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MUSOLF, D. Northern States Power Co.
RECIP. NAME RECIPIENT AFFILIATION
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SUBJECT: Forwards response to 861021 request for addl info re reanalysis of postulated high energy line breaks. Operating procedure, defining steps necessary for Div I diesel generator to power other divs, prepared.

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Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 330-5500

December 4, 1986

Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Additional Information Related to High Energy Line Break Analysis

In a letter dated October 21, 1986 from Mr John A Zwolinski, Director, BWR Project Directorate No. 1, Division of BWR Licensing, USNRC, we were requested to provide additional information related to our recent reanalysis of postulated high energy line breaks for the Monticello Nuclear Generating Plant.

The information requested in Mr Zwolinski's letter is attached.

Please contact us if you have any questions related to the information we have provided or if additional information is required to assist in NRC Staff review of this issue.

David Musolf

David Musolf
Manager Nuclear Support Services

c: NRC Resident Inspector
NRR Project Manager
G Charnoff

Attachment



A001
1/1

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT
DECEMBER 4, 1986

QUESTION 1

As specified in the June 18, 1986, submittal, verify that all additional evaluations, modifications and procedural changes identified in the NUTECH report dated June 16, 1986 have been completed and implemented. Provide a case-by-case description of the completion status with respect to the items identified in Sections 5.2, 5.3.1.1, 5.3.1.3, 5.3.2.1, 5.3.2.3, 5.3.4.1, 5.3.6.1, 5.4.8, 5.4.9, 6.1, 6.2, 6.3, 6.4 and 6.5.

RESPONSE 1:

The referenced sections in the question can be divided into six separate identified physical and/or procedural changes. The six concerns are identified below along with a discussion of how each concern was resolved prior to restart of the Monticello Plant.

The six concerns are as follows:

(1) Response to Section 5.2 - Single active failure of the required Diesel Generator following postulated High Energy Line Breaks (HELBs).

Action Taken:

An operating procedure has been prepared which defines the steps necessary for the Division I diesel generator to power the other division. In this procedure, cutting of electrical cables is required. These cables were labeled and the cutters needed to carry out this procedure were placed on the inside of the panel that contains these cables. The generation of this procedure is a direct result of the HELB evaluation.

(2) Response to Sections 5.3.1.1, 5.3.4.1 and 6.2 - Identify the path to safe shutdown resulting from a Main Steamline break or High Pressure Coolant Injection System (HPCI) Steamline break in the Main Steam Chase, which affects the Reactor Core Isolation Cooling System (RCIC) Steamline (PS17-3"-ED) and the Emergency Service Water (ESW) Line (SW30B-3"-HF) coincident with a loss of power and single active failure of the opposite division I Diesel Generator (D/G).

Action Taken:

Modifications were made to the ESW Line (SW30B-3"-HF) so that cooling flow to the diesel generators would not be affected by a HELB in the Main Steam Chase (see Figure 1, page 4).

A test was performed to determine the effects of the loss of cooling water to the Division II Core Spray (CS) and Residual Heat Removal (RHR) pump motor oil coolers and to the RHR room coolers. The results confirmed that the CS and RHR pumps could be operated for approximately 2½ hours before the pump motor cooling and the room cooler would be required. The results allow adequate time for operator action to establish flow to the room cooler with the service water pumps using emergency D/G power source.

(3) Response to Sections 5.3.1.3, 5.3.2.1 and 6.1 - Identify the path to safe shutdown resulting from a Primary Steam Break on the Steam Bypass line in the Condenser Bay Area or Condensate Line Break, which could damage both divisions of the Emergency Service Water System.

Action Taken:

The safety-related Emergency Service Water (ESW) System consists of two divisions which supply cooling water to D/Gs, the Core Spray and RHR Motors and the HPCI and RHR Pump Room Ventilation Units. Cooling water is taken from the river by the ESW pumps in the intake structure. Flow to the D/Gs is through a 4" diameter line which branches in the intake structure. The remaining flow goes to the Emergency Core Cooling (ECCs) motors and ventilation units. Changes were made to the ESW in response to the HELB concerns. First, a normally closed manual valve was added to each line to the ECCS pump motors and ventilation units so that flow to the D/Gs would not be affected by any postulated HELB (see Figure 1, page 4). As indicated in the Action Taken section of concern 2 of this question, a test was performed which indicated there would be adequate time to establish flow to the ventilation units using an emergency D/G power source to operate the service water pumps. Also, an ESW valve was moved to the Torus area so any HELB affecting the ESW system could be isolated. With this arrangement, the postulated HELBs affecting both ESW lines can be mitigated.

(4) Response to Sections 5.3.2.3 and 6.3 - Loss of both essential Motor Control Centers (MCCs), 133 and 143, resulting from postulated HELBs on the Feedwater and Condensate Lines in the Feedwater Pump Area.

Action Taken:

For the postulated Feedwater and Condensate HELBs in the Feedwater Pump Area, an evaluation was done of the postulated break locations for jet and pipe whip reaction loads. Calculated loads used for jet impingement forces were made using the criteria of the Standard Review Plan (SRP) 3.6.2 page 3.6.2-7. The results were that the analyzed ceiling locations could withstand the 6" diameter Feedwater line breaks in the Feedwater Pump Area but not the jet impingement force of one 14" diameter Feedwater line break. A jet impingement shield capable of withstanding the calculated force from this 14" Feedwater line break was designed and installed. To eliminate problems with pipe whip in the Feedwater Pump Area, one pipe whip restraint was installed on the 16" Condensate line to protect the ceiling. With these modifications, no line break can adversely affect both MCCs 133 and 143.

(5) Response to Sections 5.3.6.1 and 6.5 - Postulated Reactor Water Cleanup (RWCU) HELBs, which damage the air systems to the inflatable seals on the Primary Containment Isolation System (PCIS) Valves AO-2386 and AO-2387.

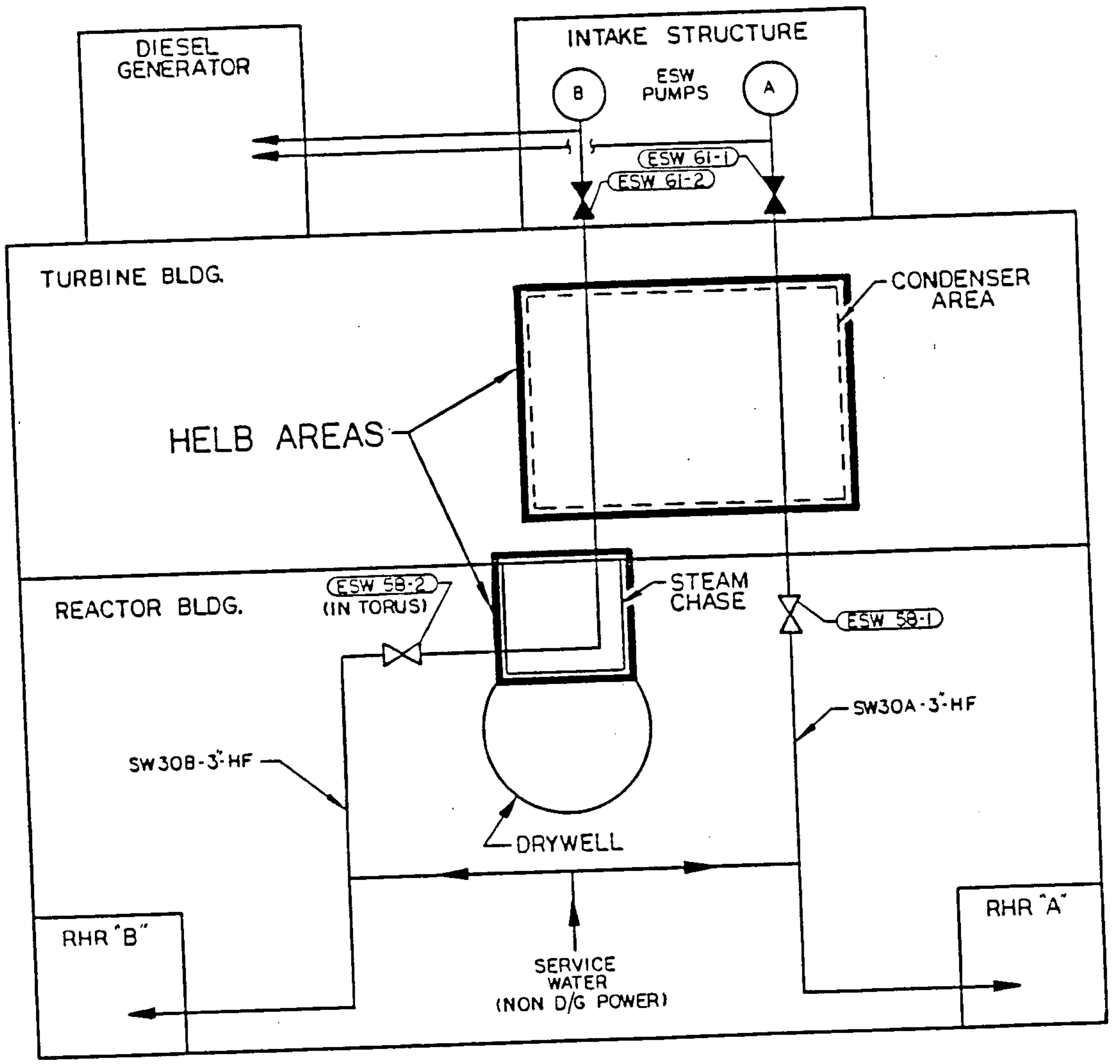
Action Taken:

The PCIS valves AO-2386 and AO-2387 are in-series, normally closed containment isolation valves on the Primary Containment Atmospheric Control System. Both valves require air to open and to maintain T-ring (seals) inflation with valve closed. These valves are opened only during the drywell purge and venting process. An ASME Section XI local leakage rate test was conducted on these valves at 42 psid without the seals inflated to identify leakage rates. The test showed that the Appendix J Technical Specification Total Primary Containment leakage rate acceptance criteria could be achieved even if these valves' T-rings were not inflated.

(6) Response to Sections 5.4.8, 5.4.9 and 6.4 - Primary Steamline Breaks in the Condenser Bay and the Steam Jet Air Ejector (SJAE) compartment and what effects do these breaks have on the essential switchgear on the 911'-0" elevation of the Turbine Building.

Action Taken:

The peak pressures and temperatures in both compartments from postulated HELBs were calculated. The results showed that the doors to these rooms could fail and the subject switchgear could be affected by the adverse environment. The doors to the SJAE compartment and the north wall entrances to the Condenser Bay have been structurally reinforced. The original concrete equipment hatch for the SJAE compartment was replaced with a steel plate hatch due to the calculated peak compartment pressure. This change allows for additional vent area from the compartment to equipment areas that do not contain safe shutdown equipment. In addition, the calculated peak pressures during the transient was found to be insufficient to cause damage to the knockout blocks on the north side of the Condenser Bay.




SIMPLIFIED PIPING AREA DRAWING

FIGURE 1

QUESTION 2

5.3.1.1, 5.3.1.2 and 5.3.1.3 of the report state that "Flooding is not a consideration since the high energy fluid is steam". On the basis of our analysis, we have concluded that sufficient steam may condense and water retention may represent a concern. Provide an analysis which indicates that sufficient steam will not condense on compartment walls such that water retention may be a problem.

RESPONSE 2:

An analysis was performed for each compartment to determine the amount of water which would collect at the bottom of each compartment. From the volume of condensate generated, flood height was determined. Mass flow rates out of breaks were postulated based upon the thermodynamic conditions of the steam in the Primary Steam Piping and the size of the breaks. Duration of flow from the breaks was based upon closure time for the Main Steam Isolation valves.

The following table identifies the postulated line break resulting in the maximum flood condition, the maximum flood height for each compartment, and any remarks.

| Break and Compartment | Maximum Flood Height | Remarks |
|---|---|---|
| Main Steam Line/ Main Steam Chase | 1 ft | Condensate does not leave the compartment as the door is 4 ft. above floor height. There is no safe shutdown equipment that would be affected by flooding in this compartment. |
| Main Steam Line/ Condenser Bay | negligible | Condenser Bay has various equipment pits 3 ft. lower than rest of area. Free volume of these areas is greater than volume of condensate generated. There is no safe shutdown equipment in the equipment pits. |
| 4" Primary Steamline/ SJAE Compartment | 8 in. | Maximum height represents total condensate that stays in SJAE compartment. The drain in the room was not considered. The compartment does not contain any safe shutdown equipment. |

The analysis assumed that all the condensate remained in the room and no credit for floor drains or vents was taken. The Condenser Bay and the SJAE Compartment are connected by a common pipe chase. Thus, a break in either area would cause steam to blow down to the other, but no credit was taken for the combined areas.

QUESTION 3

Section 5.3.2.1 of the report states that "Flooding is not a concern because of the size of the condenser area". A similar statement is made in Sections 5.3.4.1, 5.3.5.1 and 5.3.5.2. For each section, provide the maximum anticipated water depth and the depth required to affect adversely a safety-related component. In addition, Section 5.3.5.2 is relying on isolation to minimize the effects of flooding. For this section, identify the means by which the line will be isolated (e.g., automatic signal, operator action, etc.) and the length of time from the time the pipe breaks until the line is isolated. (Note: If the isolation is by operator action, the minimum acceptable isolation time is 20 minutes if the isolation can be performed from within the control room, and 30 minutes if the action must be taken outside of the control room.)

RESPONSE 3:

(a) Section 5.3.2.1 describes Condensate System Pipe Breaks in the Condenser Bay. An evaluation was performed assuming that the Condensate Pumps pumped the entire inventory available in the Condensate hotwell through the break in the line. The total volume of water pumped is approximately 80,000 gallons. Height of water in the bottom of the Condenser Bay would be less than the 911'-0" elevation because the water flowing out of the break would be contained in the L.P. Heater Drain Cooler Pit, the Condenser Pit and the Mechanical Vacuum Pump Pit via the floor drains. All of these areas have floor elevations lower than the main floor elevation (911'-0") of the Condenser Bay. No safe shutdown equipment in the Condenser Bay would be adversely affected by the flooding because all safe shutdown equipment is located well above the Condenser Bay main floor.

(b) Sections 5.3.4.1, 5.3.5.1 and 5.3.5.2 describe the results of the following:

5.3.4.1 - HPCI Steam Line Break in the Main Steam Chase.
5.3.5.1 - RCIC Steam Line Break in the Main Steam Chase.
5.3.5.2 - RCIC Steam Line Break in the Torus Area.

Flooding heights in these compartments for the specified HELBs, were calculated using mass flow rates, enthalpies, and steam thermodynamic temperatures and pressures used in the environmental analysis work done to determine peak pressures and temperatures.

Areas and volumes in the compartments were calculated based on plant arrangement drawings. No credit was taken for escape of steam out of the compartment. The results of the flooding analysis for each compartment area are as follows:

| Break and Compartment | Maximum Water Height | Remarks |
|---|---|---|
| HPCI Steamline/ Main Steam Chase | 1 ft | The floor is 4 feet below the area entrance. |
| RCIC Steamline/ Main Steam Chase | 1 inch | The floor is 4 feet below the area entrance. |
| RCIC Steamline/ Torus Compartment | .05 inches | Torus Compartment floor area is approx. 10,700 ft.$^2$. |

The HPCI and RCIC breaks were terminated based upon their break detection instrumentation sensing the rupture and the stroke time of Containment Isolation Valves. A break in either the HPCI or RCIC steamlines would cause the flow instrumentation or the area temperature monitors to close the isolation valves.

The Updated Safety Analysis Report states HPCI Containment Isolation Valves are required to have a closure time of less than or equal to 40 seconds, and the RCIC Containment Isolation Valves have a closure time of less than or equal to 30 seconds. The modeled HPCI event terminated 53.0 seconds after the break and the modeled RCIC event 43.0 seconds after the break.

QUESTION 4

Section 5.3.2.3 of the report discusses the effects of a HELB in the feedwater and condensate lines. The same HELB could fail Motor Control Centers (MCC). Specifically, it could fail MCC 133 and potentially affect MCC 143 on the elevation above. Indicate whether or not MCC 143 is the redundant motor control center to MCC 133.

RESPONSE 4:

Motor Control Centers 133 and 143 are redundant in that each supplies power to one of the two divisions of a system. As stated in Question 1, Response 4, structural modifications were done in the Feedwater Pump Area to prevent damage to MCC 133 from line breaks. Also as indicated in Question 1, Response 1, a procedure was prepared which defines the step necessary for the Division I D/G to power the other division.

QUESTION 5

5a) Provide the subcompartment environmental analysis for a main steam line break in the Main Steam Chase (II/2F) and for a reactor water cleanup line break in the RWCU compartment (II/3D). For each compartment, provide a list of all safety-related components and their environmental qualification.

5b) For each compartment and line break, provide the following information:

1) With respect to the pipe to be broken:

   (a) Type of fluid (water or steam);
   (b) Temperature;
   (c) Pressure;
   (d) Source of the fluid;
   (e) Flow rate (or assumed flow rate) versus time; and
   (f) Enthalpy versus time

2) With respect to the compartments being analyzed:

   (a) Number of compartment analyzed;
   (b) For each compartment:
      i. Initial temperature
      ii. Initial pressure
      iii. Initial humidity
      iv. Floor area, including floor space taken by equipment (square feet)
      v. Number of vents and vent areas (square feet) for each vent; and
      vi. Compartment wall height (feet; and
   (c) Simple compartment and interconnection diagram

3) All assumptions used, including but not limited to the:

   (a) Orifice coefficient;
   (b) Fluid expansion factor; and
   (c) Heat transfer coefficient for heat through the walls

4) Utility's analysis results:

   (a) Temperature versus time curve (peak temperature specified);
   (b) Pressure versus time curve (peak pressure specified); and
   (c) Humidity versus time curve (peak humidity specified)

RESPONSE 5:

(5a) A subcompartmental transient analysis has been performed for both compartments to determine the pressure and temperature transients for these compartments. The peak environmental conditions were calculated based upon a Main Steamline break in the Main Steam Chase and a RWCU Line break in the RWCU Area on the 962'-6" elevation of the Reactor Building. These peak conditions are:

| Compartment | Peak Pressure (PSIA) | Peak Temp. (°F) | Relative Humidity |
|---|---|---|---|
| Main Steam Chase | 21.7 | 198 | 100% |
| RWCU Area | 16.97 | 213 | 100% |

As stated in the June 16, 1986 Nutech report, NSP-30-102, the environmental conditions for these compartments are unchanged from those determined in the original high energy line break study. Also, no additional equipment required for safe shutdown was identified. The original study, in conjunction with detailed thermal-hydraulic analyses (described in the response to question 5b below), was used to prepare the response to IE Bulletin 79-01B, Environmental Qualification of Class 1E Equipment. This response was extensively reviewed by the NRC and its contractors and was the subject of Safety Evaluation Reports dated June 3, 1981, January 4, 1983 and December 13, 1984. Please let us know if additional information is required.

5b) The following information has been gathered relative to the breaks described in Part 5a of the response.

| Parameter | Main Steam Chase | RWCU Area |
|---|---|---|
| 1a) Type of Fluid | Steam | Water |
| b) Temperature | 540°F | 540°F |
| c) Pressure | 963 psia | 963 psia |
| d) Source of Fluid | RPV | RPV |
| e) Flow rate versus time | 3650 lbm/sec. (0-2 sec.)<br>9670 lbm/sec. (2-3.5 sec.)<br>8100 lbm/sec. (3.5-5.0 sec.)<br>3800 lbm/sec. (5.0-5.5 sec.) | 244 lbm/sec. (0-120 sec.) |
| f) Enthalpy versus time | 1194.3 BTU/lbm (0-2.0 sec.)<br>633.3 BTU/lbm (2.0-5.5 sec.) | 575.4 BTU/lbm (0-120 sec.) |
| 2a) Compartment Anaylzed | Main Steam Chase (#13 - Figure 2) | RWCU Area (#16 - Figure 2) |
| 2b.i) Initial Temperature | 140°F | 104°F |
| b.ii) Initial Pressure | 14.7 psia | 14.7 psia |
| b.iii) Initial rel. humidity | 50% | 50% |
| b.iv) Floor Area | 568 $ft.^2$ w/25% covered w/equip. | 1087 $ft.^2$ w/20% covered w/equip. |
| b.v) Vent Area | Table 1<br>Table 2 | Table 1<br>Table 2 |

| Parameter | | Main Steam Chase | RWCU Area |
|---|---|---|---|
| 2c) | Simple Compartment Interconnection Diagram | Figure 2 | Figure 2 |
| 3.) | All assumptions Used | | |
| a) | Heat transfer coefficient | 40 $BTU/hr\text{-}ft^2\text{-}F$ | 40 $BTU/hr\text{-}ft^2\text{-}F$ |
| b) | Uchida Heat transfer coefficient | 280 $BTU/hr\text{-}ft^2\text{-}F$ | |
| 4.) | Utility Analysis Results | | |
| a) | Temp. vs. time | Figure 3 | Figure 5 |
| b) | Pressure vs. time | Figure 4 | Figure 6 |
| c) | Humidity vs. time | 100% from beginning of event | 100% from beginning of event |

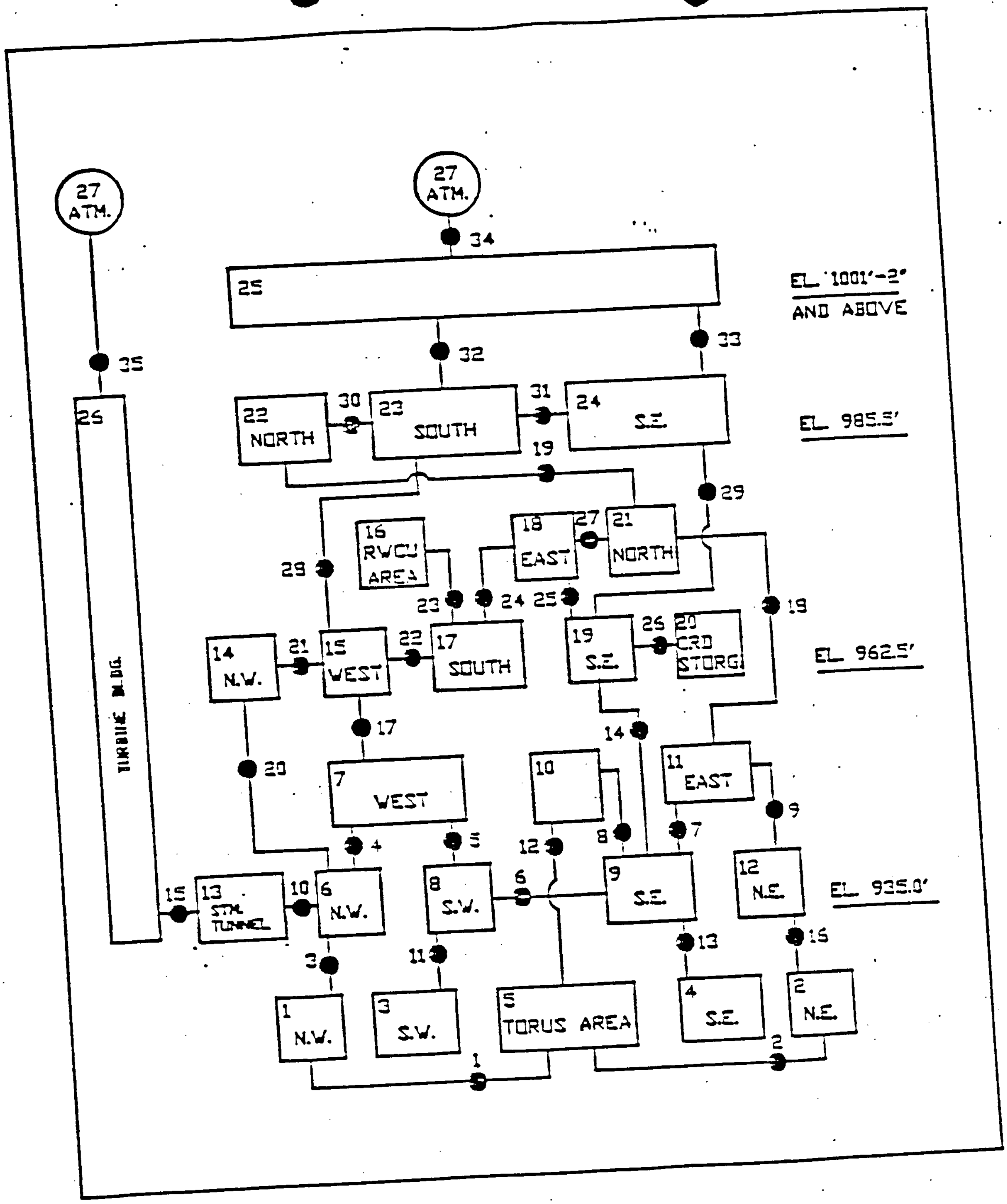



FIGURE 2

Computer Model No. 2 - Model of Reactor Building for Detailed Environmental Conditions on Selected Levels

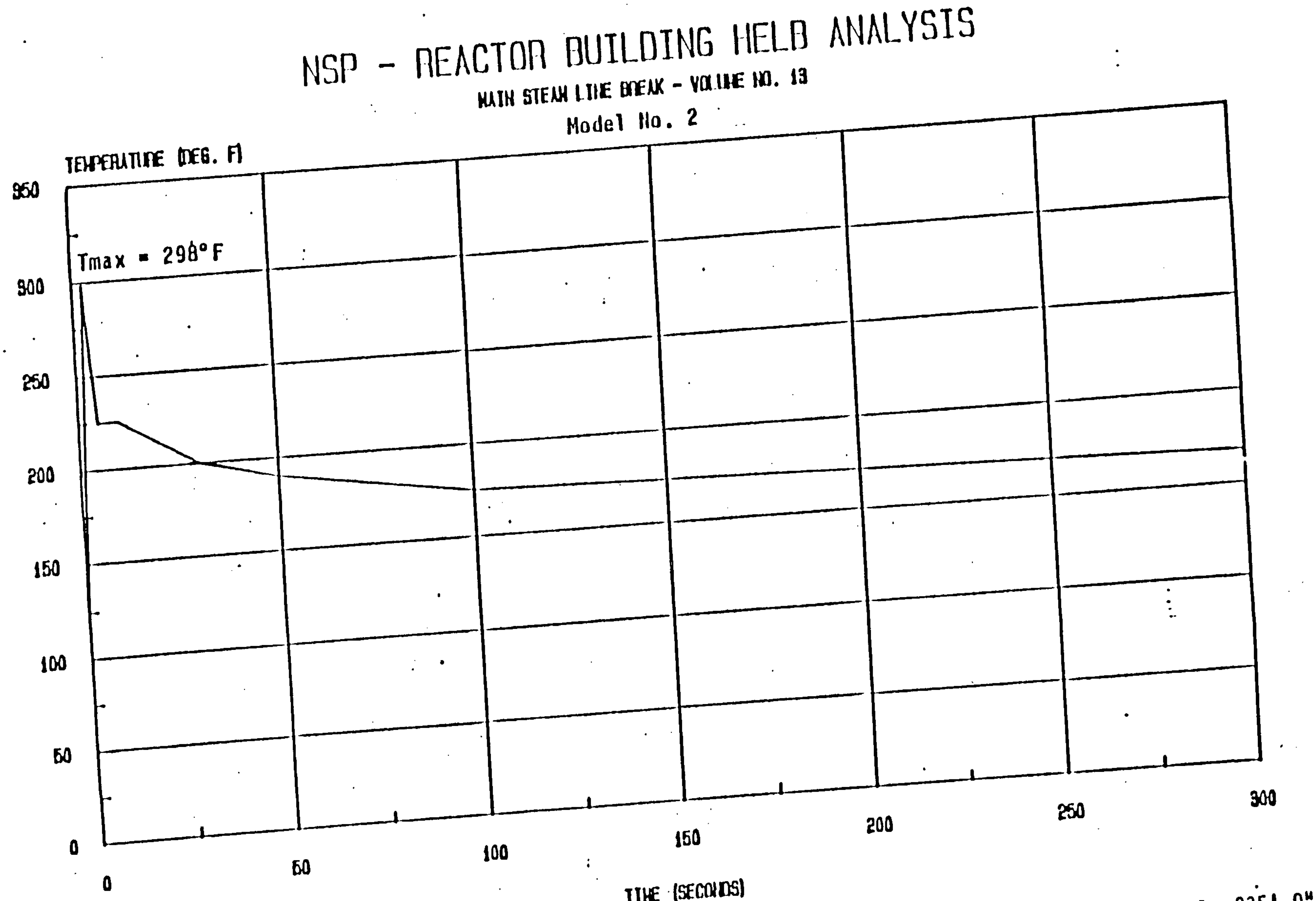



Figure 3 Temperature Time History in Steam Tunnel (Vol. 13), Reactor Building el. 935'-0"

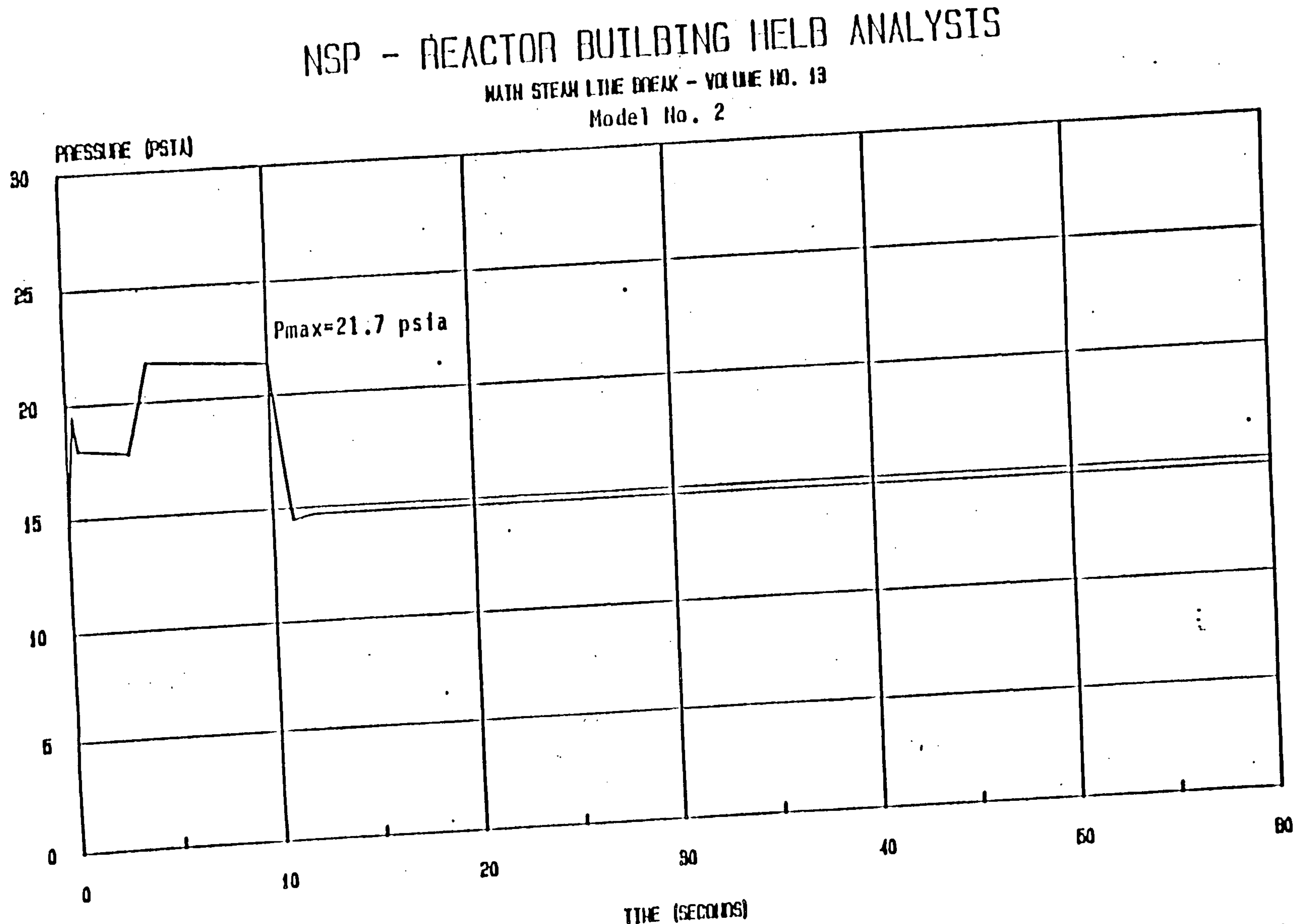



Figure 4 Pressure Time History in Steam Tunnel (Vol. 13), Reactor Building el. 935'-0"

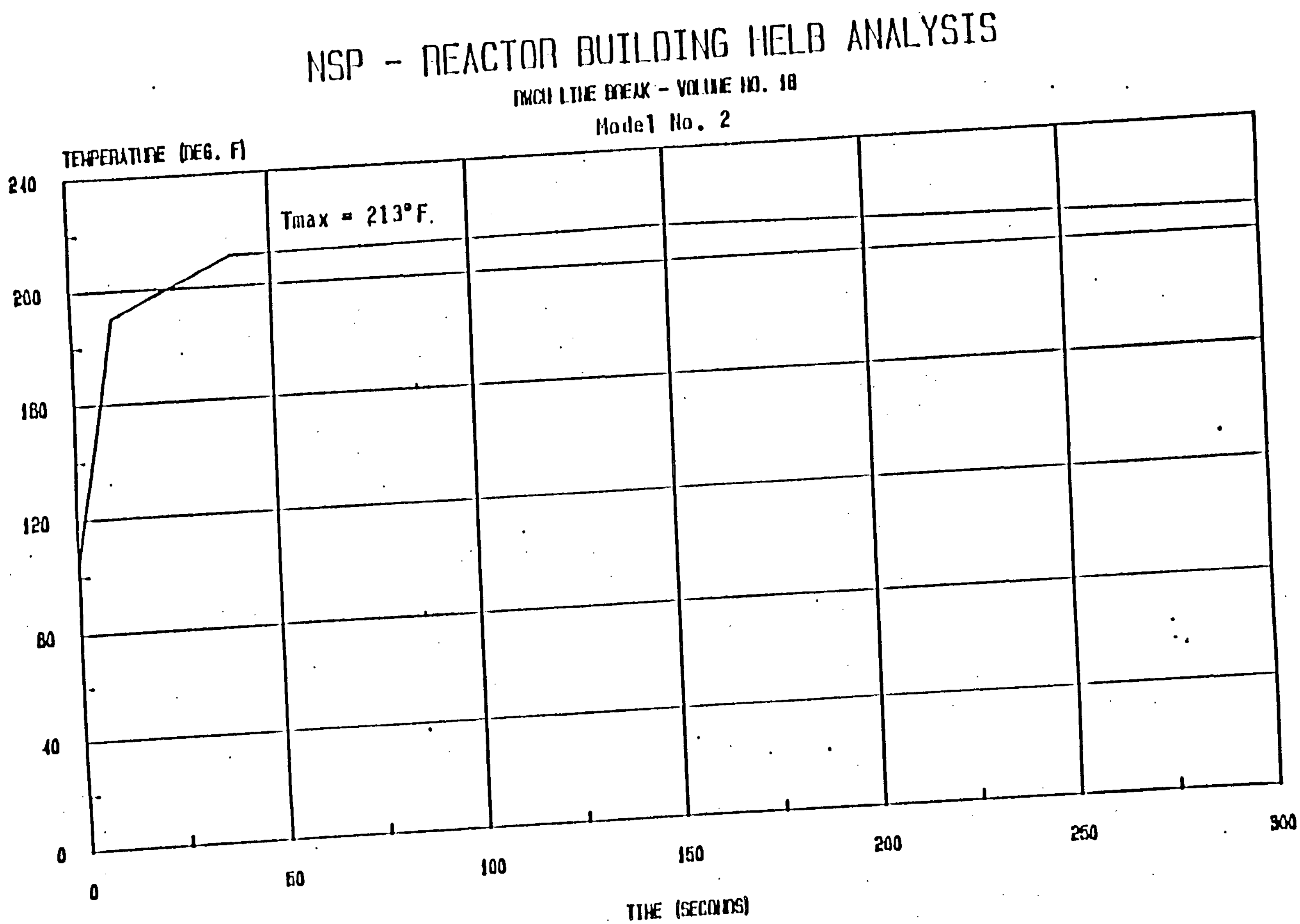



Figure 5 Temperature Time History in Volume 16, Reactor Building el. 962'-6"

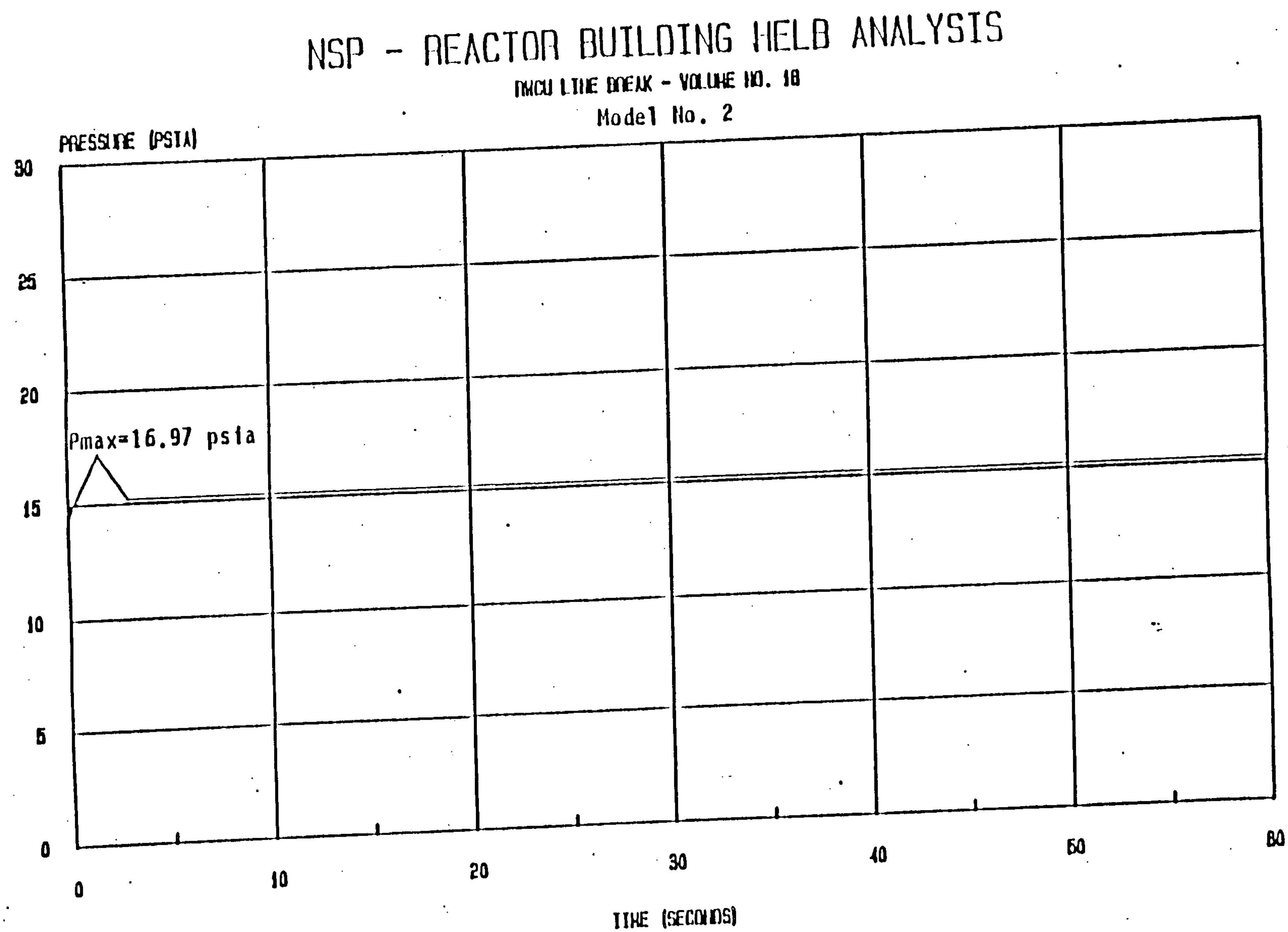



Figure 6 Pressure Time History in Volume 16, Reactor Building el. 962'-6"

## TABLE 1 - VOLUME PARAMETERS

| VOLUME # | VOLUME (ft3) | HEIGHT (ft) | FLOOR EL (ft) | HEAT SLAB AREA (ft2) | HEAT SLAB THICKNESS (ft) | REMARKS |
|---|---|---|---|---|---|---|
| 1 | 20341 | 38.75 | 896.25 | - | - | RB NW at 896' |
| 2 | 20341 | 38.75 | 896.25 | - | - | RB NE at 896' |
| 3 | 20962 | 38.75 | 896.25 | - | - | RB SW at 896' |
| 4 | 20962 | 38.75 | 896.25 | - | - | RB SE at 896' |
| 5 | 213856 | 38.75 | 896.25 | - | - | RB TORUS at 896' |
| 6 | 62762 | 27.5 | 935.0 | 2660 | 1.5 | RB NW at 935' |
| 7 | 34153 | 27.5 | 935.0 | 1050 | 1.5 | RB WEST at 935' |
| 8 | 56289 | 27.5 | 935.0 | 2806 | 1.5 | RB SW at 935' |
| 9 | 55694 | 27.5 | 935.0 | 2806 | 1.5 | RB SE at 935' |
| 10 | 19325 | 27.5 | 935.0 | - | - | RB Per Access at 935' |
| 11 | 47735 | 27.5 | 935.0 | 1600 | 1.5 | RB EAST at 935' |
| 12 | 21993 | 27.5 | 935.0 | 1875 | 1.5 | RB NE at 935' |
| 13 | 20317 | 27.6 | 931.0 | 4342 | 1.5 | STEAM TUNNEL |
| 14 | 21593 | 23.0 | 962.5 | 1670 | 1.0 | RB NW at 962' |
| 15 | 21136 | 23.0 | 962.5 | 1564 | 1.0 | RB WEST at 962' |
| 16 | 29300 | 23.0 | 962.5 | 2026 | 1.0 | RWCU AREA |
| 17 | 28450 | 23.0 | 962.5 | 1921 | 2.5 | RB SOUTH at 962' |
| 18 | 19752 | 23.0 | 962.5 | - | - | RB EAST at 962' |
| 19 | 34629 | 23.0 | 962.5 | 2331 | 1.0 | RB SE at 962' |
| 20 | 15299 | 23.0 | 962.5 | 987 | 1.0 | CRD STORAGE |
| 21 | 36002 | 23.0 | 962.5 | 2131 | 1.0 | RB NORTH at 962' |
| 22 | 48691 | 14.5 | 985.5 | 2465 | 1.0 | RB NORTH at 985' |
| 23 | 40527 | 15.7 | 985.5 | 1856 | 1.0 | RB SOUTH at 985' |
| 24 | 26796 | 15.7 | 985.5 | 1552 | 1.0 | RB SE at 985' |
| 25 | 782996 | 73.25 | 1001.2 | - | - | RB SOUTH at 1001' |
| 26 | 1443633 | 53.5 | 951.0 | - | - | TURBINE BLDG. |
| 27 | 1.E12 | 1000 | 890.0 | - | - | ATMOSPHERE |

# TABLE 2 - JUNCTION PARAMETERS

| JUNC. # | VOLUME FROM | VOLUME TO | AREA (ft2) | ELEV (ft) | LOSS COEFF. FWD. | LOSS COEFF. REV. | BLOWOUT PRESS. (PSID) | REMARKS |
|---|---|---|---|---|---|---|---|---|
| 1 | 1 | 5 | 21.0 | 899.75 | 1.5 | 1.5 | 2.33 | 3' X 7' Door |
| 2 | 2 | 5 | 21.0 | 899.75 | 1.5 | 1.5 | 2.33 | 3' X 7' Door |
| 3 | 1 | 6 | 36.0 | 935.0 | 1.5 | 1.5 | - | Stairway |
| 4 | 6 | 7 | 999.0 | 948.5 | 0.0 | 0.0 | - | Passageway |
| 5 | 7 | 8 | 925.0 | 947.5 | 0.0 | 0.0 | - | Passageway |
| 6 | 8 | 9 | 293.75 | 947.5 | 1.5 | 1.5 | - | Passageway |
| 7 | 9 | 11 | 962.5 | 947.5 | 0.0 | 0.0 | - | Passageway |
| 8 | 9 | 10 | 28.0 | 938.5 | 1.5 | 1.5 | - | Passageway |
| 9 | 11 | 12 | 42.0 | 938.5 | 1.5 | 1.5 | 0.05 | Locked Double Door |
| 10 | 13 | 6 | 21.0 | 942.5 | 1.5 | 1.5 | 2.33 | 3' X 7' Door |
| 11 | 3 | 8 | 36.0 | 935.0 | 1.5 | 1.5 | - | Stairway |
| 12 | 5 | 10 | 35.0 | 935.0 | 1.5 | 1.5 | - | Hatch |
| 13 | 4 | 9 | 36.0 | 935.0 | 1.5 | 1.5 | - | Starway |
| 14 | 9 | 19 | 297.5 | 962.5 | 1.5 | 1.5 | - | Hatch |
| 15 | 13 | 26 | 150.0 | 954.5 | 2.47 | 2.47 | 0.25 | Blowout Panel |
| 16 | 2 | 12 | 42.0 | 935.0 | 1.5 | 1.5 | - | Stairway |
| 17 | 7 | 15 | 84.0 | 962.5 | 1.5 | 1.5 | - | Stairway |
| 18 | 11 | 21 | 130.0 | 962.5 | 1.5 | 1.5 | - | Stairway |
| 19 | 21 | 22 | 130.0 | 985.5 | 1.5 | 1.5 | - | Stairway |
| 20 | 6 | 14 | 60.0 | 962.5 | 1.5 | 1.5 | - | Stairway |
| 21 | 14 | 15 | 68.3 | 967.5 | 1.5 | 1.5 | - | Passageway |
| 22 | 15 | 17 | 62.4 | 966.0 | 1.5 | 1.5 | - | Passageway |
| 23 | 16 | 17 | 21.0 | 966.0 | 1.5 | 1.5 | 2.33 | 3' X 7' Door |
| 24 | 17 | 18 | 441.0 | 973.0 | 1.5 | 1.5 | - | Passageway |
| 25 | 18 | 19 | 693.0 | 973.0 | 0.0 | 0.0 | - | Passageway |
| 26 | 19 | 20 | 42.0 | 966.0 | 1.5 | 1.5 | - | Normally Open Double Door |
| 27 | 18 | 21 | 577.5 | 973.0 | 0.0 | 0.0 | - | Passageway |
| 28 | 15 | 23 | 84.0 | 985.5 | 1.5 | 1.5 | - | Stairway |
| 29 | 19 | 24 | 297.5 | 985.5 | 1.5 | 1.5 | - | Hatch |
| 30 | 22 | 23 | 652.5 | 992.75 | 0.0 | 0.0 | - | Passageway |
| 31 | 23 | 24 | 304.5 | 992.75 | 1.5 | 1.5 | - | Passageway |
| 32 | 23 | 25 | 36.0 | 1001.2 | 1.5 | 1.5 | - | Stairway |
| 33 | 24 | 25 | 297.5 | 1001.2 | 1.5 | 1.5 | - | Hatch |
| 34 | 25 | 27 | 2160 | 1050.7 | 0. | 0. | 0.5 | RB Roof |
| 35 | 26 | 27 | 2698 | 1004.5 | 0. | 0. | 0.5 | Turbine Building Roof |