the NRC in accordance with 10 CFR 50.59. Proposed changes that involve a license amendment are reviewed and approved by the NRC prior to implementation.

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Chapter 17

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CHAPTER 17 QUALITY ASSURANCE

The Quality Assurance Program is described in Topical Report DOM-QA-1, Dominion Nuclear Facility Quality Assurance Program Description (QAPD). This topical report provides the QAPD for Dominion's nuclear power stations and independent spent fuel storage installations. The Dominion QAPD conforms to applicable regulatory requirements, such as 10 CFR 50, Appendix B, and approved industry standards, including equivalent alternatives, where identified. This program applies to activities during design, construction, operation, and decommissioning as well as siting.

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Chapter 18

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Chapter 18: Programs and Activities That Manage the Effects of Aging
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Chapter 18: Programs and Activities That Manage the Effects of Aging
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Chapter 18: Programs and Activities That Manage the Effects of Aging

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CHAPTER 18 PROGRAMS AND ACTIVITIES THAT MANAGE THE EFFECTS OF AGING

The integrated plant assessment for license renewal identified new and existing aging management programs and activities necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. The period of extended operation is defined as 20 years from the end of each units original operating license expiration date. This chapter describes these programs and activities and their planned implementation.

This chapter also discusses the evaluation results for each of the plant-specific time-limited aging analyses (TLAAs) performed for license renewal. The evaluations have demonstrated that the analyses remain valid for the period of extended operation; the analyses have been projected to the end of the period of extended operation; or that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The NRC Safety Evaluation Report, NUREG-1766, (Reference 23) for the Surry renewed operation licenses, identified commitments associated with the future development and enhancement of various aging management programs and activities. These commitments are compiled and listed in Appendix D of the NUREG and are provided herein as Table 18-1.

18.1 NEW AGING MANAGEMENT ACTIVITIES

The following sections provide a description of aging management programs and activities that were not in-place when the renewed operating licenses were issued for Surry. These programs and activities were not part of the licensing basis for the original operating license period but were identified as necessary to manage aging of various station systems, structures, and components during the period of extended operation.

18.1.1 Buried Piping and Valve Inspection Activities

Prior to the period of extended operation, buried piping and valves will be inspected for the existence of aging effects (Item 1, Table 18-1). The Buried Piping and Valve Inspection Activities will include a one-time inspection of representative samples of piping and valves for different combinations of buried material and burial condition. Visual inspections are used to detect cracking of protective coatings and loss of material from protective coatings or the substrate material. Visual inspections detect gross indications of change in material properties for copper-nickel pipe. A Dominion evaluation identifies the scope of inspections for buried piping and valves.

The inspection will be completed in accordance with the schedule provided in Item 1, Table 18-1, and will include representative valves and sample lengths (i.e., several feet) of piping for each of the following combinations of material and burial conditions:

- Carbon steel, concrete encased
- Carbon steel, coated
- Carbon steel, coated, wrapped
- Carbon steel, coated, and wrapped with cathodic protection
- Stainless steel, coated, and wrapped
- 90/10 Copper-nickel, uncoated

An engineering evaluation of the results of the buried piping and valves inspections will be performed to determine future actions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.1.2 Infrequently Accessed Area Inspection Activities

The purpose of the Infrequently Accessed Area Inspection Activities is to provide reasonable assurance that equipment and components within the scope of License Renewal, which are not readily accessible, will continue to fulfill their intended functions during the period of extended operation (Item 9, Table 18-1). A one-time inspection has been performed in accordance with the schedule provided in Item 9, Table 18-1, to assess the aging of components and structures located in areas not routinely accessed due to high-radiation, high-temperature, confined spaces, location behind security or missile barriers, or normally flooded. The external condition of structures, supports, piping, and equipment has been determined by visual inspection. These inspections detect the aging effect of loss of material.

Concrete is inspected to detect the aging effects of loss of material, cracking, and change in material properties (Item 17, Table 18-1).

Infrequently accessed areas determined to be within the scope of license renewal and the focus of inspections within these area include:

- Reactor containment - Sump areas, cabling and supports*
- Reactor containment keyway - Leakage, structural support provided by the neutron shield tank.
- Subsurface drains - Access shaft and component supports
- Cover for Containment dome plug - Structural condition

- Volume control tank cubicle - Structure, supports, and equipment.
- Black Battery Building - Supports that could affect power supply to AMSAC
- Cable spreading rooms, Cable tunnels, Upper areas of emergency switchgear rooms - Cable raceways and supports*
- New fuel storage area - Supports and structure affecting spent fuel pool cooling*
- Auxiliary Building filter and ion exchanger cubicles - Structure, supports, and equipment*
- Tunnel from Turbine Building to Auxiliary Building - Structure, supports, and piping*

Note: Representative samples will be inspected in areas denoted with an asterisk.

Inspection results are documented for evaluation and retention. Engineering evaluation assesses the severity of the visual inspection results and determines the extent of required actions or future inspections. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.1.3 Tank Inspection Activities

The purpose of the Tank Inspection Activities is to perform inspections of above ground and underground tanks to provide reasonable assurance that the tanks will perform their intended function through the period of extended operation (Item 10, Table 18-1).

A one-time inspection was performed in accordance with the schedule provided in Item 10, Table 18-1, for specified tanks that are within the scope of license renewal and could experience aging effects. The aging effect of concern for tanks is loss of material. A representative sample of tanks is designated for the one-time inspections in order to assess the condition of tanks that require aging management. The choice of representative tanks to be inspected is dependent on the material of construction for the tank, its contents, the foundation upon which the tank is based, and the type of coating. Visual inspections of internal and external surfaces are performed. Volumetric examinations are performed to look for indications of wall thinning on tanks that are founded on soil or buried. Indications of degradation are referred for evaluation by engineering.

The following tanks have been inspected or represented by suitable replacement samples:

- Emergency diesel generator tanks (fuel oil, coolant, and starting air)
- AAC diesel generator tanks (fuel oil, coolant and starting air)
- Security diesel generator tank (fuel oil)
- Underground fuel oil storage tanks

- Diesel-driven fire pump fuel oil storage tanks
- Refueling water storage tanks
- Chemical addition tanks
- Emergency condensate storage tanks
- Fire Protection/Domestic water storage tanks (re-inspection required during the Period of Extended Operation)
- Emergency service water pump diesel fuel oil storage tank

An engineering evaluation may determine that the observed condition is acceptable or requires repair; or, in the case of degraded coatings, may direct removal of the coating, non-destructive examination of the substrate material, and replacement of the coating. Re-inspections are dependent upon the observed surface condition, and the results of this engineering evaluation. For the one-time inspections, tank conditions were confirmed to be acceptable, but the fire protection/domestic water storage tanks require re-inspection during the Period of Extended Operation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

In addition to the one-time inspections of specified tanks, a second aspect of Item 10, Table 18-1 is to evaluate the need for ongoing inspections. The one-time inspection results for all tanks, except the fire protection/domestic water storage tanks, indicated acceptability during the complete Period of Extended Operation (PEO). The fire protection/domestic water storage tank will require re-inspection during the PEO based on an engineering evaluation of the one-time inspection results.

The combination of acceptable results from the one-time inspections, and the development of plans for future inspection of the fire protection/domestic water storage tanks, completes the tasks required for Item 10, Table 18-1.

18.1.4 Non-Environmental Qualification (EQ) Cable Monitoring

The purpose of the Non-EQ Cable Monitoring activities is to perform inspections on a limited, but representative, number or accessible cable jackets and connector coverings that are utilized in non-EQ applications (Item 19, Table 18-1). In order to confirm that ambient conditions are not changing sufficiently to lead to age-related degradation of the in-scope cable jackets and connector coverings, initial visual inspections for the non-EQ application insulated power cables, instrumentation cables, and control cables (including low-voltage instrumentation and control cables that are sensitive to a reduction in insulation resistance) are performed in accordance with a station procedure. Visual inspection of the representative samples of non-EQ power, instrumentation, and control cable jackets and connector coverings detect the presence of

cracking, discoloration, or bulging, which could indicate aging effects requiring management. These effects could be due to high radiation, high temperature, or wetted condition environments. Subsequent inspections to confirm ambient conditions will be performed at least once per 10 years following the initial inspection. Additionally, a commitment for the renewed operating licenses required that upon issuance of staff guidance regarding the aging management of fuse holders in power circuits, the non-EQ cable monitoring program was to be revised to address the guidance (Item 26, Table 18-1). Guidance was issued by the NRC in Revision 1 of the Generic Aging Lessons Learned (NUREG-1801). Implementation of the required surveillance included reviews of station drawings, listings of station fuses, and plant walkdowns. These efforts identified no fuse holders that are affected by this license renewal requirement.

The potentially adverse localized environment due to moisture which could lead to water-treeing in high- or medium-voltage cables that are within the scope of license renewal, is also detected by visually monitoring for the presence of water around cables. Programs utilizing periodic inspections and design features such as drains or sump pumps are used to control the cable localized environment (Item 27, Table 18-1). A maintenance surveillance procedure is used to periodically determine the extent of water in manholes of interest for license renewal so that corrective actions can be taken to avoid having the cable remain wetted. Cable found to be wetted for any significant period of time will be tested using an appropriate test method which has been proven to accurately assess the cable condition with regards to water treeing.

The source, intermediate, and power range neutron detector operate with high-voltage power supply in conjunction with low-voltage signal cables. Direct inspection and testing are performed using station procedures to identify age-related degradation in the insulation on those cables.

Any anomalies resulting from the inspections will be dispositioned by Engineering and will consider the cable environment including the potential for moisture in the areas of the anomalies. Occurrence of an anomaly that is adverse to quality will be entered into the Corrective Action System. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable. Although age-related degradation is not expected for power, instrumentation, and control cables and connectors in their normal environments, visual inspections provide reasonable assurance that the intended functions will be maintained.

18.2 EXISTING AGING MANAGEMENT ACTIVITIES

The following sections provide a description of aging management programs and activities that were essentially in-place when the renewed operating licenses were issued for Surry. These programs and activities were part of the licensing basis for the original operating license period. For some programs, however, enhancements were identified during the license renewal process as necessary to manage aging of various station systems, structures, and components during the period of extended operation.

18.2.1 Augmented Inspection Activities

The purpose of the Augmented Inspection Activities is to perform examinations of selected components and supports in accordance with requirements identified in the Technical Specifications, Technical Requirements Manual, UFSAR, license commitments, industry operating experience, and good practices for the station. Augmented inspections are outside the required scope of ASME Section XI. The scope of Augmented Inspection Activities to be performed during each refueling outage is identified by Engineering in accordance with controlled procedures. Component conditions are monitored to detect degradation due to loss of material and cracking. Inspections include visual, surface, and volumetric examinations. The extent of each component inspection is defined within the Augmented Inspection Activities program description.

Augmented Inspection Activities include:

- Sensitized stainless steel (Class 1) circumferential, longitudinal, branch connection, and socket welds on the Pressurizer spray line welds as required by Technical Specifications.
- Sensitized stainless steel (Class 2) circumferential, longitudinal, branch connection, and socket welds as required by Technical Specifications.
- High Energy Lines Outside of Containment (Main Steam and Feedwater)
- Reactor vessel incore detector thimble tubes
- Component supports
- Steam generator feedwater nozzles
- Pressurizer instrument connections
- Reactor vessel head

The scope of augmented inspections has been revised to include the core barrel hold-down spring (Item 3, Table 18-1). The inspection addresses the aging effect of loss of pre-load. Additionally, the scope for augmented inspections has been revised to include an inspection of the pressurizer surge line connection to the reactor coolant system hot-leg loop piping (Item 2, Table 18-1). The inspection addresses the aging effect of thermal fatigue failure of the weld due to environmental effects, as described in NRC Generic Safety Issue (GSI)-190. The scope, frequency, qualifications, and methods of inspection are consistent with those utilized for the Inservice Inspection Program in accordance with ASME Section XI. Inspections are performed during each 120-month inspection interval. Industry efforts to study the environmental effects on weld thermal fatigue failure will continue to be evaluated by Dominion. If warranted, alternatives to this planned inspection (re-evaluation, replacement, or repair) will be submitted to the NRC for review.

The acceptance standards for non-destructive examinations for the Augmented Inspection Activities are consistent with guidance provided in ASME Section XI or are provided within applicable examination procedures. Evidence of loss of material, loss of pre-load, or cracking requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.2 Battery Rack Inspections

The purpose of the Battery Rack Inspections is to provide reasonable assurance of the integrity of the supports for various station batteries. Loss of material due to corrosion is the aging effect. Periodic checks of the rack integrity are performed, coincident with periodic battery inspections, to determine the physical condition of the battery support racks. The condition and mechanical integrity of the battery support racks are visually inspected to provide reasonable assurance that their function to adequately support the batteries is maintained. Visual inspections are adequate to identify degradation of the physical condition of the support racks. These inspections check for corrosion of the support rack structural members.

If any material condition deterioration is sufficiently extensive to interfere with integrity of the racks, the Corrective Action System will determine the cause and appropriate action to repair and prevent recurrence of the degradation. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.3 Boric Acid Corrosion Surveillance

Leakage from borated systems inside Containment creates the potential for degradation of components. Inspections are performed to provide reasonable assurance that borated water leakage does not lead to undetected loss of material from the reactor coolant pressure boundary and surrounding components. Carbon steel is particularly susceptible, but copper also can be damaged.

In Generic Letter 88-05 (Reference 2), the NRC identified concerns with boric acid corrosion of carbon-steel reactor pressure boundary components inside Containment. In response to this generic letter, activities were developed to examine primary coolant components for evidence of borated water leakage that could degrade the external surfaces of nearby structures or components, and to implement corrective actions to address coolant leakage.

Primary coolant systems inside Containment are examined for evidence of borated water leakage. An overall visual inspection of coolant system piping is performed, with particular interest in potential leakage locations. Insulated portions of the coolant systems are examined for signs of borated water leakage through the insulation by examining accessible joints and exposed surfaces of piping and equipment. Vertical components are examined at the lowest elevation. Components and connections that are not accessible are examined by looking for borated water

leakage on the surrounding area of the floor or adjacent equipment and insulation. The inspection scope includes connections to the reactor coolant system from the normal coolant letdown and makeup piping, and from the emergency core cooling systems. Components that are in the vicinity of borated water leakage are also examined for damage resulting from the leakage.

When visual inspections indicate evidence of borated water leakage, an evaluation is performed to determine if degradation of the leaking component or nearby affected components has occurred; and whether the observed condition is acceptable without repair. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.4 Chemistry Control Program for Primary Systems

The purpose of the Chemistry Control Program for Primary Systems is to provide reasonable assurance that water quality is compatible with the materials of construction in the plant systems and equipment in order to minimize the loss of material and cracking. The Chemistry Control Program for Primary Systems creates an environment in which material degradation is minimized, therefore, maintaining material integrity and reducing the amount of corrosion product that could accumulate and interfere with equipment operation or heat transfer.

Chemistry sampling is performed and the results are monitored and trended by maintaining logs of measured parameters. Acceptability of the measurements is determined by comparison with the limits established in the Chemistry Control Program for Primary Systems. Acceptance criteria for the measured primary chemistry parameters are listed in the Chemistry Control Program for Primary Systems. The acceptance criteria reflect EPRI guidelines for parameters that have been shown to contribute to component degradation. Adherence to the guidelines minimizes the aging effects of loss of material and cracking.

Action levels are established to initiate corrective action when the established limits are approached or exceeded. Depending on the magnitude of the out-of-limit condition, plant shutdown may be performed to minimize aging effects while plant actions are being taken. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.5 Chemistry Control Program for Secondary Systems

The purpose of the Chemistry Control Program for Secondary Systems is to provide reasonable assurance that water quality is compatible with the materials of construction in the plant systems and equipment in order to minimize the loss of material and cracking. The Chemistry Control Program for Secondary Systems creates an environment in which material degradation is minimized, therefore, maintaining material integrity and reducing the amount of corrosion product that could accumulate and interfere with equipment operation or heat transfer.

Chemistry results are monitored and trended by maintaining logs of measured parameters. Acceptability of the measurements is determined by comparison with limits established by the Chemistry Control Program for Secondary Systems. Acceptance criteria for the measured secondary chemistry parameters are listed in the Chemistry Control Program for Secondary Systems. The acceptance criteria reflect EPRI guidelines for parameters that have been shown to contribute to component degradation. Adherence to the guidelines minimizes the aging effects of loss of material and cracking.

Action levels are established to initiate corrective action when the established limits are exceeded. Depending on the magnitude of the out-of-limit condition, power is reduced or the plant is shut down to minimize aging effects while plant actions are being taken. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.6 Civil Engineering Structural Inspection

The maintenance rule, 10 CFR 50.65, requires licensees to monitor the condition of structures against established goals. During the period of extended operation, the provisions of the Maintenance Rule Program will be utilized to provide reasonable assurance of the continuing capability of civil engineering structures to fulfill their intended functions. The application for the renewed operating licenses included a commitment to expand the scope of Civil Engineering Structural Inspections to include inspections required for license renewal, including annual monitoring of groundwater chemistry with consideration given to seasonal chemistry variations (Item 4, Table 18-1, Item 16, Table 18-1, & Item 28, Table 18-1). This expansion for Item 4 is implemented in accordance with the schedule provided in Table 18-1. Implementation of the license renewal requirement to include an expanded scope for Item 4 was completed with the approval of a civil engineering technical report and an engineering surveillance procedure. For Items 16 and 28, groundwater chemistry monitoring is identified as a requirement of the Civil Engineering Guideline for Monitoring Structures, and is accomplished using a plant surveillance procedure each calender quarter. This quarterly surveillance accommodates seasonal variation in groundwater conditions. Measured chemistry results that exceed established acceptance criteria are processed through the corrective action system, including engineering evaluation.

Structural monitoring inspections are visual inspections that are performed to assess the overall physical condition of the structure. For concrete structures, this includes elastomer sealant and gasket materials.

Inspections are performed by trained inspectors and include representative samples of both the interior and exterior accessible surfaces of structures. Documentation of inspection results includes a general description of observed conditions, location and size of anomalies, and the noted effects of environmental conditions. If an inaccessible area becomes accessible by such means as dewatering, excavation or installation of radiation shielding, an opportunity will exist

for additional inspections. The application for the renewed operating licenses included a commitment to provide guidance in plant procedures in accordance with the schedule provided in Item 5, Table 18-1, to take advantage of such inspection opportunities when they arise for inaccessible areas (Item 5, Table 18-1). This commitment has been fulfilled through revised implementing procedures at Surry.

Concrete is inspected to detect the aging effects of loss of material, cracking, and change in material properties.

A visual indication of: (1) loss of material for concrete and structural steel, (2) significant cracking for concrete and masonry walls, (3) cracking or change in material properties for elastomers, (4) loss of material or loss of form for soil, and (5) gross indications of change in material properties of concrete, each requires an engineering evaluation (Item 17, Table 18-1).

Inspections of masonry walls are included in this program. The inspections check for cracks of joints and missing or broken blocks.

The engineering evaluation of inspection results, including groundwater chemistry results, determines whether analysis, repair, or additional inspections or testing is required to provide reasonable assurance that structures will continue to fulfill their intended functions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.7 Fire Protection Program

Regulatory requirements associated with fire protection systems and implementation plans are provided in 10 CFR 50.48 and 10 CFR 50, Appendix R. The Fire Protection Plan includes applicable National Fire Protection Association (NFPA) commitments and maintains compliance with NRC Branch Technical Position (BTP) 9.5-1 from the Standard Review Plan (Reference 3). Aging management concerns related to fire protection involve visual inspections of fire protection equipment and barriers, including doors, walls, floors, ceilings, penetration seals, fire-retardant coatings, fire damper housings, cable-tray covers, and fire stops.

Applicable aging effects that are found by visual examination include loss of material, separation and cracking/delamination, heat transfer degradation, and change in material properties. Aging effects on piping systems (including valve bodies and pump casings) that are dry or that carry water are evaluated in the same manner as for any other mechanical system. Testing of the fire protection pumps provides indication of heat transfer degradation, and inspections of the pumps provide indication of loss of material. Verification of piping integrity to maintain a pressure boundary for the fire protection system, and the availability of water are addressed by routine plant walkdowns, by pressure/flow tests that are conducted periodically, and by the Work Control Process (Item 30, Table 18-1). Visual inspections are performed periodically for hose stations, hydrants, and sprinklers.

The work control audit for Commitment Item 22 in Table 18-1 confirms that inspections of fire protection piping are included in the Work Control Process. This provides the verification that is necessary for completing Item 30.

Provisions to replace sprinklers or test a representative sample of sprinklers that have been in service for 50 years have been incorporated into the Fire Protection Program (Item 6, Table 18-1). This task conforms to the requirements of NFPA-25, Section 2-3.1.1. If testing is performed, re-testing will be performed at 10-year intervals per NFPA-25.

Fire protection equipment is examined for indications of visible damage. Acceptable sizes for breaks, holes, cracks, gaps, or clearances in fire barriers, and acceptable amounts of sealant in penetrations are established in the inspection procedures. Any questions regarding the ability of the barrier to fulfill its fire protection function are addressed by engineering evaluation. Acceptance criteria for fire protection equipment performance tests (i.e., flow and pressure tests) are provided in the appropriate test procedures. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.8 Fuel Oil Chemistry

The Fuel Oil Chemistry program manages the loss of material by requiring that oil quality is compatible with the materials of construction in plant systems and equipment. Poor fuel oil quality could lead either to degradation of storage tanks or accumulations of particulates or biological growth in the tanks. The purpose of the Fuel Oil Chemistry program is to minimize the existence of contaminants such as water, sediment, and bacteria which could degrade fuel oil quality and damage the fuel oil system and interfere with the operation of safety-related equipment.

The Fuel Oil Chemistry program is an aging effects mitigation activity which does not directly detect aging effects. The Fuel Oil Chemistry guidelines address the parameters to be monitored and the acceptance limit for each parameter. The acceptance criteria reflect ASTM guidelines for parameters that have been shown to contribute to component degradation. Adherence to the guidelines mitigates the aging effect of loss of material. Parameters analyzed and found to be outside established limits will be reported to Engineering, an evaluation will be performed, and appropriate corrective actions will be taken. Occurrence of significant deviations that are adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.9 General Condition Monitoring Activities

General Condition Monitoring Activities are performed for the assessment and management of aging for components that are located in normally accessible areas. The results of this monitoring are the basis for initiating required corrective action in a timely manner. This monitoring is based on the observations that are made during focused inspections that are

performed on a periodic basis. Guidance is implemented in procedures regarding inspection criteria that focus on detection of aging effects during General Condition Monitoring Activities (Item 8, Table 18-1).

- Guidance is implemented in a procedure for Engineering that provides direction for performing periodic plant walkdowns to monitor equipment conditions. The walkdowns include surveillance activities and observations of maintenance tasks. Indications of age-related degradation are monitored during these walkdowns. Additional inspection information regarding the integrity of components is provided by inspections that support the implementation of the Boric Acid Corrosion Control Program.
- Guidance is implemented in procedures for Health Physics that provides direction for performing walkdowns within the Radiological Control Area to monitor for evidence of potential pressure-boundary degradation (e.g., leakage) of plant equipment.

The external condition of supports, piping, doors, and equipment will be determined by visual inspection. General Condition Monitoring Activities include inspections that are performed periodically in radiologically controlled areas to check for borated water leakage, and general system walkdowns that occur periodically during normal operation and refueling outages. Such walkdowns include inspections for the condition of structural supports and doors.

Inspection criteria for non-ASME Section XI component supports and doors, as part of General Condition Monitoring are procedurally implemented using guidance provided in an Engineering document (Item 7, Table 18-1). Doors that require inspection for age-related degradation are also designated as EQ doors. Monitoring of the EQ doors occurs as directed by the Technical Requirements Manual. Initial inspections will be completed, using the criteria, in accordance with the schedule provided in Item 7, Table 18-1.

These inspections provide information to manage the aging effects of loss of material, change in material properties, separation and cracking/delamination, and cracking.

The acceptance criteria for visual inspections are identified in procedures that direct the various monitoring activities. Responsibility for the evaluation of identified visual indications of aging effects is assigned to Engineering personnel. Evaluations of anomalies found during General Condition Monitoring Activities determine whether analysis, repair, or further inspection is required. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.10 Inspection Activities - Load Handling Cranes and Devices

The load handling cranes within the scope of license renewal are listed below:

- Containment polar cranes

- Containment jib cranes
- Containment annulus monorails
- Refueling manipulator cranes
- Fuel handling bridge crane
- New fuel transfer elevator
- Spent fuel crane
- Auxiliary Building monorails

The long-lived passive components of these cranes that are subject to aging management review include rails, towers, load trolley steel, fasteners, base plates, and anchorage. The application for the renewed operating licenses included a commitment for an internal inspection of representative sections of the box girders for the polar cranes to be implemented as a one-time only inspection (Item 13, Table 18-1). The internal visual inspections have been completed. There were no findings of age-related degradation, and an engineering assessment indicated no need for supplemental examinations.

The Inspection Activities - Load Handling Cranes and Devices has been developed in accordance with ASME B30.2 (Reference 13) and the inspection activities for monorails are developed in accordance with ASME B30.11 (Reference 14).

The Work Control Process directs structural integrity inspections of applicable cranes which include specific steps to check (visually inspect) the condition of structural members and fasteners on the cranes, the runways along which the cranes move, and the baseplates and anchorages for the runways. The applicable aging effect is identified as loss of material. If the nature of any identified discrepancies is such that corrective action can be completed within the scope of the procedure performing the inspection, no additional corrective action may be necessary. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.11 In-Service Inspection (ISI) Program - Component and Component Support Inspections

The ISI Program - Component and Component Support Inspections are performed in accordance with the requirements of Subsections IWB, IWC, and IWF of ASME XI, Rules for Inservice Inspection of Nuclear Power Plant Components. For this program, the license renewal concerns with respect to Subsection IWC include only the carbon steel piping that is susceptible to high energy line breaks in the feedwater and main steam systems. Inservice Inspection requirements may be modified by applicable Relief Requests and Code Cases, which are approved by the NRC specifically for each unit. The scope and details of the inspections to be

performed are contained in the individualized Inservice Inspection Plan for each unit. Each Inservice Inspection Plan is developed for a 120-month inspection interval and submitted to the NRC. The examinations required by ASME Section XI utilize visual, surface, and volumetric inspections to detect loss of material, cracking, gross indications of loss of pre-load, and gross indications of reduction in fracture toughness (which presents itself as cracking of cast-austenitic stainless steel valve bodies due to thermal embrittlement).

Dominion actively participates in the EPRI-sponsored Materials Reliability Project Industry Task Group on thermal fatigue which currently is developing industry guidance for the management of fatigue caused by cyclic thermal stratification and environmental effects. Dominion is committed to following industry activities related to failure mechanisms for small-bore piping and will evaluate changes to inspection activities based on industry recommendations (Item 11, Table 18-1). This commitment is closed based on an engineering evaluation indicating that no new industry initiatives have occurred regarding inspections of small-bore piping. The current ASME requirements remain in effect to perform visual inspections of small-bore socket welds, and visual and volumetric inspections of small-bore butt welds.

Acceptance standards for inservice inspections are identified in Subsection IWB for Class 1 components, Subsection IWC for included Class 2 components, and in Subsection IWF for component supports. Table IWB 2500-1 refers to acceptance standards listed in paragraph IWB 3500. Anomalous indications beyond the criteria set forth in the Code acceptance standards that are revealed by the inservice inspections of Class 1 components may require additional inspections of similar components in accordance with Section XI. Evidence of loss of material, cracking, and gross indications of either loss of pre-load or reduction of fracture toughness requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.12 ISI Program - Containment Inspection

The ISI Program - Containment Inspection for concrete containments and containment steel liners implements the requirements in 10 CFR 50.55a and Subsections IWE and IWL of ASME Section XI. The program incorporates applicable code cases and approved relief requests. The provisions of 10 CFR 50.55a are invoked for inaccessible areas within the Containment structure.

Loss of material is the aging effect for the containment steel liner. Surface degradation and wall thinning are checked by visual and volumetric examinations. The frequency and scope of examination requirements are specified in 10 CFR 50.55a and Subsection IWE. Loss of material, cracking and change in material properties are the aging effects for the containment concrete and are checked by visual examinations. The frequency and scope of examination requirements are specified in 10 CFR 50.55a and Subsections IWL. These inspections provide reasonable

assurance that aging effects associated with the containment liner and concrete are detected prior to compromising design basis requirements. The evaluations of accessible areas provide the basis for extrapolation to the expected condition of inaccessible areas, and an assessment of degradation in such areas.

During the course of containment inspections, anomalous indications are recorded on inspection reports that are kept in Station Records. Acceptance standards for the IWE inspections are identified in ASME Section XI Table IWE 2500-1 and refer to 10 CFR 50, Appendix J. For the IWL inspections, acceptance standards are identified in ASME Section XI Table IWL 2500-1. Engineering evaluations are performed for inspection results that do not meet established acceptance standards. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.13 ISI Program - Reactor Vessel

The ISI Program - Reactor Vessel is performed in accordance with the requirements of Subsection IWB of ASME XI, Rules for Inservice Inspection of Nuclear Power Plant Components. Inservice Inspection requirements may be modified by applicable Relief Requests and Code Cases, which are approved by the NRC specifically for each unit. The scope and details of the inspections to be performed are contained in the individualized Inservice Inspection Plan for each unit. Each Inservice Inspection Plan is developed and submitted to the NRC for a 120-month inspection interval. A commitment was made for Dominion to evaluate industry recommendations and enhance inspections of core support lugs as appropriate (Item 12, Table 18-1). Requests for information from EPRI and from the Nuclear Energy Institute (NEI) indicate that no updated guidance is being developed by the industry for the inspection of core support lugs. The continued use of visual inspections in accordance with ASME Section XI is proper. Examinations of the core support lugs have been completed for Surry Units 1 and 2 during the 10-year reactor vessel inservice inspections.

In accordance with ASME Section XI, reactor vessel components are inspected using a combination of surface examinations, volumetric examinations, and visual examinations to detect the aging effects of loss of material, cracking, gross indications of loss of pre-load, and gross indications of reduction in fracture toughness. Acceptance standards for inservice inspections are identified in Subsection IWB for Class 1 components. Table IWB 2500-1 refers to acceptance standards listed in paragraph IWB 3500. Anomalous indications that are revealed by the inservice inspections may require additional inspections of similar components, in accordance with Section XI. Evidence of aging effects requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.14 Reactor Vessel Integrity Management

The scope of the Reactor Vessel Integrity Management activities is focused on ensuring adequate fracture toughness of the reactor vessel beltline plate and weld materials. Neutron dosimetry and material properties data derived from the reactor vessel materials irradiation surveillance program are used in calculations and evaluations that demonstrate compliance with applicable regulations. The Reactor Vessel Integrity Management activities includes the following aspects:

- Irradiated sample (capsule) surveillance.
- Vessel fast neutron fluence calculations.
- Measurements and calculations of nil-ductility transition temperature (RT_{NDT}) for vessel beltline materials.
- Measurements and calculations of Charpy Upper Shelf Energy (C_vUSE).
- Calculation of reactor coolant system pressure-temperature (P-T) operating limits, and Low Temperature Overpressure Protection System (LTOPS) setpoints.
- Pressurized Thermal Shock (PTS) screening calculations.

Specimen capsules were placed in each of the reactors prior to initial irradiation and contain reactor vessel plate and weld material samples. The baseline mechanical properties of reactor vessel steels are determined from pre-irradiation testing of Charpy V-notch and tensile specimens. Post-irradiation testing of similar specimens provides a measure of radiation damage. Refer to Section 4.1.7.1.

Fast neutron irradiation is the cause of radiation damage to the reactor vessel beltline. The results of surveillance capsule dosimetry analyses are used as benchmarks for calculations of neutron fluence to the surveillance capsules and to the reactor vessel beltline.

Measured values of Charpy transition temperature and C_vUSE are obtained from mechanical testing of irradiation surveillance program specimens. Measured values of transition temperature are used to determine the RT_{NDT} for the limiting reactor vessel beltline material. RTNDT is a key analysis input for the determination of reactor coolant system P-T operating limits and LTOPS setpoints. Measured values of transition temperature shift are similarly utilized in PTS screening calculations required by 10 CFR 50.61. Measured values of C_vUSE are used to verify compliance with the upper shelf energy requirements of 10 CFR 50 Appendix G.

Acceptable values are established for the following parameters:

- Heatup and cooldown limits, as implemented by Technical Specifications, to ensure reactor vessel integrity.

- A PTS reference temperature that is within the screening criteria of 10 CFR 50.61.
- A fast fluence value for the surveillance capsule that bounds the expected fluence at the affected vessel beltline material through the period of extended operation.
- C_VUSE greater than limits set forth in 10 CFR 50, Appendix G.

Based on established parameters, calculations are performed to ensure that the units will remain within the acceptable values.

18.2.15 Reactor Vessel Internals Inspection

Visual inservice inspections are implemented in accordance with Category B-N-3 (Removable Core Support Structures) of ASME Section XI, Subsection IWB, to determine the possible occurrence of age-related degradation. These inspections are performed at 10-year intervals in accordance with the inspection plans submitted to the NRC. The scope of components that comprise the reactor internals includes the upper and lower core internals assemblies. This includes core support and hold-down spring components, as well as, the baffle/former bolting and barrel/former bolting. Additionally, a one-time focused inspection of the reactor vessel internals has been performed in accordance with the schedule provided in Item 14, Table 18-1. The one-time inspection looked for indications of the presence of aging effects identified in the aging management review for the reactor vessel internals. The inspection has been performed on one reactor (at either Surry or North Anna) and an engineering evaluation of results determines the need for inspections of the other units. Dominion will remain active in industry groups, including the EPRI-sponsored Materials Reliability Project Industry Task Group, to stay aware of any new industry recommendations regarding such aging management issues as neutron embrittlement, void swelling, and the synergistic effect of thermal and neutron embrittlement of internals sub-components (Item 14, Table 18-1). If future industry developments suggest the need for an alternate inspection plan during the period of extended operation, or negate the need for a one-time inspection, then Dominion will modify the proposed inspection plan.

An engineering evaluation to determine the most susceptible Virginia reactor (as mentioned in Item 14, Table 18-1) confirmed that for all aspects of the reactor vessel internals, except for the control rod guide cards, Surry Unit 1 is the most susceptible North Anna or Surry reactor with respect to aging effects. As a result, Surry Unit 1 was chosen for the enhanced inspections of reactor vessel internals. But since the guide cards for the Surry reactors have been replaced only in Unit 1, Unit 2 contains the older and more susceptible control rod guide cards. That circumstance has been addressed at Surry Power Station by inspecting the Unit 2 control rod guide cards in accordance with industry guidance during the 2012 refueling outage. The plan that accomplished the inspection of the Unit 2 control rod guide cards does not affect the commitment in Item 14, Table 18-1, since the required focus on susceptibility is maintained by inspecting the Unit 2 control rod guide cards, in addition to the Unit 1 reactor vessel internals.

Industry guidance regarding enhanced inspections consistent with the one-time focused inspection described above are contained in MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)." Dominion is implementing the one-time focused inspection for Surry Unit 1 in accordance with guidelines from MRP-227 and, in accordance with the requirements for the renewed operating licenses. The Unit 1 inspection results are used to evaluate the need for additional inspections at Surry Unit 2 and the North Anna units. As noted previously, the one-time focused inspections also will include inspections of the Unit 2 control rod guide cards.

Visual inspections are utilized to detect loss of material and cracking; as well as, gross indications of loss of pre-load and/or reduced fracture toughness. The acceptance standards for the visual examinations are summarized in ASME Subsection IWB-3520.2, Visual Examination, VT-3. These inspections are directed to be performed with the internals assemblies removed from the reactor vessel.

Acceptance standards for Reactor Vessel Internals Inspection activities are identified in ASME Section XI, Subsection IWB. Table IWB 2500-1 identifies references to the acceptance standards listed in Paragraph IWB 3500. Anomalous indications, that are revealed to be beyond the criteria in the acceptance standards by the inservice inspections, may require additional inspections. Evidence of any component degradation requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.16 Flow Accelerated Corrosion

The purpose of the Flow Accelerated Corrosion program is to identify, inspect, and trend components that are susceptible to the aging effect of loss of material as a result of Flow Accelerated Corrosion (FAC) in either single or two-phase flow conditions. This program has been implemented in accordance with NRC Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning* (Reference 15), and NUREG-1344, *Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants* (Reference 16), and EPRI Guideline NSAC-202L, Revision 4, *Recommendations for an Effective Flow Accelerated Corrosion Program* (Reference 17).

The scope of the Flow Accelerated Corrosion program includes portions of the feedwater systems, the main and auxiliary steam systems, and the steam generator blowdown lines.

The identification of components and piping segments to be included in each Flow Accelerated Corrosion effort is performed by Engineering using plant chemistry data, past inspection data, predictions from FAC-monitoring computer codes, and industry experience. Determination of whether a piping component has experienced FAC degradation is made by

measuring the current wall thickness using the UT method and comparing against previous baseline thickness measurement, if available. Visual inspections of the internals of non-piping components, such as pumps and valves, are performed as the equipment is opened for other repairs and/or maintenance, to determine whether flow-accelerated degradation is occurring.

The decision to repair or replace a component is made by Engineering. For the internal surface examinations, engineering evaluations are utilized to determine whether the results of visual inspections indicate conditions that require corrective action. Occurrences of significant degradation that are adverse to quality are entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.17 Service Water System Inspections

Compliance with Generic Letter 89-13 (Reference 18) requires a variety of inspections, non-destructive examinations, and heat transfer testing for components cooled by service water. Generic Letter 89-13 directed utilities to assess the following aspects of operational problems with service water cooling systems:

- Biofouling
- Heat Transfer Testing
- Routine Inspection and Maintenance
- Single-failure Walkdown
- Procedure Review

The Service Water System Inspections program provides reasonable assurance that corrosion (including microbiologically-influenced corrosion, MIC), erosion, protective coating failure, silting, and biofouling of service water piping and components will not cause a loss of intended function. The primary objectives of this program are to (1) remove excessive accumulations of biofouling agents, corrosion products, and silt; and (2) repair defective protective coatings and degraded service water system piping and components that could adversely affect performance. Preventive maintenance, inspection, and repair procedures have been developed to provide reasonable assurance that any adverse effects of exposure to service water are adequately addressed. The addition of biocide to the service water system reduces biological growth (including MIC) that could lead to degradation of components exposed to the service water.

Service Water System Inspections are performed to check for biofouling, damaged coatings, and degraded material condition. Heat transfer parameters for components cooled by service water are monitored. Visual inspections are performed to check for loss of material and changes in material properties. Heat transfer testing is performed to identify the aging effects of loss of material and heat transfer degradation.

The acceptance criteria for visual inspections are identified in the procedures that perform the individual inspections. The procedures identify the type and degree of anomalous conditions that are signs of degradation. In the case of service water, degradation includes biofouling as well as material degradation. Engineering evaluations determine whether observed deterioration of material condition is sufficiently extensive to lead to loss of intended function for components exposed to the service water. The degraded condition of material or of heat transfer capability may require prompt remediation. Occurrence of significant degradation that is adverse to quality is entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.18 Steam Generator Inspections

Steam Generator Inspections are performed in accordance with Technical Specifications and Inservice Inspection requirements of ASME Section XI. Steam Generator Inspections plans are based upon the guidelines established by Nuclear Energy Institute document, NEI 97-06 (Reference 4) and the Electric Power Research Institute steam generator inspection guidelines (Reference 5). Steam generator tubing inspections are performed on a sampling basis. The sample population inspected meets or exceeds the requirements of Technical Specifications. Qualified techniques, equipment and personnel are used for inspections in accordance with site-specific eddy current analysis guidelines.

Examination of steam generator sub-components other than tubes are performed as required by the governing edition and addenda of ASME Section XI, as imposed by 10 CFR 50.55a. In some cases the specific inspection requirements of ASME Section XI are modified by regulatory commitments and approved Relief Requests. Inspections of the steam generators to check for loss of material, cracking, and gross indications of loss of pre-load include a combination of visual inspections, surface examinations, and volumetric examinations. Tubing inspections are performed in accordance with ASME Section XI, Subsection IWB.

Acceptance standards for steam generator inspections are provided in ASME Section XI, Subsections IWB-3500 and IWC-3500. Evidence of component degradation requires engineering evaluation for determination of corrective action. Occurrence of significant degradation that is adverse to quality will be entered into the Corrective Action System. Corrective action provides reasonable assurance that conditions adverse to quality are promptly corrected.

18.2.19 Work Control Process

Performance testing and maintenance activities, both preventive and corrective, are planned and conducted in accordance with the station's Work Control Process. The Work Control Process integrates and coordinates the combined efforts of Maintenance, Engineering, Operations, and other support organizations to manage maintenance and testing activities. Performance testing on heat exchangers evaluates the heat transfer capability of the components to determine if heat transfer degradation is occurring. Maintenance and engineering activities provide opportunities for inspectors who are Visual Test (VT) qualified to visually inspect the surfaces (internal and

external) of plant components and adjacent piping (Item 21, Table 18-1). Adjacent piping is primarily the internal piping surfaces immediately adjacent to a system component that is accessible through the component for visual inspection. Visual inspections performed through the Work Control Process provide data that can be used to determine the effectiveness of the Chemistry Control Program for Primary Systems and Chemistry Control Program for Secondary Systems to mitigate the aging effects of cracking, loss of material, and change of material properties.

The application for the renewed operating licenses included a commitment for changes to procedures to reasonably assure that consistent inspections of components are completed during the process of performing work control process activities (Item 15, Table 18-1). Implementation of consistent inspections is accomplished using automated inspection instructions for work orders involving components and structures that have been identified as requiring aging management. The instructions consistently require inspections to identify a variety of aging mechanisms required for the renewed operating licenses.

The Work Control Process also provides opportunities through preventive maintenance sampling (predictive analysis) to collect lubricating oil and engine coolant samples for subsequent analysis of contaminants that would provide early indication of an adverse environment that can lead to material degradation.

The inspections, testing, and sampling performed under the Work Control Process provide reasonable assurance that the following aging effects will be detected:

- loss of material
- cracking
- heat transfer degradation
- separation and cracking/delamination
- change in material properties (Item 18, Table 18-1)

The change in material properties, as listed in Item 18, refers to elastomeric components. A commitment was included for license renewal to ensure that procedures are performed through the work control process to inspect the elastomeric seals used on the flood barriers (stop logs) for the emergency service water pump house at the low-level intake structure, and the elastomeric seals used in the expansion joints for the intake canal concrete liner. Separate procedures exist for these two inspections, and are managed using the work control process.

The acceptance criteria for visual inspections, testing, or sampling are currently identified in the procedures that perform the individual maintenance, testing, or sampling activity. The procedures identify the type and degree of anomalous conditions that are signs of degradation.

Whenever evidence of aging effects exists, an engineering evaluation is performed to determine whether the observed condition is acceptable without repair. Occurrence of significant aging effects that is adverse to quality is entered into the Corrective Action System. If the evaluation of an anomalous condition indicates that the occurrence was unexpected for the operational conditions involved, the Work Control Process ensures that locations with similar material and environmental conditions (both within the system and outside the system) are inspected as directed by a Station procedure (Item 29, Table 18-1).

As confirmation that the Work Control Process has inspected representative components from each component group for which the Work Control Process is credited to manage the effects of aging, periodic audits of inspections actually performed will be performed and, if Work Control Process activities are found not to be representative, supplemental inspections will be performed (Item 22, Table 18-1). Two audits of the Work Control Process are required, and each will consists of a review of the previous 10 years of historical data. The audits are performed in accordance with the schedule provided in Item 22, Table 18-1. Any required supplemental inspections are completed within 5 years after the audits are performed. The initial audit has been completed prior to the Period of Extended Operation. Results from the audit indicate that supplemental inspections are required during the 5-year follow-up period.

18.2.20 Corrective Action System

The Corrective Action System is a required element of the Quality Assurance Program outlined in the Quality Assurance Topical Report (Chapter 17 of the Updated Final Safety Analysis Report). The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, Standard Review Plan for License Renewal. The Corrective Action System activities include the elements of corrective action, confirmation process, and administrative controls; and is applicable to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal.

18.3 TIME-LIMITED AGING ANALYSIS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of TLAAs for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

18.3.1 Reactor Vessel Neutron Embrittlement

The reactor vessel is subjected to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. Analyses have been performed that address the following:

- Upper shelf energy

- PTS
- RCS P-T operating limits

18.3.1.1 Upper Shelf Energy

The Charpy V-notch test provides information about the fracture toughness of reactor vessel materials. 10 CFR 50 requires the C_VUSE of reactor vessel beltline materials to meet Appendix G requirements. If the USE of a reactor vessel beltline material is predicted to not meet Appendix G requirements, then licensees must submit an analysis that demonstrates an equivalent margin of safety at least three years prior to the time the material is predicted to not meet those requirements.

Reactor vessel calculations have been performed which demonstrated that the upper shelf energy values of limiting reactor vessel beltline materials at the end of the period of extended operation meet Appendix G requirements. Thus, the TLAA has been projected to the end of the period of extended operation and is found to be adequate.

18.3.1.2 Pressurized Thermal Shock

A limiting condition on reactor vessel integrity, known as PTS, may occur during postulated system transients, such as a loss-of-coolant accident (LOCA) or a steam line break. Such transients may challenge the integrity of the reactor vessel under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high re-pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect in the vessel wall.

The reference temperature for pressurized thermal shock (RT_{PTS}) is defined in 10 CFR 50.61. RT_{PTS} values for the limiting reactor vessel materials at the end of the period of extended operation have been recalculated by Dominion. At the end of the period of extended operation, the calculated RT_{PTS} values for the beltline materials are less than the applicable screening criteria established in 10 CFR 50.61. Thus, the TLAA has been projected to the end of the period of extended operation and is found to be adequate.

18.3.1.3 Pressure-Temperature Limits

Atomic Energy Commission (AEC) General Design Criterion (GDC) 14 of Appendix A of 10 CFR 50, *Reactor Coolant Pressure Boundary,* requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage (or rapid failure) and of gross rupture. AEC GDC 31, *Fracture Prevention of Reactor Coolant Pressure Boundary,* requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that when stressed under operating, maintenance, and testing conditions the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

Reactor vessel neutron fluence values corresponding to the end of the period of extended operation and reactor vessel beltline material properties were used to determine the limiting value of RT_{NDT}, and to calculate RCS P-T operating limits valid through the end of a period of extended operation. Maximum allowable LTOPS power operated relief valve lift setpoints have been developed on the basis of the P-T limits applicable to the period of extended operation. Revised RCS P-T limit curves and LTOPS setpoints will be submitted for review and approval prior to the expiration of the existing technical specification limits in order to maintain compliance with the governing requirements of 10 CFR 50 Appendix G.

The TLAA has been projected to the end of the period of extended operation and is found to be adequate.

18.3.2 Metal Fatigue

The thermal fatigue analyses of the station's mechanical components have been identified as TLAAs.

18.3.2.1 **ASME Boiler and Pressure Vessel Code, Section III, Class 1**

The steam generators, pressurizers, reactor vessels, reactor coolant pumps, control rod drive mechanisms (CRDMs), and pressurizer surge line piping have been analyzed using the methodology of the ASME Boiler and Pressure Vessel Code, Section III, Class 1.

The ASME Boiler and Pressure Vessel Code, Section III, Class 1, requires a design analysis to address fatigue and establish limits such that the initiation of fatigue cracks is precluded.

Experience has shown that the transients used to analyze the ASME III requirements are often very conservative. Design transient magnitude and frequency are more severe than those occurring during plant operation. The magnitude and number of the actual transients are monitored. This monitoring assures that the existing frequency and magnitude of transients are conservative and bounding for the period of extended operation, and that the existing ASME III equipment will perform its intended functions for the period of extended operation. A cycle counting program (Section 18.4.2) is in place to provide reasonable assurance that the actual transients are smaller in magnitude and within number of the transients used in the design.

Fatigue analyses for the steam generators, pressurizers, reactor vessels, reactor coolant pumps, CRDMs, and pressurizer surge lines have been evaluated and determined to remain valid for the period of extended operation.

Fatigue analyses for the reactor vessel closure studs have been re-analyzed. The analyses for these components have been projected to be valid for the period of extended operation.

18.3.2.2 **Reactor Vessel Underclad Cracking**

In early 1971, an anomaly was identified in the heat-affected zone of the base metal in a European-manufactured reactor vessel. A generic fracture mechanics evaluation by Westinghouse demonstrated that the growth of underclad cracks during a 40-year plant life would be insignificant.

The evaluation was extended to 60 years using fracture mechanics evaluation based on a representative set of design transients. The occurrences were extrapolated to cover 60 years of service life. This 60-year evaluation shows insignificant growth of the underclad cracks and is documented in WCAP-15338 (Reference 21). The plant-specific design transients are bounded by the representative set used in the evaluation.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation and has been found to be acceptable.

18.3.2.3 **ANSI B31.1 Piping**

The balance-of-plant piping and reactor coolant pressure boundary piping except the pressurizer surge line piping are designed to the requirements of ANSI B31.1, *Power Piping.*

ANSI B31.1 design requirements assume a stress range reduction factor in order to provide conservatism in the piping design while accounting for fatigue due to thermal cyclic operation. This reduction factor is 1.0, provided the number of anticipated cycles is limited to 7000 equivalent full-temperature cycles. A piping system would have to be thermally cycled approximately once every three days over a plant life of 60 years to reach 7000 cycles. Considering this limitation, a review of the ANSI B31.1 piping within the scope of license renewal has been performed to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically, these systems are subjected to continuous steady-state operation. Significant variation in operating temperatures occur only during plant heatup and cooldown, during plant transients, or during periodic testing.

The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation.

18.3.2.4 **Environmentally Assisted Fatigue**

GSI-190 (Reference 6) identifies a NRC staff concern about the effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. The reactor water's environmental effects as described in GSI-190, are not included in the CLB. As a result, the criterion specified in 10 CFR 54.3(a)(6) is not satisfied. Hence, environmental effects are not TLAAs. GSI-190, which was closed in December 1999, has concluded that environmental effects have a negligible impact on core damage frequency, and as

such, no generic regulatory action is required (Reference 7). However, as part of the closure of GSI-190, the NRC has concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs. As demonstrated in the preceding sections, fatigue evaluation in the original transient design limits remain valid for the period of extended operation. Confirmation by transient cycle counting will ensure that these transient design limits are not exceeded. Secondly, the reactor water's environmental effects on fatigue life were evaluated using the most recent data from laboratory simulation of the reactor coolant environment.

As a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories evaluated, in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Reference 8), fatigue-sensitive component locations at plants designed by all four U. S. Nuclear Steam Supply System vendors. The pressurized water reactor calculations, especially the early-vintage Westinghouse PWR calculations, are directly relevant to the Dominion stations. The description of the "Older Vintage Westinghouse Plant" evaluated in NUREG/CR-6260 applies to the Surry station. In addition, the transient cycles considered in the evaluation match or bound the design. The results of NUREG/CR-6260 analyses, and additional data from NUREG/CR-6583 (Reference 9) and NUREG/CR-5704 (Reference 10), were then utilized to scale up the plant-specific cumulative usage factors (CUFs) for the fatigue-sensitive locations to account for environmental effects.

Based on these adjusted CUFs (using the environmental fatigue penalty factor), it has been determined that the surge line connection at the reactor coolant system's hot leg pipe exceeds the design threshold of 1.0. As a consequence, management of environmentally assisted fatigue is required.

Since the Surry piping design code for the safety injection (SI) and charging line piping is USAS B31.1, no explicit fatigue analysis has been performed. However, since the physical attributes of the piping systems, nozzles, and transient characteristics are similar to those at North Anna, the evaluations for North Anna SI and charging line nozzles are applicable to Surry. The CUFs that were adjusted for environmental effects for the North Anna SI and charging line nozzles initially were determined to exceed the design threshold of 1.0. Subsequent fatigue evaluations (Reference 34), using the methodology of ASME Section III, NB-3200, confirm that the CUFs for the SI and charging line nozzles do not exceed the ASME code allowable value of 1.0, including the effects of the reactor water environment. Therefore, the North Anna (and subsequently Surry) SI and charging line nozzles will not require enhanced inspections for the management of environmentally assisted fatigue (Item 25, Table 18-1).

The approach to manage environmentally assisted fatigue for the surge line is developed from one or more of the following options and submitted to the NRC for review prior to the

period of extended operation (Item 24, Table 18-1):

1. Further refinement of the fatigue analysis (e.g., NB-3200 analysis) to lower the CUFs to below 1.0, or
2. Repair of the affected locations, or
3. Replacement of the affected locations, or
4. Inspection of the affected locations.

The surge line weld at the hot leg pipe connection is included (Item 2, Table 18-1) as an Augmented Inspection Activity (Section 18.2.1). Baseline inspections of the surge line welds were required prior to entry into the period of extended operation, and were completed for Units 1 and 2 in November 2007 and May 2008 respectively. The inspections produced no recordable indications. Inspections will also occur once per ISI Interval (i.e., once every 10 years). The inspection frequency is based upon fracture mechanics evaluation per Section XI, Appendix L (Reference 36). The results of these inspections and the results of planned research by the EPRI-sponsored Materials Reliability Program will be utilized to assess the appropriate approach for addressing environmentally assisted fatigue of the surge lines during the period of extended operation.

The use of inspections (Option 4) to manage environmentally assisted fatigue during the period of extended operation, requires inspection details such as scope, qualification, method, and frequency be provided to the NRC for review prior to entering the period of extended operation. The NRC review ensures that the inspection intervals for the periodic inspection of the affected locations would be determined by a method accepted by the NRC. Dominion Letter No. 07-0349 provides the required pressurizer surge line weld inspection summary to the NRC.

Implementation of one of the above listed options ensures that the potential effects of the reactor water environment have been addressed for the period of extended operation as required by GSI-190.

18.3.3 Environmental Qualification of Electric Equipment

10 CFR 50.49 requires that each holder of a nuclear power plant operating license establish a program for qualifying safety-related electric equipment. Such a program has been implemented at the station and is invoked by Administrative Procedure. Analyses and tests that qualify safety-related equipment for the period of extended operation are considered TLAAs.

The EQ Program (Section 18.4.1) requires that all electrical equipment important to safety located in a harsh environment shall be managed through the period of extended operation.

18.3.4 Containment Liner Plate

The accumulated fatigue effects of applicable liner loading conditions were evaluated in accordance with Paragraph N-415 of the ASME Boiler and Pressure Vessel Code, Section III, 1968. The evaluation was based on 1000 cycles of operating pressure variations, 4000 cycles of operating temperature variations, and 20 design earthquake cycles. The operating pressure variations are anticipated to be less than 100 and temperature variations are anticipated to be less than 400 for forty years of operation. Extrapolating these anticipated values for sixty years of operation results in 150 pressure variations and 600 temperature variations (Section 15.5.1.8). The number of design cycles was conservatively increased to 1500 cycles of operating pressure variations, 6000 cycles of operating temperature variation, and 30 design earthquake cycles by using a multiplication factor of 1.5, to account for the period of extended operation.

A review of the identified calculations has determined that the increase in the number of cycles due to the period of extended operation is acceptable. Effects of the Containment Type A pressure tests on fatigue of the Containment liner plate have been included in the evaluation. Therefore, the Containment liner is adequate for a 60-year operating period as currently designed. The analyses associated with the Containment liner plate have been revised and projected to be valid for the period of extended operation.

18.3.5 Plant-Specific Time-Limited Aging Analyses

18.3.5.1 Crane Load Cycle Limit

The following are cranes included in license renewal scope and in NUREG-0612 (Reference 11):

- Containment polar cranes
- Containment annulus monorails
- Fuel handling bridge crane
- Spent fuel crane
- Auxiliary Building monorails
- Containment jib cranes

NUREG-0612 requires that the design of heavy load overhead handling systems meet the intent of Crane Manufacturers Association of America, Inc. (CMAA) Specification #70. The crane load cycle provided in CMAA-70 has been identified as a TLAA, with the most limiting number of loading cycles being 100,000.

The most frequently used cranes are spent fuel cranes. Each of these cranes will experience approximately 25,000 cycles of half-load lifts to support the refueling of both units over a 60-year period. In addition, the crane is used to load new fuel into the fuel pool, to perform the various rearrangements required by operations support, to accommodate inspections by fuel vendors, and to load spent fuel casks. In such service, the crane is conservatively expected to make a total of 50,000 half-load lifts in a 60-year period.

Therefore, the analyses associated with crane design, including fatigue, are valid for the period of extended operation.

18.3.5.2 **Reactor Coolant Pump Flywheel**

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to produce high-energy missiles in the unlikely event of failure.

The aging effect of concern is fatigue crack initiation in the flywheel bore keyway. An evaluation of a failure over the period of extended operation has been performed. It demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack growth over a 60-year service life (Reference 12).

The analysis associated with the structural integrity of the reactor coolant pump flywheel has been evaluated and determined to be valid for the period of extended operation.

18.3.5.3 **Leak-Before-Break**

Westinghouse (Westinghouse Owners Group) tested and analyzed crack growth with the goal of eliminating reactor coolant system primary loop pipe breaks from plant design bases. The objective of the investigation was to examine mechanistically, under realistic yet conservative assumptions - whether a postulated crack causing a leak, will grow to become unstable and lead to a full circumferential break when subjected to the worst possible combinations of plant loading.

The detailed evaluation has shown that double-ended breaks of reactor coolant pipes are not credible, and as a result, large LOCA loads on primary system components will not occur. The overall conclusion of the evaluation was, that, under the worst combination of loading, including the effects of safe shutdown earthquake, the crack will not propagate around the circumference and cause a guillotine break. The plant has leakage detection systems that can identify a leak with margin, and provide adequate warning before the crack can grow.

The concept of eliminating piping breaks in reactor coolant system primary loop piping has been termed “leak-before-break” (LBB).

Fatigue crack growth has been identified as an LBB TLAA. The fatigue crack growth analysis has been performed using RCS design transient cycles. The original analysis, however, did not consider the effects of the thermal aging of cast austenitic stainless steel. Dominion has continued to participate in the ongoing NRC/industry program on Alloy 82/182 weld material and

implemented the findings/resolution from this effort, as appropriate (Item 20, Table 18-1). The guidance of MRP-126, Materials Reliability Program: Generic Guidance for Alloy 600 Management, has been used as the basis for a Dominion program to manage aging for inconel alloy welds.

To maintain the plant's LBB design basis, the thermal aging effect for 60 years has been revalidated. The change in the material property has been found to be insignificant. Since the number of design transient cycles will not be exceeded during 60 years of operation, the LBB analysis is projected to be valid for the period of extended operation.

18.3.5.4 **Spent Fuel Pool Liner**

The spent fuel pool liner located in the Fuel Building is needed to prevent a leak to the environment. A design calculation has been identified which documents that the spent fuel pool design meets the general industry criteria. The calculation includes a fatigue analysis to add a further degree of confidence.

The normal thermal cycles occur at each refueling, resulting in 80 cycles for both units in 60 years. Total number of thermal cycles is expected to be 90, which includes normal, upset, emergency, and faulted conditions.

The calculations show that the allowable thermal cycles for spent fuel pool liner for the most severe thermal condition, which includes a loss of cooling, is 95.

Therefore, the existing calculations remain valid for the period of extended operation.

18.3.5.5 **Piping Subsurface Indications**

Calculations have been identified that addressed piping subsurface indications detected by inspections, performed in accordance with ASME Section XI. Section XI provides the acceptance criteria for various flaw orientations, locations and sizes. The calculations determined the number of thermal cycles required for the flaws to reach unacceptable size.

Required cycles for the flaws to reach an unacceptable size are 37,500 or higher.

Since it is expected that the number of the cycles experienced by the piping will not exceed these values for sixty years of operation, the analyses have been determined to remain valid for the period of extended operation.

18.3.5.6 **Reactor Coolant Pump and ASME Code Case N-481**

Periodic volumetric inspections of the welds in the primary loop pump casings in commercial nuclear power plants was required by Section XI of the ASME Boiler and Pressure Vessel Code. Since the reactor coolant pump casings were inspected prior to being placed in service, and no significant mechanisms exist for crack initiation and propagation; it was concluded that the inservice volumetric inspection could be replaced with an acceptable alternate

inspection. In recognition of this conclusion, ASME Code Case N-481, *Alternative Examination Requirements for Cast Austenitic Pump Casings,* provided an alternative to the volumetric inspection requirement. The code case allowed the replacement of volumetric examinations of primary loop pump casings with fracture mechanics based integrity evaluations - Item (d) of the code case - supplemented by specific visual examinations. The analysis has been performed on the reactor coolant pump casing integrity in accordance with the ASME Code Case N-481 requirements. The analysis has been projected to be valid for 60 years. The 2000 Addenda of ASME Section XI removed the volumetric examination of the reactor coolant pump casing welds. Starting with the fourth inspection interval the Code Case is no longer needed.

18.3.6 Exemptions

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on TLAAs, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a TLAA. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a TLAA as defined in 10 CFR 54.3 have been identified.

18.4 TLAA SUPPORTING ACTIVITIES

18.4.1 Environmental Qualification Program

The EQ Program activities are in compliance with the requirements of 10 CFR 50.49. The EQ Program will be continued throughout the period of extended operation. Electrical equipment located in a harsh environment is evaluated for environmental qualification if they are required to function in the conditions that will exist post-accident after being subjected to the normal effects of aging. A harsh environment results from a LOCA or main steam line break inside Containment, high radiation levels due to the post-LOCA effects outside Containment, or high energy line breaks outside Containment.

The EQ Program is applicable to the following groups of components:

- Safety-related electrical equipment that is relied upon to remain functional during and following a design-basis event (DBE)
- Non-safety-related electrical equipment whose failure, under postulated environmental conditions, could prevent accomplishment of safety functions
- Certain post-accident monitoring equipment as described in Regulatory Guide 1.97 (Reference 19).

Guidance regarding environmental qualification was given in NRC Bulletin 79-01B (Reference 20).

The Equipment Qualification Master List provides a listing of electrical equipment that is important to safety and is located in a potentially harsh environment.

Based on the definitions of 10 CFR 54, certain EQ calculations are considered to be TLAAs. As stated in 10 CFR 54.21(c) and in NEI 95-10 (Reference 22), analyses for TLAAs utilize one of the following three options:

i. The analyses remain valid for the period of extended operation,

ii. The analyses have been projected to the end of extended operation, or

iii. The effects of aging will be adequately managed during the period of extended operation.

For purposes of license renewal, EQ components will be evaluated utilizing Option iii in accordance with the EQ Program. EQ concerns for license renewal will consider only those in-scope components that have a qualified lifetime greater than 40 years. Components with a qualified lifetime of less than 40 years already are included in a program of periodic replacement and are not considered TLAAs.

10 CFR 50.49(j) requires that a qualification record be maintained for all equipment covered by the EQ Rule. The qualification process verifies that the equipment is capable of performing its safety function when subjected to various postulated environmental conditions. These conditions include expected ranges of temperature, pressure, humidity, radiation, and accident conditions such as chemical spray and submergence.

The process of qualifying EQ equipment includes analysis, data collection, and data reduction with appropriate assumptions, acceptance criteria and corrective actions.

Qualification Document Reviews (QDRs) provide the basis for qualifying EQ components. The QDRs provide the following information for each piece of equipment that is qualified:

- The performance characteristics required under normal, DBE, and post-DBE conditions.
- The voltage, frequency, load, and other electrical characteristics for which equipment performance can be provided with reasonable assurance.
- The environmental conditions, including temperature, pressure, humidity, radiation, chemical spray, and submergence, at the location where the equipment must function.

18.4.2 Transient Cycle Counting

During normal, upset, and test conditions; reactor coolant system pressure boundary components are subjected to transient temperatures, pressures, and flows, resulting in cyclic changes in internal stresses in the equipment. The cyclic changes in internal stresses cause metal fatigue. Class 1 reactor coolant system components have been designed to withstand a number of

design transients without experiencing fatigue failures during their operating life. The purpose of the Transient Cycle Counting is to record the number of normal, upset, and test events, and their sequence that the station experiences during operation. Design transients are counted to provide reasonable assurance that plant operation does not occur outside the design assumptions.

The Transient Cycle Counting activities are applicable to the reactor coolant system pressure boundary components for which the design analysis assumes a specific number of design transients. A summary of reactor coolant system design transients for which transient cycle counting is performed is listed below:

- Heatups/Cooldowns < 100°F/Hr.
- Step load increase/decrease of 10%
- Large load reduction of 50%
- Loss of load > 15%
- Loss of AC power
- Loss of flow in one loop
- Full power reactor trip
- Inadvertent auxiliary pressurizer spray

The aging effect that is managed by counting transient cycles is cracking due to metal fatigue. The Transient Cycle Counting activities monitor transient cycles that have been experienced by each unit and compare the actual number of cycles to a design assumption. Any concerns related to fatigue are mitigated, as long as the number and magnitude of transient cycles are less than the design assumptions. Approaching a design limit may indicate a situation that is adverse to quality, and would initiate the Corrective Action System. Subsequently, an engineering analysis will determine the design margin remaining, taking credit for the actual magnitude of transients and their sequence to confirm that the allowable factor has not been exceeded. If warranted, component repair or replacement would be initiated.

18.5 REFERENCES

1. Working Draft of the NRC Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants.
2. Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,* March 17, 1988.

3. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,* US Nuclear Regulatory Commission. (Formerly NUREG-75/087)
4. NEI 97-06, *Steam Generator Program Guidelines,* Revision 2, Nuclear Energy Institute.
5. *Steam Generator Management Program: Pressurized Water Reactor, Steam Generator Examination Guidelines*, Revision 7, Electric Power Research Institute.
6. Generic Safety Issue (GSI)-190, *Fatigue Evaluation for Metal Components for 60-year Plant Life,* U.S. Nuclear Regulatory Commission, August 1996.
7. Memorandum from Ashok C. Thadani, to William D. Travers, U.S. Nuclear Regulatory Commission, *Closeout of Generic Safety Issue 190,* December 26, 1999.
8. NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* U.S. Nuclear Regulatory Commission, March 1995.
9. NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,* U.S. Nuclear Regulatory Commission, March 1998.
10. NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* U.S. Nuclear Regulatory Commission, April 1999.
11. NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, July 1980.
12. WCAP-14535A, *Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination,* Westinghouse Electric Corporation, November 1996.
13. American National Standards Institute: ANSI B30.2-1976, *Overhead and Gantry Cranes.*
14. American National Standards Institute: ANSI B30.11-1973, *Monorail Systems and Underhung Cranes.*
15. Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning,* May 2, 1989.
16. NUREG-1344, *Erosion/Corrosion-Induced Pipe Wall Thinning in US Nuclear Power Plants,* April 1, 1989.
17. NSAC-202L, *Recommendation for an Effective Flow Accelerated Corrosion Program,* Electric power Research Institute, Revision 4.
18. Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment,* July 18, 1989 (Supplement 1 dated 4/4/90).
19. U.S. Nuclear Regulatory Commission, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,* Regulatory Guide 1.97, December 1980.

20. IE Bulletin 79-01B, *Environmental Qualification of Class 1E Equipment,* Office of Inspection and Enforcement, January 14, 1980 (Supplement 1 dated 2/29/80; Supplement 2 dated 9/30/80; and Supplement 3 dated 10/24/80).
21. WCAP-15338, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants,* Westinghouse Electric Corporation, March 2000.
22. NEI 95-10, *Industry Guidance for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule,* Revision 2, August 2000.
23. NUREG-1766, *Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station Units 1 and 2,* December 2002.
24. Letter from Leslie N. Hartz (Dominion) to NRC, *Dominion Position Regard Fuse Holders,* Serial No. 02-691, November 4, 2002.
25. Letter from Eugene S. Grecheck (Dominion) to NRC, *Supplemental Information to Support License Renewal,* Serial No. 02-706, December 2, 2002.
26. Letter from David A. Christian (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 01-686, January 16, 2002.
27. Letter from David A. Christian (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 02-163, May 22, 2002.
28. Letter from David A. Christian (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 01-685, January 4, 2002.
29. Letter from David A. Christian (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 01-647, November 30, 2001.
30. Letter from David A. Christian (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 01-514, September 27, 2001.
31. Letter from David A. Christian (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 01-732, February 5, 2002.
32. Letter from David A. Christian (Dominion) to NRC, *License Renewal Applications - Submittal,* Serial No. 01-282, May 29, 2001.
33. Letter from Leslie N. Hartz (Dominion) to NRC, *Request for Additional Information License Renewal Applications,* Serial No. 02-332A, October 1, 2002.
34. Letter from Leslie N. Hartz (Dominion) to NRC, *Engineering Evaluation Results and Closure of Commitment for Management of Environmentally-Assisted Fatigue in Accumulator and Charging Nozzles,* Serial No. 03-616, January 6, 2004.
35. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev 0)*. EPRI, Palo Alto, CA: 2008. 1016596.

36. Structural Integrity Associates, Report No. 1601007.401, Flaw Tolerance Evaluation of Surry Hot Leg Surge Line Nozzles Using ASME Code Section XI, Appendix L, September 2017.

Table 18-1
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
1	Develop and implement inspection program for buried piping and valves.	One-time between years 30-40. Additional inspections based on results. (Complete)	Table B4.0[b], RAI B2.1.1-1	32 30
2	Add pressurizer surge line to Augmented Inspection Program.	Prior to Period of Extended Operation (Complete)	Table B4.0[b], RAI 4.3-7	32 26
3	Add core barrel hold-down spring to Augmented Inspection Program.	Prior to Period of Extended Operation (Complete)	Table B4.0[b]	32
4	Expand scope of Civil Engineering Structural Inspection to cover License Renewal requirements.	Prior to Period of Extended Operation (Complete)	Table B4.0[b]	32
5	Revise plant documents to use inspection opportunities when inaccessible areas become accessible during work activities.	Prior to Period of Extended Operation (Complete)	Table B4.0[b]	32
6	Incorporate NFPA-25, Section 2-3.1.1 for sprinklers.	Prior to year 50. If testing used, repeat every 10 years. (Complete)	Table B4.0[b]	32
7	Develop inspection criteria for non-ASME supports and doors.	Prior to Period of Extended Operation (Complete)	Table B4.0[b]	32
8	Develop procedural guidance for inspection criteria that puts focus on aging effects.	Prior to Period of Extended Operation (Complete)	Table B4.0[b]	32
9	Develop and implement inspection program for infrequently accessed areas.	One-time between years 30-40. Additional inspections based on results. (Complete)	Table B4.0[b], RAI 3.5-1, RAI 3.5.8-1	32 29 31

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For Surry Unit 1, the Period of Extended Operation is from May 26, 2012 to May 25, 2032 and for Surry Unit 2, from January 30, 2013 to January 29, 2033.

b. Table B4.0 is the table of Licensee Followup Actions located in the License Renewal Application for Surry (Reference 32).

Table 18-1 (CONTINUED)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
10	Develop and implement inspection program for tanks.	One-time between years 30-40. Additional inspections based on results. (Complete)	Table B4.0[b]	32
11	Follow industry activities related to failure mechanisms for small-bore piping. Evaluate changes to inspection activities based on industry recommendations.	On-going activity (Complete)	Table B4.0[b], RAI 3.1.1.2-2	32 29
12	Follow industry activities related to core support lugs. Evaluate need to enhance inspection activities based on industry recommendations.	Industry activities during the 3.5 years since the renewed licenses were issued have resulted in no new guidance for the inspection of core support lugs. The lugs were successfully inspected for both Surry units during the 10-year ASME inservice inspections in 2004 and 2005. (Complete)	Table B4.0[b]	32
13	Inspect representative sections of polar crane box girders.	One-time between years 30-40. Additional inspections based on results. (Complete)	Table B4.0[b]	32
14	Follow industry activities related to reactor vessel internals issues such as void swelling, thermal and neutron embrittlement, etc. Evaluate industry recommendations.	One-time inspection between years 30-40 on most susceptible single unit (Surry or North Anna). Additional inspections based on results. (Complete)	Table B4.0[b]	32

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For Surry Unit 1, the Period of Extended Operation is from May 26, 2012 to May 25, 2032 and for Surry Unit 2, from January 30, 2013 to January 29, 2033.

b. Table B4.0 is the table of Licensee Followup Actions located in the License Renewal Application for Surry (Reference 32).

Table 18-1 (CONTINUED)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
15	Implement changes into procedures to assure consistent inspection of components for aging effects during work activities.	Prior to Period of Extended Operation (Complete)	Table B4.0[b]	32
16	Incorporate groundwater monitoring into the civil engineering structural monitoring program. Consider groundwater chemistry in engineering evaluations of deficiencies.	Prior to Period of Extended Operation (Complete)	RAI 3.5-2	29
17	Incorporate management of concrete aging into the civil structural monitoring program and the infrequently accessed area inspection programs.	Prior to Period of Extended Operation (Complete)	RAI 3.5-7	29
18	Incorporate management of elastomers into the work control activities.	Prior to Period of Extended Operation (Complete)	RAI 3.5.6-4, RAI B2.2.19-3	31 29
19	Develop and implement inspection program for Non-EQ cables.	One-time between years 30-40. Additional inspections every 10 years thereafter. (Complete)	RAI 3.6.2-1	29
20	Follow industry activities related to Alloy 82/182 weld material. Implement activities based on industry recommendations, as appropriate.	On-going activity. (Complete)	RAI 4.7.3-1	28
21	Inspectors credited in the Work Control Process will be VT qualified.	Prior to Period of Extended Operation (Complete)	RAI B2.2.19-1	29

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For Surry Unit 1, the Period of Extended Operation is from May 26, 2012 to May 25, 2032 and for Surry Unit 2, from January 30, 2013 to January 29, 2033.
b. Table B4.0 is the table of Licensee Followup Actions located in the License Renewal Application for Surry (Reference 32).

Table 18-1 (CONTINUED)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
22	Perform audit of work control inspections to ensure representation by all in-scope license renewal systems and to determine need for supplemental inspections.	Prior to Period of Extended Operation and every 10 years thereafter. Supplemental inspections within 5 years of audit. (Complete)	RAI B2.2.19-3	29
23	Not Applicable for Surry.	N/A	N/A	N/A
24	Provide inspection details for pressurizer surge line inspections to the NRC for review and approval.	Prior to Period of Extended Operation (Complete)	RAI 4.3-7, RAI 4.3-6	26
25	Provide inspection details for safety injection and charging line inspections to the NRC for review and approval if analysis is not successful in reducing the Cumulative Usage Factor (CUF) below 1.0. The analysis described in Reference 34 confirms the CUFs to be below 1.0, and no inspections are required.	Prior to Period of Extended Operation (Complete)	RAI 4.3-6,	26, 33
26	Address NRC staff final guidance regarding fuse holders when issued.	When issued or prior to Period of Extended Operation, whichever is later. (Complete)	See Reference	24
27	Develop and implement a program to control water intrusion into manholes at Surry.	Prior to Period of Extended Operation (Complete)	RAI 3.6.2-1	29

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For Surry Unit 1, the Period of Extended Operation is from May 26, 2012 to May 25, 2032 and for Surry Unit 2, from January 30, 2013 to January 29, 2033.

b. Table B4.0 is the table of Licensee Followup Actions located in the License Renewal Application for Surry (Reference 32).

Table 18-1 (CONTINUED)
LICENSE RENEWAL COMMITMENTS

Item	Commitment	Schedule[a]	Source	Ref.
28	Revise procedures for groundwater testing to account for possible seasonal variations.	Prior to Period of Extended Operation (Complete)	See Reference	25
29	Inspect similar material/environment components, both within the system and outside the system, if aging identified in a location within a system cannot be explained by environmental/operational conditions at that specific location.	Prior to Period of Extended Operation (Complete)	RAI B2.2.19-3	29
30	Supplement the NFPA pressure and flowrate testing credited in each LRA as part of the fire protection program activity with the work control process activity in order to manage aging effects for the fire protection system piping.	Prior to Period of Extended Operation (Complete)	RAI B2.2.7-2	27, 29

a. The Period of Extended operation is the period of 20 years beyond the expiration date of each unit's original operating license. For Surry Unit 1, the Period of Extended Operation is from May 26, 2012 to May 25, 2032 and for Surry Unit 2, from January 30, 2013 to January 29, 2033.

b. Table B4.0 is the table of Licensee Followup Actions located in the License Renewal Application for Surry (Reference 32).

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