15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the reactor coolant system (loss of coolant accident).

(2) Major secondary system pipe ruptures.

(3) Steam generator tube rupture.

(4) Single reactor coolant pump locked rotor.

(5) Fuel handling accident.

(6) Rupture of a control rod drive mechanism housing (rod cluster control assembly ejection).

The analysis of thyroid and whole body doses, resulting from events leading to fission product release, appears in Section 15.5. The fission product inventories which form a basis for these calculations are presented in Chapter 11 and Section 15.1. Section 15.5 also includes the discussion of systems interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Loss-of-coolant accidents (LOCAs) are accidents that would result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. LOCAs could occur from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (RCS). Large breaks are defined as breaks in the reactor coolant pressure boundary having a cross-sectional area greater than or equal to 1.0 ft^2. Reference [34] documents this criterion. The large break LOCA analysis is performed to demonstrate compliance with the 10 CFR 50.46 acceptance criteria[35] for emergency core cooling systems for light water nuclear power reactors.

A large break LOCA is the postulated double-ended guillotine or split rupture of one of the RCS primary coolant pipes.

The boundary considered for loss of coolant accidents is the RCS or any line connected to the system up to the first closed valve.

The sequence of events following a limiting large break LOCA transient is presented in Table 15.4-17. Before the break occurs, the RCS is assumed to be operating normally at full power in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. A large beak is assumed to open almost instantaneously in one of the main RCS pipes. Calculations have demonstrated that the most severe transient results occur for a break in the cold leg between the pump and the reactor vessel.

Immediately following the cold leg break, a rapid system depressurization occurs along with a core flow reversal due to a high discharge of subcooled fluid into the broken cold leg and out the break. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up, while the core power shuts down due to voiding in the core. The hot water in the core, upper plenum, and upper head flashes to steam, and subsequently the cooler water in the lower plenum and downcomer begins to flash. Once the system has depressurized to the accumulator pressure, the accumulators begin to inject cold borated water into the intact cold legs. During the blowdown period a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. The bypass period ends as the system pressure continues to decrease and approaches the containment pressure, resulting in reduced break flow and consequently reduced core flow.

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water. This phase continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

During the reflood period, the core flow is oscillatory as ECCS water periodically rewets and quenches the hot fuel cladding which generates steam and causes system repressurization. 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 the break. This flow path resistance is overcome by the downcomer water elevation head which provides the gravity driven reflood force. 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.

15.4.1.1 Thermal Analysis

15.4.1.1.1 Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest reactor coolant system pipe. The reactor core and internals together are designed so that the reactor can be safely shutdown and the essential heat transfer geometry of the core preserved following the accident. The current internals is of the upflow barrel/baffle design. The ECCS, even when operating during the injection mode with the most limiting single active failure, is designed to meet the acceptance criteria.

15.4.1.1.2 Method of Thermal Analysis

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in Appendix K of 10 CFR 50.46, both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted Peak Clad Temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1) Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2) Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without loss in safety to the public. It was confirmed that some plants were being restricted in operating flexibility by the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 [50]. The SECY-83-472 [50] represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1998, 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. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. 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[44].

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 (NUREG/CR-5249[45]). 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 revised10 CFR 50.46 rules was developed by Westinghouse with support of EPRI and Consolidated Edison and has been approved by the NRC (WCAP-12945-P-A [46]).

More recently, Westinghouse developed an alternative methodology called ASTRUM, which stands for Automated Statistical TReament of Uncertainty Method (WCAP-16009-P-A [49]). This method is still based on the CQD methodology and follows the steps in the CSAU methodology (NUREG/CR-5249 [45]). However, 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 [49].

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 [49], as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A [49] was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and CQD uncertainty approach per section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this topical report.

The Watts Bar 2 ASTRUM LBLOCA uses a plant-specific adaptation of the ASTRUM methodology that includes explicit modeling of fuel thermal conductivity degradation (TCD), as well as a larger sampling range for rod internal pressure (RIP) uncertainty.

The methods used in the application of WCOBRA /TRAC to the large break LOCA with ASTRUM are described in WCAP-12945-P-A [46] and WCAP-16009-P-A [49]. 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.

WCAP-16009-P-A [49] states that the ASTRUM methodology is based on the frozen code version WCOBRA/TRAC MOD7A, Revision 6. WCOBRA/TRAC MOD7A, Revision 8-T2 was used for the execution of ASTRUM Uncertainty Studies for Watts

Bar Unit 2. The confirmatory analysis (paragraph "2) Determination of Plant Operating Conditions'') were executed with WCOBRA/TRAC MOD7A Revision 7.

The Nuclear Regulatory Commission (NRC) approved Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) ASTRUM methodology [49] is based on the PAD 4.0 fuel performance code [51]. PAD 4.0 was licensed without explicitly considering fuel thermal conductivity degradation (TCD) with burnup. Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperatures (pellet radial average temperature) as well as other fuel performance related effects beyond beginning-of-life. Since PAD provides input to the large-break LOCA analysis, this will tend to increase the stored energy at the beginning of the simulated large-break LOCA event. This in turn leads to an increase in Peak Cladding Temperature (PCT) if there is no provision to credit off-setting effects. In addition, a different fuel thermal conductivity model in WCOBRA/TRAC and HOTSPOT was used to more accurately model the fuel temperature profile when accounting for TCD.

In order to mitigate the impact of the increasing effect of pellet TCD with burnup, the large-break LOCA evaluation of second/third Cycle fuel utilized reduced peaking factors from those shown directly in FSAR Table 15.4-19. The reduced peaking factors are limited to the following application: Burndown credit for the hot rod and hot assembly is taken for higher burnup fuel in the second/third cycle of operation. The Watts Bar Unit 2 peaking factor values utilized in this analysis are shown in Table 15.4-24. Note that the beginning to middle of life values are retained at their direct Table 15.4-19 values.

It should be noted that evaluation of fuel in its second/third cycle of irradiation is beyond the first cycle considered in the approved ASTRUM Evaluation Model (EM), but was considered in the analysis when explicitly modeling TCD to demonstrate that conformance to the acceptance criteria is met for the second/third cycle fuel.

In addition to the standard uncertainty calculations, the Watts Bar 2 LBLOCA analysis sampled a larger rod internal pressure (RIP) uncertainty than originally included in the ASTRUM methodology [49]. It was discovered that the as-approved sampling range did not bound the plant-specific rod internal pressure uncertainties for Watts Bar 2. Therefore, the approved sampling range was expanded to bound the Watts Bar 2 plant-specific data.

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.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ. Dividing the liquid phase into two fields is a convenient and physically accurate way of handling flows where the liquid can appear in both film and droplet form. The droplet field permits more accurate modeling of thermal-hydraulic phenomena such as entrainment, de-entrainment, fallback, liquid pooling, and flooding.

WCOBRA/TRAC also features a two-phase, one-dimensional hydrodynamic formulation. In this model, the effect of a phase slip is modeled indirectly via a constitutive relationship which provides the phase relative velocity as a function of fluid conditions. Separate mass and energy conservation equations exist for the two-phase mixture and for the vapor.

The reactor vessel is modeled with the three-dimensional, three-field model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional model.

All geometries modeled using the three-dimensional model are represented as a matrix of cells. The number of mesh cells used depends on the degree of detail required to resolve the flow field, the phenomena being modeled, and practical restrictions such as computing costs and core storage limitations.

The equations for the flow field in the three-dimensional model are solved using a staggered difference scheme on the Eulerian mesh. The velocities are obtained at mesh cell faces, and the state variables (e.g., pressure, density, enthalpy, and phasic volume fractions) are obtained at the cell center. This cell is the control volume for the scalar continuity and energy equations. The momentum equations are solved on a staggered mesh with the momentum cell centered on the scalar cell face.

The basic building block for the mesh is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors.

One-dimensional components are connected to the vessel. The basic scheme used also employs the staggered mesh cell. Special purpose components exist to model specific components such as the steam generator and pump.

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 LOTIC-2 [5] code 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 Peak Cladding Temperature (PCT), Maximum Local Oxidation (MLO), 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 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. The confirmatory configuration analysis was performed previous to the ASTRUM uncertainty calculations prior to the identification of the TCD issue and associated PAD data. However, as no miscellaneous plant configuration changes were introduced, and the effects of TCD are minimal for the confirmatory analysis, the limiting plant configuration (Referred to as the Reference Transient) was judged to remain the same.

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 WCAP-16009-P-A [49]. The determination of the PCT uncertainty, MLO 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,MLO,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 also 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 MLO and CWO. The highest rank of PCT, MLO, 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, this range may be the desired range or may be narrower for some parameters to gain additional margin.

15.4.1.1.3 Containment Analysis

The containment pressure analysis is performed with the LOTIC-2 [5] code. Transient mass and energy releases for input to the LOTIC-2 model are obtained from the WCOBRA/TRAC code. The transient pressure computed by the LOTIC-2 code is then used in WCOBRA/TRAC for the purpose of supplying a backpressure at the break plane while computing the reflood transient. The containment pressure transients and associated parameters were computed by LOTIC-2 and are presented in Figures 15.4-40b through 15.4-40g. The data used to model the containment for the analysis is presented in Tables15.4-14 and 15.4-15. Mass and energy release rates to containment can be found in Table 15.4-16. The Table 15.4-16 mass and energy releases are taken from the ‘Reference Transient’ case of Section 15.4.1.1.2, which did not include the fuel TCD modeling. The conservatively low containment backpressure from this LOTIC-2 study is bounding since the core stored energy increases when explicitly modeling fuel TCD, which would tend to increase energy released through the break and hence increase the containment pressure.

The impact of purging on the calculated containment pressure was addressed by performing a calculation to obtain the amount of mass which exits through two available purge lines during the initial portion of a postulated LOCA transient. The maximum air loss was calculated using the transient mass distribution (TMD) computer code model, which is described in Section 6.2.1.3.4, to be 1160 lbm. The containment

pressure calculations account for a loss of 1160 lbm of air after initiation of the accident through modifying the compression ratio input to the LOTIC-2 code.

15.4.1.1.4 Results of Large Break Limiting Transient

The Watts Bar Unit 2 PCT and MLO/CWO transients are double ended cold leg guillotine breaks with an effective break area of 1.911, and 2.0968 respectively (note that the limiting MLO and CWO arise from the same case), which analyzes conditions that fall within those listed in Table 15.4-19. Traditionally, cold leg breaks have been limiting for large break LOCA. Analysis experience indicates that this break location most likely causes conditions that result in flow stagnation to occur in the core. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (WCAP-12945-P-A[46]).

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cool down 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 Figure 15.4-41 through 15.4-55. (The limiting PCT 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 (Figure 15.4-42). The region 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 15.4-41) 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 15.4-47 and 15.4-51). 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 15.4-44 shows the void fraction for one intact loop pump and the broken loop pump. This 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 the core vapor flow (Figures 15.4-45 and 15.4-46) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure (Figure 15.4-43), the accumulators begin to inject cold borated water into the intact cold legs (Figure 15.4-48). 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, is 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 15.4-52).

4) Refill Phase:

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water (Figure 15.4-48). 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) Early Reflood Phase:

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 15.4-50). 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 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 (Figure 15.4-49) 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 and the early reflood period is terminated.

6) Late Reflood Phase:

The late reflood phase is characterized by 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 metal, reduces the subcooling of water in the lower plenum and downcomer. Figure 15.4-54 illustrates the reduction in lower plenum subcooling.

The saturation temperature is dictated by the containment backpressure. For WBN, which has a low containment pressure after the LOCA, boiling does occur and has a significant effect on the gravity reflood. Vapor generated in the downcomer reduces the driving head which results in a reduced core reflood rate. The top core elevations experience a second reflood heatup, which exceeds the first (Figure 15.4-41, HOTSPOT result).

15.4.1.1.5 POST ANALYSIS OF RECORD EVALUATIONS

An evaluation of IFBA fuel including the effects of pellet TCD was performed, and shows that IFBA fuel is limiting for MLO but not for PCT. The AOR PCT and MLO results in Tables 15.4-18a and 15.4-18b reflect the higher results of IFBA/non-IFBA.

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 15.4-18a for the large break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200 °F.

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 FSAR until 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 [36], Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting. TVA provides the NRC with annual and 30-day reports, as applicable, for Watts Bar Unit 2. TVA intends to provide future reports required by 10 CFR 50,46 consistent with the approach described in WCAP-13451.

15.4.1.1.6 CONCLUSIONS - THERMAL ANALYSIS

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 at the 95-percent confidence level. Figure 15.4-41 shows the predicted

HOTSPOT cladding temperature transient at the PCT location and the WCOBRA/TRAC PCT transient, both for the limiting PCT case. The HOTSPOT PCT plot includes local uncertainties applied to the Hot Rod , whereas the WCOBRA/TRAC PCT plot does not account for any local uncertainties. Since the resulting HOTSPOT PCT for the limiting case is 1766°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F, is demonstrated. The results are shown in Table 15.4-18b.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO for the limiting case is 1.99 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Maximum Local Oxidation of the cladding less than 17 percent", is demonstrated. The results are shown in Table 15.4-18b.

(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 0.08 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 the HAR value. Since the resulting CWO is 0.08 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core Wide Oxidation less than 1 percent", is demonstrated.

(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 the fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology (WCAP-12945-P-A [46]) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation has not been calculated to occur for Watts Bar Unit 2. Therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that the 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. While WCOBRA/TRAC is typically not run past full core quench, all base calculations are run well past PCT turnaround and past the point where increasing vessel inventories are calculated. The conditions at the end of the WCOBRA/TRAC calculations indicate that the transition to long term cooling is underway even before the entire core is quenched.

Based on the ASTRUM Analysis results (Table 15.4-18b), it is concluded that Watts Bar Unit 2 maintains a margin of safety to the limits prescribed by 10 CFR 50.46.

15.4.1.1.7 PLANT OPERATING RANGE

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. Tables 15.4-19 summarizes the operating ranges as defined for the proposed operating conditions which are supported by the Best-Estimate LBLOCA analysis for Watts Bar Unit 2. Tables 15.4-14 and 15.4-15 summarize 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 (Tavg).

15.4.1.2 Hydrogen Production and Accumulation

Pursuant to NRC final rule as defined in 10 CFR 50.44 and Regulatory Guide 1.7, the new definition of design-basis LOCA hydrogen release eliminates requirements for hydrogen control systems for mitigation of releases. “All PWRs with ice condenser type containments must have the capability to control combustible gas generated from metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity. The deliberate ignition systems provided to meet this existing combustible gas source term are capable of safely accommodating even greater amounts of combustible gas associated with even more severe core melt sequences that fail the reactor vessel and involve molten core-concrete interaction. Deliberate ignition systems, if available, generally consume the combustible gas before it reaches concentrations that can be detrimental to containment integrity.” On the basis of this definition, no further analysis is required to support events considered to be outside the design basis. Deliberate ignition systems are described in FSAR Section 6.2.5

15.4.2 Major Secondary System Pipe Rupture

15.4.2.1 Major Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result 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 coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The analysis of a main steam line rupture is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck RCCA with or without offsite power and assuming a single failure in the engineered safeguards, the core remains in place and intact. Radiation doses are not expected to exceed the guidelines of 10 CFR 100.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no violation of the DNB design basis occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

The following functions provide the necessary protection for a steam line rupture:

(1) Safety injection system actuation from any of the following:

(a) Two out of three low pressurizer pressure signals.

(b) Two out of three high containment pressure signals.

(c) Two out of three low steamline pressure signals in any steamline.

(2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

(3) Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. A safety injection signal will rapidly close all feedwater control valves and main feedwater isolation valves, and trip the main feedwater pumps, condensate booster pumps, condensate demineralizer pump, and motor-operated standby feedwater pump if operating.

(4) Trip of the fast acting steam line stop valves (main steam isolation valves) (designed to close in less than 6 seconds) on:

(a) Two out of four high-high containment pressure signals.

(b) Two out of three low steamline pressure signals in any steamline.

(c) Two out of three high negative steamline pressure rate signals in any steamline.

Fast-acting isolation valves are provided in each steam line that will fully close within 6 seconds after a steamline isolation signal setpoint is reached. The time delay for actuation of the low steamline pressure safety injection actuation signal, high negative steamline pressure rate signal, high-high containment pressure signal, and manual block of the low steamline pressure safety injection actuation signal must be within 2 seconds after initiation. This, along with the main steam isolation time of approximately 6 seconds,shall not exceed a 8 second total response time for this action in the safety analysis for this event. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blowdown even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe and thus the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.4-6 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since it will vary depending upon postulated break location and details of initial conditions. Design criteria and methods of protection of safety related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.4.2.1.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the 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 LOFTRAN[11] Code has been used.

(2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, VIPRE-01[30], has been used to determine if the calculated DNBR occurs for the core conditions computed in Item 1 above.

The following conditions were assumed to exist at the time of a main steam line break accident.

(1) End-of-life shut down margin at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

(2) The negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1110 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.2-40. The effect of power generation in the core on overall reactivity is shown in Figure 15.4-9. The parameters used to determine the radioactivity releases for the steamline break are given in Table 15.5-16.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the statepoints shown on Table 15.4-7. These core analyses considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for all statepoints. The limiting statepoint is presented in Table 15.4-7. These results verified conservatism, i.e., underproduction of negative reactivity feedback from power generation.

(3) Minimum capability for injection of concentrated boric acid which is bounding for higher boric acid solution corresponding to the most restrictive single failure in the safety injection system. The emergency core cooling system consists of three systems: 1) the passive accumulators (at 1900 ppm), 2) the residual heat removal system, and 3) the safety injection system (at 2000 ppm).

The actual modeling of the safety injection system in LOFTRAN is described in Reference [11] and reflects injection as a function of RCS pressure versus flow including RCP seal injection, excluding centrifugal charging pump miniflow, and with no spilling lines. This injection analysis result is bounded

when using the minimum composite pump curve (degraded by 5% of design head) as shown in Figure 6.3-4. This corresponds to the flow delivered by one charging pump and one safety injection pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the RWST prior to the delivery of concentrated boric acid to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept, of course, before the 2000 ppm (which is bounding for higher boric acid concentrations) reaches the core. This delay, described above is inherently included in the modeling.

In cases where offsite power is not available, a 10-second delay is assumed to start the diesels and then begin loading the necessary safety injection equipment sequentially onto them.

This assumption results in additional conservatism in the analysis, which adds the 10 seconds to the 27 seconds assumed for valve alignment in the offsite power available case for a total of 37 seconds.

(4) Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

(5) Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location would have the same effect on the Nuclear Steam Supply System (NSSS) as the 1.4 square foot break. The following cases have been considered in determining the core power and reactor coolant system transients:

(a) Complete severance of a pipe, with the plant initially at no load conditions, full reactor coolant flow with offsite power available.

(b) Case a above with loss of offsite power. Loss of offsite power results in coolant pump coastdown.

(6) Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly-during the return to power phase following the steam line break.

The limiting statepoints for the two cases are presented in Table 15.4-7.

Both the cases above assume initial hot shutdown conditions at time zero since this represents the most limiting initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. 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 steam line break before the no load conditions of RCS 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 which assumes no load condition at time zero.

However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the RCS cooldown are greater for steam line breaks occurring from no load conditions.

(7) In computing the steam flow during a steam line break, the Moody Curve[9] for fl/D = 0 is used.

(8) A steam generator tube plugging level of 10% is assumed.

(9) A thermal design flowrate of 372,400 gpm is used which accounts for the 10% steam generator tube plugging level and instrumentation uncertainty.

Results

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

Core Power and RCS Transient

Figures 15.4-11a through 15.4-11c show the RCS transient and core response following a main steam line rupture (complete severance of a pipe) at initial no load condition (Case a). Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by low steamline pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high-high containment pressure or low steam line pressure signals. Even with the failure of one valve, release is limited by isolation valve closure for the other steam generators while the one generator blows down. The main steamline isolation valves are designed to be fully closed in less than 6 seconds from receipt of a closure signal.

As shown in Figure 15.4-11a the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) shortly after boron solution at 2000 ppm (which is bounding for higher boric acid concentrations) enters the reactor coolant system. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by the water flowing in the reactor coolant system prior to entering the reactor core. The concentration after mixing depends upon 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.

It should be noted that the safety injection accumulators are actuated in Case (a) due to low RCS pressure (Figure 15.4-11b). Once the accumulators actuate, 2400 ppm boron is delivered to the core and the transient is terminated before a significant return to power is achieved. Once the transient is terminated and the plant is stabilized, emergency operating procedures may be followed to recover from the MSLB event.

Figures 15.4-12a through 15.4-12c show the responses of the salient parameters for Case b which corresponds to the case discussed above with additional loss of offsite power at the time the safety injection signal is generated. The injection of borated water is conservatively delayed to 37 seconds based on the assumed 10 second diesel generator delay time plus the 27 seconds associated with the valve lineup for the offsite power available case (Case a). In this case criticality is achieved later 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. For both these cases the peak power remains well below the nominal full power value.

Unlike Case a, Case b does not result in the actuation of the safety injection accumulators. Therefore, due to the fact that less boric acid solution is delivered to the core, Case b results in a more limiting return to power than Case a.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, the remaining steam generators are still available for 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 via the steam line safety valves.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures call for operator action to limit RCS pressure and pressurizer level by terminating safety injection flow, and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following safety injection actuation.

Margin to Critical Heat Flux

A DNB analysis was performed for the limiting case. The limiting statepoints are presented in Table 15.4-7. It was found that all cases had a minimum DNBR greater than the limit value.

15.4.2.1.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. In addition, the pressure differential across the steam generator tubes that has been calculated for a postulated main feedwater line break is more limiting (i.e., dictates a minimum tube wall thickness) than the pressure differential for a postulated main steam line break. Therefore, steam generator tube rupture is not expected to occur (see Section 4.19.7.6 of Reference [34]).

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded in the criterion, the above analysis, in fact, shows that no violation of the DNB design basis occurs for any rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

If it is assumed that there is leakage from the reactor coolant system to the secondary system in the steam generators and that offsite power is lost following the steam line break, radioactivity will be released to the atmosphere through the relief or safety valves. Environmental consequences of a postulated steam line break are addressed in Section 15.5.4.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedline check valve would affect the nuclear steam supply system only as a loss of normal feedwater.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break), or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in Section 15.4.2.1. Therefore, only the reactor coolant system heatup effects are evaluated for a feedline rupture.

A feedline rupture reduces the ability to remove heat generated by the core from the reactor coolant system because of the following reasons:

(1) Feedwater to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

(2) Liquid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

(3) The break may be large enough to prevent the addition of any main feedwater after trip.

An auxiliary feedwater system is provided to assure that adequate feedwater is available such that:

(1) No substantial overpressurization of the reactor coolant system occurs; and

(2) Liquid in the reactor coolant system is sufficient to cover the reactor core at all times.

The following provides the necessary protection for a main feedwater rupture:

(1) A reactor trip on any of the following conditions:

(a) High pressurizer pressure

(b) Overtemperature ΔT

(c) Low-low steam generator water level in one or more steam generators

(d) Safety injection signals from any of the following:

(i) Low steamline pressure

(ii) Low pressurizer pressure

(iii) High containment pressure

(2) An auxiliary feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal.

15.4.2.2.2 Analysis of Effects and Consequences

The discussion of the analysis for a main feedwater break inside primary containment presented below is based on a reactor trip generated by steam generator low-low water level. Evaluations that were performed using the MONSTER[37] Code show a high containment pressure signal is generated in less than 1.0 second. In the analysis presented below, steam generator level decreases to its trip setpoint in 37.1 seconds. Thus, the following analysis is conservative and is being retained although containment pressure is the signal that will actually be used to generate a reactor trip for this event.

Method of Analysis

A detailed analysis using the LOFTRAN[11] Code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics, reactor coolant system including natural circulation, pressurizer, steam generators and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Two cases are analyzed. One case assumes that offsite electrical power is maintained throughout the transient. Another case assumes the loss of offsite electrical power at the time of reactor trip, and RCS flow decreases to natural circulation. Both cases assume a double-ended rupture of the largest feedwater pipe at full power. Major assumptions used in the analysis are as follows:

(1) The plant is initially operating at full power including applicable uncertainty.

(2) Initial reactor coolant average temperature is 6.0°F above the nominal value (bounds an instrument uncertainty of ±5°F and instrument bias of -1°F), and the initial pressurizer pressure is 50 psi below its nominal value (bounds an instrument uncertainty of ± 50 psi and instrument bias of -20 psi).

(3) The pressurizer power-operated relief valves and the safety relief valves are assumed to function. No credit is taken for pressurizer spray. Initial pressurizer level is at the nominal programmed value plus 8% uncertainty.

(4) No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

- High pressurizer pressure
- Overtemperature ΔT
- High pressurizer level
- High containment pressure (For breaks outside containment)

(5) Main feedwater to all steam generators is assumed to stop at the time the break occurs (all main feedwater spills out through the break).

(6) The initial blowdown quality from the affected steam generator is assumed to be 15% due to effects as the inventory passes back through the preheater. At the time of reactor trip, the frothing and oscillations within the steam generator are reduced and saturated liquid (0% quality) is blown out the break until all the liquid is gone. Subsequent blowdown, prior to the time of steamline isolation, is assumed to be saturated liquid (100% quality).

(7) No credit is taken for the low-low water level trip on the affected steam generator until the steam generator level reaches 0% of the narrow range span. This assumption minimizes the steam generator fluid inventory at the time of trip, and thereby maximizes the resultant heatup of the reactor coolant.

(8) A double-ended break area of 0.223 ft^2 is assumed.

(9) No credit is taken for heat energy deposited in reactor coolant system metal during the RCS heatup.

(10) No credit is taken for charging or letdown.

(11) Steam generator heat transfer area is assumed to decrease as the shellside liquid inventory decreases.

(12) The core residual heat generation is based on the 1979 version of ANS 5.1 [Ref. 33] based upon long term operation at the initial power level. The decay of U-238 capture products is included as an integral part of this expression.

(13) The auxiliary feedwater is actuated by the low-low steam generator water level signal.

The analysis addresses either TDAFWP failure with and without offsite power or MDAFWP failure with and without offsite power. The assumptions for the limiting case (MDAFWP) failure) are as follows:

a. The motor driven pump which feeds two intact steam generators is assumed to fail.

b. After steamline isolation, all flow from all pumps is initially assumed "lost" to the faulted steam generator. After the faulted steam generator pressure drops below 360 psig, a valve automatically restricts MD pump flow to the faulted steam generator, thus allowing some delivery (assumed to be 60 gpm) to an intact loop.

c. Operator action to isolate the affected steam generator is assumed to occur no later than 12 minutes from the time of the first low steam generator level signal.

d. After isolation of the faulted steam generator, the TDAFWP supplies flow to the 3 remaining steam generators while the operating MD pump supplies flow to 1 steam generator.

A 60 second delay was assumed following the low-low steam generator water level signal to allow time for startup of the emergency diesel generators and the auxiliary feedwater pumps.

Results

Figures 15.4-13a, 15.4-13b, and 15.4-13c show the calculated plant parameters following a feedline rupture for the case with offsite power. Figures 15.4-14a, 15.4-14b, and 15.4-14c show the calculated plant parameters following a feedline rupture with loss of offsite power. The calculated sequence of events for both cases analyzed is presented in Table 15.4-9.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented in the figures show that pressures in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure increases until reactor trip occurs on low-low steam generator water level. Pressure then decreases, due to the loss of heat input, until steamline isolation occurs. Coolant expansion occurs due to reduced heat transfer capability in the steam generators. The pressurizer relief valves open to maintain primary pressure at an acceptable value. The calculated relief rates are within the relief capacity of the pressurizer relief valves. Addition of the safety injection flow aids in cooling down the primary side and helps to ensure that sufficient fluid exists to keep the core covered with water.

The reactor core remains covered with water throughout the transient and the auxiliary feedwater system flow capacity is sufficient to preclude bulk boiling in the RCS throughout the transient.

15.4.2.2.3 Conclusions

Results of the analysis show that for the postulated feedline rupture, the assumed auxiliary feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to prevent the water level in the RCS from dropping to the top of the core.

15.4.3 Steam Generator Tube Rupture

15.4.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to transfer of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of radioactivity to the atmosphere takes place via the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

The steam generator tube material is Inconel-600 and is a highly ductile material; thus, it is considered that the assumption of a complete severance of a tube is somewhat conservative. 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 an accumulation of minor leaks which exceed the

limits established in the Technical Specifications is not permitted during the unit operation.

The operator is expected to readily determine that a steam generator tube rupture (SGTR) has occurred, identify and isolate the faulty steam generator on a restricted time scale in order to complete the required recovery actions to stabilize the plant, minimize contamination of the secondary system, and ensure termination of radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

(1) Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch alarm as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that steam generator from the primary side.

(2) Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or by overtemperature ΔT. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by low-low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

(3) The steam generator blowdown liquid monitor, the condenser vacuum exhaust radiation monitor and/or main steamline radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system. The steam generator blowdown liquid monitor will automatically terminate steam generator blowdown to the cooling tower and divert flow to the condensate demineralizer.

(4) The reactor trip automatically trips the turbine and if offsite power is available the steam dump valves open permitting steam dump to the condenser. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator power operated relief valves (and safety valves if their setpoint is reached).

(5) Following reactor trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of loss of offsite power, steam relief to atmosphere.

(6) Safety injection flow results in increasing RCS pressure and pressurizer water level, and the RCS pressure trends toward an equilibrium value where the safety injection flow rate equals the break flow rate.

In the event of an SGTR, the plant operators must diagnose the event and perform the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. The operator actions for SGTR recovery are provided in the plant Emergency Operating Procedures.

Operator actions are described below.

(1) Identify the ruptured steam generator.

High secondary side activity, as indicated by the condenser vacuum exhaust radiation monitor, steam generator blowdown liquid monitor, or main steam line radiation monitor, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level, a radiation protection survey, or a chemistry laboratory sample. For an SGTR that results in a reactor trip at high power, the steam generator water level as indicated on the narrow range scale will decrease significantly for all of the steam generators. The auxiliary feedwater flow will begin to refill the steam generators, distributing flow to each of the steam generators. Since primary to secondary break flow adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly than normally expected in that steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

(2) Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

Once the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by l) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary break flow.

(3) Cool down the RCS using the intact steam generators.

After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling will exist in the RCS after depressurization of the RCS to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the steam generator power operated relief valves to release steam from the intact steam generators.

(4) Depressurize the RCS to restore reactor coolant inventory.

When the cooldown is completed, safety injection flow will increase RCS pressure until break flow matches safety injection flow. Consequently, safety injection flow must be terminated to stop primary to secondary break flow. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after safety injection flow is stopped. Since break flow from the primary side will continue after safety injection flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator.

The RCS depressurization is performed using normal pressurizer spray if the RCPs are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using the pressurizer power operated relief valve or auxiliary pressurizer spray.

(5) Terminate safety injection to stop primary to secondary break flow.

The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that safety injection flow is no longer needed. When these actions have been completed, safety injection flow must be stopped to terminate primary to secondary break flow. Primary to secondary break flow will continue after safety injection flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of break flow into the ruptured steam generator.

Following safety injection termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are

performed to cooldown and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

15.4.3.2 Analysis of Effects and Consequences

An SGTR results in the transfer of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiological consequences. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The results of this analysis demonstrated that there is margin to steam generator overfill for a design basis SGTR for Watts Bar Units 1 and 2. A thermal and hydraulic analysis was also performed to determine the input for the offsite radiological consequences analysis, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting case for the analysis of the radiological consequences for an SGTR for Watts Bar. The results of the thermal and hydraulic analysis for the offsite radiological consequences analysis are discussed as follows.

Thermal and Hydraulic Analysis

A thermal and hydraulic analysis has been performed to determine the plant response for a design basis SGTR, and to determine the integrated primary to secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information has been used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences.

The plant response following an SGTR was analyzed with the LOFTTR2 program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The time of reactor trip was calculated by modeling the Watts Bar Unit 2 reactor protection system. It was assumed that the reactor is operating at full power at the time of the accident and the initial secondary mass was assumed to correspond to operation at nominal steam generator mass, minus an allowance for uncertainties. It was also assumed that a loss of offsite power occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The limiting single failure was assumed to be the failure of the power operated relief valve on the ruptured steam generator. Failure of this valve in the open position will cause an uncontrolled depressurization of the ruptured steam generator which will increase primary to secondary break flow and the mass release to the atmosphere. It was assumed that the ruptured steam generator power operated relief valve fails open when the ruptured steam generator is isolated, and that the valve was subsequently isolated by locally closing the associated block valve.

The major operator actions required for the recovery from an SGTR are discussed in Section 15.4.3.1 and these operator actions were simulated in the analysis. The operator action times which were used for the analysis are presented in Table 15.4-20. It is noted that the power operated relief valve on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator was isolated. Before proceeding with the recovery operations, the failed open power operated relief valve was assumed to be isolated by locally closing the associated block valve. It was assumed that the ruptured steam generator power operated relief valve is isolated at 11.0 minutes after the valve was assumed to fail open. After the ruptured steam generator power operated relief valve was isolated, the additional delay time of 7.15 minutes (Table 15.4-20) was assumed for the operator action time to initiate the RCS cooldown.

Transient Description

The LOFTTR2 analysis results are described below. The sequence of events for this transient is presented in Table 15.4-21.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 15.4-97a. The RCS pressure also decreases as shown in Figure 15.4-97b as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary break flow, automatic reactor trip occurs at approximately 109 seconds on an overtemperature ΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator power operated relief valves and (safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.4-97c. The loss of offsite power at reactor trip results in the termination of main feedwater and actuation of the auxiliary feedwater system. It was assumed that auxiliary feedwater flow is initiated to all steam generators at 60 seconds after reactor trip.

The RCS pressure and pressurizer level decrease more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the leak flow

continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal at approximately 155 seconds. After SI actuation, the RCS pressure and pressurizer level begin to increase and approach the equilibrium values where the safety injection flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figures 15.4-97d and 15.4-97e); however, the temperature differential subsequently increases as the reactor coolant pumps coast down and natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The hot leg temperatures reach a peak and then slowly decrease as steady state conditions are reached until the ruptured steam generator is isolated and the power operated relief valve is assumed to fail open.

Major Operator Actions

(1) Identify and Isolate the Ruptured Steam Generator

The ruptured steam generator is assumed to be isolated at either 15 minutes after initiation of the SGTR or when the narrow range level reaches 30%, whichever time is greater. Since the time to reach 30% narrow range is less than 15 minutes, it was assumed that the ruptured steam generator is isolated at 15 minutes. The failure causes the ruptured steam generator to rapidly depressurize as shown in Figure 15.4-97c which results in an increase in primary to secondary break flow. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary which results in a decrease in the ruptured loop temperatures as shown in Figure 15.4-97e. The intact steam generator loop temperatures also slowly decrease, as shown in Figure 15.4-97d until the RCS cooldown is initiated. The shrinkage of the reactor coolant due to the decrease in the RCS temperatures results in a decrease in the pressurizer level and RCS pressure as shown in Figures 15.4-97a and 15.4-97b. When the depressurization of the ruptured steam generator is terminated, the pressure begins to increase as shown in Figure 15.4-97c.

(2) Cool Down the RCS to establish Subcooling Margin

After the block valve for the ruptured steam generator power operated relief valve is closed, there is a 7.15 minute operator action time assumed prior to initiation of cooldown. The depressurization of the ruptured steam generator due to the failed-open power operated relief valve affects the RCS cooldown target temperature since it is determined based on the pressure at that time. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact steam generator power operated relief valves. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 65 °F plus an allowance for instrument uncertainty.

Because of the lower pressure in the ruptured steam generator when the cooldown is initiated, the associated temperature the RCS must be cooled to is also lower which has the net effect of extending the time required for cooldown.

The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure 15.4-97c, and the effect of the cooldown on the RCS temperature is shown in Figure 15.4-97d. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant, as shown in Figures 15-4-97a and 15.4-97b.

(3) Depressurize RCS to Restore Inventory

After the RCS cooldown, a 2.45 minute operator action time is assumed prior to the RCS depressurization. The RCS is depressurized to assure adequate coolant inventory prior to terminating safety injection flow. With the RCPs stopped, normal pressurizer spray is not available and the RCS is depressurized by opening a pressurizer power operated relief valve. The depressurization is initiated and continued until the criteria in the emergency operating procedures are satisfied. The RCS depressurization reduces the break flow as shown in Figure 15.4-97g and increases safety injection flow to refill the pressurizer as shown in Figure 15.4-97a.

(4) Terminate SI to Stop Primary to Secondary Break Flow

The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that safety injection flow is no longer needed. When these actions have been completed, the safety injection flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary break flow. The safety injection flow is terminated at this time if the safety injection termination criteria in the emergency operating procedures are satisfied.

After depressurization is completed, an operator action time of 4.07 minutes is assumed prior to initiation of safety injection termination. When termination requirements are satisfied, actions proceed to close off the safety injection flow path. After safety injection termination, the RCS pressure begins to decrease as shown in Figure 15.4-97b. The intact steam generator power operated relief valves are opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the power operated relief valves are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The differential pressure between the RCS and the ruptured steam generator is shown in Figure 15.4-97f. Figure 15.4-97g shows that the primary to secondary break flow continues after the safety injection flow is stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume for the transient is shown in Figure 15.4-97h. The mass of water in the ruptured steam generator is also shown as a function of time in Figure 15.4-97i.

Mass Releases

The mass releases are determined for use in evaluating the site boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator are determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours are used to calculate the radiation doses at the site boundary for a 2 hour exposure, and the releases for 0-8 hours are used to calculate the radiation doses at the low population zone for the duration of the accident.

The operator actions for the SGTR recovery up to the termination of primary to secondary break flow are simulated in the LOFTTR2 analysis. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary break flow into the ruptured steam generator are determined from the LOFTTR2 results for the period from the initiation of the accident until the break flow is terminated.

Following the termination of break flow, actions are taken to cooldown the plant to cold shutdown conditions. The power operated relief valves for the intact steam generators can be used to cool down the RCS to the RHR system operating temperature of 375°F, at the maximum allowable cooldown rate of 100°F/hr. The steam releases and the feedwater flows for the intact steam generators for the period from break flow termination until two hours are then determined from a mass and energy balance using the calculated RCS and intact steam generator conditions at the time of break flow termination and at 2 hours. The RCS cooldown is continued after 2 hours until the RHR system in-service temperature of 375 °F is reached. Depressurization of the ruptured steam generator can be performed to the RHR in-service pressure of 414.7 psia via steam release from the ruptured steam generator power operated relief valve. The RCS pressure is also reduced concurrently as the ruptured steam generator is depressurized. Therefore, the analysis assumes that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours are then determined for the intact and ruptured steam generators from a mass and energy balance using the conditions at 2 hours and at the RHR system in-service conditions.

After 8 hours, plant cooldown to cold shutdown as well as long-term cooling can be provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR cut-in, assumed to be reached at 8 hours.

For the time period from initiation of the accident until break flow termination, the releases are determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any

radioactivity released to the atmosphere prior to reactor trip would be through the condenser vacuum exhaust. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator power operated relief valves. The mass release rates to the atmosphere from the LOFTTR2 analysis are presented in Figure 15.4-97j and 15.4-97k for the ruptured and intact steam generators, respectively, for the time period until break flow termination. The mass releases calculated from the time of break flow termination until 2 hours and from 2-8 hours were also assumed to be released to the atmosphere via the steam generator power operated relief valves. The mass releases for the SGTR event for the 0-2 hour and 2-8 hour time intervals considered are presented in Table 15.4-22.

In addition to the mass releases, information is developed for use in performing the offsite radiation dose analysis. The time dependent fraction of rupture flow that flashes to steam and is assumed to be immediately released to the environment is presented in Figure 15.4-97e. The break flow flashing fraction is conservatively calculated assuming that 100% of the break flow comes from the hot leg side of the steam generator, whereas the break flow actually comes from both the hot leg and cold leg sides of the steam generator. The water above the steam generator tubes reduces the iodine content of the atmospheric release by scrubbing the steam bubbles as they rise from the rupture to the water surface. However, if partial tube uncovery were to occur, the increase in iodine release would be negligible. This result for tube uncovery is described in References [39] and [40]. Reference [41] provides NRC approval of References [39] and [40] and states that no further evaluation of steam generator tube uncovery is required.

15.4.3.3 Conclusions

A steam generator tube rupture will cause no subsequent damage to the reactor coolant system or the reactor core. An orderly recovery from the accident can be completed even assuming simultaneous loss of offsite power. The results of the thermal and hydraulic analysis are used to evaluate the environmental consequences of the postulated SGTR. The results of the environmental consequences analysis are presented in Section 15.5.5.

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.4.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor such as is discussed in Section 5.5.1.3.5.

Flow through the affected reactor coolant loop is rapidly reduced, leading to initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine

steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

The consequences of a locked rotor are very similar to those of a pump shaft break. The initial rate of reduction of coolant flow is greater for the locked rotor event. However, with a failed shaft, the impeller could conceivably be free to spin in the reverse direction as opposed to being fixed in position as assumed for a locked rotor. The effect of such reverse spinning is a slight decrease in the endpoint (steady-state) core flow when compared to the locked rotor. Only one analysis is performed, representing the most limiting condition for the locked rotor and pump shaft break accidents.

15.4.4.2 Analysis of Effects and Consequences

Method of Analysis

Two digital-computer codes are used to analyze this transient. The LOFTRAN[11] Code is used to calculate the resulting loop and core flow transient following the pump seizure, the time of reactor trip, based on the loop flow transients, the nuclear power following reactor trip, and the reactor coolant system peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[12] Code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN Code includes a film boiling heat transfer coefficient.

One reactor coolant pump seizure has been analyzed for a locked rotor/shaft break with four loops in operation.

The accident is evaluated without offsite power available. For the evaluation, power is assumed to be lost to the unaffected pumps instantaneously after reactor trip. At the beginning of the postulated locked rotor accident, i.e. at the time the shaft in one of the reactor coolant pumps in assumed to seize, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., maximum steady state power level, maximum steady state pressure, and maximum steady state coolant average temperature.

When the peak pressure is evaluated, the initial pressure is conservatively estimated as 70 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response shown in Figure 15.4-15 is at the point in the reactor coolant system having the maximum pressure.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin 1.2 seconds after the flow in the affected loop reaches 87% to nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these systems are expected to function and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are full open at 2580 psia and their capacity for steam relief is as described in Section 5.2.2.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core and, therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of the 'hot spot' condition represent the upper limit with respect to clad temperature and zirconium water reaction.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation[19]. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures).

The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a presounded influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 BTU/hr-ft^2-°F at the initiation of the transient. Thus the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp - \left[\frac{45,500}{1.986T}\right]$$

where:

w = amount reacted, mg/cm^2

t = time, sec

T = temperature, °K

The reaction heat is 1510 cal/gm

Results

The calculated sequence of events is shown on Table 15.4-1. The transient results without offsite power available are shown in Figures 15.4-15 through 15.4-20. The peak reactor coolant system pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerable less than 2700 °F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient. The results of these calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are also summarized in Table 15.4-10.

15.4.4.3 Conclusions

(1) Since the peak reactor coolant system pressure reached during any of the transients is less than that which cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

(2) Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F, and the amount of zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability.

15.4.5 Fuel Handling Accident

15.4.5.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly onto the fuel storage area floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. Dropping a fuel assembly in the spent fuel pool has been analyzed and will not result in criticality.[43]

15.4.5.2 Analysis of Effects and Consequences

For the analyses and consequences of the postulated fuel handling accident, refer to Section 15.5.6.

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

15.4.6.1 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 (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.6.1.1 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, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

Mechanical Design

The mechanical design is discussed in Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

(1) Each full length control rod drive mechanism housing was completely assembled and shop tested at 4100 psi.

(2) The mechanism housings were individually hydrotested after being attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed reactor coolant system.

(3) Stress levels in the mechanism are not be affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments by the design earthquake are acceptable within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

(4) The latch mechanism housing and rod travel housing are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The 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. Administrative regulations require periodic inspections of these (and other) welds.

Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated by compensating for fuel depletion and xenon oscillations with changes to the boron concentration. Typically the control rods are not deeply inserted. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full power operation, a less severe reactivity excursion could be expected to occur.

However, it may be occasionally 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 RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instruction requirements are as specified in Technical Specifications 3.1.5, 3.1.6 and 3.1.7.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference [14]. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings leading to an increase in severity of the initial accident.

Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the steel tube.

If failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not be bent because of the rigidity of multiple adjacent housings.

Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil stack assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited 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.

Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.6.1.2 Limiting Criteria

Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable 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[15]. Extensive tests of UO_2 zirconium clad 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[13] results, which 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, 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 to be below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

(2) Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits. This criteria is generically addressed in Reference [16].

(3) Fuel melting will be limited to less than the innermost 10% of the fuel pellet at the hot spot even if the average fuel pellet enthalpy at the hot spot is below the limits of criterion 1 above.

It should be noted that the FSAR included an additional criterion that the average clad temperature at the hot spot must remain below 3000°F. The elimination of this criterion as a basis for evaluating the RCCA Ejection accident results is consistent with the revised Westinghouse acceptance criteria for this event.

15.4.6.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages: first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial 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 energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference [16].

Average Core Analysis

The spatial kinetics computer code, TWINKLE[17], is used for the average core transient analysis. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback affects. In this analysis, the code is used as a one dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement and the elimination of axial feedback weighting factors. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 15.1.9.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN[12]. This computer code 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 distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong correlation[19] to determine the film boiling coefficient after DNB. The DNB heat flux is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state temperature distribution to agree with the fuel heat transfer design codes presently in use by Westinghouse. Further description of FACTRAN appears in Section 15.1.9.

System Overpressure Analysis

Because safety limits for 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 this heat flux data, a thermal hydraulic 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 reactor coolant system and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

The system overpressure is generically addressed in Reference [16].

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 15.4-12 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by synthesis method employing one dimensional and two 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 and part length rod positions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties.

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 channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, the axial weighting is not necessary. In addition, no weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a

function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis[16].

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning-of-life and end-of-life are adjusted in the nuclear code in order to obtain moderator temperature coefficients which are conservative compared to actual design conditions for the plant. For example, a Positive Moderator Temperature Coefficient (PMTC) of +5 pcm / °F was applied to both beginning-of-life rod ejection cases, although a PMTC is precluded by the plant Technical Specifications at hot full power conditions. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady state computer code with a Doppler weighting factor of 1.0. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1.0 (approximately 1.2), just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction β_{eff} typically yield values no less than 0.70% at beginning-of-life and 0.50% at end-of-life for the first cycle. The accident is sensitive to β if the ejected rod worth is equal to or greater than β as in zero power transients. In order to allow for future cycles, conservative estimates of β of 0.48% at beginning-of-cycle and 0.44% at end-of-cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-12 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.7 seconds after the start of fall. 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 important conservatism for a full-power accident. The rod ejection transient was evaluated using the thermal design flowrate.

The minimum design shutdown margin available for this plant at HZP may be reached only at end-of-life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1% Δk. Therefore, following a

reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed for a typical four-loop plant assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated on low pressurizer pressure within one minute after the break. The reactor coolant system pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about two to three minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the reactor coolant system temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2% Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains sub-critical during the cooldown.

Results

Cases are presented for both beginning and end-of-life at zero and full power.

In the full power cases, control bank D was assumed to be inserted to its insertion limit. In the zero power cases, control bank D was assumed to be fully inserted, and control banks B and C were assumed to be at their insertion limits.

The results for these cases are summarized in Table 15.4-12. In all cases the maximum fuel pellet average enthalpy is well below that which could cause sudden cladding failure, the maximum clad average temperature is below the point of clad embrittlement, and fuel melting, if any, is limited to less than 10% of the fuel cross-section at the hot spot.

The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning-of-life full power and end-of-life zero power) are presented in Figures 15.4-24 through 15.4-27.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a detailed three-dimensional THINC analysis[16]. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth 1 dollar at beginning-of-life, hot full power, indicates that the peak pressure does not exceed that which would cause the faulted condition stress limits to be exceeded[16]. Since the severity of the present analysis does not exceed this "worst case" analysis, the

accident for this plant will not result in an excessive pressure rise or further damage to the reactor coolant system.

Lattice Deformations

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 may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. Since the 17 x 17 fuel design is also under-moderated, the same effect would be observed.

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 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.

15.4.6.3 Conclusions

Even on a worst-case 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 reactor coolant system. The reference [16] analyses have demonstrated that the number of fuel rods entering DNB amounts to less than 10%, thus satisfactorily limiting fission product release.

The environmental consequences of this accident is bounded by the loss of coolant accident. See Section 15.5.3, "Environmental Consequences of a Loss of Coolant Accident." The reactor coolant system integrated break flow to containment following a rod ejection accident is shown in Figure 15.4-28.

Following reactor trip, requirements for operator action and protection system operation are similar to those presented in the analysis of a small loss of coolant event, section 15.3.1.

REFERENCES

(1) Deleted by Amendment 97

(2) Deleted by Amendment 97

(3) Deleted by Amendment 97

(4) Deleted by Amendment 97

(5) Hsieh, T., and Raymund, M., "Long Term Ice Condenser Transient Analysis (LOTIC II)," WCAP-8355 Supplement 1, May 1975 and WCAP-8354 (Proprietary), July 1974.

(6) Deleted by Amendment 97

(7) Deleted by Amendment 63.

(8) Deleted by Amendment 80.

(9) Moody, F. S., "Transactions of the ASME, Journal of Heat Transfer," Figure 3, Page 134, February 1965.

(10) Deleted Amendment 80.

(11) Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7907-P-A (proprietary) and WCAP-7907-A (non-proprietary), April 1984.

(12) Hunin, C., "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, July 1972.

(13) Liimataninen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," ANL-7225, January - June 1966, p. 177, November 1966.

(14) Burnett, T. W. T, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April 1969.

(15) Taxelius, T. G., "Annual Report - Spert Project, October 1968, September 1968," Idaho Nuclear Corporation IN-1370, June 1970.

(16) Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

(17) Barry, R. F., and Risher, D. H., Jr., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8208-A, January 1975 (Non-Proprietary).

(18) Deleted by Amendment 80.

(19) Bishop, A. A., et al., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

(20) Deleted by Amendment 97.

(21) Deleted by Amendment 97.

(22) Deleted by Amendment 97.

(23) Deleted by Amendment 97.

(24) Deleted by Amendment 97.

(25) Deleted by Amendment 97.

(26) Deleted by Amendment 97.

(27) Deleted by Amendment 97.

(28) Deleted by Amendment 80.

(29) Deleted by Amendment 80.

(30) C. W. Stewart, et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, "Volumes 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, EPRI.

(31) Deleted by Amendment 80.

(32) Deleted by Amendment 97.

(33) "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

(34) Rupprecht, S. D, et. al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary), WCAP-11372 (Non-Proprietary), October 1986.

(35) U.S. Nuclear Regulatory Commission, Code Federal Regulations - Energy 10, Chapter 1, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

(36) "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting", WCAP-13451 October 1992.

(37) Devault, R. M., Smith, J. D., and Studer, P. G., "MONSTER - A Multi-Compartment Containment System Analysis Program User Manual," System I.D. 262303, March 1993.

(38) Deleted by Amendment 97

(39) Letter from Walsh, L. A., Westinghouse Owners Group, to Jones, R. C., U.S. Nuclear Regulatory Commission, "Steam Generator Tube Uncovery Issue," OG-92-25, March 1992.

(40) "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," WCAP-13247 (Proprietary), March 1992.

(41) Letter from Jones, R. C., U.S. Nuclear Regulatory Commission, to Walsh, L. A., Westinghouse Owners Group, "Steam Generator Tube Uncovery Issue," March 10, 1993.

(42) Watts Bar "Design Basis Events Design Criteria", Document WB-DC-40-64.

(43) Criticality Analysis Summary Report for Watts Bar Nuclear Plant," Document Number PFE-R07, Tennessee Valley Authority Nuclear Fuels Department (L38 961015 802).

(44) USNRC Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performances", May 1989.

(45) Boyack, B., et al, 1989, "Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of- Coolant-Accident", NUREG/CR-5249

(46) "Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," WCAP- 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), March 1998 (Westinghouse Proprietary).

(47) "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant," WCAP-14839-P Revision 1, June 1998.

(48) Letter from W. J. Johnson of Westinghouse to R. C. Jones of the NRC, "Use of 2700 °F PCT Acceptance Limit in Non-LOCA Accidents, "NS-NRC-89-3466, October 1989.

(49) "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", WCAP-16009-P-A, January 2005 (Westinghouse Proprietary)

(50) "Emergency Core Cooling System Analysis Methods", SECY-83-472, Information Report from W. J. Dircks to the Commissioners, November 17.1983.

(51) "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),"WCAP-15063-P-A, Revision 1 with Errata (Proprietary), July 2000.

Table 15.4-1 Time Sequence Of Events For Condition IV Events (Page 1 of 2)

Accident	Event	Time (Seconds)
Major Reactor Coolant System Pipe Ruptures, Double-Ended Cold Leg Guillotine	See Table 15.4-17	
Major Secondary System Pipe Rupture		
1. Case B		
Complete severence of a pipe, loss of offsite power simultaneous with the break and initiation of safety injection signal	Steam Line Ruptures	0.0
	Low Steam Pressure Setpoint Reached	0.67
	Pressurizer Empties	12.0
	Criticality Attained	58.0
	Boron Reaches Core	46.0
	Accumulators Actuated	N/A
2. Case A		
Complete severence of a pipe, offsite power available	Steam Line Ruptures	0.0
	Low Steam Pressure Setpoint Reached	0.67
	Pressurizer Empties	11.0
	Boron Reaches Core	34.0
	Criticality Attained	44.0
	Accumulators Actuated	54
Reactor Coolant Pump Shaft Seizure (Locked Rotor/Broken Shaft)		
All pumps in operation, one shaft seizure without offsite power available	Rotor on one pump seizes	0
	Low flow trip point reached	0.02
	Rods begin to drop	1.22
	Undamaged pumps lose power and begin coasting down	1.22
	Maximum RCS pressure occurs	3.50
	Maximum clad temperature occurs	3.99

Table 15.4-1 Time Sequence Of Events For Condition IV Events (Page 2 of 2)

Accident	Event	Time (Seconds)	
Rod Ejection		BOL HFP	EOL HZP
	RCCA Ejected	0.0	0.0
	Reactor Trip Setpoint Reached	0.05	0.163
	Peak Nuclear Power	0.135	0.193
	Rods Drop	0.55	0.663
	Peak Fuel Average Temperature is Reached	2.205	1.821
	Peak Clad Temperature is Reached	2.25	1.490
	Peak Heat Flux	2.256	1.519

Table 15.4-2 Deleted by Amendment 97

Table 15.4-3 Deleted by Amendment 97

Table 15.4-4 Deleted by Amendment 97

Table 15.4-5 Deleted by Amendment 97

Table 15.4-6 Equipment Required Following A High Energy Line Break (Page 1 of 3)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on under- voltage, underfrequency, and turbine trip may be excluded).	Auxiliary feedwater system including pumps, water supply, and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident).	Steam generator power-operated relief valves (can be manually operated locally) Controls for defeating automatic safety injection actation during a cooldown and depressurization.
Safety injection system including the pumps, the refueling water storage tank, and the systems valves and piping.	Capability for obtaining a reactor coolant systemsample.	Residual heat removal system including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the reactor coolant system in a cold shutdown condition
Diesel generators and emergency power distribution equipment.	Lower compartment cooling fans must be started (a minimum of 2 of 4) 1-1/2 hours to 4 hours after the initiation of HELB.	
Essential raw cooling water system	Ice condenser.	
Containment safeguards cooling equipment.	Air return fan to recirculate air thru ice condenser.	
Main feedwater control valves* (trip closed feature).	Containment spray to maintain hot standby lower compartment temperature.	

Table 15.4-6 Equipment Required Following A High Energy Line Break (Page 2 of 3)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Bypass feedwater control valves* (trip closed feature).		
Circuits and/or equipment required to trip the main feedwater pumps.*		
Main steam line stop valves* (Main Steam Isolation Valves trip closed feature).		
Main steam line stop valve bypass valves* (trip closed feature).		
Steam generator blowdown isolation valves (automatic closure feature).		
Batteries (Class 1E).		
Control room ventilation.		
Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		

Table 15.4-6 Equipment Required Following A High Energy Line Break (Page 3 of 3)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Emergency lighting.		
Post accident monitoring system**		
Wide range Thot or Tcold for each reactor coolant loop.		
Pressurizer water level.		
Wide range reactor coolant system pressure		
Steam line pressure for each steam generator.		
Wide range and narrow range steam generator level for each steam generator.		
Containment pressure		

* Required for steam line, feed line, and steam generator blowdown line break only.

** See Section 7.5 for a discussion of the post accident monitoring system.

Table 15.4-7 Limiting Core Parameters Used In Steam Break DNB Analysis

Reactor vessel inlet temperature (°F)	
Faulted SG Loop	398.7
Intact SG Loops	479.5
RCS pressure (psia)	603.22
RCS flow fraction of nominal (%)	100
Heat flux fraction of nominal (%)	1.6
Reactivity (%)	0.015
Density (gm/cc)	0.829
Boron (ppm)	16.45
Time (seconds)	57.4

Table 15.4-8 Deleted by Amendment 80

Table 15.4-9 Time Sequence Of Events For Feedline Break

Event	Time (seconds)	
	With Offsite Power	Without Offsite Power
Feedline rupture occurs	10	10
Pressurizer relief valve setpoint reached	26.5	26.5
Low-low steam generator level reactor trip and auxiliary feedwater pump start setpoint reached in affected steam generator	37	37
Rods begin to drop	39	39
Auxiliary feedwater starts to intact steam generators	97	97
Cold auxiliary feedwater reaches intact steam generators	144	144
Low steamline pressure setpoint reached	328	392
All main steam stop (main steam isolation) valves closed	336	400
Pressurizer water relief begins	680	3196
Core power decreases to auxiliary feedwater removal capacity	764	≈3600

Table 15.4-10 Summary Of Results For Locked Rotor Transients

	4 Loops Operating Initially
Maximum reactor coolant system pressure (psia)	2672 (1)
Maximum clad temperature at core hot spot (°F)	1852
Zr-H2O reaction at core hot spot (% by weight)	0.36%

1. A generic study was performed that addressed an initial pressurizer level including the pressurizer water level uncertainty which determined that at most a 41 psi increase would result from modeling this condition. The evaluation demonstrated that sufficient margin exists and the pressure limit will continue to be met. Hence, the conclusions presented in the section remain valid.

Table 15.4-11 Deleted by Amendment 80

Table 15.4-12 Parameters Used In The Analysis Of The Rod Cluster Control Assembly Ejection Accident

Time in Life	Beginning	Beginning	End	End
Power Level, %	102	0	102	0
Ejected rod worth, %ΔK	0.200	0.725	0.210	0.970
Delayed neutron fraction, %	0.48	0.48	0.44	0.44
Trip Reactivity, % ΔK	4.0	2.0	4.0	2.0
Fq before rod ejection	2.50	--	2.50	--
Fq after rod ejection	6.70	10.60	7.25	23.0
Number of operational pumps	4	2	4	2
Results:				
Max. fuel pellet average temperature, °F	3932	2929	3804	3752
Max. fuel center temperature, °F	4948	3400	4851	4190
Max. clad average temperature, °F	2207	2175	2130	2957
Max. fuel stored energy, cal/gm	171	121	164	162
Percent of fuel melted	<10	0	<10	0

Table 15.4-13 Parameters Recommended For Determining Radioactivity Releases For Rod Ejection Accident

Failed fuel	10% of fuel rods in core
Activity released to reactor coolant from failed fuel and available for release	
Noble gases Iodines	10% of gap inventory 10% of gap inventory
Melted fuel	0.25% of core
Activity released to reactor coolant from melted fuel and available for release	
Noble gases Iodines	0.25% of core inventory 0.125% of core inventory
Steam dump from relief valves	59,000 lbs
Duration of dump from relief valves	140 sec
Time between accident and equilization of primary and secondary system pressures	300 sec

Table 15.4-14 Large-Break LOCA Containment Data (Ice Condenser Containment) Used for Calculation of Containment Pressure for Watts Bar Unit 2

Parameter	Value
Net Free Volume Distribution Between Upper (UC), Lower (LC), Ice Condenser (IC) and Dead-Ended (DE) Compartments	UC: 710,000 ft^3 LC: 253,114 ft^3 IC: 122,350 ft^3 DE: 129,900 ft^3
Initial Condition Containment Pressure	14.7 psia
Maximum Temperature for the Upper (UC), Lower (LC and Dead-Ended (DE) Compartments	UC: 110°F LC: 120°F DE: 120°F
Minimum RWST Temperature (Containment Spray Temperature)	60°F
Minimum Temperature Outside Containment	5°F
Maximum Containment Spray Flow Rate	4000 gpm/pump
Number of Spray Pumps Operating	2
Post-Accident Initiation of Spray System	25 sec
Post-Accident Delay Time for Deck Fan Actuation	490 sec
Deck Fan Flow Rate	41,690 cfm/fan
Initial Ice Mass	2,450,000 lb_m

Table 15.4-15 Large-Break Containment Data - Heat Sinks Data (Ice Condenser Containment)

Wall	Compartment(1)	Area [ft^2]	Thickness [ft]	Material
1	UC	5124.	1.6	concrete
2	UC	19992.	0.000525/1.6	coating/concrete
3	UC	4032.	0.02167/1.6	stainless steel/concrete
4	UC	11192.	0.00065/0.03908	coating/carbon steel
5	UC	47800.	0.00065/0.09252/1.0	coating/carbon steel/concrete
6	UC	273.	0.00065/0.1308	coating/carbon steel
7	LC	59000.	2.1	concrete
8	LC	17178.	0.000133/2.1	coating/concrete
9	LC	12988.	2.1	concrete
10	LC	2384.	0.02167/2.1	stainless steel/concrete
11	LC	25444.	0.00065/0.1089/1.0	coating/carbon steel/concrete
12	LC	12810.	0.00065/0.07593	coating/carbon steel
13	LC	2625.	0.00055/0.12083	coating/carbon steel
14	LC	1575.	0.00065/0.14167	coating/carbon steel
15	LC	12915.	0.00065/0.044167	coating/carbon steel
16	LC	12988.	2.1	concrete
17	LC	3439	0.1561	carbon steel

Notes:

1. UC and LC are Upper and Lower Compartment, respectively.

Table 15.4-16 Mass And Energy Release Rates Used for Calculation of Containment Pressure for Watts Bar Unit 2 (Page 1 of 2)

Time After Break (sec)	Mass Flow Rate(lbm/sec)	Energy Flow Rate (BTU/sec)
0.	9646.7	5369419.
1.	71201.5	39577048.
2.	50782.1	28747484.
3.	40475.1	23410743.
4.	34105.4	20560588.
5.	30009.1	18713303.
6.	27906.0	17640053.
7.	26130.9	16632257.
8.	24651.1	15663961.
9.	22805.6	14511306.
10.	20004.8	13053678.
11.	17472.9	11605252.
12.	14601.0	10093803.
12.4	13464.4	9420184.
14.	12172.5	7614137.
15.	12554.4	6455205.
16.	11369.7	5308157.
17.	10902.4	4491501.
18.	10124.7	3756484.
19.	9258.1	3127399.
20.	8178.7	2411114.
21.	7321.0	2120146.
22	7603.9	1977749.
23.	5474.9	1402837.
24.	4641.5	999621.
25.	6992.0	1356562.
26.	5955.4	1051498.
28.	4062.4	618361.
29.	3020.7	405978.
30.	1824.1	201868.
32.	1873.8	190499.

Table 15.4-16 Mass And Energy Release Rates Used for Calculation of Containment Pressure for Watts Bar Unit 2 (Page 2 of 2)

Time After Break (sec)	Mass Flow Rate(lbm/sec)	Energy Flow Rate (BTU/sec)
33.	1882.1	180047.
34.5	1890.2	204412.
35.	1921.9	200973.
39.	2275.7	246561.
41.	1959.7	214441.
43.	2031.8	267974.
45.	2650.2	384113.
46.	7824.1	1100908.
47.5	2842.5	400880.
50.	1811.6	373546.
51.	1764.3	397686.
55.	2254.1	544982.
57.5	1383.9	503576.
60.	1621.8	592463.
65.	790.9	338984.
80.	686.2	251517.
110.	646.9	232801.
150.	643.9	307300.
190.	654.1	229705.
226.	374.2	116811.
300.	404.3	144644.
349.	503.8	176903.

Table 15.4-17 Watts Bar Unit 2 Best-Estimate Large-Break LOCA Sequence Of Events for Limiting PCT Transient

Event	Time after break (sec)
Start of Transient	0.0
Safety Injection Signal	5
Accumulator Injection Begins	10
End of Blowdown	11
Bottom of Core Recovery	36
Accumulator Empty(1)	43
Safety Injection Begins	60
PCT Occurs	190
End of analysis time	400.0

Note:
1. Accumulator injection switches from liquid to nitrogen.

Table 15.4-18a Peak Clad Temperature Including All Penalties and Benefits, Best-Estmate Large-Break LOCA (BE LBLOCA) for Watts Bar Unit 2

PCT for Analysis-of-Record (AOR)	**1766°F**
PCT Assessments Allocated to AOR	
None	N/A
BE LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	1766°F

Table 15.4-18b Watts Bar Unit 2 Best-Estimate Large-Break LOCA Results

ASTRUM Results	AOR Value	Acceptance Criteria
95/95 PCT	1766°F	<2200°F
95/95 MLO	1.99%	<17%
95/95 CWO	0.08%	<1%

Table 15.4-19 Plant Operating Range Analyzed by the Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2 (Page 1 of 2)

Parameter		As-Analyzed Value or Range
1.0	**Plant Physical Description**	
	a) Dimensions	Nominal
	b) Pressurizer location	Modeled on an intact loop
	c) Hot assembly location	Anywhere in core interior (1)
	d) Hot assembly type	17x17 RFA-2, ZIRLO® Clad with IFMs
	e) Steam generator tube plugging level	≤ 10% Any or All SGs
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Core Power	3479.8 MWt ±0% Uncertainty(2)
	b) Peak heat flux hot channel factor (F_Q)	≤2.50 See Table 15.4-24
	c) Peak hot rod enthalpy rise hot channel factor ($F_{\Delta H}$)	≤1.65 See Table 15.4-24
	d) Hot assembly radial peaking factor ($\overline{P}_{HA}$)	≤1.65/1.04 See Table 15.4-24
	e) Hot assembly heat flux hot channel factor (F_{QHA})	≤2.50/1.04 See Table 15.4-24
	f) Axial power distribution (P_{BOT},P_{MID})	Figure 15.4-56
	g) Low power region relative power (P_{LOW})	$0.2 \le P_{LOW} \le 0.8$
	h) Hot assembly burnup	≤ 62,000 MWD/MTU, lead rod
	i) MTC	≤ 0 at hot full power (HFP)
	j) Typical cycle length	20,000 MWD/MTU
	k) Minimum beginning of cycle core average burnup	≥ 10,000 MWD/MTU
	l) Maximum steady state depletion, F_Q	2.0 See Table 15.4-24
	2.2 Fluid Conditions	
	a) T_{AVG}	$582.2°F \le T_{AVG} \le 594.2°F$
	b) Pressurizer pressure	$2180 \text{ psia} \le P_{RCS} \le 2300 \text{ psia}$
	c) Loop flow	TDF ≥ 93,100 gpm/loop
	d) Upper head temperature	$= T_{COLD}$
	e) Pressurizer level (at full power)	1067 ft^3
	f) Accumulator temperature	$100°F \le T_{ACC} \le 120°F$
	g) Accumulator pressure	$585 \text{ psig} \le P_{AC} \le 690 \text{ psig}$
	h) Accumulator liquid volume	$1005 \text{ ft}^3 \le V_{ACC} \le 1095 \text{ ft}^3$

Table 15.4-19 Plant Operating Range Analyzed by the Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2 (Page 2 of 2)

Parameter		As-Analyzed Value or Range
	i) Accumulator fL/D	5.6186 ± 20%
	j) Minimum accumulator boron	1900 ppm (4)
3.0	**Accident Boundary Conditions**	
	a) Minimum safety injection flow	Table 15.4-23
	b) Safety injection temperature	60°F ≤ SI Temp ≤ 105°F
	c) Safety injection delay (5)	40 seconds (with offsite power) 55 seconds (with LOOP)
	d) Containment modeling	Tables 15.4-14, 15.4-15, and 15.4-16 and Figure 15.4-40b
	e) Single failure	1 RHR, 1 IHSI, and 1 CH/SI Pump Operable; Containment pressure: all trains operational

Notes:

1. 44 peripheral locations will not physically be lead power assembly.
2. The core average linear heat rate is set equal to a value corresponding to 3479.8 MWt (100.6 percent of 3459 MWt), and is not ranged in the uncertainty analysis. This power level approach bounds any future plant operation whose product of nominal full power and calorimetric uncertainty of ≤ 3479.8 MWt (for example, a nominal full power of 3479.8/1.005) MWt and 0.5% calorimetric uncertainty is bounded).
3. Not Used.
4. The accumulator boron concentration used for the uncertainty analysis was 1900 ppm rather than 3000 ppm, which was the value transmitted to Westinghouse by TVA. This bounds the value transmitted by TVA and will have no impact on the results presented herein.
5. Conservatively high SI delay times were used to bound the values transmitted by TVA to Westinghouse.

Table 15.4-20 Operator Action Times For Design-Basis Steam Generator Tube Rupture Analysis

Identify and isolate ruptured SG	15.00 min or LOFTTR2 calculated time from event initiation to reach 30% narrow range level in the ruptured SG, whichever is longer
Operator action time to initiate cooldown	7.15 min
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	2.45 min
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	4.07 min
SI termination and pressure equalization	Calculated by LOFTTR2

Table 15.4-21 Steam Generator Tube Rupture Analysis Sequence Of Events

EVENT	TIME (sec)
SG Tube Rupture	0
Reactor Trip	109
Safety Injection	155
Ruptured SG Isolated	900*
Ruptured SG Atmospheric Steam Dump Valve Fails Open	906
Ruptured SG Atmospheric Steam Dump Valve Closed	1566
RCS Cooldown Initiated	1995
Flashing Stops	2253
RCS Cooldown Terminated	3152
RCS Depressurization Initiated	3303
RCS Depressurization Terminated	3392
SI Terminated	3638
Break Flow Terminated	5032

* Additional two seconds results from program limitations for simulating operator actions.

Table 15.4-22 Steam Generator Tube Rupture Analysis Mass Release Results Total Mass Flow (Pounds)

	0 - 2 HRS	2 - 8 HRS
Ruptured SG		
- Condenser	118,600	0
- Atmosphere	103,300	32,800
- Feedwater	149,600	0
Intact SGs		
- Condenser	532,400	0
- Atmosphere	492,100	900,200
- Feedwater	1,018,600	900,500
Break Flow	191,400	0

Flashing Break Point Pre-Trip = 934.4 lbm
Flashing Break Flow Post-Trip = 9142.8 lbm

Table 15.4-23 Minimum Injected Safety Injection Flow Used in Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2

Pressure [psia]	Charging Flow [gpm]	SI Flow [gpm]	RHR Flow [gpm]	Total Flow [gpm]
14.7	262.2	416.9	2715.2	3394.3
34.7	260.5	413.7	2284.6	2958.8
54.7	258.8	410.4	1811.3	2480.5
74.7	257.0	407.2	1367.7	2031.9
94.7	255.3	403.9	1156.9	1816.1
114.7	253.6	400.7	916.9	1571.2
134.7	251.9	396.8	633.2	1281.9
154.7	250.1	393.0	232.2	875.3
214.7	244.9	381.4	0.0	626.3
314.7	236.1	360.8	0.0	596.9
414.7	227.1	339.5	0.0	566.6
514.7	218.0	317.6	0.0	535.6
614.7	208.4	294.6	0.0	503.0
714.7	198.6	269.8	0.0	468.4
814.7	188.5	242.7	0.0	431.2
914.7	178.2	214.5	0.0	392.7
1014.7	167.6	185.0	0.0	352.6
1114.7	156.7	150.2	0.0	306.9
1214.7	145.4	150.9	0.0	251.3
1314.7	131.0	50.8	0.0	181.8
1414.7	115.9	0.0	0.0	115.9
1514.7	100.1	0.0	0.0	100.1
1614.7	83.0	0.0	0.0	83.0
1714.7	63.7	0.0	0.0	63.7
1814.7	44.3	0.0	0.0	44.3
1914.7	27.0	0.0	0.0	27.0
2014.7	5.8	0.0	0.0	5.8
2114.7	0.0	0.0	0.0	0.0

Table 15.4-24 Summary of Peaking Factor Burndown Analyzed by the Best-Estimate Large-Break LOCA Analysis for Watts Bar Unit 2

Hot Rod Burnup (MWD/MTU)	FdH (with uncertainties)	FQ Transient (with uncertainties)	FQ Steady-state (without uncertainties)
0	1.65(1)	2.50(1)	2.00(1)
30000	1.65(1)	2.50(1)	2.00(1)
60000	1.525	2.25	1.800
62000	1.525	2.25	1.800
Note 1: Same Value as Table 15.4-19 (Note FQ SS is titled ‘SS depletion’ therein)			

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Figure 15.4-1 Deleted by Amendment 97

Figure 15.4-1a Deleted by Amendment 97

Figure 15.4-1b Deleted by Amendment 97

Figure 15.4-2 Deleted by Amendment 97

Figure 15.4-3 Deleted by Amendment 97

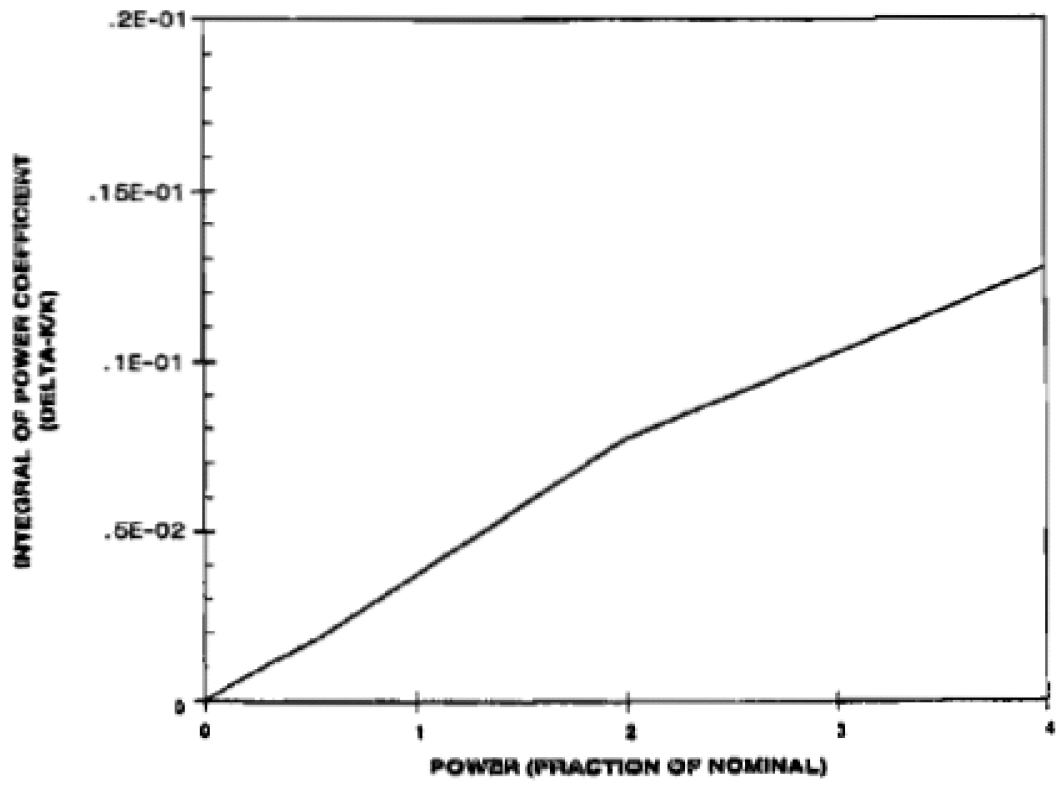
Figure 15.4-4 Deleted by Amendment 97

Figure 15.4-5 Deleted by Amendment 97

Figure 15.4-6 Deleted by Amendment 97

Figure 15.4-7 Deleted by Amendment 97

Figure 15.4-8 Deleted by Amendment 97



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VARIATION OF REACTIVITY WITH
POWER AT A CONSTANT
CORE AVERAGE TEMPERATURE
FIGURE 15.4-9

Figure 15.4-9 Variation of Reactivity with Power at Constant Core Average Temperature

Figure 15.4-10 Deleted by Amendment 89

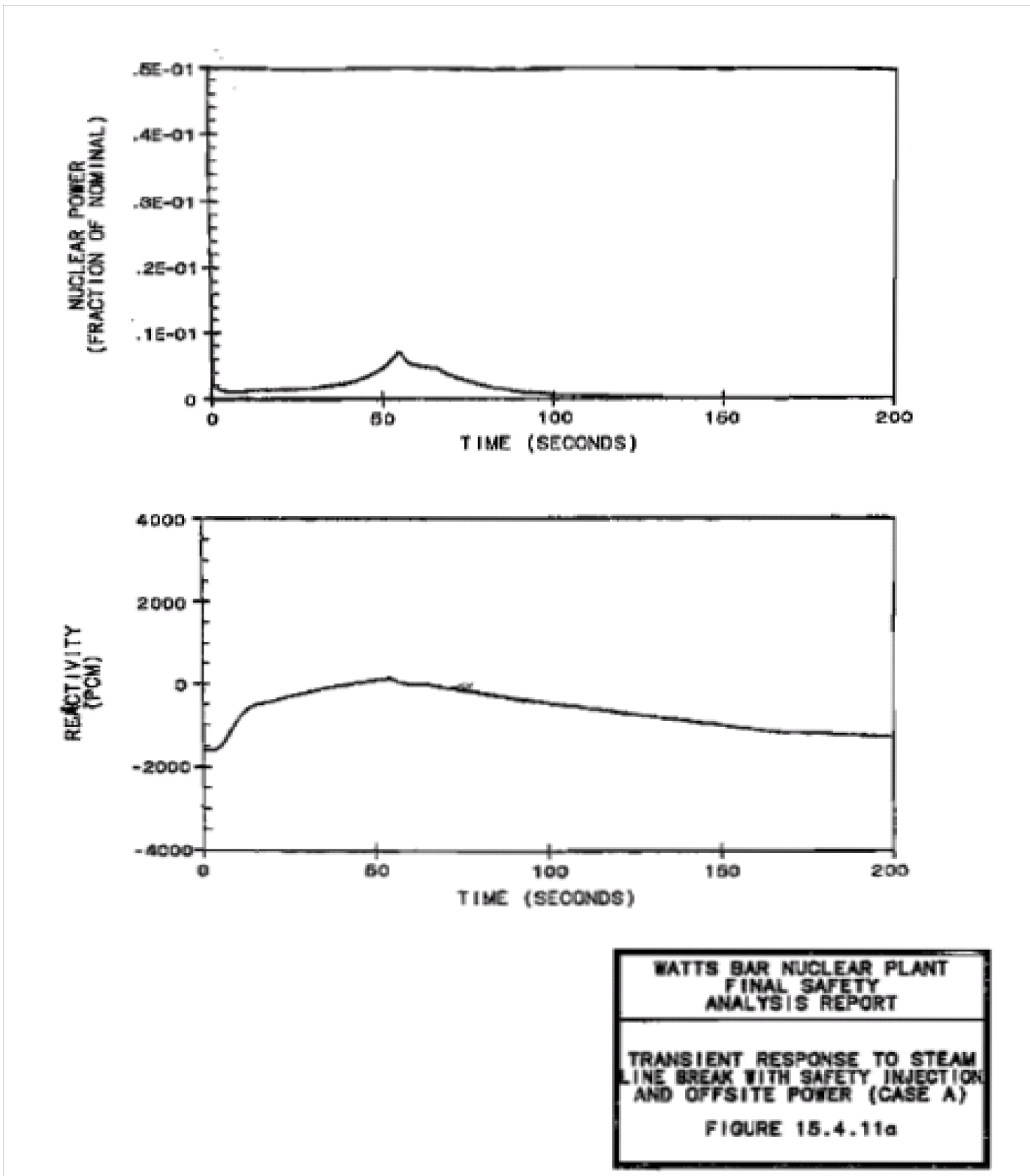


Figure 15.4-11a Transient Response to Steam Line Break with Safety Injection and Offsite Power (CASE A)

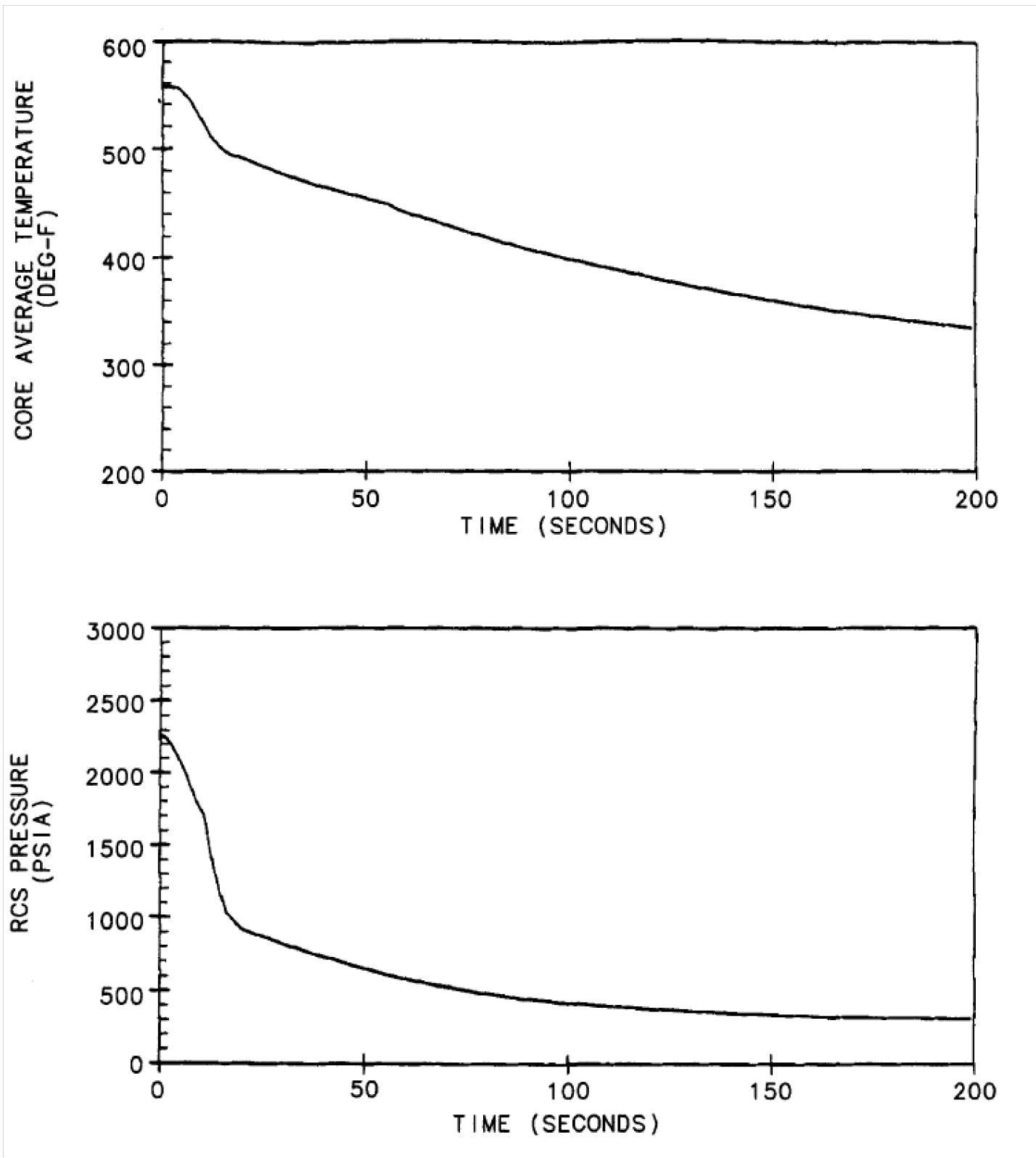
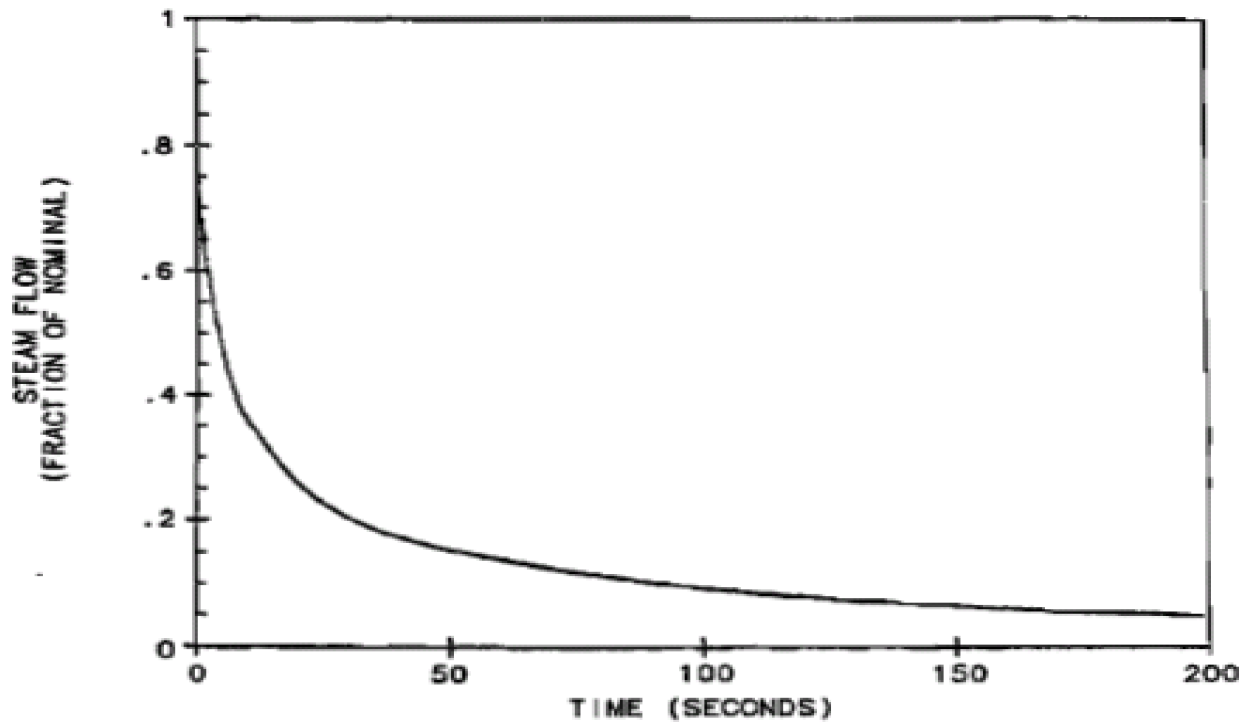


Figure 15.4-11b Transient Response to Steam Line Break with Safety Injection and Offsite Power (CASE A)



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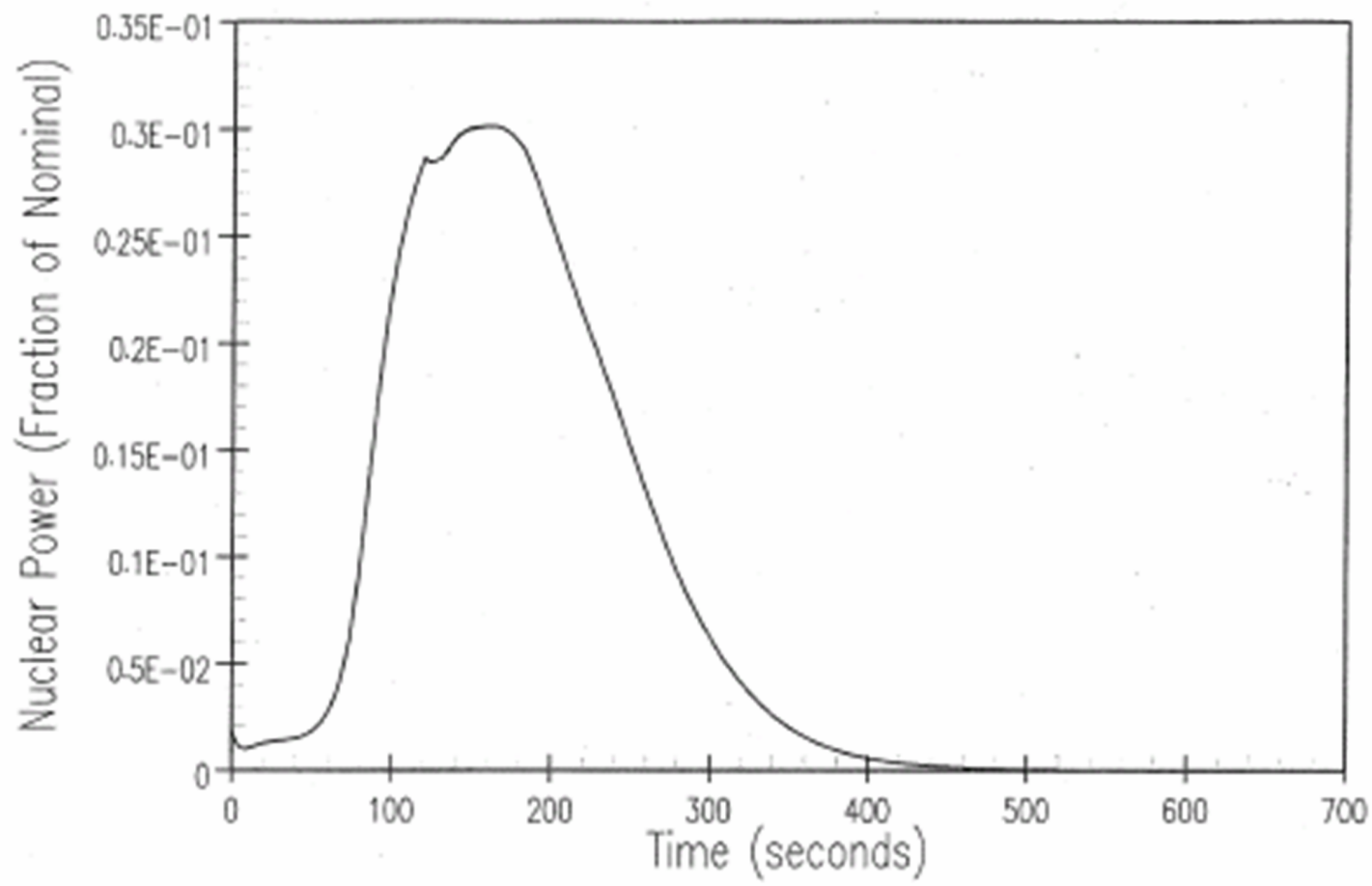
TRANSIENT RESPONSE TO STEAM LINE BREAK WITH SAFETY INJECTION AND OFFSITE POWER (CASE A)

FIGURE 15.4.11c

Figure 15.4-11c Transient Response to Steam Line Break with Safety Injection and Offsite Power (CASE A)



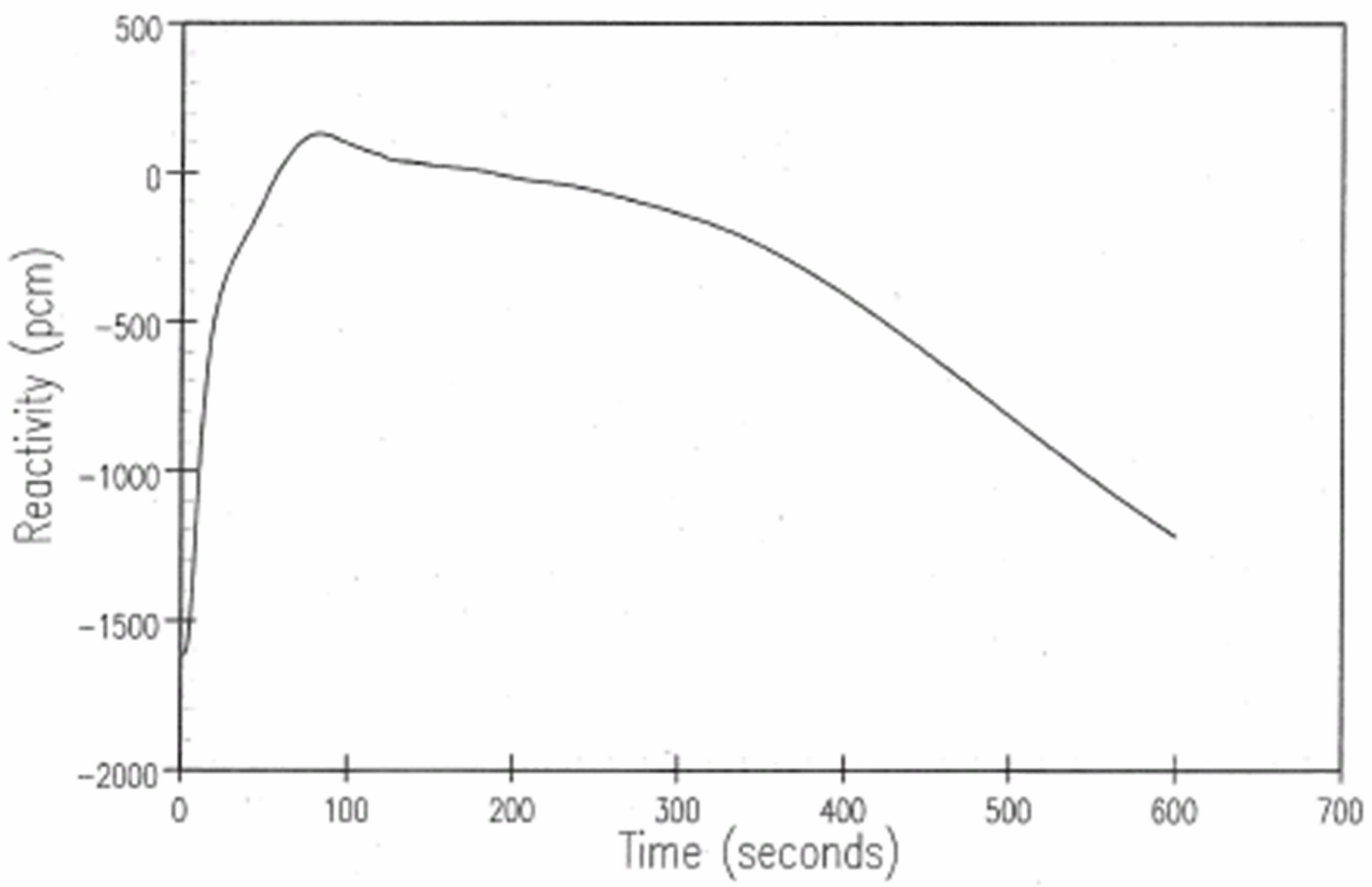
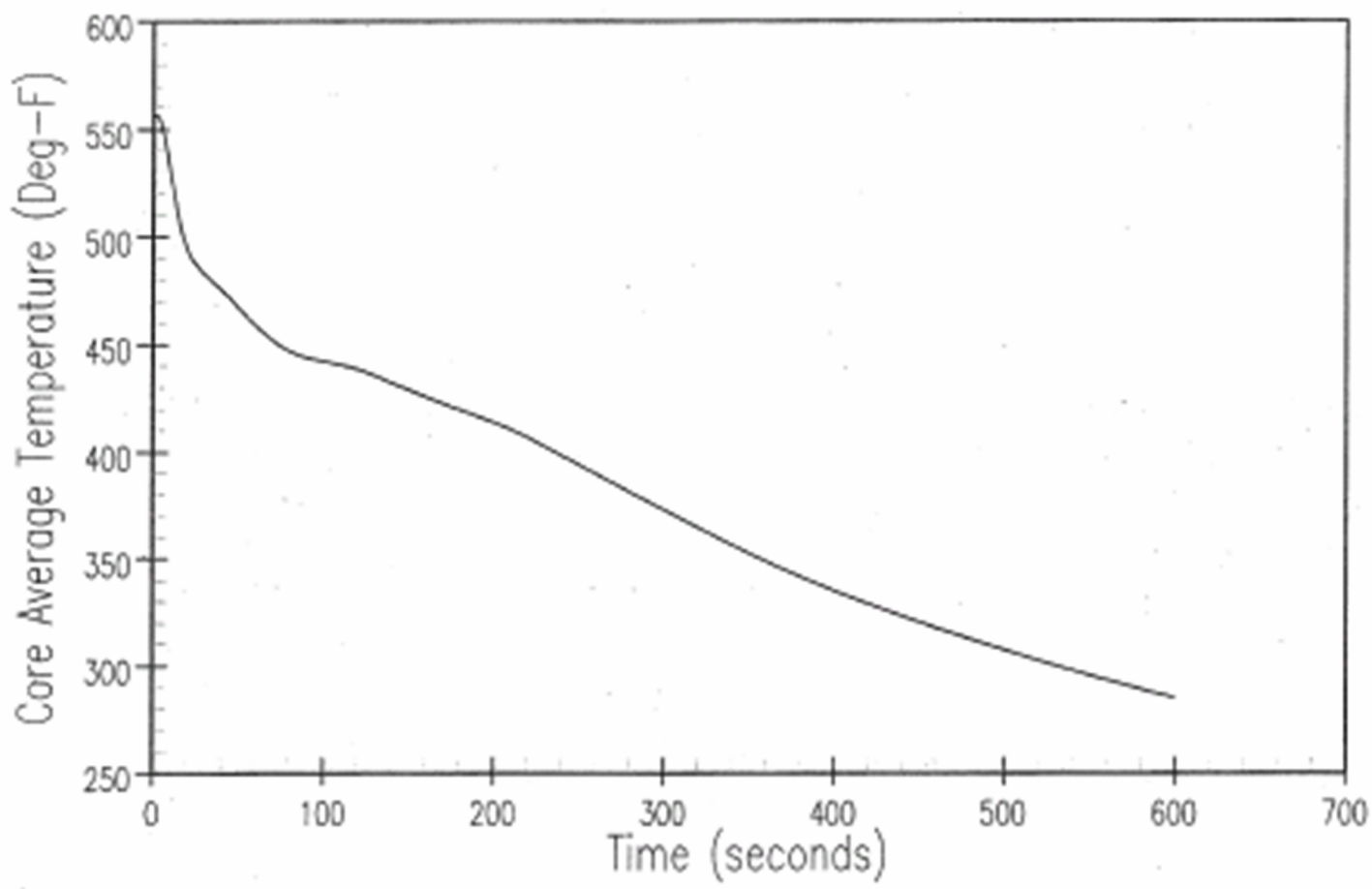


Figure 15.4-12a Transient Response to Steam Line Break without Offsite Power Nuclear Power and Reactivity Versus Time



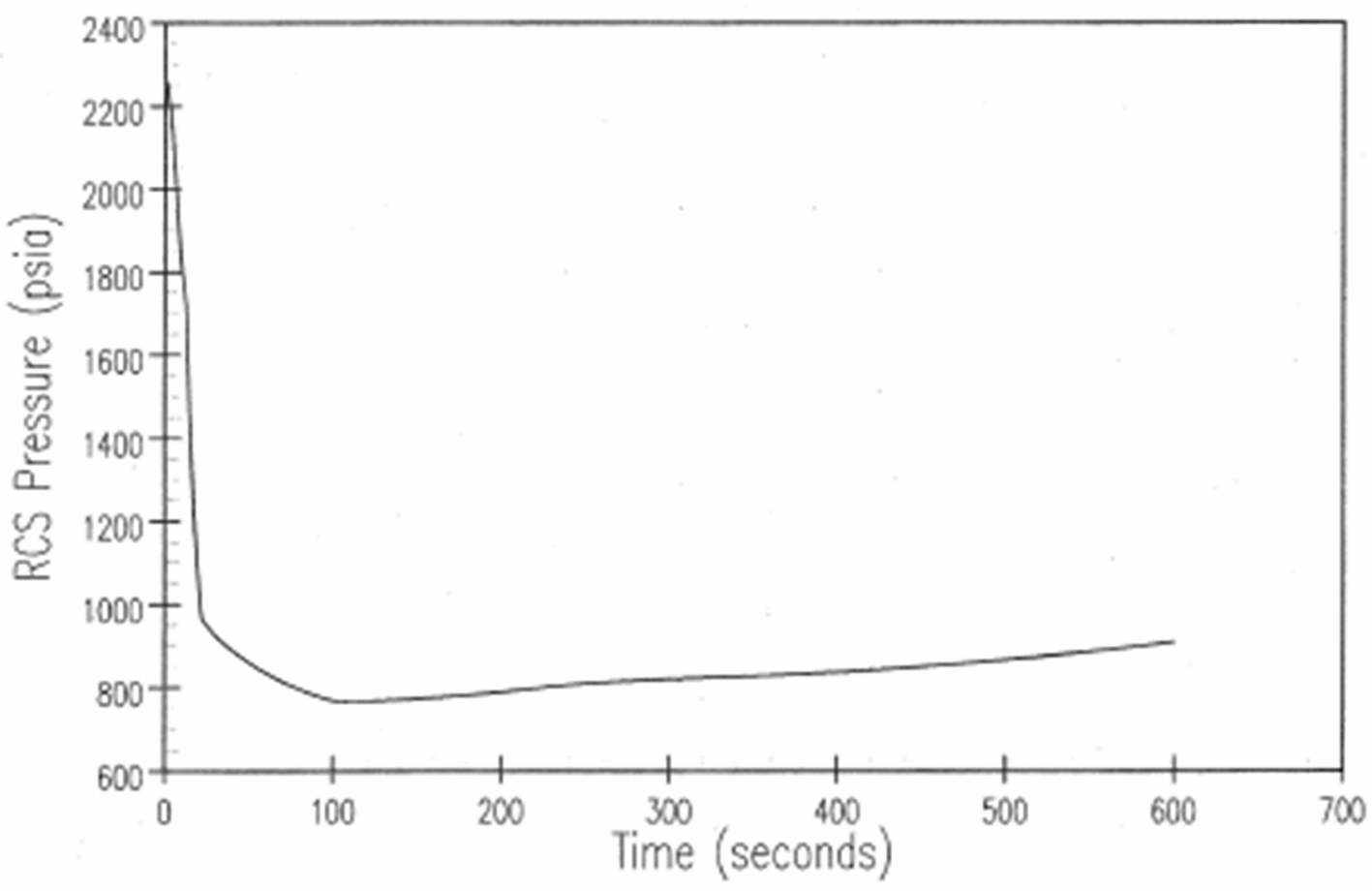


Figure 15.4-12b Transient Response to Steam Line Break without Offsite Power Core Average Temperature and RCS Pressure Versus Time

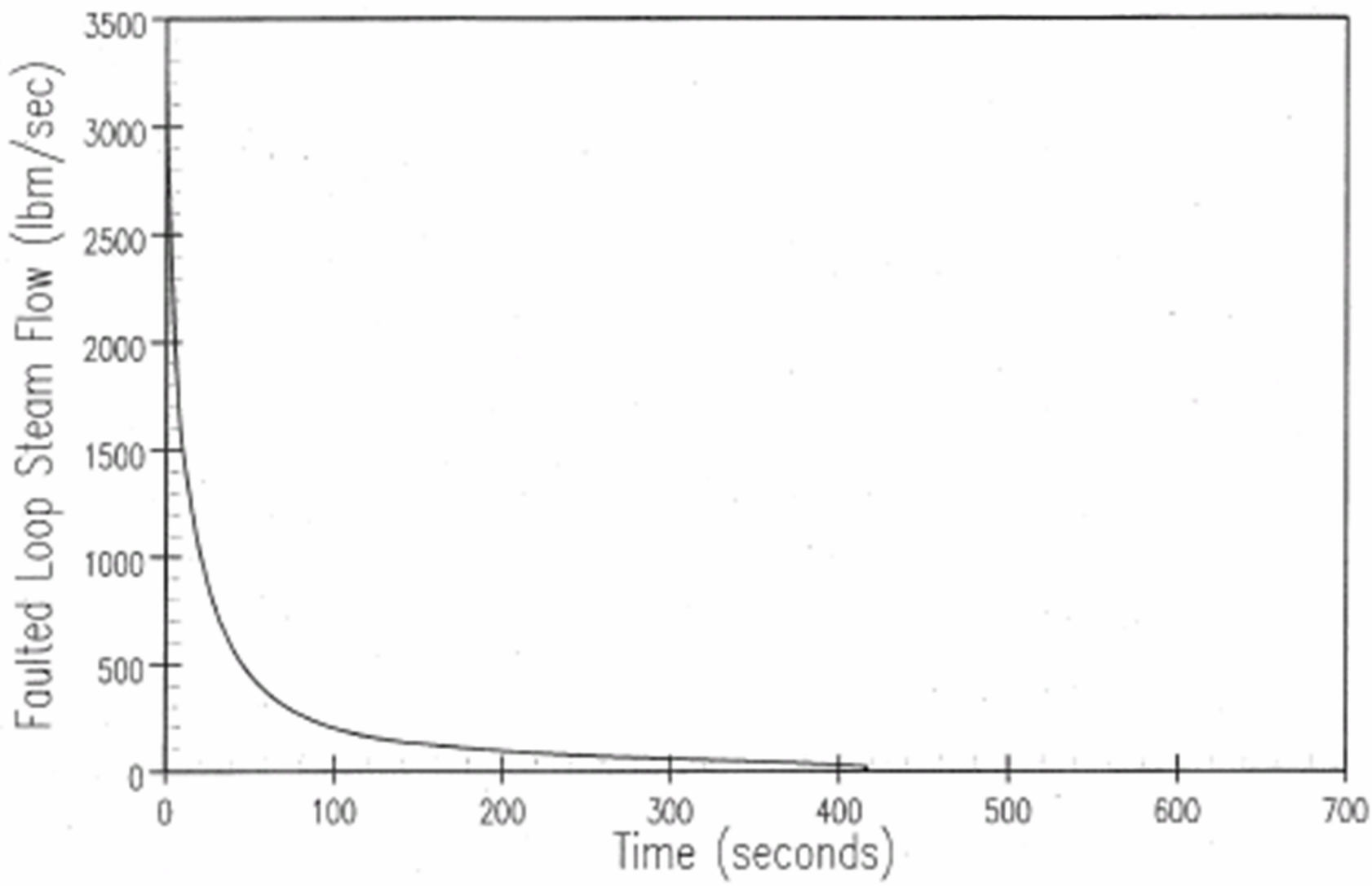


Figure 15.4-12c Transient Response to Steam Line Break without Offsite Power Faulted Loop Steam Flow Versus Time

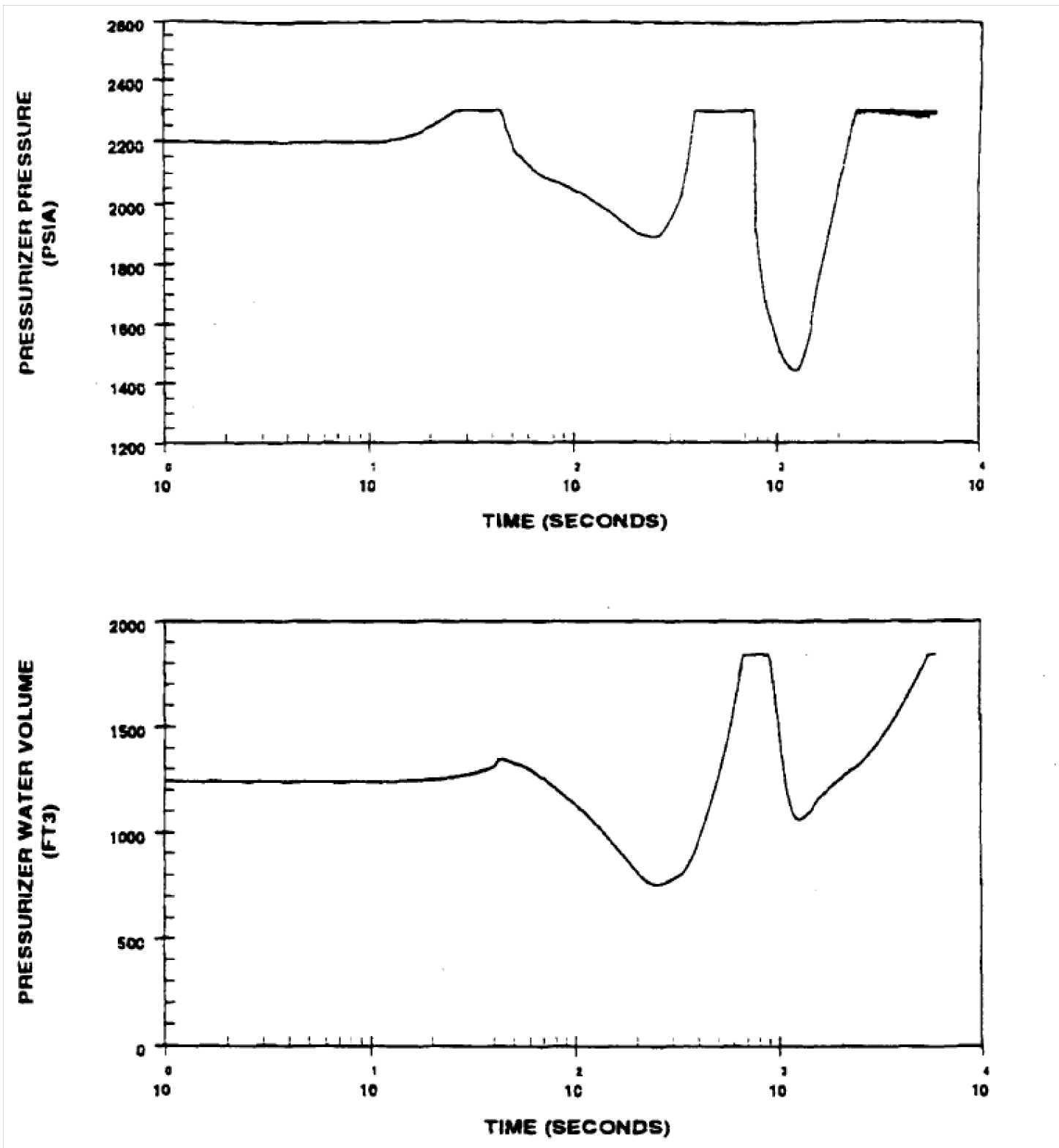


Figure 15.4-13a Pressurizer Pressure and Water Volume Transients for Main Feedline Rupture With Offsite Power

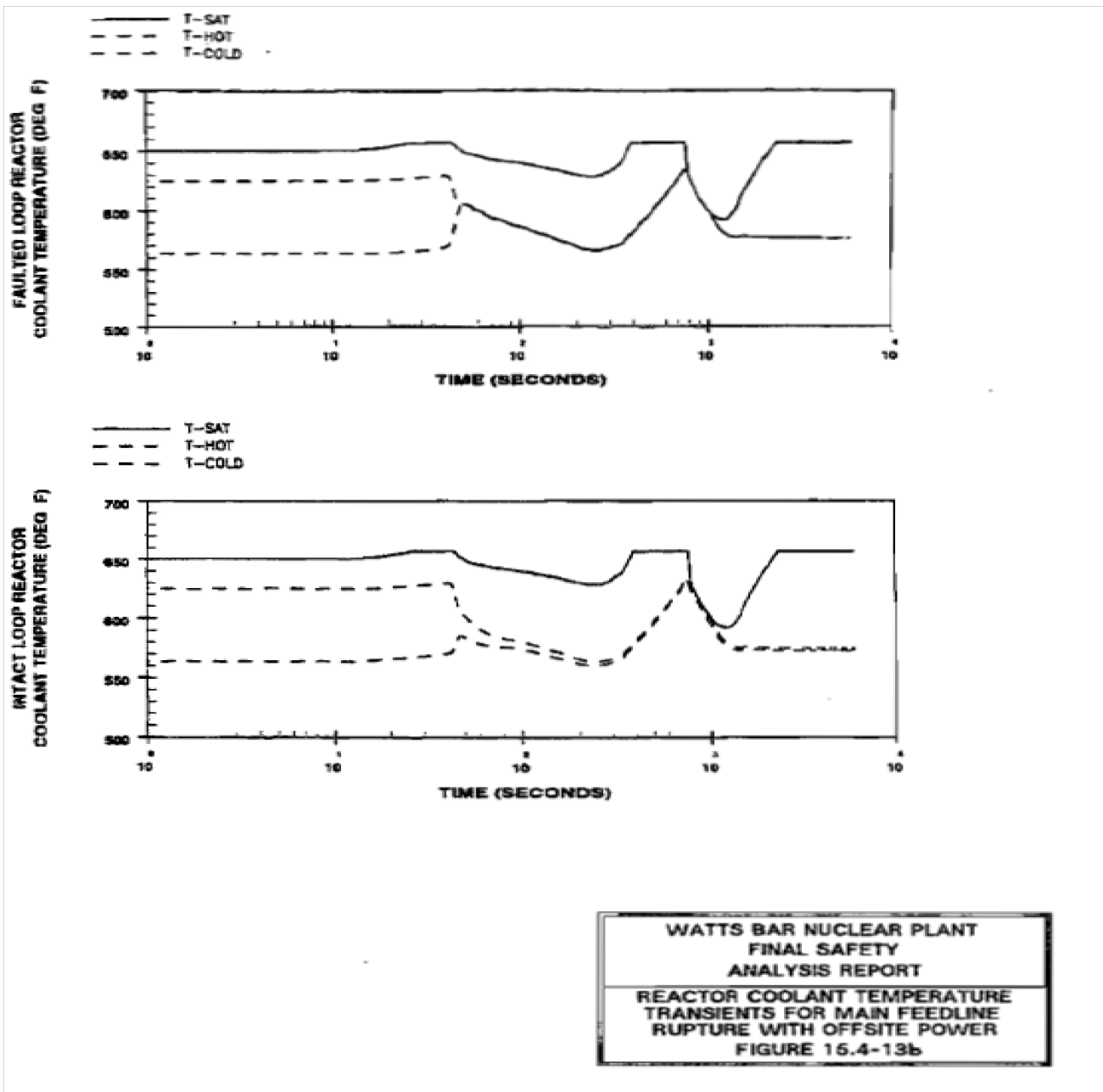


Figure 15.4-13b Reactor Coolant Temperature Transients Main Feedline Rupture With Offsite Power

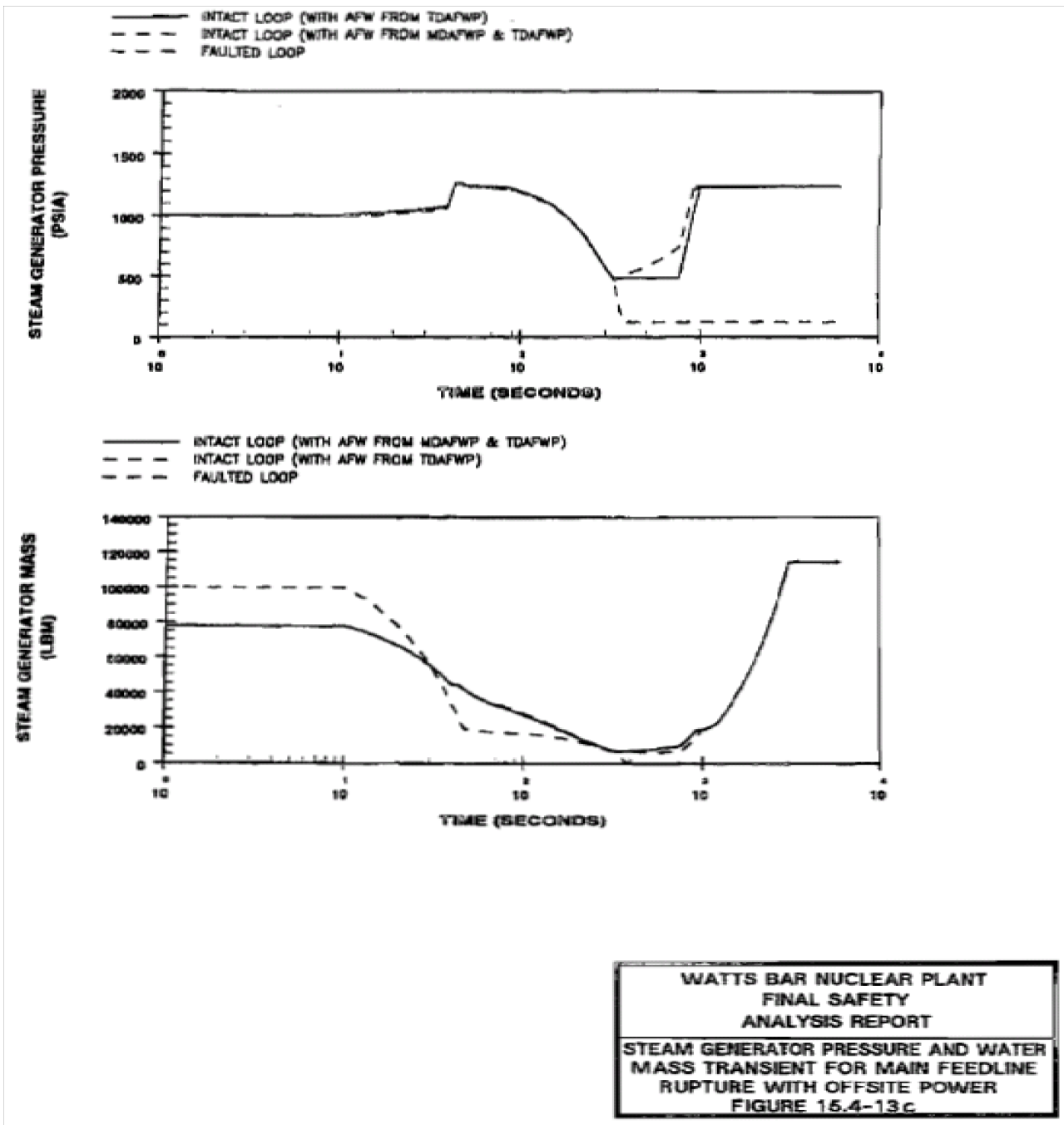


Figure 15.4-13c Steam Generator Pressure and Water Mass Transients for Main Feedline Rupture With Offsite Power

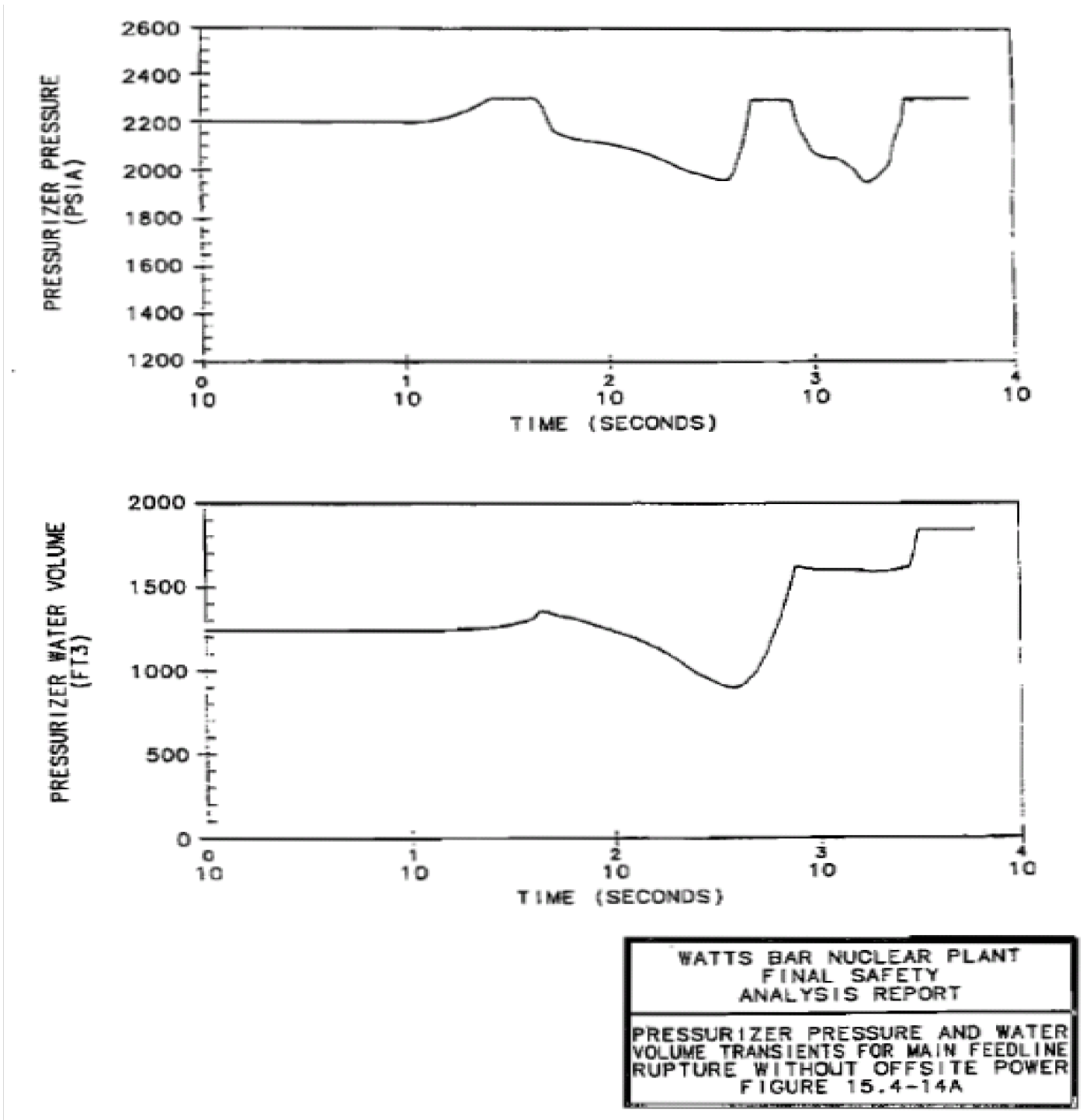


Figure 15.4-14A Pressurizer Pressure and Water Volume Transients for Main Feedline Rupture Without Offsite Power

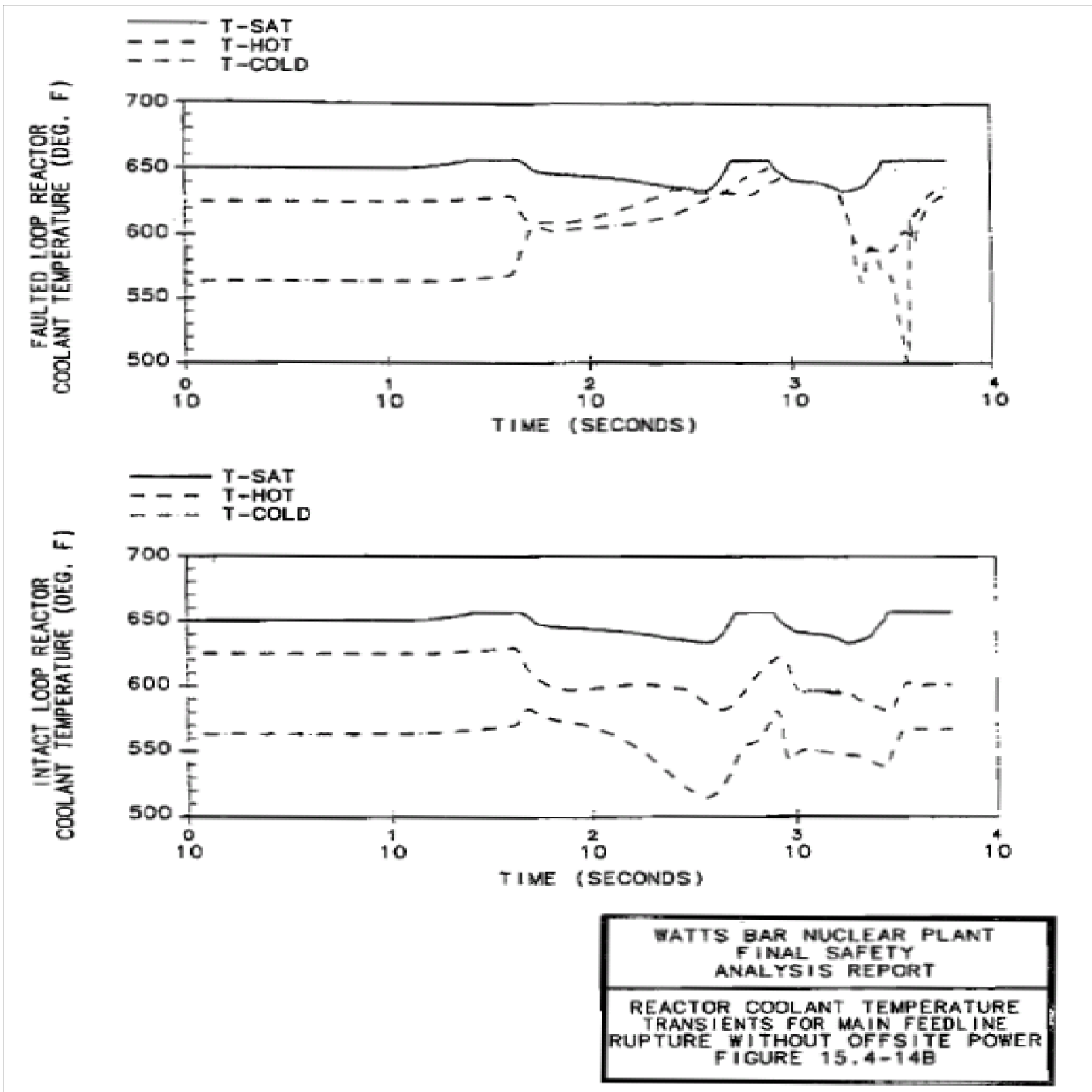


Figure 15.4-14B Reactor Coolant Temperature Transients for Main Feedline Rupture Without Offsite Power

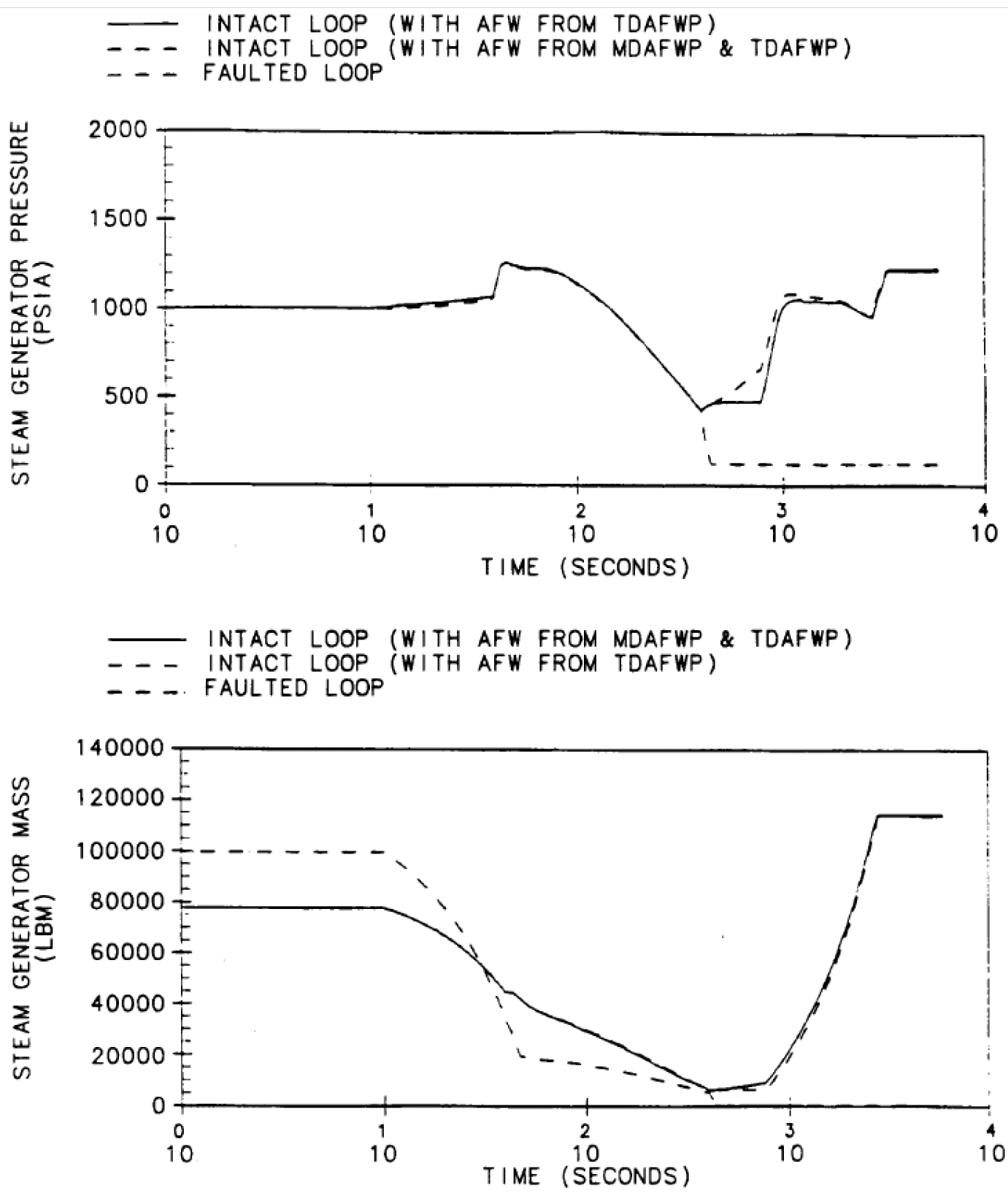
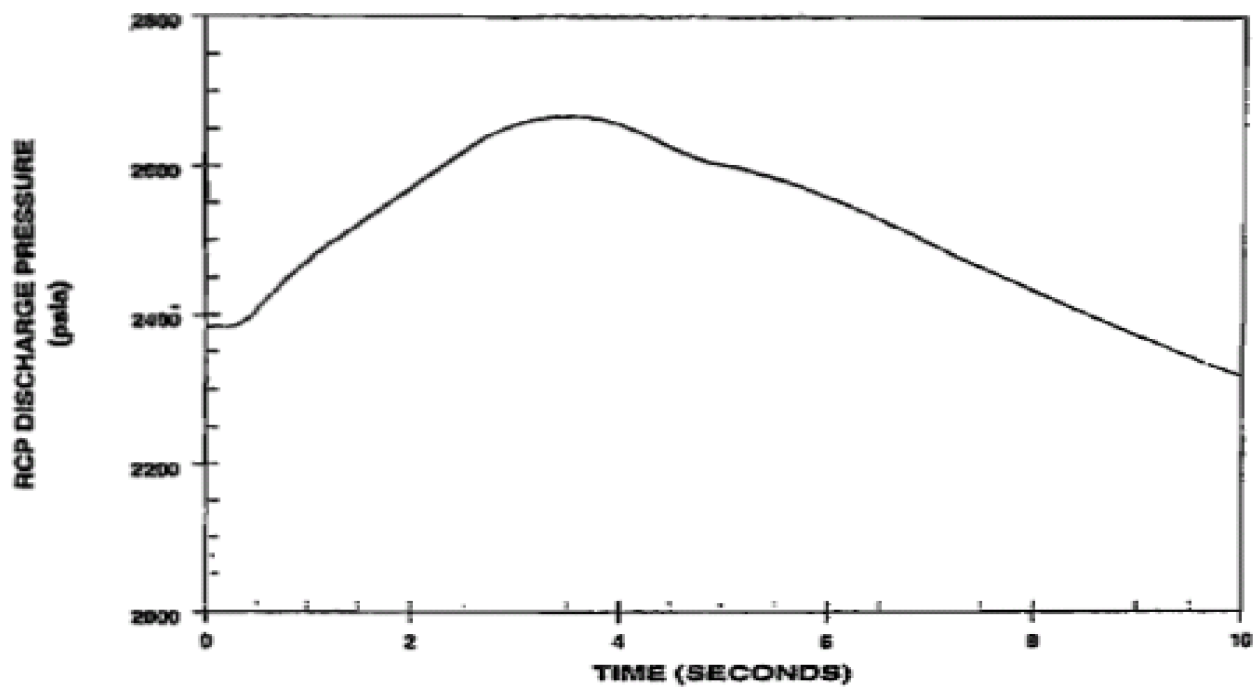


Figure 15.4-14C Steam Generator Pressure and Water Mass Transients for Main Feedline Rupture Without Offsite Power



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RCS PRESSURE TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR

FIGURE 15.4-15

Figure 15.4-15 RCS Pressure Transient; Four Pumps in Operation, One Locked Rotor

Figure 15.4-16 Deleted by Amendment 80

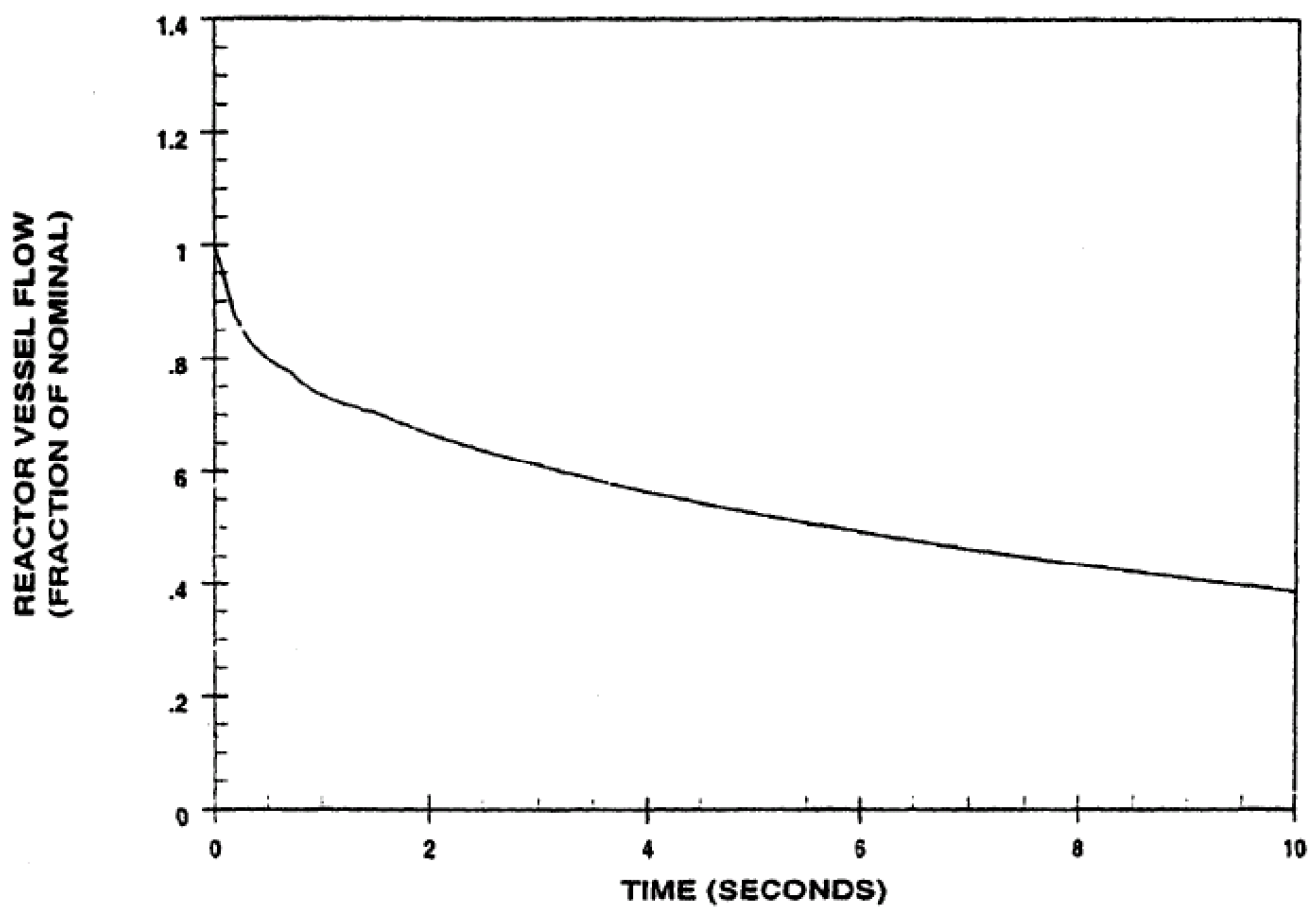


Figure 15.4-17 Reactor Vessel Flow Transient; Four Pumps in Operation, One Locked Rotor

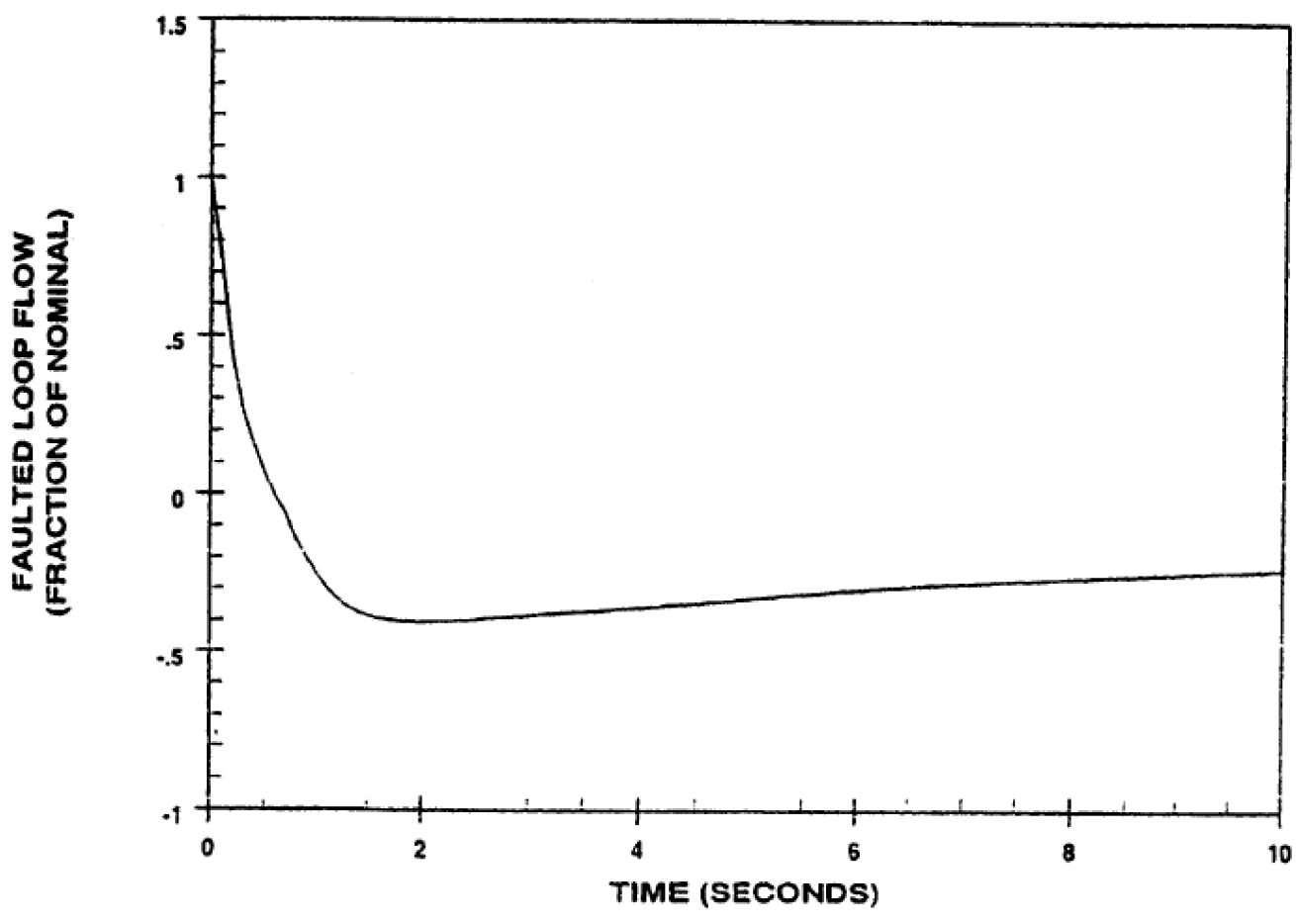
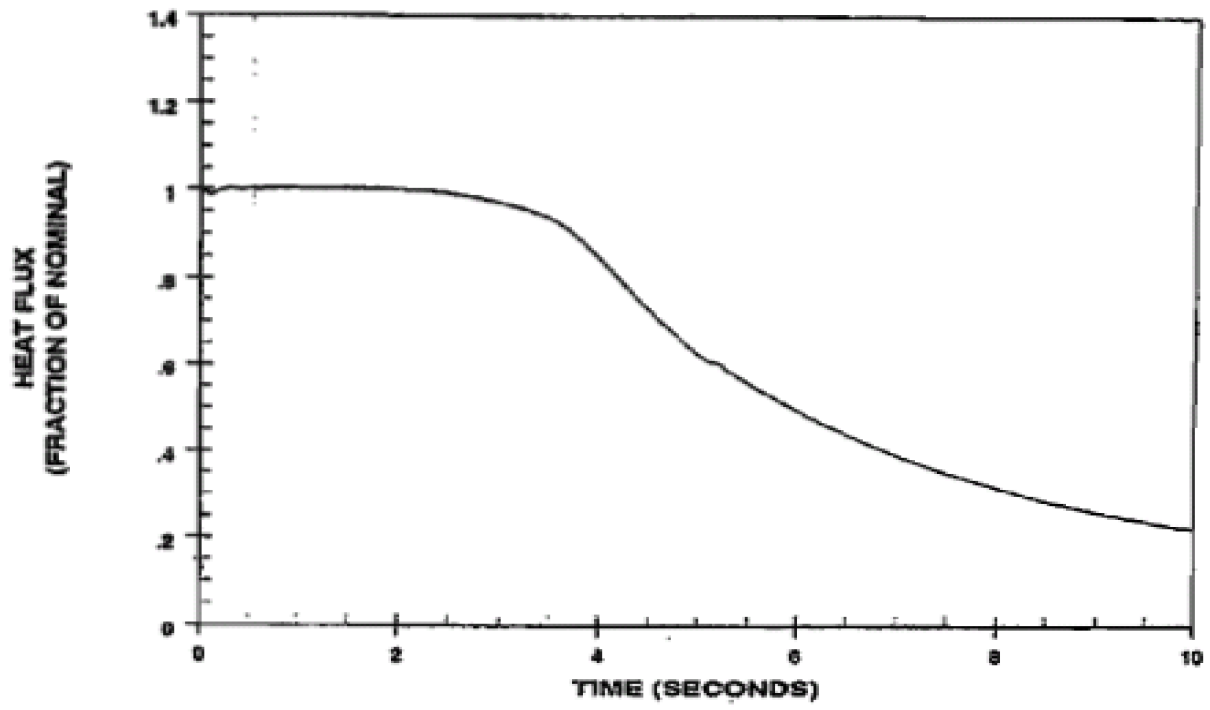


Figure 15.4-18 Loop Flow Transient; Four Pumps in Operation, One Locked Rotor



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CORE HEAT FLUX TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR

FIGURE 15.4-19

Figure 15.4-19 Core Heat Flux Transient; Four Pumps in Operation, One Locked Rotor

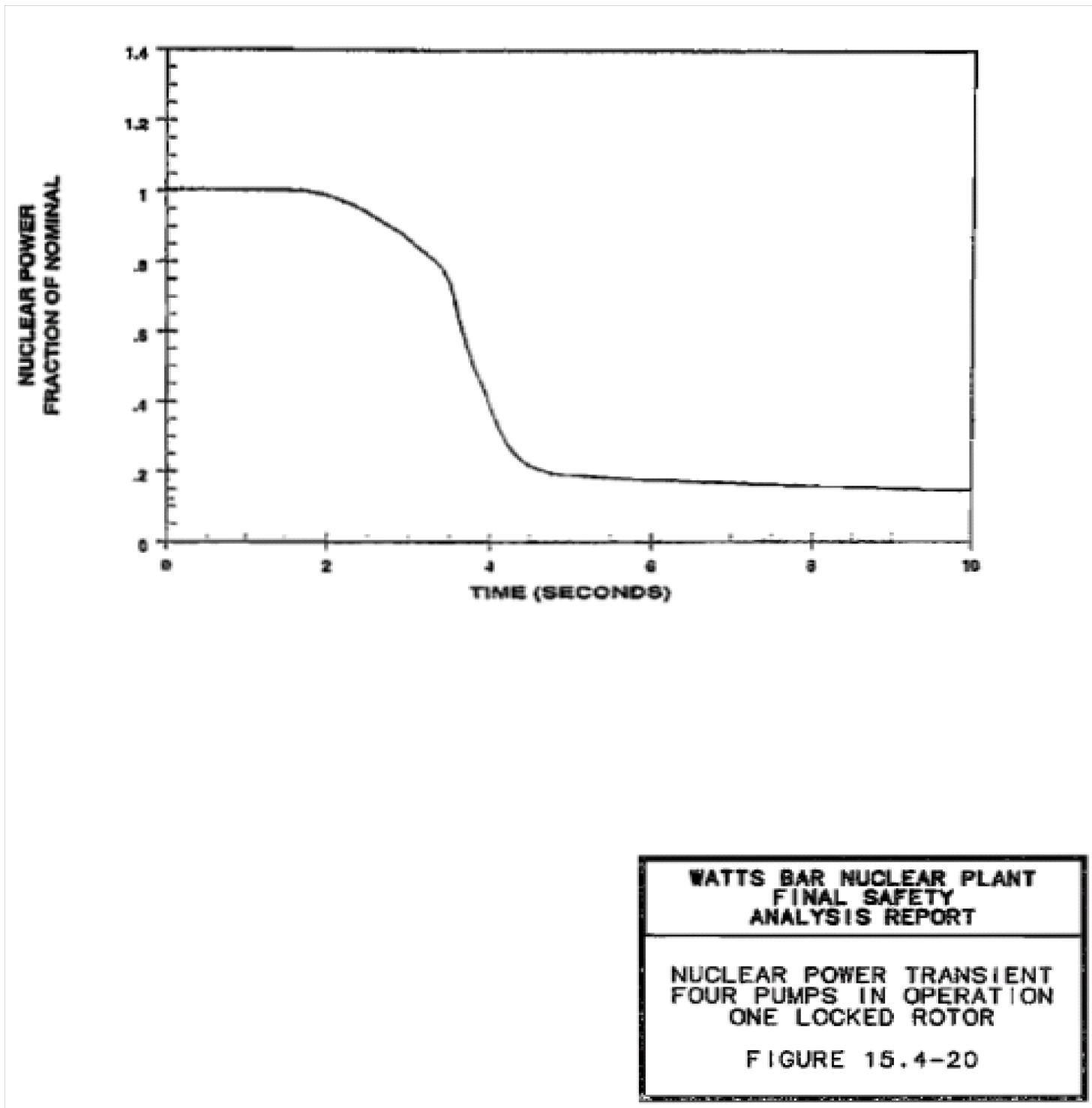
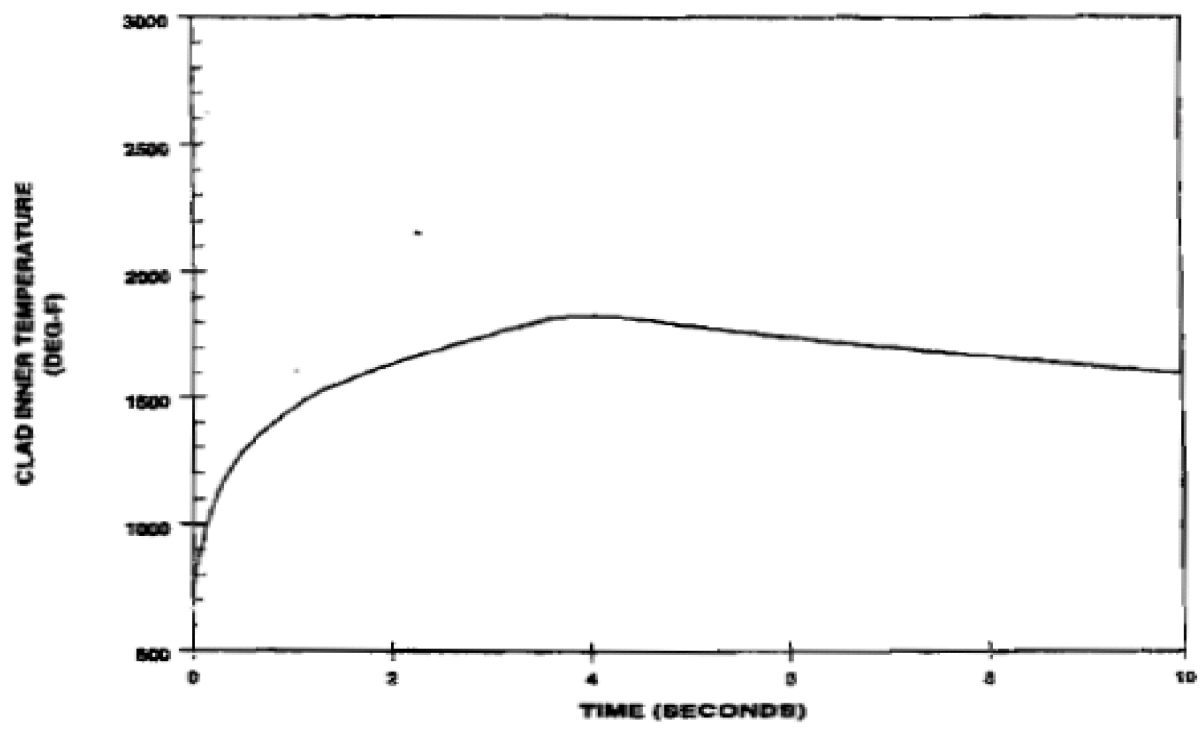


Figure 15.4-20 Nuclear Power Transient; Four Pumps in Operation, One Locked Rotor



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CLAD TEMPERATURE TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR
FIGURE 15.4-21

Figure 15.4-21 Clad Inner Temperature Transient; Four Pumps in Operation, One Locked Rotor

Figure 15.4-22 Deleted by Amendment 80

Figure 15.4-23 Deleted by Amendment 80

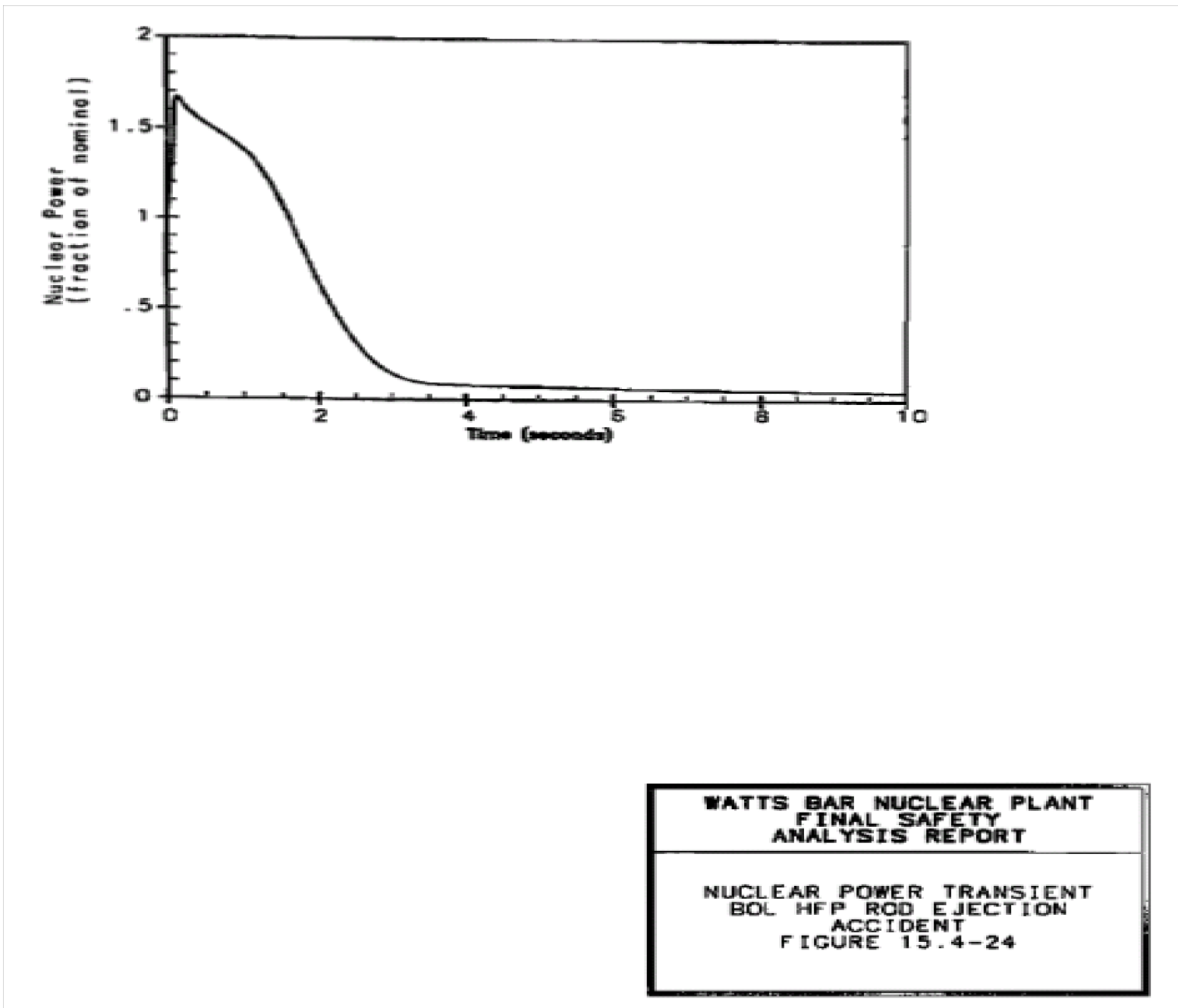
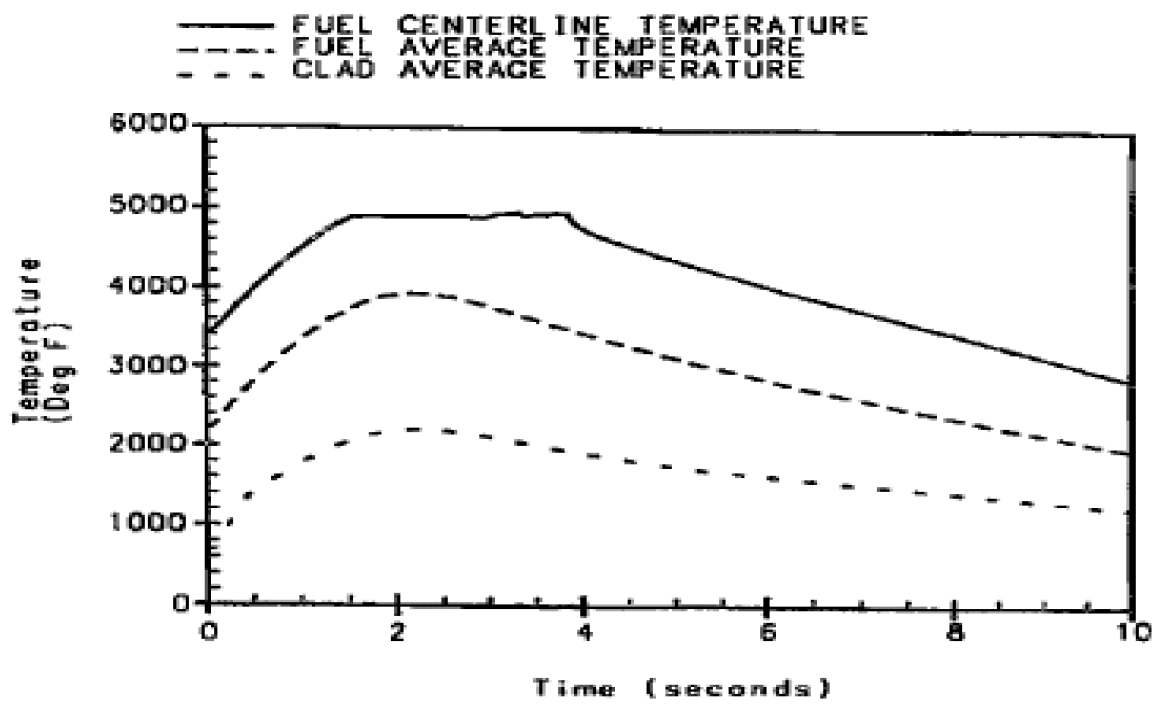


Figure 15.4-24 Nuclear Power Transient; BOL HFP Rod Ejection Accident



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HOT SPOT FUEL AND CLAD
TEMPERATURE VERSUS TIME
BOL HFP ROD EJECTION
ACCIDENT
FIGURE 15.4-25

Figure 15.4-25 Hot Spot Fuel and Clad Temperature Versus Time; BOL HFP Rod Ejection Accident

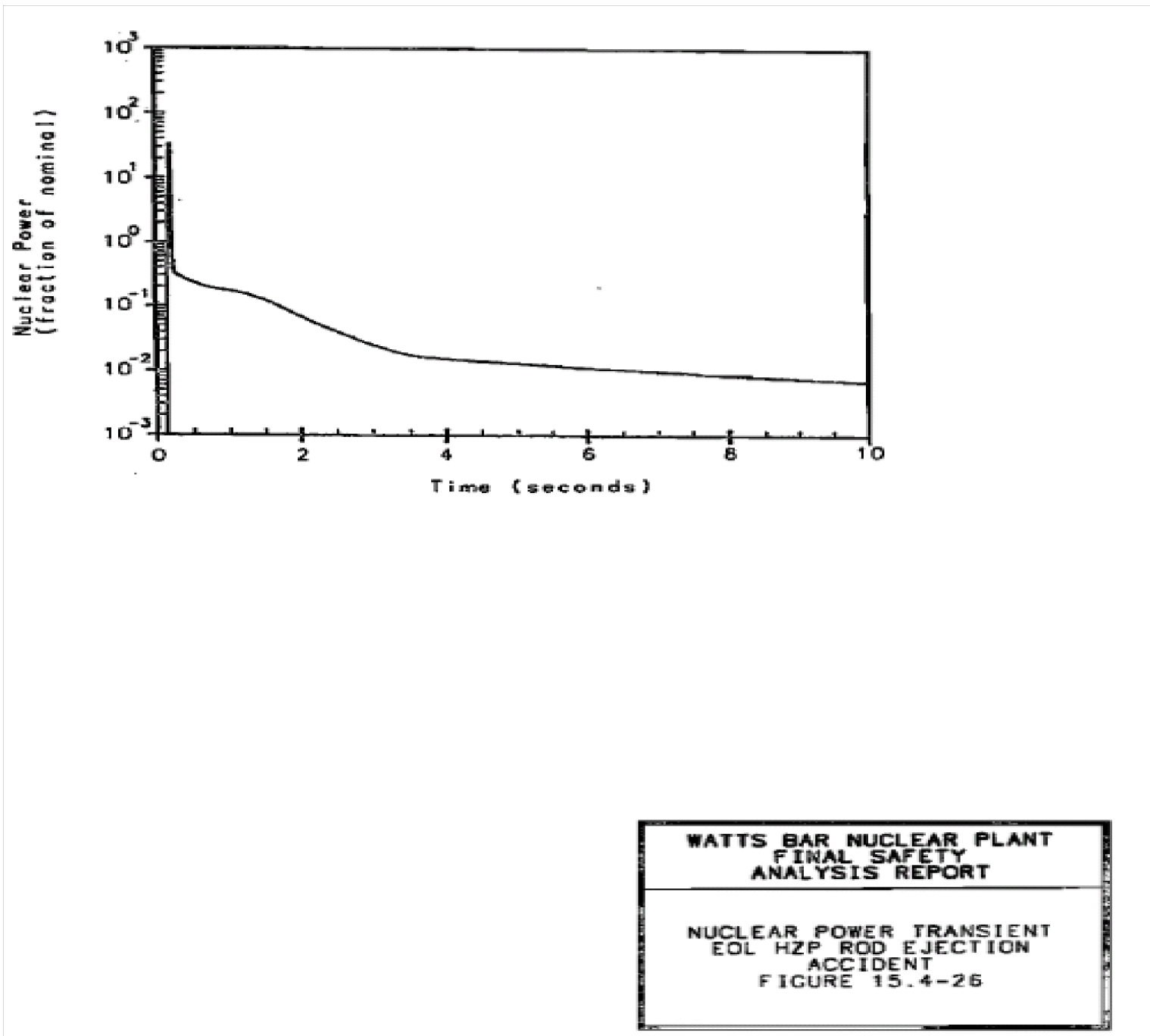


Figure 15.4-26 Nuclear Power Transient; EOL HZP Rod Ejection Accident

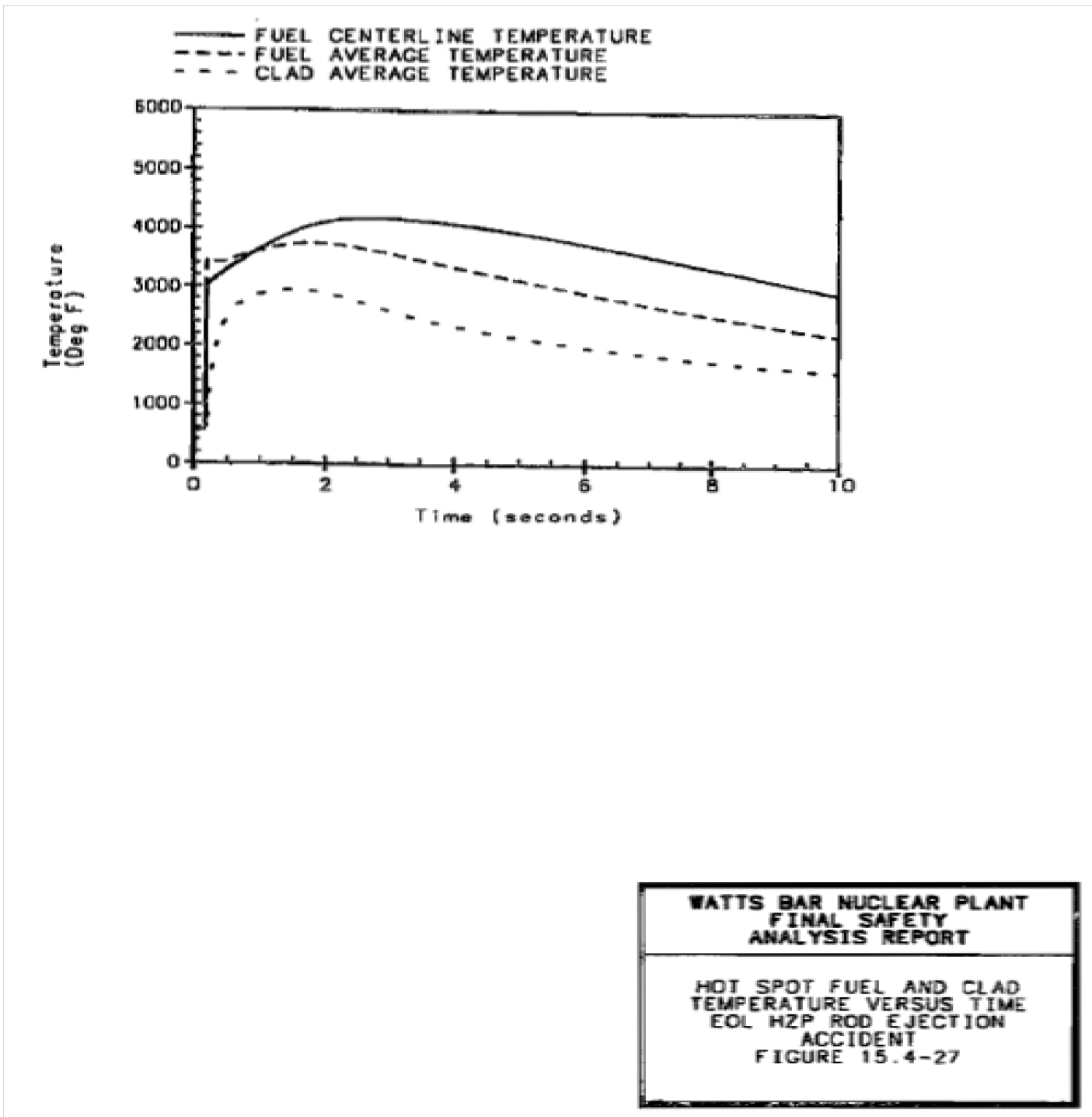


Figure 15.4-27 Hot Spot Fuel and Clad Temperature Versus Time; EOL HZP Rod Ejection Accident

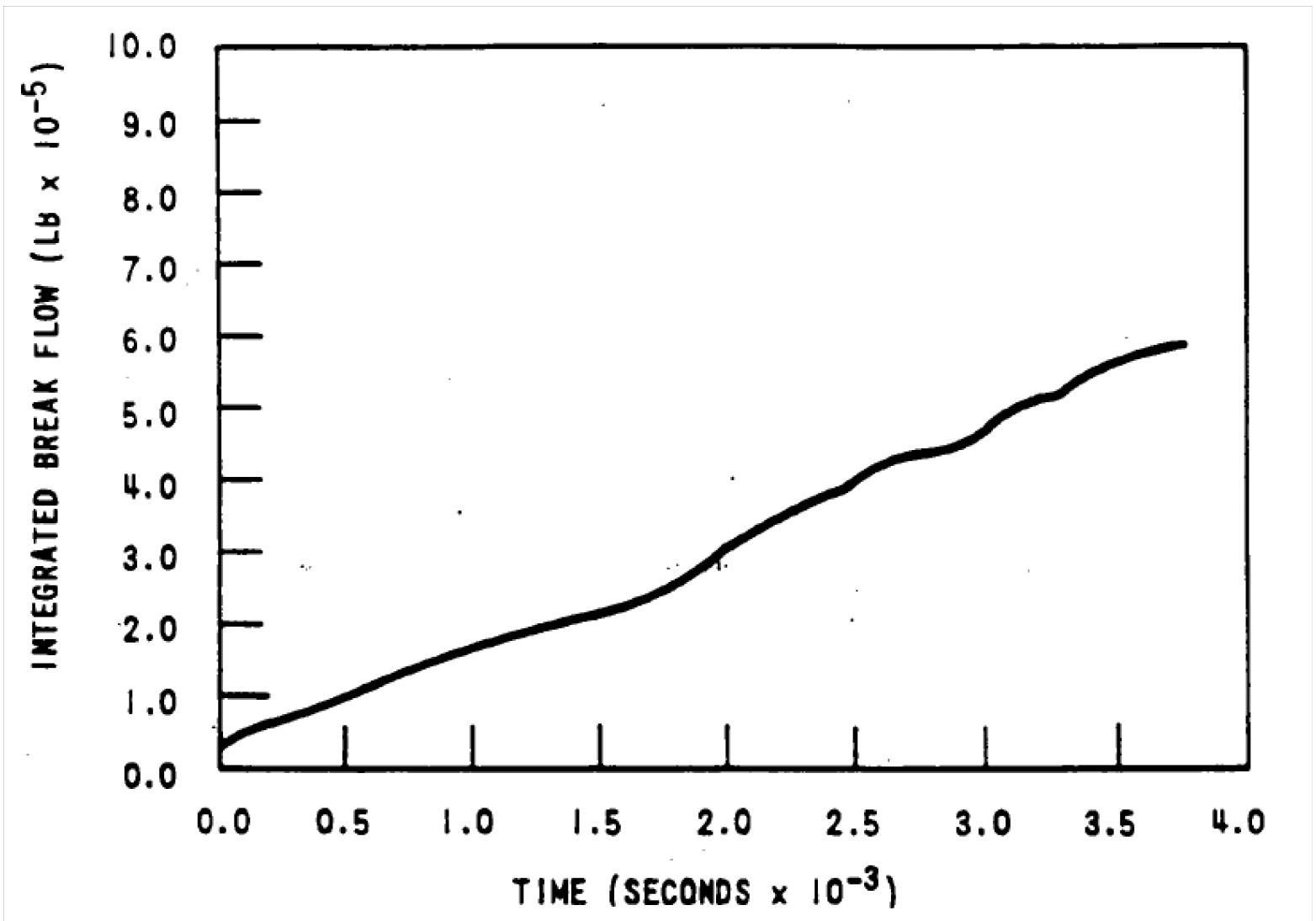


Figure 15.4-28 Reactor Coolant System Integrated Break Flow Following a Rod Ejection Accident

Figure 15.4-29 Deleted by Amendment 97

Figure 15.4-30 Deleted by Amendment97

Figure 15.4-31 Deleted by Amendment9

Figure 15.4-32 Deleted by Amendment 97

Figure 15.4-33 Deleted by Amendment 97

Figure 15.4-34 Deleted by Amendment 97

Figure 15.4-35 Deleted by Amendment 97

Figure 15.4-36 Deleted by Amendment 97

Figure 15.4-37 Deleted by Amendment 97

Figure 15.4-38 Deleted by Amendment 97

Figure 15.4-39 Deleted by Amendment 97

Figure 15.4-40a Deleted by Amendment 97

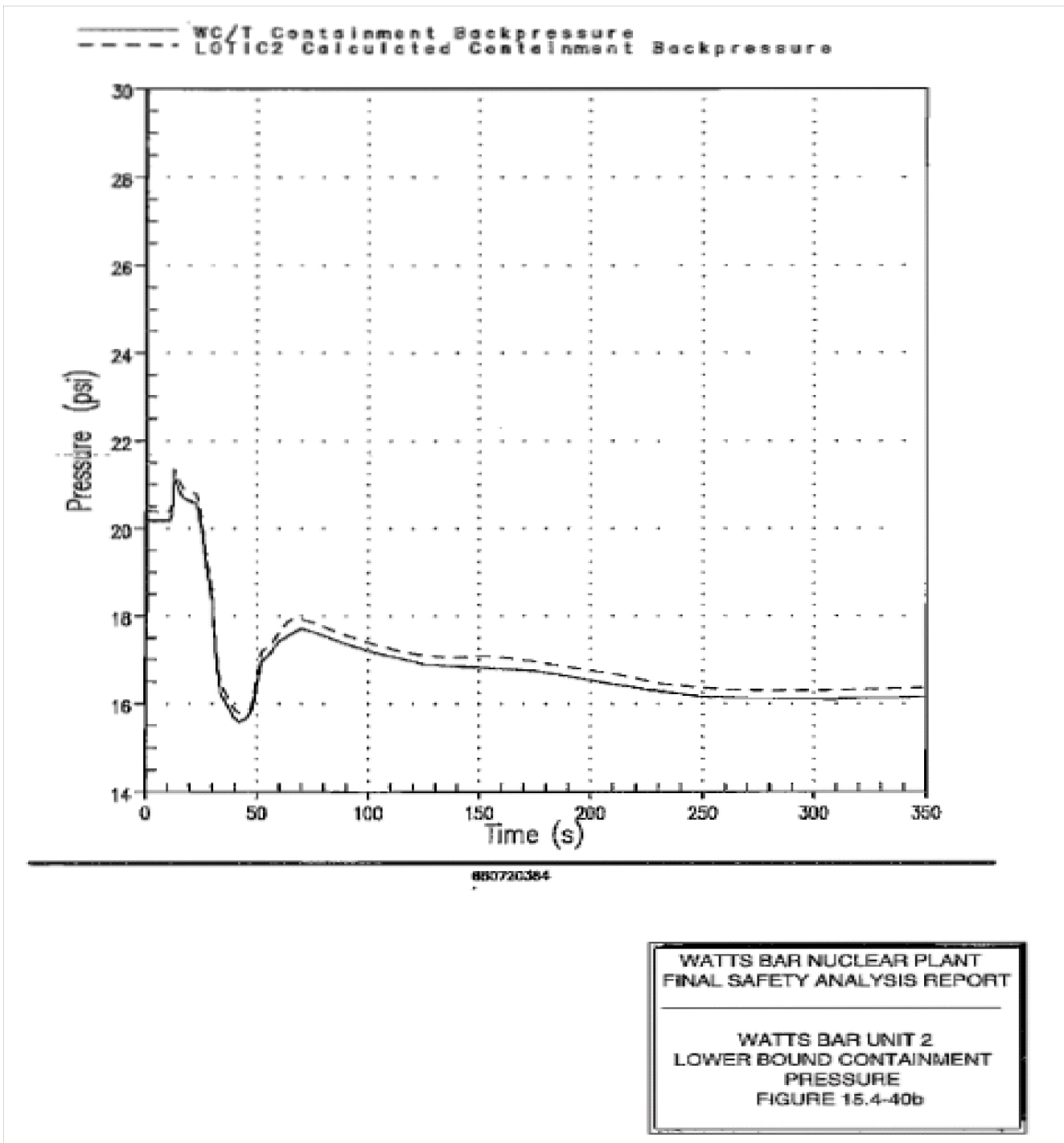


Figure 15.4-40b Watts Bar Unit 2 Lower Bound Containment Pressure

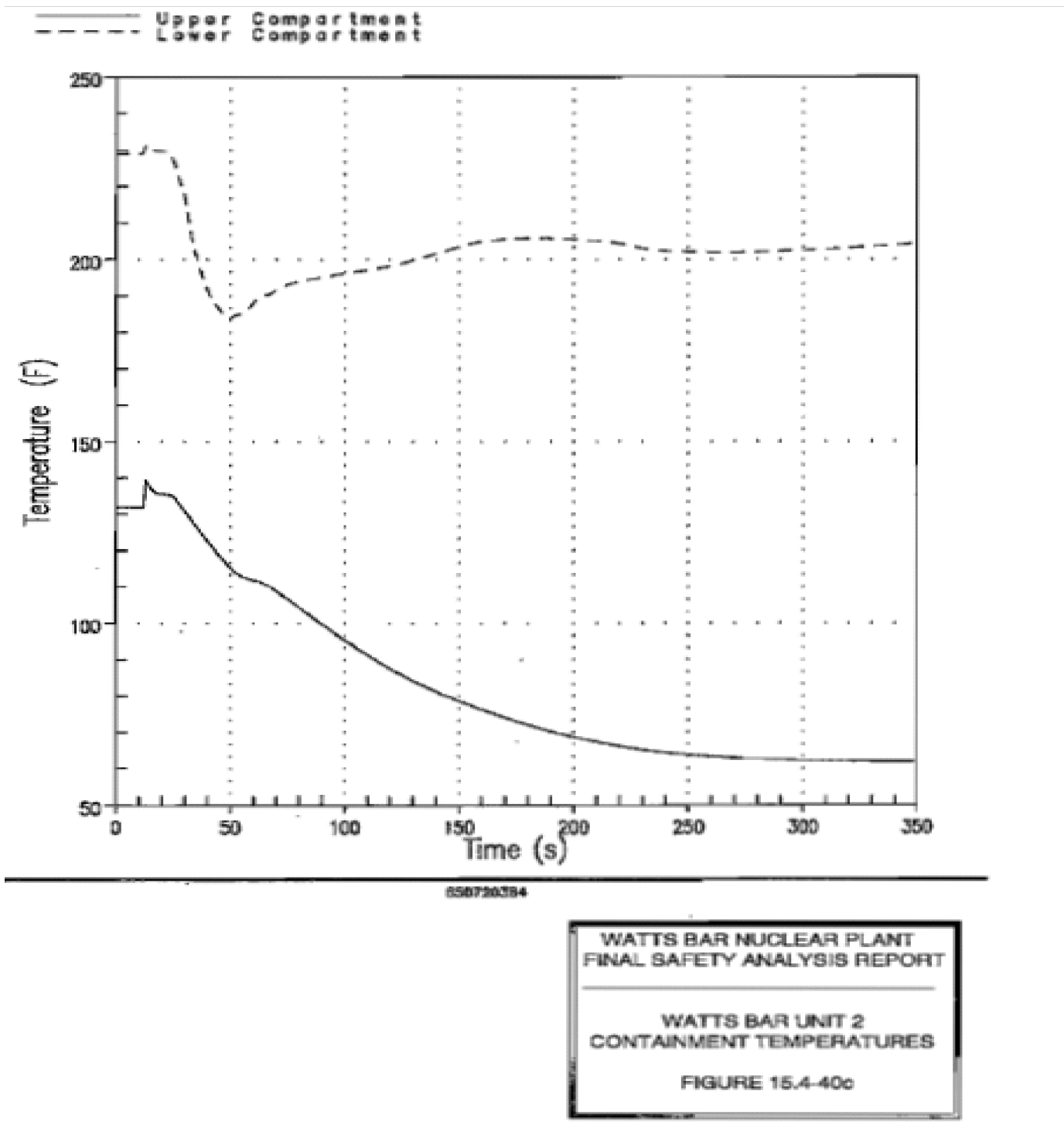


Figure 15.4-40c Watts Bar Unit 2 Containment Temperatures

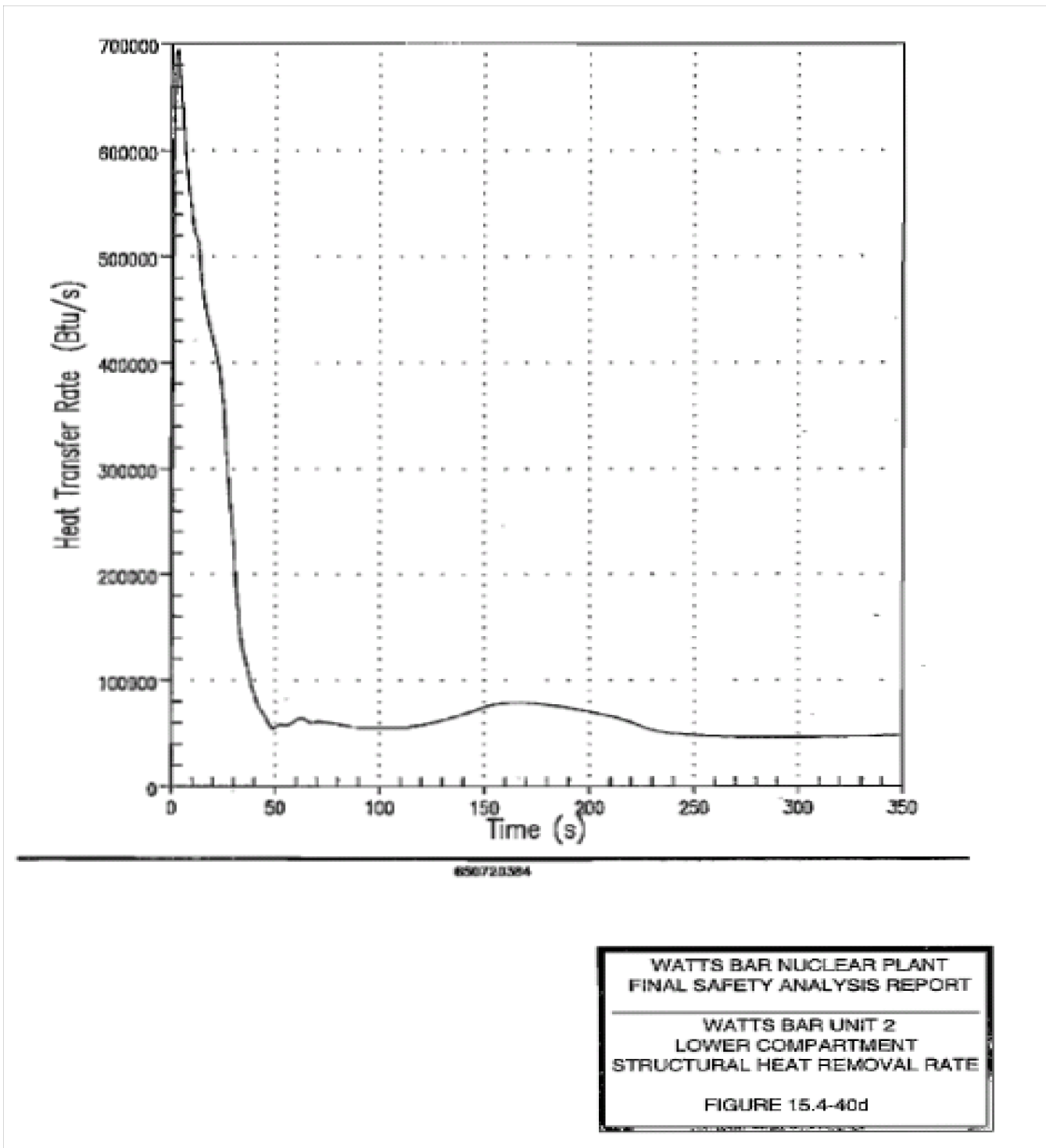


Figure 15.4-40d Watts Bar Unit 2 Lower Compartment Structural Heat Removal Rate

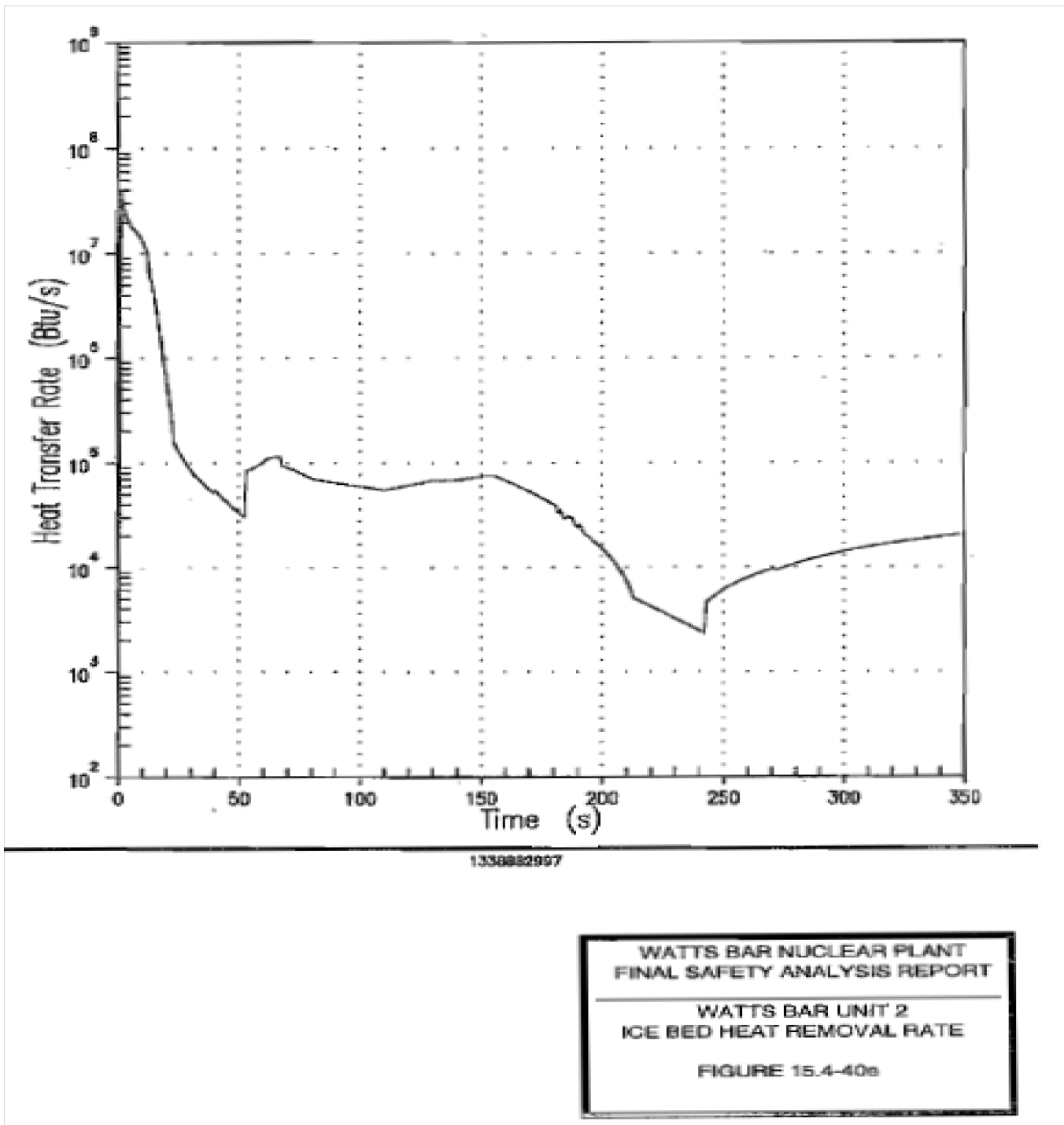


Figure 15.4-40e Watts Bar Unit 2 Ice Bed Heat Removal Rate

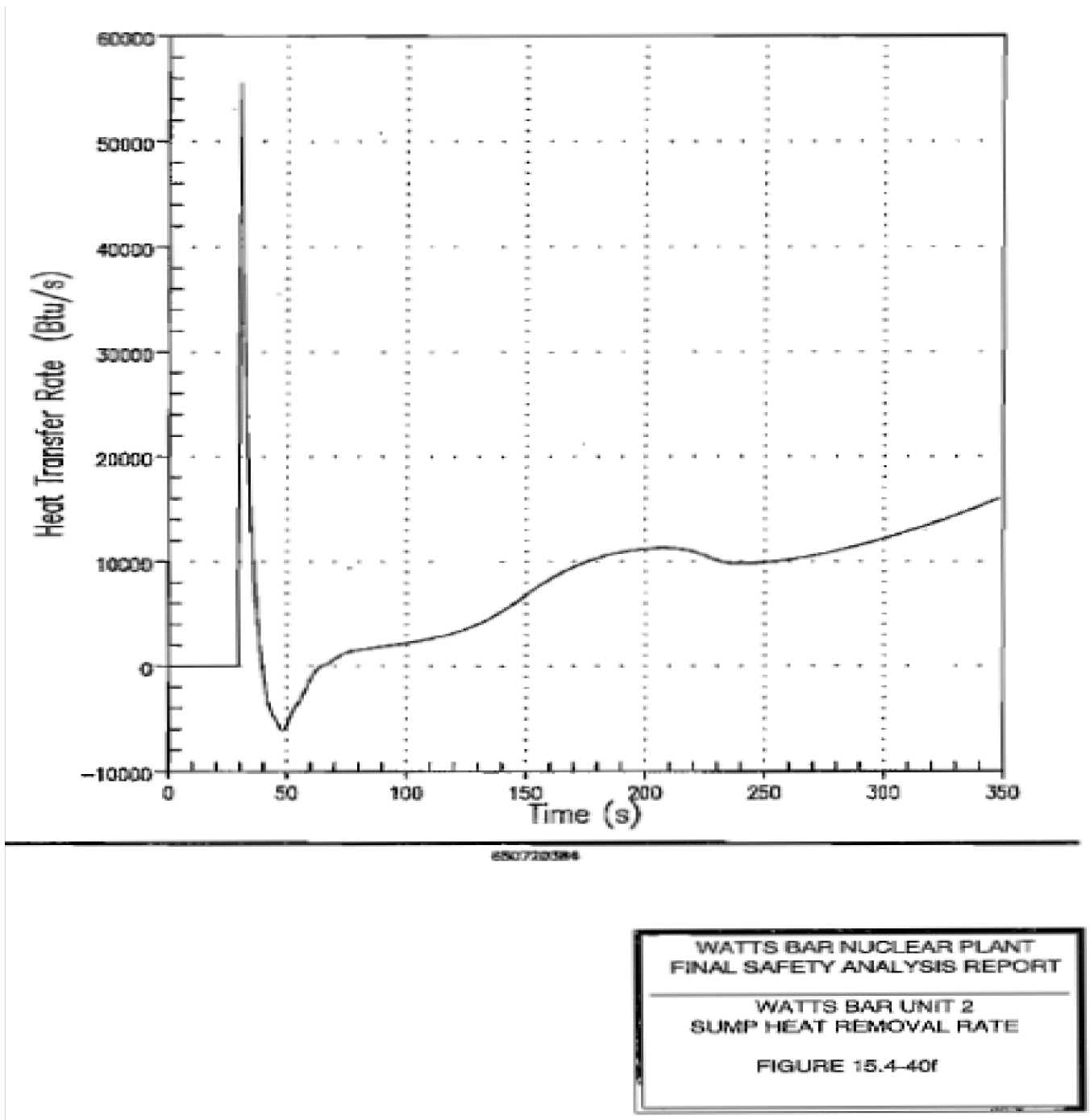


Figure 15.4-40f Watts Bar Unit2 Sump Heat Removal Rate

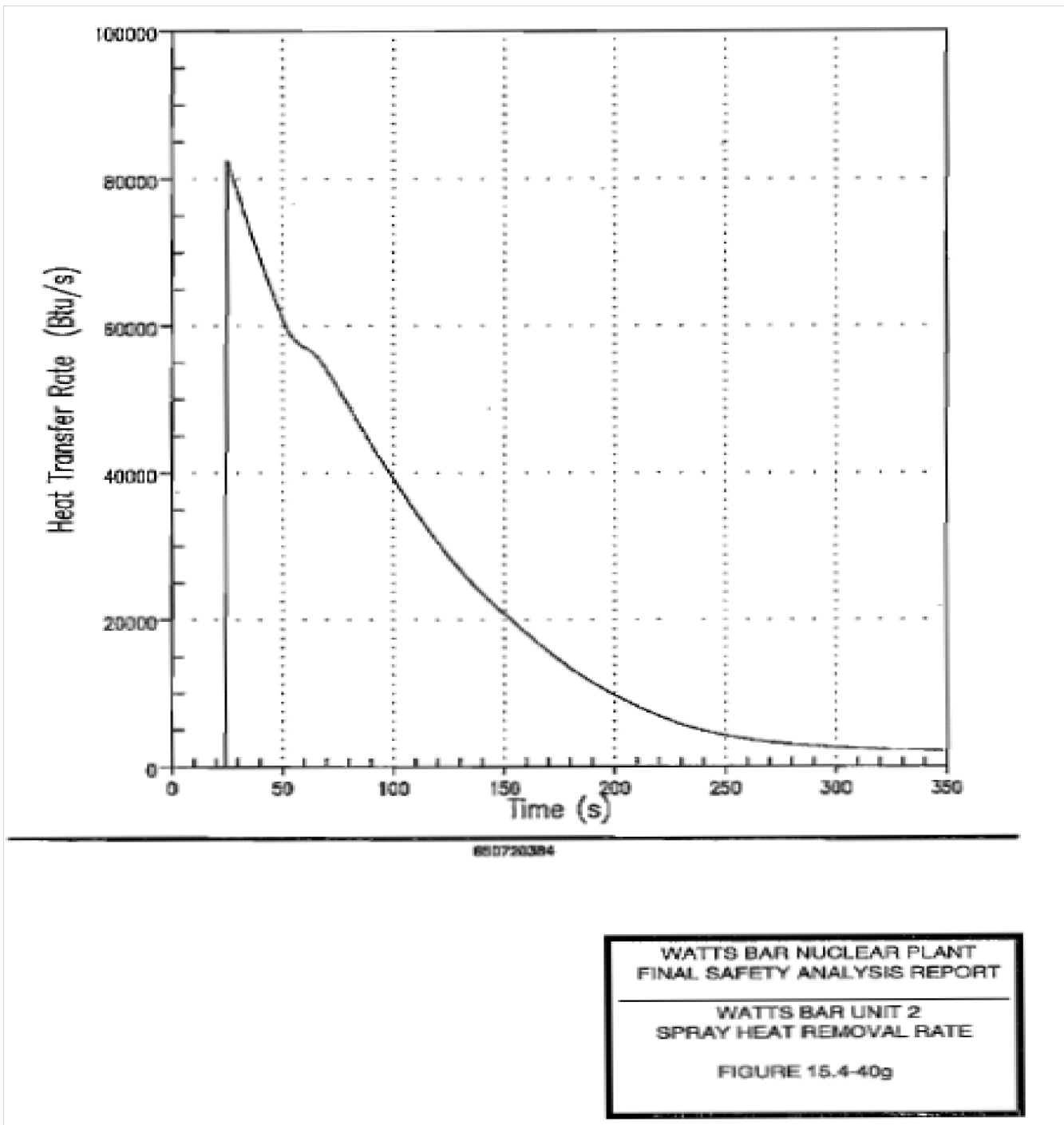


Figure 15.4-40g Watts Bar Unit2 Sump Heat Removal Rate

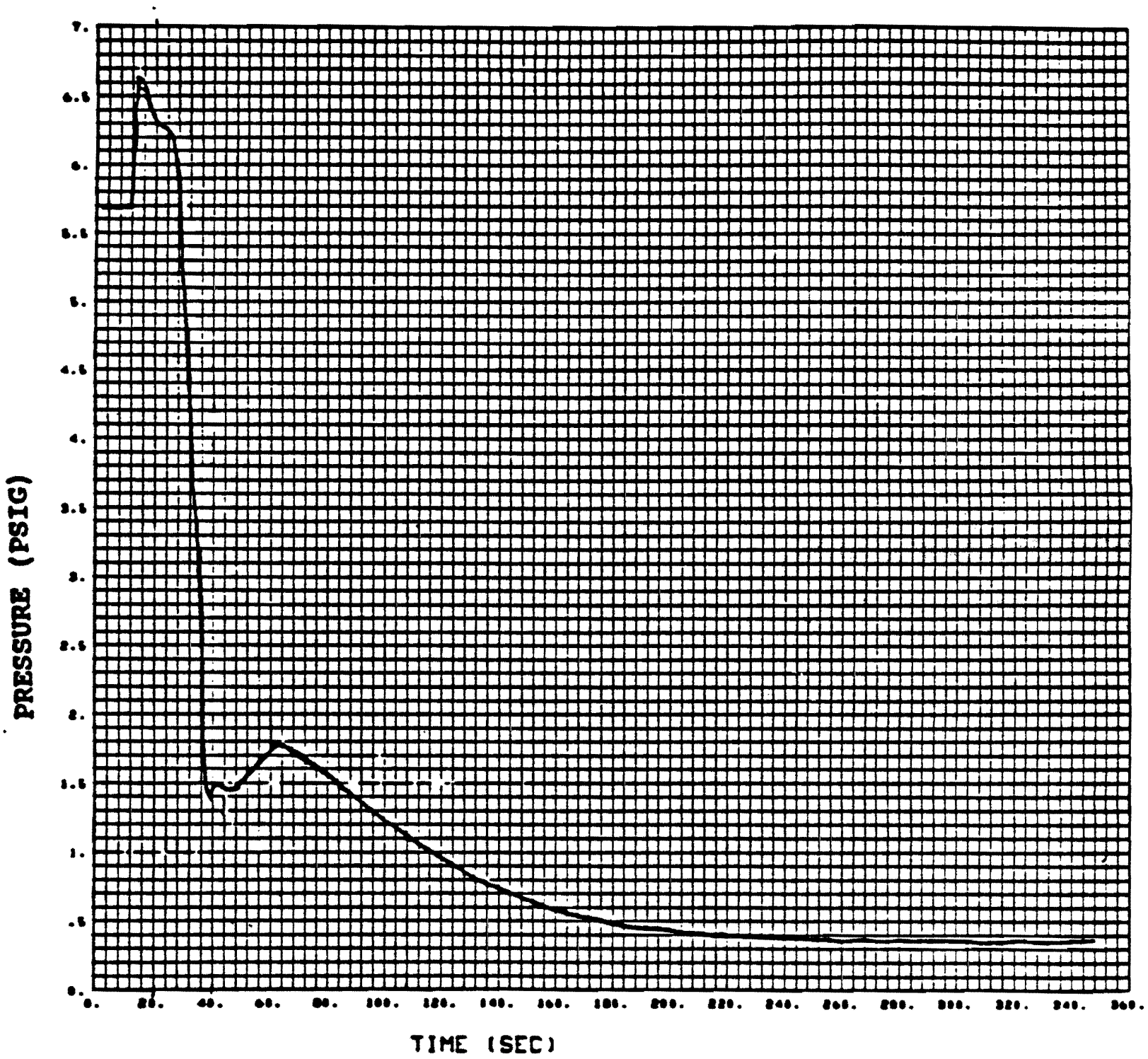


FIGURE 15.4-40h CONTAINMENT LOWER COMPARTMENT PRESSURE, MAXIMUM SAFEGUARDS, UPFLOW BARREL/BAFFLE REGION

Amendment 63

Figure 15.4-40h Containment Lower Compartment Pressure, Maximum Safeguards, Upflow Barrel/Baffle Region

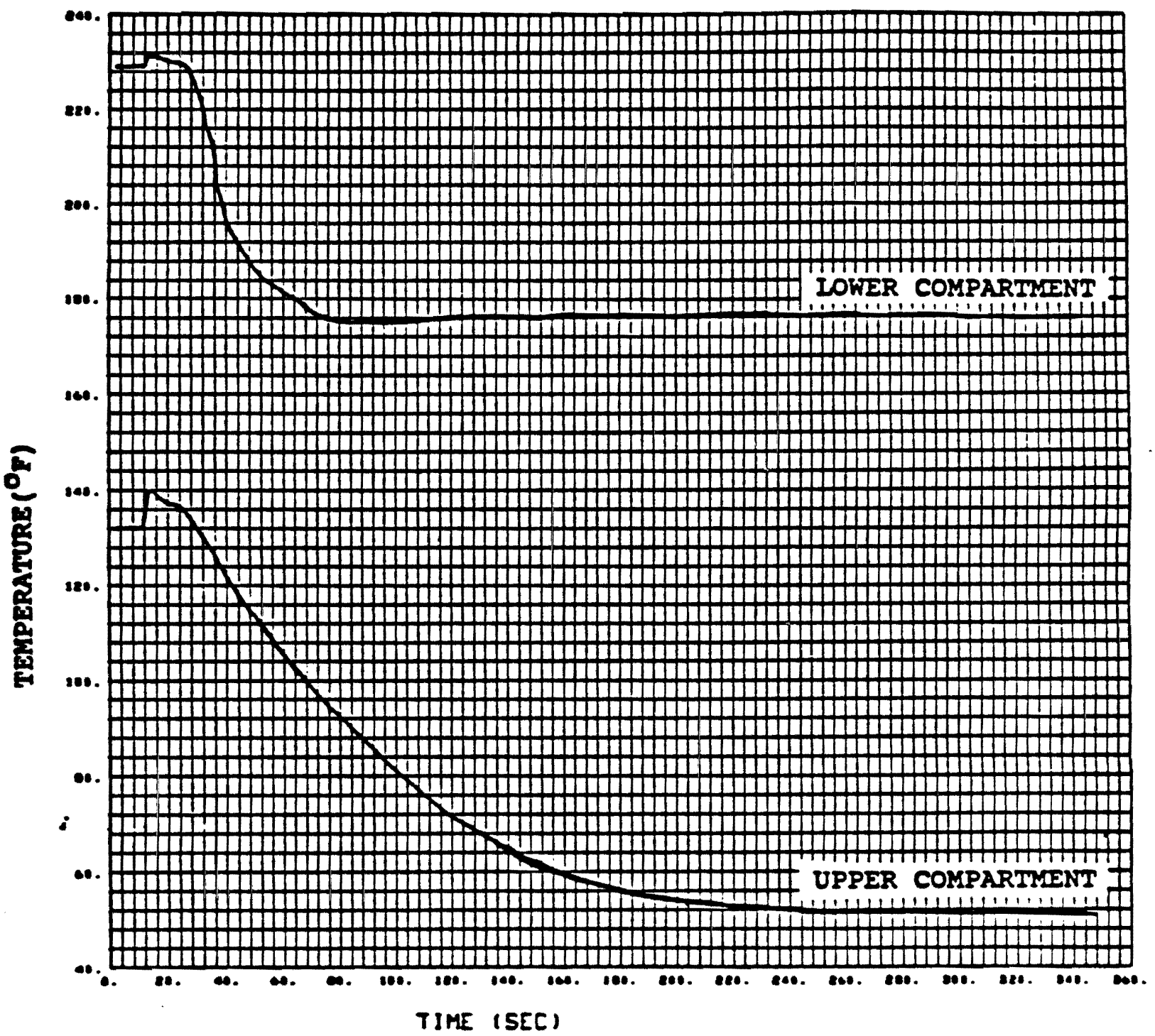


FIGURE 15.4-40i COMPARTMENT TEMPERATURES, MAXIMUM SAFEGUARDS, UPFLOW BARREL/BAFFLE REGION

Amendment 63

Figure 15.4-40i Compartment Temperatures, Maximum Safeguards, Upflow Barrel/Baffle Region

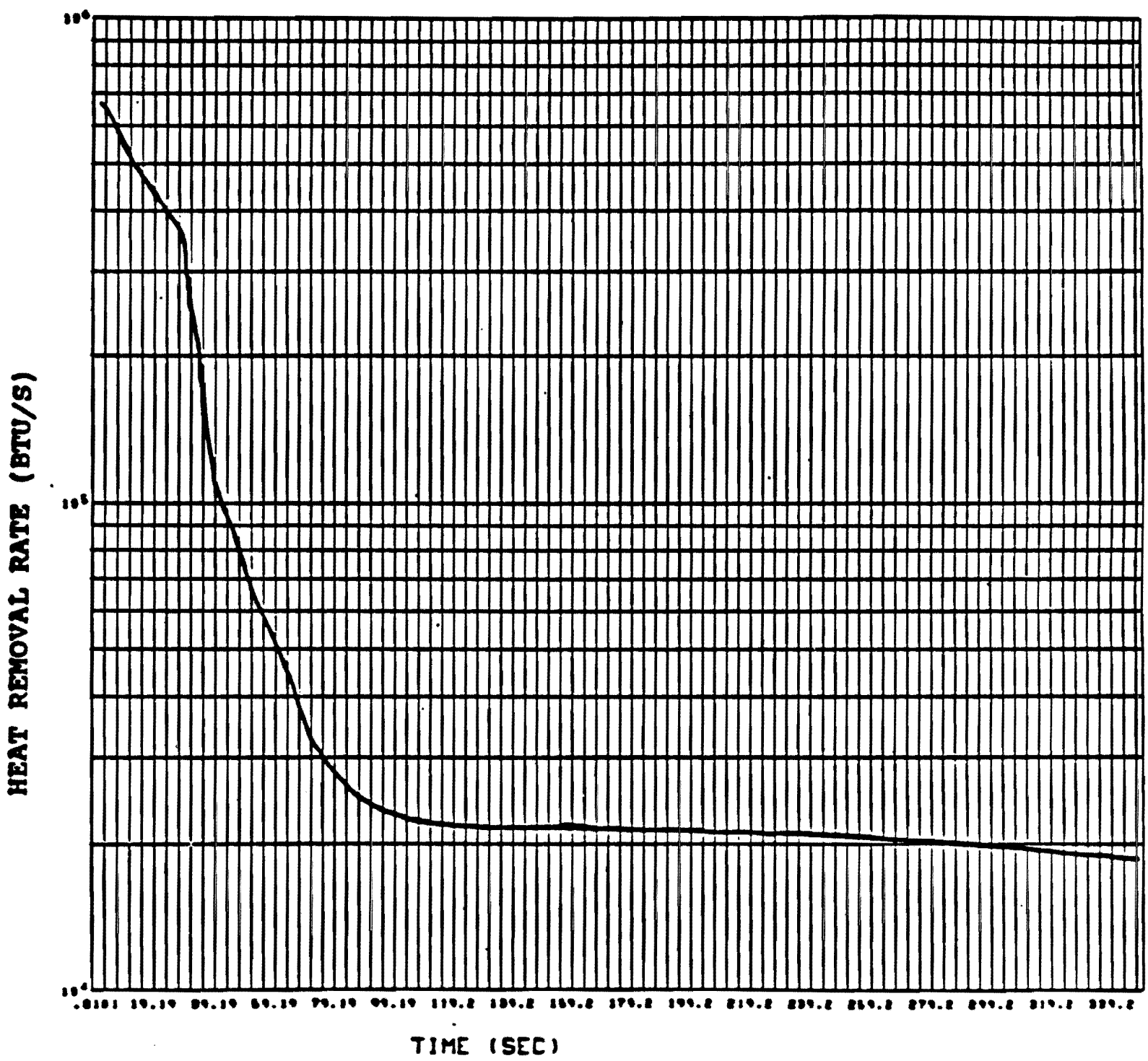


FIGURE 15.4-40j LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE, MAXIMUM SAFEGUARDS, UPFLOW BARREL/BAFFLE REGION

Amendment 63

Figure 15.4-40j Lower Compartment Structural Heat Removal Rate, Maximum Safeguards, Upflow Barrel/Baffle Region

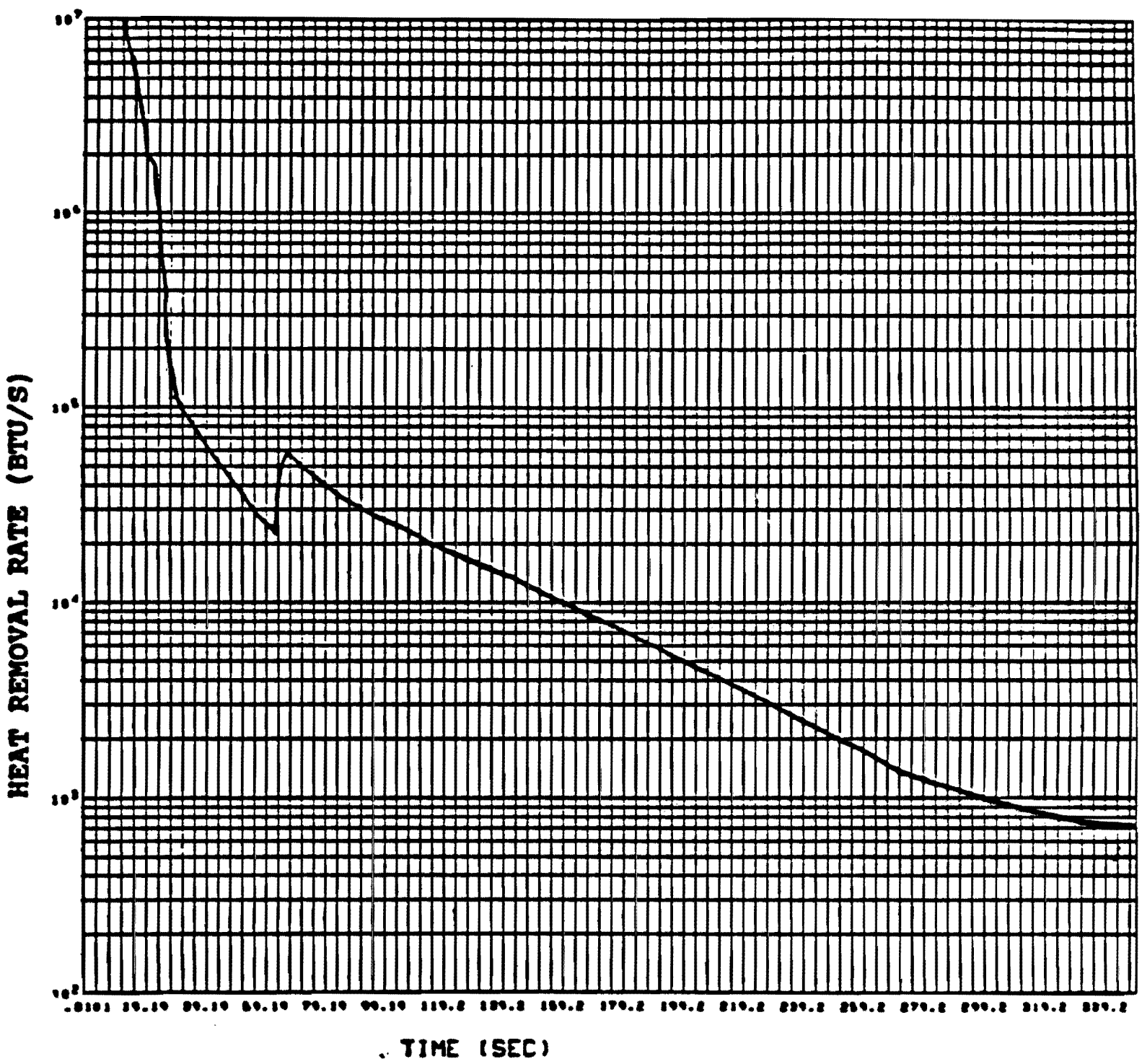


FIGURE 15.4-40k ICE BED HEAT REMOVAL RATE, MAXIMUM SAFEGUARDS, UPFLOW BARREL/BAFFLE REGION

Amendment 63

Figure 15.4-40k Ice Bed Heat Removal Rate, Maximum Safeguards, Upflow Barrel/Baffle Region

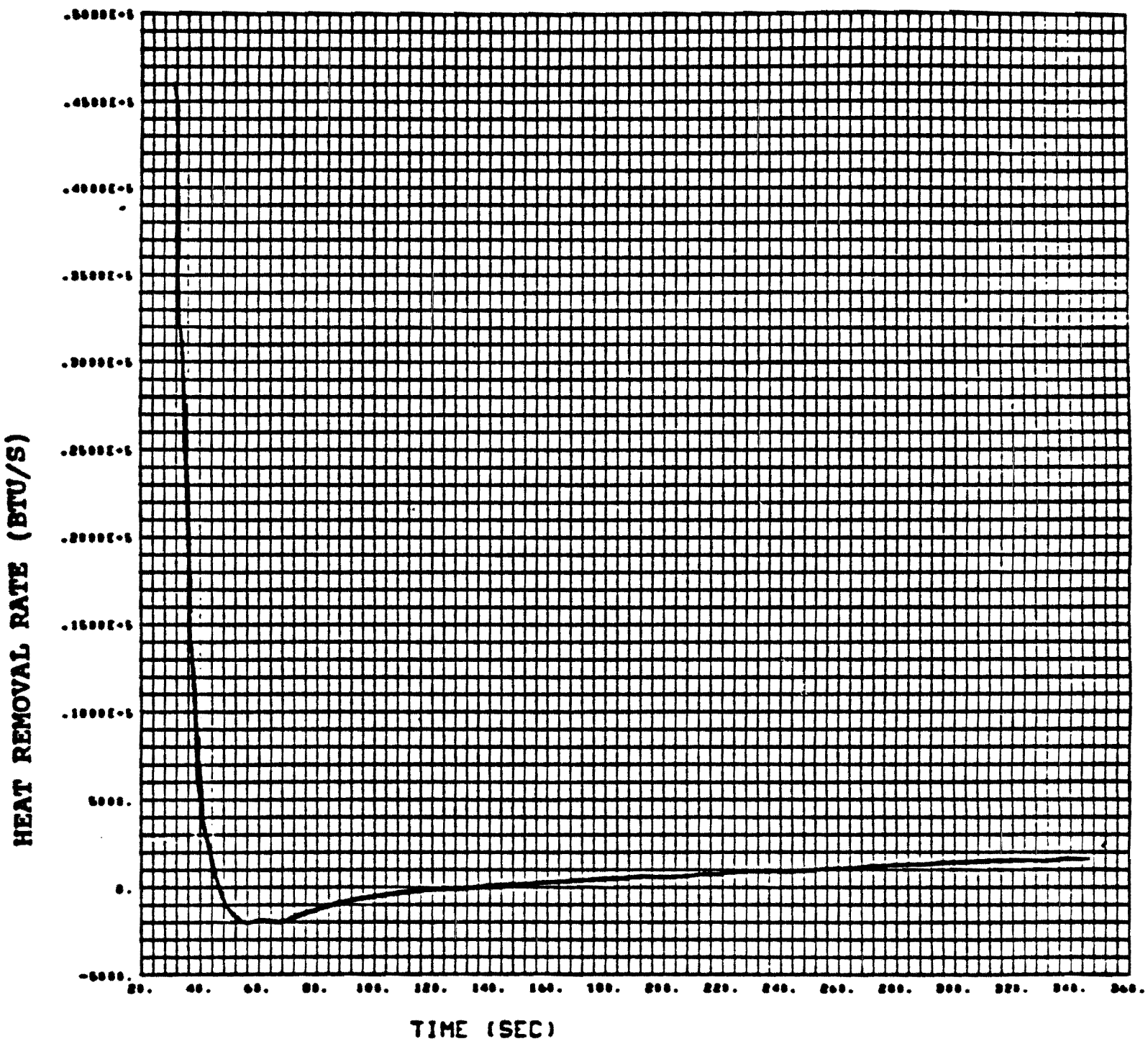


FIGURE 15.4-40l HEAT REMOVAL BY SUMP
MAXIMUM SAFEGUARDS, UPFLOW BARREL/BAFFLE REGION

Amendment 63

Figure 15.4-40l Heat Removal by Sump, Maximum Safeguards, Upflow Barrel/Baffle Region

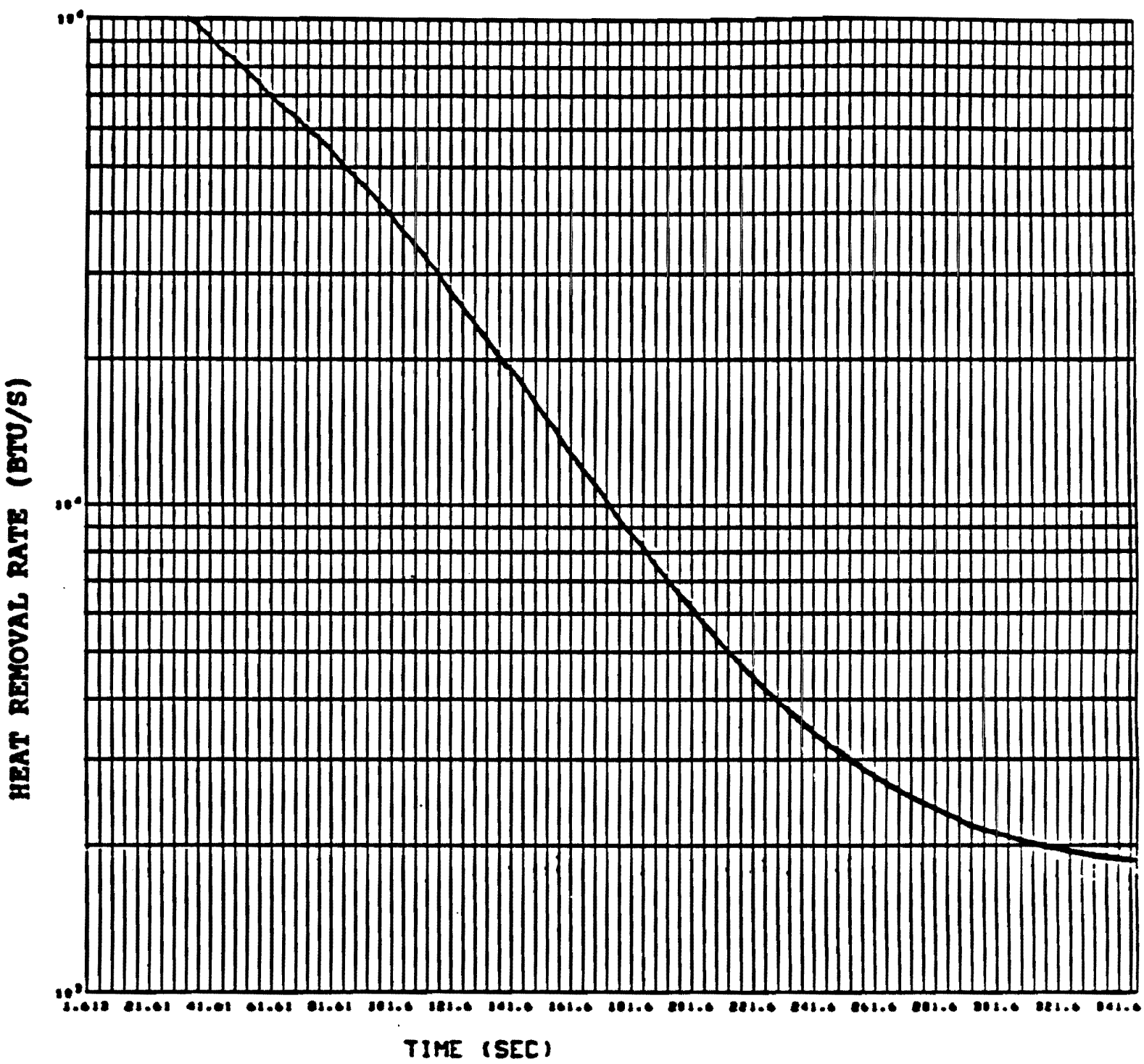


FIGURE 15.4-40m HEAT REMOVAL BY SPRAY, MAXIMUM SAFEGUARDS, UPFLOW BARREL/BAFFLE REGION

Amendment 63

Figure 15.4-40m Heat Removal by Spray, Maximum Safeguards, Upflow Barrel/Baffle Region

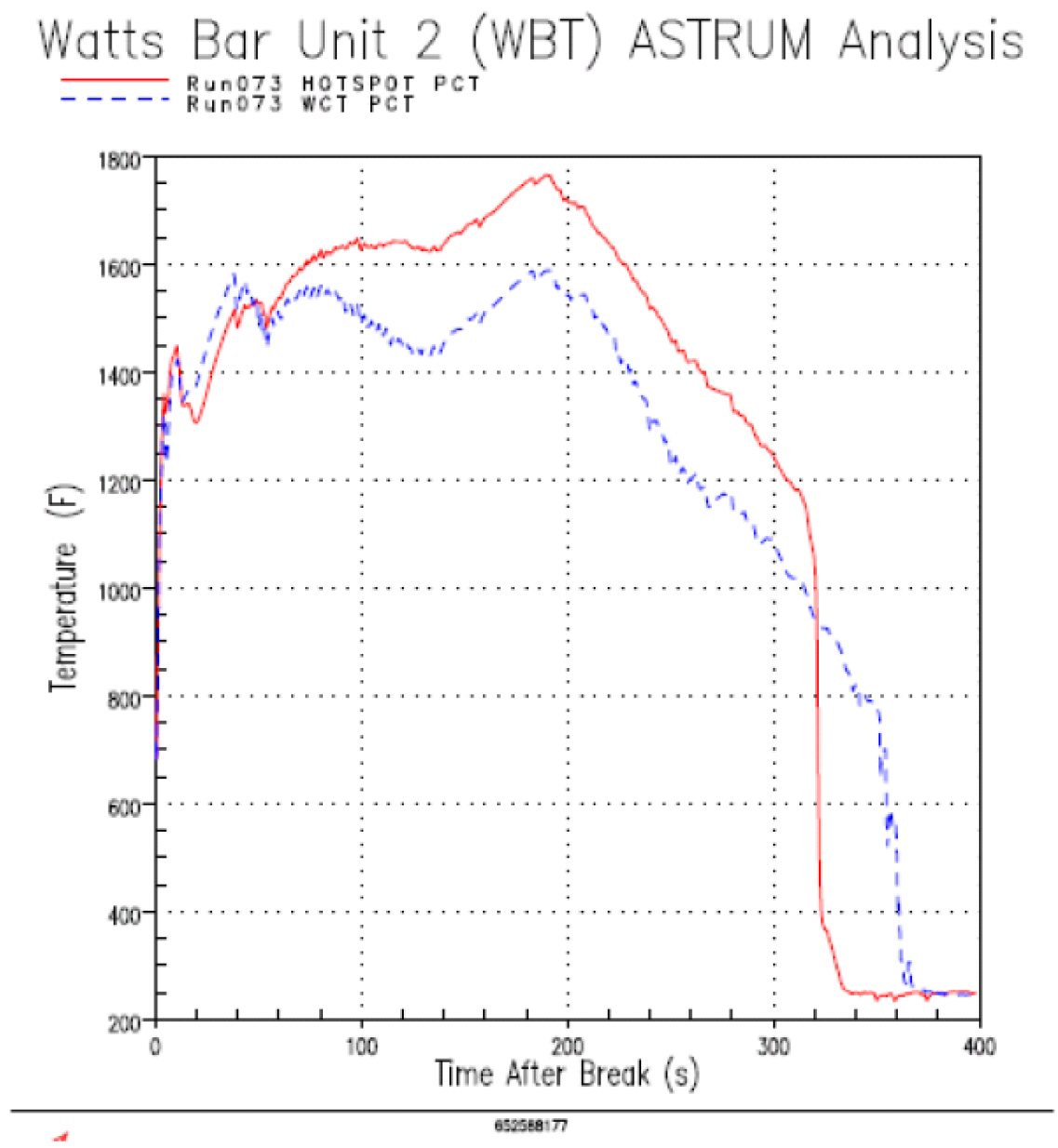


Figure 15.4-41 Watts Bar Unit 2 Limiting PCT Case (2nd Cycle, Run073, non-IFBA) HOTSPOT Clad Temperature at the Limiting Elevation and WC/T PCT

Figure 15.4-41b Deleted by Amendment 110

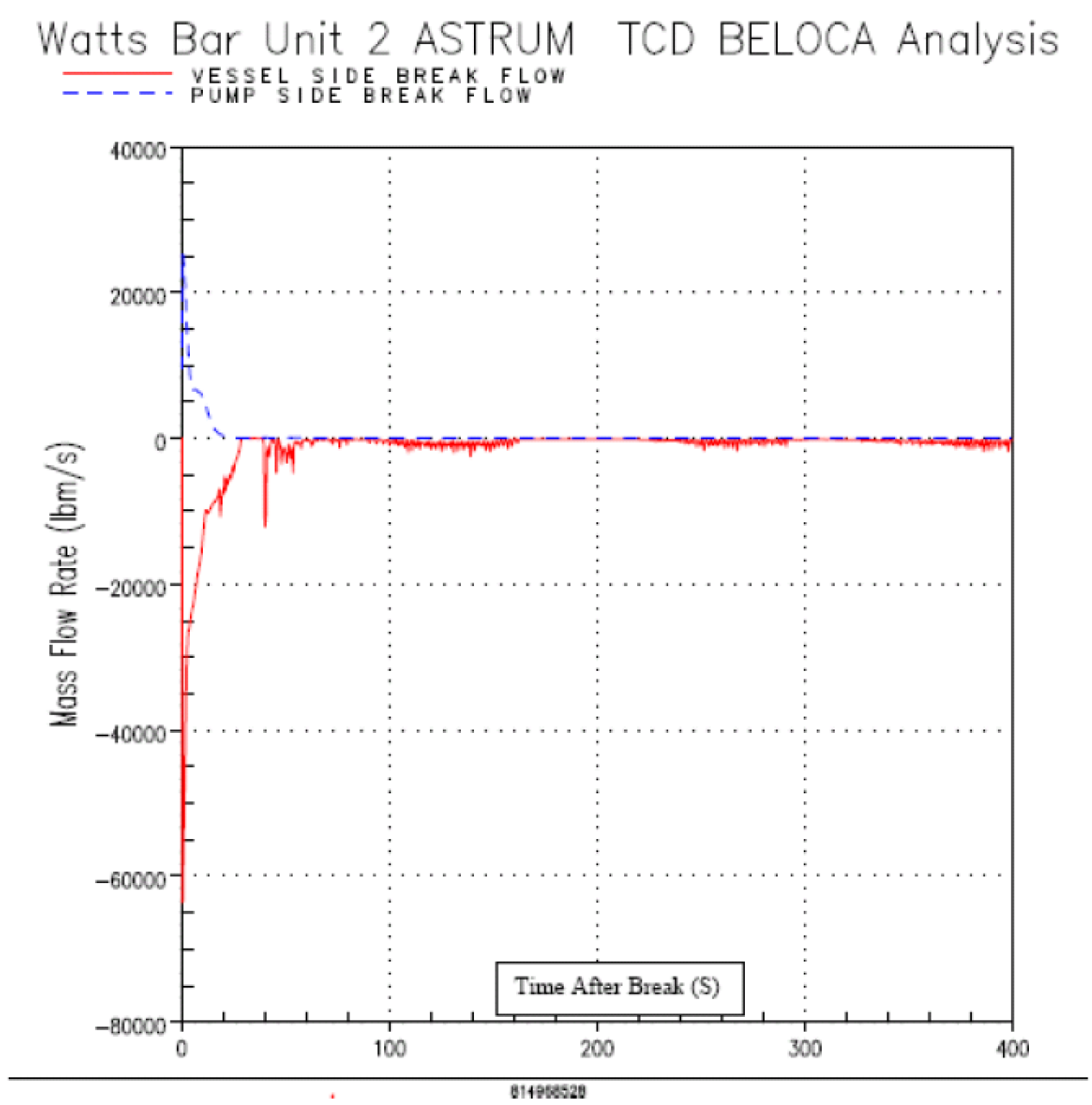


Figure 15.4-42 Watts Bar Unit 2 Limiting PCT Case Break Flow

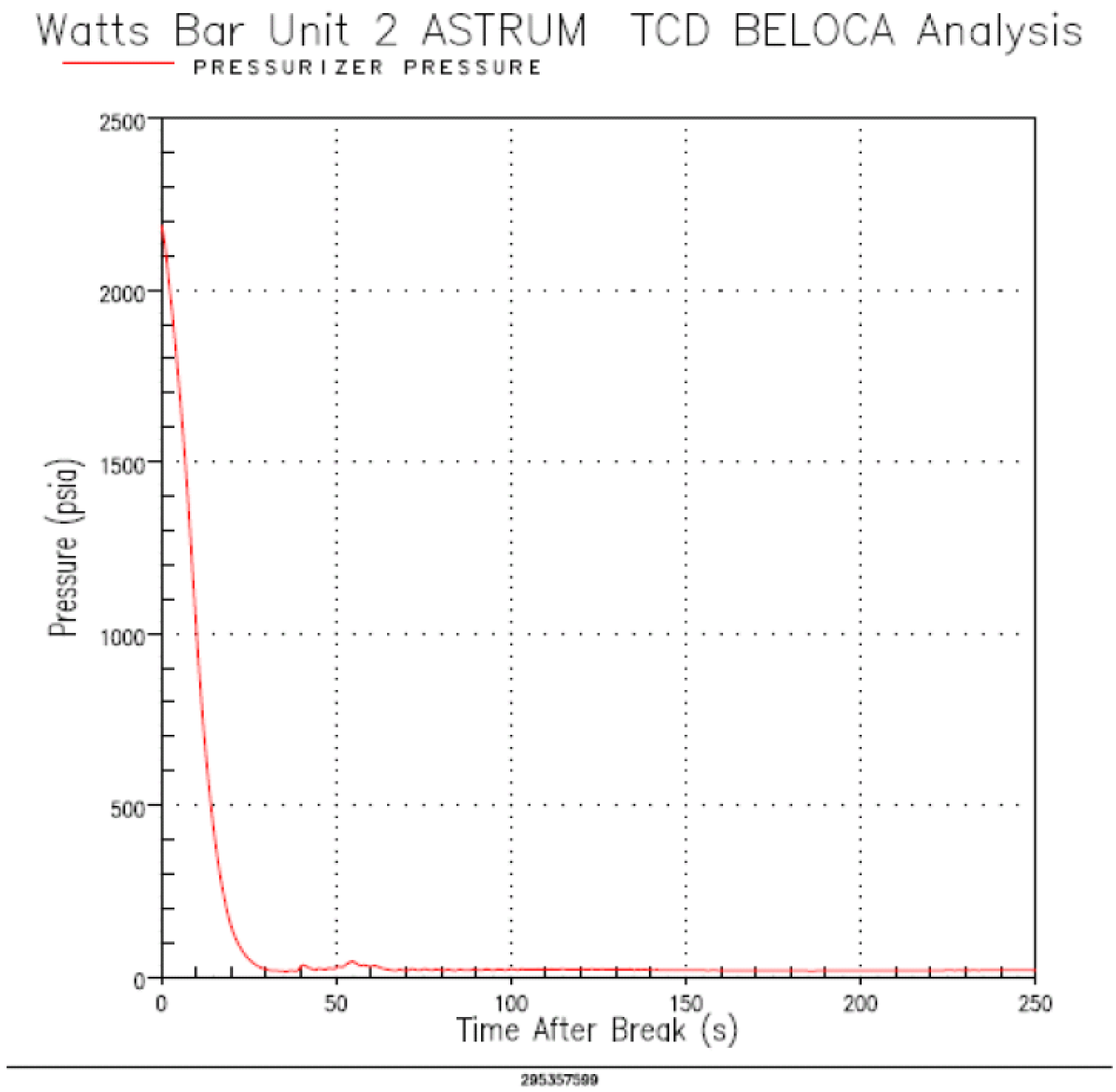


Figure 15.4-43 Watts Bar Unit 2 Limiting PCT Case Pressurizer Pressure

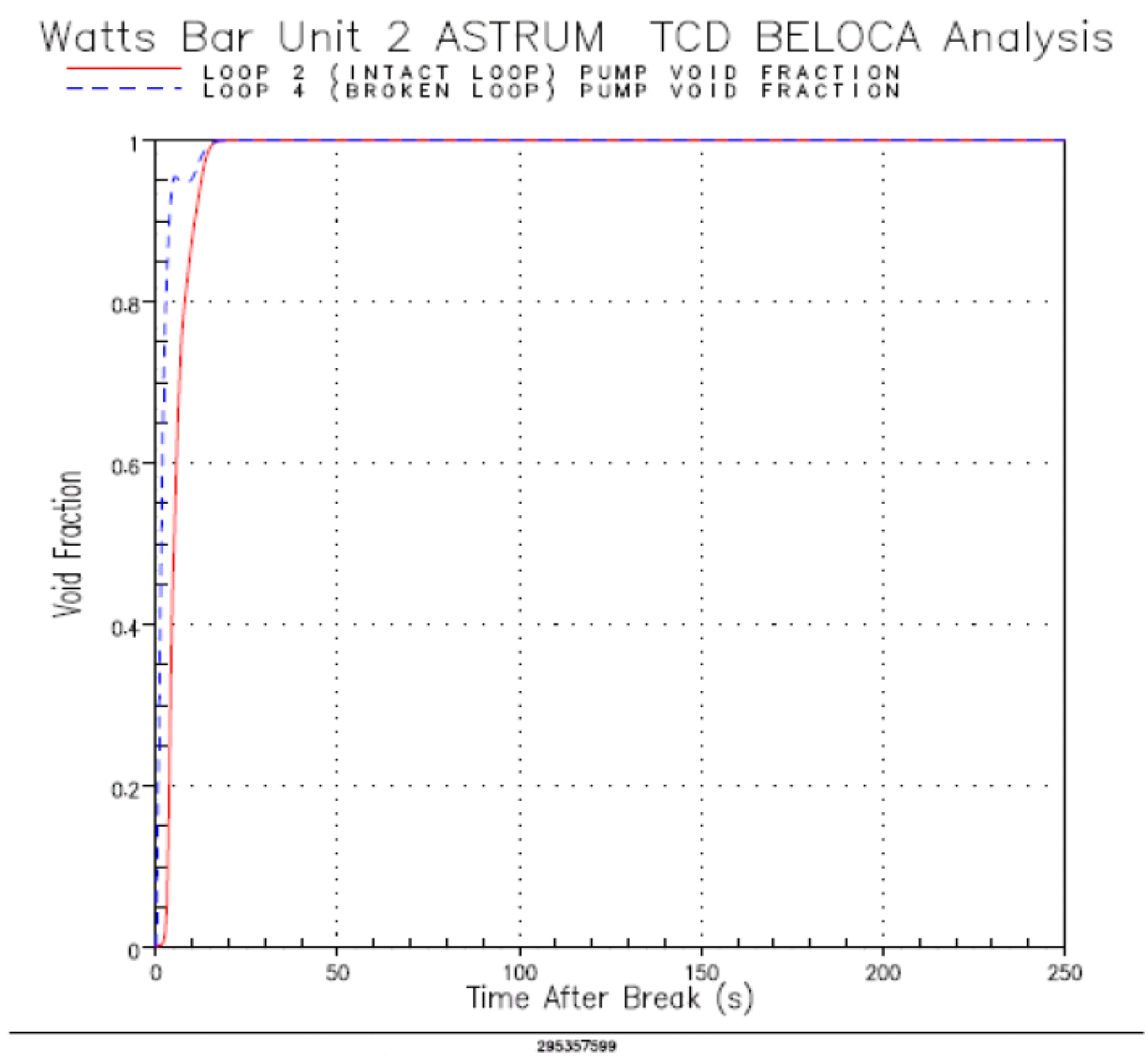


Figure 15.4-44 Watts Bar Unit 2 Limiting PCT Case Broken And Intact Loop Void Fraction

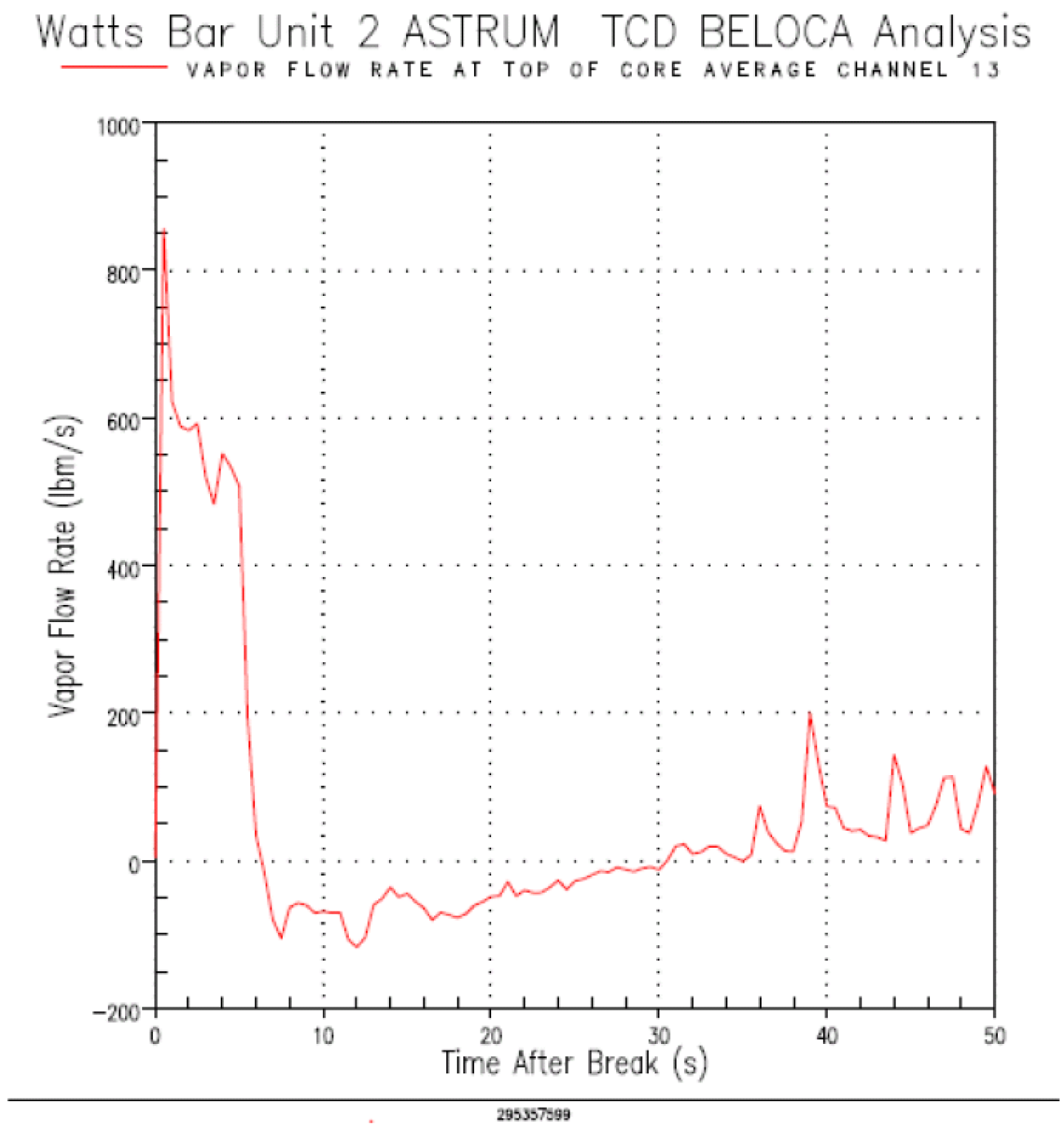


Figure 15.4-45 Watts Bar Unit 2 Limiting PCT Case Core Vapor Flow at the Top of the Core for a Core Average Channel

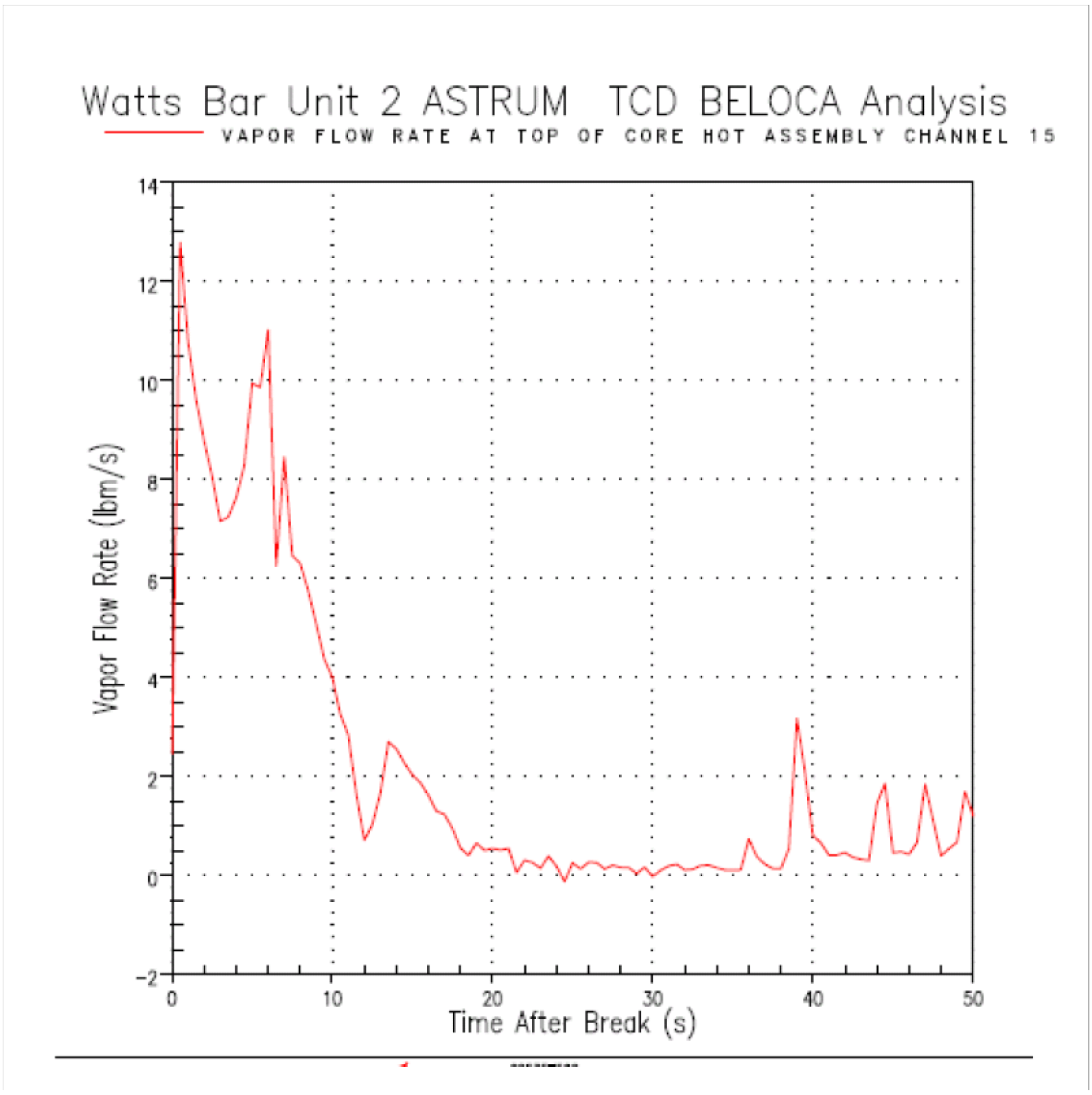


Figure 15.4-46 Watts Bar Unit 2 Limiting PCT Case Core Vapor Flow at the Top of the Core for the Hot Assembly Channel

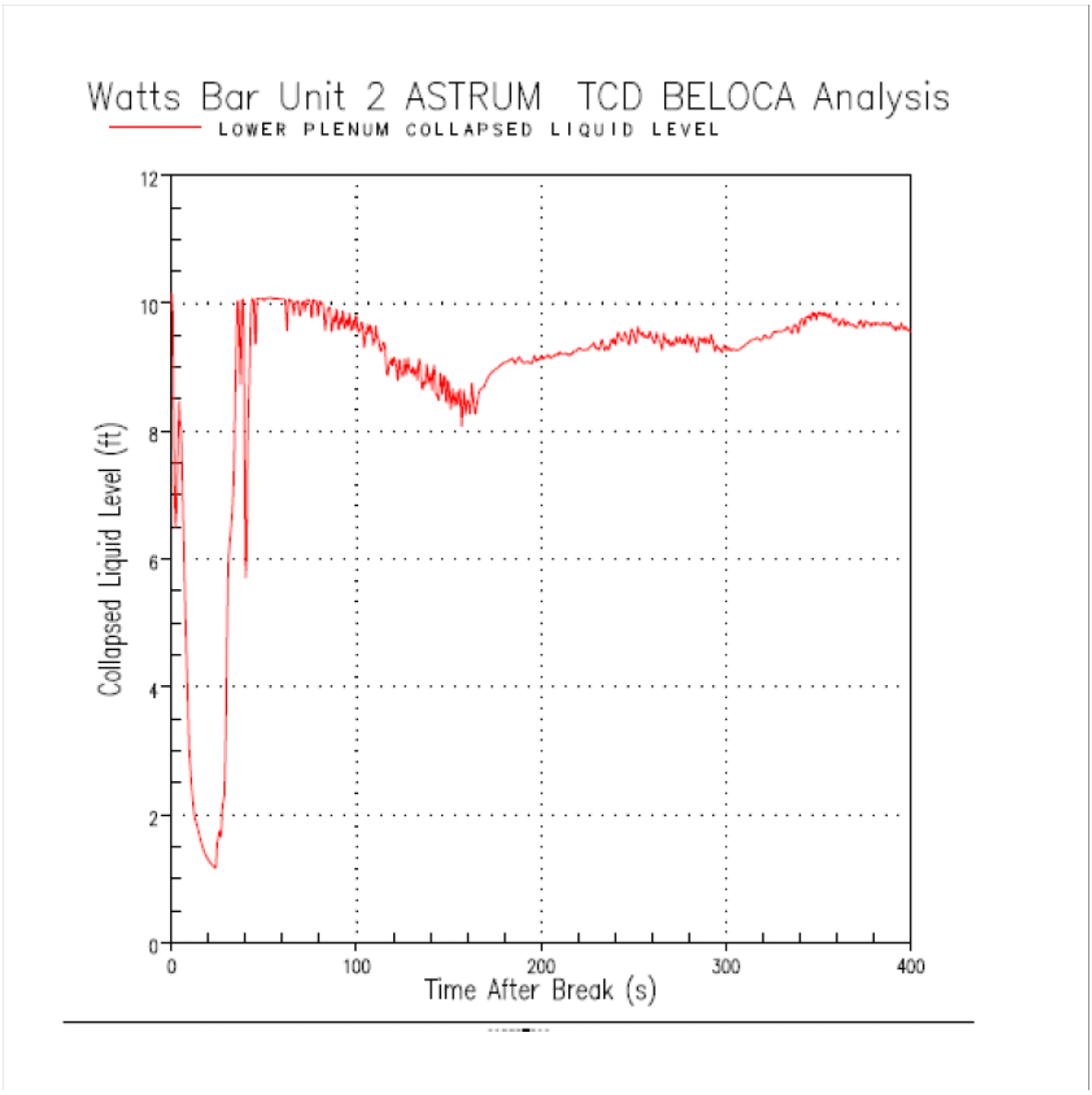


Figure 15.4-47 Watts Bar Unit 2 Limiting PCT Case Lower Plenum Collapsed Liquid Level

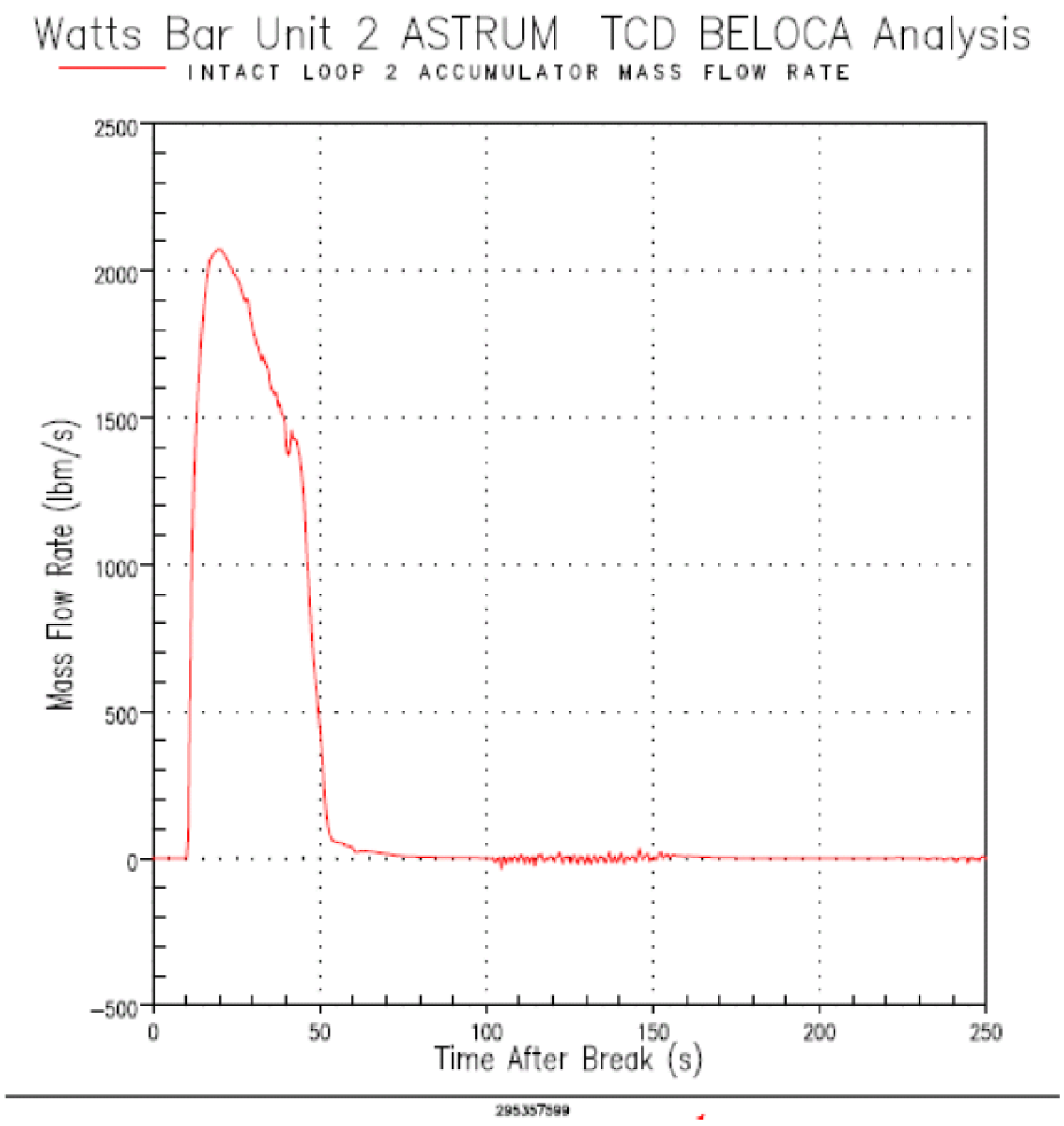


Figure 15.4-48 Watts Bar Unit 2 Limiting PCT Case Intact Loop Accumulator Flow

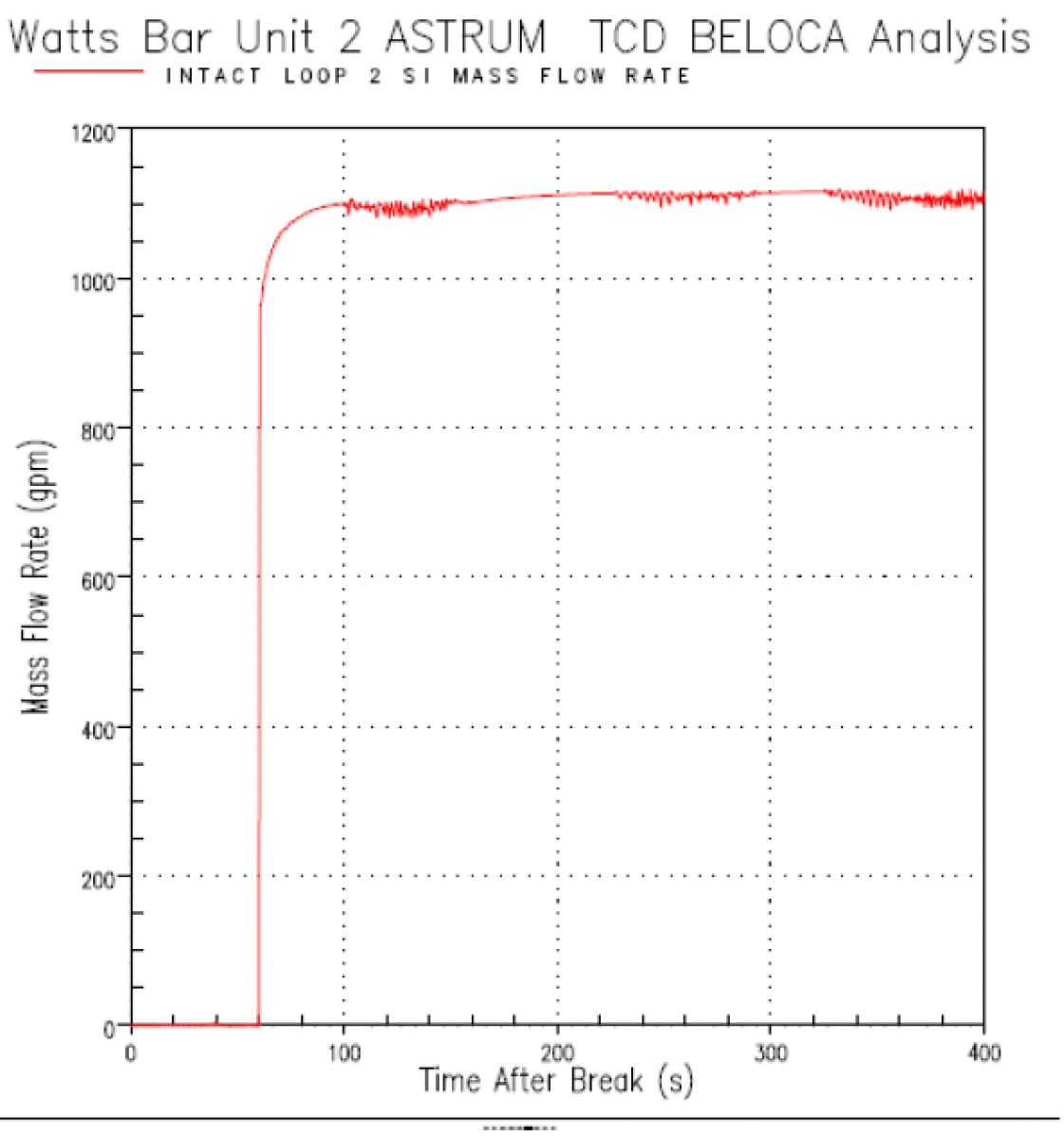


Figure 15.4-49 Watts Bar Unit 2 Limiting PCT Case Intact Loop Safety Injection Flow

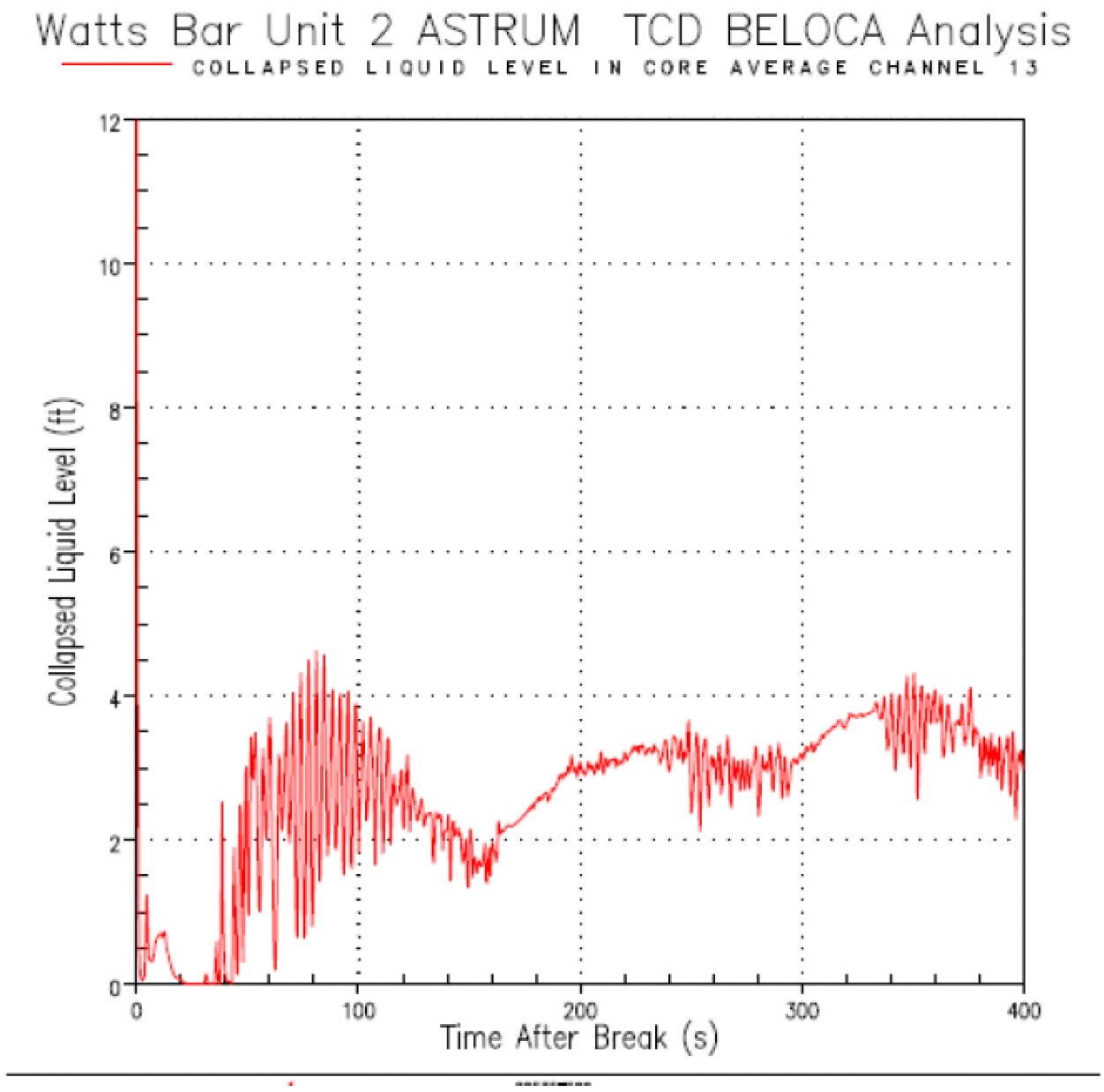


Figure 15.4-50 Watts Bar Unit 2 Limiting PCT Case Core Average Channel Collapsed Liquid Level

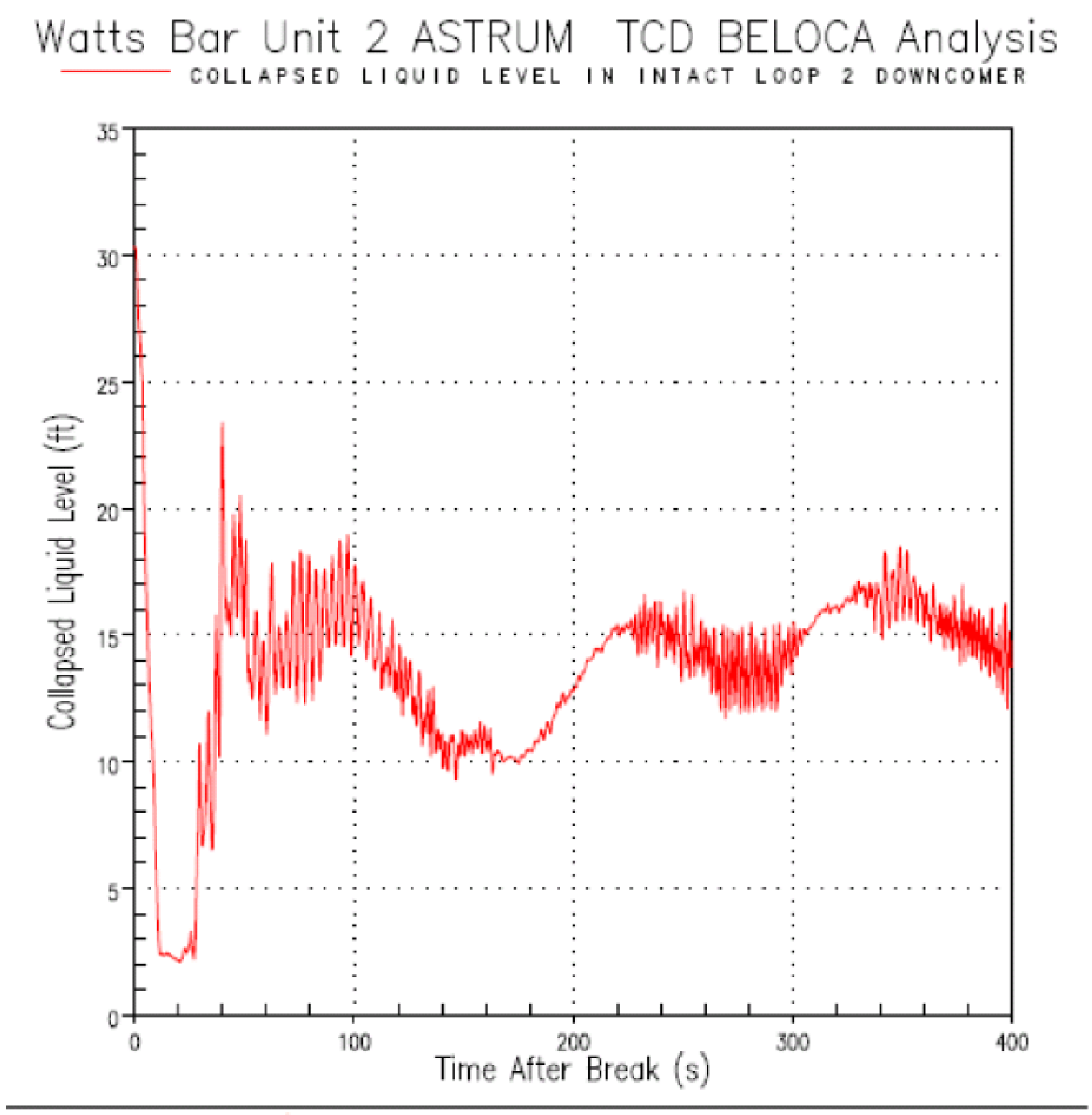


Figure 15.4-51 Watts Bar Unit 2 Limiting PCT Case Loop 2 Downcomer Collapsed Liquid Level

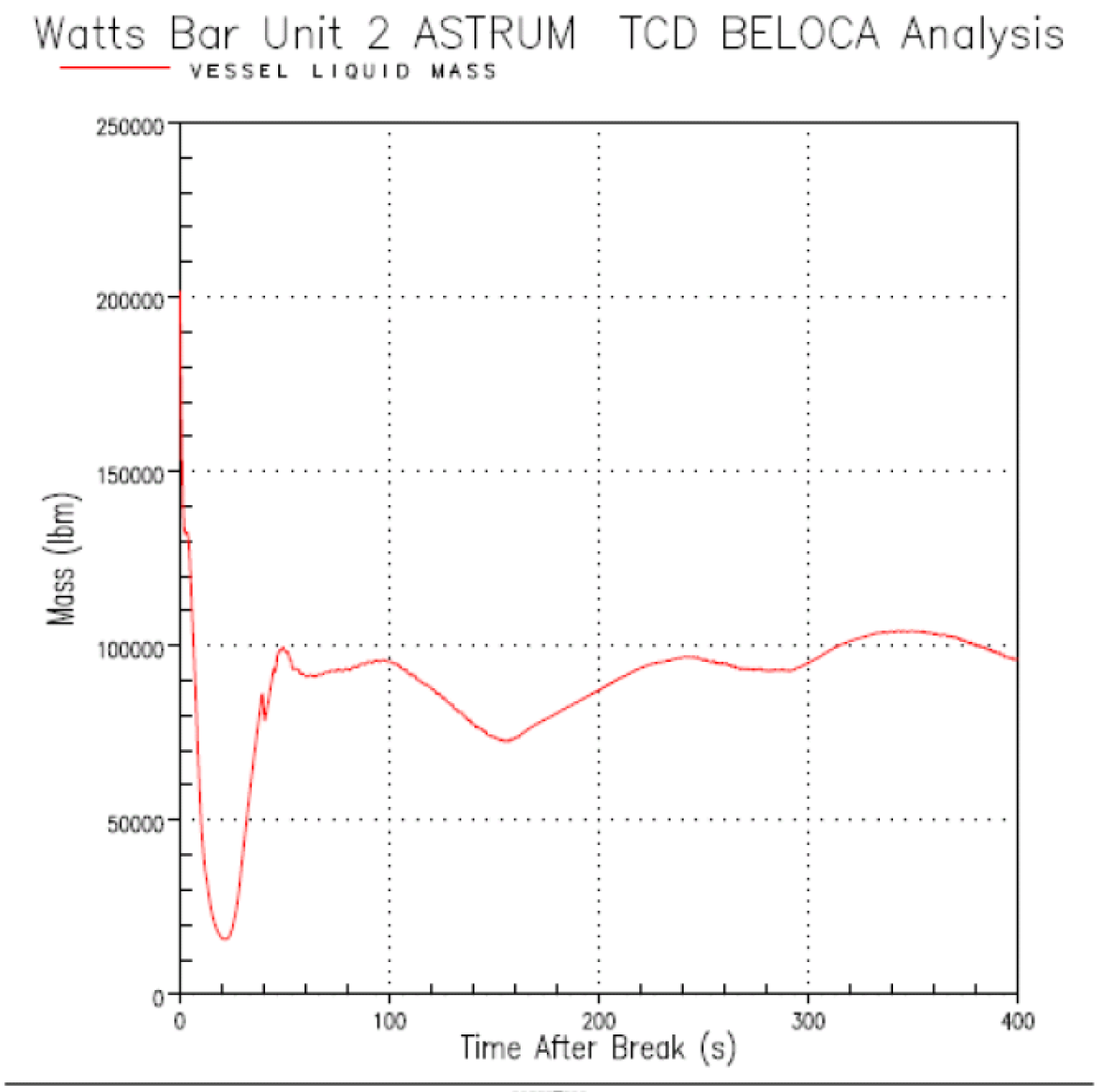


Figure 15.4-52 Watts Bar Unit 2 Limiting PCT Case Vessel Fluid Mass

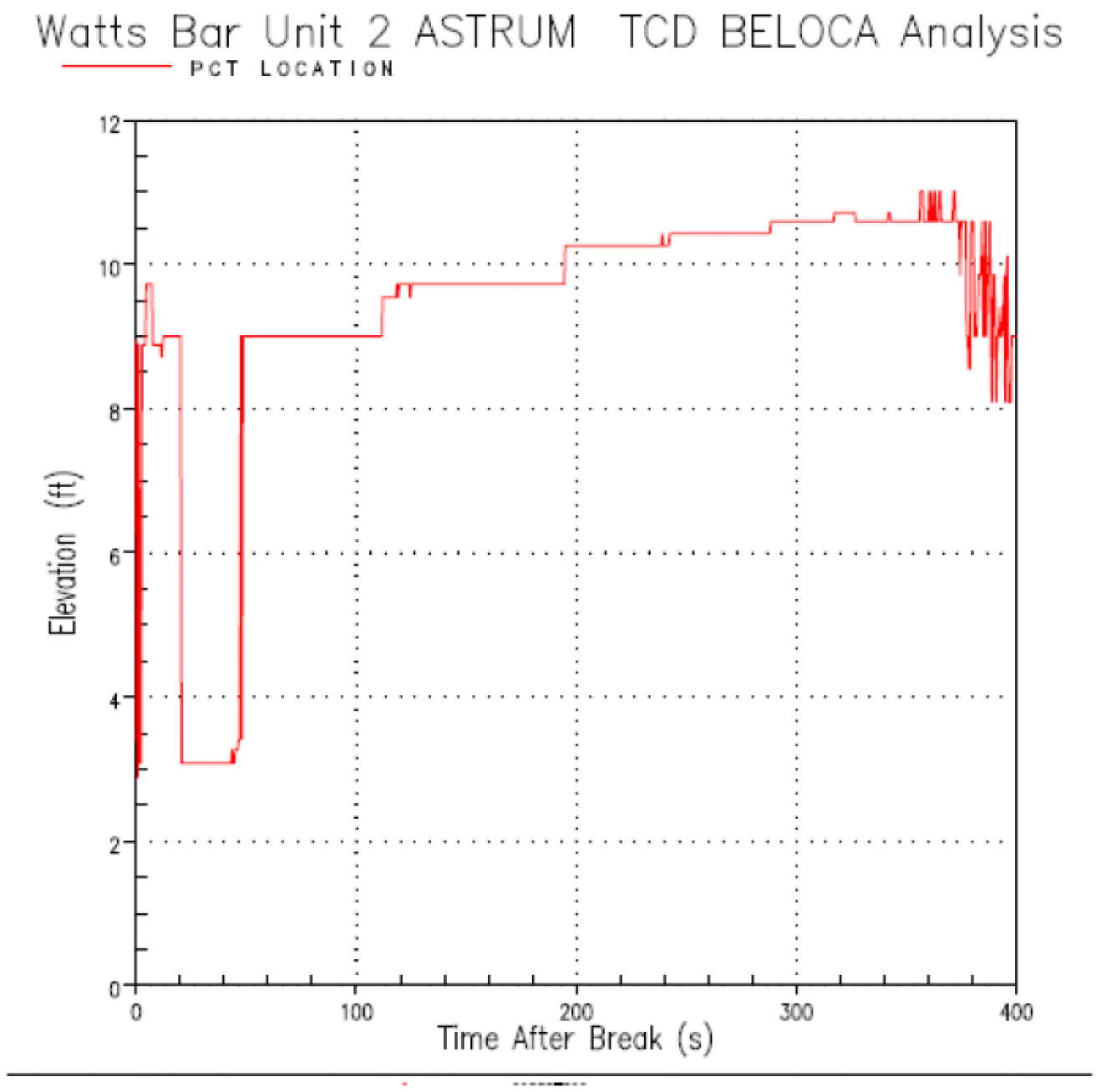


Figure 15.4-53 Watts Bar Unit 2 Limiting PCT Case PCT Location

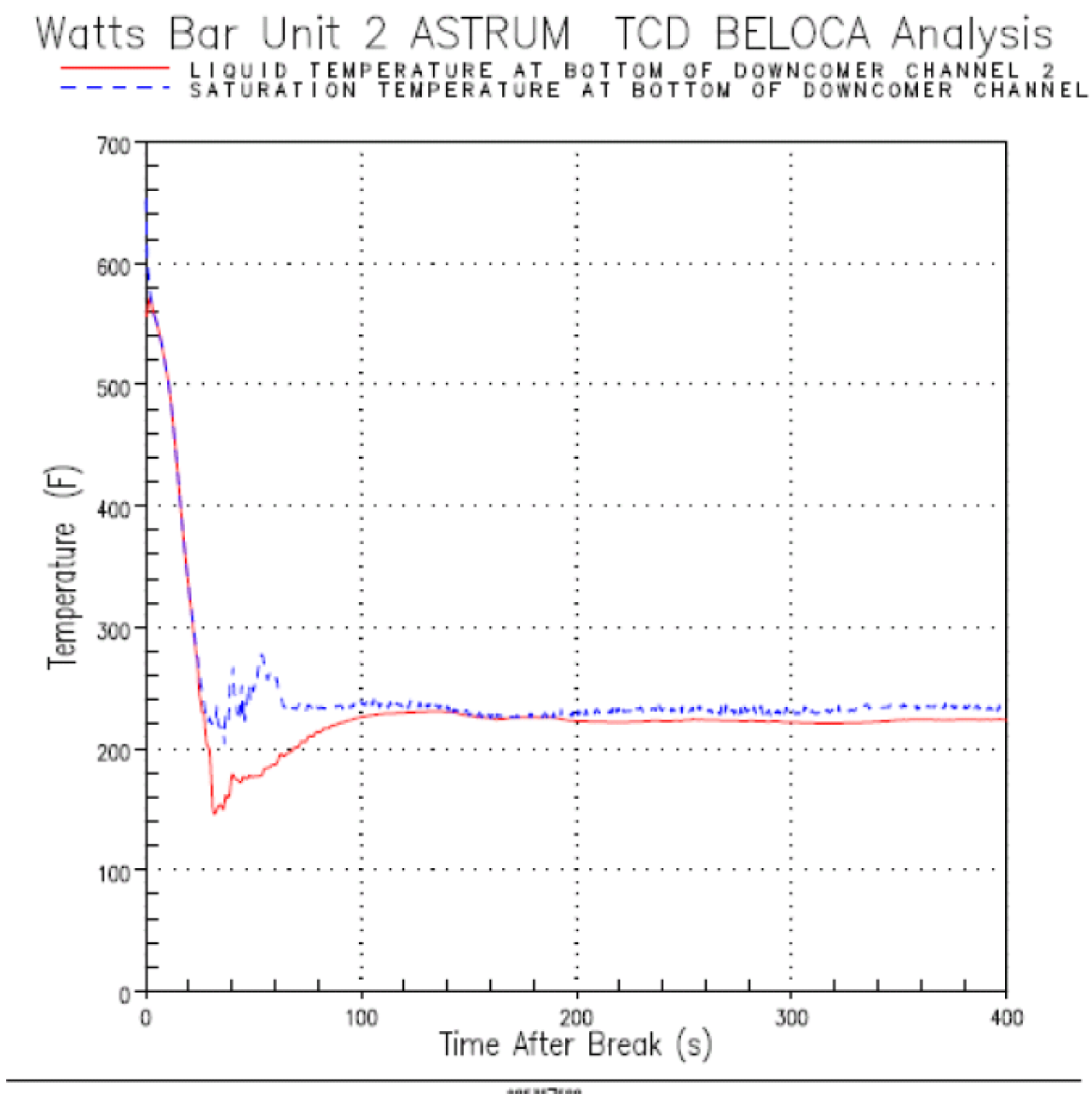


Figure 15.4-54 Watts Bar Unit 2 Limiting PCT Case Liquid And Saturation Temperature At Bottom Of Downcomer Channel

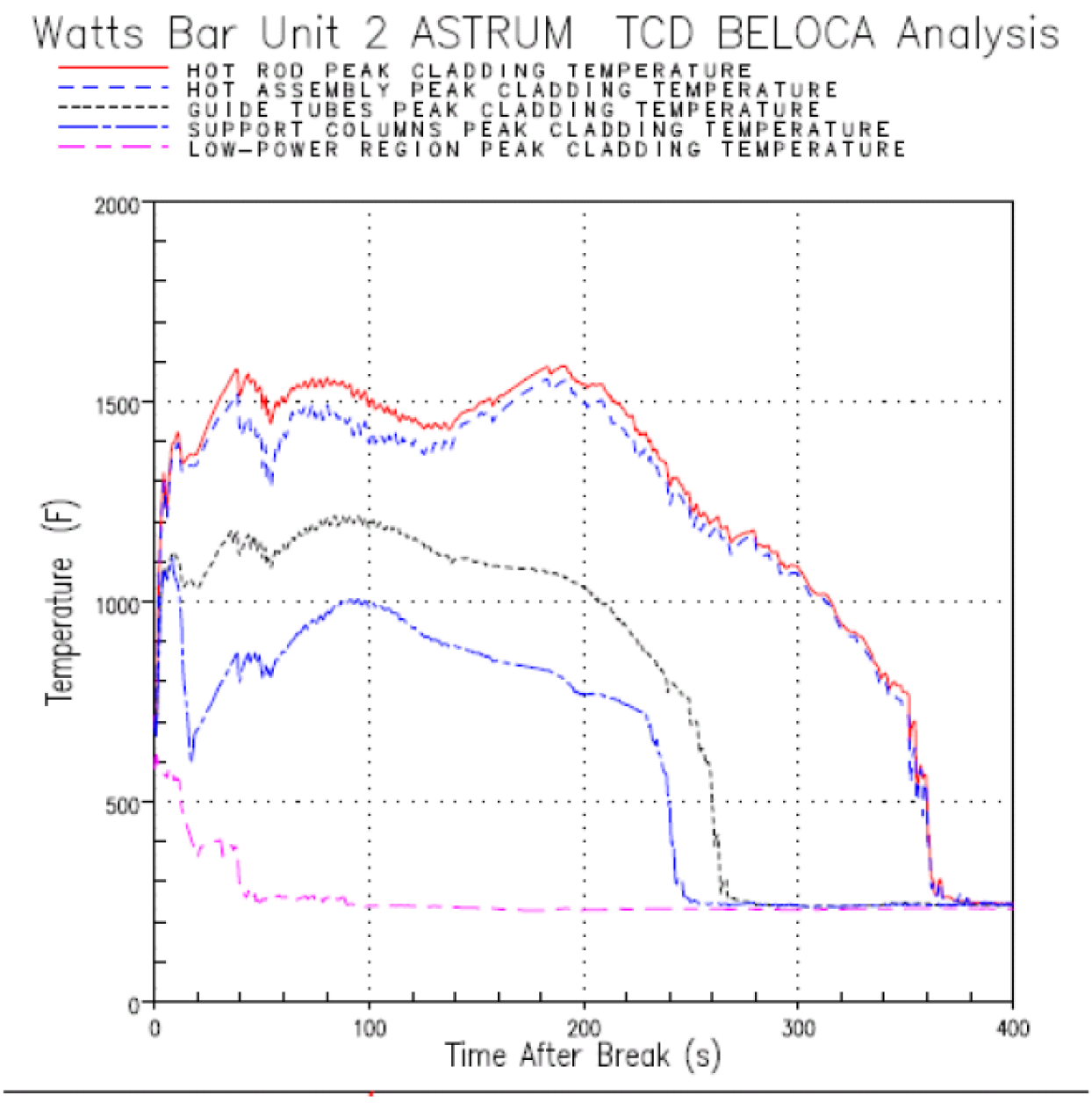
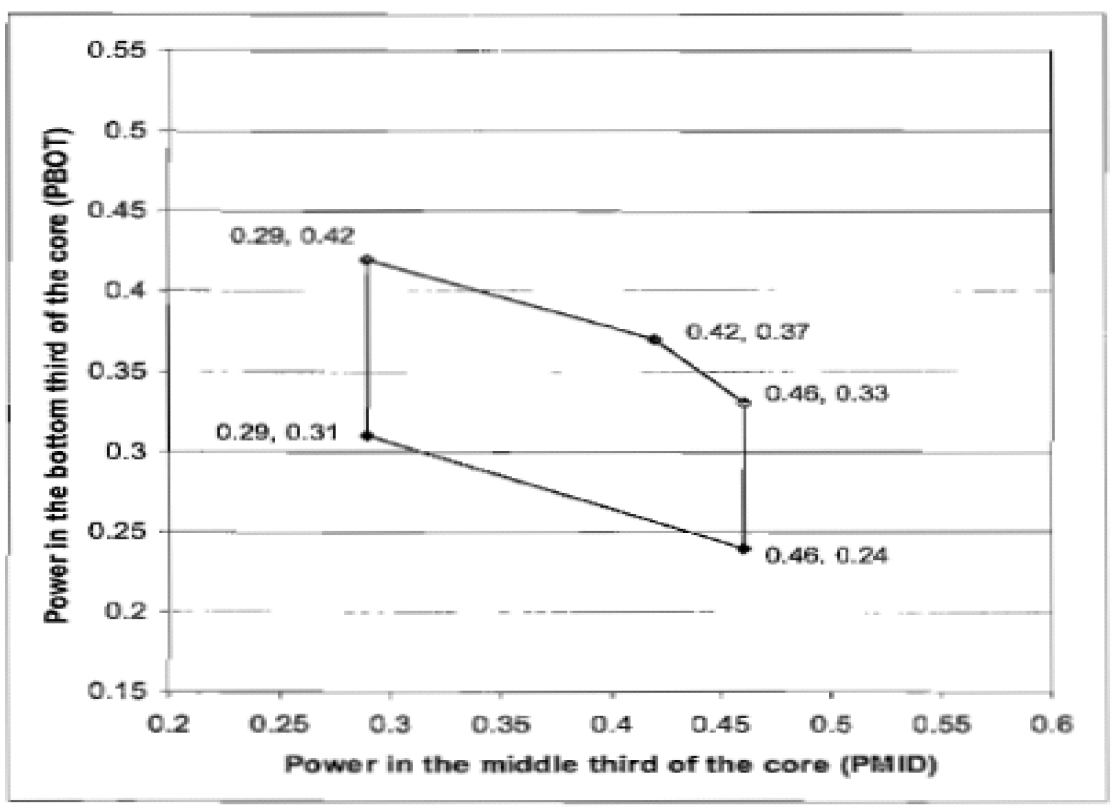


Figure 15.4-55 Watts Bar Unit 2 Limiting PCT Case PCT Case Peak Cladding Temperature For All 5 Rods



PBOT = integrated power fraction in the bottom third of the core
PMID = integrated power fraction in the middle third of the core

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WATTS BAR UNIT 2
BELOCA ANALYSIS
AXIAL POWER SHAPE OPERATING
SPACE ENVELOPE

FIGURE 15.4-56

Figure 15.4-56 Watts Bar Unit 2 BELOCA Analysis Axial Power Shape Operating Space Envelope

Figure 15.4-57 Deleted by Amendment 97

Figure 15.4-58 Deleted by Amendment 97

Figure 15.4-59 Deleted by Amendment 97

Figure 15.4-60 Deleted by Amendment 97

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Figure 15.4-66 Deleted by Amendment 97

Figure 15.4-67 Deleted by Amendment 97

Figure 15.4-68a Deleted by Amendment 97

Figure 15.4-68b Deleted by Amendment 97

Figure 15.4-68c Deleted by Amendment 97

Figure 15.4-68d Deleted by Amendment 97

Figure 15.4-68e Deleted by Amendment 97

Figure 15.4-68f Deleted by Amendment 97

Figure 15.4-68g Deleted by Amendment 97

Figure 15.4-68h Deleted by Amendment 97

Figure 15.4-68i Deleted by Amendment 97

Figure 15.4-68j Deleted by Amendment 97

Figure 15.4-68k Deleted by Amendment 97

Figure 15.4-68l Deleted by Amendment 97

Figure 15.4-68m Deleted by Amendment 97

Figure 15.4-68n Deleted by Amendment 97

Figure 15.4-69 Deleted by Amendment 97

Figure 15.4-70 Deleted by Amendment 97

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Figure 15.4-96a Deleted by Amendment 97

Figure 15.4-96b Deleted by Amendment 97

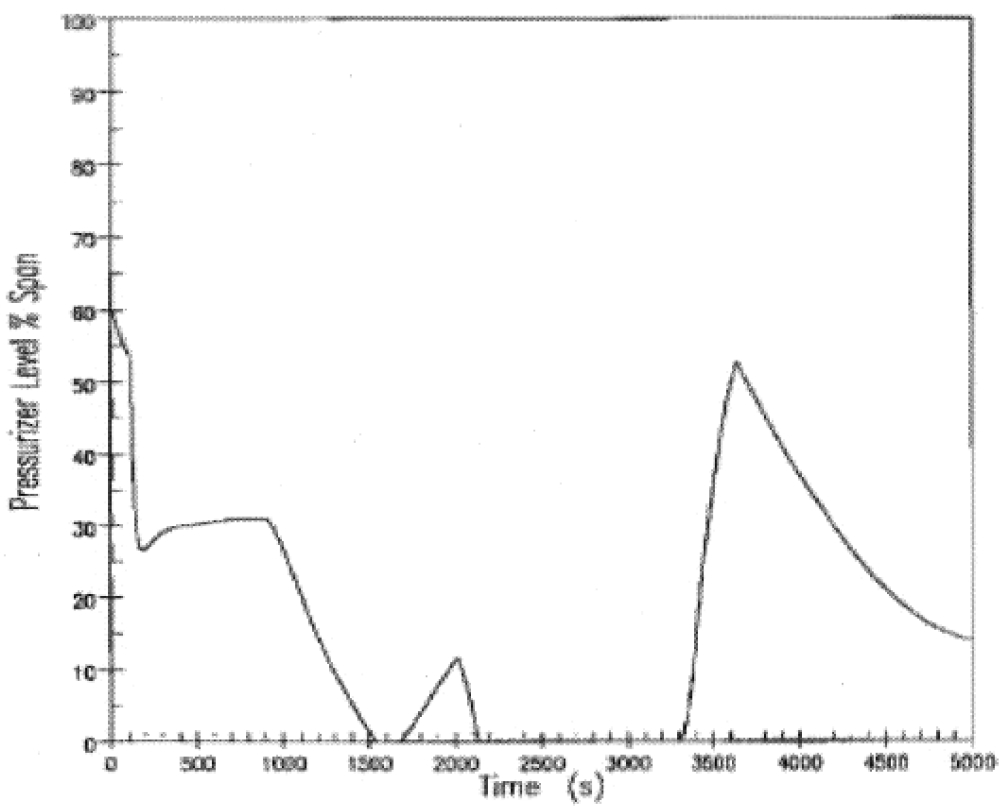
Figure 15.4-96c Deleted by Amendment 97

Figure 15.4-96d Deleted by Amendment 97

Figure 15.4-96e Deleted by Amendment 97

Figure 15.4-96f Deleted by Amendment 97

Figure 15.4-96g Deleted by Amendment 97



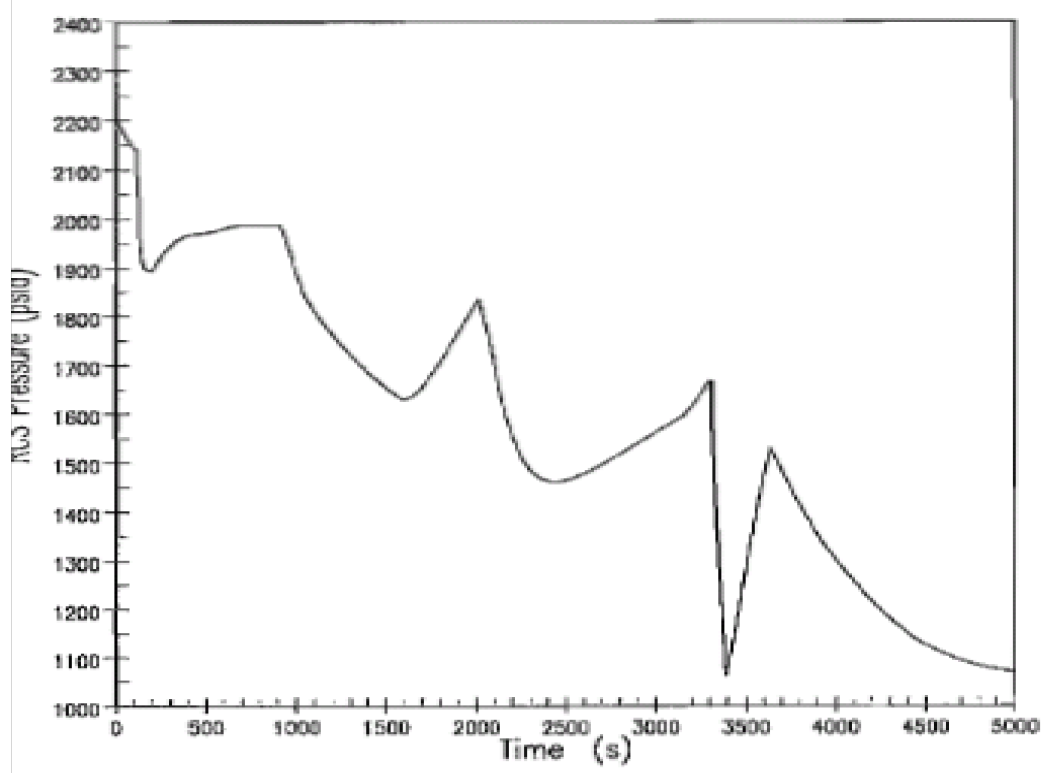
WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

PRESSURIZER LEVEL

FIGURE 15.4-97a

Figure 15.4-97a Steam Generator Tube Rupture Analysis Pressurizer Level



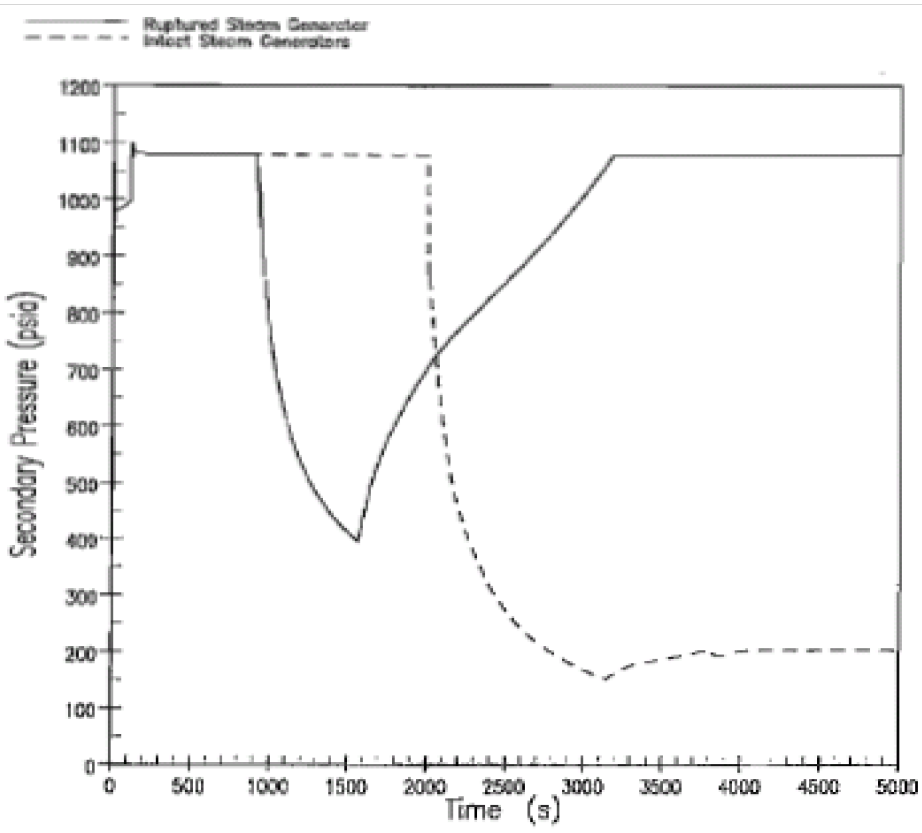
WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

RCS PRESSURE

FIGURE 15.4-97b

Figure 15.4-97b Steam Generator Tube Ruptue Analysis RCS Pressure



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE RUPTURE
ANALYSIS - SECONDARY PRESSURE

FIGURE 15.4-97c

Figure 15.4-97c Steam Generator Tube Ruptue Analysis -Secondary Pressure

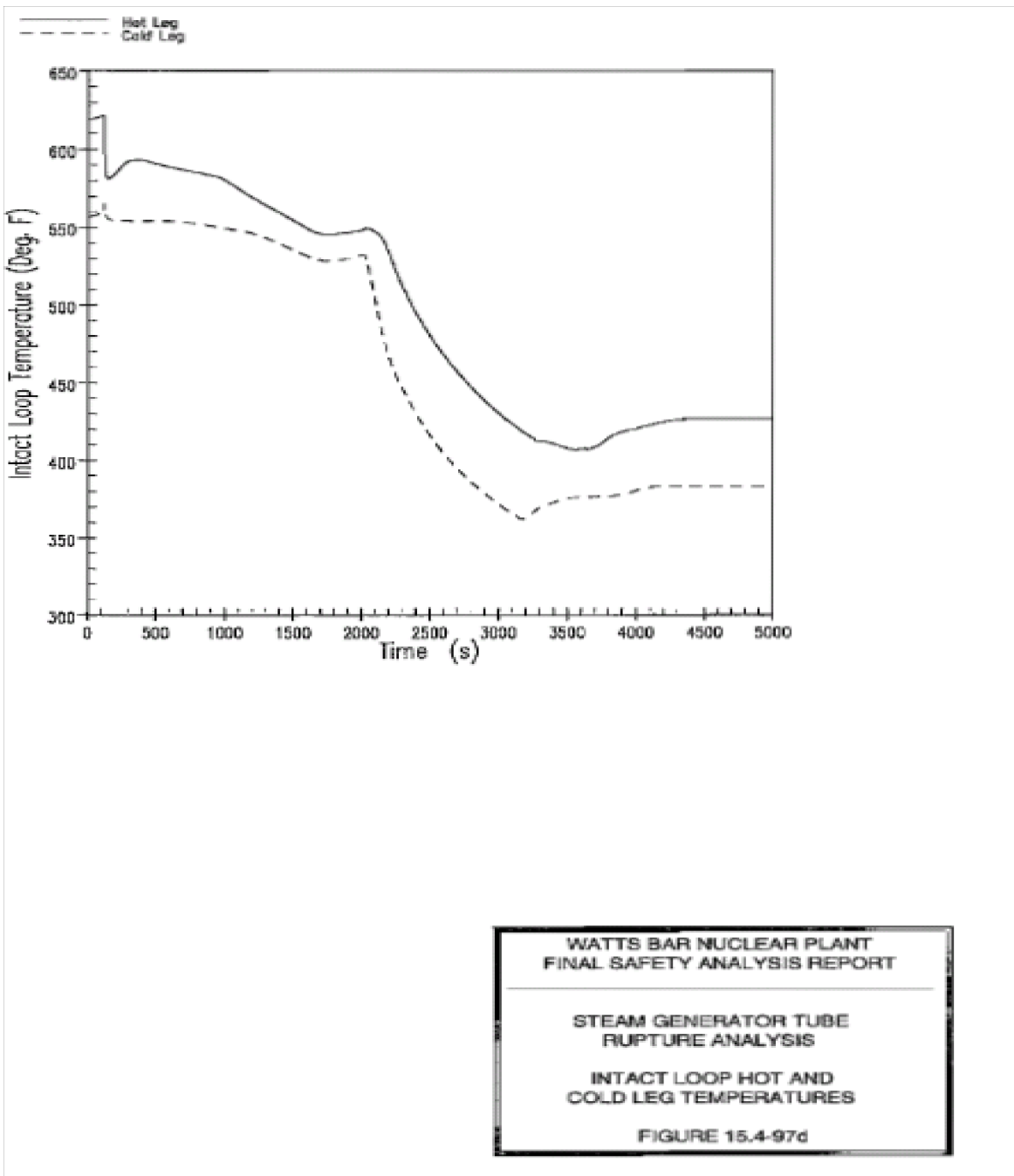
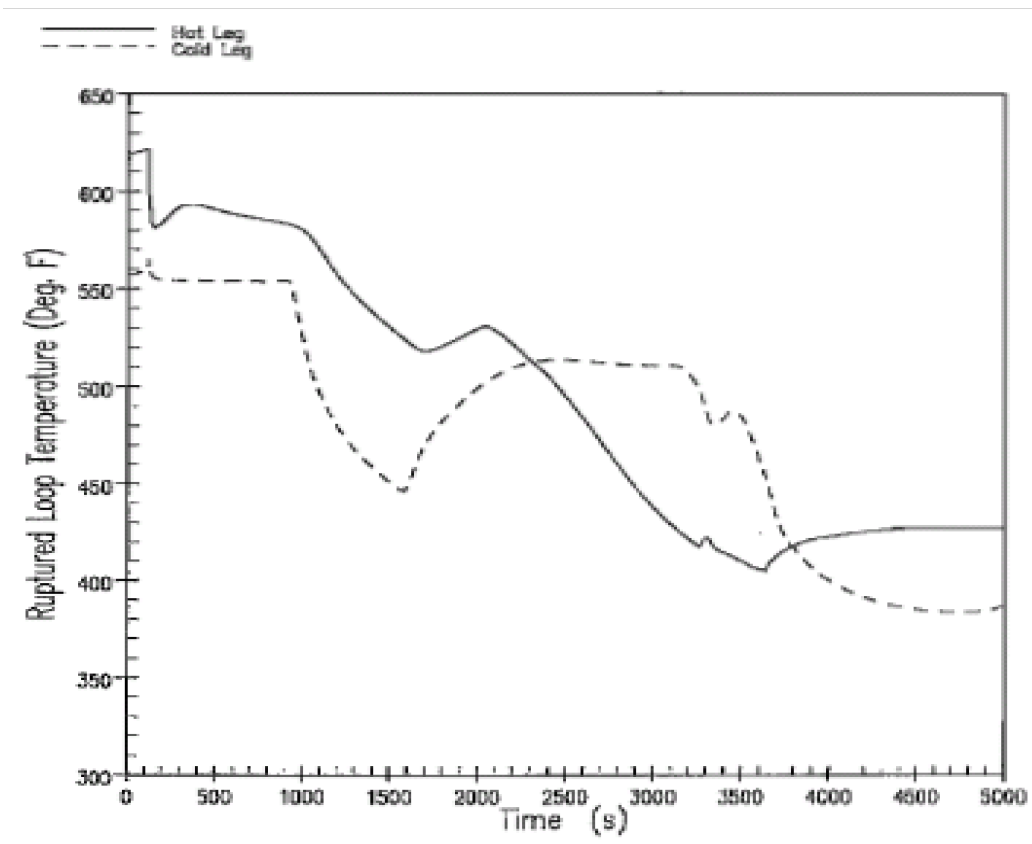


Figure 15.4-97d Steam Generator Tube Ruptue Analysis -Intact Loop Hot and Cold Leg Temperatures



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

RUPTURED LOOP HOT AND
COLD LEG TEMPERATURES

FIGURE 15.4-97e

Figure 15.4-97e Steam Generator Tube Ruptue Analysis -Ruptured Loop Hot and Cold Leg Temperatures

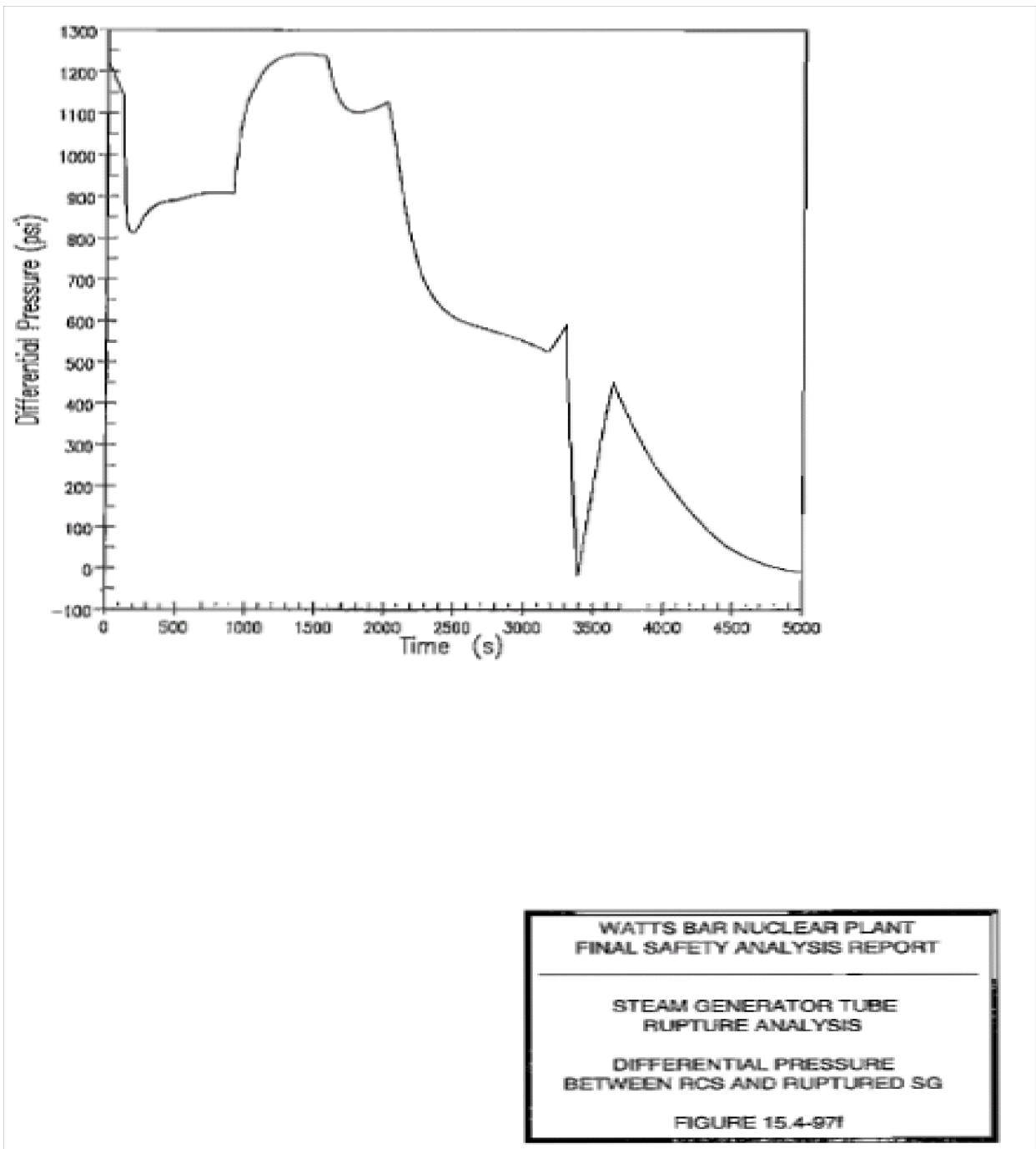
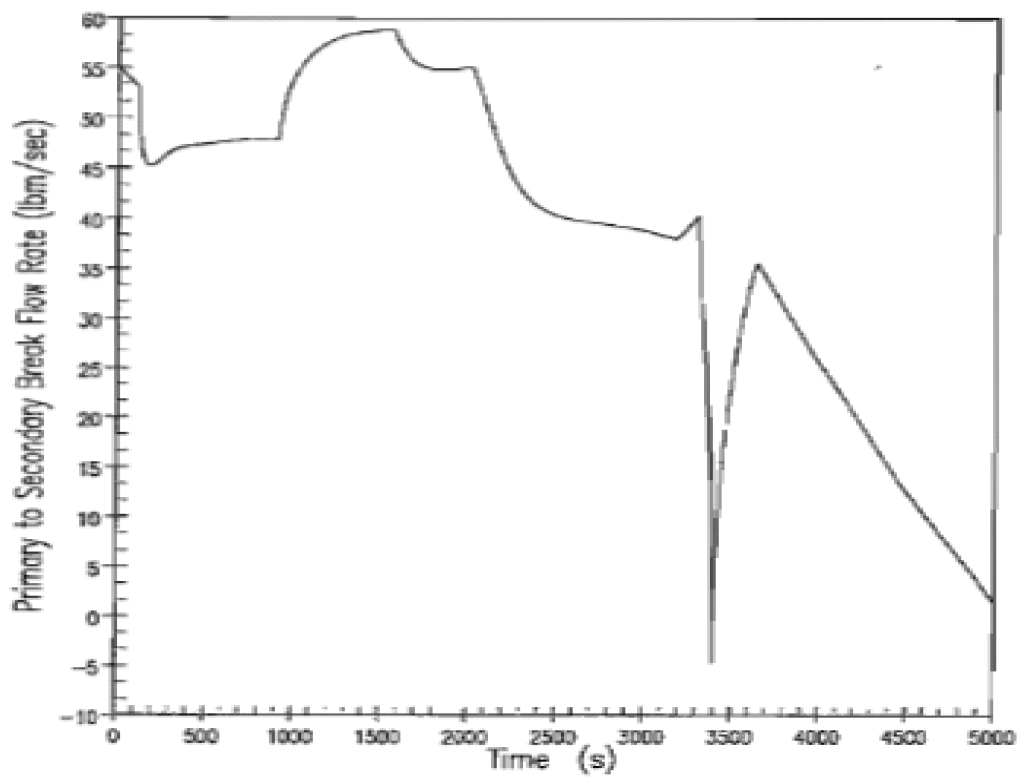


Figure 15.4-97f Steam Generator Tube Ruptue Analysis - Differential Pressure Between RCS and Ruptured SG



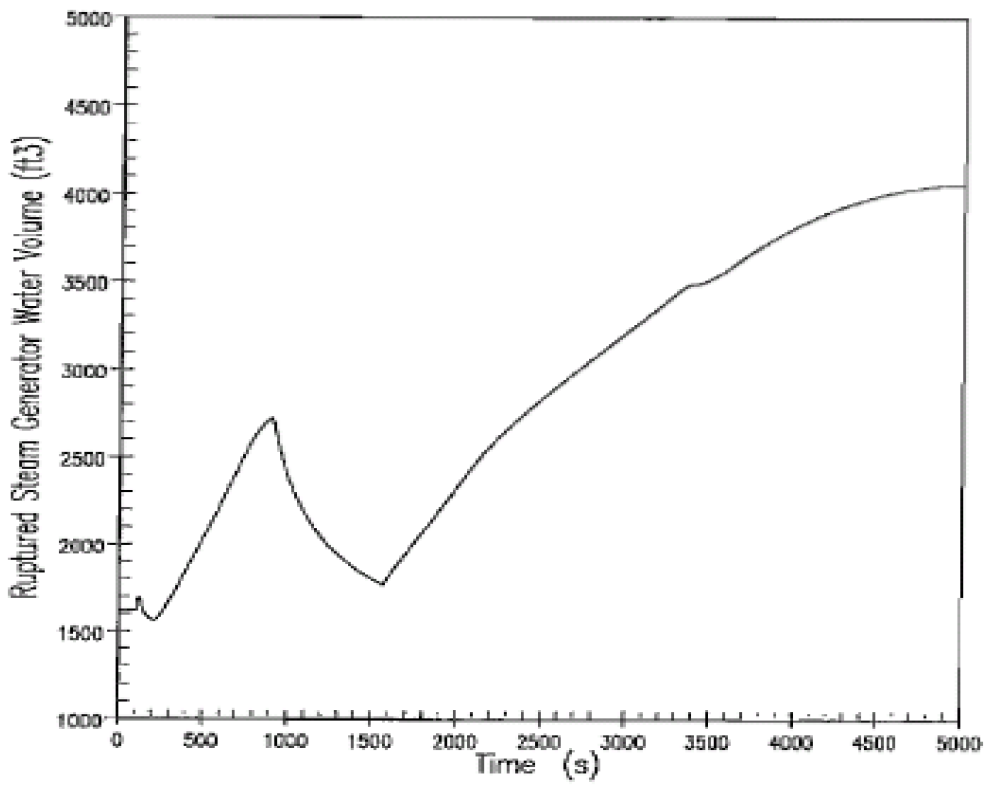
WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

PRIMARY TO SECONDARY
BREAK FLOW RATE

FIGURE 15.4-97g

Figure 15.4-97g Steam Generator Tube Ruptue Analysis - Primary to Secondary Break Flow Rate



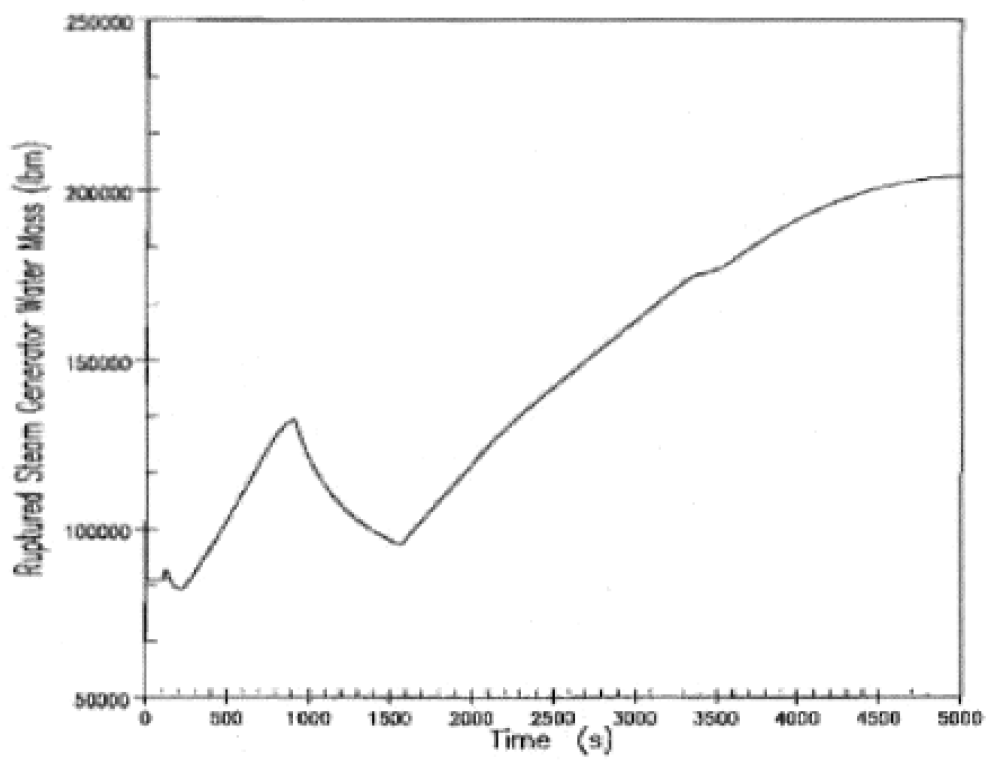
WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

RUPTURED SG WATER VOLUME

FIGURE 15.4-97h

Figure 15.4-97h Steam Generator Tube Ruptue Analysis - Ruptured SG Water Volume·



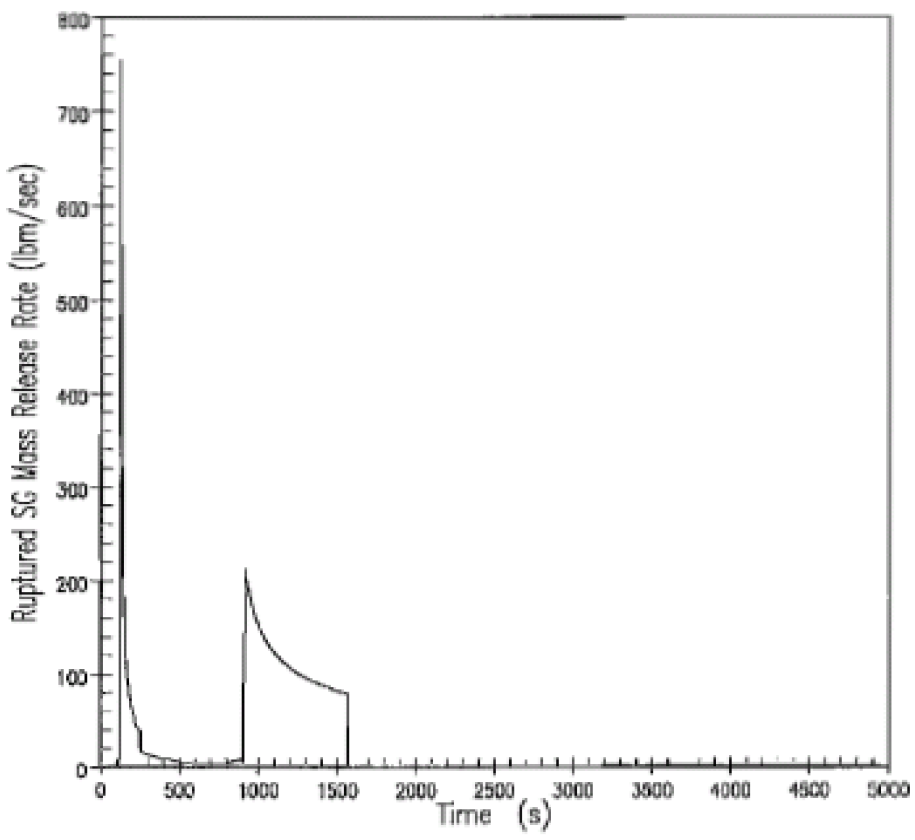
WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

RUPTURED SG WATER MASS

FIGURE 15.4-97i

Figure 15.4-97i Steam Generator Tube Ruptue Analysis - Ruptured SG Water Mass



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TUBE
RUPTURE ANALYSIS

RUPTURED SG MASS
RELEASE RATE TO THE ATMOSPHERE

FIGURE 15.4-97j

Figure 15.4-97j Steam Generator Tube Ruptue Analysis - Ruptured SG Mass Release Rate to the Atmosphere

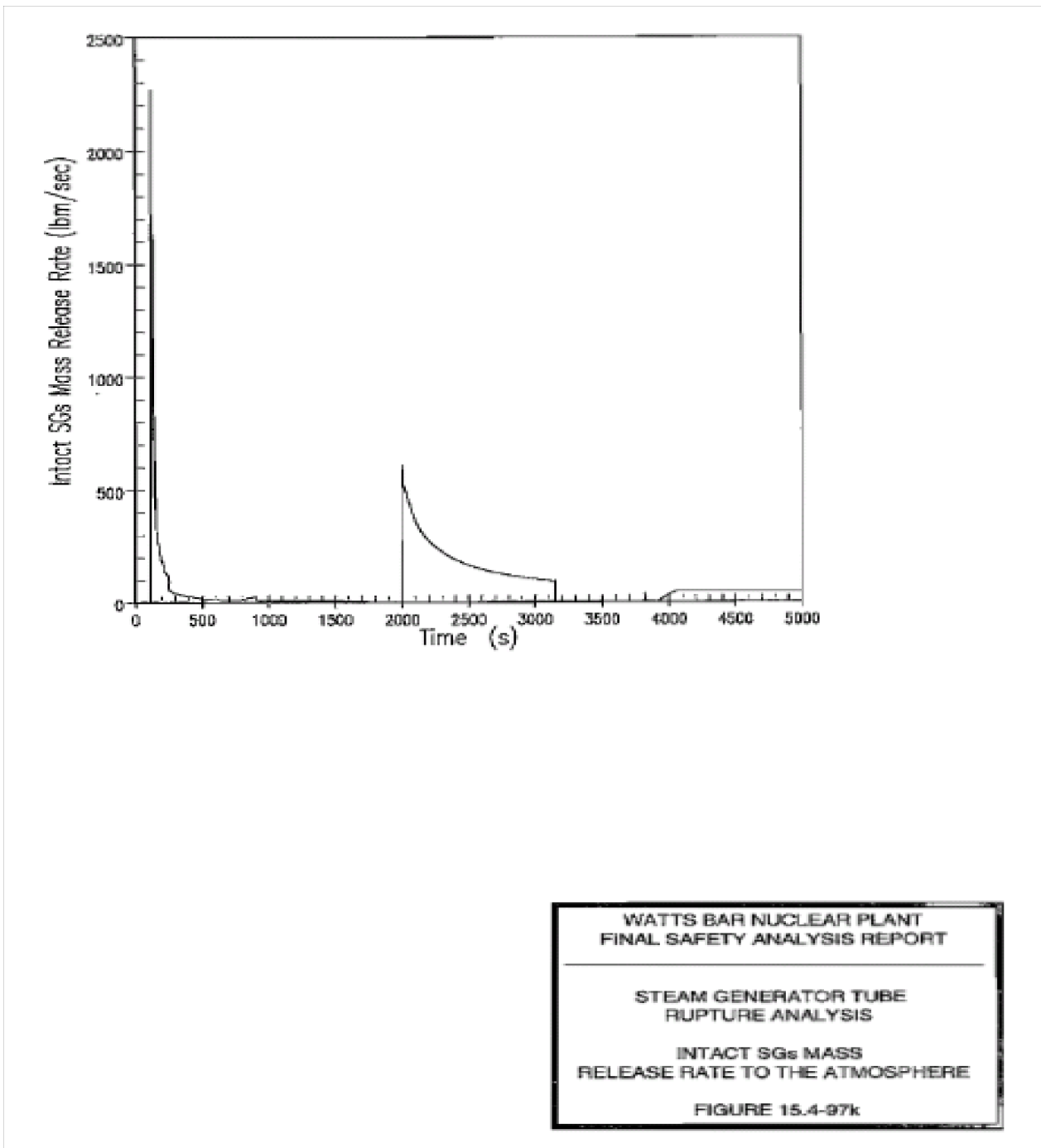


Figure 15.4-97k Steam Generator Tube Ruptue Analysis - Intact SGs Mass Release Rate to the Atmosphere

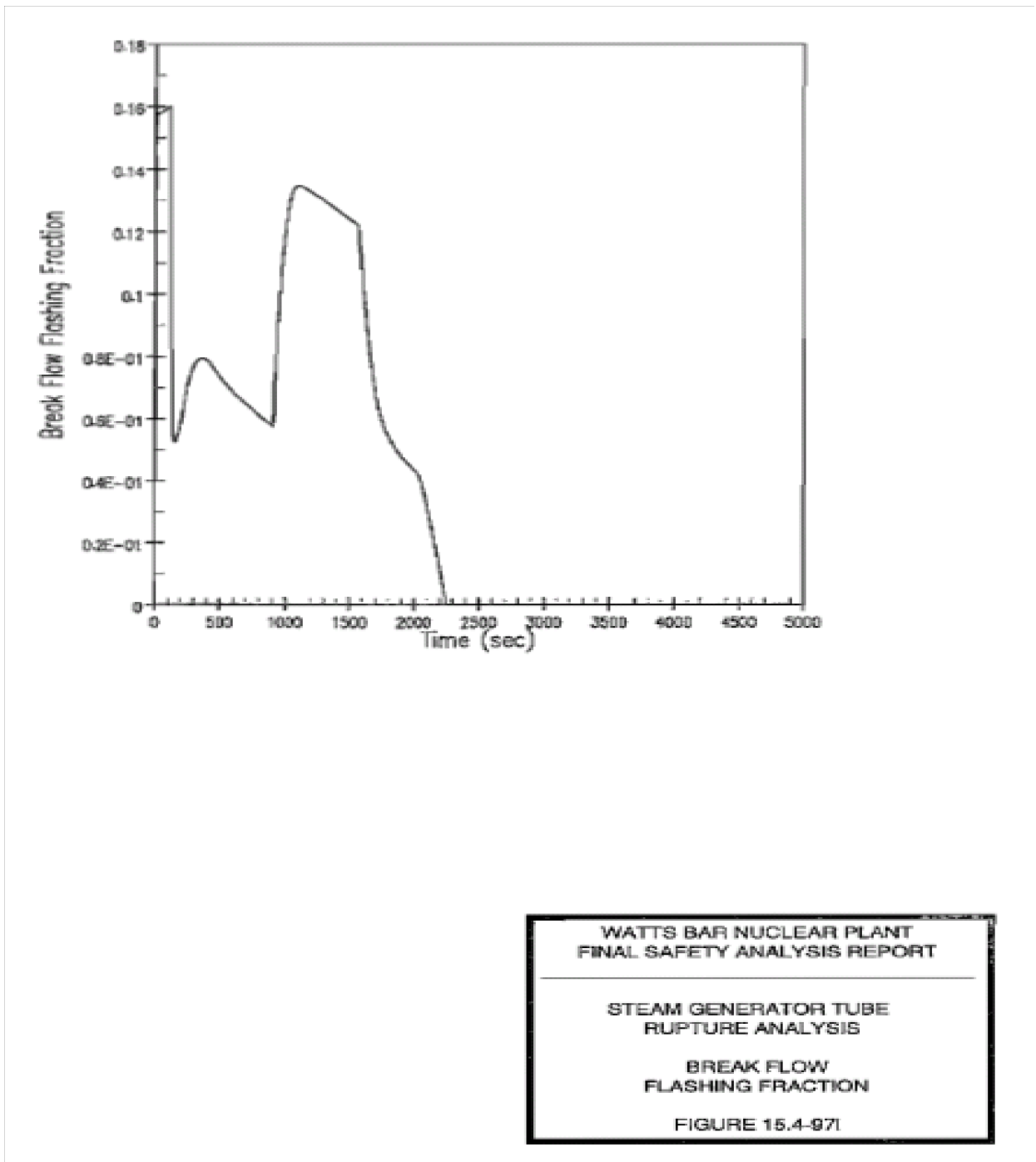


Figure 15.4-97I Steam Generator Tube Ruptue Analysis - Break Flow Flashing Fraction

Figure 15.4-97m Deleted by Amendment 97

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15.5 ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15.5.1 Environmental Consequences of a Postulated Loss of AC Power to the Plant Auxiliaries

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the reactor coolant system (RCS) to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a parameter. This analysis incorporates assumptions of a Technical Specification limit of 0.1 µCi/gm I-131 dose equivalent, and a realistic source term. Parameters used in both the realistic and conservative analyses are listed in Table 15.5-1.

The realistic assumptions that determine the equilibrium concentrations of isotopes in the secondary system are as follows:

(1) Primary coolant activity is associated with 0.125% defective fuel and is given in Table 11.1-7.

(2) The iodine partition factor in the steam generators is:

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}} = 0.01$$

(3) No noble gas is dissolved or contained in the steam generator water, i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off gas system.

(4) The 0-2 and 2-8 hour atmospheric dilution factors given in Appendix 15A and Table 15.5-14; the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable. Doses are based on the dose models in Appendix 15A .

(5) Primary and Secondary side source terms are based on ANSI/ANS–18.1–1984.

Assumptions used for the conservative analysis are the same as the realistic assumptions except the Secondary side source terms at the Technical Specification limit of 0.1 µCi/gm I-131 dose equivalent are assumed.

The steam releases to the atmosphere for the loss of AC power are in Table 15.5-1.

The gamma, beta, and thyroid doses for the loss of AC power to the plant auxiliaries at the exclusion area boundary and low population zone are in Table 15.5-2 for the realistic and conservative analyses. These doses are calculated by the FENCDOSE

computer code[16]. The doses for this accident are less than 25 rem whole body, 300 rem beta and 300 rem thyroid. This is well within the limits as defined in 10 CFR 100.

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-2. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

15.5.2 Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture

Two analyses of the postulated waste gas decay tank rupture are performed:

(1) a realistic analysis, and (2) an analysis based on Regulatory Guide 1.24 (Reference 2). The parameters used for each of these analyses are listed in Table 15.5-3.

The assumptions for the Regulatory Guide analysis are:

(1) The reactor has been operating at full power with 1% defective fuel for the RG 1.24 analysis.

(2) The maximum content of the decay tank assumed to fail is used for the purpose of computing the noble gas inventory in the tank. Radiological decay is taken into account in the computation only for the minimum time period required to transfer the gases from the reactor coolant system to the decay tank. For the Regulatory Guide 1.24 analysis, noble gas and iodine inventories of the tank are given in Table 15.5-4. For the realistic analysis, source terms are based on ANSI/ANS-18.1-1984 methodology[14].

(3) The tank rupture is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank through the Auxiliary Building vent to the outside atmosphere. The assumption of the release of the noble gas inventory from only a single tank is based on the fact that all gas decay tanks will be isolated from each other whenever they are in use.

(4) The short-term (i.e., 0-2 hour) dilution factor at the exclusion area boundary given in Appendix 15A is used to evaluate the doses from the released activity. Doses are based on the dose models presented in Appendix 15A. The gamma, beta, and thyroid doses for the gas decay tank rupture at the exclusion area boundary and low population zone are given in Table 15.5-5 for both the realistic and Regulatory Guide 1.24 analyses.

(5) The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-5. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

15.5.3 Environmental Consequences of a Postulated Loss of Coolant Accident

The results of the analysis presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident do not result in doses which exceed the reference values specified in a 10 CFR 100.

The analysis is based on Regulatory Guide 1.4[3]. The parameters used for this analysis are listed in Table 15.5-6. In addition, an evaluation of the dose to control room operators and an evaluation of the offsite doses resulting from recirculation loop leakage are presented.

Fission Product Release to the Containment

Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the emergency core cooling system keeps cladding temperatures well below melting, and limits zirconium-water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the primary containment.

In this analysis, based on Regulatory Guide 1.4[3], a total of 100% of the noble gas core inventory and 25% of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, it is further assumed that 91% is in elemental form, 4% in methyl form, and 5% in particulate form. The core inventory of iodines and noble gases is listed in Table 15.1-5.

Primary Containment Model

The quantity of activity released from the containment was calculated with a single volume model of the containment.

If it is assumed that there are no sources of activity following the initial instantaneous release of fission products to the containment, the equation which describes the time dependent activity or quantity of material in a component is:

$$\frac{dA_{ij}(t)}{dt} = -\Lambda_{ij}A_{ij}(t) + P_{ij}(t) \qquad (1)$$

where A_{ij} is the activity or quantity of material i in component j. P_{ij} is the rate at which activity or material i is added to component j, and Λ_{ij} is the rate at which activity or material i is removed or lost from component j. If both Λ and P are independent of time, then for one material and one component one obtains the solution:

$$A = A_0 e^{-\Lambda t} + \frac{P}{\Lambda}(1 - e^{-\Lambda t}) \qquad (2)$$

where A_o is the initial activity. However, in general, P is time dependent and in some cases Λ is also time dependent.

The addition of material to the component, $P_{ij}(t)$, may come from two sources: (1) flow from another component in the system may add material to the component, (2) material may be produced within the component by radioactive decay. Thus, the addition rate for material i to component j can be expressed as:

$$P_{ij}(t) = P_{ij}^{(1)}(t) + P_{ij}^{(2)}(t) \qquad (3)$$

where:

$P_{ij}^{(1)}(t) = \sum_{jj \neq j}^{n} c_{ijj-j}(t)A_{ijj}(t); c_{ijj-j}(t)$ is the transfer coefficient of i from component jj to j, and $P_{i}^{(2)}(t) = \sum_{ii}^{n} \Upsilon_{ii-i}A_{iij}(t); \Upsilon_{ii-i}$ is the rate of production of i from ii in component j. Note that $\gamma_{ii\text{-}i}$ is not normally a function of time or component.

Similarly, the loss from a component can be due to: (1) loss within the component (such as radioactive decay), (2) flow out of the component to other components, and (3) removal from the system. Thus, the loss rate from component j for material i can be expressed as:

$$\Lambda_{ij}(t) = \lambda_i + \Lambda_{ij}^{(2)}(t) + \Lambda_{ij}^{(3)}(t) \qquad (4)$$

where λ_i is the removal rate inside the component due to radioactive decay (neither time nor component dependent),

$\Lambda_{ij}^{(2)}(t) = \sum_{jj \neq j}^{n} f_{ij-jj}(t); f_{ij-jj}(t)$ is the transfer coefficient of material i from component j to jj, and $\Lambda_{ij}^{(3)}(t)$ is the removal from the system.

A computer program Source Transport Program (STP) has been developed to solve equation (1) for each isotope and for two halogen forms (i.e., elemental and or organic). From this, the isotopic concentration airborne in the containment as a function of time and the integrated isotopic leakage from the containment for a given time period can be obtained. Parameters used in the loss-of-coolant accident analysis are listed in Table 15.5-6.

Modeling of Removal Process

For fission products other than iodine, the only removal processes considered are radioactive decay and leakage.

The fission product iodine is assumed to be present in the containment atmosphere in elemental, organic, and particulate form. It is assumed that 91% of the iodine available for leakage from the containment is in elemental (i.e., I_2 vapor) form, 4% is assumed to be in the form of organic iodine compounds (e.g., methyl iodine), and 5% is assumed to be absorbed on airborne particulate matter. In this analysis it was conservatively assumed that the organic form of iodine is not subject to any removal processes other than radioactive decay and leakage from the containment. The elemental and particulate forms of iodine are assumed to behave identically.

The effectiveness of the ice condenser for elemental iodine removal is described in Section 6.5.4. For the calculation of doses, the ice condenser was treated as a time dependent removal process. The time dependent ice condenser iodine removal efficiencies for the Regulatory Guide 1.4 analysis are given in Table 15.5-7.

Ice Condenser

The ice condenser is designed to limit the leakage of airborne activity from the containment in the event of a loss-of-coolant accident. This is accomplished by the removal of heat released to the containment during the accident to the extent necessary to initially maintain that structure below design pressure and then reduce the pressure to near atmospheric. The addition of an alkaline solution such as sodium tetraborate enhances the iodine removal qualities of the melting ice to a point where credit can be assumed in the radiological analyses.

The operation of the containment deck fans (air return fans) is delayed for approximately 10 minutes following a Phase B isolation signal resulting from the loss-of-coolant accident.

This delay in fan operation yields an initial inlet steam-air mixture into the ice condenser of greater than 90% steam by volume which results in more efficient iodine removal by the ice condenser.

As a result of experimental and analytical efforts, the ice condenser system has been proven to be an effective passive system for removing iodine from the containment atmosphere following a loss-of-coolant accident. (Reference 4)

With respect to iodine removal by the ice condenser, the following assumptions were made:

(1) The ice condenser is only effective in removing airborne elemental and particulate iodine from the containment atmosphere.

(2) The ice condenser is modeled as a time dependent removal process.

(3) The ice condenser is no longer effective in removing iodine after all of the ice has been melted using the most conservative assumptions.

Primary Containment Leak Rate

The primary containment leak rate used in the Regulatory Guide 1.4 analysis for the first 24 hours is the design basis leak rate guaranteed in the technical specifications regarding containment leakage and it is 50% of this value for the remainder of the 30 day period. Thus, for the first 24 hours following the accident, the leak rate was assumed to be 0.25% per day and the leak rate was assumed to be 0.125% per day for the remainder of the 30 day period.

The leakage from the primary containment can be grouped into two categories: (1) leakage into the annulus volume and (2) through line leakage to rooms in the Auxiliary Building (see Figure 15.5-1). The environmental effects of the core release source events have been analyzed on the basis that 25% of the total primary containment leakage goes to the Auxiliary Building.

The leakage paths to the Auxiliary Building are tested as part of the normal Appendix J testing of all containment penetrations. An upper bound to leakage to the Auxiliary Building was estimated to be 25% of the total containment leakage. Selecting an upper bound is conservative because an increasing leakage fraction to the Auxiliary Building results in an increasing calculated offsite dose. This upper bound was also selected on the basis that it is large enough to be verified by testing. The periodic Appendix J testing will assure that leakage to the Auxiliary Building remains below 25%. The remaining 75% of the leakage goes to the annulus.

Bypass Leakage Paths

There are no bypass paths for primary containment leakage to go directly to the atmosphere without being filtered. For further details see the discussion on Type E leakage paths in Section 6.2.4.3.1.

Auxiliary Building Release Path

The Auxiliary Building allows holdup and is normally ventilated by the auxiliary building ventilation system. However, upon an ABI signal following a loss-of-coolant accident, the normal ventilation systems to all areas of the Auxiliary Building are shutdown and

isolated. Upon Auxiliary Building isolation, the Auxiliary Building gas treatment system (ABGTS) is activated to provide ventilation of the area and filtration of the exhaust to the atmosphere. This system is described in Section 6.2.3.2.3.

Fission products which leak from the primary containment to areas of the Auxiliary Building are diluted in the room atmosphere and travel via ducts and other rooms to the fuel handling area or the waste packaging area where the suctions for the Auxiliary Building gas treatment system are located. The mean holdup time for airborne activity in the Auxiliary Building areas other than the fuel handling area is greater than one hour with the Auxiliary Building isolated and both trains of the ABGTS operating. It has been conservatively assumed in the estimation of activity release that activity leaking to the Auxiliary Building is directly released to the environment for the first four minutes and then through the ABGTS filter system, with a conservatively assumed mean hold-up time of 0.3 hours in the Auxiliary Building before being exhausted. In the Regulatory Guide 1.4 analysis the ABGTS filter system is assumed to have a removal efficiency of 99% for elemental, organic, and particulate iodines. Minor leakage into the ABGTS and EGTS ductwork allows some unfiltered Auxiliary Building air to be released to the environment. This leakage, quantified by testing, is modeled in the LOCA analysis as indicated in Table 15.5-6 and does not significantly impact doses.

The Auxiliary Building internal pressure is maintained at less than atmospheric during normal operation (see Section 9.4.2 and 9.4.3), thereby preventing release to the environment without filtration following a LOCA. The annulus pressure is maintained more negative than the Auxiliary Building internal pressure during normal operation and after a DBA. Therefore, any leakage between the two volumes following a LOCA is into the annulus.

Shield Building Releases

The presence of the annulus between the primary containment and the Shield Building reduces the probability of direct leakage from the vessel to the atmosphere and allows holdup, dilution, sizing, and plate-out of fission products in the Shield Building. The major factor in the effectiveness of the secondary containment is its inherent capability to collect the containment leakage for filtration of the radioactive iodine prior to release to the environment. This effect is greatly enhanced by the recirculation feature of the air handling systems, which forces repeated filtration passes for the major fraction of the primary containment leakage before release to the environment. Seventy-five percent of the primary containment leakage is assumed to go to the annulus volume.

The initial pressure in the annulus is less than atmospheric. However, the dose analysis conservatively assumes the Annulus is at atmospheric pressure at event initiation. After blowdown, the annulus pressure will increase rapidly due to expansion of the containment vessel as a result of primary containment atmosphere temperature and pressure increases. The annulus pressure will continue to rise due to heating of the annulus atmosphere by conduction through the containment vessel. After a delay, the EGTS operates to maintain the annulus pressure below atmospheric pressure.

The EGTS is essentially an annulus recirculation system with pressure activated valves which allow part of the system flow to be exhausted to atmosphere to maintain

a "negative" annulus pressure. The system includes absolute and impregnated charcoal filters for removal of halogens. The EGTS combined with ABGTS ensures that all primary containment leakage is filtered before release to the atmosphere.

The EGTS suction in the annulus is located at the top of the containment dome, while nearly all penetrations are located near the bottom of the containment (see Section 6.2), thereby minimizing the probability of leakage directly from the primary containment into the EGTS.

Transfer of activity from the annulus volume to the EGTS suction is assumed to be a statistical process similar mathematically to the decay process, (i.e., the rate of removal from the annulus is proportional to the activity in the annulus). This corresponds an assumption that the activity is homogeneously distributed throughout the mixing volume. Because of the low EGTS flow rate (compared to the annulus volume), the thermal convection due to heating of the containment vessel, and the relative locations of the EGTS suctions (at the top of the dome) and the EGTS recirculation exhausts (at the base of the annulus), a high degree of mixing can be expected. It is conservatively assumed that only 50% of the annulus free volume is available for mixing of activity in the Regulatory Guide 1.4 analysis.

Tables 15.5-8 and 15.5-8A list the EGTS and recirculation flow rates as a function of time after the LOCA, which were used for calculation of activity releases for the Regulatory Guide 1.4 analysis. Table 15.5-8 flow rates are as a result of a postulated single failure loss of one train of EGTS concurrent with LOCA. Table 15.5-8A flow rates are as a result of an alternate single failure scenario resulting in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. Both EGTS fans are in service until operator action is taken to place one fan in standby between one and two hours post accident. The flow path of fission products which are drawn into the air handling systems is shown schematically in Figure 15.5-1 where:

L_0 Represents the flow of activity from primary containment to the annulus

L_1 Represents the flow of activity from primary containment to the Auxiliary Building

L Represents the flow of activity from the annulus into the EGTS

K Represents the ratio of EGTS recirculation flow to total EGTS flow rate

n_f Represents the appropriate filter efficiency

Effectiveness of Double Containment Design

The analysis has demonstrated clearly the benefits of the double containment concept. As would be expected for a double barrier arrangement, the second barrier acts as an effective holdup tank, resulting in substantial reduction in the two-hour inhalation and whole body immersion doses. The expected offsite doses for the 30-day period at the

low population zone are also substantially reduced, since the holdup process is effective for the duration of the accident.

The EGTS exhaust flow rate is dependent on the rate of air inleakage to the annulus. In fact, after about 30 minutes following blowdown of the reactor vessel the EGTS exhaust flow is approximately equal to the air inleakage rate. Studies[5] made of leak rates from typical concrete buildings of this type have resulted in leak rates from 4% to 8% per day at a pressure differential of 14 inches of water. Although the pressure differential in this case will be much lower than this value, it has been assumed that a shield building inleakage flow of 250 cfm exists throughout the 30-day period for the single failure scenario which results in loss of one EGTS train concurrent with a LOCA. The inleakage flow for the single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack (while the other train remains functional) was conservatively assumed to be greater since the resulting annulus pressure is more negative than the original single failure scenario loss of one EGTS train. The long term inleakage flow rates of 832 cfm (until operator action to place one fan in standby) and 604 cfm thereafter are used in the dose analysis. This inleakage flow includes leakage past ventilation system primary containment isolation valves assuming that a single isolation valve fails in the open position.

In order to evaluate the effectiveness of the Shield Building, the following case was analyzed:

50% Mixing Case

At the beginning of the accident, the EGTS starts exhausting filtered fission products to the environs (see Tables 15.5-8 and 15.5-8A). At approximately 114 seconds for the loss of one EGTS train the Annulus pressure becomes less than -0.25 inches w.g. and the effluents are filtered for the duration of the accident. At approximately 60 seconds (for the single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functonal) the annulus pressure becomes less than minus 0.25 inches w.g., and the effluents are filtered for the duration of the accident. All of the primary containment leakage going to the shield building is assumed to be uniformly mixed in 50% of the annulus free volume.

Emergency Gas Treatment System Filter Efficiencies

The EGTS takes suction from the annulus, and the exhaust gases are drawn through two banks of impregnated charcoal filters in series. Sufficient filter capacity is provided to contain all iodines, inorganic, organic, and particulate available for leakage. Since the air in the annulus is dry, filter efficiencies of greater than 99% are attainable as reported in ORNL-NSIC-4[6]. Heaters and demisters have been incorporated upstream of the filters resulting in a relative humidity of less than 70% in the air entering the filters which further ensures high filter efficiency.

In the Regulatory Guide 1.4 analysis however, an overall removal efficiency of 99% for elemental, organic, and particulate iodine is assumed for the two filter banks in series.

Discussion of Results

The gamma, beta, and thyroid doses for the LOCA at the exclusion area boundary and the low population zone are given in Table 15.5-9. These doses are calculated by the FENCDOSE computer code[16]. The doses are based on the atmospheric dilution factors and dose models given in Appendix 15A. The doses for this accident are less than 25 rem whole body, 300 rem beta, and 300 rem thyroid. The doses are well within the 10 CFR 100 guidelines and reflect the worst case values in consideration of both single failure scenarios.

Loss of Coolant Accident - Environmental Consequences of Recirculation Loop Leakage

Component leakage in the portion of the emergency core cooling system outside containment during the recirculation phase following a loss of coolant accident could result in offsite exposure. The maximum potential leakage for this equipment is specified is Table 6.3-6. This leakage refers to specified design limits for components and normal leakage is expected to be well below those upper limits. Recirculation is assumed in the analysis to start at 10 minutes after the loss of coolant accident. At this time the sump temperature is approximately 160°F (Figure 6.2.1-3). The enthalpy of the sump is approximately 130 BTU/lb. The enthalpy of saturated liquid at 1.0 atmosphere pressure and 212°F is greater than 130 BTU/lb. Therefore, there will be no flashing of the leakage from recirculation loop components, and an iodine partition factor of 0.1 is assumed for the total leakage.

The analysis of the environmental consequences is performed as follows:

Core iodine inventory given in Table 15.1-5 is used. The water volume is comprised of water volumes from the reactor coolant system, accumulators, refueling water storage tank, and ice melt. All the noble gases are assumed to escape to the primary containment. Ninety-seven percent of trituim was assumed to remain liquid and accumulate in the sump, while 3% was assumed to go airborne to the containment. An alternate analysis was also performed assuming 100% of the tritium goes airborne into the containment. Radioactive decay was taken into account in the dose calculation. The major assumptions used in the analysis are listed in Table 15.5-12. The offsite doses at the exclusion area boundary and low population zone for the analysis are given in Table 15.5-13 and reflect the worst case values in consideration of 3% airborne tritium or 100% airborne tritium. The atmospheric dilution factors and dose models discussed in Appendix 15A are used in the dose analysis. The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-13. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

Loss of Coolant Accident - Control Room Operator Doses

In accordance with General Design Criterion 19, the control room ventilation system and shielding have been designed to limit the whole body gamma dose during an accident period to 5 rem, the thyroid dose to 30 rem and the beta skin dose to 30 rem.

The doses to personnel during a post-accident period originate from several different sources. Exposure within the control room may result from airborne radioactive nuclides entering the control room via the ventilation system. In addition, personnel are exposed to direct gamma radiation penetrating the control room walls, floor, and roof from:

(1) Radioactivity within the primary containment atmosphere

(2) Radioactivity released from containment which may have entered adjacent structures

(3) Radioactivity released from containment which passes above the control room roof

Further exposure of control room personnel to radiation may occur during ingress to the control room from the exclusion area boundary and during egress from the control room to the exclusion area boundary.

In the event of a radioactive release incident, the control room is isolated automatically by a safety injection system signal and/or by radiation signal from beta detectors located in the air intake stream common to the air intake ports at either end of the Control Building. These redundant signals are routed to redundant controls which actuate air-operated isolation dampers downstream of the beta detectors. Operation of the emergency pressurizing fans with inline HEPA filters and charcoal adsorbers is also initiated by these signals. Simultaneously, recirculation air is rerouted automatically through the HEPA filters and charcoal adsorbers. Approximately 711 cfm of outside air, the emergency pressurization air, flows through a duct routed to the emergency recirculation system upstream of the HEPA filters and charcoal adsorbers. This flow of outside air provides the control room with a slight positive pressure relative to the atmosphere outside and to surrounding structures. In addition, the equivalent of 51 cfm of unfiltered outside air enters through the main control room doors and other sources. Isolation dampers located in each intake line may be selectively closed by control room personnel. The selection between the two would be based on the objective of admitting a minimum of airborne activity to the control room via the makeup airflow.

The control room ventilation flow system is shown in Figure 9.4-1.

To evaluate the ability of the control room to meet the requirements of General Design Criterion 19, a time-dependent model of the control room was developed. In this model, the outside air concentration enters the control room via the isolation damper bypass line and the HEPA filters and charcoal absorbers. The concentration in the room is reduced by decay, leakage out, and by recirculation through the HEPA filters

and charcoal absorbers. Credit for filtration is taken during two passes through the charcoal absorbers. Using these assumptions, the following equations for the rate of change of the control room concentrations are obtained:

$$\frac{dM}{dt} = C_o(1-K_1)L/V - (L/V)M - \frac{R_c}{V}M - \lambda M \qquad (1)$$

$$\frac{dN}{dt} = \frac{R_c}{V}(1-K_2)\ M - (L/V)N - \lambda N \qquad (2)$$

$$C(t) = M(t) + N(t) \qquad (3)$$

Where:

M(t) = Once-filtered time-dependent concentration

N(t) = Twice-filtered (or more) time-dependent concentration

C(t) = Total time-dependent concentration in control room

C_o = Concentration of isotope entering air intake

K_1 = Filter efficiency for a particular isotope during first pass

K_2 = Filter efficiency for a particular isotope during second pass

L = Flow rate of outside air into control room and leakage out of control room

R_c = Recirculated air flow rate through filters

λ = Decay constant

V = Control room free volume

These equations are readily solvable if C_o is constant or a simple function of time during a time interval. Since C_o consists of a number of terms involving exponentials, it was assumed to be constant during particular time intervals corresponding to the average concentration during each interval as described below. Solving equations (1), (2), and (3) yields:

$$C(t) = \left[\frac{1-K_1 1-K_2 C_o}{W_m V}\right] \times \left[\frac{L}{(1-K_2)}(1-e^{-Wmt}) + \frac{R_c L}{W_n V}(1-e^{-W_n t}) - L(e^{-W_n t} - e^{-W_m t})\right] \qquad (4)$$

Where:

$$W_m = \frac{(L + R_c + \lambda V)}{V}$$

$$W_n = \frac{(L + \lambda V)}{V}$$

The value of C_o used in equation (4) is determined as follows:

$$C_{oi} = (X/Q)_i \frac{\int_{t_i}^{t_{i+1}} R\,dt}{t_{i+1} - t_i} \qquad (5)$$

C_{oi}= Average concentration of activity outside control room during ith time period (Ci/m^3).

$(x/Q)_i$= Atmospheric dilution factor (sec/m^3) during the ith time period.

R= Time dependent release rate of activity from containment (Ci/sec).

The atmospheric dilution factors were determined using the accumulated meteorological data on wind speed, direction, and duration of occurrence obtained from the Watts Bar plant site applied to a building wake dilution model. The dilution factors are calculated by the ARCON96 methodology[8] and are the maximum values for each time period. The worst case is Unit 1 exhaust to intake 2. These factors are applied for the first 8 hours, at which time it is assumed that the operator selects intake 1 which has more favorable dilution factors. The values used in the analysis are given in Table 15.5-14.

Equation (4) is used to determine the concentration at any time within a time period and upon integrating and dividing by the time interval gives the average concentration during the time interval due to inflow of radioactivity with outside air as shown:

$$\overline{C_i} = \int_0^T \frac{C_i(t)dt}{T - 0} \qquad (6)$$

Where:

$T = t - t_{i-1}$

t = Time after accident

t_{i-1} = Time at end of previous time period

Further contributions to the concentration during the time period are due to the concentrations remaining from prior time periods. These contributions are obtained from the following equations:

$$C_{R(i+j)} = M_{R(i+j)} + N_{R(i+j)} \quad (7)$$

$$\frac{dM_{R(i+j)}}{dt} = -(L/V + R_c/V + \lambda)M_R(i+j) \quad (8)$$

$$\frac{dN_{R(i+j)}}{dt} = (R_c/V)(1-K_2)M_{R(i+j)} - (L/V+\lambda)N_{R(i+j)} \quad (9)$$

With initial conditions:

$M_{R(i+j)}(0) = M_{R0(i)} =$ (Once-filtered concentration at end of the ith time period.)

$N_{R(i+j)}(0) = N_{R0(i)} =$ (Twice-filtered, or more, concentration at end of the ith time period.)

Solving equations (8) and (9) and substituting certain initial condition relations, equation (7) becomes:

$$C_{R(i+j)} = C_{R0(i)}e^{-W_N(t-ti)} - M_{R0(i)}K_2(e^{-W_N(t-ti)} - e^{-W_M(t-ti)}) \quad (10)$$

Integrating equation (10) for each of the prior time periods gives the contribution from these time periods to the present time period. The average concentration is determined for these contributions using the method of equation (6).

Filter efficiencies of 95% for elemental and particulate iodine and 95% for organic iodine were deemed appropriate for the first filter pass. Since the concentrations of iodine in the main control room are such reduced as a result of this filtration, the efficiencies were reduced for the second pass to 70% for elemental and particulate iodine, and 70% for organic iodine.

To account for the unfiltered inleakage, a bypass leak rate (BPR) of 51 cfm was added to the makeup flow (L in equation (1)) of 711 cfm, and the filter factor for the first pass was decreased by the ratio L/(L+BPR). The filter efficiencies for the second pass are not affected by the unfiltered inleakage.

The filter efficiency for noble gases was taken as zero for all cases.

The above equations were incorporated into computer program COROD[17] together with appropriate equations for computing gamma dose, beta dose, and inhalation dose using these average nuclide concentrations and time periods. The whole body gamma dose calculation consists of an incremental volume summation of a point kernel over the control room volume. The principal gammas of each isotope are used to compute the dose from each isotope. The dose computations for beta activity were based on a semi infinite cloud model. Doses to thyroid were based on activity to dose conversion factors. (The equations and various data are given below.) The doses from these calculations are presented in Table 15.5-9. Gamma dose contributions from shine through the control room roof due to the external cloud and from shine through the control room walls from adjacent structures and from containment are computed using an incremental volume summation of a point kernel which includes buildup factors for the concrete shielding. For the calculation of shine through the control room roof, an atmospheric, rectangular volume several thousand feet in height and several control room widths was used. The control room roof is a 2 foot 3-inch-thick concrete slab and is the only shielding considered in this calculation. The average isotope concentrations at the control bay for each time period were used as the source concentrations. For the shine from adjacent structures, the shielding consists of the 3-foot-thick (5 feet in certain areas) control room walls. The doses are calculated similarly to the shine dose through the roof. The average isotope concentrations at the control bay intake for each time period are also used for these calculations.

The shine from the spreading room below the control room is also computed in the same manner as adjacent structures.

Shielding for this computation consists of the 8-inch-thick concrete floor. The summation of the incremental elements is performed over the volume of each room or structure of interest.

In addition to the dose due to shine from surrounding structures and from the passing cloud, the shine from the reactor containment building also contributes to the gamma whole body dose to personnel. This contribution is computed in the same manner as the methods used above. Due to the location of the Auxiliary Building between the Reactor Buildings and the control room and the thicker control room auxiliary building wall near the roof, the minimum ray path through concrete from the containment into the control room below 10 feet above the control floor, is 8 feet. All nuclides released to containment are assumed uniformly distributed and their time-dependent concentrations were used to compute the dose. The dose computed from this source is small.

Several doors penetrate the control room walls, and the dose at these areas would be larger than the doses calculated as described above. The potential shine at these doors and at other penetrations has been evaluated. As a result, hollow steel doors filled with no. 12 lead shot have been incorporated into the design of the shield wall between the control room and the Turbine Building. These doors provide shielding comparable to the concrete walls. Shine through other penetrations was found to be negligible.

Another contribution to the total exposure of control room personnel is the exposure incurred during ingress from and egress to the exclusion area boundary. The doses due to ingress and egress were computed based on the following assumptions:

(1) Five minutes are required to leave the control room and arrive at car or vice versa.

(2) The distance traveled on the access road to the site exclusion boundary is estimated to be 1500 meters. The average car speed is assumed to be 25 mph.

(3) One one-way trip first day, one round-trip/day 2nd through 30th days.

The control room occupancy factors used in this calculation were taken from Murphy and Campe[9]. They are:

100% occupancy 0-24 hours

60% occupancy 1-4 days

40% occupancy 4-30 days.

All atmospheric dilution factors were conservatively based on 5th percentile wind velocity averages.

It was also assumed that initially the makeup air intake would be through the vent admitting the highest radioisotope concentration, but that the main control room personnel would switch intake vents 8 hours after the accident in order to admit a lower amount of airborne activity to the MCR via the makeup air flow.

The whole body, beta, and thyroid doses from the radiation sources discussed above are presented in Table 15.5-9. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose Equations, Data, and Assumptions

The dose from gamma radiation originating within the control room is given by:

$$D_{\Upsilon} = 1.69 \times 10^{4} \sum_{i=1}^{\alpha} \left[\sum_{k=1}^{\beta} TCOT_{ik} \left(\sum_{l=1}^{\Upsilon} \left\{ E_{kl} f_{kl} \left(\frac{\mu_e}{\rho} \right)_l \sum_{m=1}^{\varepsilon} \sum_{n=1}^{\omega} \sum_{q=1}^{\sigma} \frac{\exp(-\mu_{al}\sqrt{x_m^2 + y_n^2 + z_q^2})}{(x_m^2 + y_n^2 + z_q^2)} \cdot \Delta x \Delta y \Delta z \right\} \right) \right] \quad (11)$$

Where:

DK = Absorbed dose in flesh in mrads

$TCOT_{ik}$=Total concentration integrated over time period i of isotope k in curies/m^3

$E_{k\ell}$ = Energy of gamma ℓ from isotope k in MeV

$f_{k\ell}$ = Number of ℓ gammas of isotope k given off per disintegration

$\left(\frac{\mu_e}{\rho}\right)_l$ = Mass attenuation coefficient for flesh determined at the energy of gamma ℓ in cm^2/gram

$\mu_{l\backslash}$=Linear attenuation coefficient for air determined at the energy of gamma ℓ in inverse meters

x_m, y_n, z_q=Coordinate distances from the dose point to the source volume element (m,n,q) in meters

$\Delta x, \Delta y, \Delta z$=Dimensions of source element (m,n,q)

α=Number of time periods

β=Number of isotopes

K=Number of gammas from an isotope

ε=Number of intervals in the x direction

ω=Number of intervals in the y direction

σ=Number of intervals in the z direction

The control room radiation dose from gamma radiation originating outside of the control room and penetrating concrete walls is given as:

$$D_\Upsilon = 1.69 \times 10^4 \sum_{i=1}^{\alpha} \left[\sum_{k=1}^{\beta} C_{o_{ik}} \left(\sum_{l=1}^{\Upsilon} \left\{ E_{kl} f_{kl} \left(\frac{\mu_e}{\rho}\right)_l \sum_{m=1}^{\varepsilon} \sum_{n=1}^{\omega} \sum_{q=1}^{\sigma} \frac{\exp(-\mu_{al}\sqrt{x_m^2 + y_n^2 + z_q^2})}{(x_m^2 + y_n^2 + z_q^2)} \cdot \exp(-\mu_{cl} t_c \sec\theta) \right.\right.\right.$$

$$\left.\left.\left. \cdot \; B_c(\mu_{cl} t_c \sec\theta) \cdot \Delta x \Delta y \Delta z \right\} \right) \right] (t_i - t_{i-1})$$

Where:

μ_{cl} = Linear attenuation coefficient of concrete determined at the energy of gamma ζ in inverse meters

t_c= Concrete shield thickness in meters

θ = Angle between a vector normal to the shield and a vector from the dose point to the source point

$B_c(\mu_{cl}t_c\sec\theta)$ = Buildup factor for concrete

C_{oik} = Average concentration of isotope k outside the control room during time period i in curies/m^3

t_{i-1},t_i = Times at the beginning and end of time period i in hours

Other parameters are defined as previously noted.

The dose from beta radiation is given by the semi-infinite cloud immersion dose:

$$D_B = (0.230)(X/Q)\left[\sum_{i=1} Q \sum_{k=1} E_{ik} f_{ik}\right] \qquad (12)$$

Where:

D_B=Dose due to beta in rem

X/Q=Atmospheric dispersion factor during time period in sec/m^3

Q_i=Accumulated activity release of isotope i during time period

E_{ik}=Average energy of beta k of isotope i

f_{ik}=Number of k betas of isotope i per disintegration

For beta dose in the control room, equation (12) becomes:

$$D_B = (0.230)\sum_{i=1}^{\delta}\sum_{i=1}^{\alpha}\overline{C}_{ij}\sum_{1}^{\beta}E_{ik}f_{ik}(tj - t_{j-1})$$

Where:

$\overline{C}_{ij}$ =Average concentration of isotope i during time period j

Inhalation Dose (Thyroid)

The inhalation dose for a given period of time has the general form:

$$D_I = (X/Q)(B)\left[\sum_{i=1}^{n}(Q_{i_j})(DCF_i)\right](t_J - t_{j-1}) \qquad (13)$$

Where:

D_I=Thyroid inhalation dose, rem

X/Q=Site dispersion factor during time period, sec/m^3

B=Breathing rate during time period, m^3/hr

Q_{ij}=Average activity release rate during time period j of iodine isotope i

DCF_i=ICRP-30 Dose conversion factor for iodine isotope i, rem/microcurie inhaled

t_j=Total time at end of period j, hours

For inhalation dose within the control room, equation (13) becomes:

$$D_I = (B)\left[\sum_{i=1}^{n} C_{ij}(DCF_i)\right](t_j - t_{j-1})$$

In this expression C_{ij}, the average concentration of isotope i during time period j, has replaced the following factor:

(X/Q) Q_{ij}

The C_{ij}'s are those determined by equations (4) and (6). The breathing rate factor B, was taken to be 3.47 x 10^{-4} m^3/sec, 1.75 x 10^{-4} m^3/sec, and 2.32 x 10^{-4} m^3/sec for the time intervals of 0-8 hours, 8-24 hours, and 24 hours - 30 days, respectively.

15.5.4 Environmental Consequences of a Postulated Main Steam Line Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is a leakage from the reactor coolant system to the secondary system in the steam generator. An acceptable primary-to-secondary leakage rate for the main steam line break (MSLB) accident is 1 gallon per minute (gpm) for the faulted steam generator loop and 150 gallons per day (gpd) for each unfaulted steam generator.

A calculation determines the offsite and main control room doses resulting from a MSLB incorporating the above primary-to-secondary criteria. The calculation determined that 1 gpm (at standard temperature and pressure) primary-to-secondary leakage in the faulted steam generator results in site boundary doses within 10CFR100 guidelines and control room doses within the 10CFR50, Appendix A, General Design Criteria (GDC)-19 limit. The calculation uses TVA computer codes STP, FENCDOSE and COROD. The STP output is input to COROD, which determines control room operator dose and FENCDOSE, which determines the 30-day low population zone (LPZ) and the 2-hour exclusion area boundary (EAB) dose.

Two methods for determining the resultant dose for a MSLB in accordance with the Standard Review Plan 15.1.5, Appendix A methodology are:

1. A pre-accident iodine spike where the iodine level in the reactor coolant spiked upward to the maximum allowable limit of 14 µCi/gm I-131 dose equivalent just prior to the initiation of the accident.

2. The reactor coolant at the maximum steady state dose of equivalent I-131 of 0.265 µCi/gm with an accident initiated iodine spike consisting of a 500 times increase on the rate of iodine release from the fuel.

In both cases, the primary-to-secondary side leak is assumed to be 1 gpm in the faulted steam generator loop and 150 gpd in each unfaulted loop. The primary side activity release uses the Technical Specification (TS) limit design reactor coolant activities, and the secondary side activity uses the Technical Specification limit of $\leq$0.1µCi/gm I-131 dose equivalent.

The steam releases to the atmosphere for the MSLB are in Table 15.5-16.

The gamma, beta and thyroid doses for the MSLB accident at the EAB and LPZ are in Table 15.5-17. The doses from this accident are less than the reference values as listed in 10CFR100 (25 rem whole body and 300 rem thyroid).

The whole body, beta and thyroid doses to control room personnel from the radiation sources discussed above are in Table 15.5-17. The doses are calculated by the COROD computer code. [17] Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] determine the dose. Dose conversion factor in ICRP-30 [25] determine thyroid doses in place of those found in TID-14844.

Assumptions for the MSLB accident:

1. RCS letdown flow of 124.39 gpm.

2. RCS letdown demineralizer efficiency is 1.0 for iodines.

3. ANSI/ASN-18.1-1984 spectrum scaled up to 0.265 or 14 µCi/gm equivalent iodine.

4. Two cases were used. In the first, pre-accident iodine spike of 14 µCi/gm I-131 dose equivalent in the RCS. In the second case, an accident initiated spike which increases the iodine concentration at the equilibrium into the reactor coolant from the fuel rods.

5. Primary side to secondary side leakage of 150 gpd (standard temperature and pressure) per steam generator in the intact loops.

6. The primary-to-secondary leakage mass release to the Environment is 1 gpm (standard temperature and pressure) from the faulted loops.

7. Steam generator secondary inventory released as steam to the atmosphere:

   a) total from the non-defective steam generators (0-2 hrs), 433,079 lbm

   b) total from the non-defective steam generators (2-8 hrs), 870,754 lbm

   c) total from the faulted steam generator (0-30 mins), 96,100 lbm

8. Iodine partition coefficients from steaming of steam generator water:

   i. non-defective steam generators initial inventory and primary-to-secondary leakage, 100.

   ii. faulted steam generator initial inventory and primary-to-secondary leakage, 1.0

9. Atmospheric dilution factors, x/Q, are in Table 15A-2 for offsite and Table 15.5-14 for control room personnel.

10. Main control room related assumptions are in Table 15.5-14.

15.5.5 Environmental Consequences of a Postulated Steam Generator Tube Rupture

Thermal and hydraulic analyses determine the plant response for a design basis steam generator tube rupture (SGTR), and the integrated primary to secondary break flow and mass releases from the ruptured and intact steam generators (SGs) to the condenser and the atmosphere (Section 15.4.3). An analysis of the environmental consequences of the postulated SGTR has also been performed, utilizing the reactor coolant mass and secondary steam mass releases determined in the base thermal and

hydraulic analysis (See Section 15.4). Table 15.5-18 summarizes the parameters used in the SGTR analysis.

The SGTR thermal and hydraulic analysis documents use WBN specific parameters and actual operator performance data, as determined from simulator exercises utilizing the appropriate emergency operating procedures (EOPs). Two cases were analyzed. Case 1: The primary side activity release use the maximum Technical Specification (TS) limit design reactor coolant activities and an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state concentration of 0.265 µCi/gm of I-131 dose equivalent. Case 2: The initial reactor coolant activity is at 14 µCi/gm of I-131 dose equivalent due to a pre-accident iodine spike caused by an RCS transient. For both cases, the secondary side releases uses expected secondary side activities, based on ANSI/ANS-18.1-l984[14] as modified for WBN, and on a 150 gpd/steam generator primary-to-secondary-side leakage. Credit was taken for flashing of the primary coolant (References [34] and [35] of Section 15.4), but "scrubbing" of the iodine in the rising steam bubbles by the water in the steam generator was conservatively neglected. A partition factor of 100 applies to iodine in the remaining unflashed coolant which will boil.

The atmospheric diffusion coefficients (X/Q) for the exclusion area boundary (EAB) and offsite dose determination are the same as those used for the LOCA analysis (Appendix 15A). The X/Q values for the control room operator were determined in the analysis. The LOCA X/Q values are for a release from the shield building vent, whereas the SGTR release is from the top of the main steam valve vault. The methodology for determination of the WBN control room X/Q values is based on computer code ARCON96.

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are in Table 15.5-19. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

The gamma, beta, and thyroid dose for the SGTR event are in Table 15.5-19. The doses at the EAB and the low population zone are less than 10% of the 10 CFR 100 limits.

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

The analysis of the fuel handling accident considers two cases. The first case is for an accident in the spent fuel pool area located in the Auxiliary Building. This case is evaluated using the Alternate Source Term based on Regulatory Guide 1.183[18], "Alternate Source Term (AST)." The second case considered is an open containment case for an accident inside containment where there is open communication between

the containment and the Auxiliary Building. This evaluation is also based on the AST and Regulatory Guide 1.183. An FHA could occur with the containment closed and the reactor building purge operating. This scenario is bounded by Case 2. The parameters used for this analysis are listed in Table 15.5-20a.

The bases for evaluation consistent with Regulatory Guide 1.183 are:

(1) The accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) Damage was assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) All of the gap activity in the damaged rods is released to the spent fuel pool and consists of 8% I-131, 10% Kr-85, and 5% of other noble gases and other halogens.

(5) Noble gases released to the Auxiliary Building spent fuel pool are released through the Auxiliary Building vent to the environment.

(6) The iodine gap inventory is composed of inorganic species (99.85%) and organic species (0.15%).

(7) The overall inorganic and organic iodine spent fuel pool decontamination factor is 200.

(8) All iodine escaping from the Auxiliary Building spent fuel pool is exhausted unfiltered through the Auxiliary Building vent.

(9) The release path for the containment scenario is changed to include 12.7 seconds of unfiltered release through the Shield Building vent, with the remainder of the unfiltered release through the Auxiliary Building vent.

(10) No credit is taken for the ABGTS or Containment Purge System Filters in the analysis.

(11) No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

(12) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used.

(13) The TEDE values for the Exclusion Area Boundary and Low Population Zone are calculated using dose conversion factors taken from EPA-400-R-92-001, “Manual of Protective Action Guides and Protective Actions of Nuclear Incidents,” May 1992. A breathing rate of 3.33E-4 m^3/sec was used for calculating the TEDE.

(14) The TEDE values for the Main Control Room are calculated using the 100% of the gamma dose calculated using a point kernel integration, 1% of Beta dose, and conversion factors taken from EPA-400-R-92-001, “Manual of Protective Action Guides and Protective Actions of Nuclear Incidents,” May 1992. A breathing rate of 3.33E-4 m^3/sec was used for calculating the TEDE. FSAR Section 15.5.3 provides a discussion of the COROD calculation methods for gamma and beta dose.

Fuel Handling Accident Results

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than for the normal auxiliary building exhaust.

The offsite doses were calculated utilizing FENCDOSE [16], while the control room doses were calculated utilizing the COROD computer code [17].

The TEDE dose is given in Table 15.5-23 for the control room, exclusion area boundary, and low population zone. The dose to control room personnel is less than the limit of 10 CFR 50.67(b)(2)(iii) of 5 rem TEDE, and the dose to the exclusion area boundary and low population zone are less than the limit of 10 CFR 50.67(b)(2)(i) and (ii), as modified by Regulatory Position C.4.4 of Regulatory Guide 1.183 of 6.3 rem TEDE.

15.5.7 Environmental Consequences of a Postulated Rod Ejection Accident

This accident is bounded by the loss-of-coolant accident. See Section 15.5.3 for the loss-of-coolant accident.

REFERENCES

(1) Styrikovich, M. A., Martynova, 0. I., Katkovska, K. YA., Dubrovski, I. YA., Smrinova, I. N., "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," translated from Atomnaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

(2) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure," Division of Reactor Standards, U.S. Atomic Energy Commission, March 23, 1972.

(3) Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Directorate of Regulatory Standards, U.S. Atomic Energy Commission, June 1974.

(4) D. D. Malinowski, "Iodine Removal in the Ice Condenser System," WCAP-7426, April 1970.

(5) NAA-SR 10100, Conventional Buildings for Reactor Containment.

(6) ORNL-NSIC-4, Behavior of Iodine in Reactor Containment Systems, February 1965.

(7) Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

(8) Ramsdell, J. V. Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes." Prepared by Pacific Northwest laboratory fo the U. S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Revision1, May 1997.

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

(10) Deleted by Amendment 80.

(11) Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents At Nuclear Power Reactors, July 2000.

(12) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Directorate of Regulatory Standards, U.S. Atomic Energy Commission, May 1974.

(13) D. B. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1, December 1971.

(14) ANSI/ANS-18.1-1984, "Radioactive Source Terms for Normal Operations of Light Water Reactors," December 31, 1984.

(15) WCAP-7664, Revision 1, "Radiation Analysis Design Manual-4 Loop Plant," RIMS Number NEB 810126 316, October 1972.

(16) Computer Code FENCDOSE, Code I.D. 262358.

(17) Computer Code COROD, Code I.D. 262347.

(18) Regulatory Guide 1.183 R0, Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, US Nuclear Regulatory Commission, July 2000.

(19) Not used

(20) NRC Safety Evaluation for Watts Bar Nuclear Plant Unit 1, Amendment 38, for Steam Generator Tubing Voltage Based Alternate Repair Criteria for Outside Diameter Stress Corrosion Cracking (ODSCC) dated February 26, 2002.

(21) NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", dated August 3, 1995.

(22) TVA Letters to NRC "Technical Specification Change No. WBN-TS-99-014 - Steam Generator Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking (ODSCC)," dated April 10, 2000, September 18, 2000, August 22, 2001, November 8, 2001 and January 15, 2002.

(23) J.J. Dinunno, et, al “Calculation of Distance Factors for Power and Test Reactor Sites”, TID-14844, March 1962.

(24) NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

(25) International Commission on Radiation Protection (ICRP) Publication 30, Limits for Intakes of Radionuclides by Workers," 1979.

Table 15.5-1 Parameters Used In Loss Of A. C. Power Analyses

	Realistic Analysis	Conservative Analysis
Core thermal power	3565 MWt	3565 MWt
Steam generator tube leak rate prior to and during accident	1 gpm	1.0 gpm
Fuel defects	ANSI/ANS 18.1 - 1984	Technical Specification limit of 0.1 µCi/gm I-131 dose equivalent
Iodine partition factor in steam generator prior to and during accident	0.01	0.01
Blowdown rate per steam generator prior to accident	25 gpm	25 gpm
Duration of plant cooldown by secondary system after accident	8 hrs	8 hrs
Steam release from 4 steam generators	444,875 lbm (0-2 hrs) 903,530 lbm (2-8 hrs)	444,875 lbs (0-2 hrs) 903,530 lbs (2-8 hrs)
Meteorology	See Tables 15A-2 & 15.5-14	See Tables 15A-2 & 15.5-14

Table 15.5-2 Doses From Loss Of A/C Power

Conservative Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	7.45E-04	4.18E-04	2.11E-04
Beta	4.53E-04	2.54E-04	2.58E-03
Thyroid - ICRP-30	4.58E-02	2.57E-02	2.23E-02

Realistic Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	1.80E-08	1.01E-08	5.09E-09
Beta	2.09E-05	1.17E-05	2.26E-04
Thyroid - ICRP-30	1.10E-06	6.18E-07	5.37E-07

Table 15.5-3 Parameters Used In Waste Gas Decay Tank Rupture Analyses

	Realistic Analysis	Regulatory Guide 1.24 Analysis
Core thermal power	3565 MWt	3565 MWt
Plant load factor	1.0	1.0
Fuel defects	ANSI/ANS-18.1, 1984	1%
Activity released from GWPS	(1)	See Table 15.5-4
Time of accident	After Tank Fill	At end of equilibrium core cycle
Meteorology	See Table 15.5-14 and Table 15A-2	See Table 15.5-14 and Table 15A-2

(1)Activity based on maximum concentrations of each isotope and actual plant flow rates of the GWPS.

Table 15.5-4 Waste Gas Decay Tank Inventory (One Unit) (Regulatory Guide 1.24 Analysis)

Isotope	Activity (Curies)
Xe-131m	8.9×10^{2}
Xe-133	6.8×10^{4}
Xe-133m	1.0×10^{3}
Xe-135	9.4×10^{2}
Xe-135m	4.8×10^{1}
Xe-137	2.7×10^{-1}
Xe-138	3.2
Kr-83m	1.7×10^{1}
Kr-85	4.2×10^{3}
Kr-85m	1.3×10^{2}
Kr-87	2.9×10^{1}
Kr-88	1.6×10^{2}
Kr-89	1.0×10^{-1}
I-131	4.8×10^{-2}
I-132	---------
I-133	3.3×10^{-2}
I-134	---------
I-135	1.2×10^{-2}

Table 15.5-5 Doses From Gas Decay Tank Rupture

Regulatory Guide 1.24 Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	5.96E-01	1.67E-01	9.44E-01
Beta	1.61E+00	4.51E-01	8.15E+00
Thyroid - ICRP-30	1.29E-02	3.60E-03	1.08E-02

Realistic Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	2.88E-02	8.05E-03	4.27E-02
Beta	1.10E-01	3.08E-02	5.62E-01
Thyroid - ICRP-30	1.21E-02	3.37E-03	1.00E-02

Table 15.5-6 Parameters Used In LOCA Analysis (Page 1 of 2)

	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt
Primary containment free volume	1.27×10^6 ft^3
Annulus free volume	3.75×10^5 ft^3
Primary containment deck (air return) fan flow rate	40,000 cfm
Number of deck (containment air return fans) fans assumed operating	1 of 2
Activity released to primary containment and available for release	
noble gases	100% of core inventory
iodines	25% of core inventory
Form of iodine activity in primary containment available for release	
elemental iodine	91%
methyl iodine	4%
particulate iodine	5%
Ice condenser removal efficiency for elemental and particulate iodine	See Table 15.5-7
Primary containment leak rate (volume percent)	0.25% per day (0-24 hours) 0.125% per day (1-30 days)
Percent of primary containment leakage to auxiliary building	25%
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Delay time of activity in auxiliary building before ABGTS operation	None
Delay time before filtration credit is taken for the ABGTS	4 min
Mean holdup time in auxiliary building after initial 4 minutes	0.3 hours
ABGTS flow rate	9000 cfm

Table 15.5-6 Parameters Used In LOCA Analysis (Page 2 of 2)

Leakage from Auxiliary Building to ABGTS downstream HVAC (bypass of filters)	27.88 cfm
Leakage from ABGTS HVAC into Auxiliary Building	8.87 cfm
Leakage from Auxiliary Building into EGTS downstream HVAC (bypass of filters)	10.7 cfm
Leakage from Auxiliary Building to environment due to single failure of ABGTS (from 30 minutes to 34 minutes post-LOCA)	9900 cfm (for 4 minutes)
Percent of primary containment leakage to annulus	75%
Emergency gas treatment system flow rates	See Table 15.5-8 and Table 15.5-8A
Percent of annulus free volume available for mixing of recirculated activity	50%
Number of emergency gas treatment system air handling units operating	1 of 2
Emergency gas treatment system filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Shield building mixing model (see Section 15.5.3)	50% mixing
Meteorology	See Table 15.5-14 and Table 15A-2

Table 15.5-7 Ice Condenser Elemental And Particulate Iodine Removal Efficiency(1,2)

Time Interval Post LOCA (Hours)	Iodine Removal Efficiency
0.0 to 0.156	0.96
0.156 to 0.267	0.76
0.267 to 0.323	0.73
0.323 to 0.489	0.71
0.489 to 0.615	0.60
0.615 to 0.768	0.58
0.768 to 0.824	0.40
0.824 to 720	0.0

(1) The ice condenser removal efficiencies given in the above table are used for the Regulatory Guide 1.4 analysis. The inlet steam/air mixture coming into the ice condenser is greater than 90% steam by volume initially due to the delaying of the operation of the containment deck fans. Without the delay of operation of the deck fans, the amount of steam by volume in the inlet mixture initially would be much lower and the ice condenser iodine removal efficiencies would be reduced.

(2) The ice bed iodine removal efficiency, O_I, has been computed on a time dependent basis and is shown in Table 15.5-7. Note that the information presented in Table 15.5-7 has been revised by Westinghouse letter WAT-D-10954. The revised efficiency information is associated with the WCAP-15699, Revision 1 analysis for reduced ice weight. A comparison of the information presented in Table 15.5-7 and the revised information contained in WAT-D-10954 shows that the information in Table 15.5-7 is conservative. Analyses supporting the plant design basis acknowledge the revised efficiency information but shall utilize the information presented in Table.15.5-7.

Table 15.5-8 EMERGENCY GAS TREATMENT SYSTEM FLOW RATES (Sheet 1 of 1)

Time Interval	Time Interval	Recirculation Rate		Exhaust Rate	
(sec)	(hours)	(cfm)	(cfh)	(cfm)	(cfh)
0-30	0-0.0083	0.00	0.00E+00	0.00	0.00E+00
30-39	0.0083-0.0108	3600.00	2.16E+05	0.00	0.00E+00
39-40	0.0108-0.0111	3286.62	1.97E+05	313.38	1.88E+04
40-41	0.0111-0.0114	2352.31	1.41E+05	1247.69	7.49E+04
41-42	0.0114-0.0117	1304.79	7.83E+04	2295.21	1.38E+05
42-43	0.0117-0.0119	362.60	2.18E+04	3237.40	1.94E+05
43-190	0.0119-0.0528	0.00	0.00E+00	3600.00	2.16E+05
190-191	0.0528-0.0531	537.28	3.22E+04	3062.72	1.84E+05
191-192	0.0531-0.0533	733.23	4.40E+04	2866.77	1.72E+05
192-193	0.0533-0.0536	735.14	4.41E+04	2864.86	1.72E+05
193-194	0.0536-0.0539	737.51	4.43E+04	2862.49	1.72E+05
194-199	0.0539-0.0553	745.23	4.47E+04	2854.77	1.71E+05
199-207	0.0553-0.0575	764.12	4.58E+04	2835.89	1.70E+05
207-215	0.0575-0.0597	790.80	4.74E+04	2809.20	1.69E+05
215-225	0.0597-0.0625	825.45	4.95E+04	2774.56	1.66E+05
225-245	0.0625-0.0681	892.72	5.36E+04	2707.29	1.62E+05
245-265	0.0681-0.0736	992.80	5.96E+04	2607.20	1.56E+05
265-285	0.0736-0.0792	1102.40	6.61E+04	2497.61	1.50E+05
285-305	0.0792-0.0847	1217.05	7.30E+04	2382.95	1.43E+05
305-446	0.0847-0.1239	1664.05	9.98E+04	1935.96	1.16E+05
446-601	0.1239-0.1669	2356.72	1.41E+05	1243.29	7.46E+04
601-602	0.1669-0.1672	2661.35	1.60E+05	938.65	5.63E+04
602-1700	0.1672-0.4722	3600.00	2.16E+05	0.00	0.00E+00
1700-1701	0.4722-0.4725	3508.13	2.10E+05	91.87	5.51E+03
1701-1702	0.4725-0.4728	3423.44	2.05E+05	176.56	1.06E+04
1702-1703	0.4728-0.4731	3410.73	2.05E+05	189.27	1.14E+04
1703-1704	0.4731-0.4733	3408.66	2.05E+05	191.34	1.15E+04
1704-1705	0.4733-0.4736	3408.17	2.04E+05	191.83	1.15E+04
1705-1706	0.4736-0.4739	3407.91	2.04E+05	192.09	1.15E+04
1706-1855	0.4739-0.5153	3395.23	2.04E+05	204.77	1.23E+04
1855-2100	0.5153-0.5833	3372.37	2.02E+05	227.64	1.37E+04
2100-30 days*	0.5833-720	3350.00	2.01E+05	250.00	1.50E+04

*Required to maintain annulus pressure when assuming 250 cfm annulus inleakage

Table 15.5-8A Emergency Gas Treatment System Flow Rates (Unit 2)

Time Interval		Time Interval		Recirculation Rate		Exhaust Rate	
(sec)	(sec)	(hours)	(hours)	(cfm)	(cfh)	(cfm)	(cfh)
0	30	0	0.0083	0	0.00E+00	0	0.00E+00
30	39	0.0083	0.0108	7200	4.32E+05	0	0.00E+00
39	40	0.0108	0.0111	6573.24	3.94E+05	626.76	3.76E+04
40	41	0.0111	0.0114	4704.62	2.82E+05	2495.38	1.50E+05
41	42	0.0114	0.0117	2609.58	1.57E+05	4590.42	2.75E+05
42	43	0.0117	0.0119	725.2	4.35E+04	6474.8	3.88E+05
43	71	0.0119	0.0197	0	0.00E+00	7200	4.32E+05
71	78	0.0197	0.0217	0	0.00E+00	7200	4.32E+05
78	79	0.0217	0.0219	1062	6.37E+04	6138	3.68E+05
79	80	0.0219	0.0222	4775	2.87E+05	2425	1.46E+05
80	102	0.0222	0.0283	4337	2.60E+05	2863	1.72E+05
102	132	0.0283	0.0367	4188	2.51E+05	3012	1.81E+05
132	165	0.0367	0.0458	3922	2.35E+05	3278	1.97E+05
165	170	0.0458	0.0472	3762	2.26E+05	3438	2.06E+05
170	210	0.0472	0.0583	3719	2.23E+05	3481	2.09E+05
210	307	0.0583	0.0853	3760	2.26E+05	3440	2.06E+05
307	498	0.0853	0.1383	4050	2.43E+05	3150	1.89E+05
498	602	0.1383	0.1672	4797	2.88E+05	2403	1.44E+05
602	603	0.1672	0.1675	5232	3.14E+05	1968	1.18E+05
603	850	0.1675	0.2361	5137	3.08E+05	1432	8.59E+04
850	1100	0.2361	0.3056	5237	3.14E+05	1332	7.99E+04
1100	1350	0.3056	0.3750	5337	3.20E+05	1232	7.39E+04
1350	1600	0.3750	0.4444	5437	3.26E+05	1132	6.79E+04
1600	1850	0.4444	0.5139	5537	3.32E+05	1032	6.19E+04
1850	2100	0.5139	0.5833	5637	3.38E+05	932	5.59E+04
2100	3600*	0.5833	1.0000	5737	3.44E+05	832	4.99E+04
3600*	30 days	1.0000	30 days	3455	2.07E+05	604	3.62E+04

*Reflects operator action to place one EGTS fan in standby mode at 1 hour.

Table 15.5-9 DOSES FROM LOSS-OF-COOLANT ACCIDENT

(rem)	2Hr EAB	30 Day LPZ	Control Room
Gamma	2.12	2.18	1.05
Beta	1.25	2.61	9.10
Thyroid - ICRP - 30	40.4	14.33	3.75

Breakdown of Control Room Personnel Dose

(rem)	Airborne	Shine	Ingress/Egress	Total
Gamma	1.02	0.005	0.027	1.05
Beta	9.04	0.000	0.060	9.10
Thyroid - ICRP - 30	3.66	0.000	0.090	3.75

Table 15.5-10 Deleted by Amendment 80

Table 15.5-11 Deleted by Amendment 80

Table 15.5-12 PARAMETERS USED IN ANALYSIS OF RECIRCULATION LOOP LEAKAGE FOLLOWING A LOCA

	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt
Recirculation sump water volume	9.63×10^4 ft^3
Activity mixed with recirculation loop water	
Noble gases	0.0
Iodines	50% of core inventory
Tritium	97% to sump (water)
Leakage of ECCS equipment outside containment	See Table 6.3-6
Iodine partition factor for leakage	0.1
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Meteorology	See Table 15.5-14 and Table 15A-2

Table 15.5-13 Doses From Recirculation Loop Leakage Following A LOCA

(rem)	2HR EAB	30 Day LPZ	Control Room
Gamma	4.14E-03	2.28E-02	1.51E-03
Beta	1.36E-03	8.54E-02	1.62E-02
Thyroid - ICRP - 30	1.40E-03	1.53E-01	3.69E-02

Table 15.5-14 Atmospheric Dilution Factors At The Control Building

DILUTION FACTOR (sec/m^3)			
Time Period (hr)	**LOCA/FHA**	**SGTR/MSLB/Loss of AC Power**	**WGDT**
0-2	1.09E-03	2.59E-03	2.56E-03
2-8	9.44E-04	2.12E-03	N/A
8-24	1.56E-04*	N/A	N/A
24-96	1.16E-04**	N/A	N/A
96-720	9.59E-05***	N/A	N/A

GENERAL CONTROL ROOM PARAMETERS

Volume	257,198 cu ft
Makeup/pressurization flow	711 cfm
Recirculation flow	2889 cfm
Unfiltered intake	51 cfm
Filter efficiency	95% first pass
	70% second pass
	0% for noble gases, Tritium
Isolation time, T	74 seconds
Occupancy factors:	
0-24 hr	100%
1-4 days	60%
4-30 days	40%

1. All FHA releases are within 2 hours. Thus, only the 0-2 hr X/Q is applicable for the FHA.

* Calculated value for U1 Shield Bldg Vent to East MCR Intake	1.26E-04
** Calculated value for U1 Shield Bldg Vent to East MCR Intake	9.53E-05
*** Calculated value for U1 Shield Bldg Vent to East MCR Intake	8.07E-05

Table 15.5-15 Deleted by Amendment 97

Table 15.5-16 Parameters Used In Steam Line Break Analysis

	Analysis Value
Steam Generator tube leak rate	
Faulted Steam Generator	1.0 gpm
Per Intact Steam Generator	150 gpd
Iodine Partition Factor	
Faulted Steam Generator	1
Intact Steam Generator	100
RCS Letdown flow rate	124.39 gpm
Steam Releases	
Faulted Steam Generator (0-30 minutes)	96,100 lbm
Three Intact Steam Generators (0-2 hrs)	433,079 lbm
Three Intact Steam Generators (2-8 hrs)	870,754 lbm

Table 15.5-17 Doses From Main Steam Line Break

1 gpm Primary-to-Secondary Leakage (ARCON-96 x/Q)	2 HR EAB	30-Day LPZ	SRP Guidance for 10CFR100 Limits (rem)	Control Room	SRP Guidance for 10CFR100 Limits (rem)
Pre-Accident Initiated Spike Case (14 µCi/gm maximum peak)					
Gamma	2.74E-02	1.11E-02	25	4.35E-03	5
Beta	8.81E-03	4.21E-03	300	4.00E-02	30
Thyroid - ICRP-30	2.41E+00	1.21E+00	300	7.44E+00	30
Accident Initiated Spike Case (0.265 µCi/gm steady state)					
Gamma	1.04E-01	1.25E-01	2.5	8.02E-03	5
Beta	2.54E-02	3.02E-02	30	6.48E-02	300
Thyroid - ICRP-30	3.09E+00	4.78E+00	30	1.03E+01	300

Table 15.5-18 Parameters Used In Steam Generator Tube Rupture Analysis

Primary Side Activity	Technical Specification Limit
Secondary Side Activity	ANSI/ANS-18.1-1984 (Expected levels, 150 gpd/SG)
Iodine Spiking Factor	Case 1: Accident initiated spike of 500 times equilibrium iodine concentration Case 2: Pre-accident spike of 14 µCi/gm I-131 dose equivalent
Iodine Partition Factor	100
Secondary Side Mass Release (Ruptured Steam Generator) 0 -2 hours 2 - 8 hours	 103,300 lbm 32,800 lbm
Secondary Side Mass Release (Intact Steam Generator) 0 - 2 hours 2 - 8 hours	 492,100 lbm 900,200 lbm
Primary Coolant Mass Release (Total) 0 - 2 hours	 191,400 lbm
Primary Coolant Mass Release (Flashed) 0 - 2 hours	 10,077.2 lbm
Meteorology	See Table 15A-2 and 15.5-14

Table 15.5-19 Doses From Steam Generator Tube Rupture

Pre-Accident Initiated Spike Case (14 µCi/gm maximum peak)			
(rem)	**2 HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	3.87E-01	1.14E-01	6.27E-02
Beta	2.28E-01	6.99E-02	7.08E-01
Thyroid - ICRP-30	1.45E+01	4.15E+00	1.32E+01
Accident Initiated Iodine Spike Case (0.265 µCi/gm steady state)			
(rem)	**2 HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	5.76E-01	1.69E-01	6.05E-02
Beta	2.66E-01	8.19E-02	7.06E-01
Thyroid - ICRP-30	7.56E+00	2.23E+00	2.20E+00

Table 15.5-20 Deleted by Amendment 111

Table 15.5-20a Parameters Used In Fuel Handling Accident Analysis

Regulatory Guide 1.183 Analysis	
Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released to spent fuel pool	
elemental iodine	99.85%(AST)
methyl iodine	0.15%(AST)
Decontamination factor in spent fuel pool	AST Overall=200
Filter efficiencies	No credit taken
Amount of mixing of activity in Auxiliary Building	None
Meteorology	See Table 15.5-14 and Table15A-2

(1) 8% I-131, 10% Kr-85, and 5% other gasses and other halogens.

Table 15.5-21 Nuclear Characteristics Of Highest Rated Discharged Assembly Used In The Analysis

Core thermal power	3565 MWt
Number of assemblies	193
Fuel rods per assembly	264
Core average assembly power	18.47 MWt
Discharged Assembly	
Radial peak to average ratio	1.65

Table 15.5-22 Deleted by Amendment 80

Table 15.5-23
Doses From A Fuel Handling Accident (FHA) (rem)

Doses from Fuel Handling Accident Regulatory Guide 1.183 Analyses

FHA in Auxiliary Building (rem)

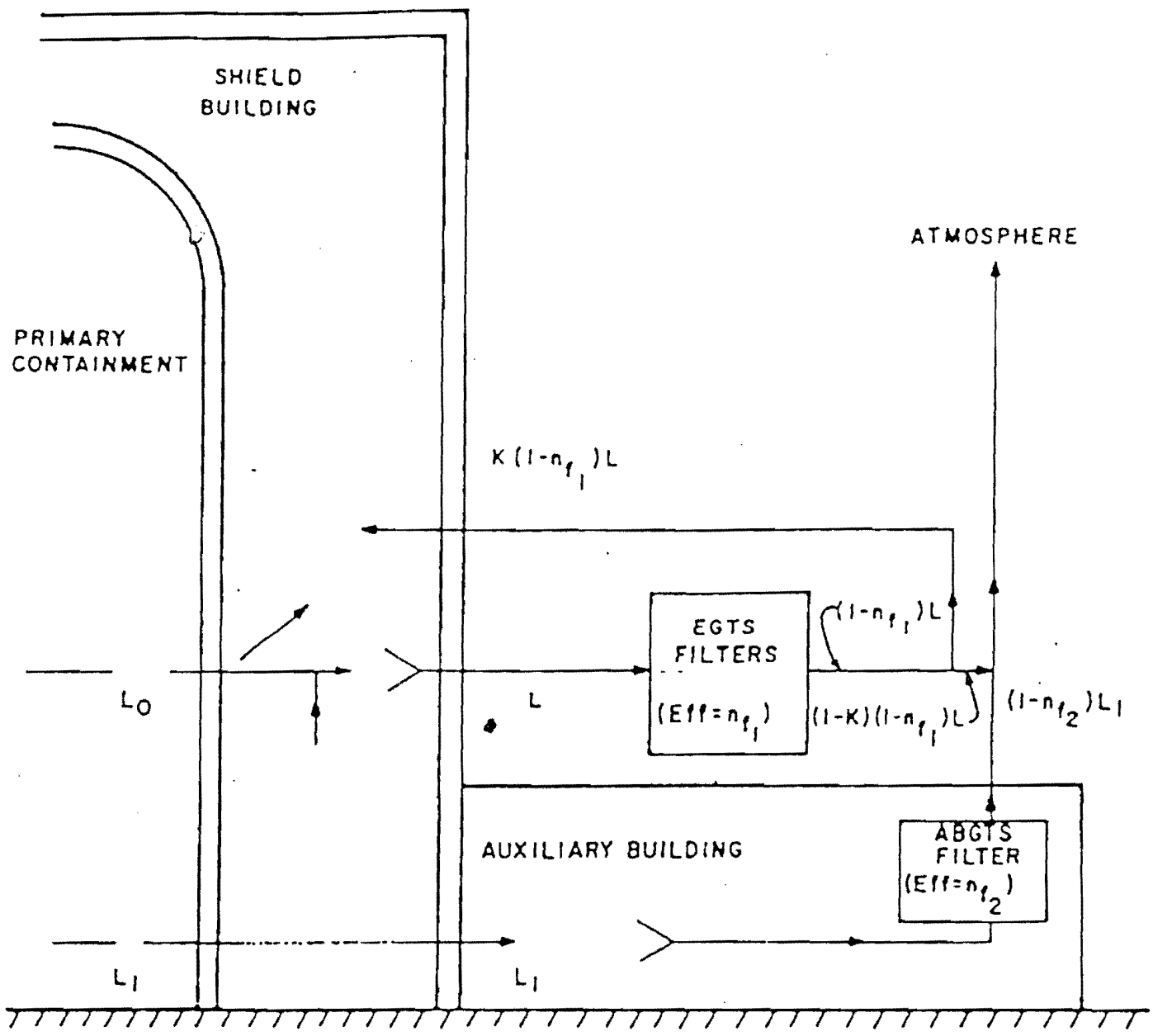
	2 HR EAB	30 DAY LPZ	CONTROL ROOM
TEDE	2.38E+00	6.66E-01	1.02E-00

FHA in Containment - Containment Open (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
TEDE	2.38E+00	6.66E-01	1.01E-00

Table 15.5-24 Deleted by Amendment 80

Table 15.5-25 Deleted by Amendment 80



NOTE:
MINOR BYPASS LEAKAGE PATHS ARE NOT SHOWN.

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SCHEMATIC OF LEAKAGE PATH

FIGURE 15.5-1

Figure 15.5-1 Schematic of Leakage Path

Figure 15.5-2 Deleted by Amendment 80

Figure 15.5-3 Deleted by Amendment 97

Figure 15.5-4 Deleted by Amendment 97

15A DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15A.1 INTRODUCTION

This Appendix identifies the models used to calculate the offsite radiological doses that would result from releases of radioactivity due to various postulated accidents. The postulated accidents are:

(1) Waste Gas Decay Tank Rupture

(2) Steam Generator Tube Rupture

(3) Steam Line Break

(4) Loss of A. C. Power

(5) Loss of Coolant Accident

15A.2 ASSUMPTIONS

The following assumptions are basic to both the model for the gamma and beta doses due to immersion in a cloud of radioactivity and the model for the thyroid dose due to inhalation of radioactivity.

(1) Direct radiation from the source point is negligible compared to gamma and beta radiation due to submersion in the radioactivity leakage cloud.

(2) All radioactivity releases are from the appropriate point of discharge.

(3) The dose receptor is a standard man as defined by the International Commission on Radiological Protection (ICRP).[1]

(4) Radioactive decay from the point of release to the dose receptor is neglected.

(5) Isotopic data such as decay rates and decay energy emissions are taken from Table of Isotopes.[2]

15A.3 GAMMA DOSE AND BETA DOSE

The gamma and beta dose delivered to a dose receptor is obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane, i.e., an "infinite semispherical cloud." The concentration of radioactive material within this cloud is taken to be uniform and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release.

The beta dose is a result of external beta radiation and the gamma dose is a result of external gamma radiation. Equations describing an infinite semispherical cloud were used to calculate the doses for a given time period as follows:[5]

$$\text{Beta Dose} = 0.23 \cdot (X/Q)_t \cdot \sum_i A_{R_i} \cdot \bar{E}_{\beta_i}$$

and

$$\text{Gamma Dose} = 0.25 \cdot (X/Q)_t \cdot \sum_i A_{R_i} \cdot \bar{E}_{\gamma_i}$$

where:

A_{Ri}=activity of isotope i released during a given time period, curies

$(X/Q)_t$=atmospheric dilution factor for a given time interval t, sec/m^3

$\bar{E}_{\beta i}$=average beta radiation energy emitted by isotope i per disintegration, mev/dis

$\bar{E}_{\gamma i}$=average gamma radiation energy omitted by isotope i per disintegration, mev/dis

15A.4 THYROID INHALATION DOSE

The thyroid dose for a given time period t, is obtained from the following expression[6]:

$$D = (X/Q)_t \cdot B \cdot \sum_i Q_i \cdot DCF_i$$

where:

D = thyroid inhalation dose, rem

$(X/Q)_t$ = site dispersion factor for time interval t, sec/m^3

B = Breathing rate for time interval t, m^3/sec

Q_i = total activity of iodine isotope i released in time period t, curies

$(DCF)_i$ = dose conversion factor for iodine isotope i, rem/curies inhaled

The isotopic data and "standard man" data are given in Table 15A-1. The atmospheric dilution factors used in the analysis of the environmental consequences of accidents are given in Chapter 2 of this report and are reiterated in Table 15A-2 of this appendix.

The gamma energies, Eγ, on Table 15A-1 include the X-rays and annihilation gamma rays if they are prominent in the electromagnetic spectrum. Also the beta energies E_4, include conversion electrons if they are prominent in the electromagnetic spectrum. The beta energies are averaged quantities in the sense that the continuous beta spectra energies are computed as one-third the maximum beta energies.

REFERENCES

(1) "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics, Vol. 3, pp. 30, 146-153, 1970.

(2) Leaderer, C. M., et. al., Table of Isotopes, 6th edition, 1968.

(3) Nuclear Data Sheets, Oak Ridge National Laboratory (ORNL) Nuclear Data Group, Vol. 7, Number 1, Academic Press, New York, January 1972.

(4) Radioactive Atoms - Supplement 1, ORNL-4923, Martin, M. J., NTIS, November 1973.

(5) Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," USAEC, June 1974.

(6) J. J. Dinunno, et. al, "Calculation of Distance Factors for Power and Test Reactor Sites", TID 14844, March 1962.

Table 15A-1 Physical Data For Isotopes

Isotope	Decay Constant** (Hr^{-1})	Gamma Energy** (Mev/Disint.)	Beta Energy** (Mev/Disint.)	Dose Conversion Factor* (Rem/Curie)
I-131	3.5833×10^{-3}	0.3810	0.1943	1.48×10^{6}
I-132	3.0401×10^{-1}	2.3332	0.5143	5.35×10^{4}
I-133	3.332×10^{-2}	0.6100	0.4090	4.00×10^{5}
I-134	7.9067×10^{-1}	2.5928	0.6102	2.50×10^{4}
I-135	1.0486×10^{-1}	1.5802	0.3680	1.24×10^{5}
Xe-131m	2.4269×10^{-3}	0.0201	0.1428	
Xe-133	5.4594×10^{-3}	0.0454	0.154	-
Xe-133m	1.2836×10^{-2}	0.0416	0.1898	-
Xe-135	7.5755×10^{-2}	0.3470	0.3168	-
Xe-135m	2.6574×10^{0}	0.4318	0.0950	-
Xe-138	2.9350×10^{0}	1.1830	0.6058	-
Kr-83m	3.7267×10^{-1}	0.0025	0.0371	
Kr-85	7.3692×10^{-6}	0.0022	0.2506	-
Kr-85m	1.5472×10^{-1}	0.1586	0.2529	-
Kr-87	5.4508×10^{-1}	0.7928	1.3237	
Kr-88	2.4755×10^{-1}	1.9629	0.3750	
Kr-89	1.3078×10^{-1}	2.0837	1.2310	

BREATHING RATES

Time Period (Hours)	Breathing Rates (M^3/Sec)
0 - 8	3.47×10^{-4}
8 - 24	1.75×10^{-4}
24 - 720	2.32×10^{-4}

* Refer to Reference [6]**

** Refer to Reference [2], [3], [4]

Table 15A-2 Accident Atmospheric Dilution Factors (sec/m^3)

Conservative And Regulatory Guide Analyses		
Time Period (hours)	**Exclusion Area Boundary***	**Low Population Zone (4828 meters)**
0-2	6.382E-04	1.784E-04
2-8		8.835E-05
8-24		6.217E-05
24-96		2.900E-05
96-720		9.811E-06

* The dilution factors were calculated for a travel distance of 1100 meters, the distance from the 100 meter radius release zone to the 1200 meter radius exclusion boundary (See Section 2.3.4).

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