Westinghouse Non-Proprietary Class 3

WCAP-15699
REVISION 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT UNIT 1
CONTAINMENT INTEGRITY ANALYSES
FOR ICE WEIGHT OPTIMIZATION
ENGINEERING REPORT

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CONTAINMENT INTEGRITY ANALYSES FOR ICE WEIGHT OPTIMIZATION
ENGINEERING REPORT

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EXECUTIVE SUMMARY

Loss-of-Coolant Long-term Containment Mass and Energy Release and Containment Integrity Analyses have been performed to support ice weight optimization at the Watts Bar Nuclear Plant Unit 1. The objective of this effort was to provide revised containment mass and energy release data using current Watts Bar specific information and more realistic models to support ice weight reduction. The analyses conducted used the WCAP-10325-P-A mass and energy release model, which is a first time application to Watts Bar but has previously used on many other Westinghouse design PWRs including Sequoyah. The containment pressure calculation is consistent with current licensed methodology.

The analyses include LOCA long-term mass and energy releases to be used to support the analytical basis and subsequently used in the LOTIC-1 Computer Code in the containment integrity response analyses.

The objective of this effort was to obtain ice weight optimization, retain current time interval (approximately 150 seconds) relationship between containment spray switchover time and ice bed melt-out and provide for peak pressure margin to design pressure.

The results of the analysis support the following:

- An ice mass of 2.029375×10^6 lbms

- A calculated containment peak pressure of 10.438 psig occurring at 6,373.5 seconds

- Ice bed meltout occurred at 3625.5 seconds

  (Containment spray switchover is completed at 3447 seconds thus the containment spray switchover ice bed meltout relationship is 178.5 seconds.)

- Ice Bed Mass limited by the Spray Switchover time of 3447 seconds and the margin between spray swithcover and ice bed meltout of at least 150 seconds. Thus, the containment pressure margin does not translate into a further reduction in ice bed mass.

- The ice bed mass of 2.029375×10^6 Lbms equates to an average of 1044 Lbm per basket. This average value recognizes that all baskets may not have the same initial weight nor have the same sublimation rate. To ensure that a sufficient quantity of ice exists in each basket to survive the blowdown phase of a LOCA, a minimum amount of ice per basket to survive the blowdown would be approximately 313 Lbm, based on Table 3-4. To ensure that an adequate distribution of ice exists in the Ice Condenser to prevent early burn -through of a localized area, 313 Lbm of ice should be the minimum weight of ice per basket at any time while also ensuring that the average weight per basket remains above 1044 Lbm.

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1.0 INTRODUCTION

A Containment Integrity analysis was performed to support ice weight optimization. The analysis effort was similar to the Watts Bar design basis containment integrity analysis currently documented in the Watts Bar Nuclear Plant UFSAR Chapter 6.2.

A Containment Integrity Analysis is performed during nuclear plant design to ensure that the pressure inside containment will remain below the containment building design pressure if a Loss-of-Coolant Accident (LOCA) inside containment should occur during plant operation. The analysis ensures that the containment heat removal capability is sufficient to remove the maximum possible discharge of mass and energy to containment from the Nuclear Steam Supply System without exceeding the acceptance criteria (13.5 psig).

This analysis utilized revised input assumptions, which eliminated analytical conservatisms from the present analysis. Several areas addressed were the assumed core stored energy, decay heat release, steam generator secondary side metal heat and ice condenser metal mass. The analysis was completed to provide the analytical basis for a reduction in the present Watts Bar design basis ice mass of 2.125 million pounds with minimal impact on current margins in peak calculated containment pressure and ice bed meltout time to containment spray switchover time.

In addition to the design basis, this analysis accounted for the effects of other plant changes that Westinghouse is aware of. These include revised minimum safety injection flows (References 12 & 13), initial condition uncertainties on RCS temperature of +6.0/-5.0°F, RCS (pressurizer)(Ref 11) pressure uncertainty of +70/-50 psia (Ref 11), and 17x17 V5H (Ref 11) fuel. It should be noted that these items were included for completeness even though any or all of the items may not currently be implemented at the Watts Bar Nuclear Plant Unit 1.

1.1 PURPOSE OF ANALYSIS

The purpose of this program was to calculate the long-term Loss of Coolant Accident (LOCA) mass and energy releases and the subsequent containment integrity response in order to demonstrate support for ice weight optimization and increased operating margins. This effort will address current Watts Bar specific plant conditions and revised models as a means of using available analytical margins to support a reduction in the amount of ice required in the ice condenser. The objective of this effort in conducting the ice weight reduction from the current design basis 2.125 million pounds will be to maintain the current time interval (150 seconds, minimum) relationship between containment spray switchover time and ice bed meltout time, and to provide peak pressure margin to design pressure.

A key element in obtaining ice mass reduction will be reducing the energy available to containment in the event of a LOCA. Areas such as core stored energy, decay heat, and available steam generator metal heat were investigated and available margins were implemented into the analysis. These margins combined with a better segmental representation of the mass and energy release transient from the computer models result in margins which reduce energy input into containment.

This program will provide the analytical basis and the results, which show that the containment design pressure is not exceeded in the event of a LOCA. The conclusions presented will

demonstrate, with respect to LOCA, that containment integrity has not been compromised. Further, since the LOCA requires the greatest amount of ice compared to other accident scenarios, the reduction in initial ice mass based on the LOCA will be acceptable.

Rupture of any of the piping carrying pressurized high temperature reactor coolant, termed a LOCA, will result in release of steam and water into the containment. This, in turn, will result in an increase in the containment pressure and temperature. The mass and energy release rates described in this document form the basis of further computations to evaluate the structural integrity of the containment following a postulated accident to satisfy the Nuclear Regulatory acceptance criteria, General Design Criterion 38. Section 2.0 presents the long-term mass and energy release analysis for containment pressurization evaluations. Section 3.0 presents the Containment Pressure Calculations.

1.2 SYSTEM CHARACTERISTICS AND MODELING ASSUMPTIONS

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Some of the most critical items are the: RCS initial conditions, core decay heat, safety injection flow, and metal and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed below. Tables 2-1 through 2-3 present key data assumed in the analysis. The data provided in Reference 11 was used, in part, to develop the plant data presented in Tables 2-1 through 2-3.

For the long-term mass and energy release calculations, operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The modeled core rated power of 3459 MWt adjusted for calorimetric error (+0.6 percent of power) was the basis in the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures, which are at the maximum levels attained in steady state operation. Additionally, an allowance of +6.0 °F is reflected in the vessel/core temperature in order to account for instrument error and deadband. The initial reactor coolant system (RCS) pressure in this analysis is based on a nominal value of 2250 psia. Also included is an allowance of +70 psi, which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2320 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS depressurizes is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2320 psia initial pressure was selected as the limiting case for the long-term mass and energy release calculations. These assumptions conservatively maximize the mass and energy in the RCS.

The selection of the fuel design features for the long-term mass and energy calculation is based on the need to conservatively maximize the core stored energy. The fuel conditions were adjusted to provide a bounding analysis for current Watts Bar Nuclear Plant Unit 1 fuel features. The following items serve as the basis to ensure conservatism in the core stored energy calculation: a conservatively high reload core loading; time of maximum fuel densification, i.e., highest BOL temperatures; and irradiated fuel assemblies are assumed to have an average burnup $\geq$15000

MWD/MTU.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled.

Regarding safety injection flow, the mass and energy calculation considered the historically limiting configuration of minimum safety injection flow.

The following summarizes the assumptions that were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the reactor coolant system (100% full power conditions).

2. An allowance in temperature for instrument error and dead band was assumed on the vessel/core inlet temperature (+6.0 degrees F).

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty).

4. Core rated power of 3459 MWt.

5. Allowance for calorimetric error (+0.6 percent of power).

6. Conservative coefficient of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer).

7. Core stored energy based on the time in life for maximum fuel densification. The assumptions used to calculate the fuel temperatures for the core stored energy calculation account for appropriate uncertainties associated with the models in the PAD code (e.g., calibration of the thermal model, pellet densification model, cladding creep model, etc.). In addition, the fuel temperatures for the core stored energy calculation account for appropriate uncertainties associated with manufacturing tolerances (e.g., pellet as-built density). The total uncertainty for the fuel temperature calculation is a statistical combination of these effects and is dependent upon fuel type, power level, and burnup.

8. An allowance for RCS initial pressure uncertainty (+70 psi).

9. A maximum containment backpressure equal to design pressure.

10. The steam generator metal mass was modeled to include only the portion of the steam generators (SG) which is in contact with the fluid on the secondary side. Portions of the SGs such as the elliptical head, upper shell and misc. internals have poor heat transfer due to location. The heat stored in these areas available for release to containment will not be able to effectively transfer energy to the RCS, thus the energy will be removed at a much slower rate and time period (>10000 seconds).

11. A provision for modeling steam flow in the secondary side through the steam generator turbine stop valve was conservatively addressed only at the start of the event. Turbine stop valve isolation time equal to 0.0 seconds was considered.

12. As noted in Section 2.4 of Reference 1, the option to provide more specific modeling pertaining to decay heat has been exercised to specifically reflect the Watts Bar Nuclear Plant Unit 1 core heat generation, while retaining the two sigma uncertainty to assure conservatism.

13. Steam generator tube plugging leveling (0% uniform)

    - Maximizes reactor coolant volume and fluid release
    - Maximizes heat transfer area across the SG tubes
    - Reduces coolant loop resistance, which reduces the Δp upstream of break and increases break flow

Thus, based on the previously noted conditions and assumptions, a bounding analysis of Watts Bar Nuclear Plant Unit 1 is made for the release of mass and energy from the RCS in the event of a LOCA to support ice weight optimization.

2.0 LONG-TERM LOCA MASS AND ENERGY RELEASE ANALYSIS

2.1 INTRODUCTION

The evaluation model used for the long-term LOCA mass and energy release calculations was the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other ice condenser plants.

This report section presents the long-term LOCA mass and energy releases that were generated in support of the Watts Bar Nuclear Plant Unit 1 ice weight optimization program. These mass and energy releases are then subsequently used in the LOTIC-1 computer code for containment integrity analysis peak pressure calculations.

2.2 LOCA MASS AND ENERGY RELEASE PHASES

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases:

1. Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state at containment design pressure.

2. Refill - the period of time when the reactor vessel lower plenum is being filled by accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively

consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the reactor vessel lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

2.2.1 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, and FROTH. These codes were used to calculate the long-term LOCA mass and energy releases for the Watts Bar Nuclear Plant Unit 1.

SATAN-VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass, energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the Emergency Core Cooling refills the reactor vessel and provides cooling to the core. The most important feature is the steam/water mixing model (See Section 2.5.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

2.3 BREAK SIZE AND LOCATION

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed for the Ice Optimization Program is the pump suction double ended guillotine, DEPSG (10.46 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for each case analyzed. The following information provides a discussion on each break location.

The hot leg double ended guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators venting directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). The mass and energy releases for the hot leg break have not been included in the scope of this containment integrity analysis because for the hot leg break only the blowdown phase of the transient is of any significance. Since there are no reflood and post-reflood phases to consider, the limiting peak pressure calculated would be the compression peak pressure and not the peak pressure following ice bed meltout.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this program.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment. This break has been determined to be the limiting break for all ice condenser plants.

In summary, the analysis of the limiting break location for an ice condenser containment has been performed and is shown in this report. The double-ended pump suction guillotine break has historically been considered to be the limiting break location, by virtue of its consideration of all energy sources in the Reactor Coolant System (RCS). This break location provides mechanism for the release of the available energy in the RCS, including both the broken and intact loop steam generators.

2.4 APPLICATION OF SINGLE FAILURE CRITERIA

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the pump suction (DEPSG) break. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the

blowdown period which is limited by the compression peak pressure.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, i.e. ECCS pumps and heat exchangers.

2.5 MASS AND ENERGY RELEASE DATA

2.5.1 Blowdown Mass and Energy Release Data

A version of the SATAN-VI code is used for computing the blowdown transient, which is the code used for the Emergency Core Cooling System (ECCS) calculation in Reference 2.

The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 2-4 presents the calculated mass and energy releases for the blowdown phase of the DEPSG break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

2.5.2 Reflood Mass and Energy Release Data

The WREFLOOD code used for computing the reflood transient, is a modified version of that used in the 1981 ECCS evaluation model, Reference 2.

The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations (during the core reflooding transient) of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e. the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving ECCS water.

This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model in recent analyses, e.g. D.C. Cook Docket [Reference 3]. Even though the Reference 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented [Reference 3]. This assumption is justified and supported by test data, and is summarized as follows:

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests [Reference 4], which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, one group corresponds almost directly to containment integrity reflood conditions. The injection flowrates from this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double ended guillotine break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam, which is condensed, that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in References 1 and 4.

Table 2-5 presents the calculated mass and energy release for the reflood phase of the pump suction double ended rupture with minimum safety injection.

The transients of the principal parameters during reflood are given in Table 2-6.

2.5.3 Post-Reflood Mass and Energy Release Data

The FROTH code [Reference 5] is used for computing the post-reflood transient.

The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two- phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. The methodology for the use of this model is described in Reference 1.

After steam generator depressurization/equilibration, the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Table 2-7 presents the two-phase post-reflood (froth) mass and energy release data for the pump suction double ended case.

2.5.4 Decay Heat Model

On November 2, 1978 the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS standard 5.1 for the determination of decay heat. This standard was used in the mass and energy release model with the following input specific for the Watts Bar Nuclear Plant Unit 1. The primary assumptions which make this calculation specific for the Watts Bar Nuclear Plant Unit 1 are the enrichment factor, minimum/maximum new fuel loadng per cycle, and a conservative end of cycle core average burnup. A conservative lower bound for enrichment of 3% was used. Table 2-2 lists the decay heat curve used in the Watts Bar Ice Weight Optimization analysis.

Significant assumptions in the generation of the decay heat curve:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and N_p-239.

2. Decay heat power from the following fissioning isotopes are included; U-238, U-235, and Pu-239.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Equation 11, of Reference 6 (up to 10,000 seconds) and Table 10 of Reference 6 (beyond 10,000 seconds).

5. The fuel has been assumed to be at full power for 1096 days.

6. The number of atoms of U-239 produced per second has been assumed to be equal

to 70% of the fission rate.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

2.5.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is Tsat at the containment design pressure. After the FROTH calculations, steam generator secondary energy is removed until the steam generator reaches Tsat at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation (Reference 7). Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used is based on the final heat removal rate calculated by FROTH. The remaining SG energy available to be released is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified equilibration pressure, assuming saturated conditions. This energy is then divided by the energy removal rate, resulting in an equilibration time.

2.6 SOURCES OF MASS AND ENERGY

The sources of mass considered in the LOCA mass and energy release analysis are given in Table 2-8. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Table 2-9. The energy sources include:

1. Reactor Coolant System Water

2. Accumulator Water

3. Pumped Injection Water

4. Decay Heat

5. Core Stored Energy

6. Reactor Coolant System Metal
   - Primary Metal (includes SG tubes)

7. Steam Generator Metal
   (includes transition cone, shell, wrapper,
   and other internals)

8. Steam Generator Secondary Energy
   (includes fluid mass and steam mass)

9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

It should be noted that the inconsistency in the energy balance tables from the end of Reflood to the time of intact loop steam generator depressurization/equilibration, i.e., "Total Available" data versus "Total Accountable" resulted from the omission of the reactor upper head in the analysis following blowdown. It has been concluded that the results are more conservative when the upper head is neglected. This does not affect the instantaneous mass and energy releases, or the integrated values, but causes an increase in the total accountable energy within the energy balance table.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

TABLE 2-1

WATTS BAR NUCLEAR PLANT UNIT 1
SYSTEM PARAMETERS
INITIAL CONDITIONS

PARAMETERS	VALUE
Core Thermal Power (MWt)	3459
Reactor Coolant System Flowrate, per Loop (gpm)	93100.
Vessel Outlet Temperature* (°F)	619.1
Core Inlet Temperature* (°F)	557.3
Vessel Average Temperature* (°F)	588.2
Initial Steam Generator Steam Pressure (psia)	980.
Steam Generator Design	Model D-3
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	122474
Accumulator	
Water Volume (ft^3)	1095/Tank plus 24.06 per line.
N_2 Cover Gas Pressure (psig)	600
Temperature (°F)	130
Safety Injection Delay (sec) (includes time to reach pressure setpoint)	32.0
Auxiliary Feedwater Flow (GPM/SG)†	~187

* (analysis value includes an additional +6.0°F allowance for instrument error and deadband)

† Auxiliary Feedwater flow necessary to maintain an SG narrow range level of 36% was modeled. Since the SG is assumed to be completely isolated, the Auxiliary Feedwater flow was only active for about 30 minutes.

TABLE 2-2

WATTS BAR NUCLEAR PLANT UNIT 1
SYSTEM PARAMETERS
DECAY HEAT CURVE

TIME (SEC)	DECAY HEAT (BTU/BTU)
10.	.0506850
15.	.0477187
20.	.0456218
40.	.0406962
60.	.0378482
80.	.0358667
100.	.0343802
150.	.0318330
200.	.0301404
400.	.0264229
600.	.0242907
800.	.0227336
1000.	.0214999
1500.	.0192069
2000.	.0175824
4000.	.0140451
6000.	.0123786
8000.	.0113975
10000.	.0107264
15000.	.0100411
20000.	.0093567
40000.	.0079090
50000.	.0071368
80000.	.0066021
100000.	.0062046
150000.	.0054924
200000.	.0050014
400000.	.0038711
600000.	.0032712
800000.	.0028872
1000000.	.0026231

Key Assumptions

-End of Cycle Core Average Burnup less than 45,000 Mwd/MTU
-Standard and V5H fuel
-Core Average Enrichment greater than 3.0%

TABLE 2-3

WATTS BAR NUCLEAR PLANT UNIT 1
SAFETY INJECTION FLOW
MINIMUM SAFETY INJECTION

INJECTION MODE

RCS Pressure (psia)	Total Flow (GPM)
15.0	4788.3
55.0	4330.4
115.0	3477.3
175.0	2067.7
215.0	886.0
315.0	852.8

INJECTION MODE (POST-REFLOOD PHASE)

RCS Pressure (psig)	Total Flow (GPM)
13.5	4637.72

RECIRCULATION MODE
(W/O RHR SPRAY)

RCS Pressure (psig)	Total Flow (GPM)
0	3757.5

RECIRCULATION MODE
(W/ RHR SPRAY)

RCS Pressure (psig)	Total Flow to RCS (GPM)
0	1855.

TABLE 2-4

WATTS BAR NUCLEAR PLANT UNIT 1

DOUBLE-ENDED PUMP SUCTION GUILLOTINE

BLOWDOWN MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
.00000	.0	.0	.0	.0
.00106	91473.7	51248.1	42757.3	23896.9
.00206	42955.1	24008.0	42573.3	23792.4
.101	42503.1	23834.3	22097.2	12338.1
.201	43339.5	24490.9	24176.6	13509.5
.301	44366.8	25335.1	24242.0	13556.5
.502	45862.8	26862.9	22296.6	12487.5
.601	45530.9	26983.4	21371.8	11974.1
.801	42484.8	25693.1	19928.8	11170.7
1.00	39544.8	24368.2	19298.7	10824.5
1.20	37020.2	23262.0	19108.0	10720.7
2.20	28255.7	19308.9	18896.7	10605.3
2.40	25402.6	17633.7	18617.8	10450.6
2.60	21678.4	15264.0	18000.3	10106.0
3.50	16360.6	11851.1	16210.1	9126.8
3.80	15077.5	10967.9	15776.6	8894.4
4.20	14035.8	10226.3	15279.0	8629.9
4.60	13441.5	9765.4	14811.9	8382.3
5.00	13257.2	9568.5	16166.4	9167.8
6.00	13191.3	9364.2	15135.9	8618.2
6.60	12777.0	9053.4	14505.5	8270.4
6.80	12788.1	9058.9	14412.3	8218.1
7.00	12210.8	8944.3	14493.5	8262.5
7.20	10650.0	8405.3	14208.2	8092.6
7.40	10261.3	8190.8	14004.9	7974.7
8.00	11126.8	8400.8	13504.2	7688.5
8.40	12274.7	8874.1	13135.7	7475.3
9.00	13232.2	9155.6	12581.1	7153.5
9.40	12766.8	8727.1	12250.8	6961.4
10.2	10649.9	7335.3	11715.6	6649.1
10.8	9517.9	6725.6	11327.3	6424.6
12.0	7946.8	5916.2	10528.8	5962.4
13.2	6868.8	5265.3	9696.9	5483.5
15.6	5560.8	4219.5	8154.6	4617.2
16.4	5016.5	4322.6	7007.9	4036.2
17.0	3745.2	4134.8	6328.7	3317.5
17.6	2755.2	3382.2	5113.6	2416.3
18.6	1896.4	2374.3	3573.7	1486.2
19.0	1573.5	1978.3	6445.1	2574.5
19.2	1422.1	1791.6	6523.6	2603.7
19.6	1205.0	1523.9	3559.5	1408.2

TABLE 2-4

WATTS BAR NUCLEAR PLANT UNIT 1

DOUBLE-ENDED PUMP SUCTION GUILLOTINE

BLOWDOWN MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
20.0	1054.7	1336.7	1900.4	739.6
20.2	962.8	1221.5	2056.3	715.9
20.6	783.8	996.6	3656.8	1161.0
23.0	273.1	350.4	1643.0	456.1
23.8	196.8	253.0	1290.4	344.3
24.4	177.0	227.8	1217.4	324.0
25.4	138.3	178.3	309.1	86.2
26.0	109.3	141.0	.0	.0
27.8	.0	.0	.0	.0

TABLE 2-5

WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
REFLOOD MASS AND ENERGY RELEASE - MINIMUM SAFETY INJECTION

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
27.8	.0	.0	.0	.0
28.8	.0	.0	.0	.0
28.9	52.4	61.0	.0	.0
29.2	9.7	11.3	.0	.0
31.9	76.8	89.4	.0	.0
33.9	103.4	120.6	.0	.0
34.9	247.4	290.2	3287.1	476.2
36.0	324.6	381.8	4337.8	632.3
36.3	324.2	381.4	4332.2	632.9
37.0	321.8	378.6	4299.4	629.8
38.0	317.0	372.9	4242.6	623.1
42.0	297.4	349.5	4004.6	592.9
43.0	292.8	344.1	3947.8	585.6
45.0	284.2	333.8	3839.4	571.4
47.0	276.2	324.3	3737.7	558.1
49.0	268.7	315.5	3642.4	545.6
51.0	261.9	307.3	3553.0	533.9
53.0	255.5	299.8	3469.0	522.9
55.0	249.6	292.7	3389.8	512.5
57.0	244.1	286.2	3315.0	502.8
59.0	238.9	280.0	3244.3	493.5
61.0	234.0	274.3	3177.1	484.7
63.0	229.4	268.9	3113.3	476.4
67.0	221.0	258.9	2994.6	460.8
71.0	213.4	250.0	2886.2	446.7
75.0	206.6	241.9	2786.4	433.6
79.0	200.3	234.5	2694.1	421.5
83.0	194.6	227.7	2608.3	410.3
84.0	254.5	297.9	250.6	123.1
85.0	341.4	401.9	286.5	173.8
86.0	341.9	402.5	286.8	174.4
93.0	308.3	362.5	271.0	152.8
97.0	291.8	342.8	264.0	143.6
101.0	278.5	327.0	258.5	136.3
105.0	267.2	313.6	253.9	130.2
109.0	257.7	302.3	250.1	125.1
121.0	237.2	278.1	241.9	114.2
133.0	225.3	264.0	237.2	108.0
145.0	218.9	256.4	234.6	104.7
161.0	215.0	251.8	233.0	102.6
165.0	215.8	252.7	233.9	102.9
173.0	218.1	255.4	238.6	104.0
181.0	220.0	257.7	244.9	105.0
197.0	220.9	258.8	259.0	105.8
205.0	219.8	257.5	266.6	105.6

TABLE 2-5

WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
REFLOOD MASS AND ENERGY RELEASE - MINIMUM SAFETY INJECTION

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY	FLOW	ENERGY
SECOND	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
213.0	217.9	255.2	275.2	105.3
215.0	217.4	254.6	277.5	105.2
225.0	213.5	250.0	289.3	104.4
241.0	203.9	238.7	308.6	102.6
245.5	200.7	234.9	314.7	102.1

TABLE 2-6

WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SAFETY INJECTION
PRINCIPAL PARAMETERS DURING REFLOOD

TIME	TEMP	FLOODING RATE	CARRY-OVER FRACTION	CORE HEIGHT	DOWN-COMER HEIGHT	FLOW FRACTION	INJECTION TOTAL	INJECTION ACCUMULATOR	INJECTION SI SPILL	ENTHALPY
SECOND	DEGREE-F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
27.8	204.9	.000	.000	.00	.00	.250	.0	.0	.0	.00
28.6	202.3	20.491	.000	.60	1.29	.000	6624.5	6624.5	.0	99.50
28.8	200.9	22.076	.000	.96	1.20	.000	6574.9	6574.9	.0	99.50
28.8	200.5	21.864	.000	1.05	1.17	.000	6550.5	6550.5	.0	99.50
29.2	200.0	2.080	.085	1.28	1.72	.205	6452.8	6452.8	.0	99.50
29.3	200.0	2.158	.099	1.30	1.99	.227	6435.5	6435.5	.0	99.50
29.4	200.0	2.143	.117	1.31	2.26	.266	6401.2	6401.2	.0	99.50
29.6	200.1	2.161	.149	1.35	2.87	.298	6356.4	6356.4	.0	99.50
30.8	200.4	1.933	.323	1.50	6.05	.353	6100.2	6100.2	.0	99.50
31.9	200.8	1.878	.436	1.61	9.07	.366	5882.3	5882.3	.0	99.50
36.0	202.0	3.520	.632	1.97	16.11	.566	5144.4	4545.7	.0	96.42
36.3	202.0	3.463	.642	2.00	16.11	.566	5104.8	4506.1	.0	96.39
37.0	202.2	3.346	.660	2.07	16.11	.566	5026.6	4427.3	.0	96.34
42.7	204.4	2.840	.723	2.51	16.11	.558	4538.5	3933.1	.0	95.97
51.3	208.7	2.490	.749	3.00	16.11	.545	4022.6	3410.1	.0	95.47
61.8	214.6	2.235	.762	3.50	16.11	.531	3571.6	2953.2	.0	94.92
73.8	221.3	2.036	.771	4.00	16.11	.518	3188.5	2565.7	.0	94.33
83.0	225.8	1.920	.776	4.34	16.11	.509	2954.0	2328.6	.0	93.90
84.0	226.2	2.380	.783	4.38	16.11	.583	620.3	.0	.0	73.03
85.0	226.7	2.911	.767	4.43	16.01	.598	590.9	.0	.0	73.03
87.0	227.7	2.860	.767	4.55	15.77	.598	591.5	.0	.0	73.03
95.9	231.6	2.511	.773	5.00	14.98	.594	603.0	.0	.0	73.02
109.0	236.2	2.201	.779	5.58	14.44	.586	611.2	.0	.0	73.03
120.0	239.3	2.045	.783	6.00	14.33	.581	615.1	.0	.0	73.03
135.0	242.8	1.923	.788	6.53	14.48	.576	618.0	.0	.0	73.03

TABLE 2-6

WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SAFETY INJECTION
PRINCIPAL PARAMETERS DURING REFLOOD

TIME	TEMP	FLOODING RATE	CARRY-OVER FRACTION	CORE HEIGHT	DOWN-COMER HEIGHT	FLOW FRACTION	INJECTION TOTAL	INJECTION ACCUMULATOR	INJECTION SI SPILL	ENTHALPY
SECOND	DEGREE-F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
149.3	245.5	1.865	.792	7.00	14.79	.574	619.3	.0	.0	73.02
161.0	247.4	1.840	.795	7.38	15.11	.573	619.8	.0	.0	73.03
165.0	247.5	1.844	.795	7.50	15.23	.574	619.7	.0	.0	73.03
180.8	246.9	1.866	.794	8.00	15.61	.579	619.0	.0	.0	73.03
183.0	246.9	1.867	.794	8.07	15.65	.580	618.9	.0	.0	73.03
197.0	247.5	1.857	.794	8.52	15.86	.584	618.7	.0	.0	73.03
212.2	247.2	1.822	.794	9.00	16.00	.588	619.2	.0	.0	73.03
229.0	247.3	1.755	.794	9.52	16.08	.592	620.3	.0	.0	73.03
245.5	247.5	1.661	.795	10.00	16.10	.592	622.2	.0	.0	73.03

TABLE 2-7

WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SAFETY INJECTION
POST REFLOOD MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY	FLOW	ENERGY
		THOUSAND		THOUSAND
SECONDS	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
245.6	208.2	259.7	431.8	117.2
245.6	208.2	259.7	431.8	117.2
250.6	207.3	258.6	432.7	117.3
255.6	207.7	259.0	432.3	117.1
260.6	206.8	257.9	433.2	117.1
265.6	207.1	258.3	432.9	117.0
270.6	206.2	257.1	433.8	117.0
275.6	206.4	257.5	433.5	116.9
280.6	205.5	256.3	434.5	117.0
285.6	205.7	256.6	434.2	116.8
290.6	204.7	255.4	435.2	116.9
295.6	204.9	255.6	435.0	116.7
300.6	203.9	254.4	436.0	116.8
305.6	204.1	254.6	435.9	116.7
310.6	203.1	253.3	436.9	116.8
315.6	203.2	253.4	436.8	116.6
320.6	202.1	252.1	437.8	116.7
330.6	202.3	252.3	437.7	116.5
335.6	201.1	250.9	438.8	116.6
345.6	201.1	250.8	438.9	116.3
350.6	199.9	249.4	440.0	116.5
370.6	199.4	248.6	440.6	116.1
390.6	197.2	245.9	442.8	116.1
425.6	194.2	242.2	445.8	115.9
430.6	194.5	242.6	445.5	115.7
440.6	193.1	240.8	446.9	115.8
445.6	193.2	241.0	446.8	115.7
475.6	190.6	237.8	449.3	115.5
480.6	190.7	237.8	449.3	115.4
510.6	187.5	233.9	452.4	115.3
515.6	187.6	234.0	452.3	115.2
545.6	184.6	230.3	455.3	115.1
550.6	184.4	230.0	455.6	115.1
555.6	82.0	102.2	558.0	137.0
771.7	82.0	102.2	558.0	137.0
771.8	79.5	98.8	560.5	132.3
775.6	79.4	98.6	560.6	132.2
1595.6	66.0	82.0	573.9	126.4
1600.3	66.0	82.0	582.5	126.8
1707.3	64.8	80.5	583.7	148.3
1712.3	64.8	80.4	453.7	134.5
2041.3	64.8	80.4	453.7	134.5
2041.4	63.0	72.5	455.5	48.8

TABLE 2-8

WATTS BAR NUCLEAR PLANT UNIT 1

DOUBLE-ENDED PUMP SUCTION GUILLOTINE

MINIMUM SAFETY INJECTION - MASS BALANCE

		Start of Accident	End of Blowdown	Bottom of Core Recovery	End of Reflood	Broken Loop SG Equilibration	Intact Loop SG Equilibration
	TIME (SECONDS)	.00	27.80	27.80	245.54	771.79	2041.34
		MASS (THOUSANDS LBM)					
INITIAL MASS in RCS and ACCUMULATORS		773.52	773.52	773.52	773.52	773.52	773.52
ADDED MASS	PUMPED INJECTION	.00	.00	.00	129.64	466.38	1239.84
	TOTAL ADDED	.00	.00	.00	129.64	466.38	1239.84
*** TOTAL AVAILABLE***		773.52	773.52	773.52	903.16	1239.90	2013.36
DISTRIBUTION	REACTOR COOLANT	497.46	72.57	72.69	134.65	134.65	134.65
	ACCUMU-LATOR	276.06	197.75	197.63	.00	.00	.00
	TOTAL CONTENTS	773.52	270.32	270.32	134.65	134.65	134.65
EFFLUENT	BREAK FLOW	.00	503.18	503.18	757.90	1094.64	1867.80
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	503.18	503.18	757.90	1094.64	1867.80
TOTAL ACCOUNTABLE		773.52	773.50	773.50	892.55	1229.30	2002.45

TABLE 2-9
WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SAFETY INJECTION - ENERGY BALANCE

		Start of Accident	End-of-Blowdown	Bottom of Core Recovery	End of Reflood	Broken Loop SG Equili-bration	Intact Loop SG Equili-bration
TIME	(Seconds)	.00	27.80	27.80	245.54	771.79	2041.34
		ENERGY (MILLION BTU)					
INITIAL ENERGY	IN RCS, ACCUM, & SG	852.47	852.47	852.47	852.47	852.47	852.47
ADDED ENERGY	PUMPED INJECTI0N	.00	.00	.00	9.47	34.06	96.83
	DECAY HEAT	.00	8.50	8.50	32.58	76.98	160.12
	HEAT FROM SECON-DARY	.00	.48	.48	.48	5.24	15.65
	TOTAL ADDED	.00	8.98	8.98	42.53	116.27	272.60
TOTAL AVAILABLE		852.47	861.45	861.45	895.00	968.74	1125.07
DISTRIBUTION							
	REACTOR COOLANT	296.96	13.17	13.18	29.85	29.85	29.85
	ACCUM-ULATOR	27.46	19.67	19.66	.00	.00	.00
	CORE STORED	25.94	14.51	14.51	3.98	3.88	3.63
	PRIMARY METAL	154.76	147.14	147.14	120.76	80.13	55.07
	SECOND-ARY METAL	66.60	67.08	67.08	60.43	46.85	28.39
	STEAM GENERAT 0R	280.76	283.33	283.33	250.37	193.10	123.42
	TOTAL CONTENT S	852.47	544.90	544.90	465.39	353.80	240.37
EFFLUEN T	BREAK FLOW	.00	315.96	315.96	417.48	602.81	858.26
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUEN T	.00	315.96	315.96	417.48	602.81	858.26
TOTAL ACCOUNTABLE		852.47	860.86	860.86	882.87	956.61	1098.63

TABLE 2-10

WATTS BAR NUCLEAR PLANT UNIT 1
SEQUENCE OF EVENTS

EVENT	TIME (Sec)
Rupture	0.0
Accumulator Flow Starts	16.0
Assumed Initiation of ECCS	35.0
End of Blowdown	27.8
Assumed Initiation of Spray System	221.
Accumulators Empty	83.79
End of Reflood	245.54
Low Level Alarm of Refueling Water Storage Tank	1571.3
Beginning of Recirculation Phase of Safeguards Operation	1631.3

3.0 LOCA CONTAINMENT INTEGRITY ANALYSIS

3.1 Description of LOTIC-1 Model

Early in the ice condenser development program it was recognized that there was a need for modeling of long-term ice condenser performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC code, described in reference 8.

The model of the containment consists of five distinct control volumes, the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartment. The ice condenser control volume with unmelted and melted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long-term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the Reactor Coolant System, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These flow rates then are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

The condensation of steam is assumed to take place in a condensing node located, for the purpose of calculation, between the two control volumes in the ice storage compartment. The exit temperature of the air leaving this node is set equal to a specific value that is equal to the temperature of the ice filled control volume of the ice storage compartment. Lower compartment exit temperature is used if the ice bed section is melted.

3.2 Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis of the double-ended pump suction guillotine case with the steam generators considered as an active heat source for the Watts Bar Nuclear Plant Containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

2. $2.029375 * 10^6$ lbs. of ice initially in the ice condenser.

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 2.5 are used.

4. The blowdown period mass and energy from Table 2-4 is conservatively compressed into a 10 second period in order to melt an amount of ice consistent with the Waltz Mill ice condenser test. (Reference 10)

5. Blowdown and post-blowdown ice condenser drain temperature of 190°F and 130°F are used. (These values are based on the Long-Term Waltz-Mill ice condenser test data described in Reference 10)

6. Nitrogen from the accumulators in the amount of 2251 lbs. is included in the calculations.

7. Hydrogen gas was added to the containment in the amount of 24,051 Standard Cubic Feet (SCF) over 24 hours. Sources accounted for were radiolysis in the core and sump post-LOCA, corrosion of plant materials (Aluminum, Zinc, and painted surfaces found in containment), reaction of 1% of the Zirconium fuel rod cladding in the core, and hydrogen gas assumed to be dissolved in the Reactor Coolant System water. (This bounds tritium producing core designs)

8. Essential service water temperature of 85°F is used on the spray heat exchanger and the component cooling heat exchanger.

9. The air return fan is effective, 10 minutes after the transient is initiated.

10. No maldistribution of steam flow to the ice bed is assumed. (This assumption is conservative, contributes to early ice bed melt out time.)

11. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

12. The initial conditions in the containment are a temperature of 100°F in the lower and dead-ended volumes, 85°F in the upper volume and a temperature 15°F in the ice condenser. All volumes are at a pressure of 0.3 psig and a 10% relative humidity, except the ice condenser which is at 100% relative humidity.

13. The minimum ECCS and Containment Spray flow rates versus time assumed in the peak containment pressure calculations were calculated based upon the assumption of loss of offsite power.

14. Containment structural heat sinks are assumed with conservatively low heat transfer rates. (See Tables 3-2 and 3-3) Note: The Dead-Ended compartment structural heat sinks were conservatively neglected.

15. The Containment compartment volumes were based on the following: Upper Compartment 645,818 ft^3; Lower Compartment 221,074 ft^3; and Dead-Ended Compartment 146,600 ft^3.

16. The operation of one containment spray heat exchanger ($UA = 2.74 * 10^6$ Btu/hr-°F), for containment cooling and the operation of one RHR heat exchanger ($UA = 1.57 * 10^6$ Btu/hr-°F) for core cooling. The component cooling heat exchanger was modeled at $7.09 * 10^6$ Btu/hr-°F.

17. The air return fan returns air at a rate of 40,000 cfm from the upper to the lower compartment.

18. An active sump volume of 51,000 ft^3 is used.

19. 100.6% of 3459 MWt power is used in the calculations.

20. Subcooling of ECC water from the RHR heat exchanger is assumed.

21. Nuclear service water flow to the containment spray heat exchanger was modeled as 5200 gpm. Also the nuclear service water flow to the component cooling heat exchanger was modeled as 5000 gpm.

22. The decay heat curve conservatively used to calculate mass and energy releases after steam generator equilibration is the same as presented in the mass and energy release section of this report.

The minimum time at which the RHR pumps can be diverted to the RHR sprays are specified in the plant operating procedures as 60 minutes after the containment isolation signal.

3.3 Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference forms accounts for transient conduction into and out of the node and temperature rise of the node for the containment structural heat sinks used in the analysis. The heat sink and material property data from Reference 11 was used to develop Tables

3-2 and 3-3.

The heat transfer coefficient to the containment structure is based primarily on the work of Tagami [Reference 9]. When applying the Tagami correlations, a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure below the design pressure.

3.4 Analysis Results

The results of the analysis shows that the maximum calculated containment pressure is 10.438 psig, for the double-ended pump suction minimum safeguards break case, assuming an ice bed mass of 2.029375×10^6 Lbm. This pressure is less than the design pressure of 13.5 psig and therefore shows the acceptability of the reduced ice mass. The pressure peak occurred at approximately 6373.5 seconds, with ice bed meltout at approximately 3625.5 seconds. It is noted that the apparent containment pressure margin between 10.438 psig and the design pressure of 13.5 psig can not be used to further reduce the ice mass. The ice bed mass is limited by the Spray Switchover time of 3447 seconds and the margin between spray swithcover and ice bed meltout of at least 150 seconds.

The following plots show the containment integrity transient, as calculated by the LOTIC-1 code.

Figure 3-1, Containment Pressure Transient
Figure 3-2, Upper Compartment Temperature Transient
Figure 3-3, Lower Compartment Temperature Transient
Figure 3-4, Active and Inactive Sump Temperature Transient
Figure 3-5, Ice Melt Transient
Figure 3-6, Comparison of Containment Pressure VS Ice Melt Transients

Tables 3-4 and 3-5 give energy accountings at various points in the transient.

Tables 3-6 through 3-8 provide data points for Figures 3-1 through 3-6.

3.5 Relevant Acceptance Criteria

The LOCA mass and energy analysis has been performed in accordance with the criteria shown in the Standard Review Plan (SRP) section 6.2.1.3. In this analysis, the relevant requirements of General Design Criteria (GDC) 50 and 10 CFR Part 50 Appendix K have been included by confirmation that the calculated pressure is less than the design pressure, and because all available sources of energy have been included. These sources include: reactor power, decay heat, core stored energy, energy stored in the reactor vessel and internals, metal-water reaction

energy, and stored energy in the secondary system.

The containment integrity peak pressure analysis has been performed in accordance with the criteria shown in the SRP section 6.2.1.1.b, for ice condenser containments. Conformance to GDC's 16, 38, and 50 is demonstrated by showing that the containment design pressure is not exceeded at any time in the transient. This analysis also demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA.

3.6 Conclusions

Based upon the information presented in this report, it may be concluded that operation with an ice weight of 2.029375 million pounds for the Watts Bar Nuclear Plant is acceptable. Operation with an ice mass of 2.029375 million pounds results in a calculated peak containment pressure of 10.438 psig, as compared to the design pressure of 13.5 psig. Further, the ice bed mass of 2.029375×10^6 Lbms equates to an average of 1044 Lbm per basket. This average value recognizes that all baskets may not have the same initial weight nor have the same sublimation rate. To ensure that a sufficient quantity of ice exists in each basket to survive the blowdown phase of a LOCA, a minimum amount of ice per basket to survive the blowdown would be approximately 313 Lbm, based on Table 3-4. To ensure that an adequate distribution of ice exists in the Ice Condenser to prevent early burn-through of a localized area, 313 Lbm of ice should be the minimum weight of ice per basket at any time while also ensuring that the average weight per basket remains above 1044 Lbm.

Thus, the most limiting case has been considered, and has been demonstrated to yield acceptable results.

4.0 REFERENCES

1. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version", WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Non-Proprietary).

2. "Westinghouse ECCS Evaluation Model - 1981 Version", WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev.1 (Non-Proprietary)

3. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D.C. Cook Nuclear Plant Unit 1", June 9, 1989.

4. EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam; 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

5. "Westinghouse Mass and Energy Release Data For Containment Design", WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Non-Proprietary).

6. ANSI/ANS-5.11979, "American National Standard for Decay Heat Power in Light Water Reactors", August 1979.

7. W. H. McAdam, Heat Transmission, McGraw-Hill 3rd edition, 1954, p.172.

8. "Long-term Ice Condenser Containment Code - LOTIC Code", WCAP-8354-P-A, April 1976 (Proprietary), WCAP-8355-A (Non-Proprietary).

9. Tagami, Takasi, Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June, 1965 (No. 1).

10. WCAP-8110, Supplement 6, (Non-Proprietary), "Test Plans and Results for the Ice Condenser System, Ice Condenser Full-Scale Section Test at the Waltz Mill Facility," May 1974

11. TVA Letter W-7567, J. E. Maddox (TVA) to J. W. Irons (W),"Design Inputs for Westinghouse LOCA M&E / Containment analysis (Reduced Ice Weight) - Data Request," May 25, 2001

12. Westinghouse Letter WAT-D-10156, J. W. Irons (W) to Mr. W. L. Elliott (TVA), "Tennessee Valley Authority, Watts Bar Nuclear Plant Units 1 & 2, Watts Bar ECCS Report Revision," October 25, 1995.

13. Westinghouse Letter WAT-D-10922, J. W. Irons (W) to Mr. J. E. Maddox (TVA), "Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, ECCS Analysis Report ," April 27, 2001

TABLE 3-1

WATTS BAR NUCLEAR PLANT UNIT 1

ECCS SWITCHOVER PUMP FLOW VS. TIME
(LOSS OF OFF-SITE POWER AT EVENT INITIATION)

Time After Safeguards Initiation (Sec)	ECCS Flow To Core (RWST) (Gpm)	Spray (Flow) (Gpm)	RHR Spray (Flow) (Gpm)	ECCS Flow To Core (Sump) (Gpm)	Comments
0	0	0	0	0	"S" - Signal
11.9	0	0	0	0	
12.0	358.9	0	0	0	CC Pump Start
16.9	359.9	0	0	0	
17.0	942.3	0	0	0	SI Pump Start
21.9	942.3	0	0	0	
22.0	*4699.8	0	0	0	RHR Pump Start
190.9	4699.8	0	0	0	
191.0	4699.8	4000	0	0	Containment Spray Start
1631.2	4699.8	4000	0	0	
1631.3	4699.8	4000	0	3757.5	RHR Switchover to Sump
1708.2	4699.8	4000	0	3757.5	
1708.3	0	4000	0	3757.5	CCP/SI Pump Switchover
3326.9	0	4000	0	3757.5	
3327.0	0	0	0	3757.5	CS Pump Stopped
3446.9	0	0	0	3757.5	
3447.0	0	4000 (Sump)	0	3757.5	CS Pump Switchover
3600.0	0	4000 (Sump)	0	3757.5	
3600.1	0	4000(Sump)	1475	1855	RHR Alignment for Auxiliary CS
End of Transient	0	4000 (Sump)	1475	1855	

*4699.8 gpm Total Flow (RWST)
358.9 gpm - 1 Centrifugal Charging Pump
583.4 gpm - 1 Safety Injection Pump
3757.5 gpm - 1 RHR Pump

TABLE 3-2

WATTS BAR NUCLEAR PLANT UNIT 1
STRUCTURAL HEAT SINK TABLE

Upper Compartment	Area (Ft^2)	Thickness (Ft)	Material
1. Operating Deck			
Slab 1	4880.	1.066	Concrete
Slab 2	18280.	0.0055	Paint
		1.4	Concrete
Slab 3	760.	0.0055	Paint
		1.5	Concrete
Slab 4	3840.	0.0208	Stainless Steel
		1.5	Concrete
2. Shell and Misc.			
Slab 5	56331.	0.001	Paint
		0.079	Steel
Lower Compartment			
1. Operating Deck, Crane Wall, and Interior Concrete			
Slab 6	31963.	1.43	Concrete
2. Operating Deck			
Slab 7	2830.	0.0055	Paint
		1.1	Concrete
Slab 8	760	0.0055	Paint
		1.75	Concrete
3. Interior Concrete and Stainless Steel			
Slab 9	2270.	0.0208	Stainless Steel
		2.0	Concrete
4. Floor*			
Slab 10	15921.	0.0055	Paint
		1.6	Concrete
5. Misc. Steel			
Slab 11	28500.	0.001	Paint
		0.0656	Steel

TABLE 3-2 (Cont'd)

WATTS BAR NUCLEAR PLANT UNIT 1
STRUCTURAL HEAT SINK TABLE

Ice Condenser	Area(Ft^2)	Thickness (Ft)	Material
1. Ice Baskets			
Slab 12	149,600.	0.00663	Steel
2. Lattice Frames			
Slab 13	75,865.	0.0217	Steel
3. Lower Support Structure			
Slab 14	28670.	0.0587	Steel
4. Ice Condenser Floor			
Slab 15	3336.	0.0055	Paint
		0.333	Concrete
5. Containment Wall Panels & Containment Shell			
Slab 16	19100.	1.0	Steel & Insulation
		0.0625	Steel Shell
6. Crane Wall Panels and Crane Wall			
Slab 17	13055.	1.0	Steel & Insulation
		1.0	Concrete

TABLE 3-3

WATTS BAR NUCLEAR PLANT UNIT 1
MATERIAL PROPERTIES TABLE

Material	Thermal Conductivity Btu/hr-ft-°F	Volumetric Heat Capacity Btu/ft^3-°F
Paint on Steel	0.21	19.9
Paint on Concrete	0.083	39.9
Concrete	0.8	31.9
Stainless Steel	9.4	53.68
Carbon Steel	26.0	53.9
Insulation on Steel	0.15	2.75

TABLE 3-4

WATTS BAR NUCLEAR PLANT UNIT 1

ENERGY ACCOUNTING

	Approx. End of Blowdown (t=10.0 sec.)	Approx. End of Reflood (t=245.5 sec.)
	(In Millions of Btus)	
Ice Heat Removal	188.54	241.128
Structural Heat Sinks*	19.6	62.03
RHR Heat Exchanger Heat Removal*	0	0
Spray Heat Exchanger Heat Removal*	0	0
Energy Content of Sump	208.69	282.64
Ice Melted (Pounds) (10^6)	0.6087	0.819

* Integrated Energies

TABLE 3-5

WATTS BAR NUCLEAR PLANT UNIT 1

ENERGY ACCOUNTING

	Approx. Time of Ice Melt Out (t=3559 sec.)	Approx. Time Peak Pressure (t=6244 sec.)
	(In Millions of Btus)	
Ice Heat Removal	543.34	543.34
Structural Heat Sinks*	80.01	117.45
RHR Heat Exchanger Heat Removal*	28.598	68.218
Spray Heat Exchanger Heat Removal*	3.479	69.089
Energy Content of Sump	804.87	806.41
Ice Melted (Pounds) (10^6)	2.029375	2.029375

* Integrated Energies

TABLE 3-6

WATTS BAR NUCLEAR PLANT UNIT 1

CONTAINMENT PRESSURE AND ICE MELT MASS

TIME (SEC)	PRESSURE (PSIG)	MELTED ICE (LBM)
2.00	7.07	121737.08
10.00	7.07	608687.44
28.09	6.97	608687.44
56.09	6.87	616145.25
89.09	6.83	636368.94
117.47	6.11	679359.88
151.38	5.87	718923.81
184.38	6.11	754901.94
217.38	6.15	790485.13
277.87	6.70	851534.69
343.87	6.88	916660.75
542.87	6.93	1103785.13
565.77	6.36	1118689.50
598.77	6.48	1134049.00
664.77	6.03	1165300.00
698.77	5.89	1181757.13
797.02	5.69	1228346.50
863.02	5.65	1257403.50
1559.02	5.67	1527298.38
1625.02	5.72	1550011.13
1723.43	5.90	1587419.63
2038.93	6.07	1725927.75
2087.76	5.77	1736785.25
2129.26	5.64	1745749.75
2212.01	5.52	1763628.38
2377.76	5.47	1798820.13
3329.76	5.61	1984581.00
3412.51	6.29	1999384.25
3445.76	6.67	2005032.75
3462.26	6.35	2007725.88
3470.51	6.28	2009034.75
3487.01	6.24	2011591.63
3520.26	6.33	2016515.63

TABLE 3-6 (Cont'd)

WATTS BAR NUCLEAR PLANT UNIT 1

CONTAINMENT PRESSURE AND ICE MELT MASS

TIME (SEC)	PRESSURE (PSIG)	MELTED ICE (LBM)
3603.01	6.84	2027240.50
3611.26	6.85	2028158.50
3619.51	7.00	2028948.25
3627.75	7.43	2029375.00
3635.92	7.59	2029375.00
3710.42	8.21	2029375.00
3793.42	8.67	2029375.00
3967.17	9.21	2029375.00
4133.92	9.53	2029375.00
4299.92	9.76	2029375.00
4630.92	10.04	2029375.00
5376.91	10.37	2029375.00
6373.50	10.438	2029375.00
6494.13	10.44	2029375.00
10628.81	9.96	2029375.00
15123.68	9.76	2029375.00
21705.14	9.26	2029375.00
27087.07	9.05	2029375.00
32841.95	8.69	2029375.00
33765.09	8.70	2029375.00
40026.00	8.25	2029375.00
62455.47	7.61	2029375.00
79723.44	7.25	2029375.00
101896.87	6.91	2029375.00
155842.67	6.36	2029375.00
199199.81	6.07	2029375.00

TABLE 3-7

WATTS BAR NUCLEAR PLANT UNIT 1

CONTAINMENT UPPER AND LOWER COMPARTMENT TEMPERATURES

	UPPER COMPARTMENT	LOWER COMPARTMENT
TIME	TEMPERATURE	TEMPERATURE
(SEC)	(DEG-F)	(DEG-F)
2.00	92.11	232.50
10.00	92.10	232.50
28.09	87.80	232.30
56.09	86.75	232.30
89.09	86.88	232.00
102.24	87.60	232.00
117.47	88.25	232.00
134.38	88.90	231.90
151.38	89.50	229.80
200.38	90.91	225.90
217.38	91.29	221.60
244.87	99.15	218.80
260.87	101.22	219.20
294.87	102.91	221.10
376.87	103.50	221.90
542.87	103.60	224.70
553.18	103.61	224.90
565.77	103.62	225.00
598.77	103.63	225.30
631.77	106.40	225.40
681.77	107.18	225.50
780.02	107.30	225.50
863.02	107.33	225.40
1095.01	107.43	208.39
1112.01	107.44	207.00
1592.02	107.64	195.30
1707.43	107.68	194.80
2038.93	107.81	193.00
2083.76	107.81	192.30

TABLE 3-7 (Cont'd)

WATTS BAR NUCLEAR PLANT UNIT 1

CONTAINMENT UPPER AND LOWER COMPARTMENT TEMPERATURES

	UPPER COMPARTMENT	LOWER COMPARTMENT
TIME	TEMPERATURE	TEMPERATURE
(SEC)	(DEG-F)	(DEG-F)
2125.01	107.82	191.50
2208.01	107.84	194.50
2377.76	107.89	198.20
3329.76	108.66	177.90
3412.51	120.43	177.80
3445.76	126.03	177.70
3454.01	122.82	177.70
3466.26	119.77	177.70
3487.01	118.45	177.60
3516.01	119.59	177.60
3615.51	127.79	177.40
3623.76	129.87	177.40
3627.75	131.17	177.30
3631.92	131.07	177.30
3743.67	138.89	177.10
3851.17	142.44	176.90
4067.92	146.40	176.60
4282.92	148.86	176.50
4497.92	150.47	176.40
5376.91	154.29	176.10
6494.13	154.87	176.10
11002.90	152.41	176.00
21245.21	146.97	175.80
28949.87	143.75	175.70
40522.43	138.35	176.40
59674.80	133.77	179.70
73885.95	131.24	179.80
113088.14	126.10	183.20
138686.45	123.81	186.70
199995.28	119.52	190.60

TABLE 3-8

WATTS BAR NUCLEAR PLANT UNIT 1

CONTAINMENT ACTIVE AND INACTIVE SUMP TEMPERATURES

	ACTIVE SUMP	INACTIVE SUMP
TIME	TEMPERATURE	TEMPERATURE
(SEC)	(DEG-F)	(DEG-F)
2.00	189.99	.00
10.00	189.97	.00
28.09	189.95	.00
56.09	188.61	.00
89.09	187.15	.00
102.24	186.06	.00
117.47	185.12	.00
134.38	184.22	.00
151.38	183.42	.00
200.38	181.37	.00
217.38	180.73	.00
244.87	179.13	.00
260.87	178.30	.00
294.87	176.69	.00
376.87	173.43	.00
542.87	168.66	.00
553.18	168.43	.00
565.77	168.28	.00
598.77	167.93	.00
631.77	167.60	.00
681.77	167.12	.00
780.02	166.26	.00
863.02	165.68	.00
1095.01	164.41	.00
1112.01	164.34	164.35
1592.02	162.72	163.47
1707.43	162.41	163.32
2038.93	160.84	162.87

TABLE 3-8 (Cont'd)

WATTS BAR NUCLEAR PLANT UNIT 1

CONTAINMENT ACTIVE AND INACTIVE SUMP TEMPERATURES

	ACTIVE SUMP	INACTIVE SUMP
TIME	TEMPERATURE	TEMPERATURE
(SEC)	(DEG-F)	(DEG-F)
2083.76	160.12	162.81
2125.01	159.44	162.74
2208.01	158.11	162.56
2377.76	155.51	162.07
3329.76	143.41	157.99
3412.51	143.03	157.89
3445.76	142.88	157.86
3454.01	142.82	157.85
3466.26	142.72	157.83
3487.01	142.54	157.81
3516.01	142.29	157.78
3615.51	141.57	157.70
3623.76	141.53	157.69
3627.75	141.51	157.69
3631.92	141.49	157.69
3743.67	141.13	157.69
3851.17	140.94	157.69
4067.92	140.77	157.69
4282.92	140.77	157.69
4497.92	140.88	157.69
5376.91	141.77	157.69
6494.13	143.10	157.69
11002.90	144.77	157.69
21245.21	141.30	157.69
28949.87	138.00	157.69
40522.43	133.56	157.69
59674.80	128.57	157.63
73885.95	126.10	157.51
113088.14	121.53	157.14
138686.45	119.47	156.89
199995.28	115.55	156.25

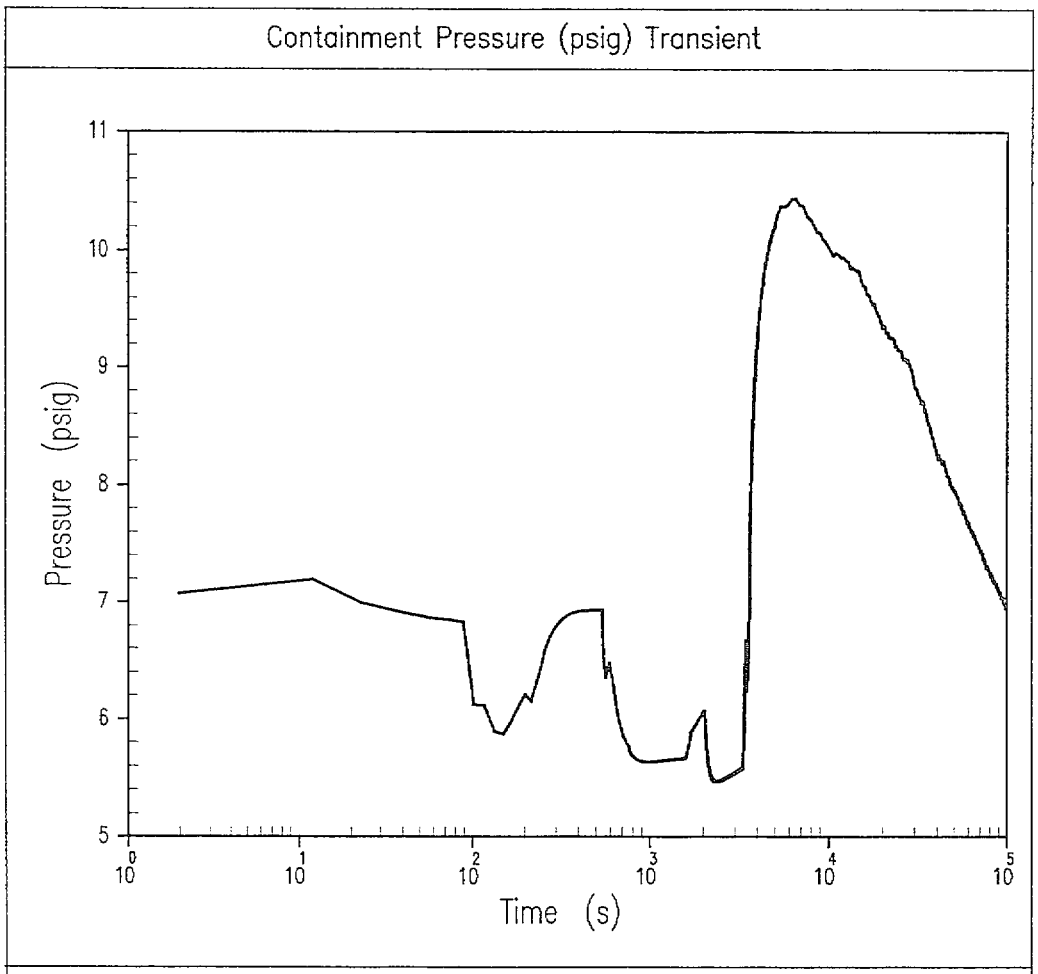
Containment Pressure (psig) Transient
Pressure (psig)
11
10
9
8
7
6
5
10^0
10^1
10^2
10^3
10^4
10^5
Time (s)

Figure 3–1

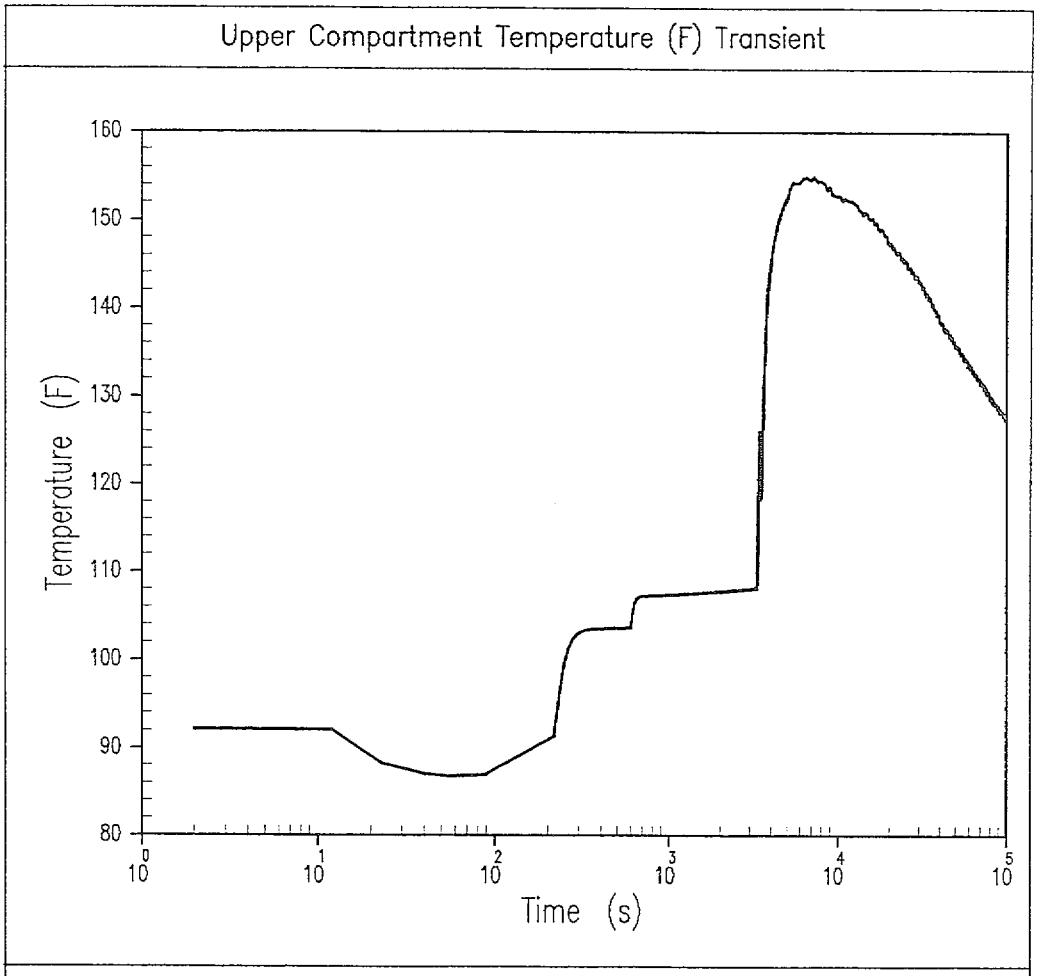
Upper Compartment Temperature (F) Transient
Temperature (F)
160
150
140
130
120
110
100
90
80
Time (s)
10^0
10^1
10^2
10^3
10^4
10^5

Figure 3–2

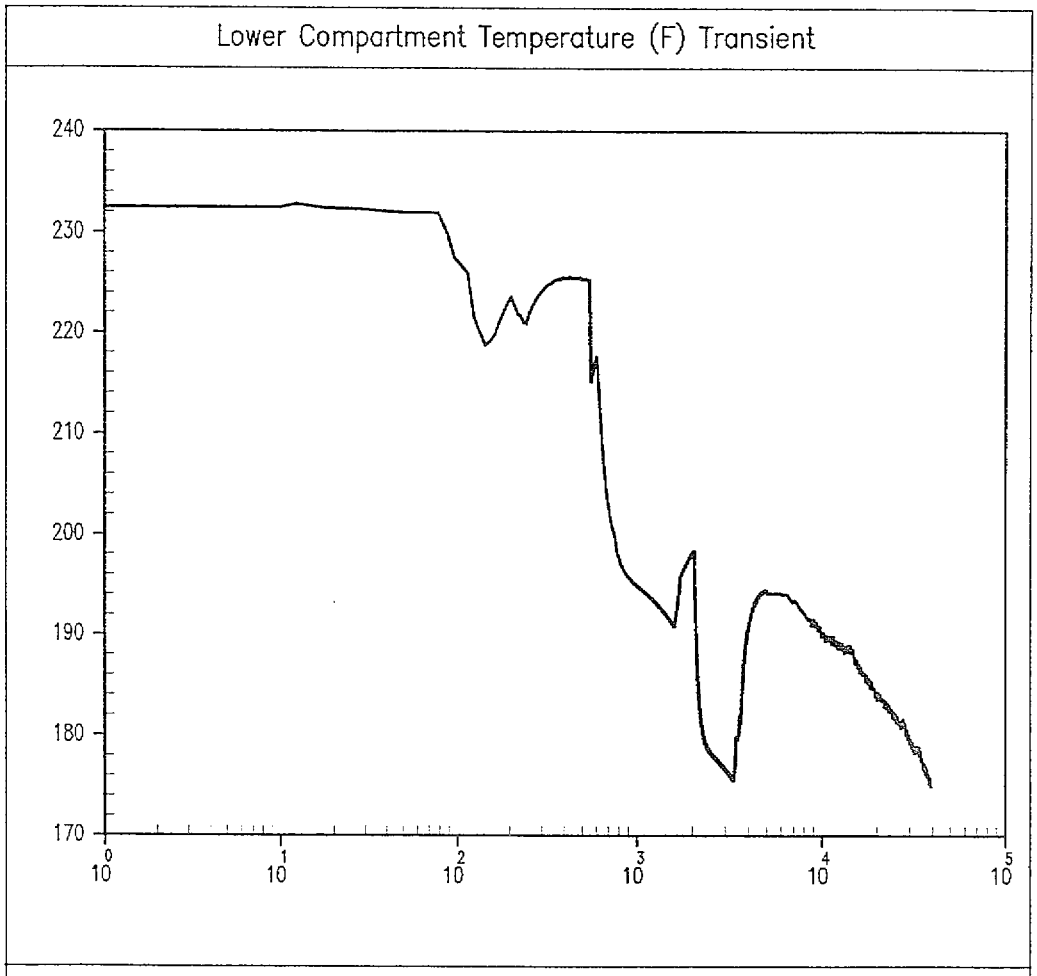
Lower Compartment Temperature (F) Transient
240
230
220
210
200
190
180
170
10^0
10^1
10^2
10^3
10^4
10^5

Figure 3–3

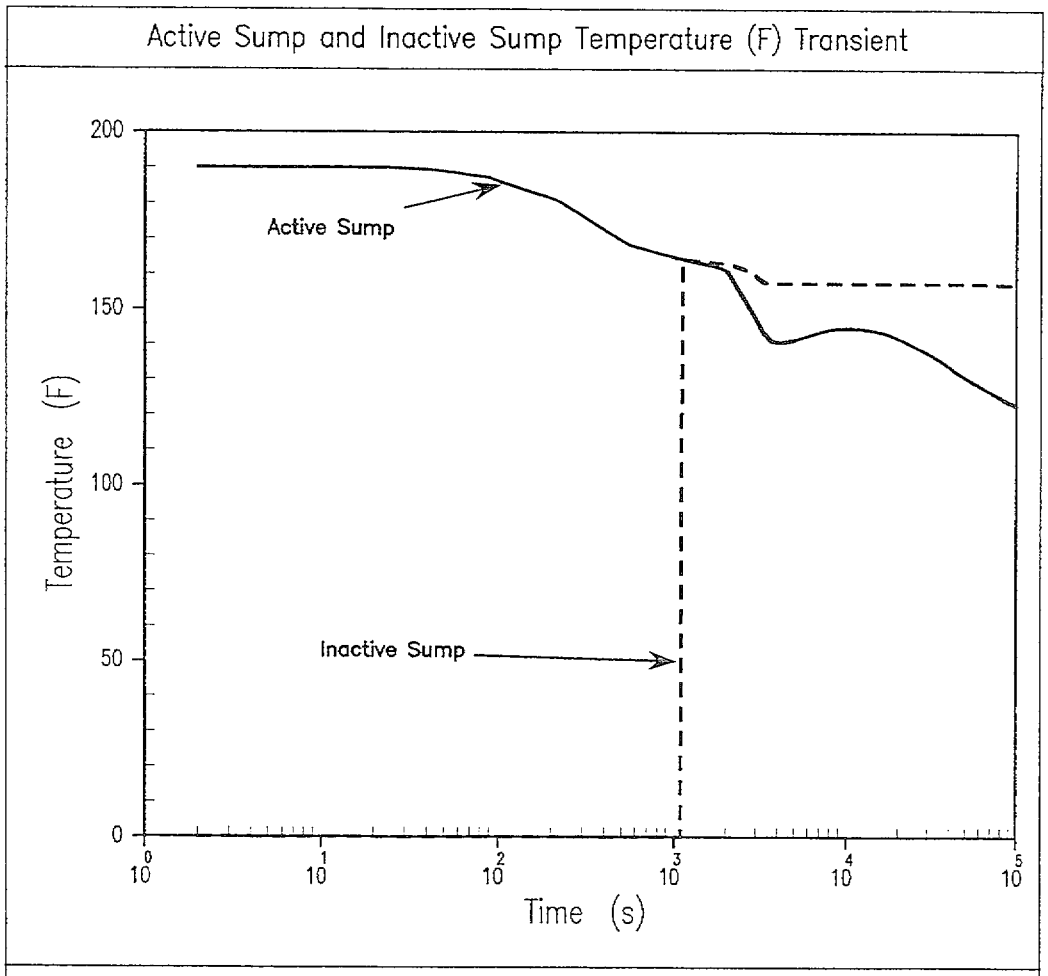
Active Sump and Inactive Sump Temperature (F) Transient
200
150
100
50
0
Temperature (F)
Active Sump
Inactive Sump
10^0
10^1
10^2
10^3
10^4
10^5
Time (s)

Figure 3–4

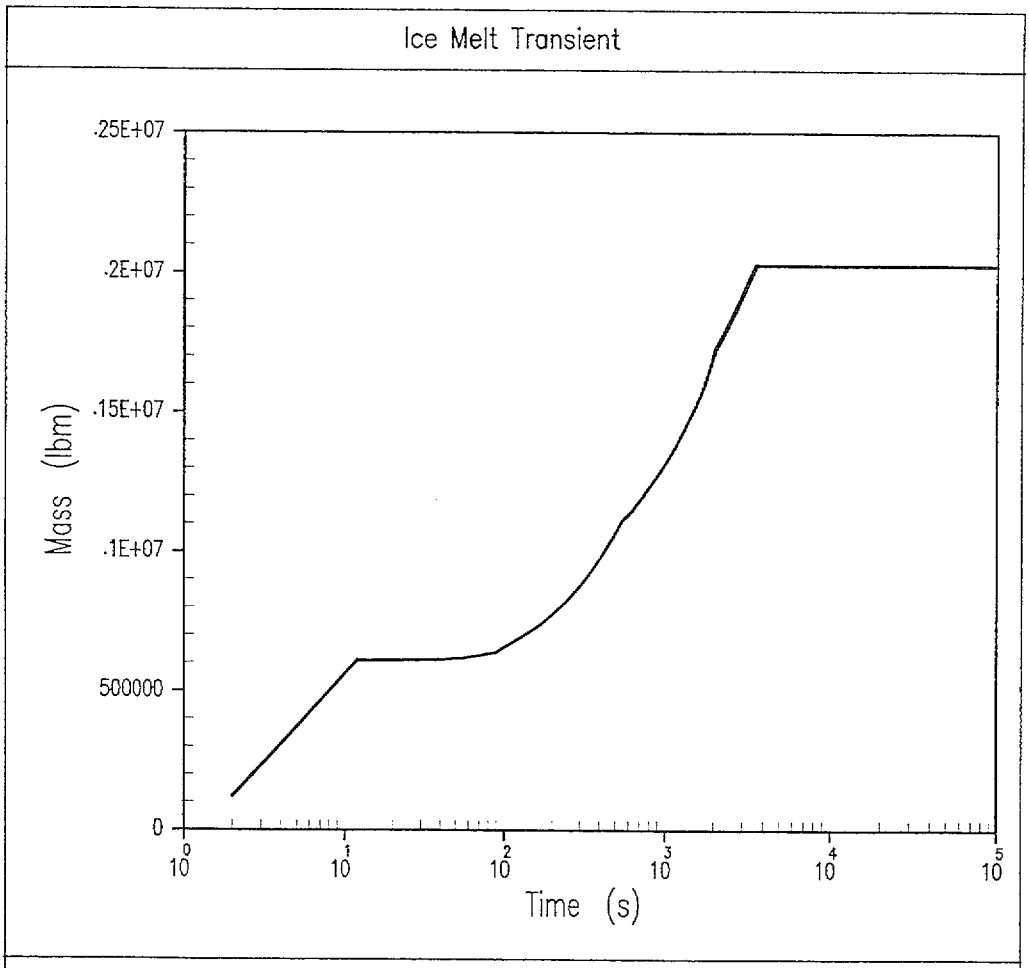
Ice Melt Transient
.25E+07
.2E+07
.15E+07
.1E+07
500000
0
Mass (lbm)
10^0
10^1
10^2
10^3
10^4
10^5
Time (s)

Figure 3–5

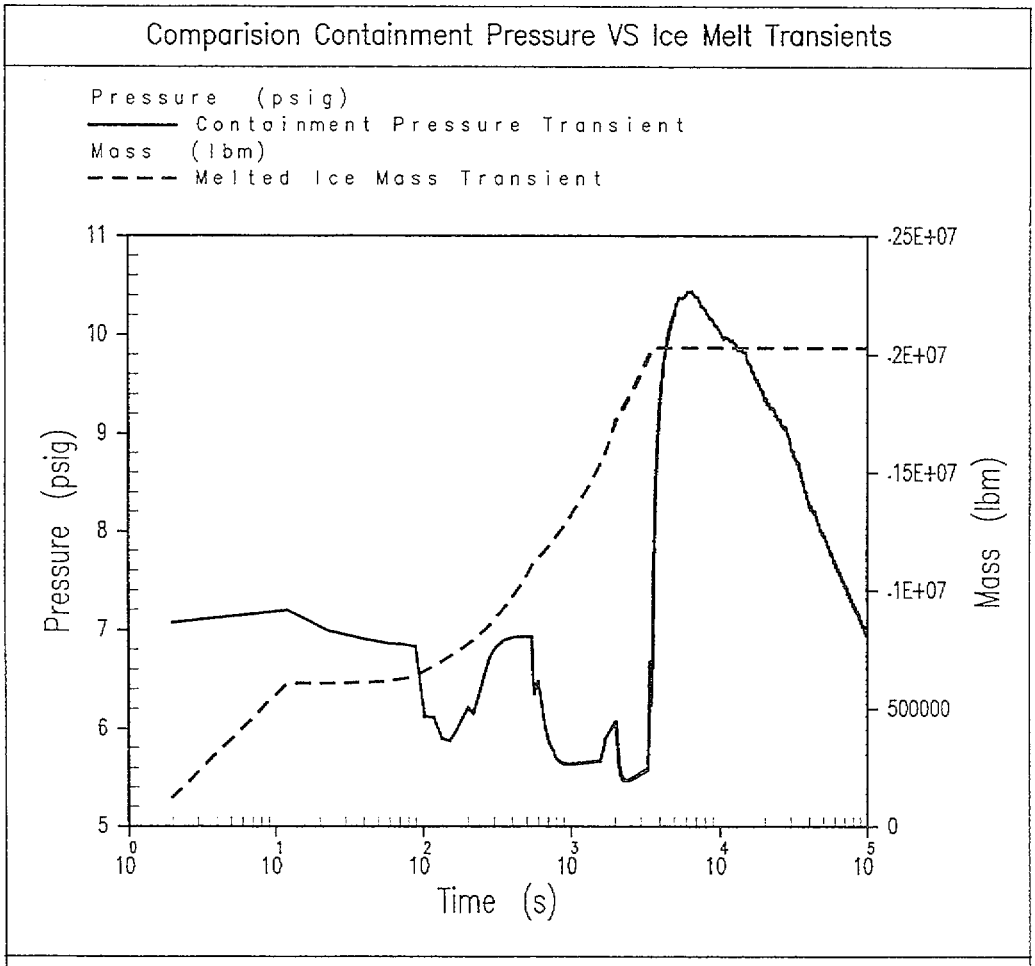
Comparision Containment Pressure VS Ice Melt Transients
Pressure (psig)
Containment Pressure Transient
Mass (lbm)
Melted Ice Mass Transient
Pressure (psig)
Mass (lbm)
Time (s)
11
10
9
8
7
6
5
.25E+07
.2E+07
.15E+07
.1E+07
500000
0
10^0
10^1
10^2
10^3
10^4
10^5

Figure 3–6

APPENDIX A - FSAR MARKUPS

3.11.6 Loss of Heating, Ventilating, and Air-Conditioning (HVAC)

Plant locations containing safety-related equipment that need a controlled environment to perform required accident mitigation operations are served by fully redundant environmental control systems, or operator actions specified to limit minimum and maximum temperatures (see Section 9.4 for details). Such redundancy and operator actions where specified, assure that no loss of safety-related equipment occurs from a single failure of HVAC equipment provided for controlling the local environment for this equipment. Data describing controlled local environmental conditions during accidents are valid for situations in which a loss of one train of HVAC is postulated.

3.11.7 Estimated Chemical and Radiation Environment

3.11.7.1 Chemical Spray

The worst case environment (normal or post-accident) chemical composition of the containment spray was based on the following sources and assumptions:

1. Ice Condenser
2. Boron Injection Tank
3. Cold Leg Injection Accumulators (4 tanks)
4. Refueling Water Storage Tank
5. Reactor Coolant System

The following assumptions were used in this analysis:

1. Calculations based on maximum pipe/tank volumes and boron concentrations and on minimum ice mass and sodium tetraborate concentration.
2. All solutions including completely melted ice mix completely.
3. Mass ratio of NaOH to boron in the ice is 1.85.
4. Density of borated water is equal to that of water.
5. Fission products, corrosion products, etc., will be neglected.

greater than

Results - The sources stated above yield a mixture of boric acid and sodium tetraborate with a pH

7.5

The locations where dose rates were calculated were chosen to conservatively calculate the dose rates in corridors, outside equipment cubicles, in adjacent rooms, and within the equipment cubicles. These dose rates were then integrated to determine equipment exposure for a 100-day period after the accident. Airborne activity in the Auxiliary Building is due to gaseous leakage from the containment which is processed and exhausted through HEPA and charcoal filters in the Auxiliary Building gas treatment system (ABGTS). The dose rates through the Reactor Shield Building from activity released into the containment atmosphere were also calculated.

Radiation exposure due to a design basis FHA is due to airborne activity and shine from the affected spent fuel bundle and affects the refueling floor and the ABGTS room. Dose rates were calculated at a single position on the refueling floor and at several locations from the ABGTS filters. These dose rates were then integrated to determine equipment exposure for a 100-day period after the FHA.

The calculation of radiation conditions outside containment in the Auxiliary Building complies with Paragraph 1.4 of NUREG-0588.

REFERENCES

1. September 30, 1986, Letter from R. Gridley to B. Youngblood, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants - Summary Status Report - Watts Bar Nuclear Plant - Unit 1".

2. Watts Bar Design Criteria, WB-DC-40-54, "Environmental Qualification to 10 CFR 50.49," Revision 2.

3. NUREG 0588, Interim Staff Report on Environmental Qualification of Safety-Related Electrical Equipment, Revision 1, July 1, 1981.

4. Watts Bar Design Criteria, WB-DC-40-42, Revision 2,"Environmental Design".

5. ~~Westinghouse letter WAT-D-10359, dated July 7, 1997~~ Reference Deleted

6. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. I-III, NUREG/CR-0200, Revision 5 (ORNL/NUREG/CSD-2/R5), March 1997.

The ice in the ice condenser is borated by adding sodium tetraborate to the ice. The aqueous solution resulting from the melted ice has a nominal boron concentration of ≥1800 ppm. In the event of an accident, this solution would be delivered to the containment sump. Containment sump pH is also controlled by the sodium tetraborate in the ice. The pH of the ice is maintained between 9.0 and 9.5, which results in a sump pH of approximately ~~8.0~~ 7.5.

Information concerning hydrogen release by the corrosion of containment metals and the control of the hydrogen and combustible gas concentrations within the containment following a LOCA is discussed in Section 6.2.5.

6.1.2 Organic Materials

For paints and coatings inside containment, the conformance with Regulatory Guide 1.54 is described in Section 6.1.4.

Organic materials within the primary containment are identified and quantified according to the following categories: electrical insulation, surface coatings, miscellaneous ALARA catch containment and shielding, ice condenser equipment, and identification tags for valves and instruments. There is no wood or asphalt inside the containment. The effects of elastomers and plastics on hydrogen generation have been evaluated and determined to be inconsequential. Therefore, the quantities identified below are considered historical and need not be revised due to design changes.

The information in this section is based on a single reactor unit.

6.1.2.1 Electrical Insulation

The typical types of electrical cable insulation/jacket material that are utilized within the primary containment are: silicon rubber, polyethylene, ethylene rubber, chlorosulfonated polyethylene, polyolefin, cross linked polyethylene, kapton. These materials are not significant contributors to hydrogen generation during a design basis accident (approximately 28,000 lbs).

6.1.2.2 Surface Coatings

Material	Mass, lbs
Concrete Surfaces	
Epoxy	2070
Phenolic-epoxy	300
Steel Surfaces:	
Phenolic epoxy	1810

6.1.3.2 Lithium Hydroxide

Lithium Hydroxide at a maximum concentration of 7.6 ppm lithium is found in the reactor coolant system for pH control.

assumed in the post-LOCA sump pH analysis

6.1.3.3 Sodium Tetraborate

7.5

Sodium tetraborate is an additive in the ice stored in the ice condenser for the purpose of maintaining containment sump pH of at least 8.0 after all the ice has melted.

1.75

The minimum ~~analysis~~ amount of ice ~~in storage~~ is 2.3 x 10^6 lbm. Boric acid and NaOH are formed during ice melt following a LOCA according to the following equation:

$$Na_2B_4O_7 + 7H_2O \rightarrow 2NaOH + 4H_3BO_3$$

6.1.3.4 Final Post-Accident Chemistry

~~In the event of an accident, the final soluble acid and soluble base concentrations for a mixture of all containment and core cooling solutions have been calculated to be 6.79 x 10^5 gram moles (boric acid equivalent) and 7.6 x 10^4 moles (sodium hydroxide equivalent), respectively. These calculations are based on the acid and base inventories of boric acid, and sodium tetraborate.~~

greater than 7.5

The final post-accident sump pH is ~~approximately 8.0~~. The estimated sump pH versus time calculation indicates that ~~within a few minutes after the LOCA, the sump pH is within a range of 7.5 to 8.5 and is within this range when the pH reaches the final estimated value of 8.0.~~

6.1.4 Degree of Compliance with Regulatory Guide 1.54 for Paints and Coatings Inside Containment

TVA is committed to adhere to Appendix B of 10 CFR 50 and ANSI N45.2 as required to produce a quality end product. Basically, it is TVA's position that the Quality Assurance Program (QA) for protective coatings inside the containment should control four activities in the coating program. The four major areas to be controlled are:

(1) The coating material itself, by extending requirements on the manufacturing process and qualification of coating systems through the use of applicable portions of ANSI Standards N101.2 and N512.

(2) The preparation of the surface to which coatings are to be applied.

the post-LOCA sump pH remains within the allowable range of 7.5 to 10.0 for the duration of the event.

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Design Bases

6.2.1.1.1 Primary Containment Design Bases

The containment is designed to assure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis accident conditions. For purposes of integrity, the containment may be considered as the containment vessel and containment isolation system. This structure and system are directly relied upon to maintain containment integrity. The emergency gas treatment system and Reactor Building function to keep out-leakage minimal (the Reactor Building also serves as a protective structure), but are not factors in determining the design leak rate.

The containment is specifically designed to meet the intent of the applicable General Design Criteria listed in Section 3.1. This section, Chapter 3, and other portions of Chapter 6 present information showing conformance of design of the containment and related systems to these criteria.

The ice condenser is designed to limit the containment pressure below the design pressure for all reactor coolant pipe break sizes up to and including a double-ended severance. Characterizing the performance of the ice condenser requires consideration of the rate of addition of mass and energy to the containment as well as the total amounts of mass and energy added. Analyses have shown that the accident which produces the highest blowdown rate into a condenser containment will result in the maximum containment pressure rise; that accident is the double-ended guillotine or split severance of a reactor coolant pipe. The design basis accident for containment analysis based on sensitivity studies is therefore the double-ended guillotine severance of a reactor coolant pipe at the reactor coolant pump suction. Post-blowdown energy releases can also be accommodated without exceeding containment design pressure.

The functional design of the containment is based upon the following accident input source term assumptions and conditions:

1. The design basis blowdown energy of ~~318~~ 316 x 10^6 Btu and mass of ~~493~~ 504 x 10^3 lb put into the containment (See Section 6.2.1.3.6).
2. A ~~reactor~~ Core power of ~~3579~~ 3459 MWt (plus ~~2%~~ 0.6% allowance for calorimetric error) (See Section 6.2.1.3.6).

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

For the purpose of calculation, the condensation of steam is assumed to take place in a condensing node located between the two control volumes in the ice storage compartment.

Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Watts Bar Nuclear Plant containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

2. 2.029375 ~~2.125~~ x 10^6 lbs. of ice initially in the ice condenser which is at 15°F. (This is less than the Technical Specification limit.)

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 6.2.1.3.6 were used.

4. Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F are used.[5]

5. Nitrogen from the accumulators in the amount of 2251 ~~2218~~ lbs. included in the calculations.

6. Essential raw cooling water temperature of 85°F is used on the spray heat exchanger and the component cooling heat exchanger.

7. The air return fan is effective 10 minutes after the transient is initiated. The actual air return fan initiation can take place in 9 ± 1 minutes, with initiation as early as 8 minutes not adversely affecting the analysis results.

8. No maldistribution of steam flow to the ice bed is assumed.

9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

10. The initial conditions in the containment are a temperature of 100°F in the lower and dead-ended volumes and a temperature of 85°F in the upper volume. All volumes are at a pressure of 0.3 psig and a 10% relative humidity.

11. A containment spray pump flow of 4000 gpm is used in the upper compartment. A diesel loading sequence for the containment sprays to energize and come up to full flow and head in 135 seconds has been used in this analysis. This initial time sequence modification was made to ensure that a frequency transient did not occur for a simultaneous LOCA and loss of offsite power (LOOP) as desired by NRC Regulatory Guide 1.9, Section C4. Subsequent analysis has changed the loading sequence to 221 seconds. However, this did not significantly affect the results obtained with the 135-second delay. It is also noted that the calculated CSS flow rate is 4550 gpm, which bounds the 4000 gpm flow rate used in the analysis, ~~and, being conservative, offsets any effect due to the sequence delay change~~.

221

12. A residual spray (~~2000~~ 475 gpm) is used starting 1 hour after the transient is initiated. The residual heat removal pump and spray pump take suction from the sump during recirculation.

The minimum time at which the RHR pumps can be diverted to the RHR sprays is specified in the plant operating procedures as one hour after the accident. A discussion of the core cooling capability of the emergency core cooling system is given in Section 6.3.1 for this mode of operation.

13. Containment structural heat sink data is found in Table 6.2.1-1.

14. The operation of one containment spray heat exchanger (UA = ~~2.446~~ 2.74 x 10^6 Btu/hr-°F) for containment cooling and the operation of one RHR heat exchanger (UA = ~~1.61~~ 1.57 x 10^6 Btu/hr-°F) for core cooling.

15. The air return fan returns air at a rate of 40,000 cfm from the upper to lower compartment.

16. An active sump volume of 51000 ft^3 is used.

17. The pump flowrates vs. time given in Table 6.2.1-2 were used. (These flow values reflect ECCS pumps at runout against the design containment pressure, using the minimum composite pump curves shown in Figures 6.3-2, 6.3-3, and 6.3-4, which are degraded by 5% and bound what is achievable in the plant. Switchover times from injection to recirculation that are achievable in the plant for each ECCS pump are also conservative in the analysis.)

18. A power rating of 100.6% of licensed power (3475 MWt)* is assumed, but not explicitly modeled. [Decay heat is based on a reactor power of 3579 MWt (+2%) for mass and energy release computations. See Section 6.2.1.3.6.]

*The additional 16 MWt is due to the contribution of heat to the primary coolant system from non-reactor sources, primarily reactor coolant pump heat.

INSERT 1

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design.

The following plots are provided:

Figure 6.2.1-1, Containment Pressure Transient,

Figure 6.2.1-2, Upper and Lower Compartment Temperature Transients,

Figure 6.2.1-3, Active and Inactive Sump Temperature Transient,

Figure 6.2.1-4, Ice Melt Transient.

Tables 6.2.1-3 and 6.2.1-4 give energy accountings at various points in the transient.

As can be seen from Figure 6.2.1-1 the maximum calculated Containment pressure is ~~11.21~~ 10.438 psig, occurring at approximately ~~2600.9~~ 6873.5 seconds. The transient shown does ~~not~~ account for hydrogen partial pressure as a result of the post DBA LOCA hydrogen production discussed in Sections 6.2.5 and 15.4.1.2. Accumulation of hydrogen prior to recombiner operation can account for approximately 0.25 psig at the time of containment peak pressure~~, for a total of 11.46 psig~~.

Also, a sensitivity study was performed varying the ice mass to determine the approximate minimum ice mass necessary for the reactor coolant pump suction pipe rupture case with the steam generators considered as the active heat source. These results are presented in Figure 6.2.1-4A.

Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference forms accounts for transient conduction into and out of the node and temperature rise of the node. Table 6.2.1-1 is a summary of the containment structural heat sinks used in the analysis. The material property data used is found in Table 6.2.1-5.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami. An explanation of the manner of application is given in Reference [3].

INSERT-1

19. Hydrogen gas was added to the containment in the amount of 24,051 Standard Cubic Feet (SCF) over 24 hours. Sources accounted for were radiolysis in the core and sump post-LOCA, corrosion of plant materials (Aluminum, Zinc, and painted surfaces found in containment), reaction of 1% of the Zirconium fuel rod cladding in the core, and hydrogen gas assumed to be dissolved in the Reactor Coolant System water.(This bounds tritium producing core designs)

6.2.1.3.6 Mass and Energy Release Data

Long-Term Mass and Energy Releases

Following a postulated rupture of the reactor coolant system (RCS), steam and water is released into the containment system. Initially the water in the RCS is sub-cooled at a high pressure. When the break occurs, the water passes through the break where a portion flashes to steam at the lower pressure of the containment. These releases continue until the RCS depressurizes to the pressure in the containment (end of blowdown).

At that time, the vessel is refilled by water from the accumulators and safety injection (SI) pumps. The analysis assumes that the lower plenum is filled with saturated water at the end of blowdown, to maximize steam releases to the containment. Therefore, the water flowing from the accumulators and SI pumps starts to fill the downcomer causing a driving head across the vessel which forces water into the hot core.

During the reflood phase of the accident water enters the core where a portion is converted to steam which entrains an amount of water into the hot legs at a high velocity. Water continues to enter the core and release the stored energy of the fuel and clad as the mixture height in the core increases. When the level, two feet below the top of the core, is reached the core is assumed to be totally quenched which leaves only decay heat to generate steam. This type of break is analyzed at three locations.

The location of the break can significantly change the reflood transient. It is for this reason that the (1) hot leg, (2) pump suction, and (3) cold leg break locations are analyzed. For a cold leg break, all of the fluid which leaves the core must vent through a steam generator and becomes superheated. However, relative to breaks at other locations, the core flooding rate (and therefore the rate of fluid leaving the core) is low because all the core vent paths include the resistance of the reactor coolant pump. For a hot leg pipe break the vent path resistance is relatively low, which results in a high core flooding rate, but the majority of the fluid which exits the core bypasses the steam generators in venting to the containment. The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and steam generator heat addition as in the cold leg break. As a result, the pump suction breaks yield the highest energy flow rates during the post blowdown period. The spectrum of breaks analyzed includes the largest cold and hot leg breaks, reactor inlet and outlet respectively, and a range of pump suction breaks from the largest to 3.0 ft^2. Because of the phenomena of reflood as discussed above, the pump suction break location is the worst case. This conclusion is supported by studies of smaller hot leg breaks which have been shown, on similar plants, to be less severe than the double ended hot leg. Cold leg breaks, however, are lower both in the blowdown peak and in the reflood pressure rise. Thus an analysis of smaller pump suction breaks is representative of the spectrum of break sizes.

As a result, the Reference 9 methodology only requires analysis of a double-ended pump suction break with minimum ECCS flows.

The LOCA analysis calculational model is typically divided into three phases which are: 1) blowdown, which includes the period from accident occurrence (when the reactor is at steady state full power operation) to the time when zero break flow is first calculated, 2) refill, which is from the end of blowdown to the time the ECCS fills the vessel lower plenum, and 3) reflood, which begins when water starts moving into the core and continues until the end of the transient. For the pump suction break, consideration is given to a possible fourth phase; that is, froth boiling in the steam generator tubes after the core has been quenched. For a description of the calculational model used for the mass and energy release analysis[9]. As per this model the flowsplit is assumed to be 100% at ~~1765 seconds for maximum safeguards and 1637~~ seconds for minimum safeguards. 2042

Basis of the Analysis

1. Assumptions

The following items ensure that the core energy release is conservatively analyzed for maximum containment pressure.

a. Maximum expected operating temperature (~~619.1~~ 625.1°F)

b. Allowance in temperature for instrument error and dead band (+~~4~~ 6°F)

c. Margin in volume (1.4%)

d. Allowance in volume for thermal expansion (1.6%)

e. ~~Margin in core power associated with use of engineered safeguards design rating (ESDR)~~

f. Allowance for calorimetric error (0.6% of ESDR)

g. Conservatively modified coefficients of heat transfer

h. Allowance in core stored energy for effect of fuel densification

~~i. Margin in core stored energy (+20%).~~

2. Initial Conditions (plus 0.6% allowance for calorimetric error)

Core Power (License Application) (MWt)	3459
~~Engineered Safeguards Design Rating (ESDR) (MWt)~~	~~3579~~
Vessel/Core Inlet Temperature (T_c) ~~(plus 0.6% allowance for calorimetric error) (°F)~~	~~557.3~~ 563.3
Vessel Average Temperature (T_{avg}) (°F)	~~588.2~~ 594.2
Vessel Outlet Temperature (T_h) (°F)	~~619.1~~ 625.3
Steam Pressure (psia)	980
Rod Array	17x17
Total Accumulator Mass (lbm)	~~210,300~~ 276,057
Accumulator Temperature (°F)	~~120~~ 130
Accumulator Pressure (psia)	600
Assumed Containment Reference Pressure (psia)	~~26.7~~ 28.2
Pumped Injection (assumed)	
Minimum (ft^3/sec)	~~10.8~~ 10.668
~~Maximum (ft^3/sec)~~	~~22.4~~
Recirculation Time (assumed) (sec)	~~1455~~ 1631

These initial conditions are based on plant operation with a 0% level of steam generator tube plugging (SGTP). An evaluation of the initial blowdown energy for conditions based on 10% SGTP and an additional 2% reduction in thermal design flow (RTDF) indicates that the energy can increase slightly, for example, as much as 2.16 x 10^6 Btu higher than the conditions for the 0% SGTP. However, this increase is small and can be offset by analysis margins, such as the margin that exists between the decay heat model in ANSI/ANS-5.1-1979 and the model used in Reference [9]. Thus, the LOCA mass and energy release analysis for the 0% SGTP conditions remains applicable for the 10% SGTP and an additional 2% RTDF conditions.

Long-Term Mass and Energy Release Data

Blowdown Results

Table 6.2.1-~~15~~ 16 ~~lists~~ Provides the calculated mass and energy releases for the double-ended pump suction blowdown phase of the LOCA. ~~various breaks analyzed, with the corresponding break size.~~

Reflood Results

~~Table 6.2.1-17 presents the hydraulic parameters used for the reflood analysis. Figures 6.2.1-21 through 6.2.1-25 present the core inlet temperature, the core flooding rate, the carry over fraction, the fraction of flow through the broken loop, and the core and downcomer water levels, respectively, for the double-ended pump suction guillotine with minimum safeguards safety injection.~~ Table 6.2.1-18 lists the table numbers for the calculated mass and energy releases for the reflood chase of the various breaks analyzed along with the corresponding safeguards assumption (maximum or minimum).

Two-Phase Post-Reflood Results

A single Two froth analyses were performed: ~~a double-ended pump suction (DEPS) guillotine break with maximum safeguards SI flow, and~~ a DEPS break with minimum safeguards SI flow. For ~~both~~ this cases the release rates are based on a reference temperature for heat stored in the steam generator secondary fluid equal to saturation temperature corresponding to reference pressure of 20.2 psia. ~~The table below presents a summary of the available secondary side energy for the broken loop and intact loop for both cases.~~

The heat content of the broken and unbroken steam, generators as a function of time is shown in Figure 6.2.1-26 for the DEPS guillotine break minimum safeguards case.

	Case 1	Case 2
Break	DEPS	DEPS
SI Assumption	Maximum	Minimum
Available Energy of Secondary Mass for Broken Loop Steam Generator (10^6 Btu)	14.9*	16.3*
Available Energy of Secondary Mass for Intact Loop Steam Generators (10^6 Btu)	179.0*	179.7*
Total Available Steam Generator Energy	193.9*	196.0*

* Referenced to 228.0°F.

Tables ~~6.2.1-20 and~~ 6.2.1-~~21~~ 19 present the calculated mass and energy release rate data for a DEPS break using ~~maximum and~~ minimum safeguards assumptions~~, respectively. These tables~~ completely replace the mass and energy release data after the end of 10-foot entrainment occurs ~~(see Tables 6.2.1-19a and 6.2.1-19b).~~

~~Depressurization Energy Release~~

~~The froth mass and energy release data presented in Tables 6.2.1-20 and 6.2.1-21 are based on a reference temperature for heat stored in the steam generator metal and secondary fluid of saturation at assumed containment back pressure (20.2 psia) up to the time at which the broken loop steam generator equilibrates.~~

Since the containment pressure remains above this value until after the time of peak pressure, depressurization energy release need not be calculated until peak pressure has occurred and the pressure returns to 20.2 psia. At this point the energy remaining in the system, presented in Table 6.2.1-22, can be added to the decay heat release by using the equation below:

$$\dot{q} = \frac{q_{total} \bullet \Delta T/\Delta t}{\Delta T_{total}}$$

$\dot{q}$ = heat release rate (Btu/sec)

q_{total} = total available heat from Table 6.2.1-22 (Btu)

$\Delta T/t$ = rate of temperature change (°F/sec)

ΔT_{total} = initial temperature differential (16°F)

Short-Term Mass and Energy Releases

The short-term mass and energy release models and assumptions are described in Reference [9]. The LOCA short-term mass and energy release data used to perform the containment analysis given in Sections 6.2.1.3.4 and 6.2.1.3.9 are listed below:

Section	Break Size and Location	Table
6.2.1.3.4	Double-Ended Cold Leg Guillotine Break Outside the Biological Shield	6.2.1-23
6.2.1.3.4	Double-Ended Hot Leg Guillotine Break Outside the Biological Shield	6.2.1-24
6.2.1.3.9	Double-Ended Pressurizer Spray Line Break	6.2.1-28
6.2.1.3.9	127 in^2 Cold Leg Break at the Reactor Vessel	6.2.1-30

6.2.1.3.7 Accident Chronology

For a double-ended pump suction loss-of-coolant accident, the major events and their time of occurrence are shown in Table 6.2.1-25 for the minimum safeguards case.

6.2.1.3.8 Energy Balance Tables

Table~~s~~ 6.2.1-26~~a through 6.2.1-26f~~ gives the initial energy distribution as well as the energy distribution at end of blowdown and end of reflood for the double-ended pump suction break ~~various break locations and sizes. The release rate transients for this case are consistent with the 10 foot entrainment calculation~~.

6.2.1.3.9 Containment Pressure Differentials

Consideration is given in the design of the containment internal structures to localized pressure pulses that could occur following a loss-of-coolant accident. If a loss-of-coolant accident were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the structures.

These subcompartments include the steam generator enclosure, pressurizer enclosure, and upper and lower reactor cavity. Each compartment is designed for the largest blowdown flow resulting from the severance of the largest connecting pipe within the enclosure or the blowdown flow into the enclosure from a break in an adjacent region.

The following paragraphs summarize the design basis calculations:

Steam Generator Enclosure

The worst break possible in the steam generator enclosure is a double-ended rupture of the steamline pipe at no load conditions. Based on an investigation of postulated break locations, the rupture is assumed to occur at the point where the steamline exits the steam generator. The blowdown for this break is given in Table 6.2.1-27a. The TMD computer code using the compressibility factor and assuming unaugmented critical flow is used to calculate the short-term pressure transients. The nodalization of the steam generator enclosure where the break occurs is shown in Figure 6.2.1-81. Node 51 is the break element and has a flow path to the adjacent steam generator enclosure which is a mirror image of the enclosure where the break occurs. Both enclosures are nodalized in the same manner; their nodal network is shown in Figure 6.2.1-82 and their input data is given in Tables 6.2.1-27b and 6.2.1-27c. This input data assumes that the insulation remains intact. The loss coefficients were computed using Reference [12]. The maximum number of nodes used is based on the geometry of the system. The steam generator compartment is essentially symmetrical with no major obstructions to flow which would introduce asymmetric pressures. In addition, the flow path to the adjacent steam generator is at the top of the enclosure. Therefore, a significant differential pressure will not occur across the steam generator vessel. The balance of plant data is similar to that presented in Section 6.2.1.3.4.

2. A quantity of 2.125×10^6 lbs of ice is assumed for the DER cases, and 2.025×10^6 lbs of ice is conservatively assumed for the small split cases, to be initially in the ice condenser.

3. The boron injection tank remains installed without heat tracing, and the boric acid concentration is reduced to zero ppm (Table 6.2.1-40).

4. The air return fan is effective 10 minutes after the transient is initiated. Actual air return fan initiation can take place in 9±1 minutes. Initiation as early as 8 minutes does not adversely affect the outcome of the analysis.

5. A uniform distribution of steam flow into the ice bed is assumed.

6. The initial conditions in the containment are a temperature of 120°F in the lower compartment, ~~120°F~~ 100°F in the dead-ended compartment, a temperature of 85°F in the upper compartment, and a temperature of 15°F in the ice condenser. All volumes are at a pressure of 0.3 psig (see Table 6.2.1-13).

7. A containment spray pump flow of 4,030 gpm is conservatively used in the upper compartment. A diesel loading sequence for the containment sprays to energize and come up to full flow and head in 135 seconds was used in the analysis. As discussed in the Section 6.2.1.3.2 list of assumptions, subsequent analysis has changed the loading sequence to 221 seconds. However, this does not significantly affect the results obtained with the 135 second delay time utilized. It is also noted that the calculated CSS flow rate is at least 4,300 gpm (in injection mode with low RWST water level), which bounds the 4,030 gpm flow rate used in the analysis and, being conservative, offsets any effect due to the loading sequence delay change.

8. Containment structural heat sinks as presented in Table 6.2.1-1 were used. The material properties are given in Table 6.2.1-5.

9. The air return fan empties air at a rate of 40,000 ft^3/min from the upper to the lower compartments. The total calculated air flow rate discharged to the dead-end compartment used is 41,885 cfm and is, therefore, bounded.

10. A series of large break cases (1.4 - 4.6 ft^2 double-ended ruptures) were run to determine the limiting large break case (Table 6.2.1-41). In addition, a series of small breaks were analyzed with LOTIC at the 30% power level (Table 6.2.1-42).

11. The mass and energy releases for the limiting breaks are given in Table 6.2.1-39. Since these rates are considerably less than the RCS double-ended breaks and their total integrated energy is not sufficient to cause ice bed melt out, the containment pressure transients generated for the RCS breaks will be more severe. However, since the steam line break blowdowns are superheated, the lower compartment temperature transients calculated in this analysis will be limited. These temperature transients are given in Figures 6.2.1-69 through 6.2.1-74.

The LOCA Sec. mass & 2,077,375 lbm would still be adequate for main steam line breaks, since these breaks only melt ~~about~~ about 1.054 x 10^6 lbm of ice.

SUBSCRIPT

a	Air
as	Air and steam
c	Suspended or entrained water
e	Energy
i	i-th compartment
j	j-th compartment
ij	from i-th compartment to j-th compartment
s	Steam

REFERENCES

1. Grimm, N. P., Colenbrander, B. G. C., "Long Term Ice Condenser, Containment Codes - LOTIC Code", WCAP-8354-P-A (Proprietary), ~~July 1974~~, and WCAP-8355-A (Non-Proprietary), ~~July 1974~~. APRIL 1976

2. "Final Report Ice Condenser Full Scale Section Test at the Waltz Mill Facility", WCAP-8282 (Proprietary), February 1974, WCAP-8211 and Appendix (Non-Proprietary), May 1974.

3. Hsieh, T., and Raymond, M., "Long Term Ice Condenser Containment Code - LOTIC Code", WCAP-8354-P-A Supplement 1 (Proprietary), June 1975, and WCAP-8355-A Supplement 1 (Non-Proprietary), June 1975. Hsieh, T., and Liparulo, N. J., "Westinghouse Long Term Ice Condenser Containment Code - LOTIC-III Code," WCAP-8354-P-A Supplement 2 (Proprietary), February 1979.

4. Salvatori, R. (approved), "Ice Condenser Containment Pressure Transient Analysis Method," WCAP-8078, March 1973.

5. Salvatori, R. (approved), "Ice Condenser Full-Scale Section Test at the Waltz Mill Facility," WCAP-8110, Supplement 6, May 1974.

6. Deleted by FSAR Amendment 85.

7. Deleted by FSAR Amendment 85.

8. Deleted by FSAR Amendment 85.

9. ~~R. M. Shepard, et al, "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8312-A, August 1975.~~

10. Deleted by FSAR Amendment 85.

9. WCAP-10325-A "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983

WBNP-1
TABLE 6.2.1-1
(Sheet 1 of 2)

STRUCTURAL HEAT SINKS

A. Upper Compartment

	Area (ft^2)	Thickness (ft)	
1. Operating Deck			
Slab 1	4880	1.1	Concrete
Slab 2	18280	.0005	Paint
		1.4	Concrete
Slab 3	760	.00125	Paint
		1.5	Concrete
Slab 4	3840	.0208	Stainless Steel
		1.5	Concrete
2. Shell and Misc			
Slab 5	56331	.000625	Paint
		.08	Steel

B. Lower Compartment

1. Operating Deck, Crane Wall, and Interior Concrete			
Slab 6	31963	1.43	Concrete
2. Operating Deck			
Slab 7	2830	.00125	Paint
		1.0	Concrete
Slab 8	760	.0005	Paint
		1.75	Concrete
3. Interior Concrete and Stainless Steel			
Slab 9	2770	.021	Stainless Steel
		2.0	Concrete

Replace

WBNP-0
TABLE 6.2.1-1
(Sheet 2 of 2)

STRUCTURAL HEAT SINKS (Cont'd)

B. Lower Compartment (Cont'd)

	Area (ft^2)	Thickness (ft)	
4. Floor*			
Slab 10	15921	.0005	Paint
		1.6	Concrete
2. Misc Steel			
Slab 11	28500	.000625	Paint
		.066	Steel

C. Ice Condenser

1. Ice Baskets			
Slab 12	180,628	.00663	Steel
2. Lattice Frames			
Slab 13	76,650	.0217	Steel
3. Lower Support Structure			
Slab 14	28,670	.0267	Steel
4. Ice Condenser Floor			
Slab 15	3,336	.000833	Paint
		.333	Concrete
5. Containment Wall Panels & Containment Shell			
Slab 16	19,100	1.0	Steel & Insulation
		.0625	Steel Shell
6. Crane Wall Panels and Crane Wall			
Slab 17	13,055	1.0	Steel & Insulation
		1.0	Concrete

*In contact with sump.

Replace

WBNP-1
TABLE 6.2.1-1
(Sheet 1 of 2)
STRUCTURAL HEAT SINKS

A. Upper Compartment

	Area (Ft^2)	Thickness (Ft)	Material
1. Operating Deck			
Slab 1	4880.	1.066	Concrete
Slab 2	18280.	0.0055	Paint
		1.4	Concrete
Slab 3	760.	0.0055	Paint
		1.5	Concrete
Slab 4	3840.	0.0208	Stainless Steel
		1.5	Concrete
2. Shell and Misc.			
Slab 5	56331.	0.001	Paint
		0.079	Steel

B. Lower Compartment

1. Operating Deck, Crane Wall, and Interior Concrete			
Slab 6	31963.	1.43	Concrete
2. Operating Deck			
Slab 7	2830.	0.0055	Paint
		1.0	Concrete
Slab 8	760	0.0055	Paint
		1.75	Concrete
3. Interior Concrete and Stainless Steel			
Slab 9	2270.	0.0208	Stainless Steel
		2.0	Concrete
4. Floor*			
Slab 10	15921.	0.0055	Paint
		1.6	Concrete
5. Misc. Steel			
Slab 11	28500.	0.001	Paint
		0.0656	Steel

* In Contact with Sump

WBNP-1
TABLE 6.2.1-1
(Sheet 2 of 2)
STRUCTURAL HEAT SINKS

C. Ice Condenser

	Area(Ft^2)	Thickness (Ft)	Material
1. Ice Baskets			
Slab 12	149600.	0.00663	Steel
2. Lattice Frames			
Slab 13	75865.	0.0217	Steel
3. Lower Support Structure			
Slab 14	28670.	0.0587	Steel
4. Ice Condenser Floor			
Slab 15	3336.	0.0055	Paint
		0.333	Concrete
5. Containment Wall Panels & Containment Shell			
Slab 16	19100.	1.0	Steel & Insulation
		0.0625	Steel Shell
6. Crane Wall Panels and Crane Wall			
Slab 17	13055.	1.0	Steel & Insulation
		1.0	Concrete

TABLE 6.2.1-2
Sheet 1 of 1

PUMP FLOW RATES VS. TIME

Time after Safeguards Initiation (sec)	SIS Flow to Core (gpm)	Spray Flow (gpm)	RHR Spray Flow (gpm)
0	0	0	0
15	460	0	0
20	1065	0	0
25	4853	0	0
135	4853	000	0
1768	4853	4000	0
1788	4853 *	4000	0
1938	3788 **	4000	0
2754	3788	4000	0
2774	3788	0	0
2894	3788	4000 **	0
3600	1078	4000	2000
End of transient	1078	4000	2000

* 3788 gpm from sump
** All flow from sump from this point until end of transient

Replace

TABLE 6.2.1-2
Sheet 1 of 1

PUMP FLOW RATES VS TIME

Time After Safeguards Initiation (Sec)	ECCS Flow To Core (RWST) (Gpm)	Spray (Flow) (Gpm)	RHR Spray (Flow) (Gpm)	ECCS Flow To Core (Sump) (Gpm)	Comments
0	0	0	0	0	"S" - Signal
11.9	0	0	0	0	
12.0	358.9	0	0	0	CC Pump Start
16.9	359.9	0	0	0	
17.0	942.3	0	0	0	SI Pump Start
21.9	942.3	0	0	0	
22.0	*4699.8	0	0	0	RHR Pump Start
190.9	4699.8	0	0	0	
191.0	4699.8	4000	0	0	Containment Spray Start
1631.2	4699.8	4000	0	0	
1631.3	4699.8	4000	0	3757.5	RHR Switchover to Sump
1708.2	4699.8	4000	0	3757.5	
1708.3	0	4000	0	3757.5	CCP/SI Pump Switchover
3326.9	0	4000	0	3757.5	
3327.0	0	0	0	3757.5	CS Pump Stopped
3446.9	0	0	0	3757.5	
3447.0	0	4000 (Sump)	0	3757.5	CS Pump Switchover
3600.0	0	4000 (Sump)	0	3757.5	
3600.1	0	4000(Sump)	1475	1855	RHR Alignment for Auxiliary CS
End of Transient	0	4000 (Sump)	1475	1855	

TABLE 6.2.1-3
Sheet 1 of 1

ENERGY BALANCES

Sink	Approx. End of Blowdown (Btu)	Approx. End of Reflood (Btu) (t=216 sec)
*Ice Heat Removal	186 (10^6)	298 (10^6)
*Structural Heat Sinks	20 (10^6)	58 (10^6)
*RHR Heat Exchanger Heat Removal	0	0
*Spray Heat Exchanger Heat Removal	0	0
Energy Content of Sump	170 (10^6)	246 (10^6)
Ice Melted	0.6 (10^6)	1.05 (10^6)

*Integrated energies, Btu

Replace

TABLE 6.2.1-3
Sheet 1 of 1

ENERGY BALANCES

Sink	Approx. End of Blowdown (t=10.0 sec.)	Approx. End of Reflood (t=245.0 sec.)
	(In Millions of Btus)	
Ice Heat Removal	188.54	241.128
Structural Heat Sinks*	19.6	62.03
RHR Heat Exchanger Heat Removal*	0	0
Spray Heat Exchanger Heat Removal*	0	0
Energy Content of Sump	208.69	282.64
Ice Melted (Pounds) (10^6)	0.6087	0.819

* Integrated Energies

WBNP-0

TABLE 6.2.1-4
Sheet 1 of 1

ENERGY BALANCES

Sink	Approx. Time of Ice Bed Melt Out (Btu) (t=2990)	Approx. Time of Peak Pressure (Btu) (t=3600.9)
*Ice Heat Removal	557 (10^6)	567 (10^6)
*Structural Heat Sinks	71.4 (10^6)	88.9 (10^6)
*RHR Heat Exchanger Heat Removal	34.7 (10^6)	48.5 (10^6)
*Spray Heat Exchanger Heat Removal	20.9 (10^6)	50.3 (10^6)
Energy Content of Sump	644 (10^6)	611 (10^6)
Ice Melted	2.125 (10^6)	2.125 (10^6)

*Integrated energies, Btu

Replace

TABLE 6.2.1-4
Sheet 1 of 1

ENERGY BALANCES

Sink	Approx. Time of Ice Melt Out (t=3559 sec.)	Approx. Time Peak Pressure (t=6244 sec.)
	(In Millions of Btus)	
Ice Heat Removal	543.34	543.34
Structural Heat Sinks*	80.01	117.45
RHR Heat Exchanger Heat Removal*	28.598	68.218
Spray Heat Exchanger Heat Removal*	3.479	69.089
Energy Content of Sump	804.87	806.41
Ice Melted (Pounds) (10^6)	2.029375	2.029375

* Integrated Energies

TABLE 6.2.1-5
Sheet 1 of 1

MATERIAL PROPERTY DATA

Material	Thermal Conductivity Btu/hr-ft -°F	Volumetric Heat Capacity Btu/ft^3-°F
Paint on Steel	.21	14.0
Paint on Concrete	.083	28.4
Concrete	.8	28.8
Stainless Steel	9.4	56.4
Carbon Steel	26.0	56.4

Replace

TABLE 6.2.1-5
Sheet 1 of 1

MATERIAL PROPERTY DATA

Material	Thermal Conductivity Btu/hr-ft-°F	Volumetric Heat Btu/ft^3-°F
Paint on Steel	0.21	19.9
Paint on Concrete	0.083	39.9
Concrete	0.8	31.9
Stainless Steel	9.4	53.68
Carbon Steel	26.0	53.9
Insulation on Steel	0.15	2.75

TABLE 6.2.1-13
(Sheet 1 of 1)

WATTS BAR ICE CONDENSER DESIGN PARAMETERS

Reactor Containment Volume (net free volume, ft^3)	
Upper Compartment	651,000
Ice Compartment	110,520
Lower Compartment	253,114
Lower Compartment (dead-ended)	129,900
Total Containment Volume	1,144,534
NSSS	
Fraction of Nominal (FON) based on NSSS Power of, MWt	3,475[1]
Analysis weight of ice in condenser, lbs (100% & 0% power DER cases)	2.125×10^6 *
Analysis weight of ice in condenser, lbs (30% power, small split cases)	2.025×10^6
Core Nuclear Power - % FON:	
100% power cases	1.006
30% power cases	0.30
0% power cases	Critical at 0.0

1. Includes RCP power (16 MWt)

* The LOCA value of 2.029375×10^6 is governing for main steamline break (MSLB), since MSLB only melts approximately half of the available ice.

TABLE 6.2.1-15
(Sheet 1 of 1)

BLOWDOWN DATA SUMMARY

Breaks	Tables
Double Ended Pump Suction	6.2.1-16a
~~.6 Double Ended Pump Suction~~	~~6.2.1-16b~~
~~3 ft^2 Pump Suction Split~~	~~6.2.1-16c~~
~~Double Ended Hot Leg~~	~~6.2.1-16d~~
~~Double Ended Cold Leg~~	~~6.2.1-16e~~

delete

TABLE 6.2.1-16
Sheet 1 of 2

DOUBLE-ENDED PUMP SUCTION GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
.00000	.0	.0	.0	.0
.00106	91473.7	51248.1	42757.3	23896.9
.00206	42955.1	24008.0	42573.3	23792.4
.101	42503.1	23834.3	22097.2	12338.1
.201	43339.5	24490.9	24176.6	13509.5
.301	44366.8	25335.1	24242.0	13556.5
.502	45862.8	26862.9	22296.6	12487.5
.601	45530.9	26983.4	21371.8	11974.1
.801	42484.8	25693.1	19928.8	11170.7
1.00	39544.8	24368.2	19298.7	10824.5
1.20	37020.2	23262.0	19108.0	10720.7
2.20	28255.7	19308.9	18896.7	10605.3
2.40	25402.6	17633.7	18617.8	10450.6
2.60	21678.4	15264.0	18000.3	10106.0
3.50	16360.6	11851.1	16210.1	9126.8
3.80	15077.5	10967.9	15776.6	8894.4
4.20	14035.8	10226.3	15279.0	8629.9
4.60	13441.5	9765.4	14811.9	8382.3
5.00	13257.2	9568.5	16166.4	9167.8
6.00	13191.3	9364.2	15135.9	8618.2
6.60	12777.0	9053.4	14505.5	8270.4
6.80	12788.1	9058.9	14412.3	8218.1
7.00	12210.8	8944.3	14493.5	8262.5
7.20	10650.0	8405.3	14208.2	8092.6
7.40	10261.3	8190.8	14004.9	7974.7
8.00	11126.8	8400.8	13504.2	7688.5
8.40	12274.7	8874.1	13135.7	7475.3
9.00	13232.2	9155.6	12581.1	7153.5
9.40	12766.8	8727.1	12250.8	6961.4
10.2	10649.9	7335.3	11715.6	6649.1
10.8	9517.9	6725.6	11327.3	6424.6
12.0	7946.8	5916.2	10528.8	5962.4
13.2	6868.8	5265.3	9696.9	5483.5
15.6	5560.8	4219.5	8154.6	4617.2
16.4	5016.5	4322.6	7007.9	4036.2
17.0	3745.2	4134.8	6328.7	3317.5
17.6	2755.2	3382.2	5113.6	2416.3
18.6	1896.4	2374.3	3573.7	1486.2

TABLE 6.2.1-16
Sheet 2 of 2

DOUBLE-ENDED PUMP SUCTION GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
19.0	1573.5	1978.3	6445.1	2574.5
19.2	1422.1	1791.6	6523.6	2603.7
19.6	1205.0	1523.9	3559.5	1408.2
20.0	1054.7	1336.7	1900.4	739.6
20.2	962.8	1221.5	2056.3	715.9
20.6	783.8	996.6	3656.8	1161.0
23.0	273.1	350.4	1643.0	456.1
23.8	196.8	253.0	1290.4	344.3
24.4	177.0	227.8	1217.4	324.0
25.4	138.3	178.3	309.1	86.2
26.0	109.3	141.0	.0	.0
27.8	.0	.0	.0	.0

TABLE 6.2.1-16a
(Sheet 1 of 1)

BLOWDOWN DOUBLE-ENDED PUMP SUCTION BREAK

Time (sec)	Mass Rate (lbs/sec)	Energy Rate (Btu/sec)
1.00000E-08	7.01135E+04	3.92484E+07
2.50331E-02	7.01135E+04	3.92484E+07
1.25218E-01	7.74512E+04	4.34598E+07
2.50276E-01	8.27959E+04	4.67551E+07
3.50283E-01	7.96880E+04	4.53219E+07
4.50334E-01	7.33320E+04	4.21250E+07
5.75504E-01	7.02176E+04	4.07680E+07
7.25475E-01	6.71328E+04	3.94242E+07
8.75455E-01	6.41607E+04	3.80113E+07
1.07552E+00	6.05893E+04	3.61407E+07
1.35026E+00	5.66415E+04	3.40634E+07
1.65024E+00	5.19535E+04	3.16012E+07
1.90023E+00	4.75583E+04	2.92134E+07
2.75014E+00	3.86340E+04	2.43129E+07
4.25035E+00	3.15758E+04	2.32677E+07
5.75034E+00	2.88951E+04	1.86041E+07
7.25076E+00	2.60857E+04	1.69035E+07
8.75090E+00	2.30263E+04	1.55946E+07
1.02500E+01	2.05454E+04	1.41215E+07
1.20020E+01	1.76466E+04	1.23493E+07
1.37519E+01	1.46272E+04	1.04894E+07
1.52508E+01	1.24415E+04	8.99497E+06
1.67506E+01	1.01873E+04	7.17505E+06
1.82507E+01	7.28281E+03	4.82533E+06
1.97504E+01	4.15947E+03	2.87425E+06
2.15006E+01	2.26931E+03	1.32581E+06
2.32505E+01	6.49016E+02	3.73793E+05
2.40056E+01	9.37511E+01	2.71096E+04
2.40112E+01	0	0

Delete

TABLE 6.2.1-16b
(Sheet 1 of 1)

0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE

Time (sec)	Mass Rate (lbs/sec)	Energy Rate (Btu/sec)
1.00000E-08	4.79640E+04	2.68810E+07
2.50299E-02	4.79640E+04	2.68810E+07
1.25329E-01	6.31848E+04	3.54951E+07
2.75628E-01	6.36324E+04	3.59799E+07
4.00528E-01	6.17803E+04	3.51421E+07
5.25375E-01	5.93017E+04	3.40101E+07
6.75264E-01	5.66199E+04	3.27632E+07
8.25324E-01	5.30648E+04	3.09105E+07
1.00043E+00	4.99152E+04	2.92173E+07
1.25040E+00	4.88309E+04	2.87375E+07
1.55034E+00	4.80839E+04	2.84306E+07
1.85026E+00	4.30301E+04	2.79501E+07
2.75030E+00	3.96863E+04	2.41877E+07
4.50032E+00	2.95092E+04	1.86615E+07
6.25035E+00	2.64339E+04	1.68526E+07
7.75033E+00	2.42472E+04	1.55843E+07
9.25022E+00	2.09389E+04	1.41847E+07
1.10014E+01	1.80135E+04	1.25018E+07
1.30015E+01	1.58736E+04	1.11191E+07
1.47511E+01	1.37137E+04	9.77383E+06
1.65009E+01	1.18047E+04	8.41174E+06
1.82505E+01	9.64089E+03	7.02915E+06
2.00008E+01	7.78744E+03	5.24305E+06
2.20006E+01	4.46210E+03	3.08245E+06
2.37505E+01	2.87242E+03	1.79967E+06
2.55004E+01	1.85402E+03	1.01407E+06
2.75002E+01	6.52229E+02	3.30052E+05
2.85410E+01	9.00244E+01	2.51400E+04
2.85821E+01	0	0

Delete

TABLE 6.2.1-16c
(Sheet 1 of 1)

3.0 FT^2 PUMP SUCTION SPLIT BREAK

Time (sec)	Mass Rate (lbs/sec)	Energy Rate (Btu/sec)
1.00000E-08	2.68603E+04	1.49186E+07
2.50198E-02	2.68603E+04	1.49186E+07
1.50202E-01	4.56233E+04	2.55286E+07
3.25255E-01	4.50196E+04	2.52160E+07
4.75198E-01	4.38802E+04	2.47597E+07
6.50203E-01	4.19298E+04	2.39053E+07
8.25258E-01	3.97821E+04	2.29166E+07
1.05018E+00	3.73363E+04	2.17363E+07
1.40021E+00	3.44804E+04	2.02816E+07
1.75031E+00	3.24718E+04	1.92259E+07
2.70033E+00	2.73104E+04	1.63721E+07
4.50033E+00	2.34960E+04	1.43062E+07
6.50058E+00	2.11243E+04	1.29734E+07
8.75048E+00	1.95321E+04	1.20108E+07
1.07504E+01	1.69029E+04	1.08458E+07
1.27520E+01	1.45476E+04	9.63951E+06
1.52518E+01	1.32818E+04	8.89267E+06
1.80006E+01	1.16929E+04	7.98823E+06
2.10011E+01	9.93231E+03	6.97085E+06
2.40011E+01	8.21381E+03	5.98077E+06
2.67507E+01	6.58449E+03	5.08452E+06
2.90006E+01	5.35300E+03	4.12997E+06
3.10006E+01	4.08384E+03	2.92654E+06
3.32504E+01	2.42247E+03	1.71390E+06
3.52502E+01	1.76877E+03	1.06222E+06
3.70004E+01	1.86478E+03	7.60013E+05
3.92505E+01	1.33534E+02	1.64594E+05
4.09874E+01	2.44842E+01	2.94524E+04
4.14743E+01	0	0

delete

TABLE 6.2.1-16d
(Sheet 1 of 1)

DOUBLE-ENDED HOT LEG GUILLOTINE BREAK

Time (sec)	Mass Rate (lbs/sec)	Energy Rate (Btu/sec)
1.00000E-08	6.96547E+04	4.58031E+07
2.51261E-02	6.96547E+04	4.58031E+07
1.00235E-01	8.15972E+04	5.37630E+07
2.00266E-01	7.34083E+04	4.81065E+07
3.00398E-01	6.98929E+04	4.54532E+07
4.25354E-01	6.68622E+04	4.31261E+07
5.50405E-01	6.46194E+04	4.14575E+07
6.50302E-01	6.31444E+04	4.04248E+07
7.75231E-01	6.16152E+04	3.94045E+07
9.00263E-01	6.01975E+04	3.84834E+07
1.07512E+00	5.87096E+04	3.75333E+07
1.30026E+00	5.71064E+04	3.65080E+07
1.55026E+00	5.50804E+04	3.52799E+07
1.85030E+00	5.18404E+04	3.33336E+07
2.50026E+00	4.53849E+04	2.94695E+07
3.50033E+00	3.89711E+04	2.54537E+07
4.75050E+00	3.55118E+04	2.31258E+07
6.00122E+00	3.37713E+04	2.18334E+07
7.25178E+00	2.94095E+04	1.96306E+07
8.75165E+00	2.47901E+04	1.70620E+07
1.05023E+01	2.07998E+04	1.45049E+07
1.22515E+01	1.66694E+04	1.19824E+07
1.37503E+01	1.28715E+04	9.54153E+06
1.50005E+01	9.30597E+03	6.76967E+06
1.62505E+01	5.50193E+03	4.22690E+06
1.77502E+01	2.38147E+03	2.11086E+06
1.90003E+01	6.88399E+02	5.41005E+05
1.96416E+01	2.41188E+02	2.02620E+05
1.97828E+01	0	0

delete

TABLE 6.2.1-16e
(Sheet 1 of 1)

DOUBLE-ENDED COLD LEG GUILLOTINE BREAK

Time (sec)	Mass Rate (lbs/sec)	Energy Rate (Btu/sec)
1.00000E-08	5.74645E+04	3.28466E+07
2.50459E-02	5.74645E+04	3.28466E+07
1.00114E-01	9.03492E+04	5.18057E+07
2.00119E-01	9.18449E+04	5.26979E+07
3.25172E-01	9.06893E+04	5.20413E+07
4.50202E-01	8.90442E+04	5.11007E+07
5.50125E-01	8.80348E+04	5.05383E+07
6.75058E-01	8.65350E+04	4.97178E+07
8.00023E-01	8.48007E+04	4.87821E+07
9.00050E-01	8.32524E+04	4.79554E+07
1.02506E+00	8.23989E+04	4.75621E+07
1.25015E+00	7.96964E+04	4.61989E+07
1.50024E+00	7.55285E+04	4.40130E+07
1.70015E+00	7.26571E+04	4.25099E+07
2.15012E+00	6.40333E+04	3.76596E+07
3.00013E+00	5.19763E+04	3.10462E+07
4.25057E+00	4.26906E+04	2.65828E+07
5.50071E+00	3.70909E+04	2.37146E+07
6.75047E+00	3.23565E+04	2.07472E+07
8.00094E+00	2.70288E+04	1.74891E+07
9.25178E+00	2.09152E+04	1.41635E+07
1.07513E+01	1.51420E+04	1.10073E+07
1.20002E+01	9.28263E+03	7.77219E+06
1.32504E+01	6.13172E+03	5.28681E+06
1.45009E+01	3.85781E+03	3.30780E+06
1.57503E+01	2.39742E+03	2.03887E+06
1.72504E+01	7.18563E+02	8.07354E+05
1.82102E+01	1.57874E+02	2.01111E+05
1.84200E+01	0	0

delete

TABLE 6.2.1-17
(Sheet 1 of 1)

19 ELEMENT W REFLOOD MODEL
(6 Broken Loop Elements, 13 Unbroken Loop Elements)

	Element	Broken Loop Area (Ft^2)	Unbroken Loop Area (Ft^2)	Form Factor K	Equivalent Length (Ft)	Hydraulic Diameter (Ft)
1.	Hot Leg Nozzle	4.59	13.77	.181	0.0	2.42
2.	Hot Leg Piping	4.59	13.77	.447	0.0	2.42
3.	Steam Generator Inlet Plenum	4.59	13.77	.442	0.0	2.42
4.	Steam Generator Tubes	11.24	33.72	3.01	55.9	.055
5.	Steam Generator Outlet Plenum	5.24	15.72	.317	0.0	2.58
6.	Crossover Leg Piping	5.24	15.72	.691	0.0	2.58
7.	Pump (forward)		13.50	(1)	0.0	2.4
8.	Cold Leg Piping		12.36	.310	0.0	2.29
9.	Cold Leg Inlet Nozzle		12.36	.373	0.0	2.29
10.	Around Downcomer (est.)(2)		20.0	0.01	8.0	4.0
11.	Cold Leg Inlet Nozzle		4.12	.373	0.0	2.29
12.	Cold Leg Piping		4.12	.310	0.0	2.29
13.	Pump (reverse)		4.50	(1)	0.0	2.4

delete

(1) The analysis accounts for transient pump resistances due to pump coastdown.

(2) The path around the downcomer is specified only to provide a loop reference point for pressure at top of downcomer. The frictional pressure drop data are estimated and provide negligible pressure drop.

TABLE 6.2.1-18
(Sheet 1 of 1)

REFLOOD DATA SUMMARY

Breaks	Tables
Double-Ended Pump Suction Minimum SI	6.2.1-19a
~~Double-Ended Pump Suction Maximum SI~~	~~6.2.1-19b~~
~~0.6 Double-Ended Pump Suction Maximum SI~~	~~6.2.1-19c~~
3 ~~ft^2 Pump Suction Split Maximum SI~~	~~6.2.1-19d~~
~~Double-Ended Hot Leg Maximum SI~~	~~6.2.1-19e~~
~~Double-Ended Cold Leg Maximum SI~~	~~6.2.1-19f~~

TABLE 6.2.1-19
Sheet 1 of 3
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
REFLOOD MASS AND ENERGY RELEASE - MINIMUM SAFETY INJECTION

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
27.8	.0	.0	.0	.0
28.8	.0	.0	.0	.0
28.9	52.4	61.0	.0	.0
29.2	9.7	11.3	.0	.0
31.9	76.8	89.4	.0	.0
33.9	103.4	120.6	.0	.0
34.9	247.4	290.2	3287.1	476.2
36.0	324.6	381.8	4337.8	632.3
36.3	324.2	381.4	4332.2	632.9
37.0	321.8	378.6	4299.4	629.8
38.0	317.0	372.9	4242.6	623.1
42.0	297.4	349.5	4004.6	592.9
43.0	292.8	344.1	3947.8	585.6
45.0	284.2	333.8	3839.4	571.4
47.0	276.2	324.3	3737.7	558.1
49.0	268.7	315.5	3642.4	545.6
51.0	261.9	307.3	3553.0	533.9
53.0	255.5	299.8	3469.0	522.9
55.0	249.6	292.7	3389.8	512.5
57.0	244.1	286.2	3315.0	502.8
59.0	238.9	280.0	3244.3	493.5
61.0	234.0	274.3	3177.1	484.7
63.0	229.4	268.9	3113.3	476.4
67.0	221.0	258.9	2994.6	460.8
71.0	213.4	250.0	2886.2	446.7
75.0	206.6	241.9	2786.4	433.6
79.0	200.3	234.5	2694.1	421.5
83.0	194.6	227.7	2608.3	410.3
84.0	254.5	297.9	250.6	123.1
85.0	341.4	401.9	286.5	173.8
86.0	341.9	402.5	286.8	174.4
93.0	308.3	362.5	271.0	152.8
97.0	291.8	342.8	264.0	143.6
101.0	278.5	327.0	258.5	136.3
105.0	267.2	313.6	253.9	130.2
109.0	257.7	302.3	250.1	125.1
121.0	237.2	278.1	241.9	114.2
133.0	225.3	264.0	237.2	108.0
145.0	218.9	256.4	234.6	104.7
161.0	215.0	251.8	233.0	102.6
165.0	215.8	252.7	233.9	102.9
173.0	218.1	255.4	238.6	104.0

TABLE 6.2.1-19
Sheet 2 of 3
DOULBE-ENDED PUMP SUCTION GUILLOTINE
REFLOOD MASS AND ENERGY RELEASE - MINIMUM SAFETY INJECTION

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY THOUSAND	FLOW	ENERGY THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
181.0	220.0	257.7	244.9	105.0
197.0	220.9	258.8	259.0	105.8
205.0	219.8	257.5	266.6	105.6
213.0	217.9	255.2	275.2	105.3
215.0	217.4	254.6	277.5	105.2
225.0	213.5	250.0	289.3	104.4
241.0	203.9	238.7	308.6	102.6
245.5	200.7	234.9	314.7	102.1
245.6	208.2	259.7	431.8	117.2
245.6	208.2	259.7	431.8	117.2
250.6	207.3	258.6	432.7	117.3
255.6	207.7	259.0	432.3	117.1
260.6	206.8	257.9	433.2	117.1
265.6	207.1	258.3	432.9	117.0
270.6	206.2	257.1	433.8	117.0
275.6	206.4	257.5	433.5	116.9
280.6	205.5	256.3	434.5	117.0
285.6	205.7	256.6	434.2	116.8
290.6	204.7	255.4	435.2	116.9
295.6	204.9	255.6	435.0	116.7
300.6	203.9	254.4	436.0	116.8
305.6	204.1	254.6	435.9	116.7
310.6	203.1	253.3	436.9	116.8
315.6	203.2	253.4	436.8	116.6
320.6	202.1	252.1	437.8	116.7
330.6	202.3	252.3	437.7	116.5
335.6	201.1	250.9	438.8	116.6
345.6	201.1	250.8	438.9	116.3
350.6	199.9	249.4	440.0	116.5
370.6	199.4	248.6	440.6	116.1
390.6	197.2	245.9	442.8	116.1
425.6	194.2	242.2	445.8	115.9
430.6	194.5	242.6	445.5	115.7
440.6	193.1	240.8	446.9	115.8
445.6	193.2	241.0	446.8	115.7
475.6	190.6	237.8	449.3	115.5
480.6	190.7	237.8	449.3	115.4
510.6	187.5	233.9	452.4	115.3
515.6	187.6	234.0	452.3	115.2
545.6	184.6	230.3	455.3	115.1

TABLE 6.2.1-19
Sheet 3 of 3
DOULBE-ENDED PUMP SUCTION GUILLOTINE
REFLOOD MASS AND ENERGY RELEASE - MINIMUM SAFETY INJECTION

TIME	BREAK PATH NO.1		BREAK PATH NO.2	
	FLOW	ENERGY	FLOW	ENERGY
		THOUSAND		THOUSAND
SECOND	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
550.6	184.4	230.0	455.6	115.1
555.6	82.0	102.2	558.0	137.0
771.7	82.0	102.2	558.0	137.0
771.8	79.5	98.8	560.5	132.3
775.6	79.4	98.6	560.6	132.2
1595.6	66.0	82.0	573.9	126.4
1600.3	66.0	82.0	582.5	126.8
1707.3	64.8	80.5	583.7	148.3
1712.3	64.8	80.4	453.7	134.5
2041.3	64.8	80.4	453.7	134.5
2041.4	63.0	72.5	455.5	48.8

TABLE 6.2.1-19a
Sheet 1 of 1

MASS AND ENERGY RELEASES
POST-BLOWDOWN DEPS GUILLOTINE MINIMUM SAFEGUARDS

Time (sec)	Mass Rate (lbm/sec)	Energy Rate (Btu/sec)
2.4000000E+01	0	0
2.5080000E+01	4.4535613E+02	5.7759761E+05
2.5210000E+01	2.3648755E+02	3.0670323E+05
2.6010000E+01	3.6059006E+02	4.6764922E+05
3.1010000E+01	9.7221735E+02	1.2588249E+06
3.2010000E+01	1.0546636E+03	1.3644830E+06
3.2010000E+01	1.0579200E+03	1.3685351E+06
3.6010000E+01	1.0194988E+03	1.3153768E+06
4.7010000E+01	9.2782863E+02	1.1882574E+06
5.0000000E+01	9.0445180E+02	1.1559287E+06
5.4010000E+01	8.7837558E+02	1.1200505E+06
6.4010000E+01	7.2257377E+02	9.1625920E+05
6.4010000E+01	7.2186668E+02	9.1534335E+05
7.4010000E+01	6.1354674E+02	7.7492356E+05
8.4010000E+01	5.4344647E+02	6.8418096E+05
1.0000000E+02	5.0943384E+02	6.3848652E+05
1.4401000E+02	4.2507584E+02	5.2736718E+05
1.9499900E+02(1)	4.0363473E+02	4.9586537E+05
1.9500100E+02	1.5140372E+02	1.8596837E+05
2.0000000E+02	1.5046056E+02	1.8480579E+05
5.0000000E+02	1.0753474E+02	1.3189885E+05
1.0000000E+03	8.3065608E+01	1.0172258E+05
1.4999990E+03	7.3412585E+01	8.9786862E+04
1.5000010E+03	8.2843218E+01	1.0131827E+05
2.0000000E+03	7.6853863E+01	9.3879329E+04
5.0000000E+03	5.8619500E+01	7.1208424E+04
1.0000000E+04	4.8076742E+01	5.8075932E+04
2.0000000E+04	3.9523537E+01	4.7433722E+04
1.0000000E+06	1.1999430E+01	1.3945929E+04

Notes:

(1) Entrainment ends at 195.00 seconds

delete

TABLE 6.2.1-19b
Sheet 1 of 1

MASS AND ENERGY RELEASES
POST-BLOWDOWN DOUBLE-ENDED PUMP SUCTION
GUILLOTINE MAXIMUM SAFEGUARDS

Time (sec)	Mass Rate (lbm/sec)	Energy Rate (Btu/sec)
2.4000000E+01	0	0
2.5010000E+01	3.2911002E+02	4.2682516E+05
2.6010000E+01	3.5178251E+02	4.5623152E+05
2.7010000E+01	5.0915921E+02	6.6025506E+05
3.1010000E+01	1.0174371E+03	1.3171506E+06
3.2010000E+01	1.0403702E+03	1.3534354E+06
3.2010000E+01	1.0456572E+03	1.3524546E+06
3.4010000E+01	1.0250232E+03	1.3239151E+06
3.7010000E+01	9.9838583E+02	1.2869778E+06
4.5010000E+01	9.3259202E+02	1.1959739E+06
5.0000000E+01	8.9624439E+02	1.1458270E+06
5.4010000E+01	8.6810003E+02	1.1073057E+06
7.0010000E+01	7.6576687E+02	9.6822679E+05
8.4010000E+01	6.9297570E+02	8.7060582E+05
1.0000000E+02	6.3350166E+02	7.9081213E+05
1.4401000E+02	4.6917311E+02	5.7788924E+05
1.6699900E+02(1)	4.0929915E+02	5.0188401E+05
1.6700100E+02	1.5871940E+02	1.9457442E+05
2.0000000E+02	1.5026145E+02	1.8417356E+05
5.0000000E+02	1.0753474E+02	1.3163706E+05
1.0000000E+03	8.3065608E+01	1.0153356E+05
1.4999990E+03	7.3412585E+01	8.9629853E+04
1.5000010E+03	8.2843218E+01	1.0114041E+05
2.0000000E+03	7.6853863E+01	9.3721244E+04
5.0000000E+03	5.8619500E+01	7.1117839E+04
1.0000000E+04	4.80767425+01	5.8025135E+04
2.0000000E+04	3.9523537E+01	4.7410502E+04
5.0000000E+04	3.0543703E+01	3.6316472E+04
1.0000000E+06	1.1999430E+01	1.3945907E+04

Notes:

(1) Entrainment ends at 167.00 seconds

delete

TABLE 6.2.1-19c
Sheet 1 of 1

MASS AND ENERGY RELEASES
0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE

Time (sec)	Mass Rate (lbm/sec)	Energy Rate (Btu/sec)
2.8600000E+01	0	0
2.9670000E+01	5.8926898E+02	7.6569765E+05
3.0110000E+01	2.8303831E+02	3.6775932E+05
3.0610000E+01	3.5426500E+02	4.6029821E+05
3.5610000E+01	1.0271490E+03	1.3321612E+06
3.9610000E+01	9.9780491E+02	1.2906958E+06
4.8610000E+01	9.2177392E+02	1.1854595E+06
5.0000000E+01	9.1222348E+02	1.1722791E+06
5.3610000E+01	8.8265601E+02	1.1315763E+06
6.0610000E+01	8.3700178E+02	1.0689368E+06
7.8610000E+01	7.3032468E+02	9.2426316E+05
9.9650000E+01	6.3186100E+02	7.9295345E+05
1.0000000E+02	6.2785516E+02	7.8768698E+05
1.2861000E+02	5.1883900E+02	6.4501941E+05
1.4861000E+02	4.5833808E+02	5.6763498E+05
1.7059900E+02	4.0715255E+02	4.9687693E+05
1.7060100E+02	3.9935071E+02	4.9207727E+05
1.7079900E+02(1)	3.9934552E+02	4.9207062E+05
1.7080100E+02	1.5819036E+02	1.9489679E+05
2.0000000E+02	1.5109456E+02	1.8612162E+05
5.0000000E+02	1.0771956E+02	1.3248813E+05
1.0000000E+03	8.3103659E+01	1.0203180E+05
1.4999990E+03	7.3420381E+01	9.0016658E+04
1.5000010E+03	8.2851725E+01	1.0157696E+05
2.0000000E+03	7.6855683E+01	9.4100179E+04
5.0000000E+03	5.8619500E+01	7.1339029E+04
1.0000000E+04	4.8076742E+01	5.8152101E+04
2.0000000E+04	3.9523537E+01	4.7471394E+04
1.0000000E+06	1.1999430E+01	1.3945923E+04

delete

Notes:

(1) Entrainment ends at 170.80 seconds

TABLE 6.2.1-19d
Sheet 1 of 1

MASS AND ENERGY RELEASES
3 FT^2 PUMP SUCTION SPLIT

Time (sec)	Mass Rate (lbm/sec)	Energy Rate (Btu/sec)
4.1500000E+01	0	0
4.2680000E+01	6.7191477E+02	8.7107099E+05
4.3510000E+01	2.8142791E+02	3.6477252E+05
4.9510000E+01	9.1074969E+02	1.1783529E+06
5.0000000E+01	9.7405084E+02	1.2598686E+06
5.6510000E+01	9.1843783E+02	1.1825077E+06
6.1510000E+01	8.8458777E+02	1.1354443E+06
6.6510000E+01	8.5389483E+02	1.0928981E+06
7.1510000E+01	8.2181484E+02	1.0485744E+06
8.1510000E+01	7.6530291E+02	9.7113123E+05
9.1510000E+01	7.1653478E+02	9.0491471E+05
1.0000000E+02	6.8022180E+02	8.5581957E+05
1.1374000E+02	6.2007089E+02	7.7568913E+05
1.4151000E+02	5.1920500E+02	6.4333099E+05
1.6151000E+02	4.5633792E+02	5.6224391E+05
1.8250900E+02(1)	4.0140774E+02	4.9234933E+05
1.8251100E+02	2.4851142E+02	3.0475763E+05
1.8289900E+02	2.4851115E+02	3.0475730E+05
1.8290100E+02	1.5640830E+02	1.9180789E+05
2.0000000E+02	1.5254775E+02	1.8705735E+05
5.0000000E+02	1.0824998E+02	1.3255538E+05
1.0000000E+03	8.3211201E+01	1.0173360E+05
1.4999990E+03	7.3440869E+01	8.9676032E+04
1.5000010E+03	8.2874008E+01	1.0119188E+05
2.0000000E+03	7.6858829E+01	9.3735455E+04
5.0000000E+03	5.8617894E+01	7.1100225E+04
1.0000000E+04	4.8075425E+01	5.7995931E+04
2.0000000E+04	3.9522455E+01	4.7377723E+04
1.0000000E+06	1.1999101E+01	1.3945329E+04

Notes:

~~(1) Entrainment ends at 182.51 seconds~~ delete

TABLE 6.2.1-19e
Sheet 1 of 1

MASS AND ENERGY RELEASES
DOUBLE-ENDED HOT LEG GUILLOTINE

Time (sec)	Mass Rate (lbm/sec)	Energy Rate (Btu/sec)
1.9800000E+01	0	0
2.0000000E+01	1.2255363E+03	3.1953427E+05
2.0410000E+01	1.1062387E+03	4.5042751E+05
2.0510000E+01	1.0687712E+03	4.5157073E+05
2.0710000E+01	4.4278860E+02	4.1377824E+05
2.1610000E+01	6.8216035E+02	4.8675568E+05
2.6810000E+01	1.9629021E+03	8.1212676E+05
2.9810000E+01	1.9949716E+03	8.2479356E+05
3.7810000E+01	1.8417703E+03	7.8488364E+05
5.0000000E+01	1.5319554E+03	7.0257474E+05
5.5810000E+01	1.3848108E+03	6.6410252E+05
6.3810000E+01	1.1690957E+03	6.0798705E+05
6.9810000E+01	1.0667600E+03	5.8010146E+05
7.9810000E+01	9.8498857E+02	5.5580778E+05
9.9810000E+01	9.2584339E+02	5.3341112E+05
1.0000000E+02	9.2449665E+02	5.3291239E+05
1.2929900E+02(1)	7.7958780E+02	4.8695008E+05
1.2930100E+02	1.7010402E+02	2.0092702E+05
2.0000000E+02	1.5016167E+02	1.7736875E+05
5.0000000E+03	1.0736836E+02	1.2681619E+05
1.0000000E+03	8.3031351E+01	9.8065589E+04
1.4999990E+03	7.3405566E+01	8.6692914E+04
1.5000010E+03	8.2835559E+01	9.7829762E+04
2.0000000E+03	7.6852224E+01	9.0759231E+04
5.0000000E+03	5.8619500E+01	6.9210825E+04
1.0000000E+04	4.8076742E+01	5.6745978E+04
2.0000000E+04	3.9523537E+01	4.6627441E+04
5.0000000E+04	3.0543703E+01	3.5992603E+04
1.0000000E+06	1.1999430E+01	1.3996067E+04

Notes:

~~(1) Entrainment ends at 129.30 seconds~~

delete

TABLE 6.2.1-19f
Sheet 1 of 1

MASS AND ENERGY RELEASES
DOUBLE-ENDED COLD LEG GUILLOTINE

Time (sec)	Mass Rate (lbm/sec)	Energy Rate (Btu/sec)
1.8400000E+01	4.0206092E+01	5.2178964E+04
2.0000000E+01	3.6096903E+02	4.7095009E+05
2.0640000E+01	1.6422536E+02	2.1425283E+05
2.2410000E+01	1.9902935E+02	2.5965535E+05
2.5410000E+01	2.6405340E+02	3.4446796E+05
2.8410000E+01	2.7290008E+02	3.5598031E+05
3.8410000E+01	2.6793330E+02	3.4941184E+05
5.0000000E+01	2.6735934E+02	3.4856086E+05
5.8410000E+01	2.6665917E+02	3.4757169E+05
7.8410000E+01	2.5637723E+02	3.3399541E+05
9.8410000E+01	2.4678501E+02	3.2133790E+05
1.0000000E+02	2.4609000E+02	3.2041854E+05
1.1841000E+02	2.4070208E+02	3.1326385E+05
2.0000000E+02	2.1941085E+02	2.8498083E+05
2.1841000E+02	2.1466465E+02	2.7869617E+05
3.1841000E+02	1.9143203E+02	2.4797559E+05
4.1841000E+02	1.6864794E+02	2.1802051E+05
4.9699900E+02(1)	1.4895299E+02	1.9230197E+05
4.9700100E+02	1.0753159E+02	1.4090801E+05
5.0000000E+02	1.0753150E+02	1.4090790E+05
1.0000000E+03	8.3020034E+01	1.0820367E+05
1.4999990E+03	7.3403247E+01	9.5261958E+04
1.5000010E+03	8.2833029E+01	1.0749028E+05
2.0000000E+03	7.6851683E+01	9.9318426E+04
5.0000000E+03	5.8619500E+01	7.4302652E+04
1.0000000E+04	4.8076742E+01	5.9723887E+04
2.0000000E+04	3.9523537E+01	4.8014843E+04
5.0000000E+04	3.0543703E+01	3.6285953E+04
1.0000000E+06	1.1999430E+01	1.3945396E+04

Notes:

~~(1) Entrainment ends at 497.00 seconds~~

delete

TABLE 6.2.1-20
Sheet 1 of 1

WATTS BAR MAXIMUM SI

POST-REFLOOD MASS AND ENERGY RELEASE INFORMATION

TIME SECONDS	STEAM FLOW MASS LB_m/SEC	STEAM FLOW ENERGY 10^3 BTU/SEC	WATER FLOW MASS LB_m/SEC	WATER FLOW ENERGY 10^3 BTU/SEC
167	131	156	1250	199
202	128	152	1260	199
302	126	150	1260	188
402	126	149	1260	186
502	126	148	1260	190
602	126	148	1260	186
702	127	148	1260	183
727	127	148	1260	182
732	92.3	106	1290	189
802	90.3	104	1290	187
902	87.7	101	1290	190
1002	85.5	98.9	1300	186
1102	83.5	96.6	1300	188
1302	80.2	92.7	1300	188
1502	77.4	89.5	1310	186
1637	75.7	87.6	1310	187
INTEGRATED	$10^3 LB_m$	10^6 BTU	$10^3 LB_m$	10^6 BTU
1637	146	171	1890	277

delete

TABLE 6.2.1-21
Sheet 1 of 1

WATTS BAR MINIMUM SI

POST-REFLOOD MASS AND ENERTY RELEASE INFORMATION

TIME SECONDS	STEAM FLOW MASS LB_m/SEC	STEAM FLOW ENERGY 10^3 BTU/SEC	WATER FLOW MASS LB_m/SEC	WATER FLOW ENERGY 10^3 BTU/SEC
195	297	354	370	72.9
200	297	353	370	72.9
300	297	344	371	73.0
305	297	344	371	73.0
310	149	172	519	102
400	140	161	528	104
500	131	152	536	106
600	131	152	536	106
700	124	143	543	107
800	118	136	549	108
900	118	137	549	108
1000	112	129	556	109
1200	105	121	563	111
1400	100	116	567	112
1600	96.6	112	571	112
1765	97.1	112	570	112
INTEGRATED	10^3 LB_m	10^6 BTU	10^3 LB_m	10^6 BTU
1765	200	232	848	167

delete

TABLE 6.2.1-22
Sheet 1 of 1

AVAILABLE ENERGY BETWEEN 20.2 PSIA AND 14.7 PSIA

Broken Loop Steam Generator	3.696×10^6 Btu
Unbroken Loop Steam Generator	10.934×10^6 Btu
Metal Energy (THIN + THICK)	4.816×10^6 Btu
Core Stored	$.604 \times 10^6$ Btu
TOTAL	20.05×10^6 Btu

delete

TABLE 6.2.1-25
Sheet 1 of 1

DOUBLE-ENDED PUMP SUCTION LOCA

Event	Time (sec)
Rupture	0
Accumulator flow starts	~~15.5~~ 16.0
Assumed initiation of ECCS	~~24.0~~ 35.0
End of blowdown	~~24.0~~ 27.8
Assumed initiation of spray system	~~55.0~~ 221.
Accumulators empty	~~56.1~~ 63.79
End of reflood	~~167.0~~ 245.54
Low level alarm of refueling water storage tank	~~1095~~ 1571.3
Beginning of recirculation phase of safeguards operation	~~1455~~ 1631.3

TABLE 6.2.1-26a
Sheet 1 of 1

WATTS BAR FOUR LOOP PLANT

DOUBLE-ENDED PUMP SUCTION GUILLOTINE, MAX. S.I. W/FROTH

MASS BALANCE

	0.0	EOB	EOE	EOF	EOFIL
Time (seconds)	0.00	24.00	167.00	727.00	1642.00
Mass (10^3 lbm)					
AVAILABLE					
Initial RCS & Acc	714.94	714.94	714.94	714.94	714.94
ADDED MASS					
Pumped Injection	0.00	0.00	173.01	940.91	2213.82
Total Added	0.00	0.00	173.01	940.91	2213.82
TOTAL AVAILABLE	714.94	714.94	887.95	1655.85	2928.76
DISTRIBUTION					
Reactor Coolant	504.64	64.99	148.27	148.27	148.27
Accumulator	210.30	156.67	0.00	0.00	0.00
Total Contents	714.94	221.66	148.27	148.27	148.27
EFFLUENT					
Break Flow	0.00	493.21	587.76	657.89	734.41
ECCS Spill	0.00	0.00	151.92	849.69	2046.08
Total Effluent	0.00	493.21	739.68	1507.58	2780.49
TOTAL ACCOUNTABLE	714.94	714.87	887.95	1655.85	2928.76

ENERGY BALANCE

	0.0	EOB	EOE	EOF	EOFIL
Time (seconds)	0.00	24.00	167.00	727.00	1642.00
Energy (10^6 Btu)					
AVAILABLE					
In RCS, Acc, & S Gen	816.61	816.61	816.61	816.61	816.61
ADDED ENERGY					
Pumped Injection	0.00	0.00	15.22	67.44	154.00
Decay Heat	0.00	9.86	31.63	91.29	168.24
**Heat from Sec.	0.00	-3.58	-3.58	1.09	8.83
Total Added	0.00	6.28	43.27	159.82	331.07
TOTAL AVAILABLE	816.61	822.89	859.89	976.43	1147.68
DISTRIBUTION					
Reactor Coolant	302.29	17.33	28.61	28.61	28.61
Accumulator	18.51	13.79	0.00	0.00	0.00
Core Stored	28.38	11.12	4.03	4.03	4.03
Thin Metal	25.01	20.50	9.44	9.44	9.44
Thick Metal	30.78	30.78	23.74	23.74	23.74
Steam Generator	411.65	411.00	343.05	262.31	161.39
Total Contents	816.61	504.51	408.86	328.13	227.21
EFFULENT					
Break Flow	0.00	318.39	437.48	520.34	608.85
ECCS Spill	0.00	0.00	13.37	118.15	291.00
Total Effluent	0.00	318.39	450.85	638.49	899.85
TOTAL ACCOUNTABLE	816.61	822.90	859.71	966.62	1127.06

** Steam out and feedwater into the steam generator

delete (Replacement)

TABLE 6.2.1-26b
Sheet 1 of 1

WATTS BAR FOUR LOOP PLANT

DOUBLE-ENDED PUMP SUCTION GUILLOTINE, MIN. S.I., W/FROTH

MASS BALANCE

	0.0	EOB	EOE	EOF	EOFIL
Time (seconds)	0.00	24.00	195.00	310.00	1770.00
Mass (10^3 lbm)					
AVAILABLE					
Initial RCS & Acc	714.94	714.94	714.94	714.94	714.94
ADDED MASS					
Pumped Injection	0.00	0.00	105.76	182.51	1156.92
Total Added	0.00	0.00	105.76	182.51	1156.92
TOTAL AVAILABLE	714.94	714.94	820.70	897.45	1871.86
DISTRIBUTION					
Reactor Coolant	504.64	64.99	148.27	148.27	148.27
Accumulator	210.30	156.67	0.00	0.00	0.00
Total Contents	714.94	221.66	148.27	148.27	148.27
EFFLUENT					
Break Flow	0.00	493.21	589.64	623.74	790.41
ECCS Spill	0.00	0.00	82.80	125.45	933.18
Total Effluent	0.00	493.21	672.44	749.19	1723.59
TOTAL ACCOUNTABLE	714.94	714.87	820.70	897.46	1871.86

ENERGY BALANCE

	0.0	EOB	EOE	EOF	EOFIL
Time (seconds)	0.00	24.00	195.00	310.00	1770.00
Energy (10^6 Btu)					
AVAILABLE					
In RCS, Acc, & S Gen	816.61	816.61	816.61	816.61	816.61
ADDED ENERGY					
Pumped Injection	0.00	0.00	9.31	14.53	80.79
Decay Heat	0.00	9.86	35.25	49.10	177.93
**Heat from Sec.	0.00	-3.58	-3.58	-3.10	3.05
Total Added	0.00	6.28	40.98	60.53	261.77
TOTAL AVAILABLE	816.61	822.89	857.59	877.14	1078.38
DISTRIBUTION					
Reactor Coolant	302.29	17.33	28.61	28.61	28.61
Accumulator	18.51	13.79	0.00	0.00	0.00
Core Stored	28.38	11.12	4.03	4.03	4.03
Thin Metal	25.01	20.50	9.44	9.44	9.44
Thick Metal	30.78	30.78	22.70	22.70	22.70
Steam Generator	411.65	411.00	343.01	315.50	155.42
Total Contents	816.61	504.51	409.79	380.28	220.20
EFFULENT					
Break Flow	0.00	318.39	440.26	480.34	673.10
ECCS Spill	0.00	0.00	7.29	15.68	174.65
Total Effluent	0.00	318.39	447.55	496.02	847.75
TOTAL ACCOUNTABLE	816.61	822.90	857.33	876.30	1067.95

** Steam out and feedwater into the steam generator

delete

TABLE 6.2.1-26c
Sheet 1 of 1

WATTS BAR FOUR LOOP PLANT - 0.6 DOUBLE-ENDED PUMP SUCTION GUILLOTINE

MASS BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	28.58	170.80	1500.00
Mass (10^3 lbm)				
AVAILABLE				
Initial RCS & Acc	714.94	714.94	714.94	714.94
ADDED MASS				
Pumped Injection	0.00	0.00	184.22	1978.94
Total Added	0.00	0.00	184.22	1978.94
TOTAL AVAILABLE	714.94	714.94	899.16	2693.88
DISTRIBUTION				
Reactor Coolant	504.64	77.61	160.89	160.89
Accumulator	210.30	144.97	0.00	0.00
Total Contents	714.94	222.58	160.89	160.89
EFFLUENT				
Break Flow	0.00	492.37	584.17	711.87
ECCS Spill	0.00	0.00	154.10	1821.12
Total Effluent	0.00	492.37	738.27	2532.99
TOTAL ACCOUNTABLE	714.94	714.95	899.16	2693.88

ENERGY BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	28.58	170.80	1500.00
Energy (10^6 Btu)				
AVAILABLE				
In RCS, Acc, & S Gen	817.91	817.91	817.91	817.91
ADDED ENERGY				
Pumped Injection	0.00	0.00	16.21	174.15
Decay Heat	0.00	10.84	32.24	157.24
**Heat from Sec.	0.00	-4.05	-4.05	-4.05
Total Added	0.00	6.79	44.40	327.34
TOTAL AVAILABLE	817.91	824.70	862.32	1145.25
DISTRIBUTION				
Reactor Coolant	302.29	20.12	31.40	31.40
Accumulator	18.51	12.76	0.00	0.00
Core Stored	28.38	10.23	4.03	4.03
Thin Metal	25.01	20.09	9.44	9.44
Thick Metal	30.78	30.78	23.77	11.79
Steam Generator	412.95	414.26	348.72	340.02
Total Contents	817.91	508.23	417.36	398.67
EFFULENT				
Break Flow	0.00	316.48	432.48	589.38
ECCS Spill	0.00	0.00	13.56	160.26
Total Effluent	0.00	316.48	446.04	749.64
TOTAL ACCOUNTABLE	817.91	824.71	863.40	1146.31

** Steam out and feedwater into the steam generator

Delete

TABLE 6.2.1-26d
Sheet 1 of 1

WATTS BAR FOUR LOOP PLANT - 3 FT^2 PUMP SUCTION

MASS BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	41.50	182.50	1500.00
Mass (10^3 lbm)				
AVAILABLE				
Initial RCS & Acc	714.94	714.94	714.94	714.94
ADDED MASS				
Pumped Injection	0.00	0.00	171.82	1950.74
Total Added	0.00	0.00	171.82	1950.74
TOTAL AVAILABLE	714.94	714.94	886.76	2665.68
DISTRIBUTION				
Reactor Coolant	504.64	100.22	183.50	183.50
Accumulator	210.30	127.36	0.00	0.00
Total Contents	714.94	227.58	183.50	183.50
EFFLUENT				
Break Flow	0.00	487.28	575.98	702.28
ECCS Spill	0.00	0.00	127.28	1779.90
Total Effluent	0.00	487.28	703.26	2482.18
TOTAL ACCOUNTABLE	714.94	714.86	886.76	2665.68

ENERGY BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	41.50	182.50	1500.00
Energy (10^6 Btu)				
AVAILABLE				
In RCS, Acc, & S Gen	812.86	812.86	812.86	812.86
ADDED ENERGY				
Pumped Injection	0.00	0.00	15.12	171.67
Decay Heat	0.00	13.23	33.83	157.33
**Heat from Sec.	0.00	-18.89	-18.89	-18.89
Total Added	0.00	-5.66	30.06	310.11
TOTAL AVAILABLE	812.86	807.20	842.92	1122.97
DISTRIBUTION				
Reactor Coolant	302.29	24.06	35.34	35.34
Accumulator	18.51	11.21	0.00	0.00
Core Stored	28.38	7.65	4.03	4.03
Thin Metal	25.01	19.08	9.44	9.44
Thick Metal	30.78	30.78	23.82	11.80
Steam Generator	407.90	401.58	335.52	327.62
Total Contents	812.86	494.35	408.14	388.22
EFFULENT				
Break Flow	0.00	312.84	424.59	579.14
ECCS Spill	0.00	0.00	11.20	156.63
Total Effluent	0.00	312.84	435.79	735.77
TOTAL ACCOUNTABLE	812.86	807.19	843.93	1123.99

** Steam out and feedwater into the steam generator

delete

TABLE 6.2.1-26e
Sheet 1 of 1

WATTS BAR FOUR LOOP PLANT

DOUBLE-ENDED HOT LEG GUILLOTINE, MAX. S.I.

MASS BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	19.80	129.30	1500.00
Mass (10^3 lbm)				
AVAILABLE				
Initial RCS & Acc	714.94	714.94	714.94	714.94
ADDED MASS				
Pumped Injection	0.00	0.00	138.77	1989.52
Total Added	0.00	0.00	138.77	1989.52
TOTAL AVAILABLE	714.94	714.94	853.71	2704.46
DISTRIBUTION				
Reactor Coolant	504.64	66.31	241.15	272.69
Accumulator	210.30	164.85	0.00	0.00
Total Contents	714.94	231.16	241.15	272.69
EFFLUENT				
Break Flow	0.00	482.76	612.56	746.76
ECCS Spill	0.00	0.00	0.00	1685.01
Total Effluent	0.00	482.76	612.56	2431.77
TOTAL ACCOUNTABLE	714.94	713.92	853.71	2704.46

ENERGY BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	19.80	129.30	1500.00
Energy (10^6 Btu)				
AVAILABLE				
In RCS, Acc, & S Gen	814.71	814.71	814.71	814.71
ADDED ENERGY				
Pumped Injection	0.00	0.00	12.21	175.08
Decay Heat	0.00	8.83	26.33	156.93
**Heat from Sec.	0.00	-0.18	-0.18	-0.18
Total Added	0.00	8.65	38.36	331.83
TOTAL AVAILABLE	814.71	823.35	853.07	1146.53
DISTRIBUTION				
Reactor Coolant	302.29	17.63	36.96	39.74
Accumulator	18.51	14.51	0.00	0.00
Core Stored	28.38	9.73	4.03	4.03
Thin Metal	25.01	21.02	9.44	9.44
Thick Metal	30.78	30.78	25.11	11.78
Steam Generator	409.74	406.21	389.32	386.72
Total Contents	814.71	499.87	464.86	451.71
EFFULENT				
Break Flow	0.00	323.39	389.59	547.99
ECCS Spill	0.00	0.00	0.00	148.28
Total Effluent	0.00	323.39	389.59	696.27
TOTAL ACCOUNTABLE	814.71	823.26	854.45	1147.98

** Steam out and feedwater into the steam generator.

delete

TABLE 6.2.1-26f
Sheet 1 of 1

WATTS BAR FOUR LOOP PLANT - DOUBLE-ENDED COLD LEG GUILLOTINE, MAX. S.I.

MASS BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	18.42	497.00	1500.00
Mass (10^3 lbm)				
AVAILABLE				
Initial RCS & Acc	714.94	714.94	714.94	714.94
ADDED MASS				
Pumped Injection	0.00	0.00	640.46	1994.73
Total Added	0.00	0.00	184.22	1994.73
TOTAL AVAILABLE	714.94	714.94	1355.40	2709.67
DISTRIBUTION				
Reactor Coolant	504.64	40.50	123.78	123.78
Accumulator	210.30	119.07	0.00	0.00
Total Contents	714.94	159.57	123.78	123.78
EFFLUENT				
Break Flow	0.00	502.29	601.39	686.99
ECCS Spill	0.00	52.60	630.23	1898.91
Total Effluent	0.00	554.89	1231.62	2585.90
TOTAL ACCOUNTABLE	714.94	714.46	1355.40	2709.90

ENERGY BALANCE

	0.0	EOB	EOE	REC
Time (seconds)	0.00	18.42	497.00	1500.00
Energy (10^6 Btu)				
AVAILABLE				
In RCS, Acc, & S Gen	816.50	816.50	816.50	816.50
ADDED ENERGY				
Pumped Injection	0.00	0.00	56.36	175.54
Decay Heat	0.00	8.47	68.97	156.77
**Heat from Sec.	0.00	-3.71	-3.71	-3.71
Total Added	0.00	4.76	121.62	328.59
TOTAL AVAILABLE	816.50	821.25	938.11	1145.09
DISTRIBUTION				
Reactor Coolant	302.29	11.35	22.63	22.63
Accumulator	18.51	10.48	0.00	0.00
Core Stored	28.38	16.03	4.03	4.03
Thin Metal	25.01	21.09	9.44	9.44
Thick Metal	30.78	30.78	15.76	11.78
Steam Generator	411.54	410.71	374.72	362.62
Total Contents	816.50	500.43	426.57	410.50
EFFULENT				
Break Flow	0.00	316.24	444.84	556.54
ECCS Spill	0.00	4.63	55.46	167.10
Total Effluent	0.00	320.87	500.30	723.64
TOTAL ACCOUNTABLE	816.50	821.30	926.87	1134.14

** Steam out and feedwater into the steam generator.

delete

TABLE 6.2.1-26A

WATTS BAR NUCLEAR PLANT UNIT 1

DOUBLE-ENDED PUMP SUCTION GUILLOTINE

MINIMUM SAFETY INJECTION - MASS BALANCE

		Start of Accident	End of Blowdown	Bottom of Core Recovery	End of Reflood	Broken Loop SG Equilibration	Intact Loop SG Equilibration
	TIME (SECONDS)	.00	27.80	27.80	245.54	771.79	2041.34
		MASS (THOUSANDS LBM)					
INITIAL MASS in RCS and ACCUMULATORS		773.52	773.52	773.52	773.52	773.52	773.52
ADDED MASS	PUMPED INJECTION	.00	.00	.00	129.64	466.38	1239.84
	TOTAL ADDED	.00	.00	.00	129.64	466.38	1239.84
*** TOTAL AVAILABLE***		773.52	773.52	773.52	903.16	1239.90	2013.36
DISTRIBUTION	REACTOR COOLANT	497.46	72.57	72.69	134.65	134.65	134.65
	ACCUMU-LATOR	276.06	197.75	197.63	.00	.00	.00
	TOTAL CONTENTS	773.52	270.32	270.32	134.65	134.65	134.65
EFFLUENT	BREAK FLOW	.00	503.18	503.18	757.90	1094.64	1867.80
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	503.18	503.18	757.90	1094.64	1867.80
TOTAL ACCOUNTABLE		773.52	773.50	773.50	892.55	1229.30	2002.45

TABLE 6.2.1-26B
WATTS BAR NUCLEAR PLANT UNIT 1
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SAFETY INJECTION - ENERGY BALANCE

		Start of Accident	End-of-Blowdown	Bottom of Core Recovery	End of Reflood	Broken Loop SG Equili-bration	Intact Loop SG Equili-bration
TIME	(Seconds)	.00	27.80	27.80	245.54	771.79	2041.34
		ENERGY (MILLION BTU)					
INITIAL ENERGY	IN RCS, ACCUM, & SG	852.47	852.47	852.47	852.47	852.47	852.47
ADDED ENERGY	PUMPED INJECTI0N	.00	.00	.00	9.47	34.06	96.83
	DECAY HEAT	.00	8.50	8.50	32.58	76.98	160.12
	HEAT FROM SECON-DARY	.00	.48	.48	.48	5.24	15.65
	TOTAL ADDED	.00	8.98	8.98	42.53	116.27	272.60
TOTAL AVAILABLE		852.47	861.45	861.45	895.00	968.74	1125.07
DISTRIBUTION							
	REACTOR COOLANT	296.96	13.17	13.18	29.85	29.85	29.85
	ACCUM-ULATOR	27.46	19.67	19.66	.00	.00	.00
	CORE STORED	25.94	14.51	14.51	3.98	3.88	3.63
	PRIMARY METAL	154.76	147.14	147.14	120.76	80.13	55.07
	SECOND-ARY METAL	66.60	67.08	67.08	60.43	46.85	28.39
	STEAM GENERAT 0R	280.76	283.33	283.33	250.37	193.10	123.42
	TOTAL CONTENT S	852.47	544.90	544.90	465.39	353.80	240.37
EFFLUEN T	BREAK FLOW	.00	315.96	315.96	417.48	602.81	858.26
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUEN T	.00	315.96	315.96	417.48	602.81	858.26
TOTAL ACCOUNTABLE		852.47	860.86	860.86	882.87	956.61	1098.63

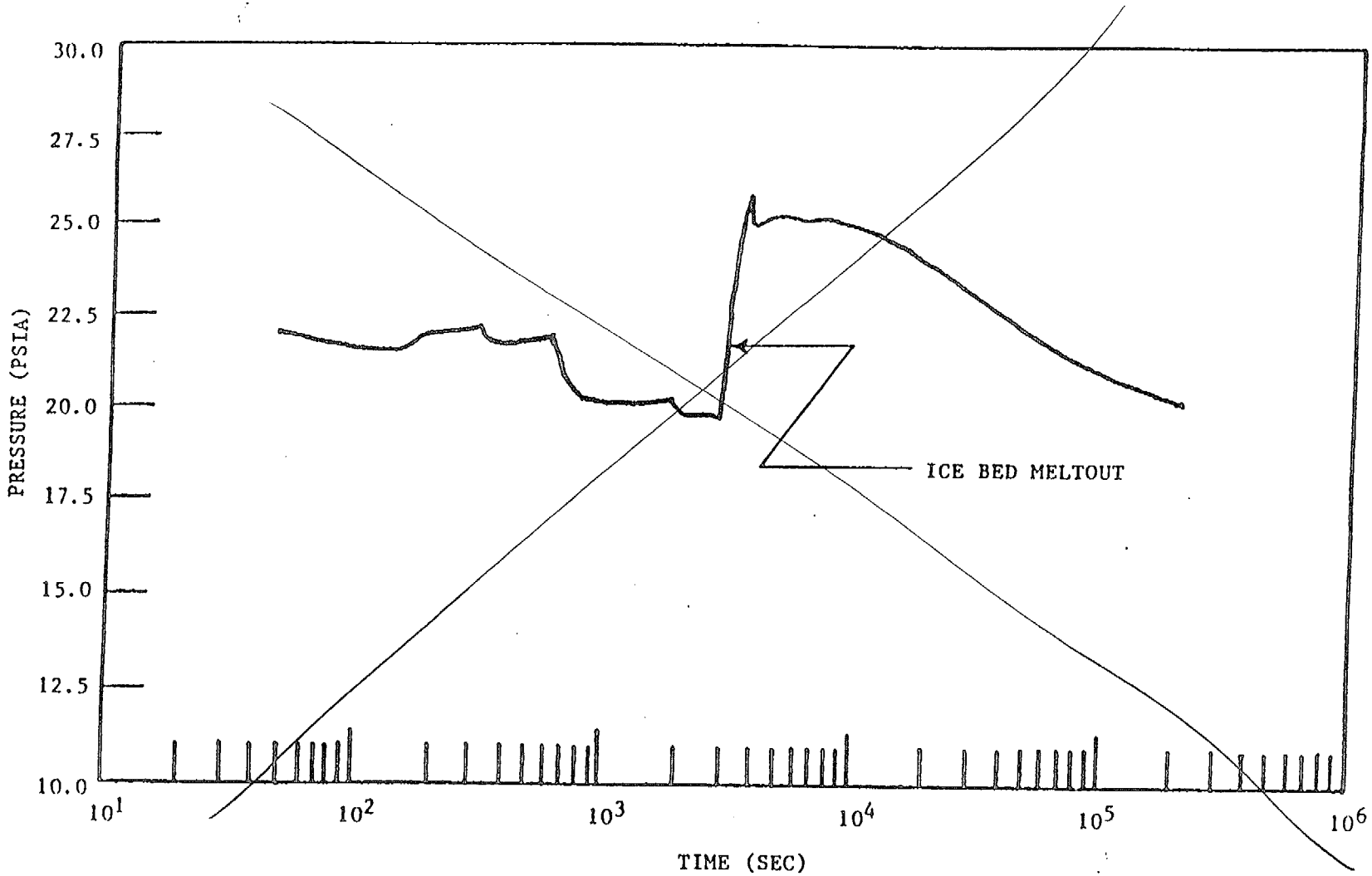

FIGURE 6.2.1-1. PRESSURE VS. TIME

Containment Pressure (psig)

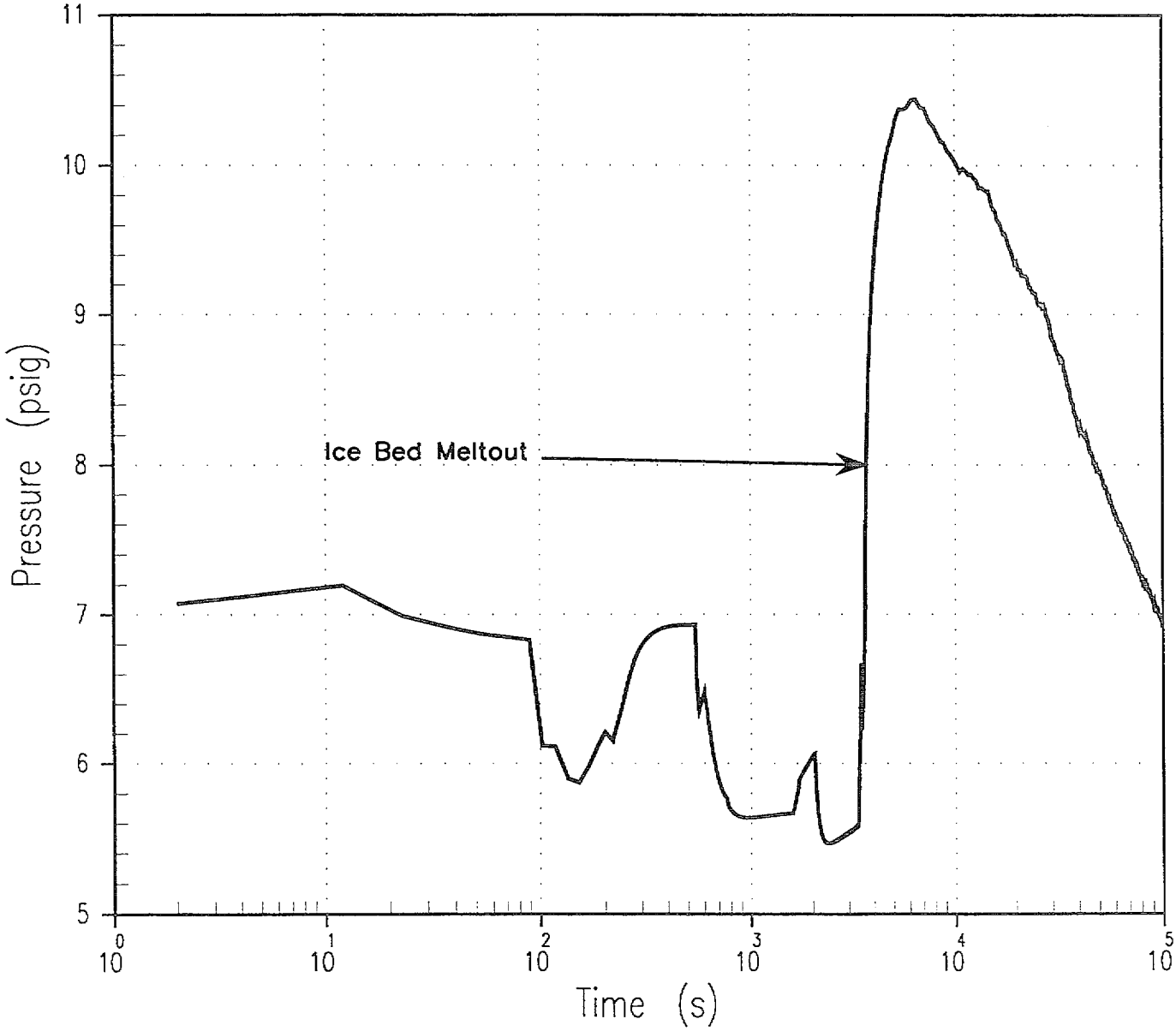

Figure 6.2.1–1

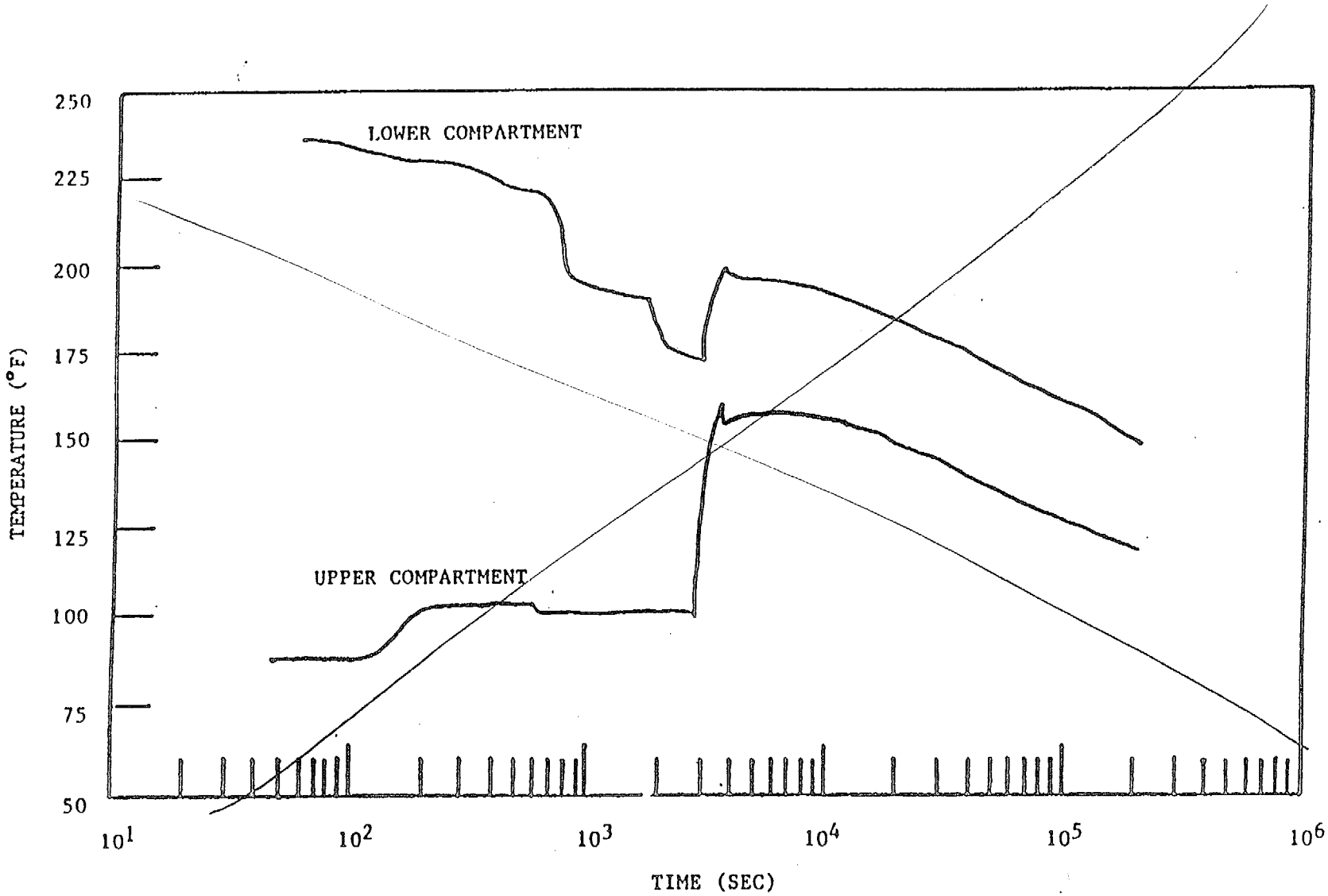

FIGURE 6.2.1-2. TEMPERATURE VS. TIME

Upper & Lower Compartment Temperatures (F)

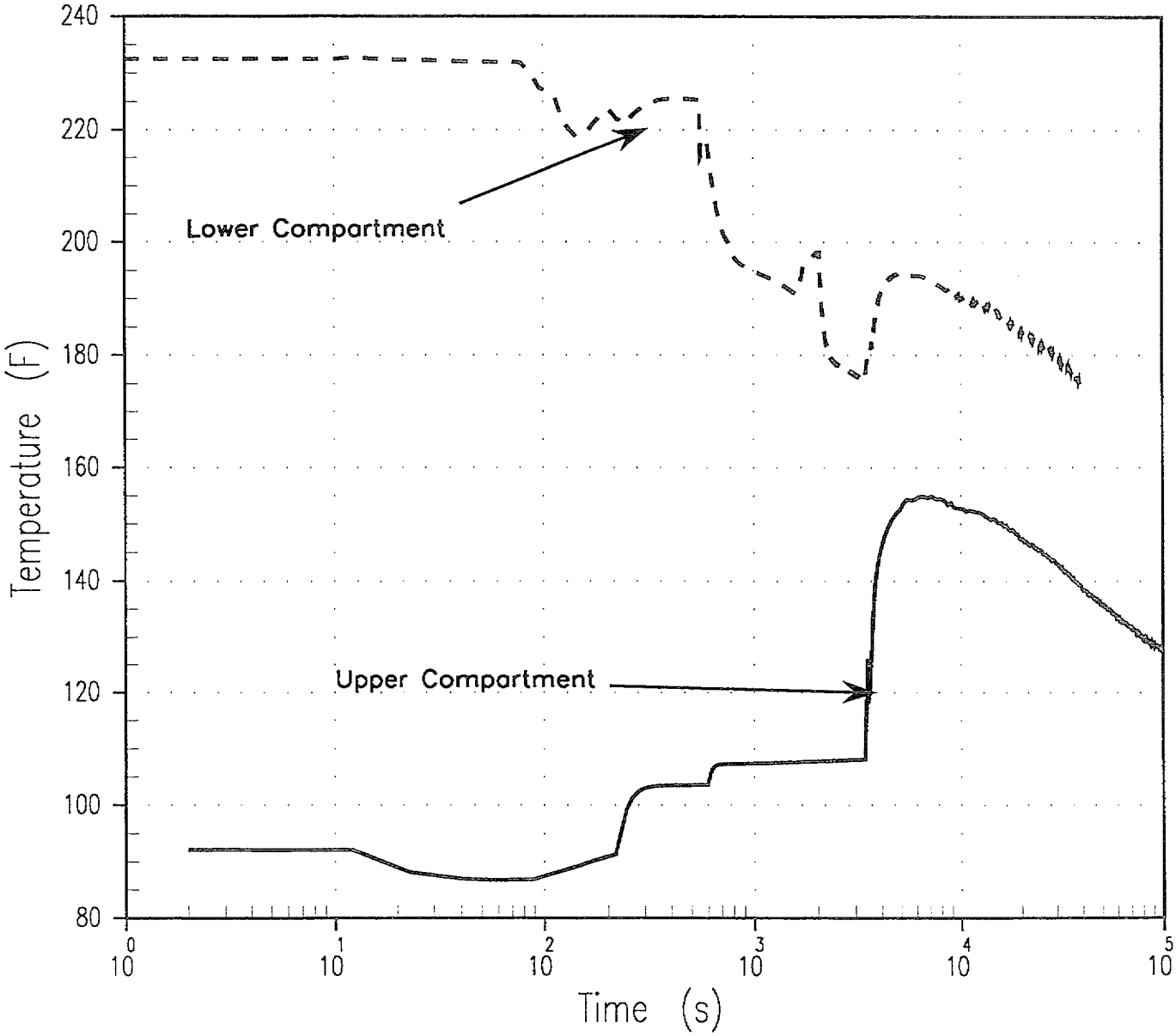

Figure 6.2.1-2

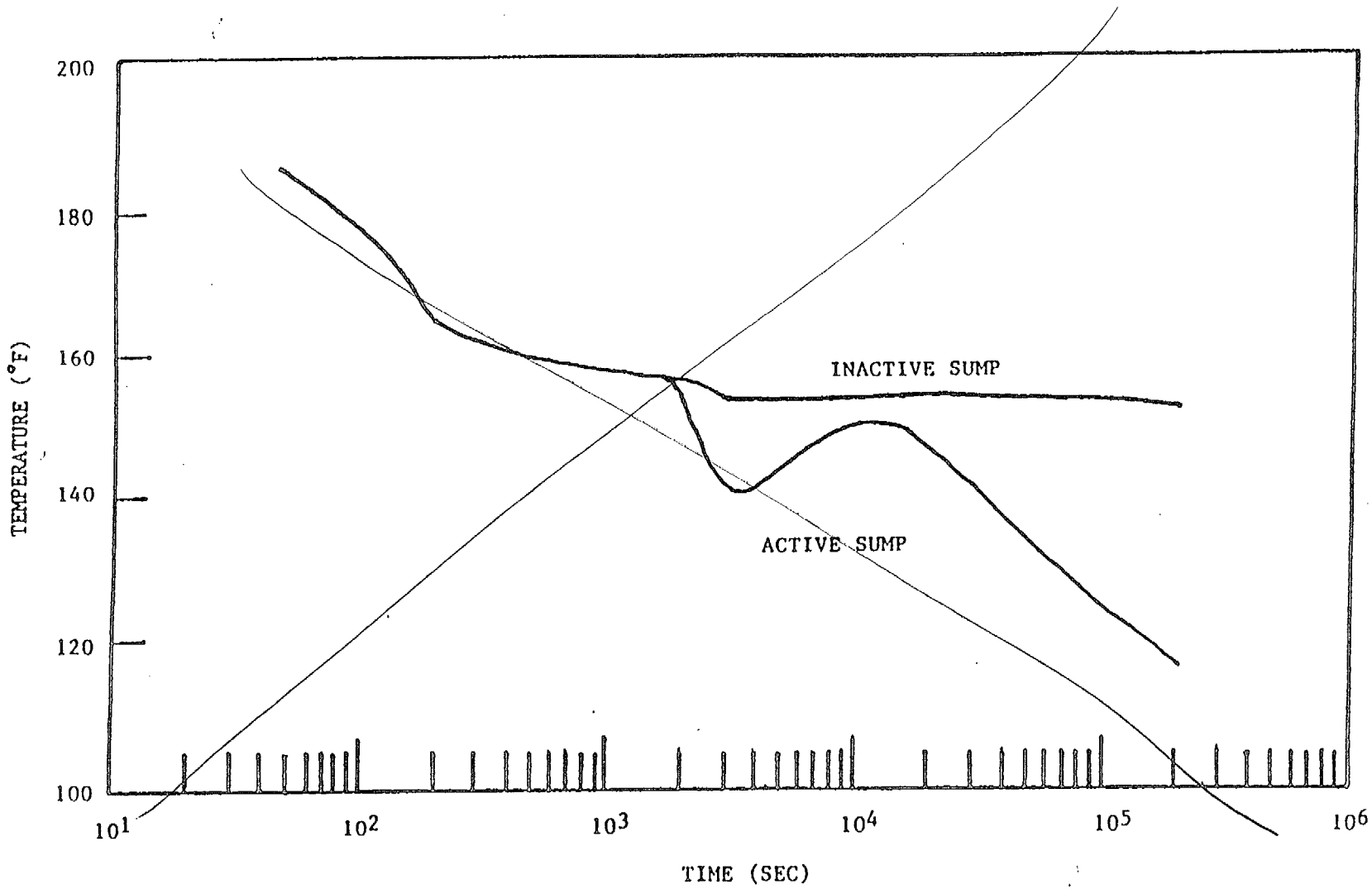

FIGURE 6.2.1-3. ACTIVE AND INACTIVE SUMP TEMPERATURE TRANSIENTS

Active Sump and Inactive Sump Temperature (F)

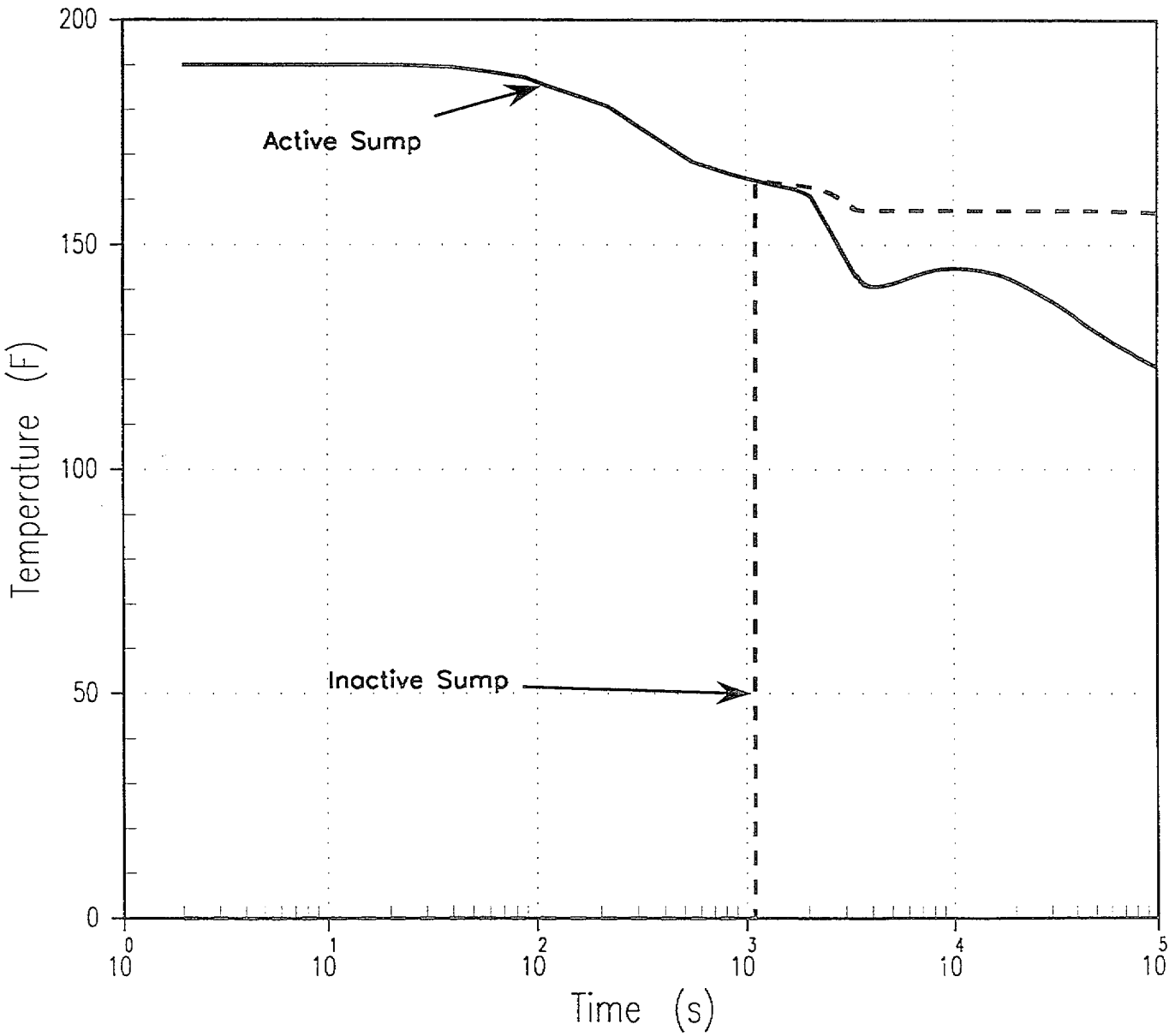

Figure 6.2.1–3

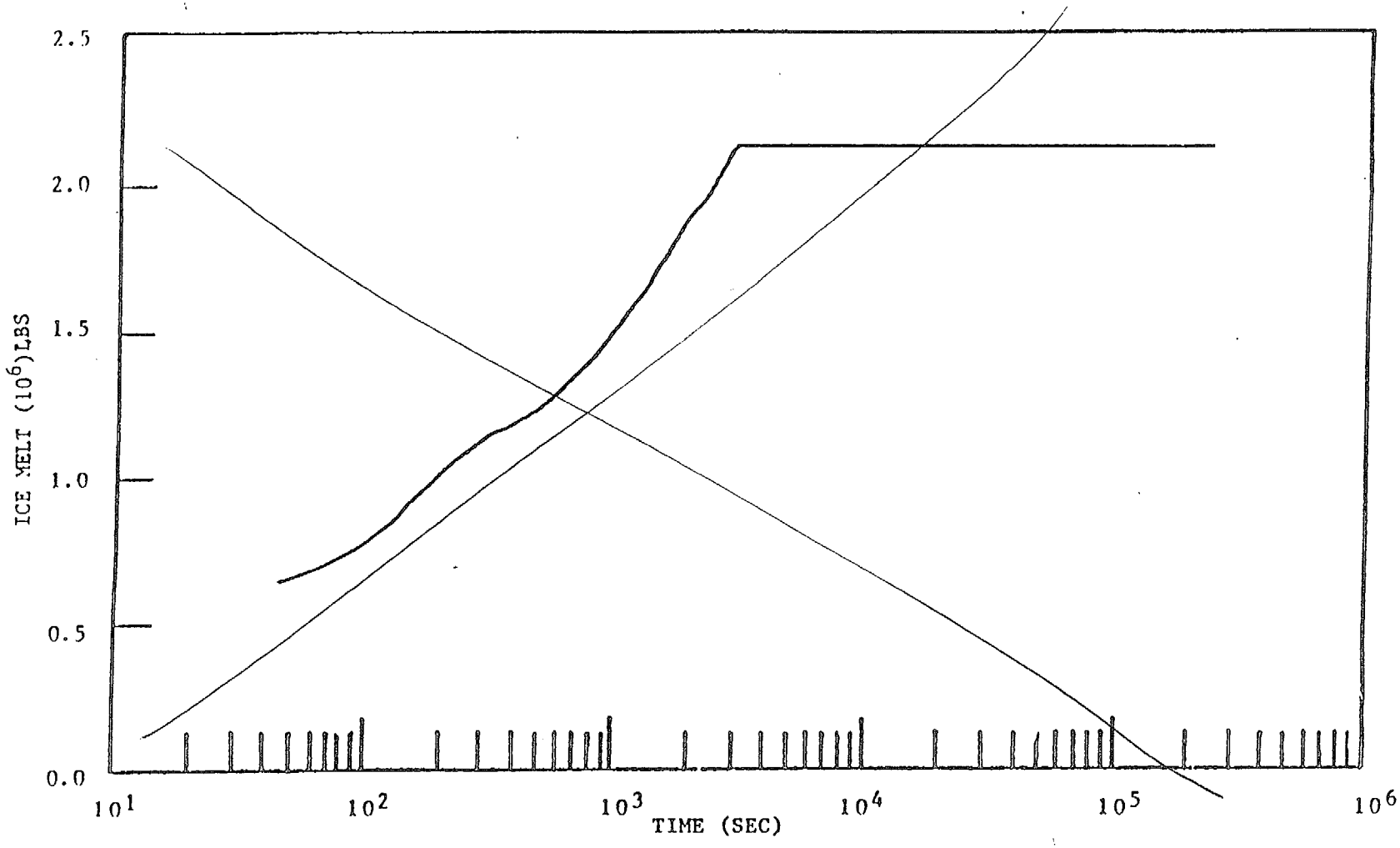

FIGURE 6.2.1-4. ICE MELT TRANSIENT

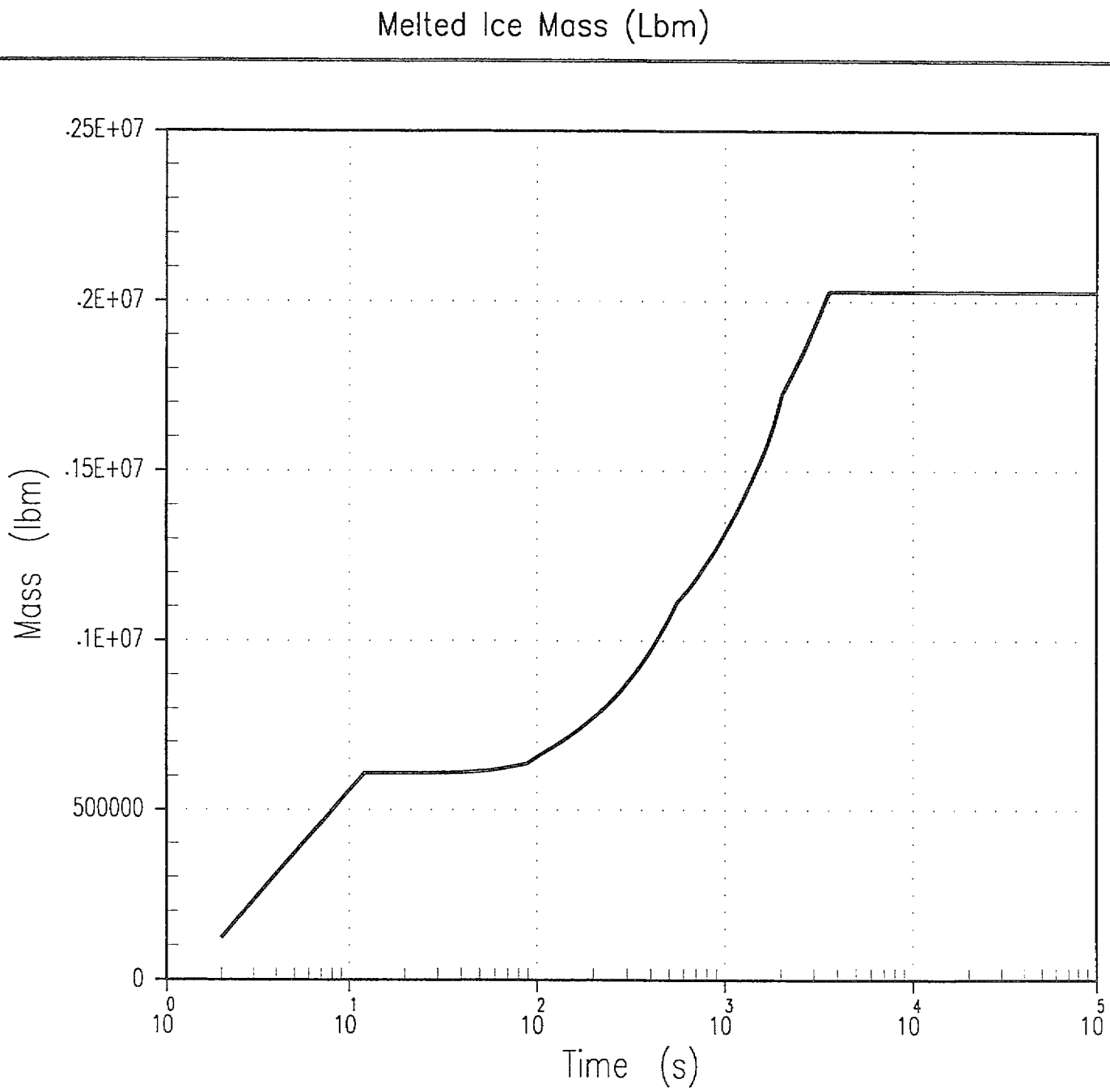
Melted Ice Mass (Lbm)
.25E+07
.2E+07
.15E+07
.1E+07
500000
0
Mass (lbm)
10^0
10^1
10^2
10^3
10^4
10^5
Time (s)

Figure 6.2.1–4

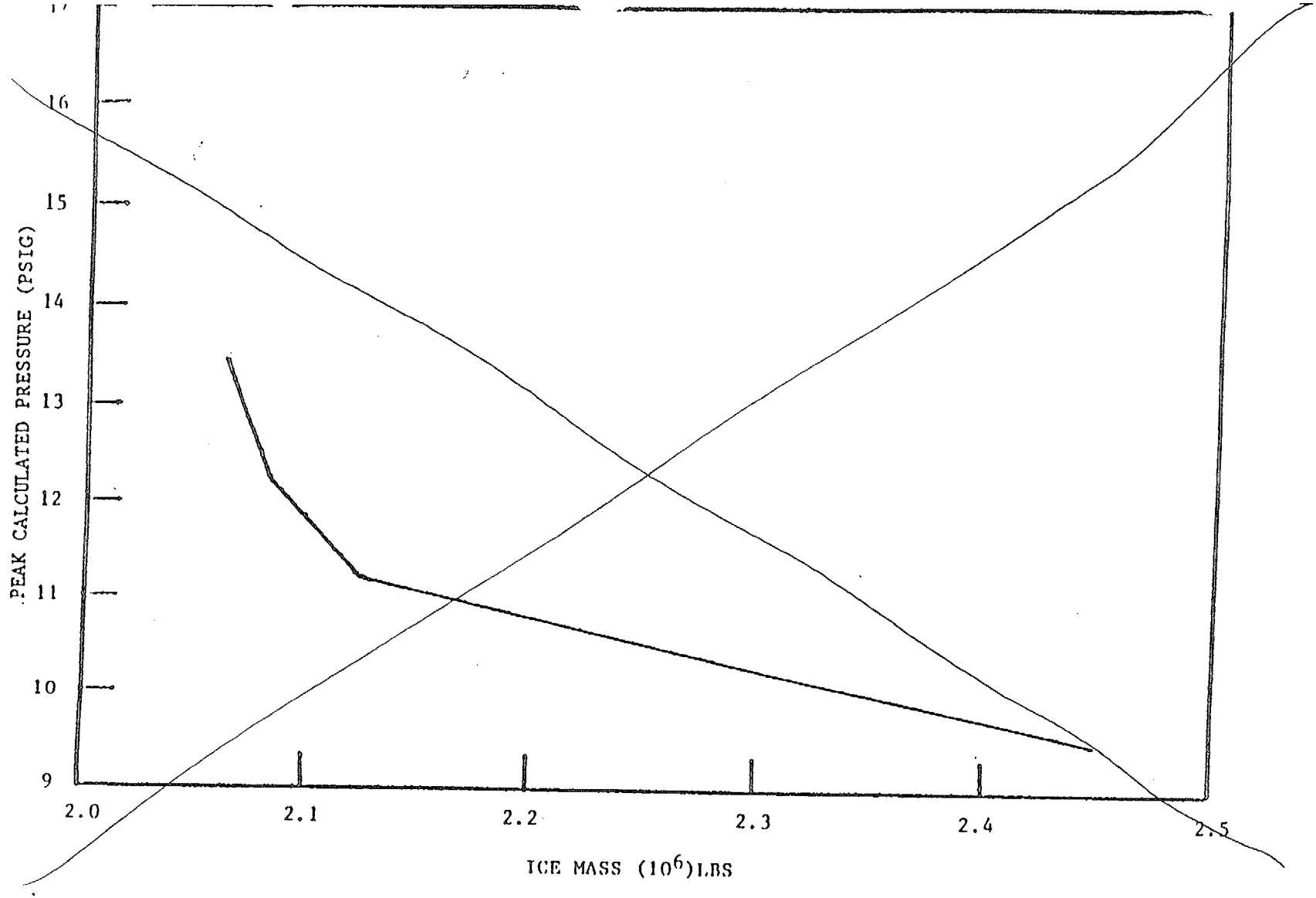

FIGURE 6.2.1-4A. ICE MASS VS. PRESSURE

delete

4. Basket Loading

The ice baskets are capable of being loaded by a pneumatic ice distribution system. The baskets contain a minimum of ~~2.125~~ 2.029375 x 10^6 pounds of ice.

5. External Basket Design

The baskets are designed to minimize any external protrusions which would interfere with lifting, weighing, removal and insertion.

6. Basket Coupling

Baskets are capable of being coupled together in 48-foot columns.

7. Basket Couplings and Stiffening Rings

Couplings or rings are located at 6 feet intervals along the basket and have internal inserts to support the ice from falling down to the bottom of the ice column during and after a DBA and/or SSE.

Design and Test Loads

The minimum test and basic design loads are given in Table 6.7-2.

6.7.4.2 System Design

The ice condenser is an insulated cold storage room in which ice is maintained in an array of vertical cylindrical columns. The columns are formed by perforated metal baskets with the space between columns forming the flow channels for steam and air. The ice condenser is contained in the annulus formed by the containment vessel wall and the crane wall circumferentially over a 300° arc.

The ice columns are composed of four baskets approximately 12 feet long each, filled with flake ice. The baskets are formed from a 14 gage (.075) perforated sheet metal, as shown in Figure 6.7-8. The perforations are 1.0 in. x 1.0 in. holes, spaced on a 1.25-inch center. The radius at the junction of the perforation is 1/16 inch. The ice basket material is made from ASTM-569 which is a commercial quality, low carbon steel. The basket component parts are corrosion protected by a hot dip galvanized process. The perforated basket assembly has an open area of approximately 64% to provide the necessary surface area for heat transfer between the steam/air mixture and the ice to limit the containment pressure within design limits. The basket heat transfer performance was confirmed by the autoclave test.

The crane travels on two circular rails that run through the ice condenser area as shown in Figure 6.7-11. The circular diameters of the rails are 95 and 109 feet. The top flange plate and rail section are continuously welded to the web plate under controlled conditions. The top flange and web plates are A-441 steel heat treated and normalized, fine grain practice, and the lower rail section is special analysis steel with a hard non-peening rolling surface.

6.7.5.3 Design Evaluation

The crane rails and supporting structures are analyzed as a part of the top deck structure (see Section 6.7.10). All stresses were maintained within limits prescribed in the design criteria, Section 6.7.16, for all design conditions defined in Section 6.7.5.1.

6.7.6 Refrigeration System

6.7.6.1 Design Basis

Function

The refrigeration system serves to cool down the ice condenser from ambient conditions of the reactor containment and to maintain the desired equilibrium temperature in the ice compartment. It also provides the coolant supply for ice machines A, B, and C during ice loading. The refrigeration system additionally includes a defrost capability for critical surfaces within the ice compartment.

During a postulated loss-of-coolant accident the refrigeration system is not required to provide any heat removal function. However, the refrigeration system components which are physically located within the containment must be structurally secured (not become missiles) and the component materials must be compatible with the post-LOCA environment.

Design Conditions

1. Operating Conditions

   See individual component sections:

   A. Floor cooling - Section 6.7.1

   B. Air handling units (AHUs) - Section 6.7.7

2. Performance Requirements

   A. The mandatory design parameters that relate to refrigeration performance are:

      i. Nominal initial total weight of ice in columns 3.0×10^6 lbs

      ii. Minimum total weight of ice in columns ~~2.125~~ $\times 10^6$ lbs
      ~~2.029~~
      2.028375

TABLE 6.7-18

(Sheet 2 of 2)

REFRIGERATION SYSTEM PARAMETERS (cont'd)

2.4 Refrigeration Medium (glycol) - UCAR Thermofluid 17 or equal

Concentration, ethylene glycol in water - 50 weight % or 47.8 volume %

At temperature:	-5°F	0°F	100°F
Specific gravity	1.083	1.082	1.056
Absolute viscosity (centipoises)	25.0	20.5	2.3
Kinematic viscosity (centistokes)	23.1	18.9	2.18

3.0 Ice Condenser (per one containment unit)

3.1 Ice Bed

Amount of ice initially stored per unit, nominal	3.0×10^6 lbs
Minimum amount of ice	2.029375 → ~~2.125~~ $\times 10^6$ lbs –
Ice displacement per year, design objective	2%
Design predicted ice displacement per year to wall panels for normal operation	<0.3%
Ice melt during maximum LOCA, calculated, approx. SEE SECT 6.2.1	~~10^6 lbs.~~
Temperature of ice & static air	15-20°F nominal
Pressure at lower doors due to cold head, nominal	1 psf
Inlet opening pressure	1 psf

3.2 Air Handling Units - 30 dual packages installed per Containment

Refrigeration requirements per containment, calculated, nominal	51.5 tons
Gross capacity per dual package rated	2.5 tons
Glycol entering temperature, approx.	-5°F
Glycol exit temperature, approx.	1°F
Glycol flow per air handler (1/2 package)	6 gpm nominal
Total glycol flow, 30 x 2 x 6	360 gpm nominal
Glycol pressure drop, estimated	50 feet
Air blower head	2' H_2O
Air entering temperature, estimated	15°F
Air exit temperature	10°F nominal

APPENDIX B - TECH SPEC MARKUPS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.11.2	Verify total weight of stored ice is ≥ ~~2,403,800~~ 2,158,000 lb by: a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains ≥ ~~1230~~ 1110 lb of ice; and b. Calculating total weight of stored ice, at a 95% confidence level, using all ice basket weights determined in SR 3.6.11.2.a.	18 months
SR 3.6.11.3	Verify azimuthal distribution of ice at a 95% confidence level by subdividing weights, as determined by SR 3.6.11.2.a, into the following groups: a. Group 1-bays 1 through 8; b. Group 2-bays 9 through 16; and c. Group 3-bays 17 through 24. The average ice weight of the sample baskets in each group from radial rows 1, 2, 4, 6, 8, and 9 shall be ≥ ~~1230~~ 1110 lb.	18 months
SR 3.6.11.4	Verify, by visual inspection, accumulation of ice on structural members comprising flow channels through the ice bed is ≤ 15 percent blockage of the total flow area for each safety analysis section.	18 months

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.11 Ice Bed

BASES

BACKGROUND

The ice bed consists of over ~~2,403,800~~ 2,158,000 lbs of ice stored in 1944 baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment, which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal plant operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets contain the ice within the ice condenser. The ice bed is considered to consist of the total volume from the bottom elevation of the ice baskets to the top elevation of the ice baskets. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

The weighing program is designed to obtain a representative sample of the ice baskets. The representative sample shall include 6 baskets from each of the 24 ice condenser bays and shall consist of one basket from radial rows 1, 2, 4, 6, 8, and 9. If no basket from a designated row can be obtained for weighing, a basket from the same row of an adjacent bay shall be weighed.

The rows chosen include the rows nearest the inside and outside walls of the ice condenser (rows 1 and 2, and 8 and 9, respectively), where heat transfer into the ice condenser is most likely to influence melting or sublimation. Verifying the total weight of ice ensures that there is adequate ice to absorb the required amount of energy to mitigate the DBAs.

If a basket is found to contain < ~~1236~~ 1110 lb of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The average weight of ice in these 21 baskets (the discrepant basket and the 20 additional baskets) shall be ≥ ~~1236~~ 1110 lb at a 95% confidence level.

Weighing 20 additional baskets from the same bay in the event a Surveillance reveals that a single basket contains < ~~1236~~ 1110 lb ensures that no local zone exists that is grossly deficient in ice. Such a zone could experience early melt out during a DBA transient, creating a path for steam to pass through the ice bed without being condensed. The Frequency of 18 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 18 month Frequency, the weight requirements are maintained with no significant degradation between surveillances.

(continued)

APPENDIX C - RESPONSES TO PREVIOUS NRC RAIs

Below are the responses to a Sequoyah Nuclear Plant request for additional information (Reference 1) as they apply to the Watts Bar WCAP-15699, Revision 1 analysis."

Question 1:

In WCAP-12455 Rev.1 containment integrity analysis, it is indicated that this analysis utilized revised input assumptions which eliminated analytical conservatism from the present analysis. Please provide the comparision and basis for the difference in assumptions for the following:

a) Core Stored Energy
b) Decay Heat Release
c) Bounding Condition for steam generator equilibrium and depressurization to reflect actual plant conditions.

Application to Watts Bar analysis in WCAP-15699 Rev.1.

Response

Item a.

The current analysis for Watts Bar is based on a core stored energy of 5.4 full power seconds. The new analysis reported in WCAP-15699 Rev.1, being based on improved core predictive models, used a core stored energy of 4.23 full power seconds. The reduction is a result of the improved predictive models. The core stored energy is still based on a full core (193 assemblies) of fresh fuel. Thus no credit for core burnup has been taken.

Item b.

The current analysis for Watts Bar is based on the ANS 1971 decay heat standard. The new analysis uses the ANSI/ANS-5.1 1979 Decay Heat Standard, including 2 sigma uncertainty. This is a feature of the new WCAP-10325-P-A methodology. As provided for in WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version," page 2-10, the decay heat was further modified to account for a maximum end of cycle core average burnup of 45,000 Mwd/MTU. Table 2-2 of WCAP-15699 Rev.1 provides a detailed table of the decay heat curve used in the analysis.

Item c.

The WCAP-15699 Rev.1 analysis is the first application of WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version," to the Watts Bar Nuclear Plant. Thus, bounding values were chosen for the steam generator equilibrium depressurization.

Question 2:

In the WCAP-12455 Rev.1 report, it is indicated that the analysis accounted for the effects of other plant changes that Westinghouse is aware of. Please list the changes and provide a comparison and basis for such changes between the two analyses.

Response

All plant and system changes are adequately indicated in the WCAP and Appendix A, "FSAR Markups". However, as discussed in References 3 and 4, additional modeling refinements were applied in the Reference 2 analysis. The following will address these modeling refinements for the Watts Bar analysis in WCAP-15699 Rev.1, Reference 5.

1) No credit for steam removal from the steam generators prior to turbine throttle valve closure was taken in the Reference 5 analysis. Instantaneous closure of the turbine throttle valve at accident initiation was assumed.

2) Item 10 on pg 3 of Reference 4 states that the metal mass was removed from the steam generators (similarly to the Reference 2 analysis for Sequoyah). Item 10 states,"... The steam generator metal mass was modeled to include only the portion of the steam generators (SG) which is in contact with the fluid on the secondary side. Portions of the SGs such as the elliptical head, upper shell and misc. internals have poor heat transfer due to location. The heat stored in these areas available for release to containment will not be able to effectively transfer energy to the RCS, thus the energy will be removed at a much slower rate and time period (>10000 seconds)...".

3) The Reference 5 analysis used the latest computer code for calculation for the mass and energy releases. With this version, an increased number of data points (up to 40) for the mass and energy release rates (specifically during the post-blowdown phase) are generated. This improved segmental representation of the data has resulted in some reduction in the mass and energy releases when compared to the earlier code version used in present analysis.

It should be noted that the Reference 5 analysis incorporates only items considered to be within the bounds of the currently approved licensing basis model.

References:

1) NRC Letter to TVA dated March 21, 1997, "Request for Additional Information - Technical Specification Change Request TS 96-02 for Sequoyah Nuclear Plant Units 1 and 2 (TAC Nos. M96592 and M96593)

2) WCAP-12455, Revision 1, "Tennessee Valley Authority, Sequoyah Nuclear Plant Units 1 and

2,Containment Integrity Analyses For Ice Weight Optimization Engineering Report," September 1995

3) TVA-92-027, John W. Irons (W) to Mr. Joseph Valente (TVA), "Sequoyah Nuclear Plant, Responses to NRC questions - Long-Term Containment Integrity Evaluation,"March 27, 1997

4) Letter from R. H. Shell (TVA) to US NRC, "Sequoyah Nuclear Plant (SQN) - Response for Additional Information Regarding Technical Specification Change 96-02," March 28, 1997

5) WCAP-15699 Rev.1, Non-Proprietary Class 3, "Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1 Containment Integrity Analyses for Ice Weight Optimization Engineering Report," August 2001

APPENDIX D - TECHNICAL REQUIREMENTS MANUAL MARKUPS

BASED ON WESTINGHOUSE LETTER WAT-D-10940 Rev.1, John W Irons (W) to Mr. J. C. Kammeyer (TVA), Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, Watts Bar Unit One TRM & FSAR Markups - Post-LOCA Sump pH," 08/16/01

Attached are markups of the Watts Bar Technical Requirements Manual Bases to address the reduction in the post-LOCA sump pH which results from the proposed ice weight reduction (WCAP-15699 Rev.1). Sump pH analyses performed for the Tritium program with a reduced ice weight, as well as increased boron in the RWST and the Accumulators, demonstrated that the minimum sump pH would be greater than 7.5. In addition, for that program it was determined that the resulting sump pH will not create a corrosion issue, and that it is acceptable with respect to minimizing the potential for chloride induced stress corrosion cracking and maintaining iodine retention in the sump solution. The attached markups are the same as those previously proposed for the Tritium program with respect to sump pH. It should be noted that these changes are also appropriate for the Tritium program.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.5.1 (continued)

is greater than or equal to 60°F. With ambient air temperature greater than 60°F, the RWST solution temperature should not decrease below this limit, therefore, monitoring is not required.

TSR 3.1.5.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is ≥ 2,500 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between ~~8.0~~ 7.5 and ~~10.5~~ 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.3

This surveillance requires verification every 7 days that the RWST borated water volume is ≥ 62,900 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between ~~8.0~~ 7.5 and ~~10.5~~ 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience. The 62,900 gallon volume requirement includes 11,100 gallons for shutdown margin, adjustments for minimum safety limit level in the RWST, and adjustments for instrument error.

TSR 3.1.5.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does not account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.5.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is between 6,120 ppm and 6,990 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between ~~8.0~~ 7.5 and ~~10.5~~ 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.6

This surveillance requires verification every 7 days that the BAT borated water volume is ≥ 3,800 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between ~~8.0~~ 7.5 and ~~10.5~~ 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

2. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar, Unit 1," Revision 00, April 1993.

3. TVA Calculation, EPM-PDM-071197, Revision 0, "Boric Acid Concentration Analysis for BAT and RWST."

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.2 (continued)

boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.6.3

This surveillance requires verification every 7 days that the RWST borated water volume is within the required limit of ≥ 370,000 gallons (value does not account for instrument error). This will ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.6.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does not account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

TSR 3.1.6.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is in accordance with Figure 3.1.6 of TR 3.1.6. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a

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