May 29, 2001
NG-01-0721

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 – Extended Power Uprate. (TAC # MB0543)

Reference: NG-01-0660, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 – Extended Power Uprate. (TAC # MB0543)," dated May 11, 2001.

File: A-117, SPF-189

Dear Sir(s):

As discussed in the Referenced letter, we agreed to provide the requested additional information in multiple submittals to allow the Staff and their contractor to begin their audit calculations as soon as possible. Consequently, Attachment 1 to this letter contains our second installment of Responses to that Request for Additional Information (RAI). Please note that we have repeated the information contained in the Referenced letter for completeness. The few remaining items have been annotated as "LATER." We anticipate providing the remaining Responses within the next few days.

On May 9, 2001, a new set of questions was received electronically from the Staff. As a result of a conference call with the Staff on May 23, 2001, an additional question was added to the May 9 RAI. These new questions and our Responses have been appended to Attachment 1, as they are of a similar technical nature to those in the original RAI.

Please note that the response in Attachment 1 contains information that the General Electric Company (GE) considers to be proprietary in nature and subsequently, pursuant to 10 CFR 9.17(a)(4), 2.790(a)(4) and 2.790(d)(1), requests that such information be withheld from public disclosure. The portion of the text containing the proprietary information is identified with vertical sidebars in the right margin. An affidavit supporting this request is provided as Attachment 2 to this letter. Attachment 3 is the redacted version of Attachment 1, with the GE proprietary material removed, suitable for public disclosure.

No new commitments are being made in this letter.

A001

Please contact this office should you require additional information regarding this matter.

This letter is true and accurate to the best of my knowledge and belief.

NUCLEAR MANAGEMENT COMPANY, LLC

By Gary Van Middlesworth
Gary Van Middlesworth
DAEC Site Vice-President

State of Iowa
(County) of Linn

Signed and sworn to before me on this 29th day of May, 2001,

by Gary Van Middlesworth.

Nancy S. Franck
Notary Public in and for the State of Iowa

NANCY S. FRANCK
MY COMMISSION EXPIRES
9-28-01

Commission Expires

Attachments:
1) DAEC Responses to NRC Containment Systems Branch Request for Additional Information Regarding Proposed Amendment for Power Uprate
2) General Electric Affidavit of Proprietary Information
3) Redacted Version of DAEC Response to NRC Containment Systems Branch Request for Additional Information Regarding Proposed Amendment for Power Uprate

cc: T. Browning
R. Anderson (NMC) (w/o Attachments 1 & 2)
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
D. McGhee (State of Iowa) (w/o Attachments 1 & 2)
NRC Resident Office
Docu

General Electric Company

AFFIDAVIT

I, **George B. Stramback,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the enclosure to letter GEDA –AEP-555, *Response to NRC RAI Regarding the Containment Analysis*, (GE Company Proprietary), dated May 29, 2001. The proprietary information is delineated by bars marked in the margin adjacent to the specific material in the *Enclosure 3 to Letter GEDA-AEP-555 Clarified Response to NRC RAI Request #10.*

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains further details regarding the GE proprietary report NEDC-32980P, *Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate,* Class III (GE Proprietary Information), dated November 2000, which contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to

perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR").

The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 29th day of May 2001.

George B. Stramback
General Electric Company

Subscribed and sworn before me this 29th day of May 2001.

VICKY D. SCHROER
Commission # 1224251
Notary Public - California
Santa Clara County
My Comm. Expires Jun 12, 2003

Notary Public, State of California

Redacted Version of
DAEC Responses to NRC
Containment Systems Branch
Request for Additional Information
Regarding Proposed Amendment for Power Uprate

1. Provide input to the computer calculation of short-term containment pressure and temperature. Details of the input and method of transmittal to the NRC will be decided mutually in a conference call with the licensee, NRC staff and NRC contractor[1]. Preliminary list is given in Attachment 1 below.

DAEC Response:

See Attached Table – 1.

2. Provide input to computer calculation of long-term containment pressure and temperature calculations. Details of the input, method of transmittal to the NRC will be decided mutually in a conference call with the licensee, NRC staff and the NRC contractor. Preliminary list is given in Attachment 1 below.

DAEC Response:

See Attached Tables – 1 and 2 for the complete set of inputs for the long-term DBA-LOCA, NPSH, and the representative main steamline break. The "time-dependent" information, other than the decay heat values in Tables 2a and 2b, will be provided in a subsequent submittal.

3. Provide short term and long term results (curves or tables of calculated values as a function of time) of Duane Arnold calculations for:

(A)drywell short term pressure and temperature,
(B)suppression pool short term temperature
(C)wetwell atmosphere short term pressure and temperature
(D)suppression pool long term temperature
(E)wetwell atmosphere long term pressure and temperature

If the long term calculation results are different from those used for calculating NPSH, provide the suppression pool long term temperature and wetwell atmosphere long term pressure and temperature used for the NPSH calculation.

DAEC Response:

For containment analysis results, refer to the following figures for the requested information:

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[1] Information Systems Laboratories, Inc., Rockville, Maryland

(A) Short-term drywell pressure – Figure 1
Short-term drywell temperature – Figure 2

(B) Short-term suppression pool temperature – Figure 2*

(C) Short-term wetwell atmosphere pressure – Figure 1
Short-term wetwell atmosphere temperature – Figure 2*

* Note that the M3CPT methodology for short-term containment response assumes that the wetwell airspace is in thermal equilibrium with the suppression pool at all times to maximize the containment pressure and temperature response.

(D) Long-term suppression pool temperature:
- DBA-LOCA - Figure 5
- NPSH - Figure 8

(E) Long-term wetwell atmosphere pressure
- DBA-LOCA - Figure 3
- NPSH – Figure 6

Long-term wetwell atmosphere temperature
- DBA-LOCA - Figure 4
- NPSH - Figure 7

In addition, based upon the conference call with the Staff, a representative main steamline break should be included in the data request. Consequently, we are also providing the following figures of the results of that analysis:

(F) Drywell and wetwell atmosphere pressure – Figure 9
Short-term Drywell temperature – Figure 10
Long-term Drywell temperature – Figure 11
Wetwell atmosphere temperature – Figure 12
Suppression Pool temperature – Figure 13

4. Explain why the 31.7 GWD/Short ton is conservative for Duane Arnold decay heat calculations.

DAEC Response:

The exposure used in the decay heat calculations was taken from the power uprate equilibrium fuel cycle, as outlined in PUSAR Section 2.1, and then adjusted to provide extra conservatism to account for any future cycle-to-cycle differences. The core average exposure for the DAEC power uprate equilibrium fuel cycle was 34 GWd/Metric Ton. This value was rounded upward to 35 GWd/Metric Ton, which is equivalent to 31.7

GWd/Short Ton. This value was used to evaluate various decay heat parameters and to estimate the average in-core irradiation time. A direct calculation of the irradiation time using this exposure resulted in a value of 3.31 years. This was rounded upward to 3.5 years for an extra degree of conservatism. Finally, an allowance of two-sigma uncertainty was applied to the decay heat table. All of these individual conservatisms ensure that the overall calculation for the decay heat curve (Table 2, attached) is bounding for the DAEC power uprate conditions.

5. Explain why a higher power level increases subcooling in the vessel downcomer region.

<u>DAEC Response:</u>

This is a characteristic of a constant pressure power uprate with no increase in core flow. The following values were taken from results of the reactor heat balance summarized in PUSAR Section 1.3.1, Table 1-2 and Figure 1-1. Core flow remains at 49 Mlb/hr. Reactor pressure remains at 1040 psia, so the reactor coolant saturation temperature remains at 540 °F. The increase in vessel steam flow from 7.172 Mlb/hr to 8.352 Mlb/hr reduces the return flow to the downcomer (i.e., core flow minus vessel steam flow) from 41.83 Mlb/hr to 40.65 Mlb/hr at 540 °F. There is a corresponding increase in Feedwater flow from 7.221 Mlb/hr at 424 °F to 8.414 Mlb/hr at 431.4 °F. Despite the increase in Feedwater temperature, the bulk average downcomer temperature decreases from 531 °F to 529 °F. Therefore, the downcomer subcooling increases from about 9 °F to 11 °F, at the saturation pressure of 1040 psia in the reactor steam dome. A similar decrease in core inlet enthalpy can be seen in Table 1-2.

6. Describe any changes made in assumptions or physically to the ultimate heat sink and the residual heat removal heat exchanger which affect the long term containment analysis.

<u>DAEC Response:</u>

No changes in the current analysis assumptions for either the Ultimate Heat Sink (PUSAR 6.4.5) or the Residual Heat Removal heat exchanger (PUSAR 3.9.2 and 6.4.1.1.2) were made.

7. Describe the steam line break scenario which requires calculating a 120 day containment response. Describe in more detail the calculation which is done for this scenario or provide the calculation.

<u>DAEC Response:</u>

In accordance with the requirements of NUREG-0737, Item II.K.3.28, the Automatic Depressurization System (ADS) valves are required to be capable of actuating for 100 days, post-accident. To ensure that these valves remain environmentally qualified, the limiting breaks for containment temperature, steamline break inside containment, were analyzed. For extra conservatism, the scenerio was extended out to 120 days.

A representative steamline break analysis is provided in our Response to Questions 2, 3 and 10.

8. Explain why the EPU peak drywell gas temperature is less than for the current licensing basis (Table 4-1 of NEDC-32980P).

DAEC Response:

The current licensing basis analysis for main steamline breaks did not credit heat sinks in the containment (i.e., drywell shell and vents). The EPU analysis removed the extra conservatism and included these heat sinks (See Table – 1), which lowered the peak gas temperature from the current analysis. This was done to ensure that the existing Equipment Qualification peak temperature in the Drywell remained bounding.

9. Explain the reason for redoing the subcompartment analyses assuming the break flow is subcooled liquid (Section 4.1.2.3 of NEDC-32980P).

DAEC Response:

As noted in PUSAR Section 4.1, "... blowdown flowrate is dependent on the reactor initial themal hydraulic conditions, primarily the vessel dome pressure and the vessel subcooling, and to a much lesser extent, on the decay heat and vessel liquid and metal mass and energy." In accordance with the extended power uprate (EPU) licensing topical reports (ELTRs), the re-analysis was initially identified based on the changes in downcomer subcooling as a result of EPU, as there is no dome pressure increase associated with the DAEC EPU. Subsequently, once the new analysis was initiated, additional considerations were identified which also supported the decision for re-analysis of the current licensing basis. It was found that the original analysis had not been updated during the previous power uprate in 1985, which increased reactor pressure 15 psi. Therefore, it was necessary to update this analysis to address the impact of the previous pressure increase. It was also noted that the original break flow model used (Moody's two-phase slip-flow model) assumed only saturated fluid conditions. This model generates a higher critical mass flux if subcooled liquid conditions are present. Analysis results are influenced by both the additional subcooling due to EPU, as well as the previous pressure increase.

As discussed in UFSAR Section 3.6.2.2.4.8, the original DAEC analysis is a double-ended break of the recirculation inlet nozzle safe end inside the reactor shield wall. Using Moody's two-phase slip-flow model, based on a pressure of 1050 psia (1040 psia dome + 10 psi hydrostatic head at the break) and a break flow enthalpy of 550 BTU/lbm, yielded a maximum blowdown mass flux of 8000 lbm/sec per ft^2. It should be noted that this is a saturated liquid condition.

The UFSAR analysis also considered the maximum pipe dislocation (restrained by the shield wall penetration) to yield an effective break cross-sectional area of 1.61 ft^2. This yielded a maximum break flowrate of 12,860 lbm/sec.

Using the Moody Slip Flow model with subcooled liquid blowdown, the EPU increase in subcooling, updated break pressure of 1065 psia (1050 psia original analysis + 15 psi increase from 1985 uprate) and break flow enthalpy 529 BTU/lbm, yields a maximum blowdown mass flux of 9316 lbm/sec per ft^2. Again, using the 1.61 ft^2 effective break area, results in a break flowrate of 14,999 lbm/sec.

Most of the increase in the blowdown flowrate is a result of the change in methodology to consider subcooled break flow, secondarily by the previous increase in reactor pressure, and a very small amount from the increased subcooling due to EPU. Of the above increase in blowdown mass flux, 92% is due to the change in methods and pressure, only 8% is due to the change in subcooling associated with EPU.

The consequences of this increase in break flow were determined to remain within the design limits of the shield wall and end pugs. For the limiting case of the recirculation outlet nozzle (N1) shield plug, the pressurization transient inside the shield wall increases the peak pressure from 16.25 psi in the original analysis (UFSAR Table 3.6-3) to 18.98 psi including all the above changes. For the N1 shield plug, the design pressure is 20 psi. Thus, considering all the above changes results in a margin of 5.1% to this limit.

10. Describe or reference the methods and assumptions used to calculate mass and energy release for the short term and long-term pressure and temperature calculations. Verify that the HEM model is being used in a manner consistent with the staff SER on NEDO-21052, September 1975.

DAEC Response:

Methods and Key Assumptions

1. Computer Codes used in Analysis

1.1 GE Computer Code M3CPT

[[General Electric Proprietary Information Redacted]]

1.2 GE Computer Code SHEX

[[General Electric Proprietary Information Redacted]]

2. Key Assumptions

2.1 Key Assumptions for Short-Term Containment Response

1. The HEM break flow model is used to calculate the containment pressure and temperature response.
2. The power level for the power/flow point analyzed includes an additional 2% power, consistent with Regulatory Guide 1.49.
3. The break is an instantaneous double-ended rupture of a recirculation suction line.
4. No credit is taken for the passive structural heat sinks.
5. The initial vent submergence and the suppression pool volume correspond to that of the Technical Specification (TS) High Water Level (HWL).
6. The initial drywell and wetwell pressures and drywell relative humidity are selected so as to maximize the initial mass of non-condensable gases.
7. The wetwell airspace is in thermal equilibrium with the suppression pool at all times to maximize the containment pressure and temperature.

2.2 Key Assumptions for Long-Term Containment Response

2.2.1 Key Assumptions for Long-Term DBA-LOCA

The following are the key assumptions for the long-term DBA-LOCA. Most of these assumptions are applicable to the DBA-LOCA for long-term NPSH evaluation and the steam line break cases. Specific assumptions for the other cases that are different from those for the DBA-LOCA are specified separately.

1. The reactor is operating at 102% of 120% ORTP (i.e., 1950 MWt) with an initial reactor pressure of 1055 psia.
2. The reactor core power includes fission energy, fuel relaxation energy, metal-water reaction energy and ANS 5.1 + 2σ decay heat for fuel applicable up to GE14 with 24-month fuel cycle.
3. Reactor blowdown flow rates are based on Moody's HEM.
4. The reactor vessel control volume is assumed to include the fluid and structural masses and energy of the primary system components including reactor vessel, recirculation loops, main steam lines to the inboard isolation valve, RCIC steam line, RHR shutdown line, LPCI line, Core Spray line and HPCI steam line.
5. Concurrent with the postulated LOCA, a loss of offsite power occurs.

6. Only minimum diesel power is available. This results in only one RHR loop with one heat exchanger available for containment cooling, starting at 10 minutes (600 seconds).

7. The portion of the feedwater inventory at a temperature higher than the peak suppression pool temperature, after absorbing additional energy from the feedwater piping as it flows toward the vessel, is injected into the vessel. This assumption is used to maximize the suppression pool temperature. This hot portion of the feedwater inventory is transferred to the vessel regardless of the availability considerations of feedwater and condensate pumps.

8. Heat and mass transfer from the suppression pool to the wetwell airspace is determined mechanistically.

9. The DBA-LOCA is the instantaneous double-ended guillotine break of the recirculation suction line at the reactor vessel nozzle safe-end to pipe weld. The effective break area is 2.523 ft^2, which includes the bottom head drain line.

10. The initial suppression pool water volume corresponds to the TS Low Water Level (LWL) to maximize the suppression pool temperature response.

11. The RHR service water temperature is at the maximum value of 95°F to maximize the suppression pool temperature.

12. Passive heat sinks in the drywell, wetwell airspace and suppression pool are conservatively neglected to maximize the suppression pool temperature. Heat transfer from the primary containment to the reactor building is also conservatively neglected.

13. Drywell fan coolers are inactive.

14. Control rod drive flow is zero.

15. All Core Spray and LPCI/RHR pumps have 100% of their motor horsepower rating converted to pump heat which is added either to the Reactor Pressure Vessel (RPV) liquid or suppression pool water. This assumption is used to maximize the suppression pool temperature response.

16. Main Steam Isolation Valve (MSIV) closure starts at 0.5 seconds after the initiation of the event and full closure is achieved at 3.0 seconds after closure is initiated.

17. Only 6 wetwell-to-drywell vacuum breakers are assumed to be active.

18. CST water inventory is not available for vessel makeup.

2.2.2 Specific Assumptions for DBA-LOCA for Long-Term NPSH Evaluation

The assumptions listed in Section 2.2.1 are applicable, with the following exceptions.

1. Minimum initial drywell and wetwell pressures and maximum initial drywell relative humidity are assumed. This is done to minimize the mass of non-condensables in the containment.
2. Containment cooling is achieved by operating the RHR loop, with heat exchanger, in the containment spray mode (drywell and wetwell sprays), instead of the vessel injection mode.
3. Passive heat sinks in the drywell and wetwell airspace are modeled.
4. Containment leakage effects are considered.
5. There is a single failure of a diesel generator so that only one RHR loop, with one heat exchanger, is available for containment cooling.
6. All 7 wetwell-to-drywell vacuum breakers are assumed to be active.

2.2.3 Specific Assumptions for Steam Line Breaks

The assumptions listed in Section 2.2.1 are applicable, with the following exceptions.

1. Breaks of 0.01 ft^2, 0.1 ft^2, 0.25 ft^2 and 1.0 ft^2 at a Main Steam Line occur.
2. Containment cooling is achieved by operating the RHR loop, with heat exchanger, in the containment spray mode (drywell and wetwell sprays), instead of the vessel injection mode.
3. Passive heat sinks in the drywell and wetwell airspace are modeled.
4. HPCI is operational, while RCIC is not available.

3. References

1. "Mark I Containment Program Load Definition Report," NEDO-21888, Revision 2, November 1981.
2. "The GE Pressure Suppression Containment Analytical Model," NEDO-10320, April 1971.
3. NUREG-0800, U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 6.2.1.1.C, "Pressure-Suppression Type Containments," Revision 6, August 1984.
4. "Safety Evaluation Report – Mark I Containment Program," NUREG-0661, July 1980.
5. "The General Electric Mark III Pressure Suppression Containment System Analytical Model," NEDO-20533, June 1974.

6. Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate", NEDC-32424P-A, Class III, February 1999 (ELTR-1).
7. "Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," Letter from Ashok Thadani (NRC) to Gary L. Sozzi (GE), July 13, 1993.
8. "Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate," NEDC-32980P, Revision 1, April 2001.
9. "Review of General Electric Topical Report NEDO-21052, 'Maximum Discharge of Liquid-Vapor Mixtures from Vessels'," Letter from D. Eisenhut (NRC) to L.J. Sobon (GE), MFN-004-79, December 27, 1978.

11. Other than the effects due to the increase in power, verify that no other changes in assumptions in the NPSH calculation have been made in the assumptions or input since the Duane Arnold responses to GL 97-04.

DAEC Response:

As stated in PUSAR Section 4.1.1.1(b), the only assumption that was changed from the NPSH analysis for GL 97-04 was in the decay heat curve which assumed a finite versus infinite fuel exposure.

DAEC Responses to
Second NRC Set of
Requests for Additional Information from
Containment Systems Branch
Regarding Proposed Amendment for Power Uprate

1.0 With regard to internal flooding resulting from main steam line break (MSLB) and feedwater line break (FWLB), Nuclear Management Company (NMC) stated, "there is no impact on feedwater line break flooding for the extended power uprate (EPU), because flooding for a feedwater line break is bounded by flooding for a main steam line break." However, NMC did not provide justification or discussion of the evaluation to demonstrate why MSLB would be the bounding break with regard to internal flooding outside the containment.

Please provide a detailed description of the analysis performed to evaluate the effects of flooding outside the containment resulting from FWLB/MSLB under EPU conditions.

DAEC Response:

The original PUSAR statement was based upon HELB analyses that are primarily interested in the effects of the break on the compartment pressure and temperature response; compartment flooding is generally a secondary concern in these analyses. It would have been more accurate to state that neither the MSLB or FWLB event create the potential for a significant flooding event outside containment. The worst-case internal flooding in the turbine building is from a pipe break in the Circulating Water System (Ref. UFSAR 10.4.5.3), which releases approximately 20.0 Mlbm of water into the turbine building. This is substantially more than the inventory available from either the Feedwater/Condensate system, assuming total runout (approximately 1.2 Mlbm), or in the Main Steam piping up to the time of Main Steamline Isolation Valve (MSIV) closure (<0.1 Mlbm) (Ref. UFSAR 15.6.5.1.3), following a line break in these systems outside containment.

The resulting pressure increase from a pipe break (either MSLB or FWLB) in the steam tunnel would cause the blowout panel to open (opening pressure = 0.5 psid) almost immediately and the break inventory would cascade from the tunnel into the turbine building, as the blowout panel starts at the floor elevation of the steam tunnel. Thus, a break in the steam tunnel becomes a flooding event in the turbine building.

These evaluations are not impacted by EPU as there is no change in the Circulating Water System (PUSAR Section 6.4.2), the Feedwater/Condensate system inventory, primarily the main condensor hotwell capacity, is unchanged (PUSAR Section 7.2), and the amount of break flow from a MSLB is not increased by EPU (PUSAR Section 10.1.1.1), as there is no pressure increase associated with EPU.

2.0 The environmental qualification(EQ) of non-metallic components, (i.e. seals, gaskets, lubricants, diaphragms, etc.)[1] has not been addressed. Please demonstrate that plant operations at the proposed EPU level will have no impact on the EQ of mechanical equipment with non-metallic components located inside and outside containment.

DAEC Response:

The design control program described in PUSAR Section 10.3.2 ensures that non-metallic components, such as seals, gaskets, lubricants, and diaphragms, are properly specified and procured for the environment in which they are intended to function.

3.0 In Section 4.7 for Post-LOCA Combustible Gas Control, it is indicated that to maintain the containment atmosphere below the 5% flammability limit for seven days post-LOCA, the minimum stored volume of nitrogen requirement increases from 50,000 scf for the current power to 67,000 cfm for EPU power level and that the CAD nitrogen storage system has sufficient capacity to accommodate this. Please indicate what is the storage capacity of the CAD system and how the minimum required capacity will be assured.

DAEC Response:

The CAD system consists of ten 51 ft^3 tanks, with a maximum pressure of 2450 psig. Based on the ideal gas law, when filled to maximum capacity (2450 psig), at the worst expected ambient temperature (these tanks are outside the building) of 120°F, would provide a minimum volume of approximately 75,000 scf. Thus, the tanks have sufficient capacity for the new required volume.

Per Technical Specification (TS) requirement SR 3.6.3.1.1, the CAD volume must be verified every 31 days by checking tank temperature and pressure to ensure the minimum volume specification is met. As stated in PUSAR Table 11-1, the TS required volume for CAD is increased from its current value (50,000 scf) to the EPU value of 67,000 scf.

4.0 Section 4.1.2.3 of the PUSAR indicates "The results of the updated calculations including the effects of the EPU indicate that the biological shield wall and component designs remain adequate, because there is sufficient pressure margin available." Please provide clarification of the numerical values of these margins.

DAEC Response:

See Response to Question #9 above.

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1 SPLB does not have the review responsibility for the mechanical component (i.e. pumps, valves, heat exchangers, etc.).

TABLE – 1
Input Parameter to Containment Analysis for
DAEC Extended Power Uprate

Parameter \ Case	Unit	Short-Term DBA-LOCA	Long-Term DBA-LOCA	Long-Term DBA-LOCA for NPSH	Long-Term 0.01 ft^2 Steam Line Break
GE Computer Code used		M3CPT	SHEX	SHEX	SHEX (NOTE: See Question 10 above.)
Break Critical Flow Model		HEM	HEM	HEM	HEM
Containment Volumes					
Drywell (including free volume of vents)	ft^3	130,000	130,000	130,000	130,000
Wetwell Atmosphere	ft^3	94,070	96,670	96,670	96,670
Wetwell Liquid	ft^3	61,500	58,900	58,900	58,900
Initial RPV Water Volume (include liquid in recirc, LPCI, CS, HPCI, RCIC and RHR shutdown piping)	ft^3	8,224(1)	7,431	7,431	7,431
Recirc Suction Nozzle Inside Diameter	in	19.75	19.75	19.75	19.75
Break Area					
Recirc Suction Nozzle	ft^2	2.127	2.127	2.127	NA
Jet Pump Nozzles	ft^2	0.387	0.380	0.380	NA
Bottom Head Drain Nozzle	ft^2	0.(2)	0.016	0.016	NA

TABLE – 1
Input Parameter to Containment Analysis for
DAEC Extended Power Uprate

Parameter \ Case	Unit	Short-Term DBA-LOCA	Long-Term DBA-LOCA	Long-Term DBA-LOCA for NPSH	Long-Term 0.01 ft^2 Steam Line Break
Total	ft^2	2.514(2)	2.523	2.523	0.01
Torus Geometry Description					
Inside Diameter	ft	25.667	25.667	25.667	25.667
Upper Torus Thickness	ft	0.042	0.042	0.042	0.042
Lower Torus Thickness (not modeled)	ft	0.045	0.045	0.045	0.045
Baffles Dimensions, Mass and Location		Not Modeled	Not Modeled	Not Modeled	Not Modeled
Initial Suppression Pool Depth	ft	10.46(3)	7.449(4)	7.449(4)	7.449(4)
Suppression Pool Surface Area (assumed constant)	ft^2	7,763	7,763	7,763	7,763
Pump Suction Location		Not Modeled	Not Modeled	Not Modeled	Not Modeled
Initial Drywell Air and Vapor Masses are based on the following Conditions					
Pressure	psig	2.3	2.3	0.5	2.3
Temperature	°F	135	135	135	135
Relative Humidity	%	20	20	100	100
Initial Wetwell Air and Vapor Masses are based on the following Conditions					
Pressure	psig	2.3	2.3	0.5	2.3
Temperature	°F	95	95	95	95

TABLE – 1
Input Parameter to Containment Analysis for
DAEC Extended Power Uprate

Parameter \ Case	Unit	Short-Term DBA-LOCA	Long-Term DBA-LOCA	Long-Term DBA-LOCA for NPSH	Long-Term 0.01 ft^2 Steam Line Break
Relative Humidity	%	100	100	100	100
Initial Suppression Pool Water Mass	lbm	3.817E6	3.656E6	3.656E6	3.656E6
Initial Submergences					
Downcomers (Vents)	ft	3.359	3.026	3.026	3.026
SRV Discharge Lines / Quenchers	ft	Not Modeled	6.125	6.125	6.125
Heat Structures Properties					
Drywell and Wetwell Internal Metal Structures and Vents (Steel)		Not Modeled	Not Modeled		
Density	lbm/ft^3	NA	NA	490	490
Specific Heat	Btu/lbm-°F	NA	NA	0.11	0.11
Thermal Conductivity	Btu/hr-ft-°F	NA	NA	26	26
Drywell and Wetwell Air Properties					
Constant Volume Specific Heat, C_v	Btu/lbm-°F	0.171	0.171	0.171	0.171
Ratio of Specific Heats, γ		1.4	1.4	1.4	1.4
Ideal Gas Constant, R	lb_f-ft/lbm-°F	53.34	53.34	53.34	53.34
Suppression to Drywell Vacuum Breakers		Not Modeled			

TABLE – 1
Input Parameter to Containment Analysis for
DAEC Extended Power Uprate

Parameter \ Case	Unit	Short-Term DBA-LOCA	Long-Term DBA-LOCA	Long-Term DBA-LOCA for NPSH	Long-Term 0.01 ft^2 Steam Line Break
Number		NA	6	7	6
Flow Area (per Vacuum Breaker)	ft^2	NA	1.396	1.396	1.396
Differential Pressure Setpoint	psid	NA	0.35	0.35	0.35
Loss Coefficient (per Vacuum Breaker)		NA	2.41	2.41	2.41
RX Building to Suppression Vacuum Breakers		Not Modeled	Not Modeled	Not Modeled	Not Modeled
Number		NA	NA	NA	NA
Flow Area (per Vacuum Breaker)	ft^2	NA	NA	NA	NA
Differential Pressure Setpoint	psid	NA	NA	NA	NA
Opening Time	sec	NA	NA	NA	NA
Loss Coefficient (per Vacuum Breaker)		NA	NA	NA	NA
Drywell Spray Flow Rate	gpm	Not Modeled	Not Modeled	4,560	4,560
Wetwell Spray Flow Rate	gpm	Not Modeled	Not Modeled	240	240
Normalized Decay Heat vs Time		Table 2a	Table 2b	Table 2b	Table 2b
Operating Pumps					
RHR/LPCI Pumps		Not Modeled			
Number of Operating Pumps		NA	2 before 600 sec	2 before 600 sec	2 before 600 sec

TABLE – 1
Input Parameter to Containment Analysis for
DAEC Extended Power Uprate

Parameter \ Case	Unit	Short-Term DBA-LOCA	Long-Term DBA-LOCA	Long-Term DBA-LOCA for NPSH	Long-Term 0.01 ft^2 Steam Line Break
			1 after 600 sec	1 after 600 sec	1 after 600 sec
Heat Exchanger K-factor (per HX)	Btu/sec-°F	NA	135	141	135
Pump Heat (per Pump)	hp	NA	600	600	600
Core Spray Pumps		Not Modeled			
Number of Operating Pumps		NA	1	1	1
Pump Heat (per Pump)	hp	NA	700	700	700
Modeling of Condensation on Containment Walls		Not Modeled	Not Modeled	Yes (Uchida)	Yes (Uchida)

Notes:

(1) The higher value used in the short-term analysis is due to the inclusion of liquid mass in the feedwater piping in the RPV liquid mass, whereas the long-term analysis treats the feedwater in a different manner (see Attachment 2).

(2) The short-term analysis did not model the bottom head drain flow path. The peak values for the parameters of primary interest for this analysis (peak drywell pressure and temperature) occur in the early stages of the DBA-LOCA (at about 17 seconds). During this early stage, the flow contribution through the recirculation line flow paths is sufficient to maintain a high enough pressure at the break location to ensure critical flow through the break. Because this flow is sufficient to maintain critical flow conditions at the break location, the contibution of any flow through the bottom head drain flow path is negligible.

(3) The initial suppression pool depth is the adjusted value determined by the M3CPT calculations, based on the suppression pool volume input corresponding to the TS HWL.

(4) The torus-shaped suppression pool is converted into a rectangle in the SHEX code by retaining the same pool surface area, same vent and SRV quencher submergences and preserving the total pool volume. The mass of the vertical water columns projected by the downcomers from the pool surface to the bottom of the rectangle is first subtracted from the total pool volume before calculating the pool depth. Thus, the calculated pool depth times the pool surface does not equal the total pool volume.

TABLE – 2a

Normalized May-Witt Shutdown Power Values
for DAEC Extended Power Uprate
(Sum of Decay Heat, Fuel Relaxation, and Metal-Water Reaction Energy)

Time (sec)	Shutdown Power

[[General Electric Proprietary Information Redacted]]

TABLE – 2b

Normalized ANS 5.1 + 2σ Shutdown Power Values
for DAEC Extended Power Uprate
(Sum of Decay Heat, Fuel Relaxation and Metal-Water Reaction Energy)

Time (sec)	Shutdown Power

[[General Electric Proprietary Information Redacted]]

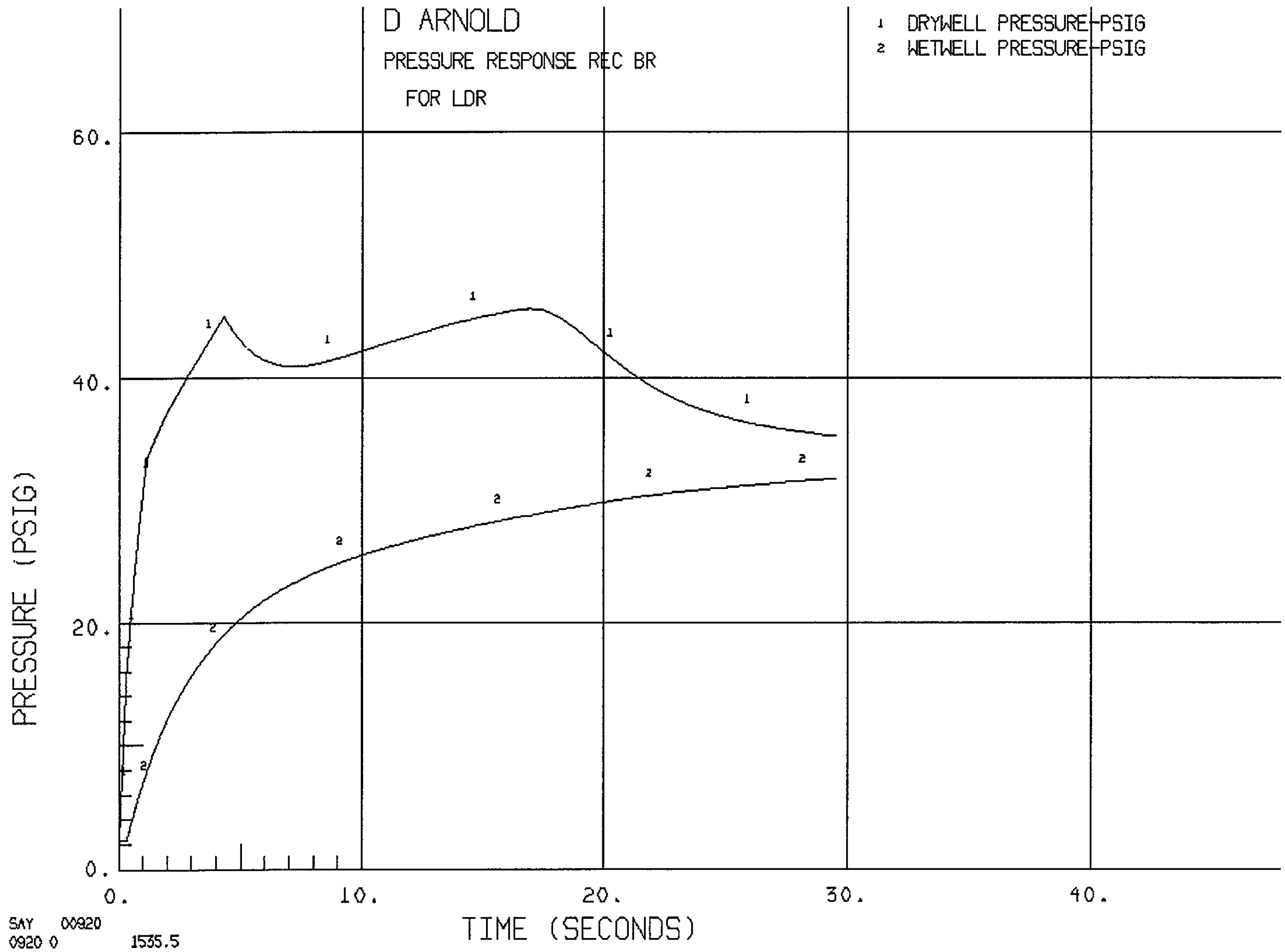

Figure 1: Containment Pressure Response For Short-Term DBA LOCA

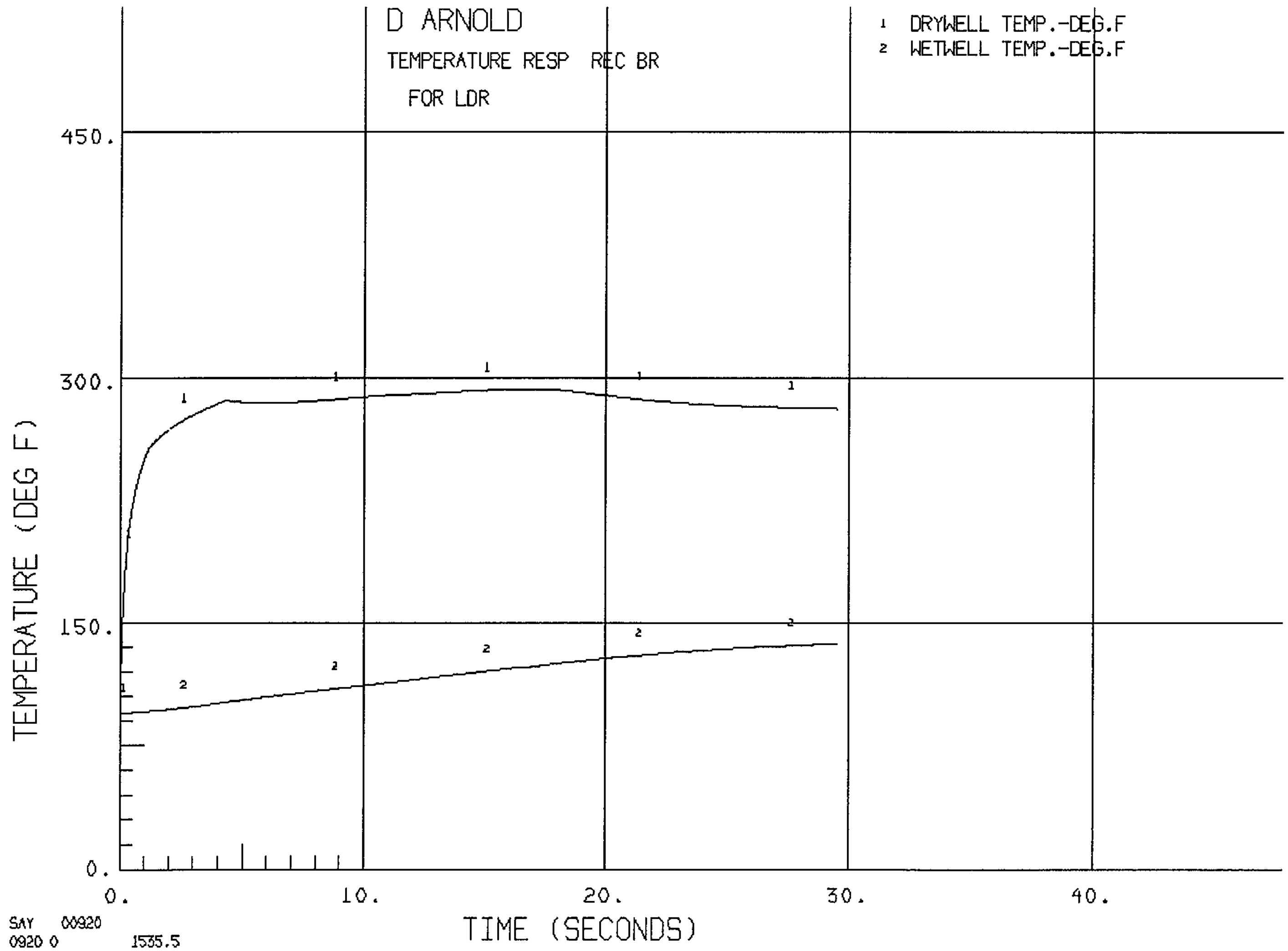

Figure 2: Containment Temperature Response For Short-Term DBA LOCA

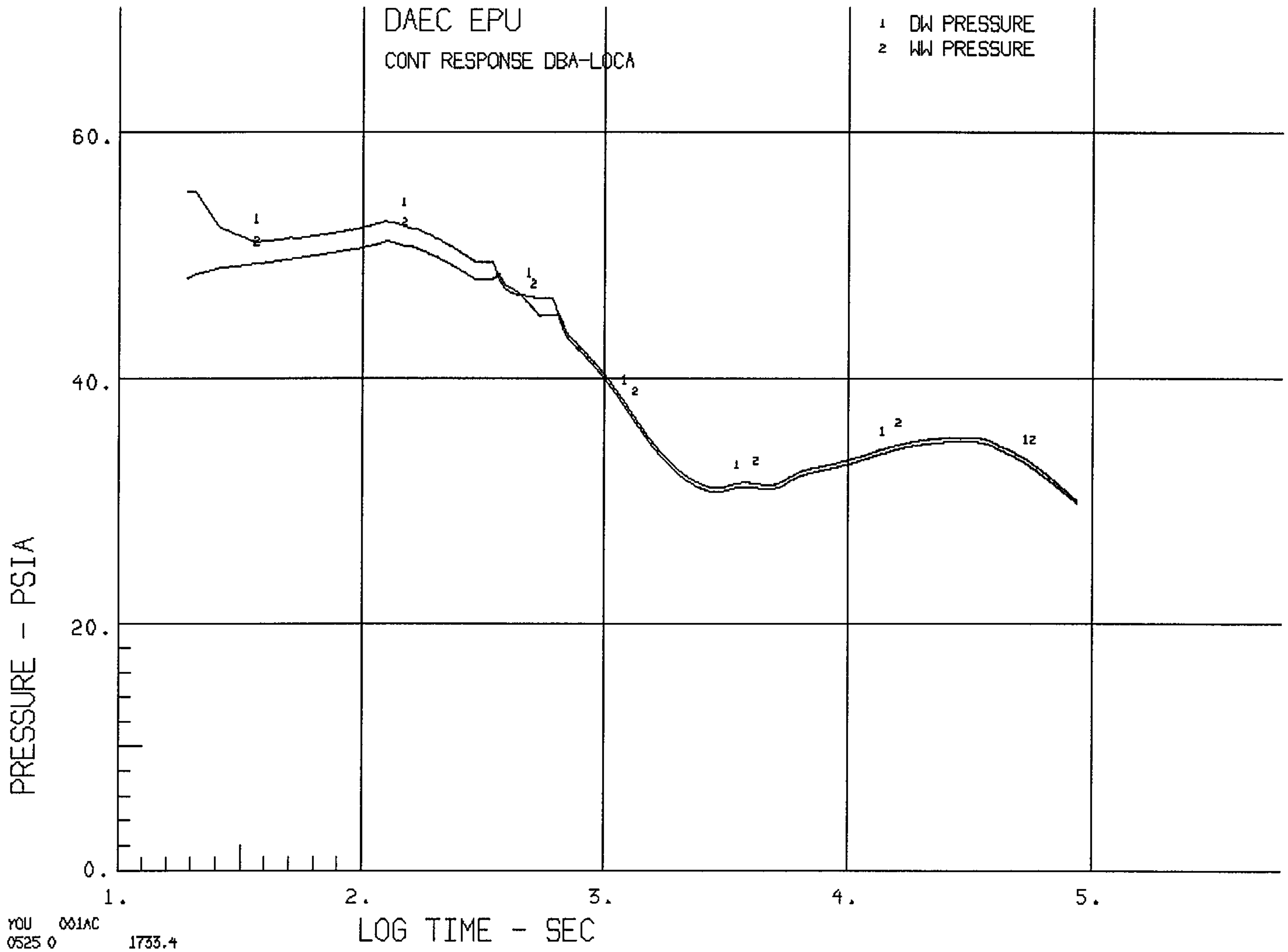

Figure 3: Containment Pressure Response For Long-Term DBA LOCA (UFSAR Case 4)

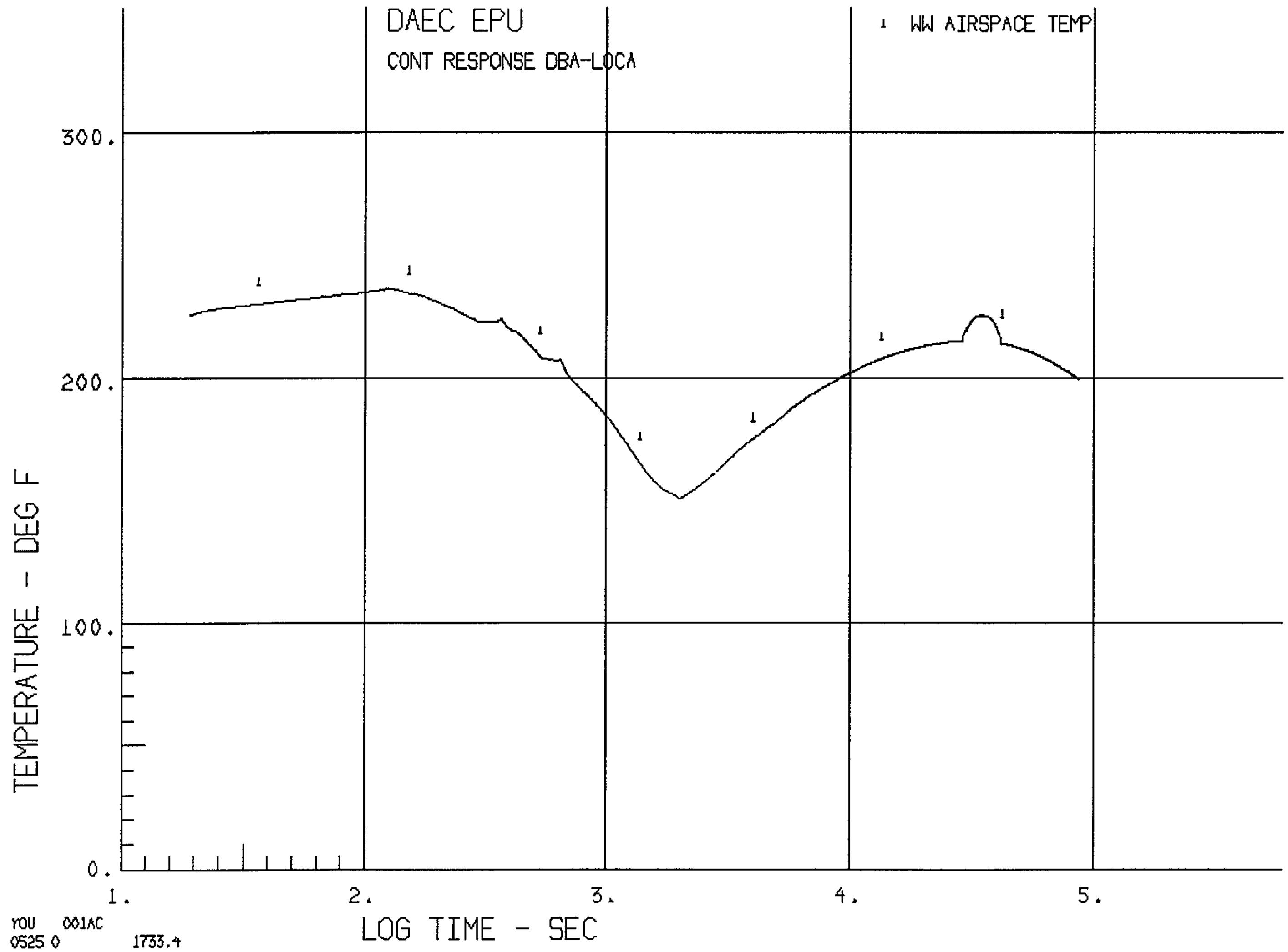

Figure 4: Wetwell Airspace Temperature Response For Long-Term DBA LOCA (UFSAR Case 4)

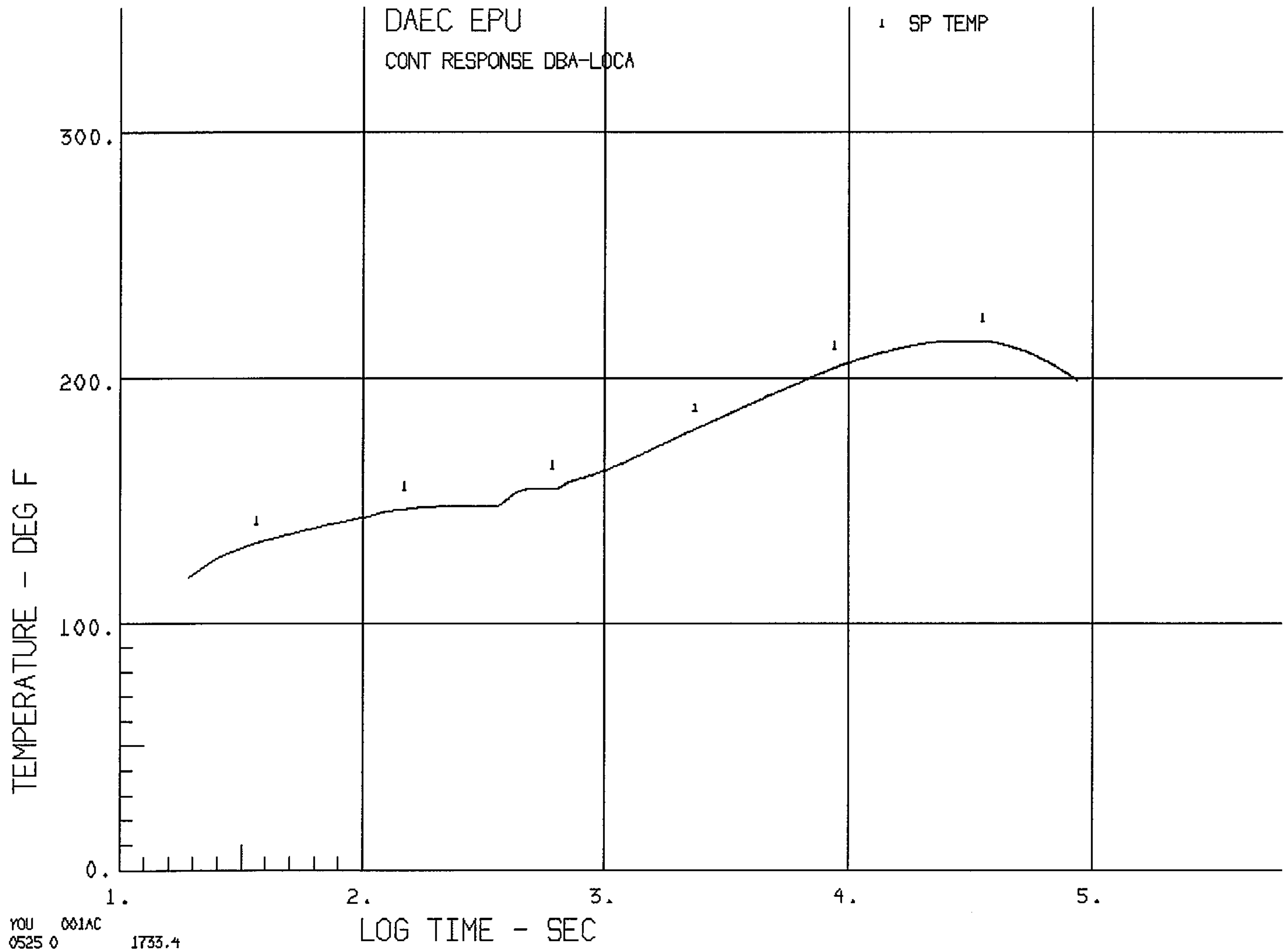

Figure 5: Suppression Pool Temperature Response For Long-Term DBA LOCA (UFSAR Case 4)

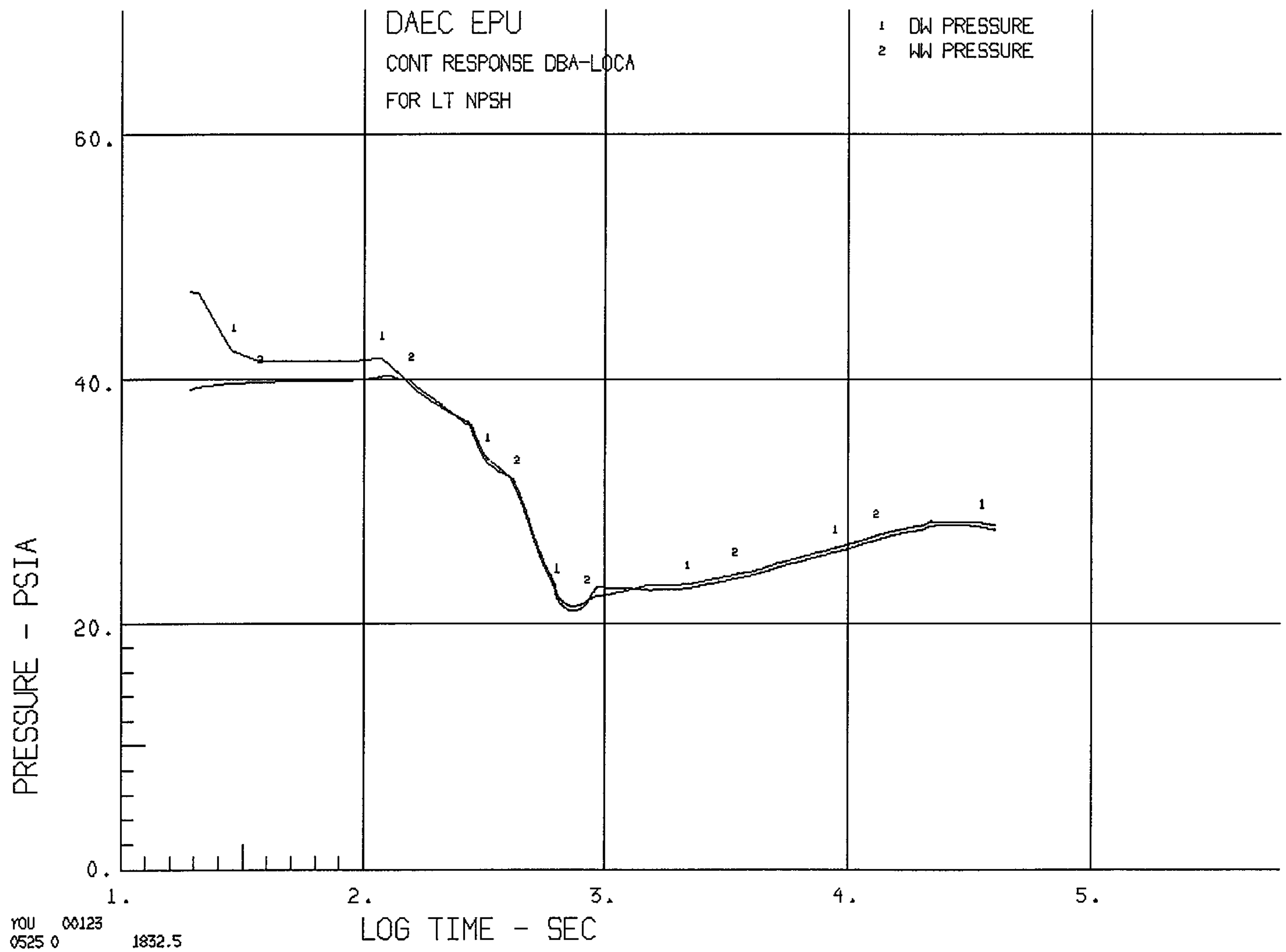

Figure 6: Containment Pressure Response For Long-Term NPSH

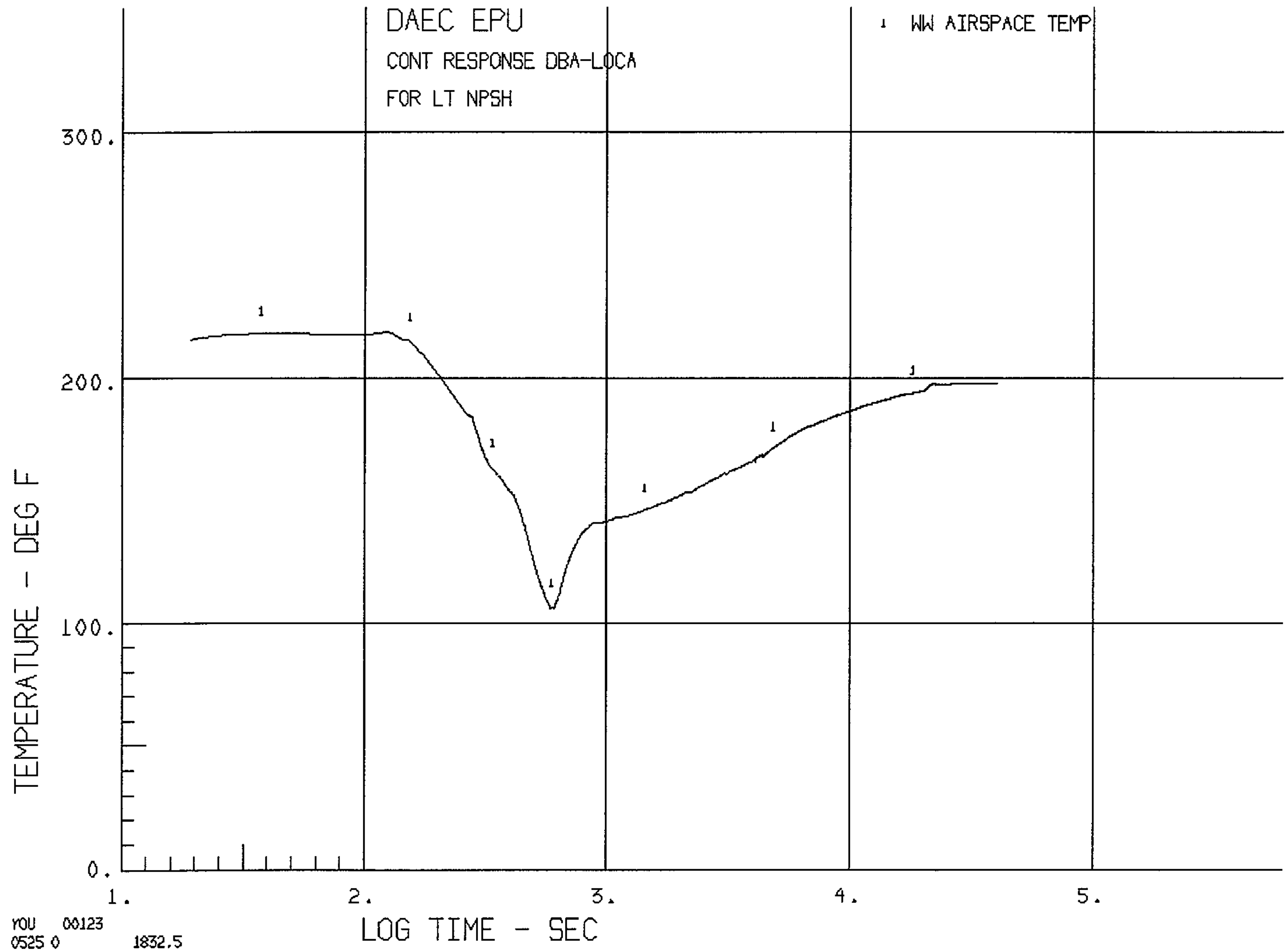

Figure 7: Wetwell Airspace Temperature Response For Long-Term NPSH

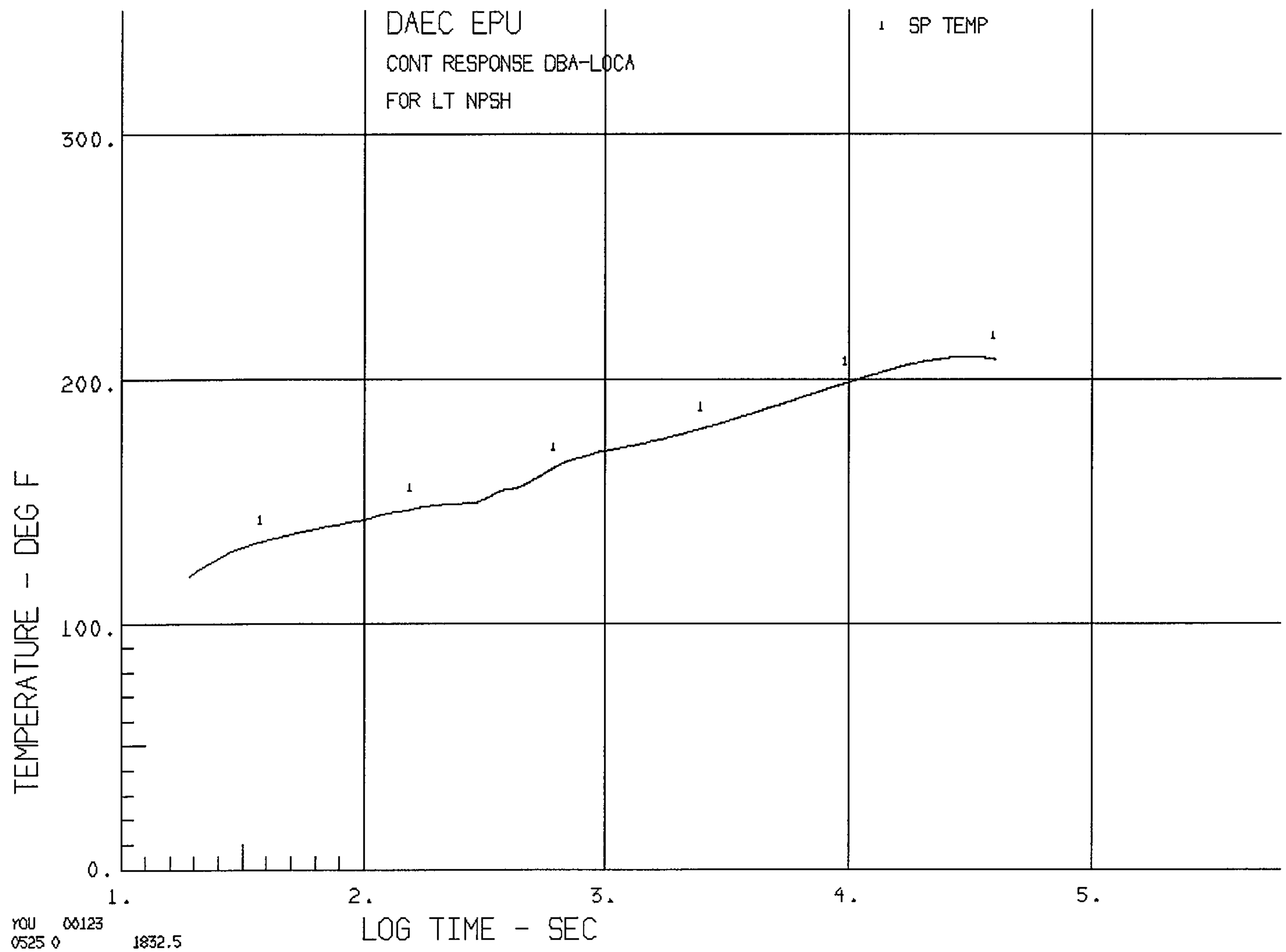

Figure 8: Suppression Pool Temperature Response For Long-Term NPSH

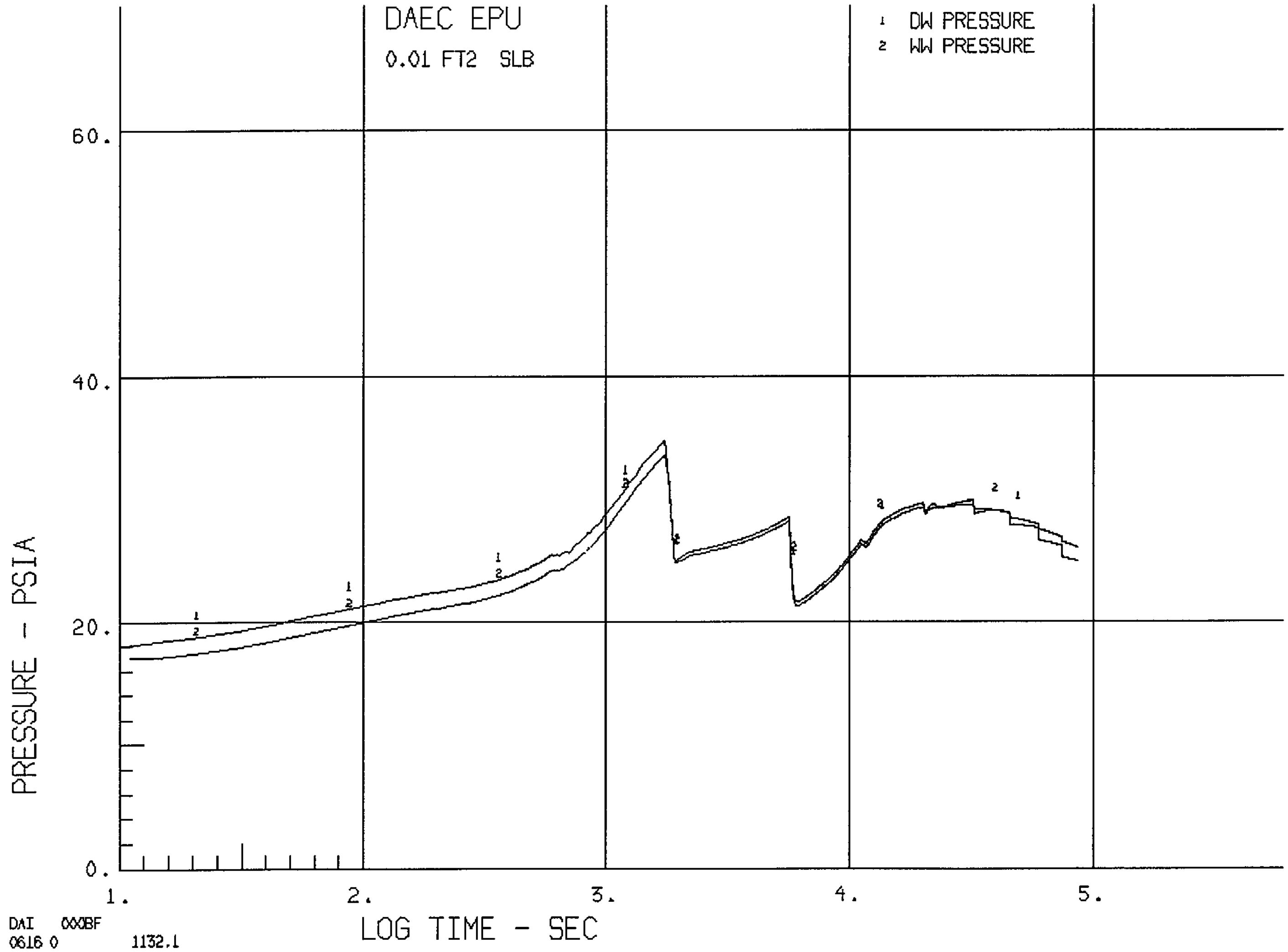

Figure 9: Containment Pressure Response For 0.01 ft² Steam Line Break

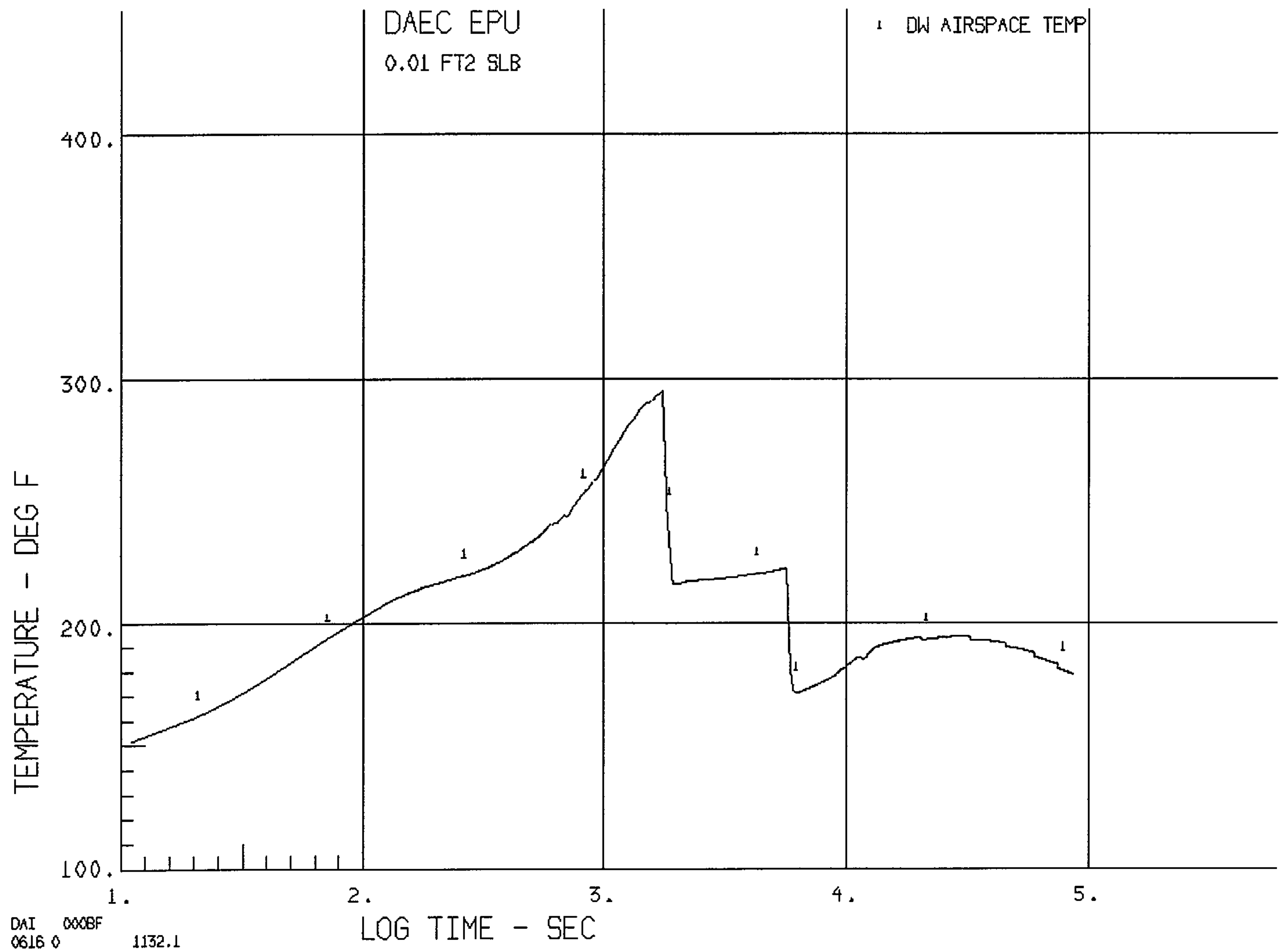

Figure 10: Drywell Airspace Temperature Response For 0.01 ft^2 Steam Line Break

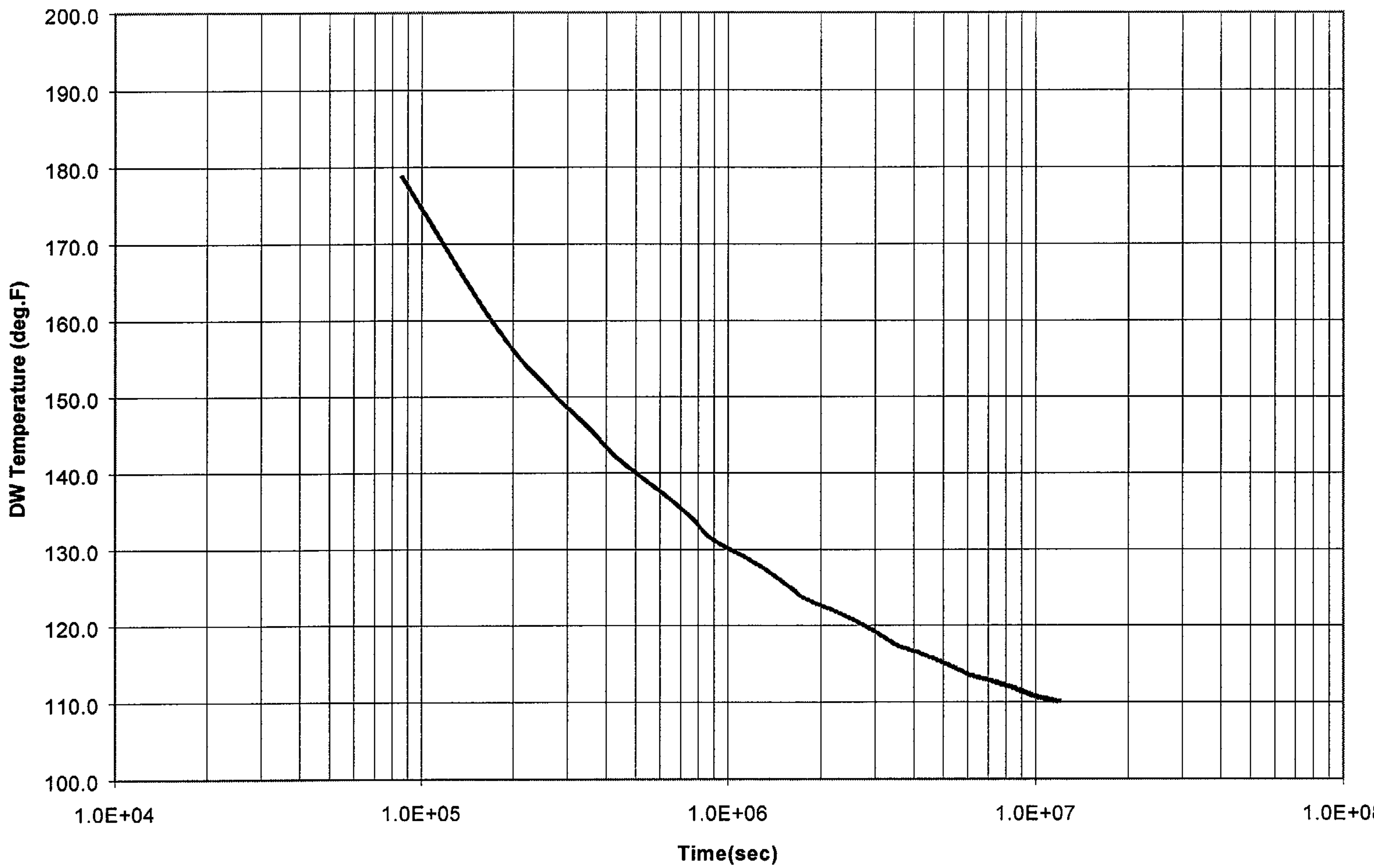

Figure 11: Drywell Airspace Temperature Response For 0.01 ft^2 Steam Line Break (Beyond 1 Day)

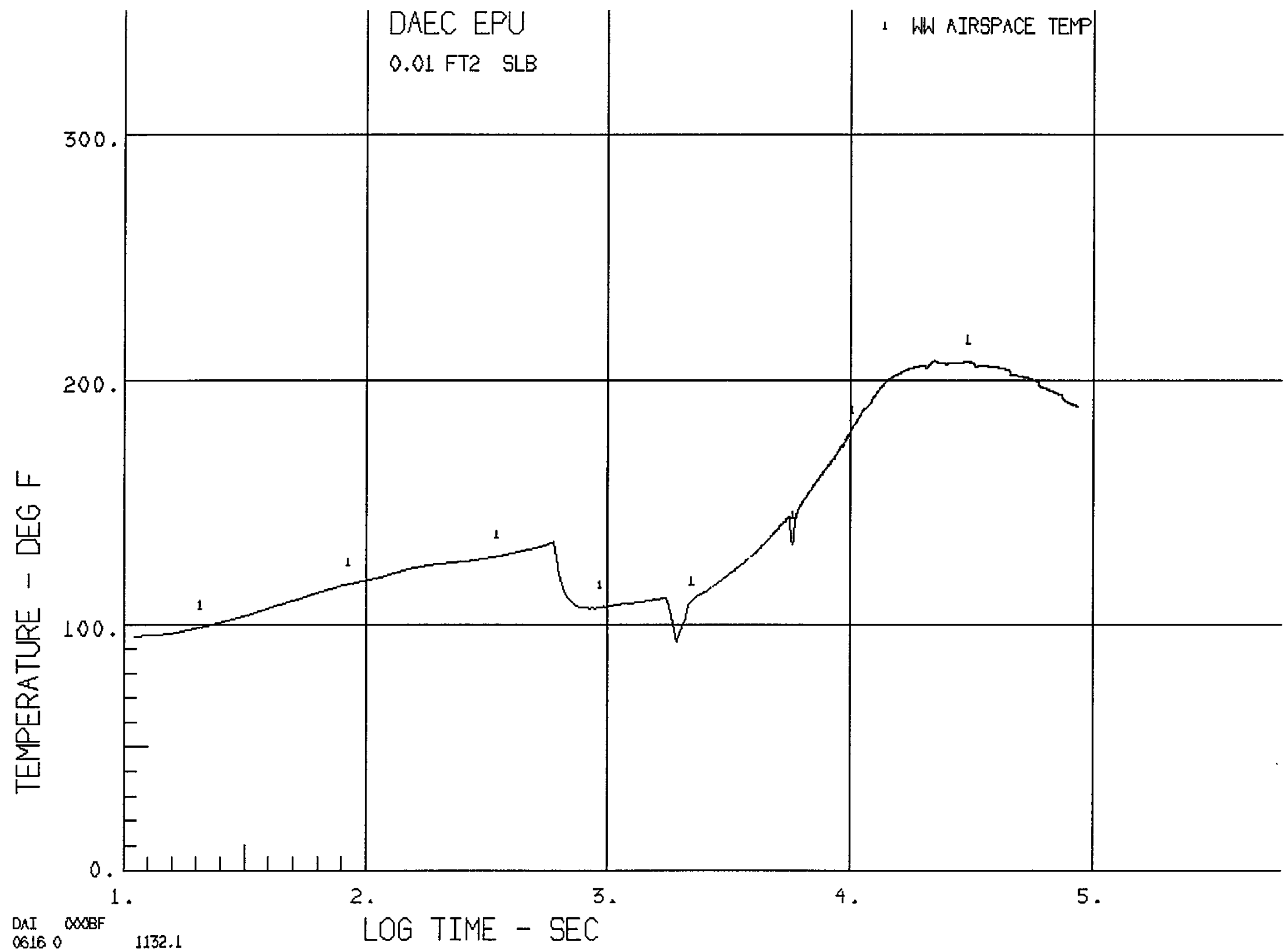

Figure 12: Wetwell Airspace Temperature Response For 0.01 ft^2 Steam Line Break

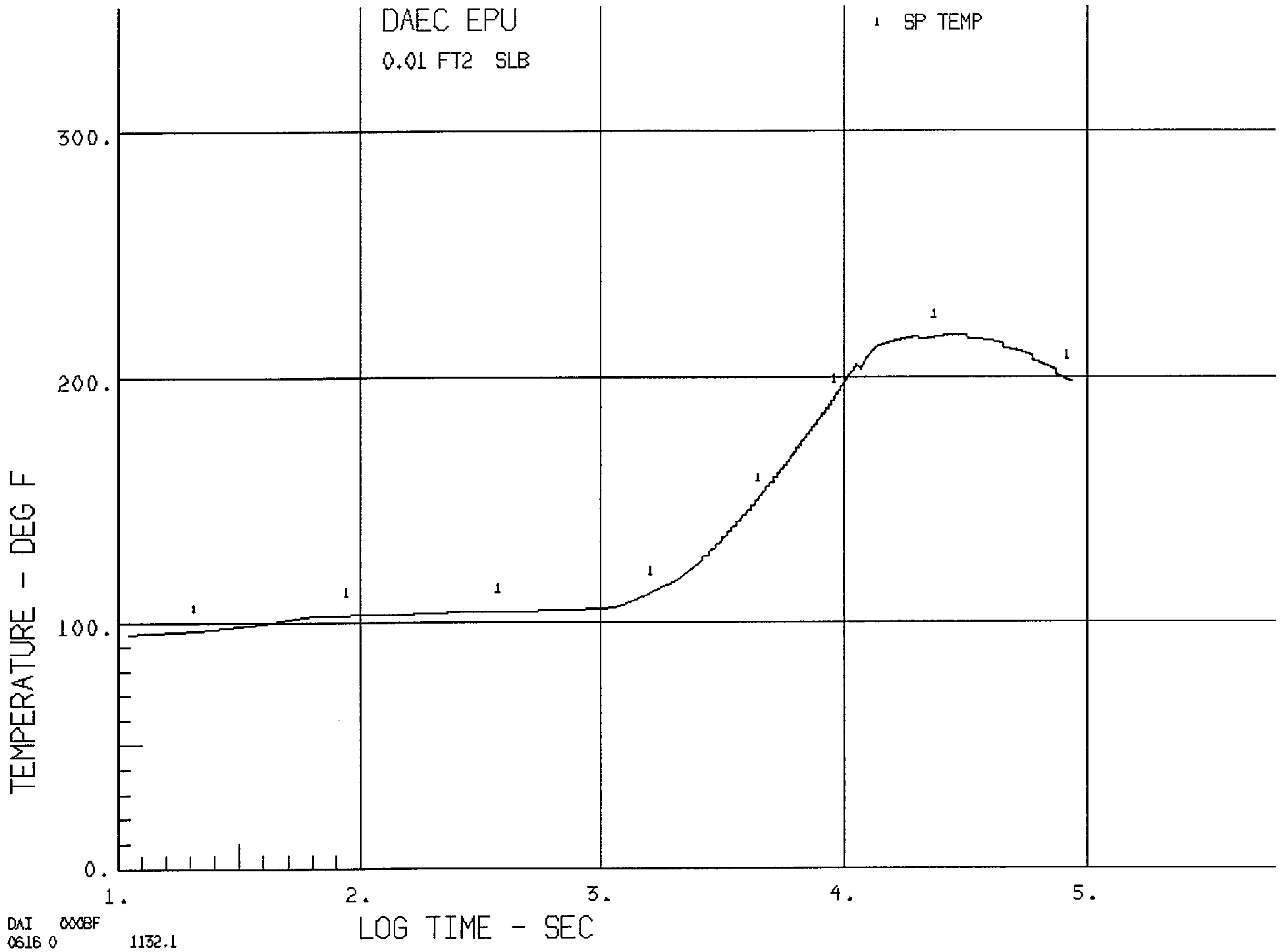

Figure 13: Suppression Pool Temperature Response For 0.01 ft² Steam Line Break