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January 7, 1994

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Braidwood Station Units 1 and 2
Response to Request for Additional Information on Environmental Qualification of Okonite Tape Splices for Local Submergence at Braidwood Station dated November 17, 1993,
NRC Docket Nos. 50-456 and 50-457

References:
1. T. Simpkin letter to A. B. Davis dated November 21, 1992, transmitting Commonwealth Edison's Response to Unresolved Items Identified in Inspection Report 456(457)/89018
2. R. Assa letter to D. Farrar dated November 17, 1993, Requesting Additional Information on Environmental Qualification of Okonite Tape Splices for Local Submergence

Reference 1 provided the initial response to Unresolved Items 50-456/89018-04 and 50-457/89018-04. Reference 2 provided the Nuclear Regulatory Commission's Request for Additional Information (RAI) following their review of reference 1.

Attached is Commonwealth Edison Company's (CECo) response to the RAI pertaining to the environmental qualification of Okonite tape splices. Also, attached are excerpts from the Braidwood UFSAR and Braidwood Station procedure, BwHP 4006-008, "Repairing, Determinating, Terminating, Splicing, Taping, Cable Jacket Repair and Application of Ray Chem Kit on Cable," which may facilitate your review of the attached response.

CECo believes that the basis for establishing the environmental qualification of the Okonite T-95 tape splices is Byron/Braidwood UFSAR Section 3.11. As stated previously, sufficient basis exists to demonstrate that for low voltage power and control circuits, Okonite tape splices are environmentally qualified for a "local submergence" condition due to a postulated line break to 10CFR50.49, NUREG 0588 Category 1 and IEEE 323-1974 requirements.

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If you have any questions concerning this response please contact me at (708) 663-6484.

Sincerely,

Denise M. Saccomando
Nuclear Licensing Administrator

Attachment

cc: R. Assa, Braidwood Project Manager - NRR
S. Dupont, Senior Resident Inspector - Braidwood
J. Martin, Regional Administrator - Region III
Office of Nuclear Facility Safety - IDNS

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1. Duration of Submergence: Regulatory Guide 1.89, Section 3.a states: "Electric equipment that could be submerged should be identified and qualified by testing in a submerged condition to demonstrate operability for the duration required. Analytical extrapolation of results for test periods shorter than the required should be justified."

During the September 13, 1993, phone call, the licensee stated that the splices must operate post-LOCA for up to one year. However, the test reports submitted do not document justification that a 5-minute immersion in tap water is equivalent to one year of submerged operation. They do not show that the splices were actually submerged for 25 or 72 hours during the tests as claimed by the licensee, or that these times are representative of one year post-LOCA operation. The Licensee also stated that the 72-hour submergence test did not envelop in-containment applications.

The licensee should provide documentation to justify that the splices will be operable for the worst case environment for these splices.

The qualification of the Okonite tape splices for postulated periods of local submergence was addressed in the previous submittal (see Sections 7.2 through 9.0, Pages 7 to 10 of Calculation Report No. 92-001S, dated November 1992 submitted with the previous response). However, this request for more information regarding the qualification of the Okonite tape splices at Braidwood appears to focus primarily on the following aspects and concerns about the splices and local submergence conditions in general:

- duration of local submergence based on the accident and post-accident conditions, and

- representation of local submergence conditions by the type testing.

The 5 minute post-LOCA high voltage withstand test is performed as required by IEEE Standard 383-1974. As stated, a 5 minute post-LOCA high voltage withstand test does not envelop extended periods of postulated local submergence. However, performing this test on specimens that have been pre-aged to their end-of-life conditions does demonstrate the insulation's integrity because higher than normal voltage stresses are applied combined with the immediate application of a ground plane on the insulation surface. Any faulty point in the insulation (weak point) would be easily detected during this test. A specimen that successfully passes such a test demonstrates that the insulation material has retained its capability to insulate for both normal conditions and conditions expected during and after an accident.

In the response previously submitted, the identification of worst-case local submergence was determined to occur in the pull boxes located inside containment which have no weep holes. This was detailed in Figure 1 of the previous submittal. The potential for local submergence results from line break scenarios and only trace amounts of spray ingress past degraded pull box cover gaskets. Because of the installation methods of electrical conduit and enclosure systems at Braidwood Station, the installed configurations

essentially preclude the ingress and accumulation of significant amounts of directly impinging spray but are open enough for steam entry and consequent steam condensation.

A discussion of the potential line break accidents inside containment at Braidwood is given in Section 6.2 of the UFSAR. Per Table 6.2-1 of the Braidwood UFSAR, an MSLB combined with failure of a steamline isolation valve is shown to be the limiting line break scenario for inside containment. At the onset of this line break, all surfaces of pull boxes, conduits, cables and splices are initially at a lower temperature (approximately 120°F per Table 6.2.2 of the Braidwood UFSAR) than the corresponding saturation temperature of the partial pressure of the air/steam phase of the containment atmosphere (reference Figure 6.2-13 of the UFSAR), hence, condensation will take place on these surfaces during their heating by the steam blowdown. The air/steam phase of the containment atmosphere reaches its peak temperature of 318°F at 59 seconds into the scenario (reference Tables 6.2-1 and 6.2-9, Braidwood UFSAR). During this time, the steam is superheated and has a lower moisture content (compare Figures 6.2-13 and 6.2-14 of the Braidwood UFSAR). In addition, steam conditions only exist for less than one hour following the line break, at which time the containment atmosphere only contains an air/water phase (reference the containment temperature and pressure response curves given in Section 6.2 of the Braidwood UFSAR). Accordingly, the condensate formed within the conduit system will be negligible. Furthermore, the condensate will be held in place on the various surfaces or will be trapped at points where the cables press against the conduit walls due to surface tension effects in the thin film of water. Therefore, minimal amounts of water (condensate) will migrate through the conduits into those pull boxes which have no weep holes. At most, as shown in the previous submittal, the maximum potential height of water in the pull box is 3/8", limited by the bottom lip of the pull box.

At Braidwood Station, the pull boxes are typically wall mounted. When the containment atmosphere starts to depressurize, the residual temperature of the wall maintains the pull box, cable and splice system at temperatures greater than the corresponding saturation temperature of the containment atmospheric pressure for days following a line break accident (the containment steel liner/concrete wall system attains a peak temperature of 208°F, long-term, see Table 6.2-66 of the Braidwood UFSAR). Thus, conditions are favorable for the evaporation of accumulated condensate. Additionally, the as installed configuration does not necessarily allow the splices to be in full contact with the bottom plate of the box where all of the condensate collects.

Local submergence conditions do not occur immediately following the line break accident, but develop over an extended time period. First, at the onset of the line break, condensate will form on all surfaces, and some time later when sufficient amounts of condensate form, it will accumulate in pull boxes. The initial period prior to the occurrence of local submergence is precisely simulated by any type tests using steam as the heating source, such as those included in the previous submittal. During these tests, the splice specimens were enclosed in free standing junction boxes and placed in a steam-heated testing chamber. Although these tests were representative of an actual installation, they are considered overly conservative in representing them for post-line break local submergence

conditions. Following the initial heating by steam, condensation, and subsequent accumulation, some splice specimens were subjected to local submergence conditions for the remainder of the 25 or 72 hour tests. The test splice specimens were exposed to extended periods of local submergence because the specimens (and junction boxes) essentially cooled-down at the same rate as the ambient conditions of the test chamber. Thus, the condensate formed and collected during the test was not able to evaporate as readily as in a typical Braidwood installation. Hence, the 25 hour or the 72 hour type testing programs previously reviewed and submitted sufficiently qualifies the subject Okonite taped splices for the postulated local submergence conditions at Braidwood Station.

Therefore, the qualification of the subject Okonite tape splices at Braidwood Station for local submergence is accomplished through the following attributes:

- Condensing steam conditions at Braidwood exist only for brief durations (less than 1 hour).

- Minimal amounts of condensate may be formed and collected in pull boxes having no weep holes, but will be limited to a maximum of 3/8".

- Only partial submergence of the splices is expected because of the lip height and installed configurations.

- From 2 hours after and beyond following a worst-case line break at Braidwood Station, evaporative conditions are prevalent, thus reducing postulated local submergence conditions to a duration less than approximately 24 to 36 hours.

- Overly conservative, actual submergence testing durations of nearly 72 hours for various Okonite T-95 and No. 35 tape splice specimens having configurations more susceptible to grounding faults than those installed at Braidwood (not as many insulating tape layers and/or amount of overlap).

Therefore, the subject Okonite tape splices are considered qualified for postulated local submergence conditions through the type testing and evaluation documentation already submitted to the NRC staff.

2. Chemical Solution for submergence: NUREG-0588, Section 2.2.(8) states: "Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the on-site spray systems actuate."

The licensee has not shown that the splices will operate in post-accident submergence in the same solution they would be exposed to during a design basis event. Tap water or condensate were not shown to be equivalent to the chemical solution. The composition of the 25 hour chemical spray was not shown to be equivalent to the Braidwood environment.

The licensee should provide documentation to justify that the chemical solution that splices could be exposed to in the plant is represented by test conditions.

Per the Byron/Braidwood Stations Environmental Qualification Program Design Basis Document (Sargent & Lundy File No. PMED-BB-EQ-DBD-00, Revision 00) the post-accident chemical spray for inside containment consists of the following composition:

> 2000 ppm Boron as boric acid buffered with sufficient Sodium Hydroxide (NaOH) to bring the pH of the solution to the range of 8.5 to 10.5 at a spray flow rate of 0.15 gpm/Ft^2.

From the previous submittal, a test report was reviewed for its chemical spray composition and submergence of Okonite taped splices (Appendix C, of the submittal report). In this report (Wyle Laboratories Test Report No. 17947-01, Page VI-26), the following chemical spray composition was used:

> 2500 ppm of Boron (as boric acid H_3BO_3) buffered with Sodium Hydroxide (NaOH) to obtain a pH of 10.7. This solution was maintained at a pH of 10.5 to 11.0 throughout the test duration of nearly 25 hours. The spray flow rate during the test was 0.15 gpm/Ft^2.

From the above, it is seen that the chemical spray composition was more caustic during the test than the postulated spray at Braidwood since it had a higher pH value (this also provides an opportunity for conducting greater leakage currents during the LOCA test since the specimens were energized). In addition, the splice specimens were submerged in this chemical solution for the duration of the spray test. Hence, the chemical spray conditions postulated for Braidwood were enveloped and represented by the test.

The validity of the high voltage withstand tests conducted post-LOCA using tap water are described below. Per U.S. Department of Health, Education, and Welfare, Public Health Service Drinking Water Standards, Revised in 1962, tap water may contain a maximum concentration of the following chemicals (typical of tap water in the United States):

Arsenic	0.01 mg/l
Chlorine	250 mg/l
Copper	1 mg/l
Carbon Chlorophyll Extracts	0.2 mg/l
Iron	0.3 mg/l
Magnesium	0.05 mg/l
Nitrate	45 mg/l
Phenols	0.001 mg/l
Sulfates	250 mg/l
Total Dissolved Solids	500 mg/l
Zinc	5 mg/l
Cyanide	0.01 mg/l

If a splice assembly had developed any potential leakage paths during the environmental qualification type testing, any one of the above conductive materials found in tap water would have provided current flow during the hi-pot testing.

This information is being provided to demonstrate that tap water has conductive properties in a similar manner to chemical solutions. However, since the test information already reviewed and discussed in the previous submittal (which was restated above) already demonstrates that the chemical spray solution of the test envelops the postulated spray exposure at Braidwood Station, the need to include any further discussion on the equivalence of tap water is not warranted.

3. Test Voltage: NUREG-0588, Section 2.2.(7) states: "Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability." Section 2.4.3.2 of IEEE Standard 383-1974 states: "The cable should function electrically throughout its exposure to the environmental extremes within the specified electrical parameters" (rated voltage and current).

The voltage used during the tests was insufficient to qualify the 4kV/600VAC/250VDC applications.

The licensee should provide documentation to show test voltages that are equivalent to the voltages used in the plant applications.

As discussed in the previous submittal, the installed Okonite taped splices are used in 4kV, 480VAC, and 250VDC power and 120VAC and 125VDC control applications. The 4kV power splices are not contained in junction or pull boxes; their only application is at the 4kV ECCS pump motors which are not subject to postulated local submergence. Hence, the 4kV in-line motor splices are not subjected to local submergence.

Other cables used at the station are rated for 600V. The highest potential that the installed cables experience is 480VAC (rms, phase-to-phase, or 277VAC rms phase-to-ground). Including a 10% margin from IEEE 323-1974, stipulates a testing value of 528VAC. The electrical loading of the test specimens as shown in Appendices A through D of the previous submittal is sufficient to qualify the installed configurations because the test voltages applied to the specimens represented a more significant voltage stress level than what is to be experienced during plant operation (note, the voltages listed below are AC rms, unless otherwise stated, which are equivalent to the same DC voltage level). The electrical loading of the test specimens during those tests which had submerged splice specimens are shown below:

Test	Voltage	Remarks
Wyle 17856-2 & 17856-3 (Appendix B of subm.)	528 VAC/132VAC	Energization circuit contained fuses for both excessive leakage and load currents for the 24 splice specimens in the test, 8 intentionally submerged.
Wyle 17947-01 (Appendix C of subm.)	633 VAC/137.5 VDC	Specimens 7.1, 8.1, and 9.1 were monitored for failure via the circuit load protection fuse. These were intentionally submerged and were continuously energized throughout the test. Total of 13 Okonite T-95 and No. 35 tape splice specimens in test.
Wyle 17961-01 (Appendix D of subm.)	305 VAC	Specimens with an "A" designation were held affixed to junction box bottom to ensure submergence. These specimens had leakage and load currents monitored continuously. Total of 63 splice specimens in test (21 intentionally submerged).

These voltages were maintained throughout the LOCA simulations. In all cases, the current or voltage source had an appropriately sized fuse for limiting the leakage current or for the connected loads (solenoid valves, etc.). Since these fuses did not open, it can be concluded that no excessive leakage currents were generated by the local submergence condition of the splices. The test set-up configuration of the fuse, cable, splice, and load are representative of typical power and control circuit applications of the Okonite tape splices at Braidwood. In addition, the leakage currents measured during the tests were reviewed in the previous submittal (see Section 8.0, page 9).

Additionally, the T-95 insulating tape has been designed for use on high voltage, high temperature splicing applications. It has a nominal thickness of 20 mils and a dielectric strength of 600 V/mil (VAC rms or VDC, since they are considered equivalent). In a typical splice application at Braidwood (per Braidwood splicing procedure BwHP 4006-008, Attachments AC through AF), a total thickness of 5/16" is required for the T-95 tape layer. This is well over 300 mils of insulation. The expected dielectric breakdown voltage for an installed splice is thus considered to be significantly greater than the level of applied maximum nominal circuit voltage of 480VAC (calculations using the 600V/mil value result in dielectric strength values greatly in excess of 10,000V for the as-installed splices). Hence, the construction method and configuration for Okonite tape splices ensures significant dielectric strength margin for the normal applied voltage levels used in a typical splice application. Furthermore, a comparison between the test specimens' construction and those field fabricated at Braidwood clearly shows that the voltage stress per mil of insulation was greater during testing than any installed splice will experience. The test specimens typically only had two half-lapped layers of Okonite T-95 tape yielding a minimal thickness of 40 to 80 mils of insulation. This corresponds to a dielectric voltage stress of 633V/(40 to 80 mils) ≈ 8 to 16V/mil versus a dielectric voltage stress of 277V/300 mils < 1V/mil for the installed splices at Braidwood. Therefore, the voltage levels applied during testing, for specimens that successfully passed the test, fully envelop and demonstrate successful performance characteristics of the Okonite tape splices while being exposed to postulated design basis accidents at Braidwood Station.

4. Similarity to Field Splices: IEEE Standard 323-1974, Section 5.1 states that the type testing of actual equipment using simulated service conditions is the preferred method of testing.

The licensee has not provided documentation to show that the splices used in the type tests are the same materials and construction as the splices in service at Braidwood.

The licensee should provide documentation, including procedural guidance, to justify that the splices in the plant are the same as those tested.

During the 1988 Braidwood Station walk-down conducted by the NRC inspection team, the as-installed configuration of the Okonite tape splices was confirmed to be similar to the type test configuration. The issue of similarity was addressed at that time. However, enclosed for further NRC staff review is a copy of the relevant sections of Braidwood Station Procedure BwHP 4006-008 concerning Okonite Tape splice configurations.

In addition as highlighted in the response to Question 3, above, the field installed splices at Braidwood Station have more Okonite T-95 insulating tape layers applied than was used in the various test programs already reviewed and discussed in the previous submittal (reference the attached Braidwood splicing procedure BwHP 4006-008).

Because the field installed splices have more insulating tape layers than the type test specimens, the performance of the installed splices will be superior to the test specimen's performance, consequently, the type test specimens represent the field installed splices. Since the test specimens successfully demonstrated their performance under enveloping simulated accident conditions, the installed Okonite tape splices are considered qualified for their postulated worst-case design basis accidents.

6.2 CONTAINMENT SYSTEMS

The containment systems include the containment heat removal systems, the containment isolation system, and the containment combustible gas control system. The containment and the containment systems function to prevent or control the release of radioactive fission products that might be released into the containment atmosphere following a postulated LOCA, secondary system pipe break, or fuel-handling accident.

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

The containment is a prestressed-concrete shell structure made up of a cylinder with a shallow dome roof and a flat foundation slab. The entire containment structure is lined on the inside with steel plate, which acts as a leak tight membrane. The containment completely encloses the entire pressurized water reactor, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safeguard features (ESF) systems.

6.2.1.1.1 Design Bases

The containment systems are designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary system pipe, the containment peak pressure remains below the design pressure, with adequate margins, as presented in Table 6.2-1. These margins are maintained by the containment system even assuming the worst single active failure affecting the operation of the emergency core cooling system, containment spray system, and the reactor containment fan coolers during the injection phase, and the worst active or passive single failure during the recirculation phase. For primary system breaks, loss of offsite power is assumed. For secondary system breaks, loss of offsite power is not assumed, since this would reduce releases to the containment.

The analyses presented in this section are based on assumptions which are conservative with respect to the design of the containment systems (i.e., minimum heat removal, maximum containment pressure). Subsections 6.2.1.2, 6.2.1.3, and 6.2.1.4 present the mass and energy releases used in the design evaluation.

The results of analyses for the pressurizer spray line in the upper pressurizer cubicles and

the main steamline in the steam line pipe chases are included in Subsection 6.2.1.2.

6.2.1.1.2 Design Features

Containment design features include:

a. A secondary shield wall constructed of 4-foot, 6-inch-thick reinforced concrete, extending from the base mat (elevation 377 feet 0 inch) to the operating floor (elevation 426 feet 0 inch). The shield wall encloses the reactor coolant pumps, reactor vessel and its primary shield wall, steam generators, the pressurizer up to elevation 426 feet 0 inch, and the refueling cavity. The secondary shield wall supports the operating floor, which together with the shield wall prevents the containment liner from being impacted by potential internal missiles and from the effects of pipe whip. Refer to Chapter 3.0 for a more detailed discussion.

b. The Byron/Braidwood design does not use a pressure-suppression-type containment.

c. Since the reactor containment fan coolers are utilized during normal operation with chilled water supplied to a non-safety-related cooling coil in each fan cooler, inadvertent changes to the accident mode will produce no significant effect upon containment internal pressure. The chilled water coils of the reactor containment fan coolers (RCFC) are designed to remove the containment heat in conjunction with essential service water coils during normal plant operation only and hence are non-safety-related. However, the essential service water coils and other components of the RCFC are safety-related as they would be required to operate following a LOCA for heat removal in the containment. For further description of the RCFC System operation and design requirements under post-LOCA conditions, refer to Subsections 6.2.2.2 and 6.2.2.3. For a description of RCFC system operation and design requirements under normal operation, refer to Subsection 9.4.9.1.

d. Water may fill the refueling cavity until it empties into the reactor vessel cavity and floods the cavity up to base mat (elevation of 377 feet 0 inch). Other than this, there are no locations within the containment where water may be trapped and prevented from returning to the containment sump. Water that condenses within the reactor

containment fan cooler housing is drained to the containment base mat.

e. The containment and subcompartment atmospheres are maintained during normal operation within prescribed pressure, temperature, and humidity limits by means of the containment chilled water systems which deliver 40°F water to the dehumidifying coils within each reactor containment fan cooler. Containment penetrations cooling is accomplished by means of supplying component cooling water to the penetrations that have cooling coils. Containment ventilation systems such as the CRDM booster fans and the CRDM cooling fans are used during normal operation and require no periodic testing to ensure functional capability.

6.2.1.1.3 Design Evaluation

The short-term pressure subcompartment analysis considers a loss of offsite power. Consideration of single active failures is of no consequence, since none of the safety equipment functions during the initial seconds of the postaccident transient. The maximum calculated differential pressure in the loop compartment is 20.27 psi resulting from a double-ended hot leg (DEHL) break in volume 3 (see Table 6.2-10 for listing of volumes). The maximum calculated differential pressure in the upper pressurizer cubicle is 10.24 psi resulting from a spray line double-ended break. The maximum calculated differential pressure in the steamline pipe chase is 13.43 psi resulting from a main steamline break in volume 26.

The containment subcompartment differential pressure analysis is described in detail in Subsection 6.2.1.2. The results of the pressure transient analysis of the containment for the loss-of-coolant accidents are shown in Figures 6.2-1 through 6.2-6. Containment temperature curves are presented in Figures 6.2-7 through 6.2-12. The cases examined in this analysis determine the effects of the full range of large reactor coolant break sizes up to and including a double-ended break. Cases illustrating the sensitivity to break location are also shown. All of these cases show that the containment pressure will remain below design pressure with margin. After the peak pressure is attained, the performance of the safeguards system reduces the containment pressure. At the end of the first day following the accident, the containment pressure has been reduced to a low value. The peak pressures and margins are shown in Table 6.2-1.

The smaller pump suction breaks, the hot leg break and the cold leg break mass and energy releases assumed that the sump water (which is pumped back through the core when the RWST empties) is at a constant temperature of saturation at the design pressure of the containment. As required by the NRC, the full

double-ended pump suction breaks (which is the worst case) assumes a depressurization of the containment.

The results of the pressure transient analysis of the containment for the secondary side breaks are presented in Table 6.2-1. The pressure and temperature curves for the most limiting steam break are presented in Figures 6.2-13 and 6.2-14.

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code, COCO (Reference 1). The COCO code has been used and found acceptable to calculate containment pressure transients for the H.B. Robinson (Docket Number 50-261) and Zion (Docket Number 50-295) plants.

The analyses performed to evaluate the containment temperature and pressure response to a postulated main steamline break (MSLB) inside containment utilized the Westinghouse containment model developed for the IEEE-323-1971 Westinghouse Supplemental Equipment Qualification Program. These models and their justification (experimental and analytical) are detailed in References 18 through 22.

The analysis of these containment models has been compared to the analyses for other Westinghouse plants where models were used which conform to those presented in CSB "Interim Evaluation Model for the Main Steam Line Break Inside Containment."

These comparisons indicate:

a. The conservatism of the Westinghouse large steamline break containment model when used with dry steam blowdowns.

b. The Westinghouse small steamline break containment model, using the convective heat flux heat transfer model, results in peak temperatures comparable to those calculated using the NRC Interim Proposed Model with 8% revaporization.

Based on the above information, the reanalysis of the MSLB inside containment is not required.

Details of the analytical model used to conservatively determine the maximum containment temperature and pressure for a spectrum of postulated main steam line breaks for various reactor power levels are provided below:

a. Single failure in the safety grade systems required to mitigate the consequences of a spectrum of main steamline breaks inside containment were evaluated to determine the limiting set of conditions for Byron/Braidwood. One such failure is the failure of a main steam isolation valve to close. This

increases the steam pipe volume available for blowdown through the break. When all valves operate, only the volume between the steam generator and the first isolation valve adds to the blowdown. When the isolation valve fails to close, the volume between the break and the isolation valves in the other steamline becomes available as well as the volume of the safety and relief valve headers and other connecting lines.

Failure of the feedwater isolation valve to close was evaluated. This failure increases the volume of feedwater not isolated from the steam generator which would be available for blowdown. The volume increase is due to water between the feedwater isolation valve and the feedwater regulating valve including all headers and connecting lines.

While the steamline and feed water isolation valves are closing, the flow is considered as unrestricted until the time of complete closure.

Auxiliary feedwater flow that continues after isolation of the steam and feedlines was included in the analysis. Excessive blowdown of auxiliary feedwater through the depressurized steam generator is prohibited by the flow limiting orifices installed in each auxiliary feedwater supply line. Within 30 minutes, auxiliary feedwater flow to the depressurized steam generator is isolated manually by the operator from the control room. Auxiliary feedwater flow to the intact steam generators is assumed to continue.

A blowdown increase due to a failure of the main feedwater pump trip was included in the analysis.

Single failure of the containment cooling systems with reduction of heat removal capability was included. These were by a train failure or by a diesel failure after a loss of offsite power.

All NSSS equipment to be relied on for the transients are safety grade.

b. The design temperature for the liner is 280°F. The design temperature of the internal containment structure is 280°F.

The peak containment temperature for a primary side break is 267°F and for a secondary side break is 318°F. Refer to Table 6.2-1.

The justification for the design temperatures, provided in Table 6.2-66, selected for the liner and internal containment structures is that they are conservative when the duration of the peak temperature for the secondary side break, the temperature lag between the containment atmosphere and the passive heat sinks such as the containment liner and

internal structures, and the resistance to heat transfer provided by the materials used, are considered.

The design temperatures defined for the qualification of Westinghouse PWR-SD supplied safety-related instrumentation inside containment are provided in Supplement 1 (Rev. 1, November, 1978) to WCAP-8587 "Methodology for Qualifying Westinghouse PWR-SD Supplied NSSS Safety Related Electrical Equipment."

The temperature used to qualify safety-related instruments inside the containment exceeds the peak temperature for a secondary side break and is therefore conservative.

c. The piping volumes from the plant layout between the affected steam generator and the various steamline isolation valves and feedwater isolation valves and control valves are the following. The maximum volume between the affected steam generator and: (1) the main steam isolation valve is 749 ft^3, (2) the main steam isolation valves for the intact steam generators is 11,358 ft^3, (3) the main feedwater isolation valve is 175 ft^3, and (4) the main feedwater control valve is 655 ft^3. These volumes are shown diagrammatically in Figures 6.2-38 and 6.2-39.

The main steam and main feedwater isolation valves have a 5-second maximum closure time. The feedwater preheater bypass valves have a 6-second maximum closure time. The main feedwater control valves must close with no specified closure time. Automatic closure of auxiliary feedwater isolation valves is not a requirement.

Blowdown from the broken steamline is assumed to be saturated steam.

Transient phenomena within the reactor coolant system affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into two systems. The first system consists of the air-steam phase; the second is the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and

subcooled or saturated water. Switching between states is handled automatically by the code. The following are the major assumptions made in the analysis:

a. Discharge mass and energy flow rates through the reactor coolant system break are established from the analysis in Subsection 6.2.1.3.

b. For the steam break analysis and the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the break point. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

c. Homogeneous mixing is assumed. The steam-air mixture and the water phase each have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and water phase.

d. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

e. For large steamline breaks the saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

f. For small steamline breaks the model described in Section 2 of reference 2 was utilized.

Subsections 6.2.1.3 and 6.2.6.1.4 present the mass and energy releases used for the analysis.

Initial Conditions

An analysis of containment response to the break of the reactor coolant system must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis.

Also, values for the temperature of the service water and refueling water storage tank solution are assumed, along with the initial water inventory of the refueling water storage tank. All of these values are chosen conservatively, as shown in Table 6.2-2.

In each of the transients, the safeguards systems shown in Table 6.2-3 are assumed to operate at the times indicated. The assumed spray flow rate is based on one of two trains operating.

Heat Removal

The significant heat removal source during the early portion of the transient is structural heat removal. Provision is made in the containment pressure transient analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for transient conduction into and out of the node and temperature rise of the node. Tables 6.2-4 and 6.2-5 are summaries of the containment structural heat sinks used in the analysis. To generate the values in Table 6.2-4, a complete and detailed list of surface areas and thicknesses of structures and equipment in the containment was compiled. An uncertainty of from 0 to -25% was assigned to each calculated area. The containment wall area, which was assumed to have 0% uncertainty, was used as calculated and all other areas were reduced to the minimum value in the uncertainty range specified. Thicknesses were reduced to give conservatively small total volume when several items of varying thickness were combined into one table entry. This procedure resulted in a conservatively small estimate of the available heat sinks.

The heat transfer coefficient to the containment structure is calculated by the code based primarily on the work of Tagami (Reference 3). From this work, it was determined that the value of the heat transfer coefficient increases parabolically to peak value at the end of blowdown for LOCA and increases parabolically to peak at the time of steamline isolation. The value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam to air weight ratio.

Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed," defined as follows:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment volume) (time interval to peak pressure)}}$$

From this the maximum h of steel is calculated:

$$h_{max} = \frac{(E^{0.60})}{t_{pv}} \qquad (6.2\text{-}1)$$

where:

h_{max} = maximum value of h (Btu/hr ft^2°F),

t_p = time from start of accident to end of blowdown for LOCA and steamline isolation for secondary breaks (sec),

V = containment volume (ft^3), and

E = coolant energy discharge (Btu).

The parabolic increase to the peak value is given by:

$$h_s = h_{max} \sqrt{\frac{t}{t_p}}, \text{ for } 0 \leq t \leq t_p \qquad (6.2\text{-}2)$$

where:

h_s = heat transfer coefficient for steel (Btu/hr ft^2°F), and

t_p = time from start of accident (sec).

For concrete, the heat transfer coefficient is taken as 40% of the value calculated for steel.

The exponential decrease of the heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})\, e^{-0.05\,(t-t_p)}, \text{ for } t < tp$$

where:

h_{stag} = $2 + 50X$, for $0 \leq X \leq 1.4$,

h_{stag} = h for stagnant conditions (Btu/hr ft^2°F), and

X = steam-to-air weight ratio in containment.

The steel heat transfer coefficients calculated for the double-ended pump suction, double-ended cold leg, and double-ended hot leg cases are shown in Figure 6.2-15. The heat transfer coefficient for the most limiting steamline break is presented in Figure 6.2-16.

For a large break the safety features are quickly brought into operation. Because of the brief period of time required to depressurize the reactor coolant system, the safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after the blowdown and maintain a low long-term pressure. Also, although the containment

structure is not as effective a heat sink as during the reactor coolant system blowdown, it still contributes significantly as a form of heat removal during the long-term cooling period.

During the injection phase of postaccident operation, the emergency core cooling system pumps water from the refueling water storage tank into the reactor vessel. Since this water enters the vessel at refueling water storage tank temperature, which is less than the temperature of the water in the vessel, it can absorb heat from the core until saturation temperature is reached. During the recirculation phase of operation, water is taken from the containment sump and cooled in the residual heat removal heat exchanger. The cooled water is then pumped back to the reactor vessel to absorb more decay heat. The heat is removed from the residual heat removal heat exchanger by component cooling water.

Another containment heat removal system is the containment spray. During the injection phase of operation, the containment spray pumps draw water from the refueling water storage tank and sprays it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the refueling water storage tank, the entire heat capacity of the spray from the refueling water storage tank temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase of postaccident operation, water is drawn from the sump and sprayed into the containment atmosphere.

When a spray drop enters the hot, saturated, steam-air containment environment following a loss-of-coolant accident, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the atmosphere and the drop will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam and the condensation will cease. The temperature of the drop will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling drop are as follows:

$$\frac{d}{dt}(Mu) = mh_g + q \qquad (6.2\text{-}3)$$

$$\frac{d}{dt}(M) = m \qquad (6.2\text{-}4)$$

where:

$$q = h_c A (T_s - T), \text{ and}$$

$$m = k_g A (P_s - P_v).$$

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu'.

Both Nu and Nu' may be calculated from the equations of Ranz and Marshall (Reference 4).

$$Nu = 2 + 0.6 (Re)^{1/2} (Pr)^{1/3} \qquad (6.2\text{-}5)$$

$$Nu' = 2 + 0.6 (Re)^{1/2} (Sc)^{1/3}$$

Thus, Equations 6.2-3 and 6.2-4 can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean drop produced by the spray nozzles rises to a value within 99% of the bulk containment temperature in less than 2 seconds.

Drops of this size will reach temperature equilibrium with the steam-air containment atmosphere after falling through less than half the available spray fall height.

Detailed calculations of the heatup of spray drops in post-accident containment atmospheres by Parsly (Reference 5) show that drops of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment.

These results confirm the assumption that the containment spray will be 100% effective in removing heat from the atmosphere.

Nomenclature

A = area,

h_c = coefficient of heat transfer,

k_g = coefficient of mass transfer,

h_g = steam enthalpy,

M = droplet mass,

m = diffusion rate,

Nu = Nusselt number for heat transfer,

Nu' = Nusselt number for mass transfer,

P_s = steam partial pressure,

P_v = droplet vapor pressure,

Pr = Prandtl number,

q = heat flow rate,

Re = Reynolds number,

Sc = Schmidt number,

T_s = droplet temperature,

T = steam temperature,

t = time, and

u = internal energy.

The reactor containment fan coolers are a final means of heat removal. The main aspect of a fan cooler from the heat removal standpoint are the fan and the banks of cooling coils. The fans draw the dense atmosphere through banks of finned cooling coils and mix the cooled steam/air mixture with the rest of the containment atmosphere. The coils are kept at a low temperature by a constant flow of cooling water. Since this system does not use water from the refueling water storage tank, the mode of operation remains the same both before and after the spray system and emergency core cooling system change to the recirculation mode. Fan cooler heat removal performance is shown in Figures 6.2-17 and 6.2-25a.

Inadvertent Spray Actuation

In the event of inadvertent spray, the containment would depressurize until the temperature of the air was approximately the temperature of the spray. A calculation was performed to calculate the maximum outside to inside pressure differential. The following initial conditions were assumed:

a. The containment is initially at 120°F, which maximizes the temperature differential between the containment atmosphere and the spray, which is at a temperature of 35°F.

b. The containment is at 14.7 psia.

c. The relative humidity is at a maximum value of 100%.

As the air temperature is reduced, the partial pressure of the air decreases from 13.009 to 11.103 psi. The steam partial pressure decreases from 1.691 to 0.117 psi as the spray cools the atmosphere. Thus a containment pressure of 11.22 psia is

produced, causing a differential pressure of 3.48 psi across the containment shell.

Accident Chronology

For the double-ended pump suction, double-ended hot leg, and double ended cold leg loss-of-coolant accidents, the major events and their times of occurrence are shown in Tables 6.2-6 through 6.2-8. Table 6.2-9 presents the accident chronology for the limiting steamline break. In the event of a LOCA or main steamline break in conjunction with loss of offsite power, the diesel generators will be at full power within 10 seconds. Table 8.3-5 shows that the RCFCs are loaded on the emergency buses approximately 20 seconds following initiation of the safety injection signal. The table also shows that containment spray pumps are available at 25 seconds after loss of offsite power or if not required at 25 seconds, the pumps will be available at 50 seconds. Delay times in Table 8.3-5 may be exceeded by 2 seconds. This will account for switching and signal transmission times.

The RCFCs will operate at full power 5 seconds after loading onto emergency power. The containment spray (CS) pumps operate at full power 2.1 seconds after startup. The CS containment isolation valves require 10 seconds to open. The valves are immediately loaded onto the ESF buses. The containment spray system is kept full at least to the 407 foot elevation (isolation valve) by the RWST. It will require less than 29 seconds to fill the spray system and achieve full flow.

In the LOCA or MSLB case, the RCFCs will require the following startup times after loss of offsite power:

RCFC Time Delay	20 Seconds
RCFC Speedup	5 Seconds
Allowance for Signals	2 Seconds
TOTAL	27 Seconds

This is well within the 40 seconds allowed in the analyses.

For the LOCA case, the availability of power will be the limiting factor in the containment spray startup. The spray actuation signal (hi-3 containment pressure) will be received before the spray pumps are loaded onto the emergency power. The containment spray valves (CS019A/B and CS007A/B) will be open when the spray pump is started. The containment spray will be available after the following times:

Pump Loading	25 Seconds
Pump Startup	2 Seconds
Spray System Filling	29 Seconds
Allowance for Signals	2 Seconds
TOTAL	58 Seconds

This time is longer than the 45 seconds used in the analysis. However, it should be noted that gravity flow will begin filling the spray system when the valve is open because the RWST level is 50 feet above the isolation valve. The pump will be partially effective during startup. The spray will initiate from the lower rings prior to full system flow. For these reasons and the margin shown by the analyses, the current assessment is considered satisfactory.

The results of the MSLB analysis shown in Figures 6.2-13 and 6.2-14 is an updated analysis. Table 6.2-9 has been updated to reflect the correct accident sequence and timing.

As discussed previously the RCFC startup time of 58 seconds is very conservative. As shown in Figure 6.2-13, the containment pressure will reach 20-22 psig and initiate the spray at between 50 and 80 seconds after the break. The containment spray will be available after the initiation (high pressure) signal is received plus the following times:

Valve Opening Time	30 Seconds
Pump Startup Time	2 Seconds
Spray System Fill	29 Seconds
Allowance for Signals	2 Seconds
TOTAL	63 Seconds

The total time for spray initiation is less than 121 seconds. This is conservatively represented by the 137.4 second time as shown in the Table 6.2-9.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Basis

Based on the main coolant loop piping leak-before-break analysis performed by Westinghouse (Westinghouse Reports WCAP-10553 and 10554) and the scope outlined in General Design Criterion 4, the containment subcompartments need not be designed for pressurization loads due to postulated primary coolant loop piping breaks. The subcompartments are not necessary to containment function, and therefore, dynamic or non-static pressurization need not be considered.

The reactor cavity and RPV nozzle inspection cavity, therefore, need not consider the pressurization loads due to the primary coolant loop cold leg nozzle breaks. In particular, the inspection cavity shield doors no longer are required to serve as a mechanism to vent the cavity into the main containment.

The loop compartment design bases loads are also no longer the primary coolant loop breaks, although the evaluation for these loads are retained in the UFSAR. These breaks are controlling when compared to other high energy line breaks which occur in these compartments.

a. Subcooled forced convection and

b. Nucleate boiling (using the Jens-Lottes' correlation).

For a more detailed discussion on heat transfer models used in MARVEL, refer to WCAP-7635 and WCAP-8843.

A feedwater pipe break inside containment is not analyzed because it is not as severe as the main steamline break, since the break effluent is at a lower specific enthalpy.

6.2.1.4.1 Pipe Break Blowdowns - Spectra and Assumptions

A series of steamline breaks were analyzed to determine the most severe break condition for containment temperature and pressure response. The following assumptions were used in the analysis:

a. Breaks were assumed to be either double-ended breaks occurring at the nozzle at one steam generator or split breaks.

b. Blowdown from the broken steamline is assumed to be saturated steam.

c. Steamline and feedwater line isolation are completed at 7 seconds after the break occurs. The steamline isolation signal is generated by either a low steamline pressure, high-2 containment pressure, or high steamline rate of pressure decrease. The feedwater isolation signal is generated by either a safety injection, P-4 (reactor trip), or S/G high-2 level. Maximum closure time for both the steamline and feedwater line isolation valves is 5 seconds, thus allowing 2 seconds for signal generation, processing, and delay.

d. Plant power levels of 102% of nominal full load power, 70% of nominal full load power, 30% of nominal full load power, and zero power were considered.

e. The double-ended breaks were evaluated for a full double-ended guillotine, 0.7 ft^2, 0.6 ft^2, 0.5 ft^2, 0.4 ft^2, and 0.2 ft^2. The split breaks were evaluated at 0.94 ft^2, 0.91 ft^2, 0.86 ft^2, and 0.4 ft^2. Steamline flow restrictions in the steam generators limit the effective break area of a full double-ended pipe break to a maximum of 1.4 ft^2 per steam generator.

f. Failures of a main steam isolation valve, a diesel generator, and a feedwater isolation valve were considered.

g. The auxiliary feedwater system is manually realigned by the operator within 30 minutes by isolating auxiliary feedwater flow from the depressurized steam generator.

The consideration of liquid entrainment was used in the MSLB blowdown model for Byron/Braidwood. Entrainment was based on results of the TRANFLO computer code (Reference 23), which was developed specifically to model Westinghouse steam generators. The TRANFLO code has been found to yield conservative entrainment results with respect to actual experimental data.

Experiments have been carried out on test vessels at Battelle Northwest Laboratories (Reference 24), and Battelle Institute, Frankfurt/Main (Reference 25). Also, several blowdowns from a series conducted by CISE as part of the CIRENE-3 program (References 26 and 27), were used to substantiate results of the TRANFLO code. Comparison is made between the actual test case results and the results of the TRANFLO code in WCAP-8821 (TRANFLO Steam Generator Code Description) currently under NRC review.

Also included in the WCAP is a discussion of the effect of steam separators on liquid entrainment. In particular, the WCAP discusses both the swirl vane and peerless chevron separators, both included in the Westinghouse Model D series steam generator design. Although experimental data on the performance of these devices is limited, studies were conducted to assure that the modeling used is conservative for defining moisture entrainment in blowdowns. Based on model sensitivity studies and backed up with field data available for Model 51 steam generators currently in operation, the computer code was found to provide conservative entrainment results with respect to steam separator modeling.

In summation, the Byron/Braidwood MSLB blowdown analysis considered entrainment based on TRANFLO results consistent with those in WCAP-8822, which were generated for Westinghouse Model D steam generators. The TRANFLO computer code has been substantiated by experimental tests mentioned previously and discussed in more detail in WCAP-8821. Also included in WCAP-8821, is a discussion of code modeling including the effect of steam separators on liquid entrainment. The conclusion to be drawn is that the TRANFLO code overpredicts entrainment and leads to conservative results in the MSLB analysis.

6.2.1.4.2 Description of Blowdown Modeling

The following is a description of the break flow modeling of the blowdown of the steam generators and plant steam piping:

a. Steam Generator Blowdown

Break flows and enthalpies from the steam generators are calculated using the Westinghouse MARVEL Code (Reference 9). Blowdown mass and energy release were determined using the MARVEL Code, including effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick metal heat storage, and reverse steam generator heat transfer.

b. Steam Plant Piping Blowdown

The contribution to the mass and energy releases from the secondary plant steam piping is included in the mass and energy release calculations. For all breaks, the steam piping volume blowdown begins at the time of the break and continues at a uniform rate until the entire piping inventory is released. The flow rate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. Following the piping blowdown, reverse flow from the intact steam generators continues for 7 seconds to simulate the reverse steam generator flow prior to steamline isolation.

The blowdown model is further discussed in References 10 and 11.

6.2.1.4.3 Single Failure Analysis

The following single failures were evaluated to determine the limiting set of conditions for this analysis:

a. Failure of a main steam isolation valve increases the volume of steam piping which is not isolated from the break. When all valves operate, the piping volume capable of blowing down is located between the steam generator and the first isolation valve. If this valve fails, the volume between the break and the isolation valves in the other steamlines including safety and relief valve headers and other connecting lines will feed the break.

b. Failure of a diesel generator would result in the loss of one containment safeguards train resulting in minimum heat removal capability.

c. Loss of the feedwater condensate tank and switchover to the service water system for auxiliary feedwater pump suction would result in a more conservative auxiliary feedwater flow rate.

d. Failure of the main feedwater pump trips would result in additional inventory supplied to the steam generator before feedwater isolation.

e. Failure of a feedwater isolation valve could only result in additional inventory in the feedwater line which would not be isolated from the steam generator. The mass in this volume can flush into the steam generator and exit through the break. Both the feedwater isolation valve and the feedwater regulating valve close in no more than 5 seconds, precluding the pumping of any additional feedwater into the steam generator. The additional line volume available to flush into the steam generator is that between the feedwater isolation valve and the feedwater regulating valve, including all headers and connecting lines.

The resultant mass and energy release rates for the limiting steam pipe break are presented in Table 6.2-50. The containment peak pressures and temperatures for the secondary side breaks are listed in Table 6.2-1.

The worst case failure assumed in each of the secondary side break cases of Table 6.2-1 is a failure of one MSIV to close and the loss of one containment spray train.

In addition, the mass and energy releases given in Table 6.2-50 are based on the worst case failure. This case is a failure of one MSIV to close and the loss of one containment spray train. The feedwater and main steam isolation valve closure times associated with the mass and energy release data in Table 6.2-50 are given in Table 6.2-9.

In the case of a loss of one heat removal train (two RCFC and one spray train) the peak containment pressure and temperature is 33 psig and 315°F respectively.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System (PWR)

The containment backpressure used for the limiting case $C_D = 0.6$, DECLG break for the ECCS analysis presented in Section 15.6.5 is presented in Figure 6.2-24. The containment backpressure is calculated using the methods and assumptions described in Appendix A of Reference 12. Input parameters, including the containment initial conditions; net free containment volume; passive sink materials, thicknesses, and surface areas; and starting time and number of containment cooling systems used in the analysis, are described in Subsections 6.2.1.5.1 through 6.2.1.5.8.

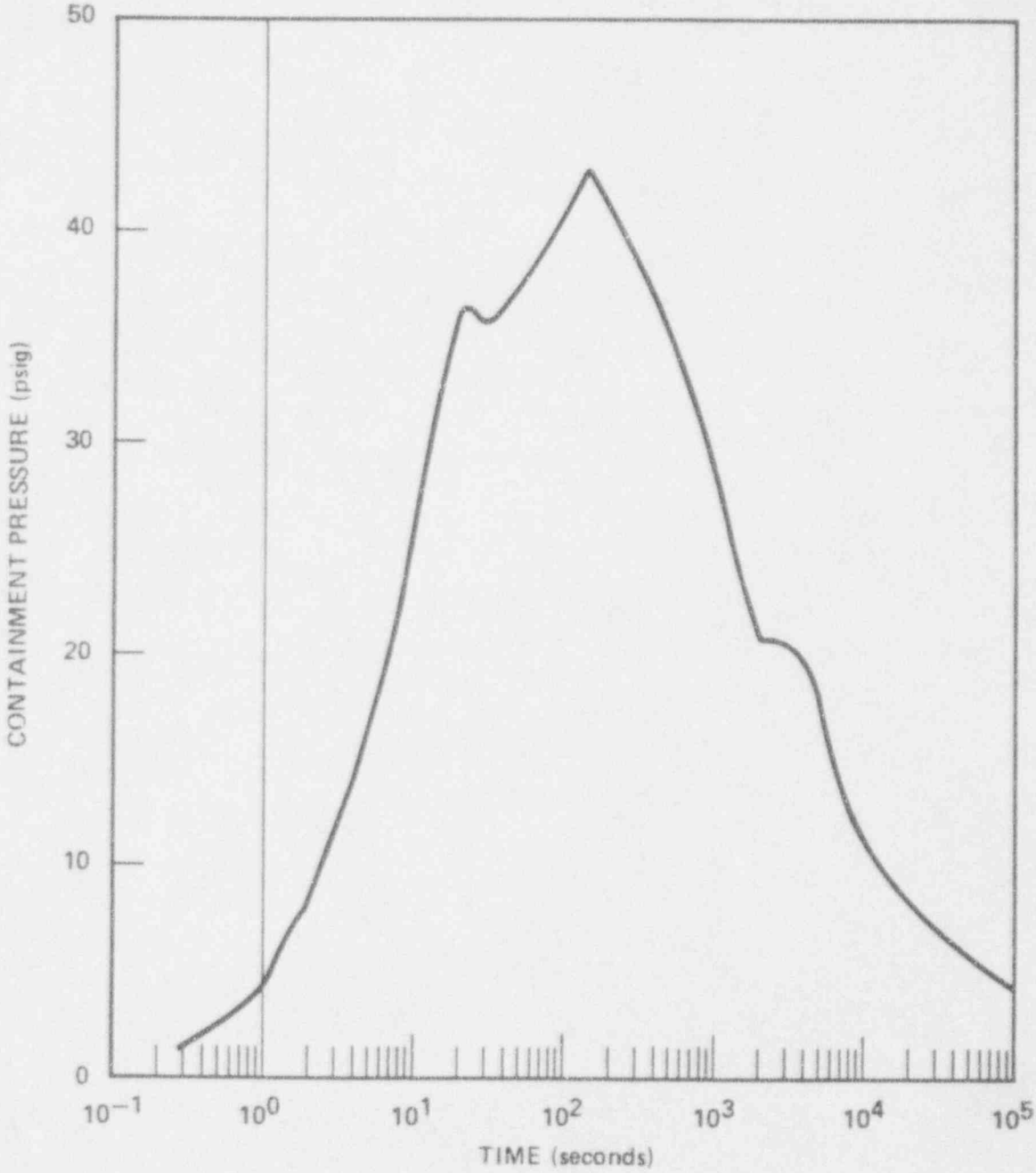

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FIGURE 6.2-1

CONTAINMENT PRESSURE RESPONSE FOR DOUBLE ENDED PUMP SUCTION MAX SI

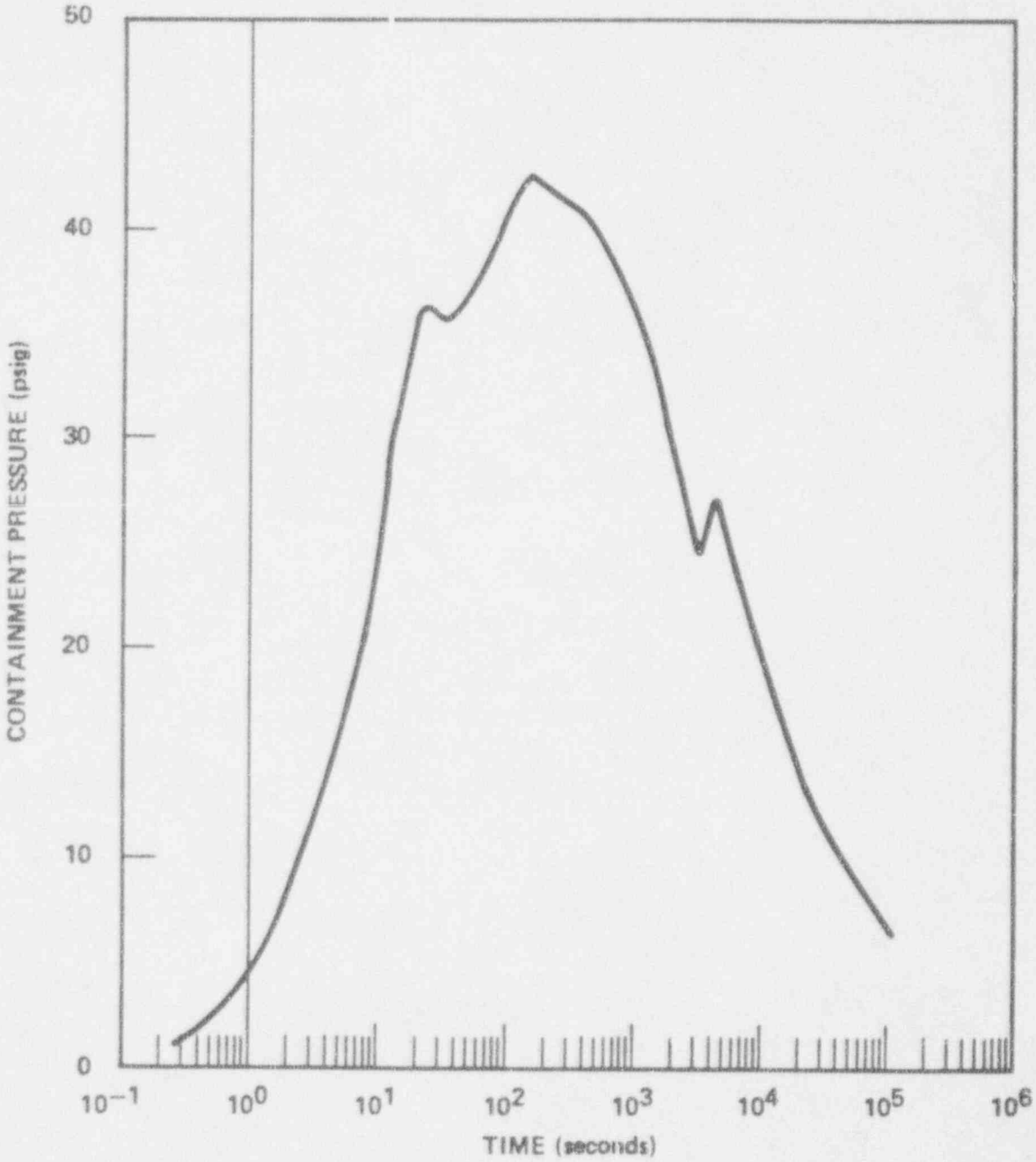

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FIGURE 6.2-2

CONTAINMENT PRESSURE RESPONSE FOR DOUBLE ENDED PUMP SUCTION MIN SI

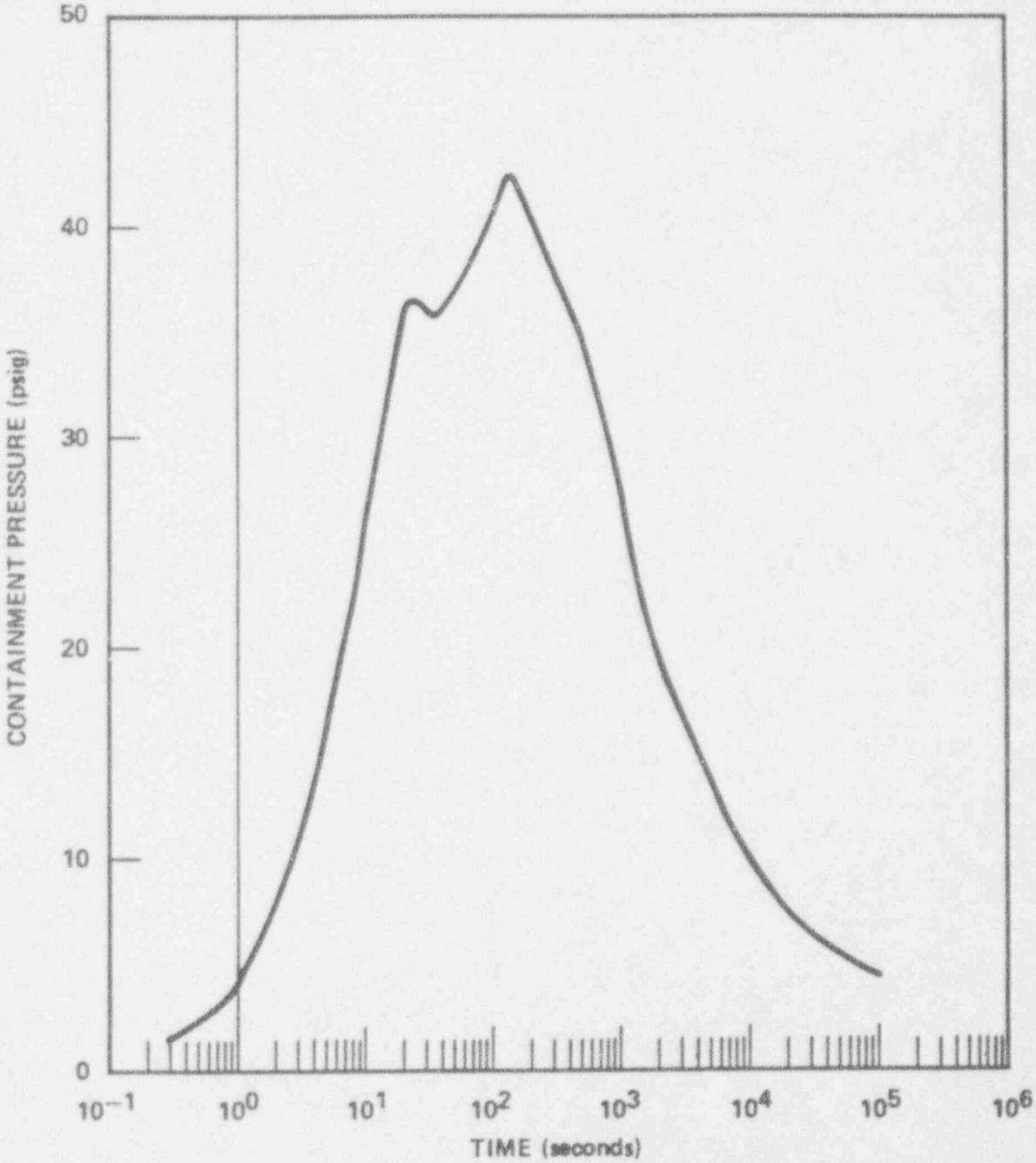

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FIGURE 6.2-3

CONTAINMENT PRESSURE RESPONSE FOR
0.6 DOUBLE ENDED PUMP SUCTION

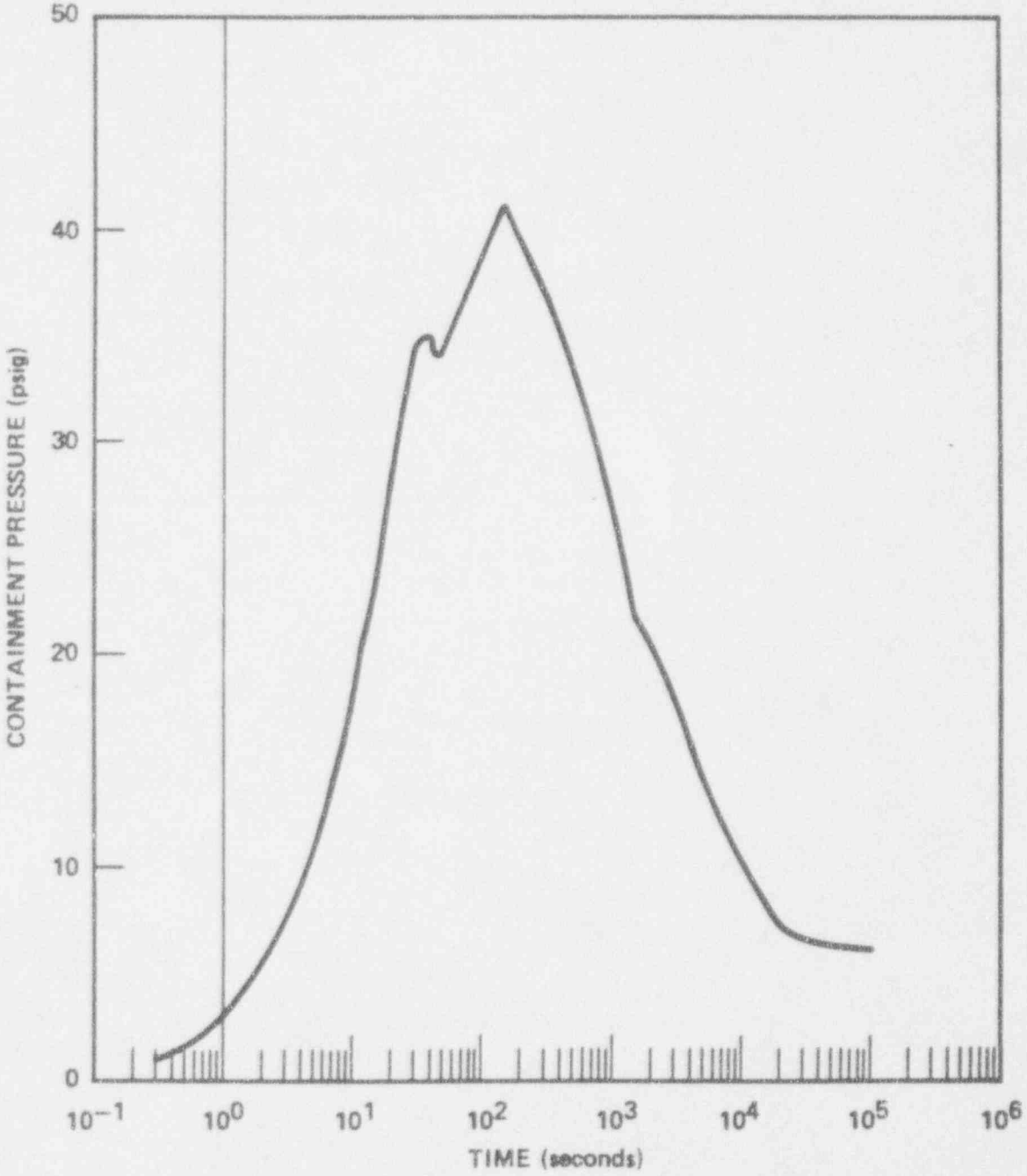

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FIGURE 6.2-4

CONTAINMENT PRESSURE RESPONSE
FOR 3 FT^2 PUMP SUCTION SPLIT

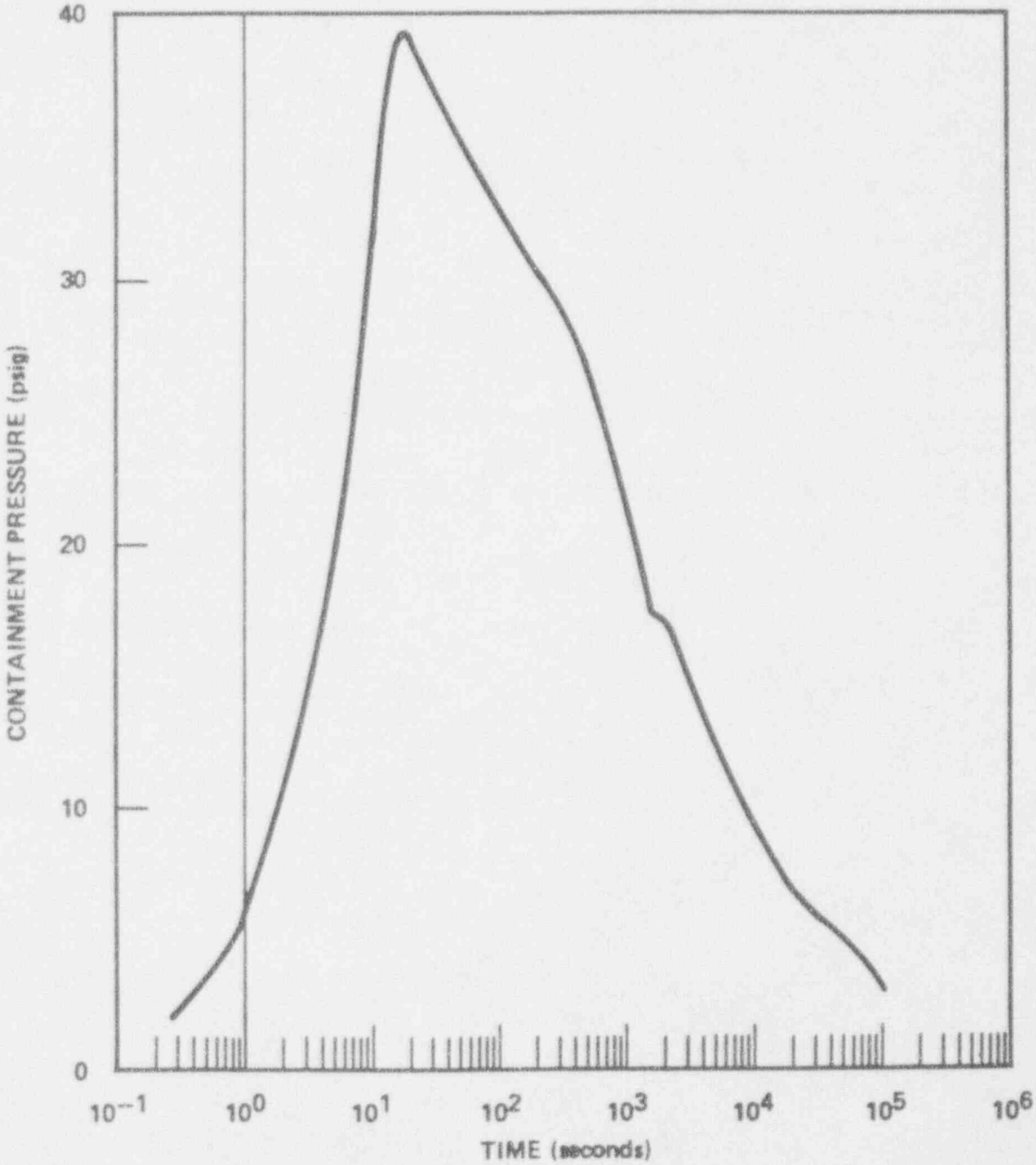

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FIGURE 6.2-5

CONTAINMENT PRESSURE RESPONSE FOR DOUBLE ENDED HOT LEG

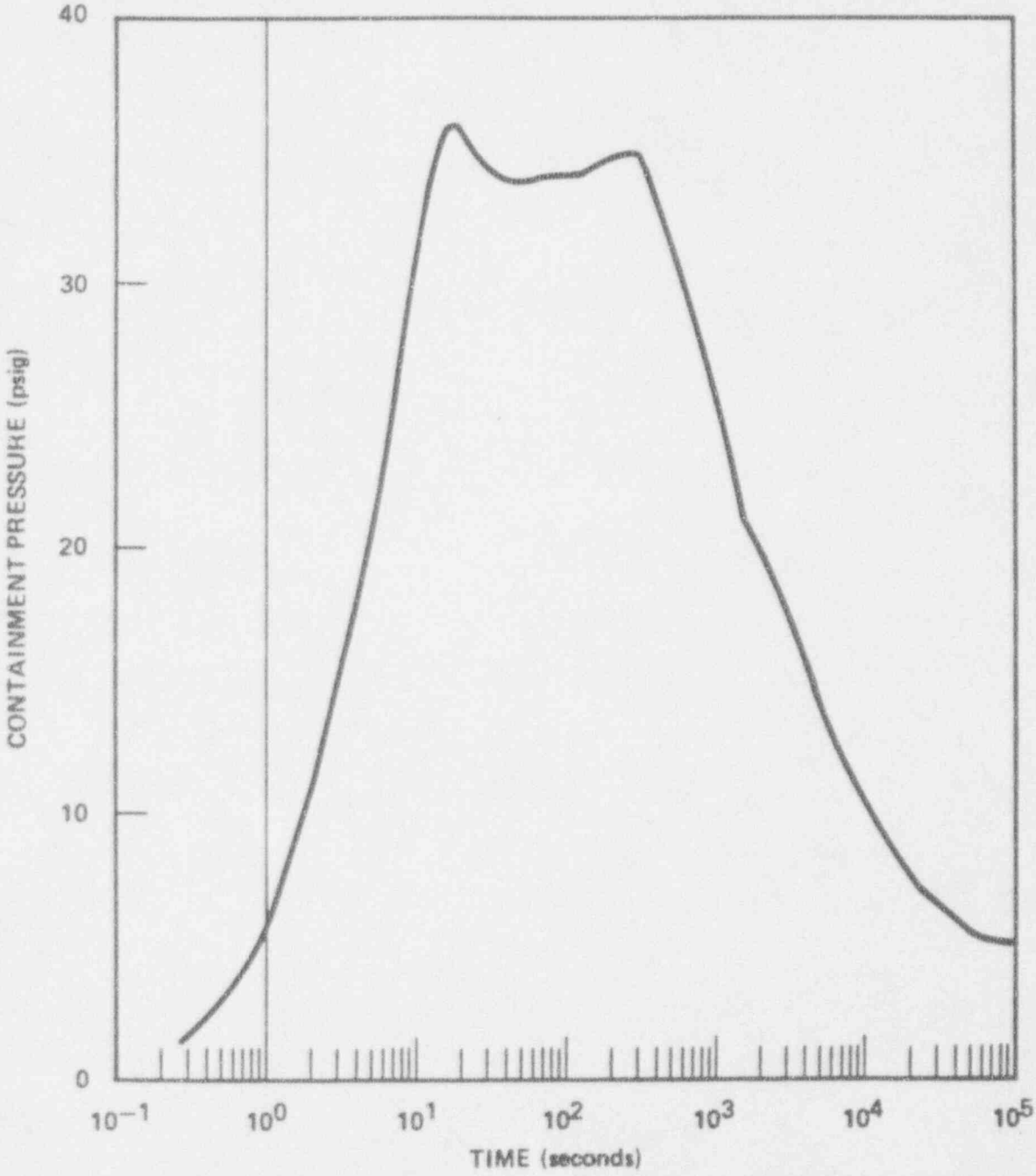

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FIGURE 6.2-6

CONTAINMENT PRESSURE RESPONSE FOR DOUBLE ENDED COLD LEG

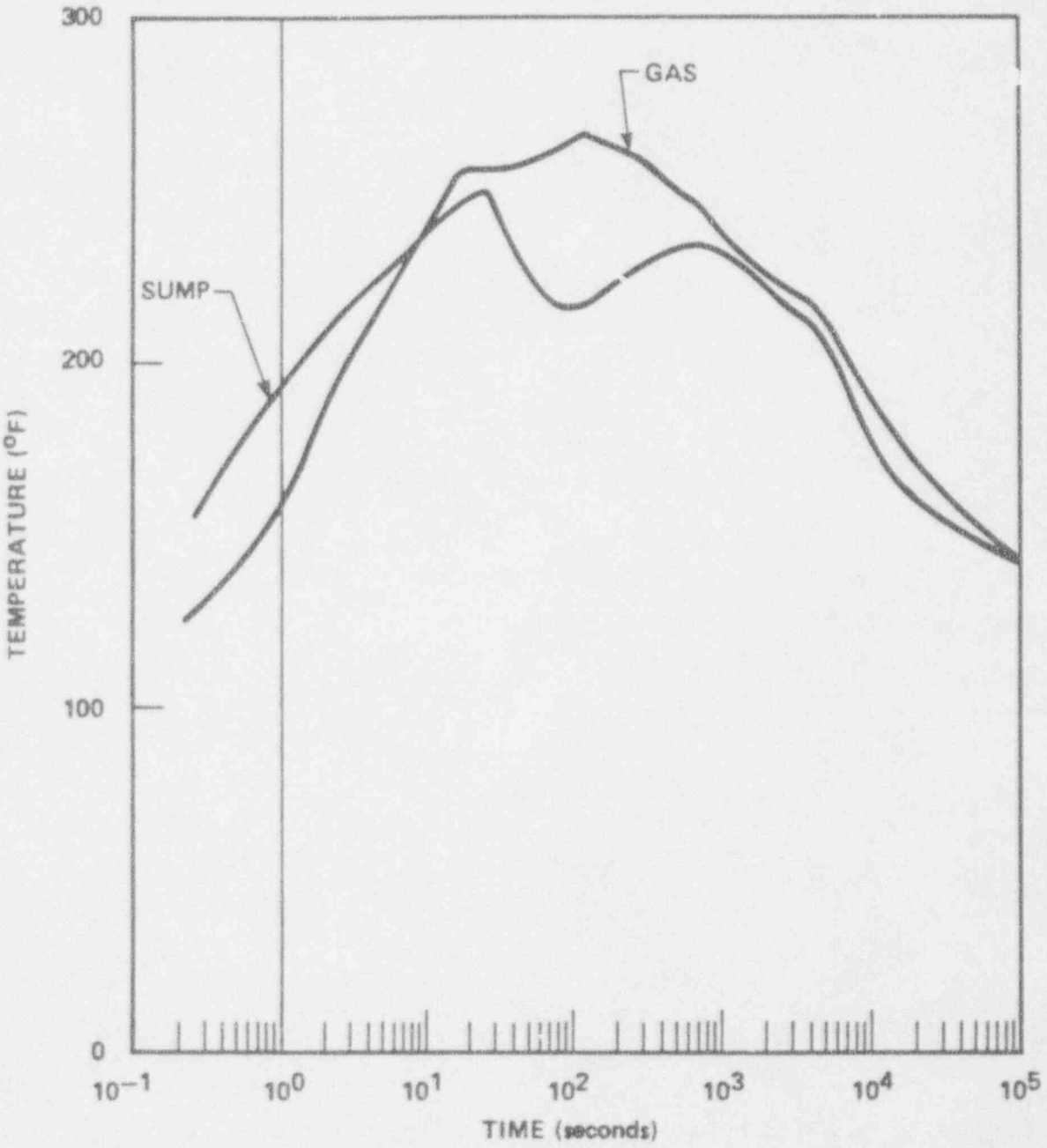

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FIGURE 6.2-7

CONTAINMENT TEMPERATURE RESPONSE FOR DOUBLE ENDED PUMP SUCTION MAX SI

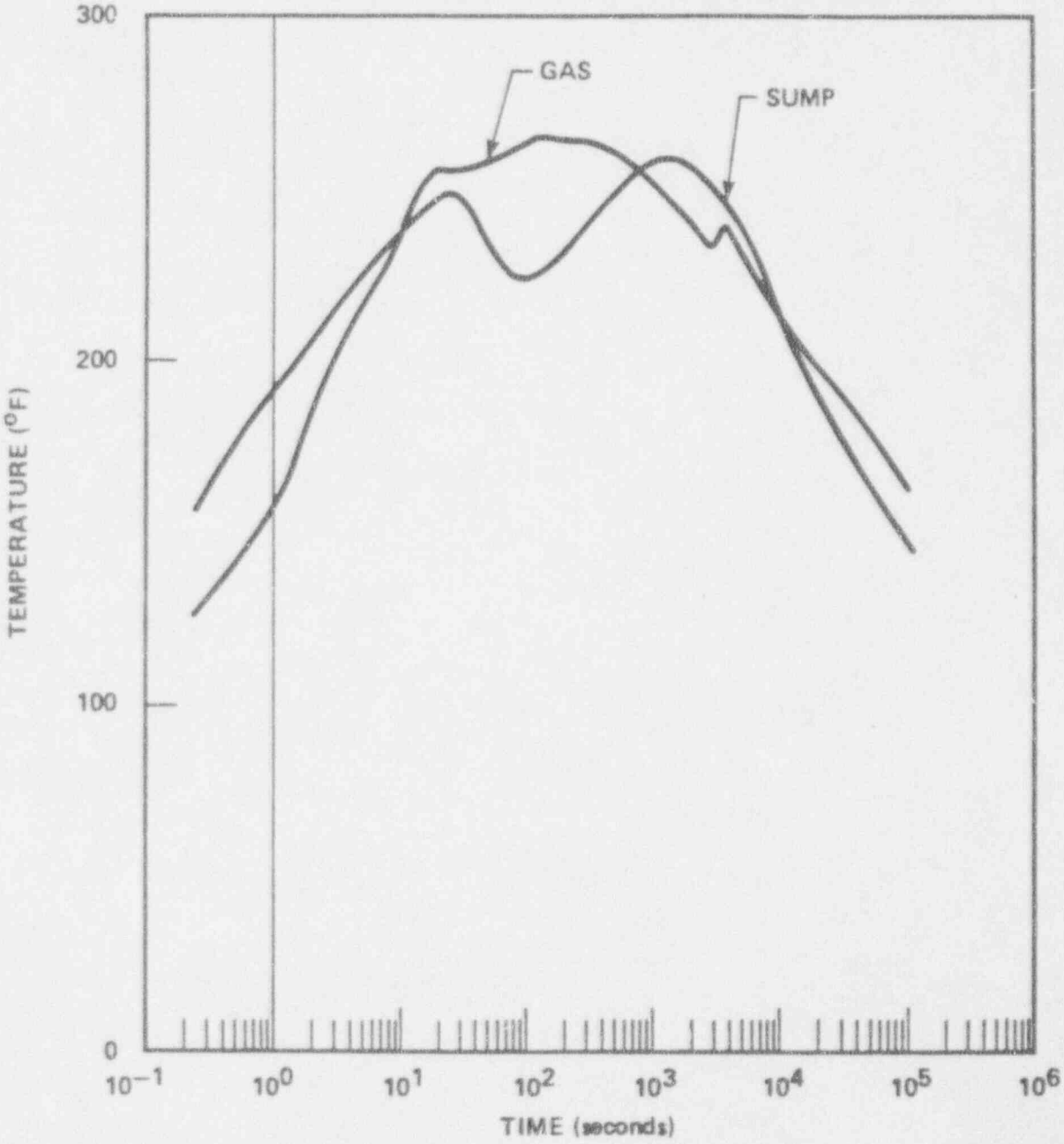

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FIGURE 6.2-8

CONTAINMENT TEMPERATURE RESPONSE FOR DOUBLE ENDED PUMP SUCTION MIN SI

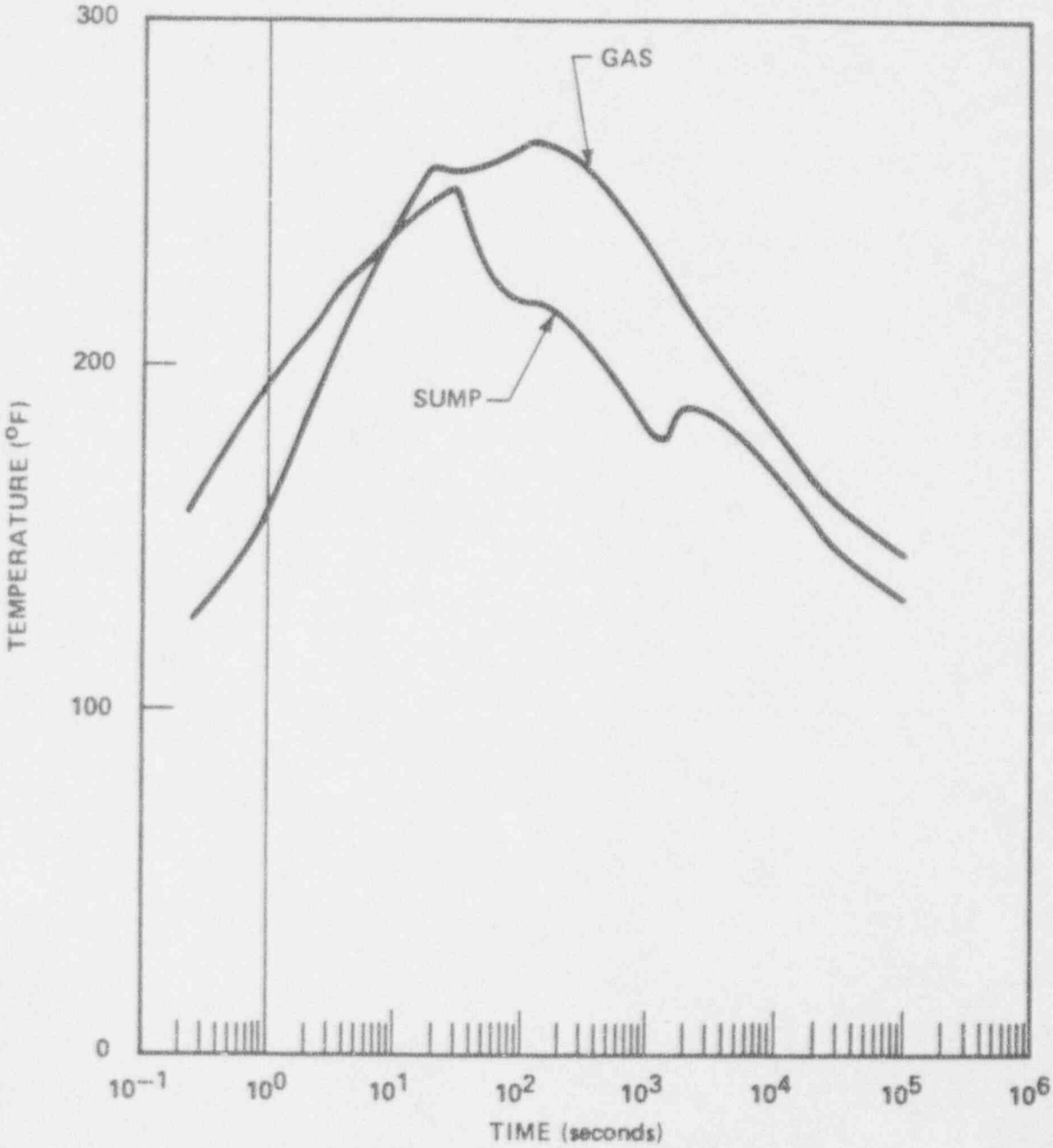

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FIGURE 6.2-9

CONTAINMENT TEMPERATURE RESPONSE FOR 0.6 DOUBLE ENDED PUMP SUCTION

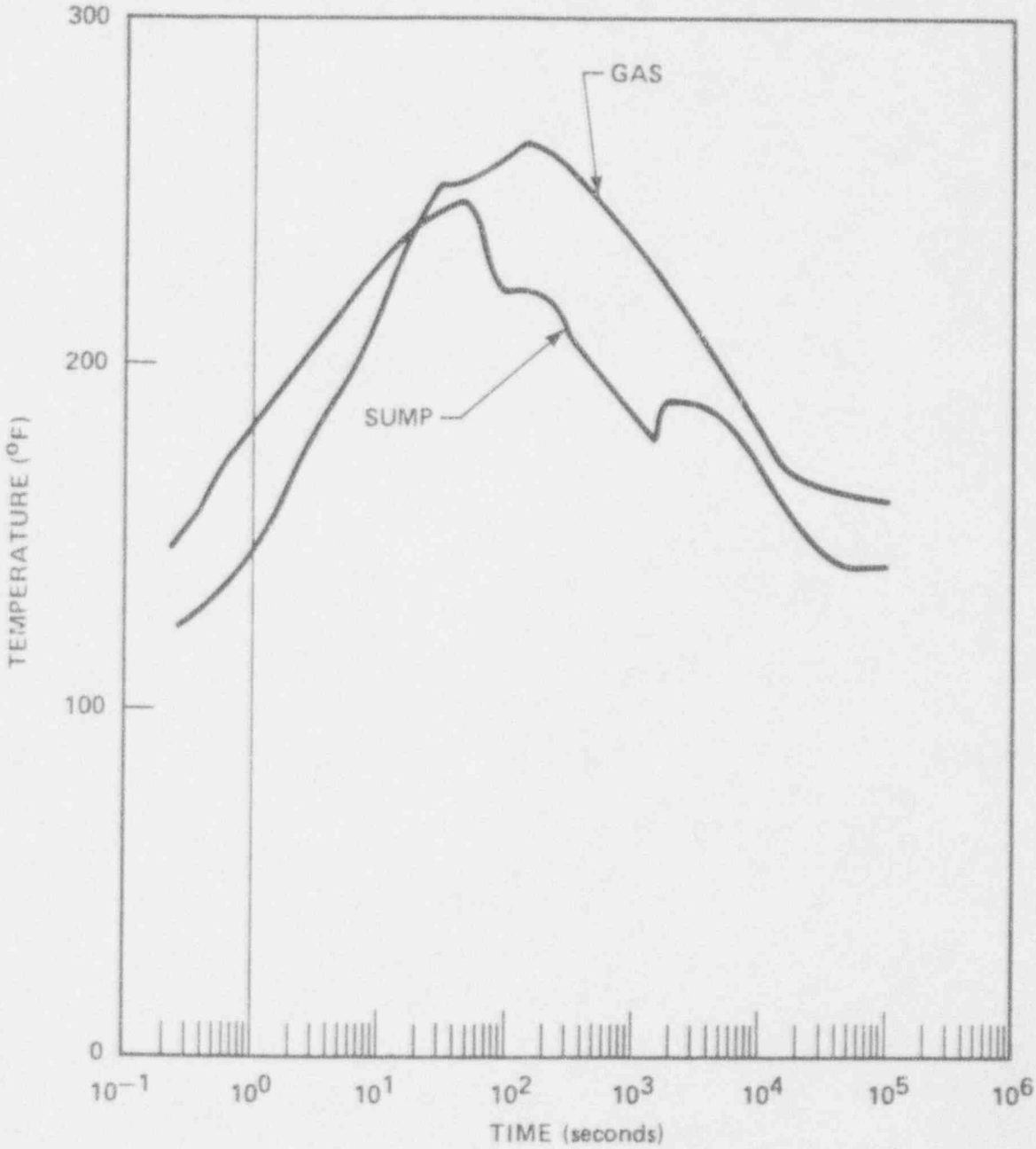

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FIGURE 6.2-10

CONTAINMENT TEMPERATURE RESPONSE
FOR 3 FT^2 PUMP SUCTION

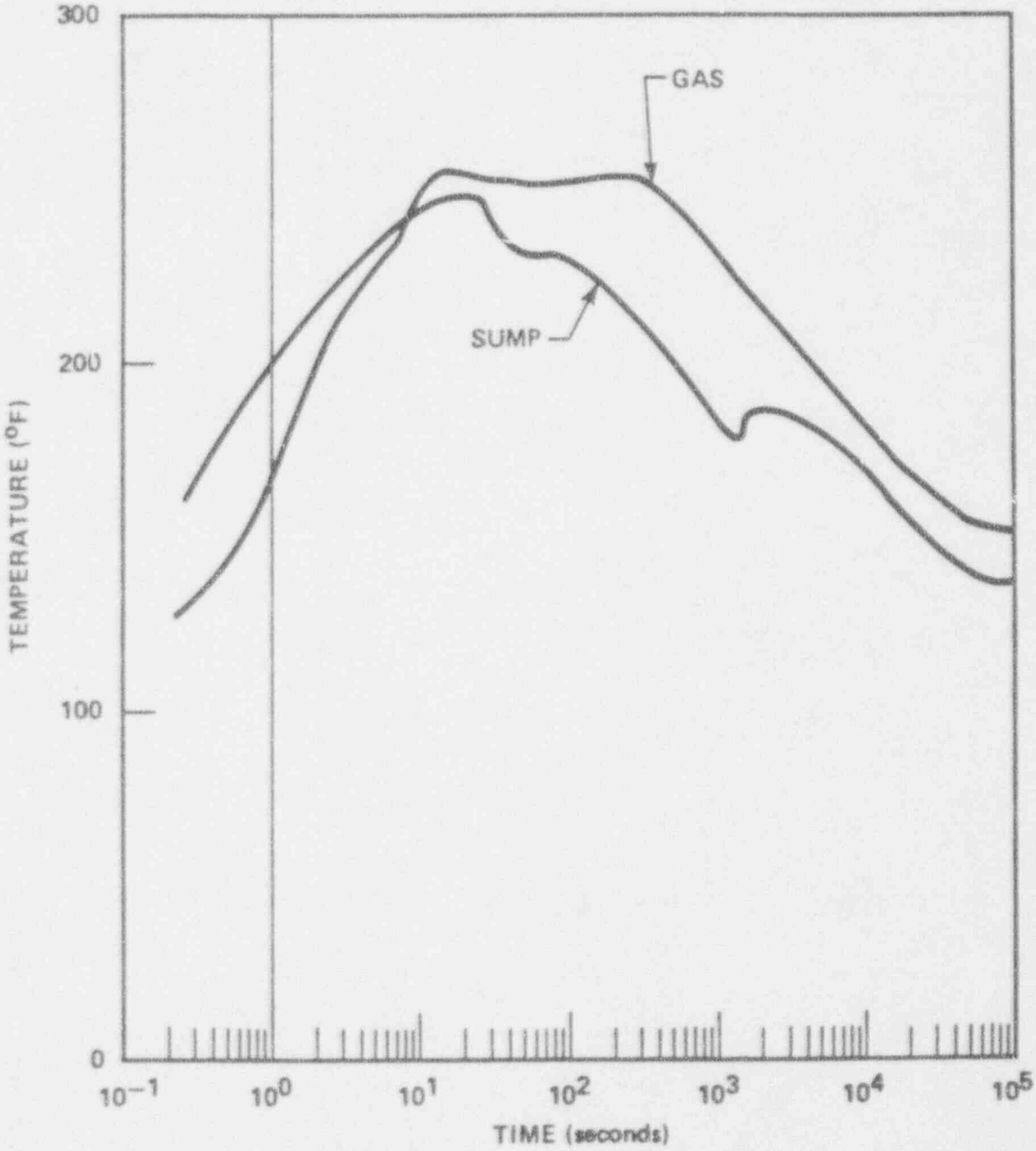

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FIGURE 6.2-11

CONTAINMENT TEMPERATURE RESPONSE FOR DOUBLE ENDED HOT LEG

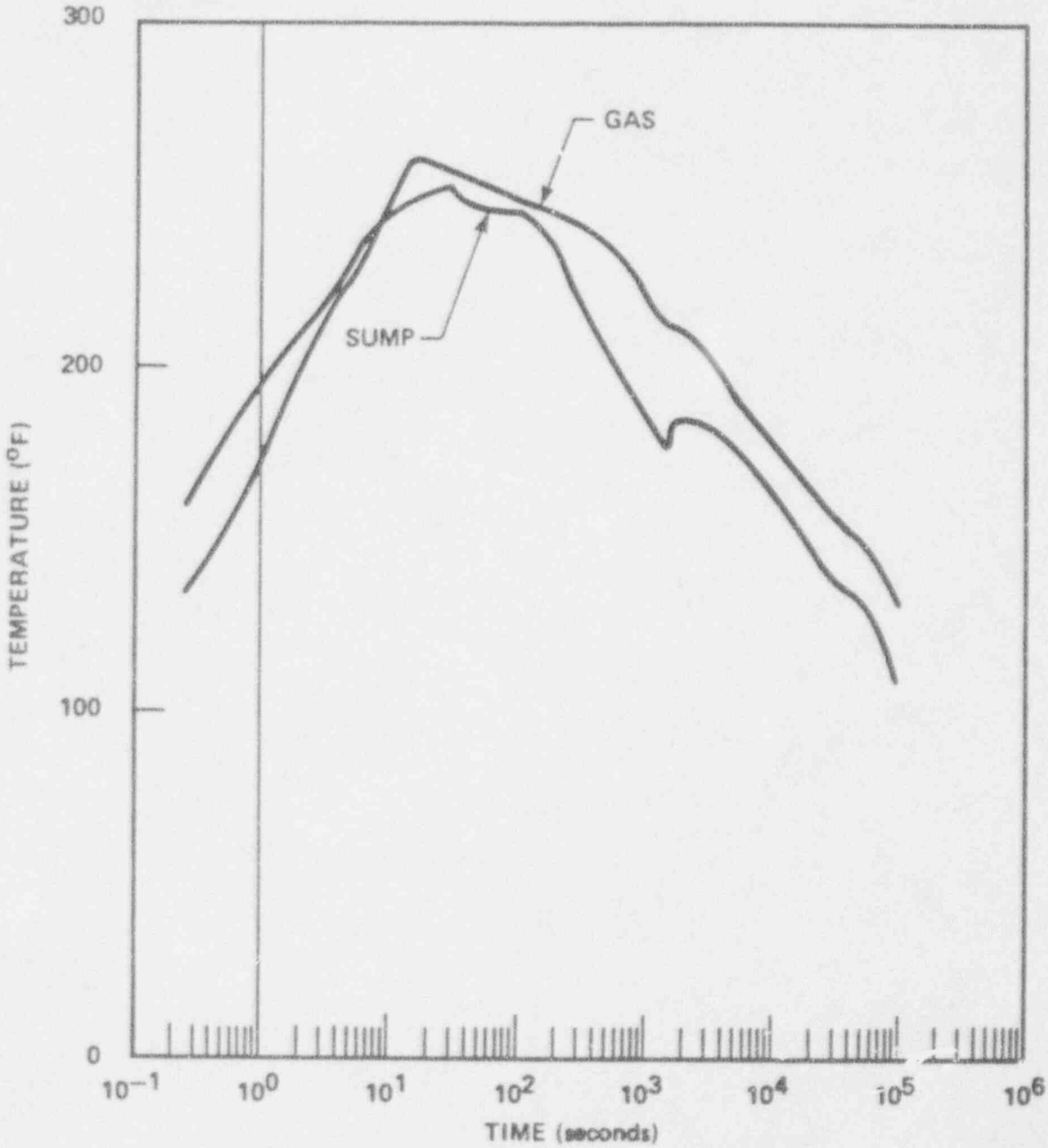

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FIGURE 6.2-12

CONTAINMENT TEMPERATURE RESPONSE FOR DOUBLE ENDED COLD LEG

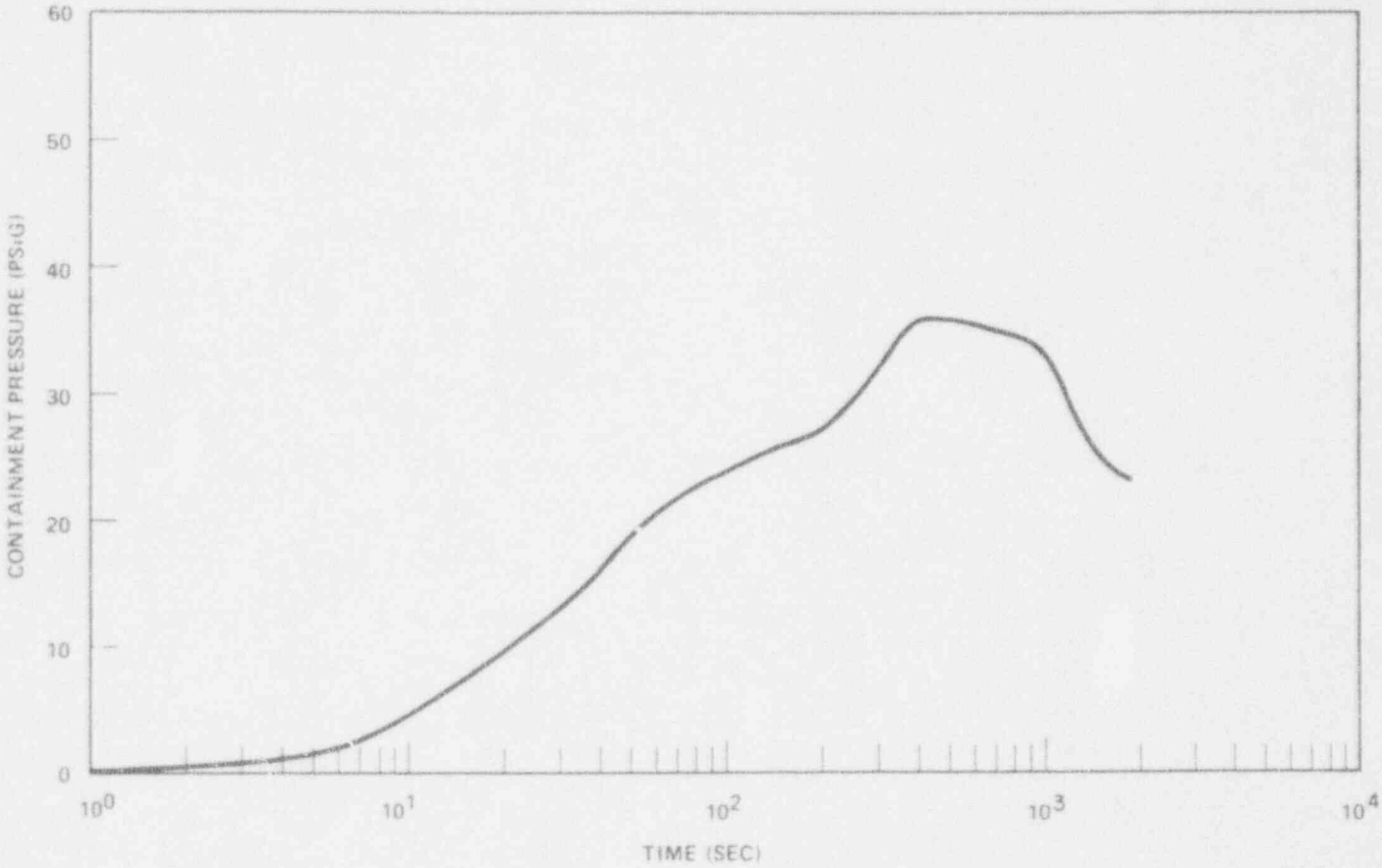

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FIGURE 6.2-13

CONTAINMENT PRESSURE RESPONSE FOR
0.942 FT^2 SPLIT BREAK AT 30% POWER
WITH STEAMLINE STOP VALVE FAILURE

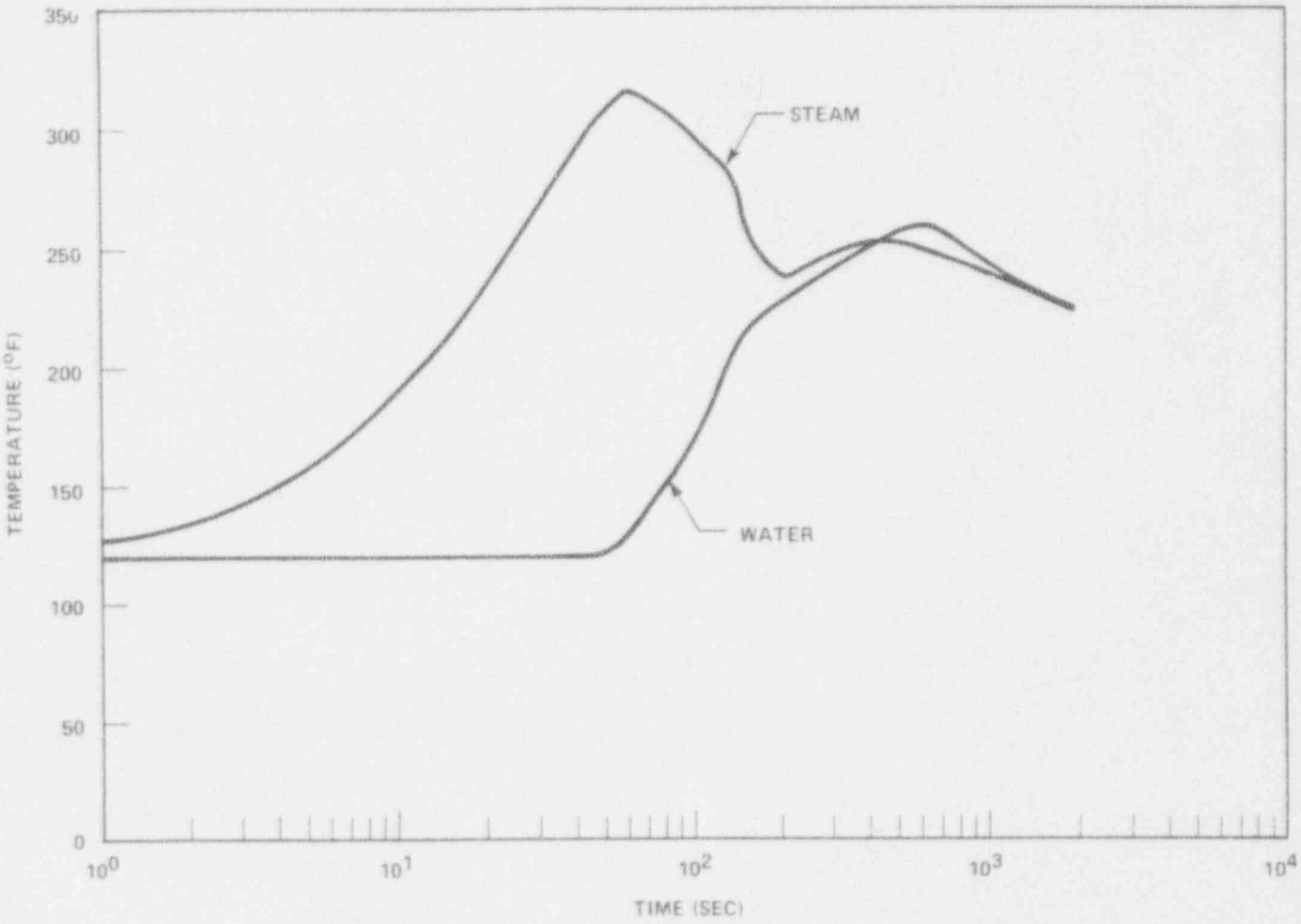

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FIGURE 6.2-14

CONTAINMENT TEMPERATURE RESPONSE FOR 0.942 FT^2 SPLIT BREAK AT 30% POWER WITH STEAMLINE STOP VALVE FAILURE

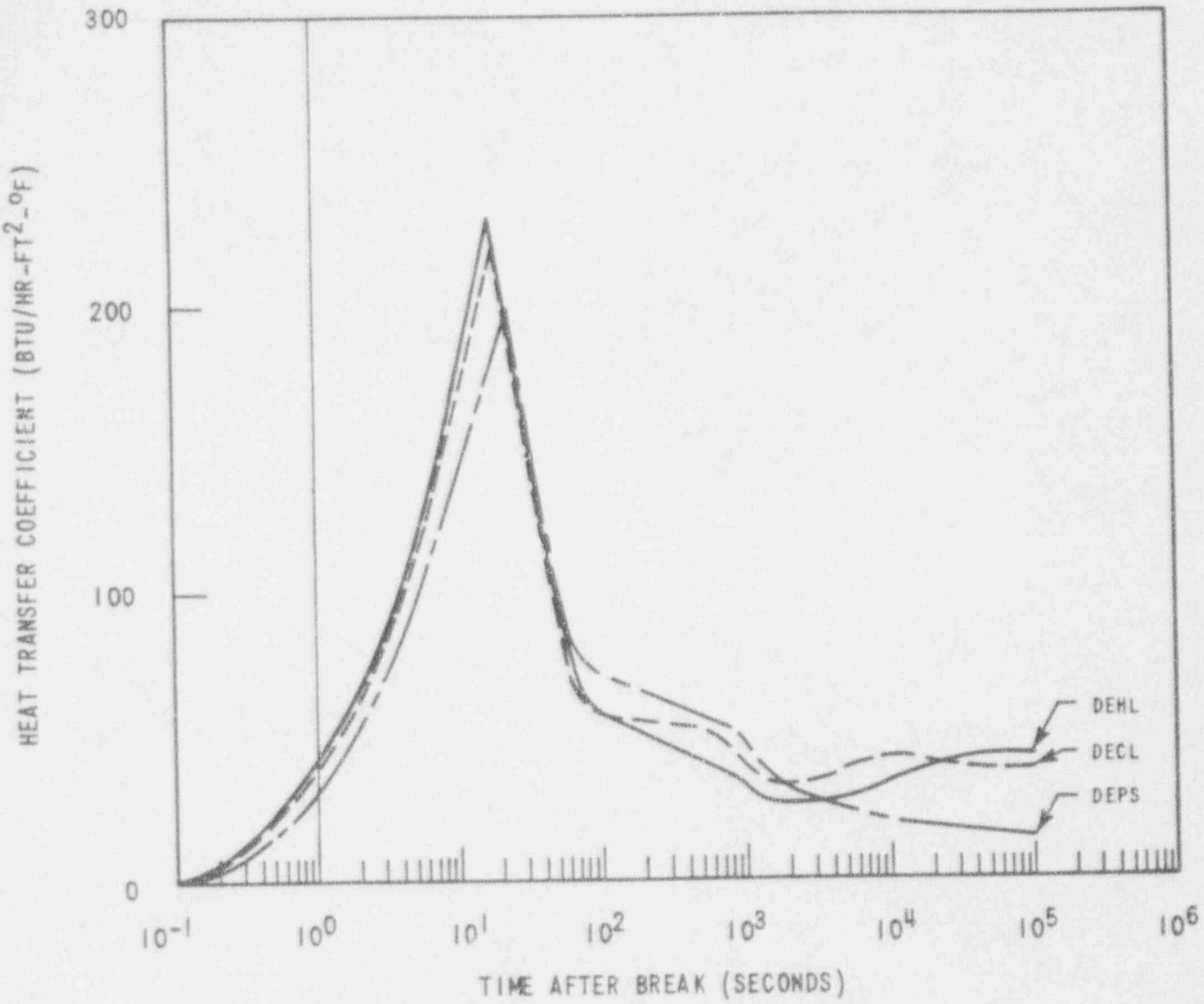

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FIGURE 6.2-15

HEAT TRANSFER COEFFICIENT, DEPS, DECL, AND DEHL

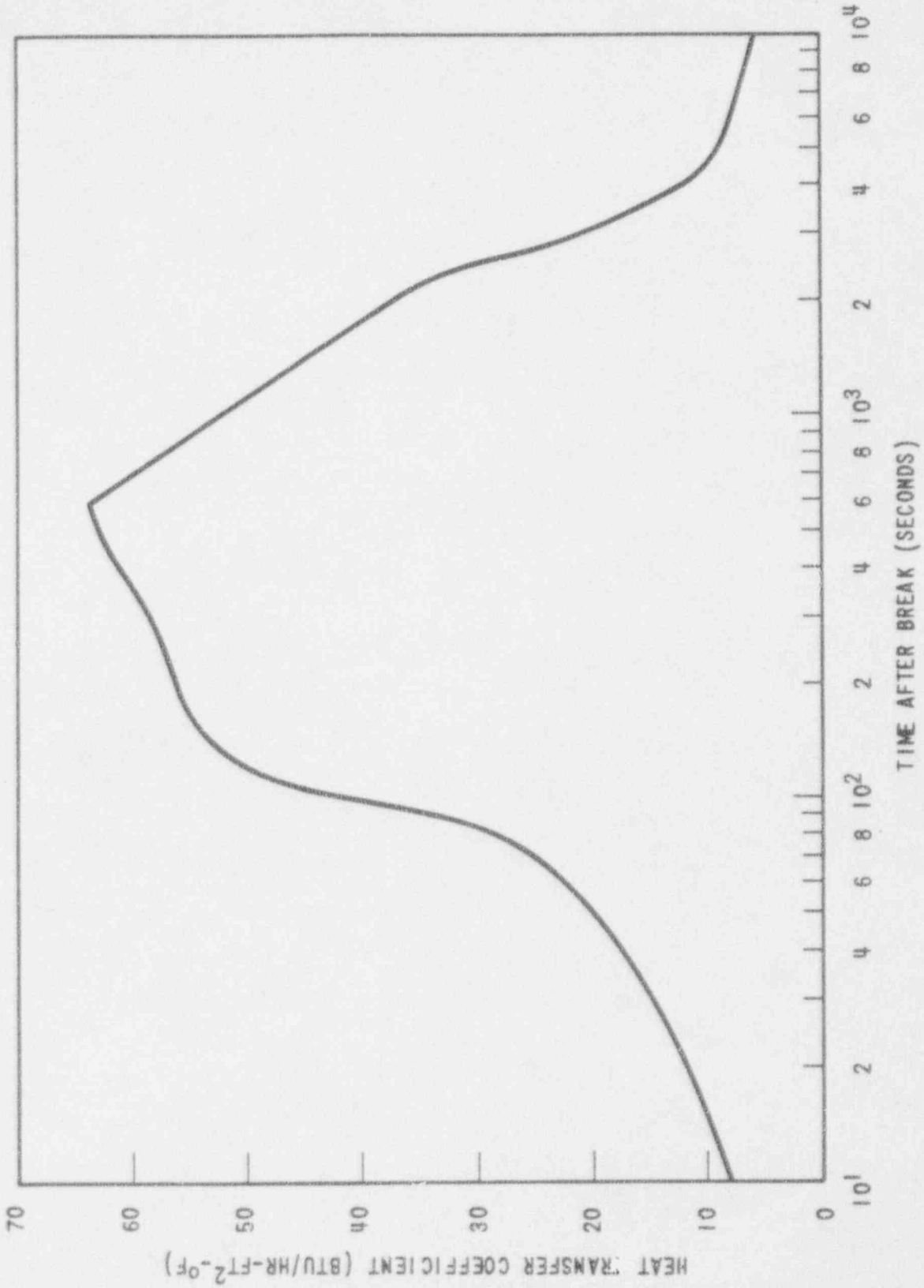

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FIGURE 6.2-16

HEAT TRANSFER COEFFICIENT,
STEAMLINE BREAK

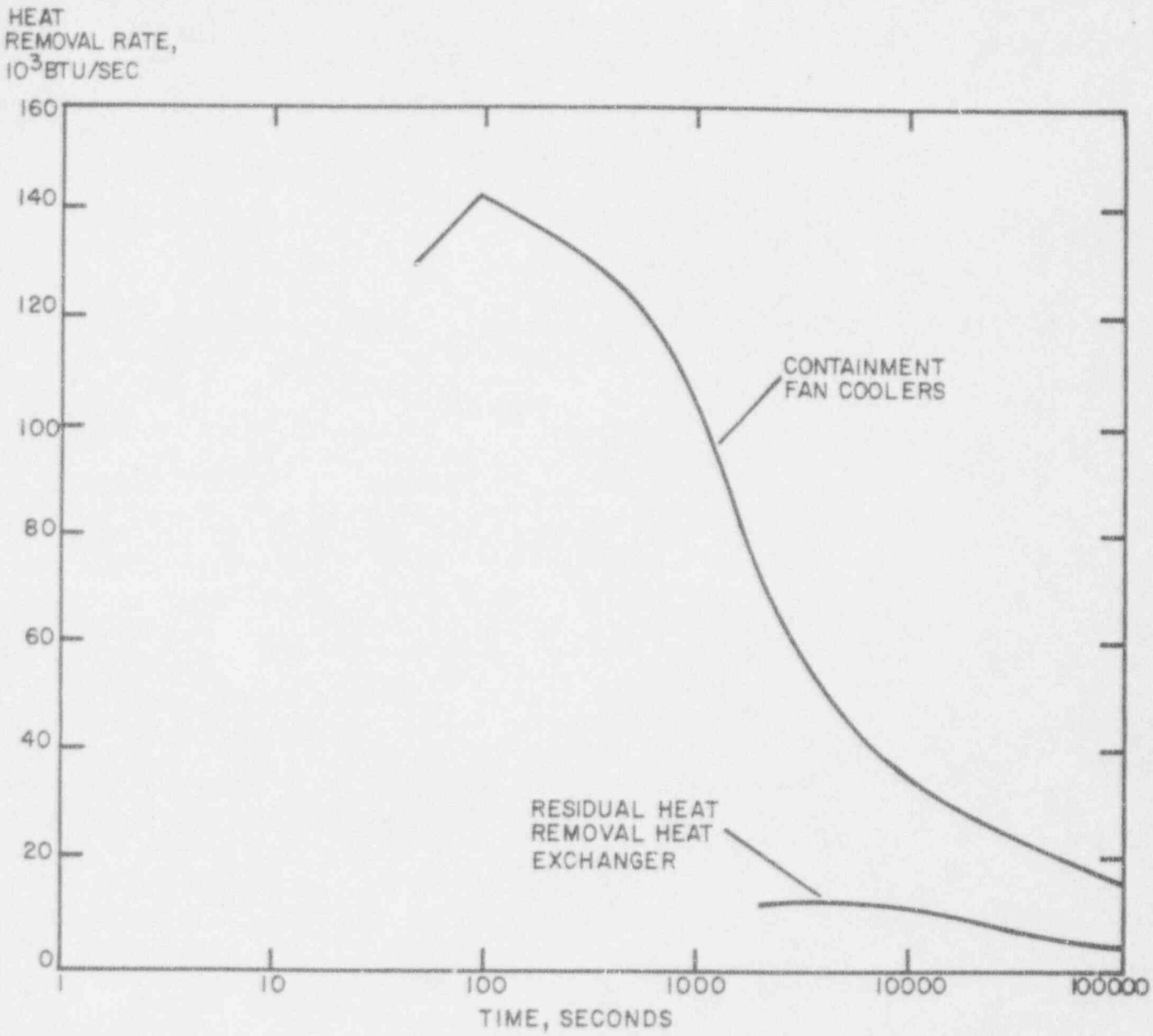

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FIGURE 6.2-17

HEAT REMOVAL FROM CONTAINMENT

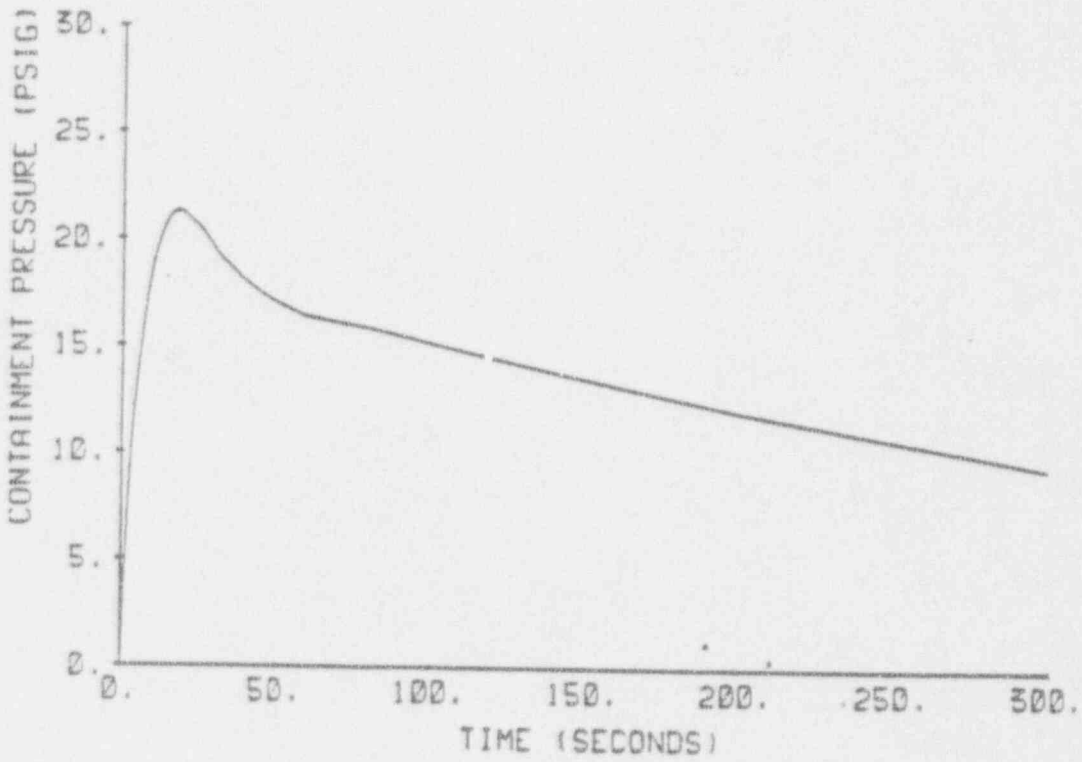

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FIGURE 6.2-24

CONTAINMENT PRESSURE
DECLG (C_D = 0.6) MINIMUM SAFEGUARDS
NOMINAL THOT (619.3 DEG-F)

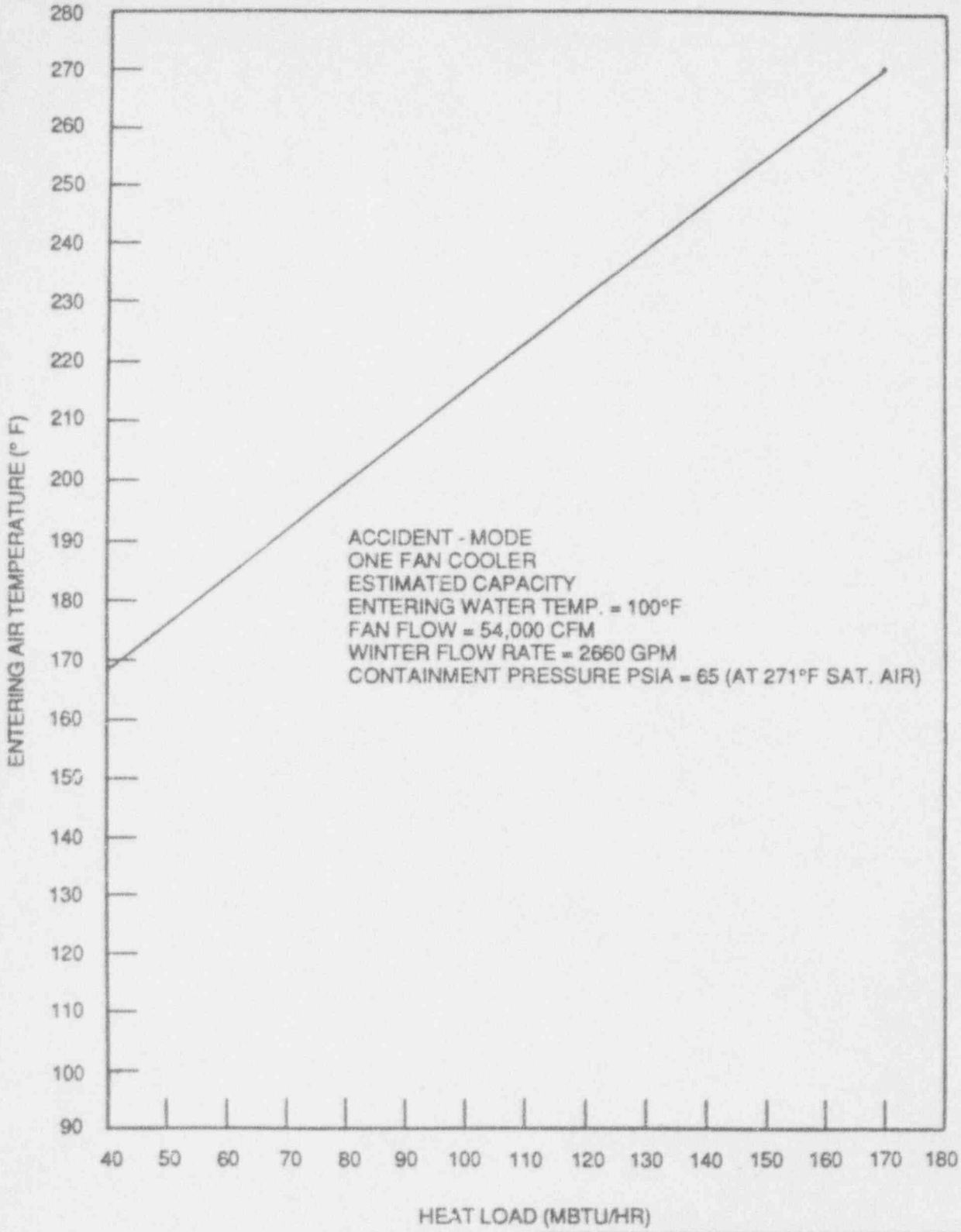

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FIGURE 6.2-25a

ONE FAN COOLER ESTIMATED HEAT REMOVAL CAPACITY (USED IN CONTAINMENT)

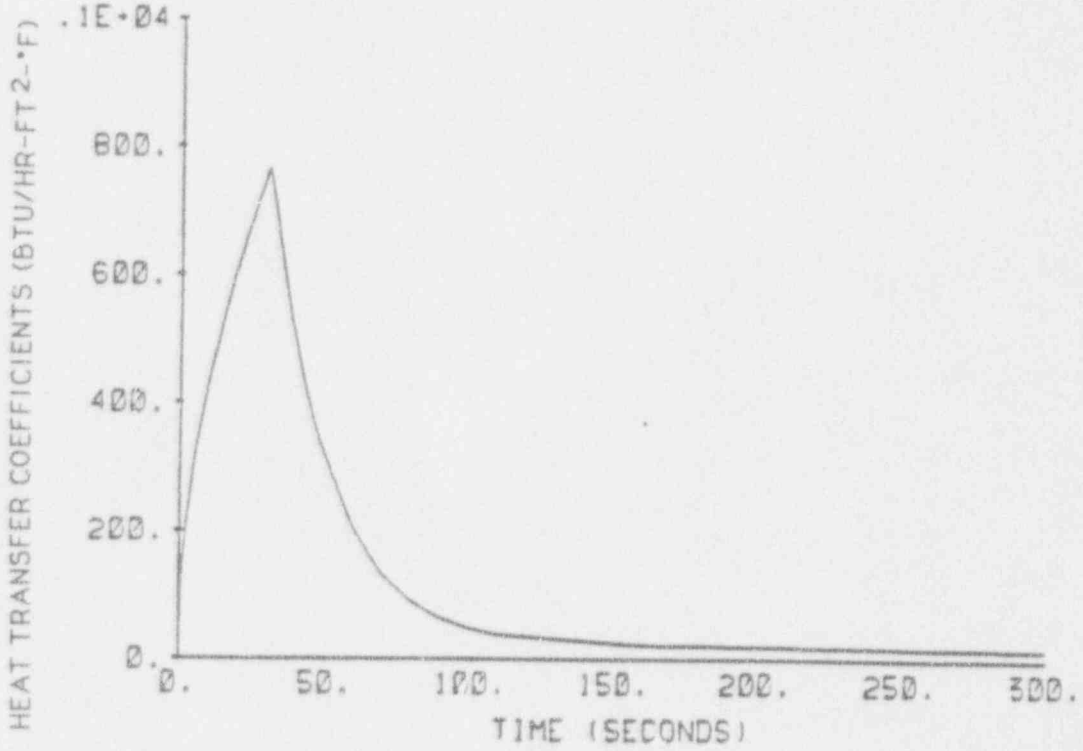

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FIGURE 6.2-26

HEAT TRANSFER COEFFICIENT VERSUS TIME
DECLG (C_D = 0.6) MINIMUM SAFEGUARDS
NOMINAL THOT (619.3 DEG-F)

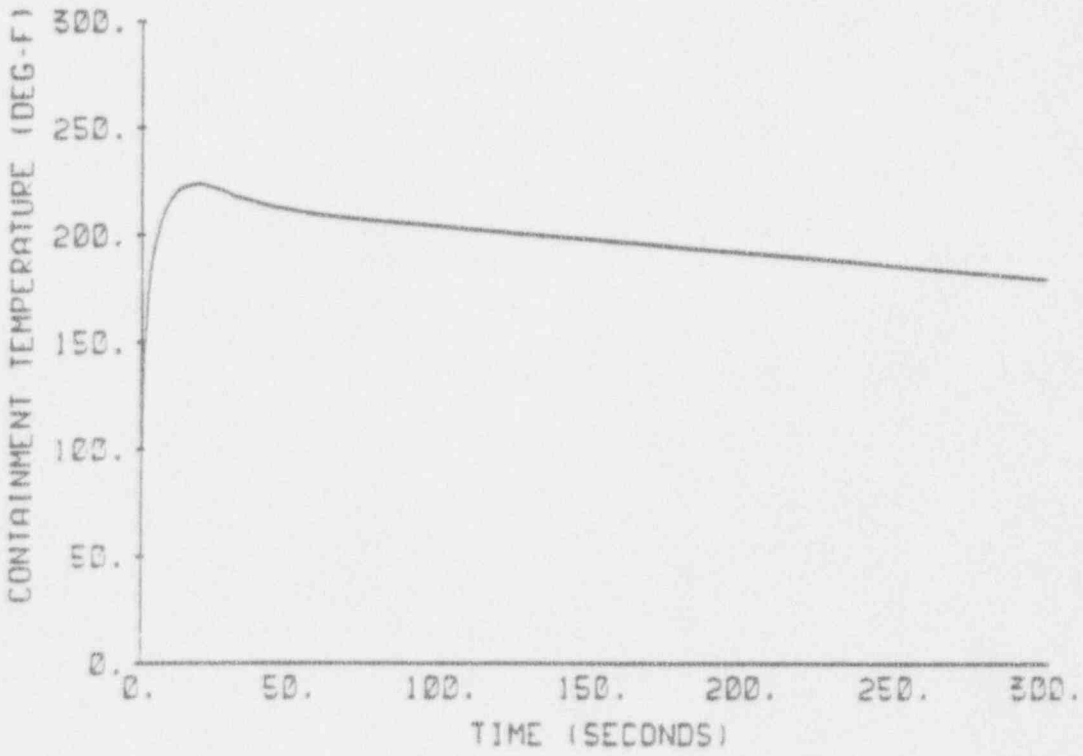

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FIGURE 6.2-27

CONTAINMENT TEMPERATURE VERSUS TIME
DECLG (C_D = 0.6) MINIMUM SAFEGUARDS
NOMINAL THOT (619.3 DEG-F)

TABLE 6.2-1

CONTAINMENT PEAK PRESSURE AND TEMPERATURE

BREAK	PEAK* PRESSURE (psig)	AVAIL-ABLE MARGIN (psi)	PEAK TEMPER-ATURE (°F)
Primary Side Ruptures			
Double-ended pump suction - Max SI	43.6	6.4	267
Double-ended pump suction - Min SI	43.2	6.8	266
0.6 double-ended pump suction	42.2	7.8	265
3-ft^2 pump suction	40.3	9.7	262
Double-ended hot leg	38.3	11.7	258
Double-ended cold leg	35.3	14.7	254
Secondary Side Breaks			
Full double-ended break - 102% power (steamline stop valve failure)	29.7	20.3	251
Full double-ended break - 70% power (steamline stop valve failure)	31.0	19.0	254
Full double-ended break - 30% power (steamline stop valve failure)	32.1	17.9	256
Full double-ended break - Hot Shut-down (steamline stop valve failure)	33.5	16.5	258
0.86-ft^2 split break - 102% power (steamline stop valve failure)	32.0	18.0	315
0.908-ft^2 split break - 70% power (steamline stop valve failure)	32.8	17.2	318
0.942-ft^2 split break - 30% power (steamline stop valve failure)	35.1	14.9	317
0.4-ft^2 split break - hot shutdown (steamline stop valve failure)	31.2	18.8	272

*If initial pressure is increased to 15.7 psia (maximum allowed by Technical Specifications) the peak pressures in Table 6.2-1 would be increased by 0.8 psi.

TABLE 6.2-2

ASSUMPTIONS FOR CONTAINMENT ANALYSIS - PART 1

Service water temperature (°F)	120
Refueling water temperature (°F)	120
RWST water volume (gal)	350,000
Initial Containment	
Temperature (°F)	120
Initial pressure (psia)	15.0*
Initial relative humidity (%)	50
Net free volume (ft^3)	2.75×10^6

*If initial pressure is increased to 15.7 psia (maximum allowed by Technical Specifications) the peak pressures in Table 6.2-1 would be increased by 0.8 psi.

TABLE 6.2-6

ACCIDENT SEQUENCE FOR DOUBLE-ENDED PUMP SUCTION BREAK MAX SI

	EVENT	TIME OF OCCURRENCE (sec)
1.	Accumulators begin injecting	15.0
2.	Safety injection begins	22.0
3.	End of blowdown	22.0
4.	Start fan coolers	40.0
5.	Start spray	45.0
6.	Stop accumulators injection	62.4
7.	End of reflood	106.8
8.	Containment peak pressure	129.1
9.	Containment depressurized to 25 psig	1130.0
10.	End of steam generator energy release	1284.0
11.	Recirculation, injection	1712.0
12.	Recirculation, spray	1712.0

TABLE 6.2-7

ACCIDENT SEQUENCE FOR DOUBLE-ENDED COLD LEG BREAK

EVENT	TIME OF OCCURRENCE (sec)
Accumulators begin injecting	12.0
Containment peak pressure	16.0
Safety injection begins	19.0
End of blowdown	19.0
Start fan coolers	40.0
Start spray	45.0
Stop accumulators injecting	37.2
End of reflood	263.6
Recirculation, injection	1,740.0
Recirculation, spray	1,740.0
Containment depressurized to 25 psig	25,000.0

TABLE 6.2-8

ACCIDENT SEQUENCE FOR DOUBLE-ENDED HOT LEG BREAK

EVENT	TIME OF OCCURRENCE (sec)
Accumulators begin injecting	13.0
Safety injection begins	18.5
End of blowdown	18.5
Start fan coolers	40.0
Containment peak pressure	40.6
Start spray	45.0
Stop accumulators injecting	48.7
End of reflood	73.9
Recirculation, injection	1,776.0
Recirculation, spray	1,776.0
Containment depressurized to 25 psig	250,000.0

TABLE 6.2-9

0.942 FT^2 SPLIT BREAK - 30% POWER - STEAMLINE STOP VALVE FAILURE/ACCIDENT SEQUENCE

EVENT	TIME OF OCCURRENCE (sec)
Main feedwater isolation	25.7
Steamline isolation	38.5
Start fan coolers	58.74
Containment peak temperature	59.0
Start sprays	137.4
Steam generator dryout	595.0
Containment peak pressure	387.0

TABLE 6.2-55

PASSIVE HEAT SINK DATA FOR MINIMUM

POST LOCA CONTAINMENT PRESSURE

A. Heat Sink Description

Slab Number	Description	Slab Material	Material Thickness (ft)	Surface Area (ft^2)
1	Structural Steel	Carbon Steel	0.02083	212107.98
2	Structural Steel	Carbon Steel	0.25583	320.84
3	Structural Steel	Carbon Steel	0.13250	88.23
4	Structural Steel	Carbon Steel	0.19792	299.25
5	Structural Steel	Carbon Steel	0.20833	892.5
6	Structural Steel	Carbon Steel	0.23958	782.25
7	Structural Steel	Carbon Steel	0.12500	1107.75
8	Structural Steel	Carbon Steel	0.10400	906.09
9	Structural Steel	Carbon Steel	0.04167	42144.90
10	Structural Steel	Carbon Steel	0.02500	44799.19
11	Structural Steel	Carbon Steel	0.16667	531.82
12	Structural Steel	Carbon Steel	0.18750	131.67
13	Internal Concrete	Concrete	1.0	101391.04
14	Internal Concrete	Concrete	1.0	14766.67
15	Containment Floor	Concrete	0.5	
	Containment Floor	Steel	0.03362	828.13
	Containment Floor	Concrete	0.5	
16-19	Foundation and Sump	Concrete	0.5	
	Foundation and Sump	Steel	0.02292	1850.20
	Foundation and Sump	Concrete	0.5	

TABLE 6.2-55 (Cont'd)

Slab Number	Description	Slab Material	Material Thickness (ft)	Surface Area (ft^2)
	Foundation and Sump	Concrete	0.5	
	Foundation and Sump	Steel	0.01563	10134.80
	Foundation and Sump	Concrete	0.5	
	Foundation and Sump	Concrete	0.5	
	Foundation and Sump	Steel	0.04899	23489.55
	Foundation and Sump	Concrete	0.5	
	Foundation and Sump	Concrete	0.5	
	Foundation and Sump	Steel	0.15276	3022.63
	Foundation and Sump	Concrete	0.5	
20	Containment Wall	Concrete	0.5	
	Containment Wall	Steel	0.02083	115872.75
	Containment Wall	Concrete	4.0	

TABLE 6.2-55 (Cont'd)

B. Thermophysical Properties

	Density lb/ft^3	Specific Heat Btu/lb °F	Thermal Conductivity Btu/hr-ft °F
Concrete	145	0.156	0.92
Carbon Steel	490	0.12	27.0

TABLE 6.2-66

LINER AND CONCRETE DESIGN TEMPERATURES

I. INSIDE FACE OF CONTAINMENT

A. Normal operating

Liner and Concrete	120°F

B. Accident Condition

Inside Face of Liner	280°F (Short-Term)
Inside Face of Concrete	208°F

II. OUTSIDE FACE OF CONTAINMENT

A. Normal Operating

1. Below Grade

a. Winter	50°F
b. Summer	80°F

2. Above Grade

a. Winter	0°F
b. Summer	100°F

3. Wall Area Neighboring Auxiliary Building	70°F (Operating Temp. of Auxiliary Building)

B. Accident Condition

The temperature at below and above grade are the same as those for Normal Operating.