NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

RESPONSE TO THE EQUIPMENT QUALIFICATION SAFETY EVALUATION REPORT

Prepared For:

NORTHERN STATES POWER COMPNAY

Prepared by:

EDS Nuclear Inc.
Walnut Creek, California

8110060206 810928
PDR ADOCK 05000263
Q PDR

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1.0 INTRODUCTION

This report has been prepared in response to the Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) [1], concerning environmental qualification of safety related equipment for the Monticello Nuclear Generating Plant. This report addresses all concerns identified by the NRC in the Monticello SER.

The Equipment Qualification Branch of the Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission (NRC), has required all licensees of operating reactors to submit a re-evaluation of the qualification of safety related electrical equipment which may be exposed to a harsh environment. This requirement was implemented primarily by the issuance, on January 14,1980, of IE Bulletin No 79-01B with subsequent clarifying supplements in February, September, and October, 1980.

The Bulletin required that a master list of safety related systems and equipment be generated, all accident service conditions be defined, and the equipment be evaluated in accordance with guidelines in the Bulletin.

Northern States Power has provided three responses to the Bulletin in May 1980, August 1980 and November 1980. The May 27, 1980 response included identification and evaluation of safety related electrical equipment. In particular, the environments for equipment outside containment were identified and a general plan was developed to complete the qualification of equipment.

The August 27, 1980 response incorporated newly acquired qualification data. Additionally the normal and accident environments for equipment both inside and outside containment were finalized. The November 1, 1980 response summarized additional evaluations that had been performed. This response completely superceded all previous responses. Also, plans were described, which had been developed to assure the qualification of equipment identified as having "outstanding items."

On June 3, 1981, the NRC issued the Monticello Nuclear Generating Plant Safety Evaluation Report (SER), which summarized their assessment of the November 1, 1980 response and also requested additional information. A summary of all Monticello Nuclear Generating Plant environmental qualification activities is provided in this report.

This response incorporates newly acquired equipment qualification data. All remaining outstanding items have also been identified and plans developed for their resolution. Plans are described to establish necessary maintenance and surveillance programs to identifiy age degraded equipment. Also, justification for continued operation of the Monticello Plant is provided.

All work concerning the evaluation and response to evaluation deficiencies of environmental qualification have been fully documented in accordance with the procedures and requirements of the EDS Quality Assurance Program, so as to provide an auditable package. This program complies with 10CFR50, Appendix B, ANSI N45.2 and applicable daughter standards.

2.0 SUMMARY

This document summarizes the evaluation of environmental qualification of safety related electrical equipment performed in response to IE Bulletin No 79-01B. It provides documentation of the Environmental Qualification Program that is being undertaken by Northern States Power Company. The program ensures that all safety related equipment is capable of performing its safety related function during postulated accident conditions.

The present status of the evaluation is as follows:

- All safety related plant systems have been identified.
- The safety related electrical equipment in these systems has been identified.
- Equipment qualification service conditions have been developed for all areas of the plant.
- Information sources have been contacted in order to obtain qualification documentation.
- Type-test reports have been received for most safety related components. This data has been evaluated for compliance with the required qualification standards.
- A summary of the current evaluation status of all equipment for which qualification documentation has been obtained is presented.
- Responses to all deficiencies identified in the SER are presented.
- Corrective action plans to resolve any outstanding qualification items are presented. These plans are currently being implemented by Northern States Power.

This report is submitted as a preliminary response to the NRC Safety Evaluation Report concerning environmental qualifications to safety related electrical equipment at Monticello.

3.0 GENERAL CONCERNS

Section 3.0 of the Safety Evaluation Report [1] identified various general concerns. The following present our responses to these concerns.

3.1 Safety Related Equipment List

The NRC staff's review of the Safety Related Equipment List determined that the equipment list provided by Northern States Power was acceptable with one exception. The exception was that a complete list of all display instrumentation mentioned in the LOCA and HELB Emergency Procedures was not provided. Display instrumentation is discussed in Section 3.1.1.

Subsequent to the previous IE Bulletin 79-01B submittal for Monticello [2], the Safety Related Equipment List was reevaluated. The review determined that some components could be deleted from the list of equipment required to function in harsh environments. Deleted equipment is discussed in Section 3.1.2.

3.1.1 Display Instrumentation

Display instrumentation mentioned in the Monticello LOCA and HELB emergency procedures and subject to a harsh environment was included in our November 1, 1980 submittal (2). Each item was listed with its appropriate system. For clarity, these items are listed listed in Table 3-1 by system, plant ID number, and function. For consistency, the component evaluation worksheets for these instruments will be included in their appropriate system.

3.1.2 Deleted Equipment

A review of the safety related equipment list for Monticello indicated that various components in three systems could be deleted from the list of equipment requiring qualification to harsh environmental service conditions. It was determined that various components in the following systems could be deleted.

1. Reactor Core Isolation Cooling System (RCIC)

2. Containment System

3. Nuclear Boiler Vessel Instrumentation System

Equipment deleted from these systems is discussed below.

3.1.2.1 Reactor Core Isolation Cooling System

In our previous submittal (2) the limiting accident in the RCIC room was assumed to be a RCIC steamline break. A review of FSAR Section 7.2.3.2 determined that an RCIC steamline break would disable the RCIC system due to initiation of the RCIC steamline isolation by the RCIC system temperature sensors in either the RCIC room or the torus compartment. The other HELB that could possibly affect the RCIC room is a HPCI steamline break in the torus compartment. However, this break will also disable the RCIC system due to steamline isolation by the RCIC system temperature sensors in the torus area. Therefore, qualification of the equipment listed in Table 3-2 for the environment resulting from a HELB is not required.

Also, in the FSAR analysis, the equipment in Table 3-2 is not assumed to operate to mitigate the consequences of a LOCA. Therefore, qualification to LOCA induced environmental conditions such as radiation is not required.

The previous discussion indicates that the equipment listed in Table 3-2 is not required to be qualified to harsh environmental conditions. This equipment is now considered to be located in a mild environment and will be included in the mild environment equipment qualification evaluation.

Level transmitters LT 1358 and LT 1359 provide indication of condensate storage tank level. Operator use of these non-safety related instruments has been reviewed. It was determined by Northern States Power that post-accident failure would not mislead the operator. Therefore, qualification of these instruments is not required. This position is consistant with the requirements of Reg. Guide 1.97, Revision 2 [9], in which these instruments are identified as Category 3.

3.1.2.2 Containment System

Level transmitter LT 2996 provides indication of Torus water level. As dictated by TMI lessons learned requirements, qualified wide range Torus level transmitters (LT 7338A, B) have been installed. Because these new instruments provide the same indication function as LT 2996, they will supercede this instrument on the safety related equipment master list. Therefore, qualification of LT 2996 is not required and it has been deleted from the response.

3.1.2.3 Nuclear Boiler Vessel Instrumentation

Level transmitters LT 6-52A,B provide indication of reactor water level. Operator use of the information provided by these level transmitters has been reevaluated. It was determined that the same information provided by LT 6-52A,B is provided by redundant level transmitters LITS 2-3-59A,B. Therefore, qualification of LT 6-52A,B is not required and they have been deleted from the response.

TABLE 3-1 DISPLAY INSTRUMENTATION
SUBJECT TO A HARSH ENVIRONMENT

SYSTEM	PLANT ID	INDICATION FUNCTION
NBVI	LT 2-3-61	Reactor Vessel Level
NBVI	PT 6-53A,B	Reactor Vessel Pressure
RHR	FT 10-109A,B	LPCI Flow
RHR	FT 10-111A,B	Containment Cooling Flow
Core Spray	FT 14-40A,B	Core Spray Loop Flow
HPCI	FT 23-82	HPCI Pump Flow
Containment	PT 2994A,B	Drywell, Torus Narrow Range Pressure
Containment	PT 7348	Drywell Wide Range Pressure
HPCI	LITS 2-3-59A,B	High Reactor Level HPCI Trip
RHR	LITS 2-3-73A,B	2/3 Core Flood Cont Spray Interlock
RHR	DPT 10-91A,B	RHR Heat Exchanger DP Control

TABLE 3-2

RCIC SYSTEM EQUIPMENT DELETED
DUE TO LOCATION

PLANT ID	MANUFACTURER/MODEL NUMBER
MO-2078	Limitorque SMB-00
MO-2106	Limitorque SMB-00
PS 13-72 A, B	Barksdale D2H-M150 SS
P210	GE 5CD10C10A
PT 13-60	General Electric 551
PT 13-65	General Electric 551
PT 13-68	General Electric 551
PT 13-70	General Electric 551
MO 2100	Rotork 12A/EC
MO 2101	Rotork 12A/EC
MO 2102	Rotork 12A/EC
SV 2104	ASCO T-HT-83212
FS 13-57	Barton 289
FT 13-58	General Electric 553
PS 13-67	Mercoid DAW-443-4132-R26E
--	Woodward Governor Turbine Controls

3.2 Service Conditions Inside Containment

The NRC staff identified four service conditions inside containment that were considered deficient or required further justification. These service conditions are as follows:

1. Pressure Conditions
2. Temperature Conditions
3. Chemical Spray
4. Radiation Conditions

3.2.1 Pressure Conditions

The containment pressure profile in the FSAR was confirmed in the Mark 1 Containment Program plant unique load definition report for Monticello [3]. Calculation techniques used in this analysis are described in the Mark 1 Containment Program Load Definition Report [4] and are equivalent to those used in GESSAR Appendix 3B. These techniques were accepted by the NRC in the Safety Evaluation Report, Mark 1 Containment Long-Term Program, NUREG 0661 [5].

For the Monticello plant, the plant unique Mark 1 containment load analysis of drywell pressure predicts a peak of 41.1 psig, which is consistent with the value of 41.5 psig used for equipment qualification. Additionally, the peak calculated drywell temperature during the LOCA is 282°F, which compares favorably with the 289°F reported to the NRC.

The previous discussion justifies our use of FSAR Figure 5-2-14, Containment Pressure Response, for equipment qualification. The pressure profile used for equipment qualification inside containment is shown in Figure B.1, of Appendix B.

3.2.2 Temperature Conditions

The NRC staff concluded that the minimum temperature profile for equipment qualification should include a margin to account for analytical uncertainties in the calculated temperature profiles for postulated accidents. A margin of 20°F above steam saturation temperature is considered to be appropriate for either a postulated LOCA or MSLB, whichever is controlling as to potential adverse environmental effects on equipment.

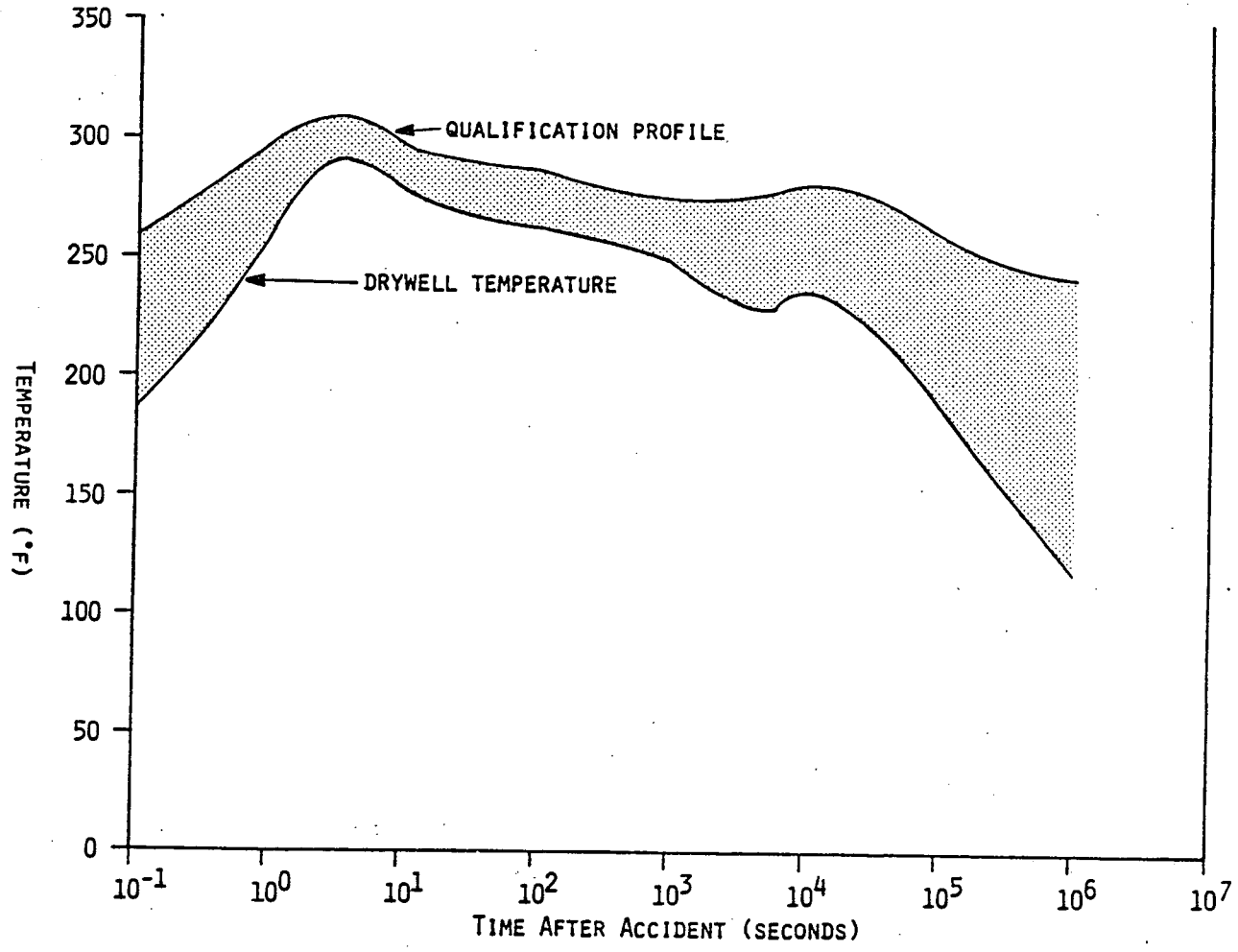
The limiting accident inside containment analyzed in the FSAR is a double ended recirculation line break. The pressure response for this accident is shown in FSAR 5-2-14 (6). The calculated temperature response for this accident is shown in FSAR Figure 5-2-15 (6) and was used for equipment qualification. To satisfy the new NRC temperature margin requirement, a new temperature profile will be used. This new profile is the steam saturation temperature plus 20°F corresponding to the pressure curve shown in FSAR Figure 5-2-14 (6). Figure 3-1 illustrates the temperature margin resulting from the new temperature profile. The temperature profile used for equipment qualification inside containment is shown in Figure B.1, of Appendix B.

3.2.3 Chemical Spray

In our previous submittal [2], the containment spray system was not considered safety-related and credit for its use was not taken in the analysis. The NRC staff concluded that because the system is available and could be used, any equipment upon which the spray impinges must be qualified for the spray parameter.

To satisfy the chemical spray requirement, all components located inside containment were reevaluated considering a spray solution of demineralized water. All equipment potentially impinged by the spray were determined to be qualified for this condition

FIGURE 3-1


350
300
250
200
150
100
50
0
TEMPERATURE (°F)
QUALIFICATION PROFILE
DRYWELL TEMPERATURE
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}
TIME AFTER ACCIDENT (SECONDS)

3.2.4 Radiation Conditions

During the environmental equipment qualification meeting in Bethesda, July 7 through 10, 1981, the NRC Staff presented their revised "screening value" for radiation service conditions inside containment. The Staff's screening value was 4.0×10^7 rads (gamma plus beta) which included a 2.0×10^7 rad normal dose and a 2.0×10^7 rad accident dose.

The maximum radiation condition used in our previous submittal (2) for equipment qualification inside containment was 3.3×10^7 rads gamma. This is an accident dose and does not include normal radiation

FSAR Section 5-2.2.4.3 indicates the maximum design normal radiation dose rate inside containment is 25 R/Hr neutron plus gamma continuous for the plant design life. Use of this dose rate continuous for the 40 year plant design life results in a total integrated normal dose of 8.77×10^6 rads gamma. This normal dose when added to the 3.3×10^7 Rad accident dose equals 4.18×10^7 Rads. This total radiation dose exceeds the NRC screening value.

3.2.5 Submergence

Our previous submittal [2] indentified a flood elevation of 922 ft. inside containment. No equipment inside containment was found to be subject to submergence. The NRC staff found this acceptable.

3.3 Service Conditions Outside Containment

3.3.1 Thermal Hydraulic Conditions

Temperature, pressure and relative humidity conditions as a result of a High Energy Line Break (HELB) outside containment were previously submitted for the following plant areas:

1. Steam Chase
2. Turbine Building
3. Torus Compartment
4. HPCI Room
5. RCIC Room
6. RHR Rooms
7. RWCU Pump Room
8. RWCU Heat Exchanger Room
9. Open Space at Elevations 935 ft. and 962 ft.

The environmental service conditions provided for these areas were deemed acceptable by the NRC staff.

The thermal hydraulic service conditions have been reevaluated for many areas of the Monticello Plant. The open spaces at elevations 935 ft. and 962 ft. 6 in. were modeled in more detail. This effort was performed in order to more accurately define the required accident service conditions for future equipment installation, relocation and replacement.

The results of the refined thermal hydraulic analyses are presented in Reference 7. Plant layout drawings depicting the layout of thermal-hydraulic control volumes for the environmental analyses are shown in Figures B.18 through B.21, of Appendix B.

In addition, the environmental conditions in the HPCI room due to a RCIC steamline break in the torus compartment were analyzed. The results of this analysis are presented in reference 8.

The temperature and pressure profiles used for equipment qualification outside containment are shown in Figure B.2 through B.17, of Appendix B. The post accident relative humidity condition for each of these locations is assumed to be 100% as indicated on the Component Evaluation Worksheet.

3.3.2 Submergence

This parameter will be discussed later.

3.3.3 Radiation

The radiation conditions for equipment located outside containment were deemed acceptable by the NRC staff. Therefore, no changes have been made in the radiation levels for equipment outside containment.

3.4 Aging

When Northern States Power Company received their operating license for Monticello, aging was not required for equipment qualification. Therefore, aging was not addressed in the original equipment qualification efforts.

The DOR guidelines presented in the IE Bulletin No. 79-01B [10] required that component aging be addressed in the evaluation of equipment qualification. While the DOR guidelines do not require a qualified life be established for all safety related equipment, the following actions are required:

1. Make a detailed comparison of existing equipment and the materials identified in appendix C of the DOR guidelines. The first supplement to IEB 79-01B requires licencees to utilize the table in Appendix C and identify any additional materials as the result of their effort.

2. Establish an ongoing program to review surveillance and maintenance records to identify potential age related degradations.

3. Establish component maintenance and replacement schedules which include considerations of aging characteristics of the installed components.

At this time, efforts required by Action Item (1) have been completed. An extensive amount of manpower was spent in these efforts. Although primary emphasis was placed on evaluating the affects of thermal aging, other significant aging mechanisms were addressed. The following steps were involved in performing the aging evaluations:

1. Identify the potentially age susceptible subcomponents of an equipment item. This was done by listing all non-metallic parts. It was assumed that metallic components were not susceptible to significant age related degradation.

2. Identify the specific non-metallic component materials. In most cases, a bill of materials provided by the equipment vendor defined these materials. However, many components were of a vintage that the vendor could not accurately identify the actual materials of construction. In these cases, typical materials used in similar applications were listed and the material most susceptible to aging was assumed.

3. Identify aging properties of the previously identified materials. This involved an extensive literature search. Data sources included:

   a. Appendix C of the DOR Guidelines
   b. Material Manufacturer Information
   c. EPRI Report No. NP-1558
   d. U.L. Temperature Ratings
   e. Test Lab Data
   f. Other Sources

   The data was reviewed to determine whether it applied to the material property of concern, such as retention of elongation for a cyclic diaphragm.

4. Determine susceptibility to age related degradation. The literature search indicated that certain materials, such as thermosetting plastics and silicons, were not susceptible to significant thermal degradation. This determination was made only when adequate basis was present in the literature. It should be noted that this conclusion was made for very few materials.

   The Arrhenius Model was used to evaluate the affects of thermal aging for materials that were determined to be potentially susceptible to significant thermal degradation. The Arrhenius evaluations extrapolated test conditions and published materials information to determine a qualified life for a material in a certain application. Conservatism was included in all Arrehnius calculations in the following manner:

   a. The lowest apparent activation energy for a material was used unless a specific justification for a higher value was provided.

   b. A conservatively high normal operating temperature was used to calculate a qualified life.

   c. A conservative activation energy of 0.5 eV was assumed for materials where adequate aging data was unavailable.

   d. A conservatively high material temperature was assumed when evaluting equipments that generate heat such as solenoids and motors.

5. Evalute effect of material degradation. When a qualified life for a material was less than required, the failure of this material was evaluated. If it was determined that failure of this material or subcomponent would not affect the functionability of the component, this material was no longer considered the limiting material and the next most limiting material was used to establish the qualified life. An example of this situation would be a cover gasket on a component not subject to harsh temperature, pressure

or relative humidity conditions. In this situation, failure of the cover gasket due to age related degradation would not have a deleterious affect on the component's functionability.

Also, for equipments exposed to no or small accident temperature transients, the normal age related degradation will not be significantly accelerated during post accident conditions. Therefore, catastrophic failure due to thermal aging is very unlikely.

4.0 COMPONENT DEFICIENCIES

Section 4.0 of the Safety Evaluation Report [1] separated the safety related equipment into three categories. These three categories are discussed below:

4.1 Equipment Requiring Immediate Action

No safety related equipment at Monticello was included in this category.

4.2 Equipment Requiring Additional Information

Appendix B of the Safety Evaluation Report [1] lists equipment requiring additional information and/or corrective action. Ninety six percent of the safety related equipment in the previous submittal [2] was included in this category. Various deficiencies were identified for these components.

Information obtained during the Environmental Equipment Qualification meeting in Bethesda, Maryland, through discussions with the NRC and by reviewing the identified deficiencies against the Component Evaluation Worksheets indicates that only additional information is required. We have attempted to identify what additional information is necessary by a component by component comparison of the Component Evaluation Worksheets and the identified deficiency. This additional information is included on the revised worksheets included in Appendix C of this report.

4.3 Equipment Considered Conditionally Acceptable

Appendix C of the Safety Evaluation Report [1] lists equipment considered conditionally acceptable. All equipment listed in the category have aging identified as the only deficiency.

Because the number of components in this category is small and the aging deficiency is not significantly different than those previously discussed, the revised Component Evaluation Worksheets for these components are included in Appendix C of this report. The additional information describing each aging evaluation is included on the worksheets.

5.0 CORRECTIVE ACTIONS

Corrective actions required to resolve outstanding items identified during the equipment qualification evaluations have been developed. These corrective actions are discussed below.

5.1 Component Specific Items

Component specific corrective action plans have been developed to resolve outstanding qualification items. The corrective action plans include the following:

1. Component Replacement
2. Component Relocation
3. Component Modification
4. Component Testing

The specific action plan for each component is indicated on the component evaluation worksheets included in Appendix C.

5.2 Aging

The NRC staff requested additional information regarding aging of all safety related equipment. Section 3.4 of this report describes the analysis methods used to address the aging requirement.

As required by the NRC, a program is being established at Monticello to review component maintenance and surveillance records to identify potential age related degradations. Information obtained from this review program and the specific component aging analyses is being incorporated into existing component maintenance and replacement schedules.

Maintenance and surveillance procedures currently in effect track the degradation of many components. Also, replacement schedules exist for components with known aging problems. These procedures are being reviewed to determine the impact of the recently compiled and ongoing aging evaluations being performed by Northern States Power. Component specific maintenance and surveillance procedures will be modified or developed as required by our evaluations.

6.0 JUSTIFICATION FOR CONTINUED OPERATION

Justification for continued plant operation has been provided for all components for which complete qualification documentation could not be obtained. These justifications are found as notes to the applicable Component Evaluation Worksheets included in Appendix C.

For components where no environmental test data has been found, justification is based on discussions with the vendor, review of the materials of construction, design specifications, operating experience and similarity to qualified equipment. Additionally, when applicable, justification included consideration of the distance between the equipment and the location of the postulated HELB and the availability of redundant qualified equipment.

7.0 REFERENCES

1. Safety Evaluation Report by the Office of Nuclear Reactor Regulation Equipment Qualification Branch for Northern States Power Company Monticello Nuclear Generating Plant Document Number 80-263.

2. Northern States Power Company; "Monticello Nuclear Generating Plant - Response to IE Bulletin 79-01B", EDS Report No 01-0910-1087, Rev. 2, November 1, 1980.

3. NEDO 24576.

4. NEDO 21888.

5. NUREG 0661 - Mark I Containment Long Term Program Safety Evaluation Report, Resolution of Generic Technical Activity A-7. February 1977 to December 1979, July 1980.

6. Northern States Power Company; "Monticello Nuclear Generating Plant-Final Safety Analysis Report"; Docket No. 50-263, Oct. 1969, License No. DPR-22.

7. EDS Report No. 01-0910-1137; "Monticello Nuclear Generating Plant Environmental Effects Due to Pipe Rupture," Rev. 0; Dec. 1980.

8. Addendum to EDS Report No. 01-0910-1137; May 1981.

9. Regulatory Guide 1.97 - "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 2; Dec. 1980.

10. Environmental Qualification of Class 1E Equipment, NRC Office of Inspectin and Enforcement Bulletin No. 79-01B, January 14, 1980.

APPENDIX A

SAFETY RELATED ELECTRICAL EQUIPMENT

MASTERLIST

SYSTEM: Main Steam (MSS)				
PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 2373	Valve Motor Operator	Limitorque	SMB-000	B.1
MO 2374	Valve Motor Operator	Limitorque	SMB-000	B.7
SV 2-71 (A-H)	Solenoid Valve	Automatic Valve Co.	C-5450	B.1
SV 2-80 (A-D)	Solenoid Valve	Automatic Valve Co.	C-4988-15	B.1
SV-2-86 (A-D)	Solenoid Valve	Automatic Valve Co.	C-4988-15	B.7
TS 2-121 (A-D)	Temperature Switch	Fenwal	17002-40	B.7
TS 2-122 (A-D)	Temperature Switch	Fenwal	17002-40	B.7
TS 2-123 (A-D)	Temperature Switch	Fenwal	17002-40	B.7
TS 2-124 (A-D)	Temperautre Switch	Fenwal	17002-40	B.7
dPIS 2-116 (A-D)	Differential Pressure Indicating Switch	Barton	278	B.9
dPIS 2-117 (A-D)	Differential Pressure Indicating Switch	Barton	278	B.9
dPIS 2-118 (A-D)	Differential Pressure Indicating Switch	Barton	278	B.9
dPIS 2-119 (A-D)	Differential Pressure Indicating Switch	Barton	278	B.9

Table A.1

SYSTEM: Nuclear Boiler-Vessel Instrumentation (NBVI)				
PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
PS 2-3-49 (A,B)	Pressure Switch	Barksdale	B2T-A12SS	Radiation Only
PS 2-3-50 (A,B)	Pressure Switch	Barksdale	B2T-A12SS	Radiation Only
PS 2-3-51 (A-D)	Pressure Switch	Barksdale	B2T-A12SS	B.13
PS 2-3-52(A)	Pressure Switch	Barksdale	B2T-A12SS	B.13
PS 2-3-52(B)	Pressure Switch	Barton	288	B.13
PS 2-3-53 (A,B)	Pressure Switch	Barksdale	B2T-A12SS	B.8
LIS 2-3-57 (A,B)	Level Indicating Switch	Yarway	4418C	B.13
LIS 2-3-58 (A,B)	Level Indicating Switch	Yarway	4418C	B.13
LIS 2-3-72 (A-D)	Level Indicating Switch	Yarway	4418C	B.13
LITS 2-3-73 (A,B)	Level Indicating Switch	Yarway	4418EC	B.8(A) B.9(B)
LITS 2-3-59 (A,B)	Level Indicating Transmitter-Switch	Yarway	4418CE	B.13
LT 2-3-61	Level Transmitter	GE	533	B.13
PT 6-53 (A,B)	Pressure Transmitter	GE	551	B.13

Table A.2

SYSTEM: Reactor Recirculation

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 2-53 (A,B)	Valve Motor Operator	Limitorque	SMB	B.1
MO 2-54 (A,B)	Valve Motor Operator	Limitorque	SMB	B.1
SV 2790	Solenoid Valve	ASCO	NP 8321A1E	B.1
SV 2791	Solenoid Valve	ASCO	THT831723	B.11
PS 2-128 (A,B)	Pressure Switch	Static-O-Ring	6N-AA3	B.8
dPIS 2-129 (A-D)	Differential Pressure Indicating Switch	Barton	288	B.8 B.9
dPIS 2-136 (A,B)	Differential Pressure Indicating Switch	Barton	288	B.8 B.9
dPIS 2-137 (A,B)	Differential Pressure Indicating Switch	Barton	288	Radiation Only
dPIS 2-138 (A,B)	Differential Pressure Indicating Switch	Barton	288	Radiation Only
dPIS 2-139 (A,B)	Differential Pressure Indicating Switch	Barton	288	Radiation Only

Table A.3

SYSTEM: Control Rod Drive (CRD)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 3-140 (A,B)	Solenoid Valve	ASCO	WPLBX 831636	B.9
SV 3-31 (A,B)	Solenoid Valve	ASCO	HVA-90-441-1A	B.8
SV 3-13-117	Solenoid Valve	ASCO	HVA-90-405	B.8
SV 3-13-118	Solenoid Valve	ASCO	HVA-90-405	B.8

Table A.4

SYSTEM: Residual Heat Removal (RHR)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 1989	Valve Motor Operator	Rotork	30A S/N S2221	B.4
MO 2002	Valve Motor Operator	Rotork	150A S/N S2221	B.4
MO 2003	Valve Motor Operator	Rotork	150A S/N S2222	B.4
MO 2006	Valve Motor Operator	Rotork	35A S/N S2207	B.3
MO 2007	Valve Motor Operator	Rotork	35A S/N S2208	B.15
MO 2008	Valve Motor Operator	Rotork	70A S/N S2219	B.3
MO 2009	Valve Motor Operator	Rotork	70A S/N S2220	B.3
MO 2010	Valve Motor Operator	Rotork	14AMKII S/N S2205	B.3
MO 2011	Valve Motor Operator	Rotork	14AMKII S/N S2205	B.3
MO 2012	Valve Motor Operator	Limitorque	SMB-5	B.10
MO 2013	Valve Motor Operator	Limitorque	SMB-5	B.10
MO 2014	Valve Motor Operator	Limitorque	SMB-2	B.10
MO 2015	Valve Motor Operator	Limitorque	SMB-2	B.10
MO 2020	Valve Motor Operator	Rotork	30A S/N S2215	B.9
MO 2021	Valve Motor Operator	Rotork	30A S/N S2212	B.13
MO 2022	Valve Motor Operator	Rotork	30A S/N S2213	B.9
MO 2023	Valve Motor Operator	Rotork	30A S/N S2214	B.13

Table A.5i

SYSTEM: Residual Heat Removal (cont'd)				
PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 2026	Valve Motor Operator	Limitorque	SMB-00	B.13
MO 2027	Valve Motor Operator	Limitorque	SMB-00	B.1
MO 2029	Valve Motor Operator	Limitorque	SMB-00	B.1
MO 2030	Valve Motor Operator	Limitorque	SMB-4	B.10
MO 2032	Valve Motor Operator	Rotork	12A S/N S2204	B.3
MO 2407	Valve Motor Operator	Rotork	12A S/N S2315	B.3
SV 1728	Solenoid Valve	ASCO	G:T-HP-830081RU	B.4
SV 1729	Solenoid Valve	ASCO	G:T-HP-830081RU	B.4
SV 1994	Solenoid Valve	ASCO	A:T-HT-831723	B.4
SV 1995	Solenoid Valve	ASCO	A:T-HT-831723	B.4
SV 1996	Solenoid Valve	ASCO	A:T-HT-831723	B.4
SV 1997	Solenoid Valve	ASCO	A:T-HT-831723	B.4
E/P 1728	Electro-Pneumatic Transducer	Fisher	546	B.4
E/P 1729	Electro-Pneumatic Transducer	Fisher	546	B.4
dPT 10-91 (A,B)	Differential Pressure Transmitter	Barton	296	B.4
PS 7192	Pressure Switch	Square D	GHG551	B.16
PS 10-105 (E-H)	Pressure Switch	Static-O-Ring	5N AA3X	B.4

Table A.5ii

SYSTEM: Residual Heat Removal (Cont'd)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
PS 10-119 (A-D)	Pressure Switch	Static-O-Ring	12N-AA4	B.13
FS 10-121 (A-D)	Flow Switch	Peeco	HP-F	B.4
P-202 (A-D)	Pump Motor	GE	5K6329XC4A	B.4
PS 7193	Pressure Switch	Square D	GHG551	B.15
MO 1986	Valve Motor Operator	Rotork	30A S/N S2186	B.4
MO 1987	Valve Motor Operator	Rotork	30A S/N S2185	B.4
MO 1988	Valve Motor Operator	Rotork	30A S/N S2209	B.4
PS 10-100 (A-D)	Pressure Switch	Static-O-Ring	12N AA4	B.13
PS 10-101 (A-D)	Pressure Switch	Static-O-Ring	12N-AA4	B.13
FT 10-109 (A,B)	Flow Transmitter	GE	553	B.4
FT 10-111 (A,B)	Flow Transmitter	GE	553	B.4 (A) B.15 (B)
K-10 (A,B)	Aux. Compressor Motor	GE	5K145A1246	B.16(A) B.15(B)
K-10 (A,B)	Aux. Compressor Motor Starter	GE	CR106	B.16(A) B.15(B)
PS 10-105 (A-D)	Pressure Switch	Mericoid	DAW-23-156	B.4
K-10 (A,B)	Aux. Comp Line Switch	GE	THN 3361, mod 2	B.16(A) B.15(B)
N3347	Aux. Comp Disconnect	GE	THN 3361, mod 2	B.16
N4347	Aux. Comp Disconnect	GE	THN 3361, mod 2	B.15

Table A.5iii

SYSTEM: Core Spray (CSS)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 1749	Valve Motor Operator	Rotork	30A	B.3
MO 1750	Valve Motor Operator	Rotork	30A	B.3
MO 1751	Valve Motor Operator	Limitorque	SMB-2	B.12
MO 1752	Valve Motor Operator	Limitorque	SMB-2	B.11
MO 1753	Valve Motor Operator	Limitorque	SMB-2	B.12
MO 1754	Valve Motor Operator	Limitorque	SMB-2	B.11
PS 14-44 (A-D)	Pressure Switch	Barksdale	B2T-A12SS	B.4
P-208 (A, B)	Pump Motor	GE	5K6338XC39	B.4
FT 14-40 (A, B)	Flow Transmitter	GE	533	B.4

Table A.6

SYSTEM: High Pressure Coolant Injection (HPCI)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 2034	Valve Motor Operator	Limitorque	SMB-0	B.1
MO 2035	Valve Motor Operator	Limitorque	SMB-0	B.7
MO 2036	Valve Motor Operator	Limitorque	SMB-1	B.6
MO 2061	Valve Motor Operator	Rotork	16AMKII	B.6
MO 2062	Valve Motor Operator	Rotork	16AMKII	B.6
MO 2063	Valve Motor Operator	Rotork	16AMKII	B.6
MO 2067	Valve Motor Operator	Limitorque	SMB-4	B.6
MO 2068	Valve Motor Operator	Limitorque	SMB-4	B.7
Mo 2071	Valve Motor Operator	Limtorque	SMB-4	B.6
PS 23-68 (A-D)	Pressure Switch	Barksdale	B2T-A12SS	B.9
PS 23-97 (A, B)	Pressure Switch	Mercoid	DA-7043-804	B.6
PS 23-84	Pressure Switch	Mercoid	DAW-443-4132 -R26E	B.6
dPIS 23-76 (A, B)	Differential Pressure Indicating Switch	Barton	288A	B.8
dPIS 23-77 (A,B)	Differential Pressure Indicating Switch	Barton	288A	B.8
TS 23-101 (A-D)	Temperature Switch	Fenwal	17023-6	B.2
TS 23-102 (A-D)	Temperature Switch	Fenwal	17023-6	B.2
TS 23-103 (A-D)	Temperature Switch	Fenwal	17023-6	B.3(A) B.3(B-D)

Table A.7i

SYSTEM: High Pressure Coolant Injection (cont'd)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
TS 23-104 (A-D)	Temperature Switch	Fenwal	17023-6	B.3(A) B.3(B-D)
P-217	Motor	Baldor	669	B.6
---	EGM	Woodward Governor	8270-811	B.6
---	EGR	Woodward Governor	A 8250-133	B.6
SV 2065	Solenoid Valve	ASCO	T-HT-83212	B.6
FS 23-78	Flow Switch	Barton	289	B.6
FT 23-82	Flow Transmitter	GE	553	B.6
LS 23-91 (A, B)	Level Switch	Magnetrol	249-C	B.3
LS 23-74	Level Switch	Robertshaw	SL-412-A1	B.8
LS 23-75	Level Switch	Robertshaw	SL-412-A1	B.8
---	Limit Switch	NAMCO	EA170-34101	B.6
---	Magnetic Pick-up	Woodward Governor	1680-622	B.6
---	Ramp Generator	Woodward Governor	8271-083	B.6

Table A.7ii

SYSTEM: Reactor Core Isolation Cooling (RCIC)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 2075	Valve Motor Operator	Limitorque	SMB-000	B.1
MO 2076	Valve Motor Operator	Limitorque	SMB-000	B.7
TS 13-79 (A-D)	Temperature Switch	Fenwal	17023-6	B.3(A) B.3(B-D)
TS 13-80 (A-D)	Temperature Switch	Fenwal	17023-6	B.3(A) B.3(B-D)
TS 13-81 (A-D)	Temperature Switch	Fenwal	17023-6	B.3(A) B.3(B-D)
TS 13-82 (A-D)	Temperature Switch	Fenwal	17023-6	B.3(A) B.3(B-D)
dPIS 13-83	Differential Pressure Indicating Switch	Barton	288	B.9
dPIS 13-84	Differential Pressure Indicating Switch	Barton	288	B.9
PS 13-87 (A-D)	Pressure Switch	Meletron	372-6SS49A	B.9
MO 2107	Valve Motor Operator	Limitorque	SMB-00	B.7

Table A.8

SYSTEM: Primary Containment and Atmosphere Control

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 2377	Solenoid Valve	ASCO	8300C64U	B.3
SV 2378	Solenoid Valve	ASCO	8300C64U	B.3
SV 2379	Solenoid Valve	ASCO	8262A212	B.3
SV 2380	Solenoid Valve	ASCO	8262A212	B.3
SV 2381	Solenoid Valve	ASCO	8300C64U	B.3
SV 2383	Solenoid Valve	ASCO	8300C64U	B.3
SV 2384	Solenoid Valve	ASCO	T-HT-8317A23	B.3
SV 2385	Solenoid Valve	ASCO	T-HT-831723	B.11
SV 2386	Solenoid Valve	ASCO	8300C64U	B.11
SV 2387	Solenoid Valve	ASCO	8300C64U	B.11
SV 2896	Solenoid Valve	ASCO	8300C64U	B.3
dPS 2573	Differential Pressure Switch	Barton	289A	Radiation Only
dPS 2572	Differential Pressure Switch	Barton	289A	Radiation Only
SV 7440	Solenoid Valve	ASCO	T-HT-8317B23	B.3

Table A.9

SYSTEM: Standby Gas Treatment (SBGT)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 2944	Solenoid Valve	ASCO	A:T-HT-831723	Isolated
SV 2945	Solenoid Valve	ASCO	A:T-HT-831723	Isolated
SV 2978	Solenoid Valve	ASCO	A:T-HT-831723	Isolated
SV 2979	Solenoid Valve	ASCO	A:T-HT-831723	Isolated
SV 2982 (A,B)	Solenoid Valve	ASCO	A:T-HT 831723	Isolated
PS 3462	Pressure Switch	Furnas	69GAV7	Isolated
FT 2942	Flow Transmitter	Leeds & Northup	1912-2-10-0-000-0100-0100	Isolated
V-EF-17 (A, B)	Fan Motor	GE	5K 54AK205	Isolated
FT 2943	Flow Transmitter	Leeds & Northrup	1912-2-10-0-000-0100-0100	Isolated
FS 2950	Flow Switch	McDonnell & Miller	AF1S	Isolated
FS 2951	Flow Switch	McDonnell & Miller	AF1S	Isolated
TS 3368	Temperature Switch	Chromalox	AR 2529	Isolated
TS 3369	Temperature Switch	Chromalox	AR 2529	Isolated
K-11	Aux. Compressor Motor	GE	5K43KG802	Isolated
E/P 2942	Electro- Pneumatic Transducer	Leeds & Northrup	10970-1	Isolated
E/P 2943	Electro-Pneumatic Transducer	Leeds & Northrup	10970-1	Isolated
LC-1 (A-B)	Contactor	Cutler Hammer	G-10-2	Isolated

Table A.10i

SYSTEM: Standby Gas Treatment (Cont'd)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
LC-2 (A-B)	Fused Disconnect	Cutler Hammer	4105H311H	Isolated
E-34 (A-B)	Thermostat	Honeywell	T451A	Isolated
E-34 (A-B)	Unit Heater	ILG Industries	H7133	Isolated
K-11	Aux. Compressor Motor Starter	Furnas	14BA32BC	Isolated
K-11	Aux. Compressor Line Switch	GE	THN 3361 THN 3362	Isolated
T1	Control XFMR	Heavy Duty Electric	SZO	Isolated
Allen-Bradley Bulletin 1492-CD3	Terminal Board	Allen-Bradley	1492-CD3	Isolated
----	Wire	GE	SI-57275	Isolated
----	Cable	GE	SI-58175	Isolated
----	Cable	GE	SI-58170	Isolated
----	Cable	Carol Wire & Cable	1050	Isolated
----	Limit Switch	Micro Switch	OP-AR	Isolated
----	Limit Switch	Cutler Hammer	Type L	Isolated

Table A.10ii

SYSTEM: Reactor Building Cooling Water

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 1426	Valve Motor Operator	Rotork	14AMKII	B.3

Table A.11

SYSTEM: Reactor Water Cleanup (RWCU)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MO 2397	Valve Motor Operator	Limitorque	SMB-00	B.1
MO 2398	Valve Motor Operator	Limitorque	SMB-00	B.11
MO 2399	Valve Motor Operator	Rotork	30A	B.17

Table A.12

SYSTEM: Reactor Building Instrument Air

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 1478	Solenoid Valve	ASCO	THT 831723	B.7
SV 7956	Solenoid Valve	ASCO	THT 8317B23	B.3

Table A.13

SYSTEM: Automatic Depressurization (ADS)				
PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 2-71 (A-C)	Solenoid Valve	Automatic Valve Co.	C-5450	B.1

Table A.14

SYSTEM: Reactor Protection

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
PS 5-14 (A-D)	Pressure Switch	Barksdale	B2T-A12SS	B.14
PS 5-12 (A-D)	Pressure Switch	Static-O-Ring	12-K4	B.13

Table A.15

SYSTEM: Radwaste

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 2541 (A,B)	Solenoid Valve	ASCO	THT 831723	B.3
SV 2561 (A,B)	Solenoid Valve	ASCO	THT 831723	B.3

Table A.16

SYSTEM: Primary Containment Nitrogen Control

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
SV 3267	Solenoid Valve	ASCO	T-HT-831723	B.3
SV 3268	Solenoid Valve	ASCO	T-HT-831723	B.3
SV 3269A	Solenoid Valve	ASCO	T-HT-831723	B.8
SV 3305	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3306	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3307	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3308	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3309	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3310	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3311	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3312	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3313	Solenoid Valve	ASCO	T-HT-831723	B.15
SV 3314	Solenoid Valve	ASCO	T-HT-831723	B.15

Table A.17

SYSTEM: Containment

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
LT 7338 (A, B)	Level Transmitter	Rosemount	1153A	B.3
PT 2994B	Pressure Transmitter	GE	552	B.2
PT 7348	Pressure Transmitter	GE	551	B.13
TE 2995 (F, G)	Thermocouple	Thermo-Electric	CE S16-2-516-T	B.3

Table A.18

SYSTEM: Components

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
----	Electrical Cable	GE	SI-58007	B.7
----	Electrical Cable	GE	SI-58136	Various
----	Electrical Cable	GE	SI-58081	Various
----	Electrical Cable	GE	SI-58109	B.1
----	Electrical Cable	Rockbestos	Firewall III	B.1
----	Instrument Cable	Samuel Moore	#1802, 1852, 1862	Various
----	Electrical Cable	GE	SI-58042	Various
JX-105 (A, C, D)	Electrical Penetration	GE	NSO-2, NSO-3, NSO-4	B.1
----	Field Space	Raychem	WCSF-N	B.1
----	Terminal Board	GE	CR151D3	Various
----	Limit Switch	NAMCO	EA08021100	B.3
----	Limit Switch	NAMCO	D2400X ST	B.3
----	Limit Switch	NAMCO	EA 17014100	B.3
----	Limit Switch	NAMCO	D 1200G	B.4
----	Limit Switch	NAMCO	SL5-C3L	B.3
----	Limit Switch	NAMCO	SL3-B2W	B.7
----	Limit Switch	NAMCO	EA-740-50100	B.1

Table A.19i

SYSTEM: Components (cont'd)

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
----	Limit Switch	NAMCO	EA-740-8000	B.1
----	Limit Switch	Micro Switch	BZE6-2RN	B.3
----	Limit Switch	Micro Switch	BZE6-2RQ2	B.3
----	Banana Plug	E. F. Johnson	108-0300-01	B.7

Table A.19ii

SYSTEM: 250VDC

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
D 311	Motor Control Center	GE	IC-7700	B.2 B.6
D 312	Motor Control Center	GE	IC-7700	B.5

Table A.20

SYSTEM: 480 VAC Station Auxiliary

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
MCC 142	Motor Control Center	GE	DA-7700	B.14
MCC 143 (A, B)	Motor Control Center	GE	DA-7700	B.14

Table A.21

SYSTEM: Heating and Ventilating

PLANT I.D. NUMBER	COMPONENT	MANUFACTURER	MODEL	ENVIRON-MENT
V-AC-4	Fan Motor	U.S. Motors	S/N 4152941	B.4
V-AC-5	Fan Motor	U.S. Motors	S/N 4152940	B.4
V-AC-6	Fan Motor	U.S. Motors	F-1088-01-268	B.5
V-AC-8 (A, B)	Fan Motor	U.S. Motors	F-1323-02-268	B.2

Table A.22

APPENDIX B

TEMPERATURE AND PRESSURE PROFILES

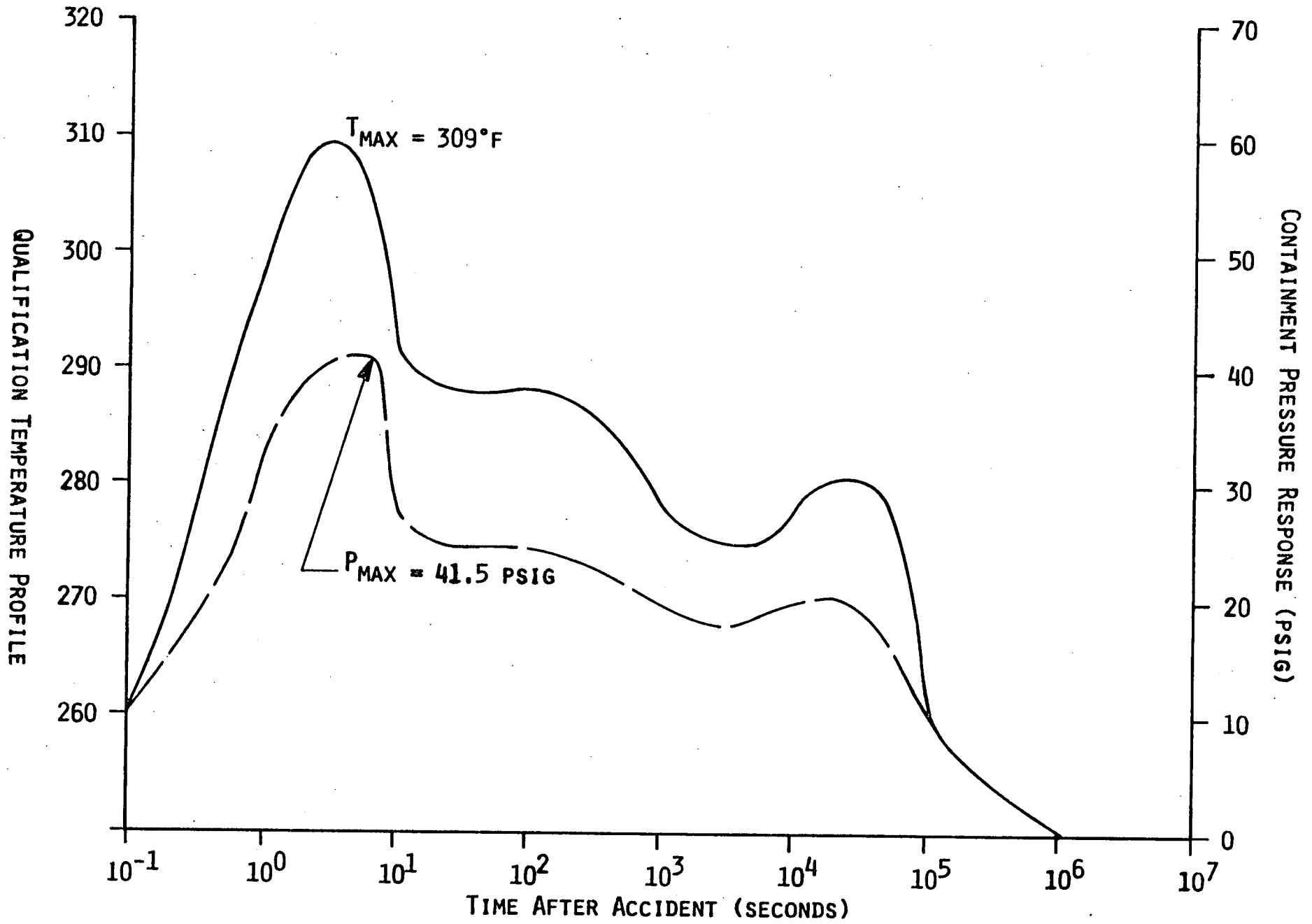


FIGURE B-1 DRYWELL PRESSURE RESPONSE AND TEMPERATURE PROFILE USED FOR EQUIPMENT QUALIFICATION

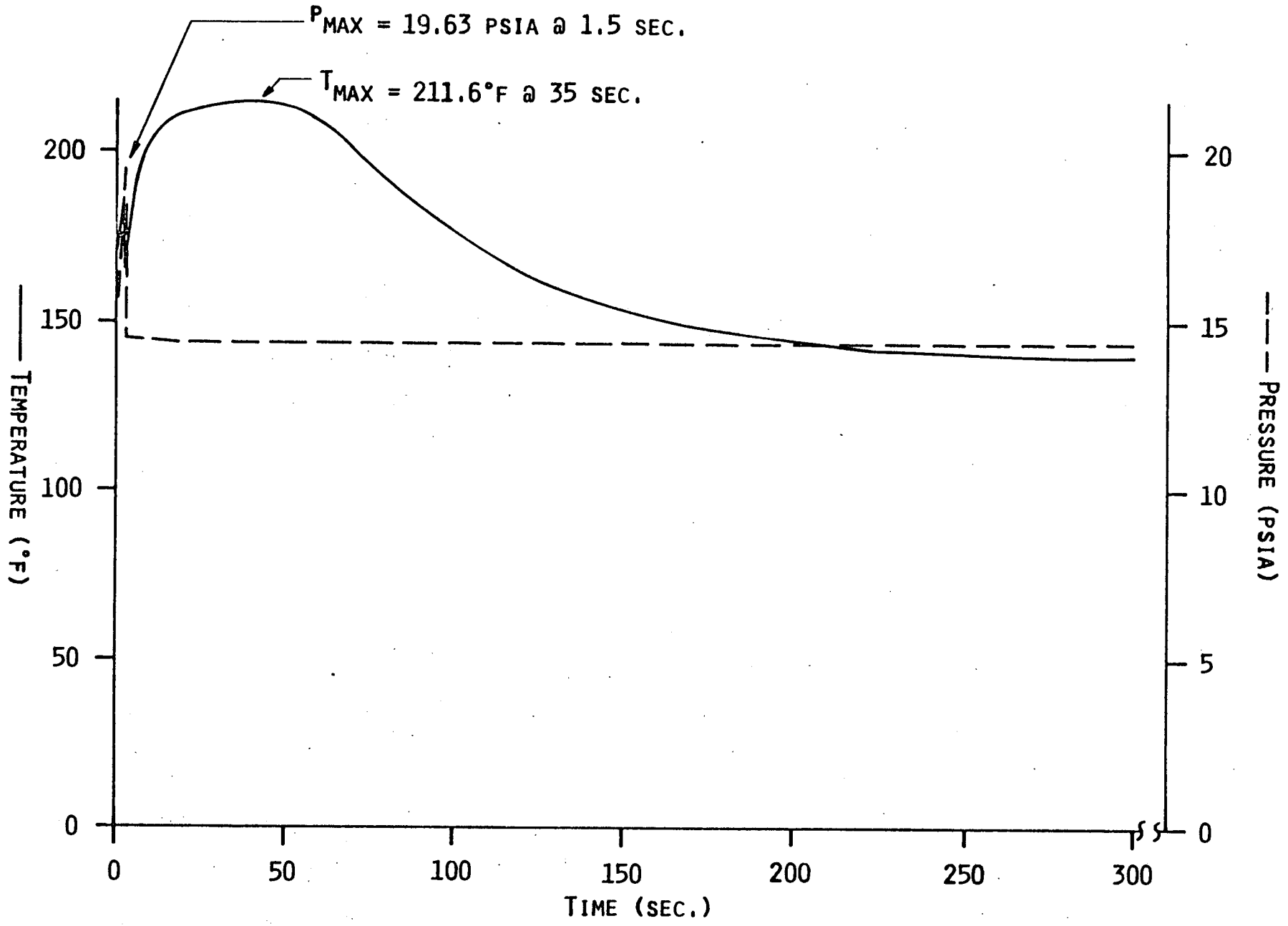


FIGURE B.2 - HPCI STEAM LINE BREAK IN HPCI ROOM
TEMPERATURE AND PRESSURE TRANSIENT FOR HPCI ROOM

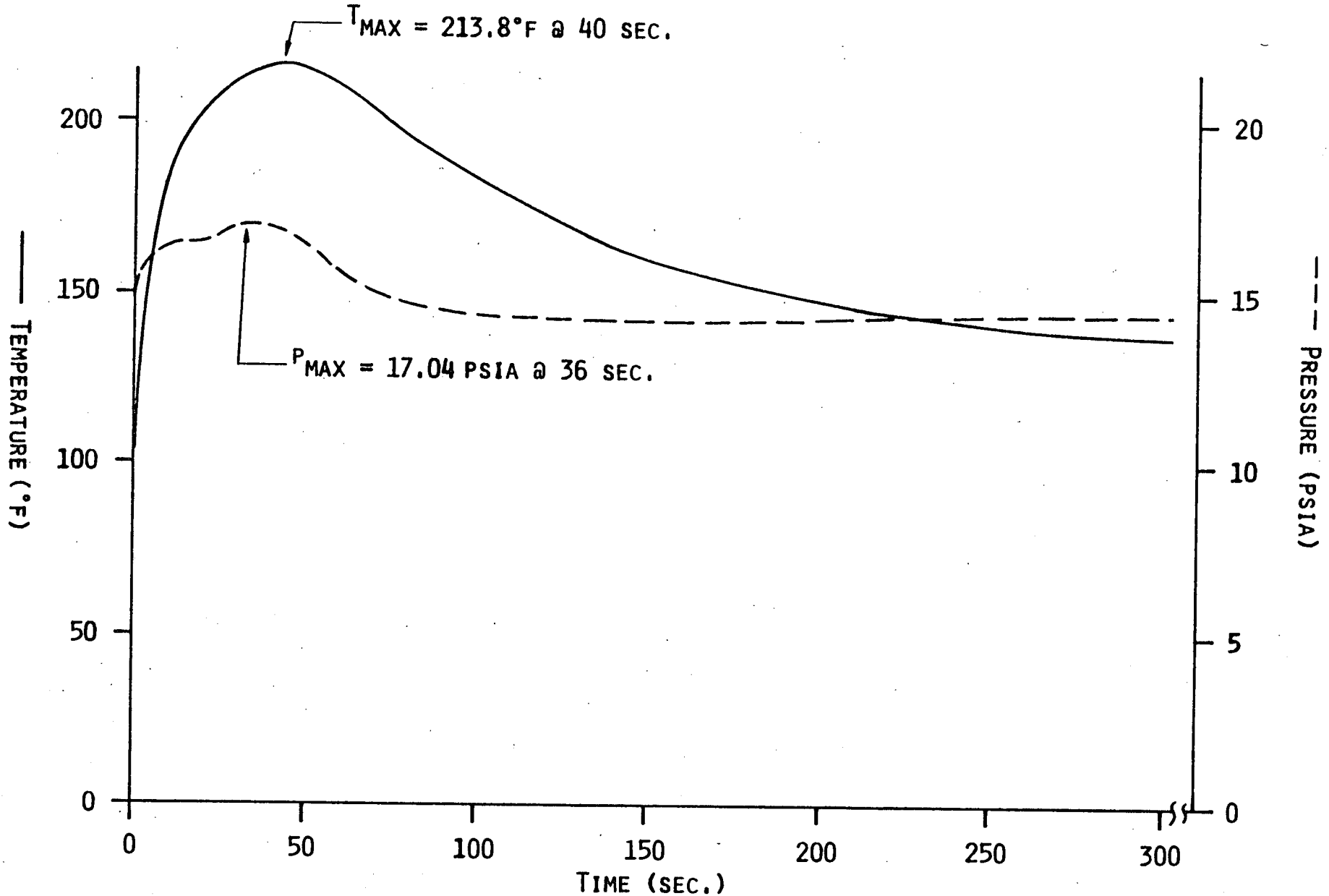


FIGURE B.3 - HPCI STEAM LINE BREAK IN TORUS COMPARTMENT
TEMPERATURE AND PRESSURE TRANSIENT FOR TORUS COMPARTMENT

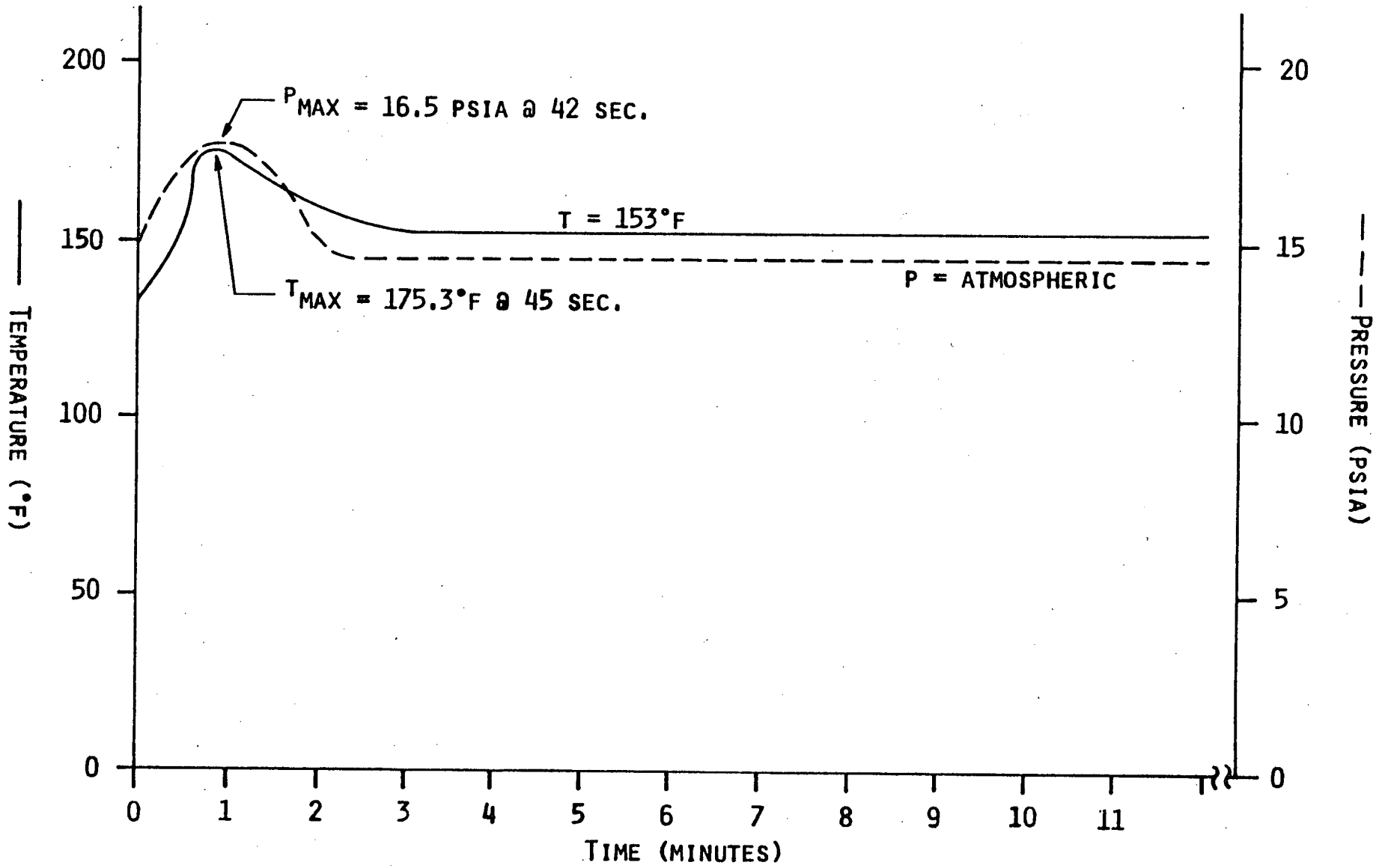


FIGURE B.4 - HPCI STEAM LINE BREAK IN TORUS COMPARTMENT TEMPERATURE AND PRESSURE TRANSIENT FOR RHR ROOMS

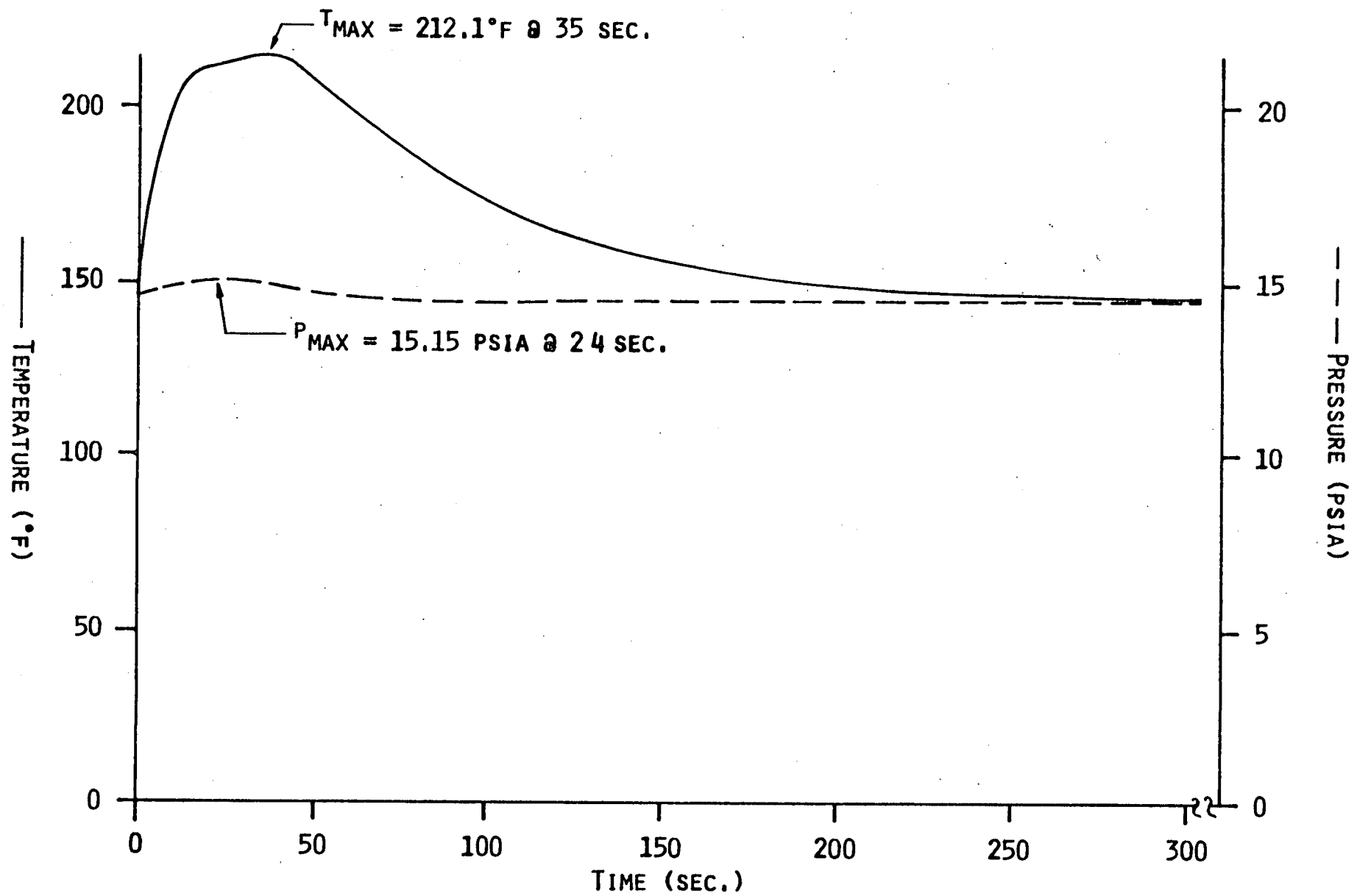


FIGURE B.5 - RCIC STEAM LINE BREAK IN RCIC ROOM
TEMPERATURE AND PRESSURE TRANSIENT FOR RCIC ROOM

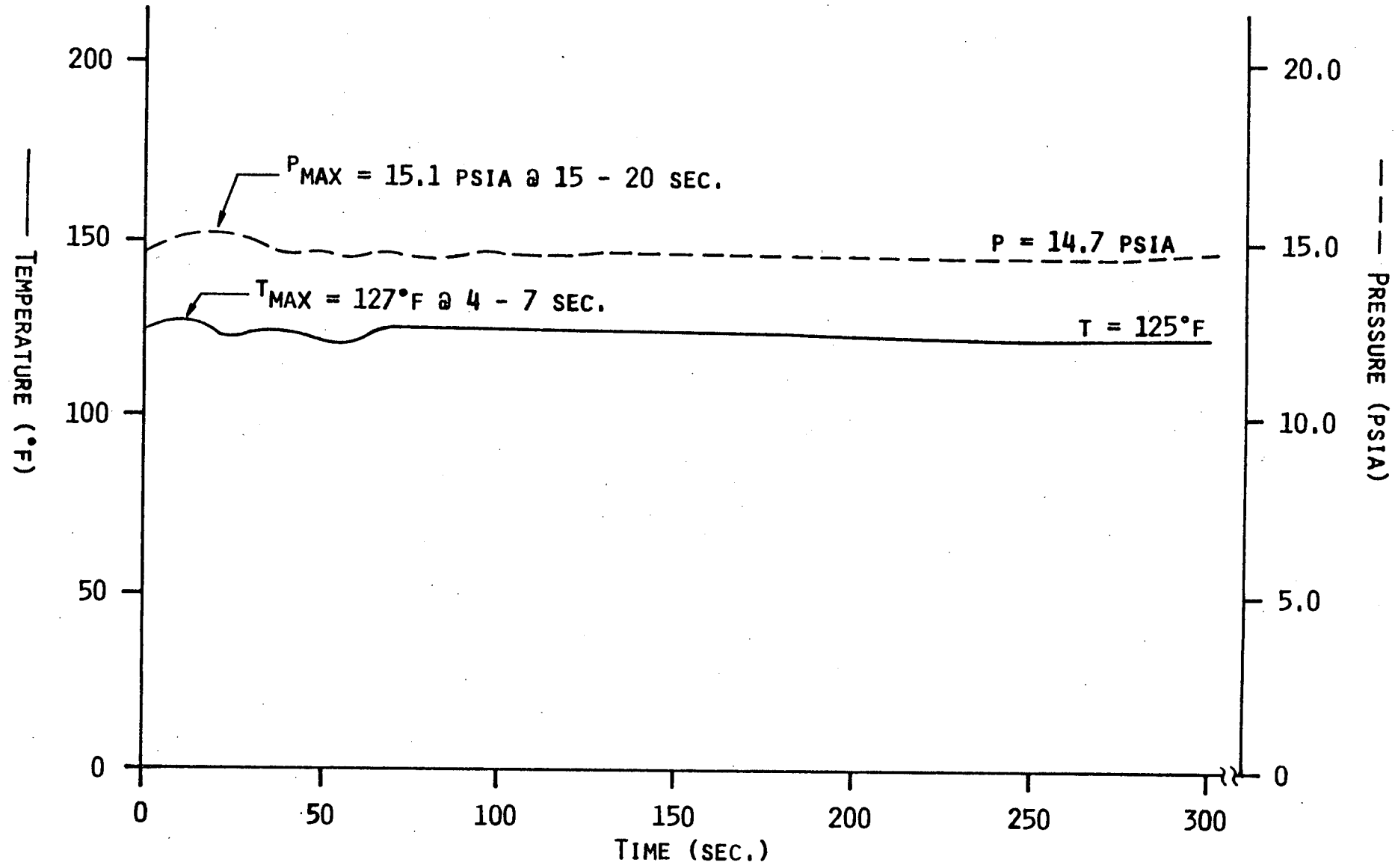


FIGURE B.6 - RCIC STEAM LINE BREAK IN TORUS COMPARTMENT TEMPERATURE AND PRESSURE TRANSIENT FOR HPCI ROOM

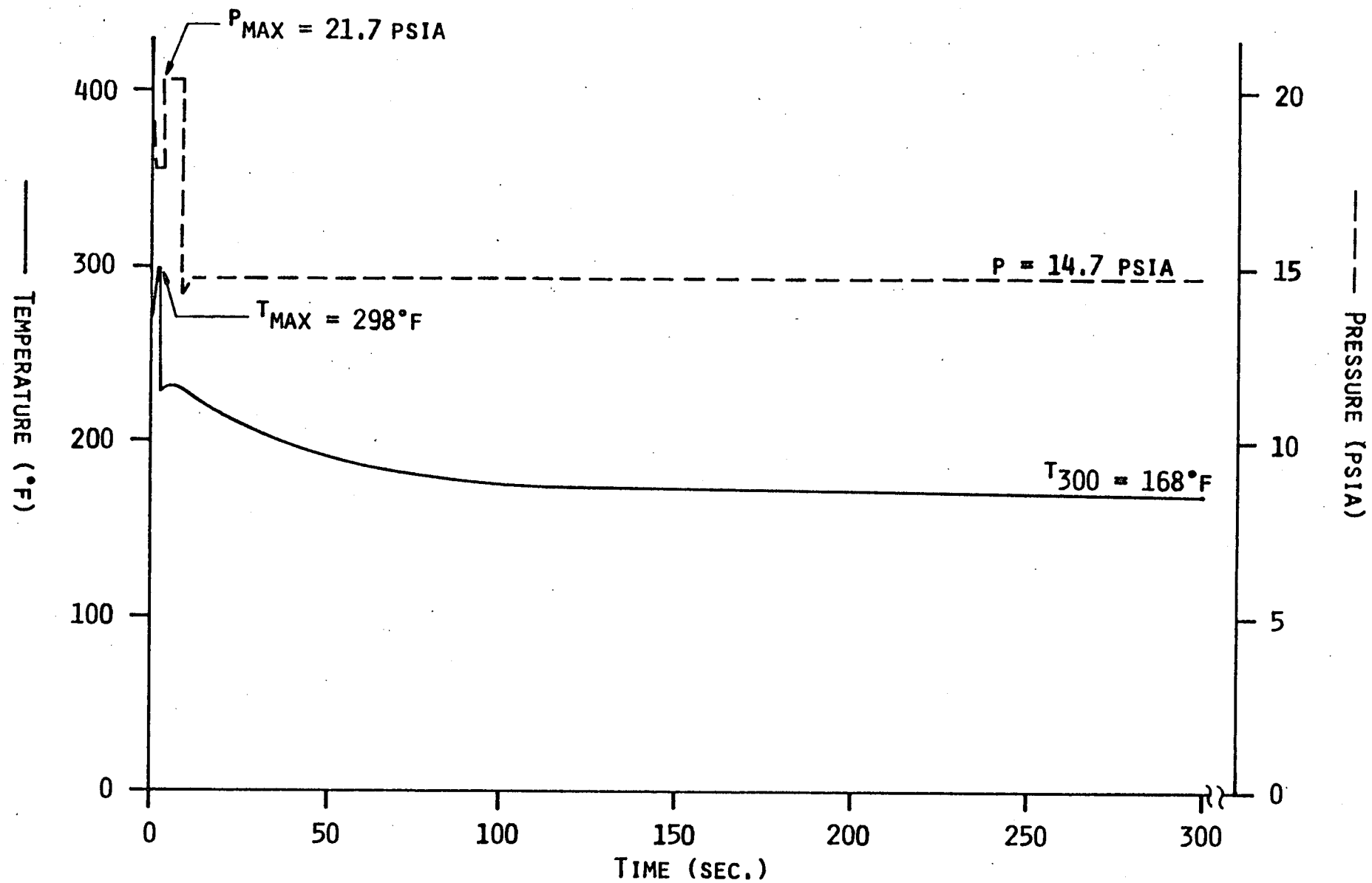


FIGURE B.7 - MAIN STEAM LINE BREAK IN MAIN STEAM TUNNEL
TEMPERATURE AND PRESSURE TRANSIENT FOR MAIN STEAM TUNNEL

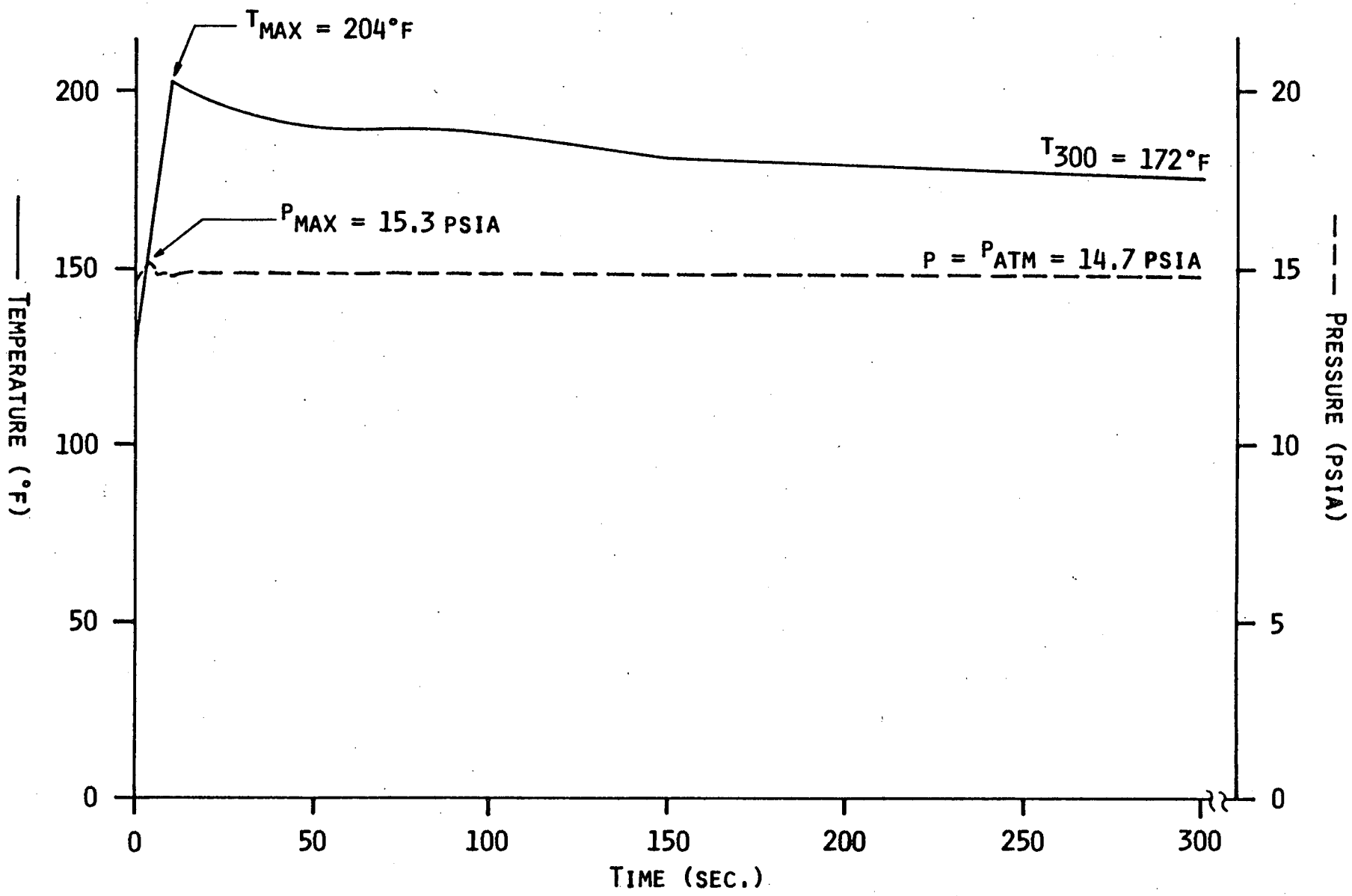


FIGURE B.8 - MAIN STEAM LINE BREAK IN MAIN STEAM TUNNEL TEMPERATURE AND PRESSURE TRANSIENT FOR WEST SIDE REACTOR BLDG. EL. 935'-0"

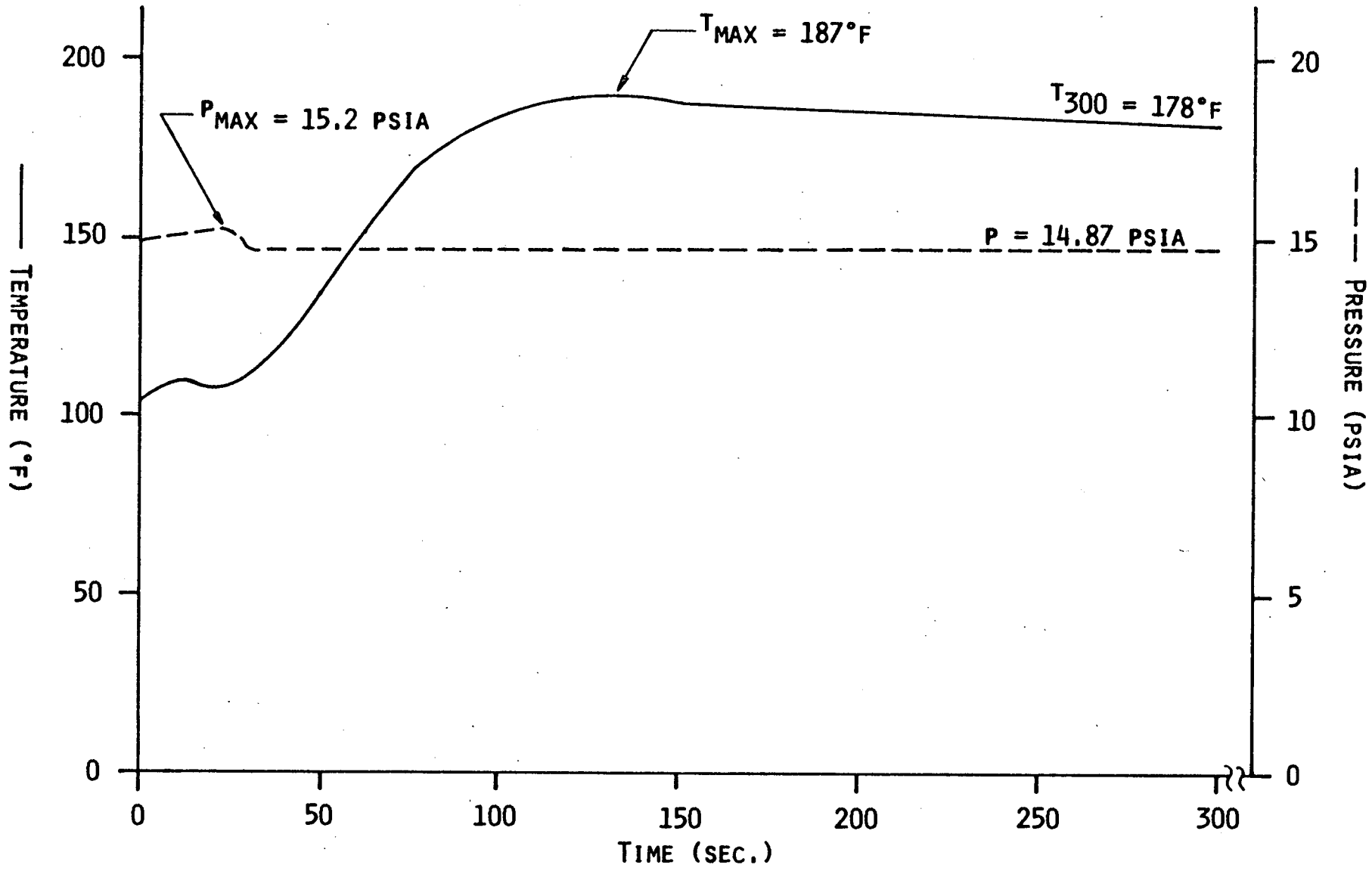


FIGURE B.9 - RWCU LINE BREAK IN WEST SIDE REACTOR BLDG.
TEMPERATURE AND PRESSURE TRANSIENT FOR EAST SIDE REACTOR BLDG.
EL. 935'-0"

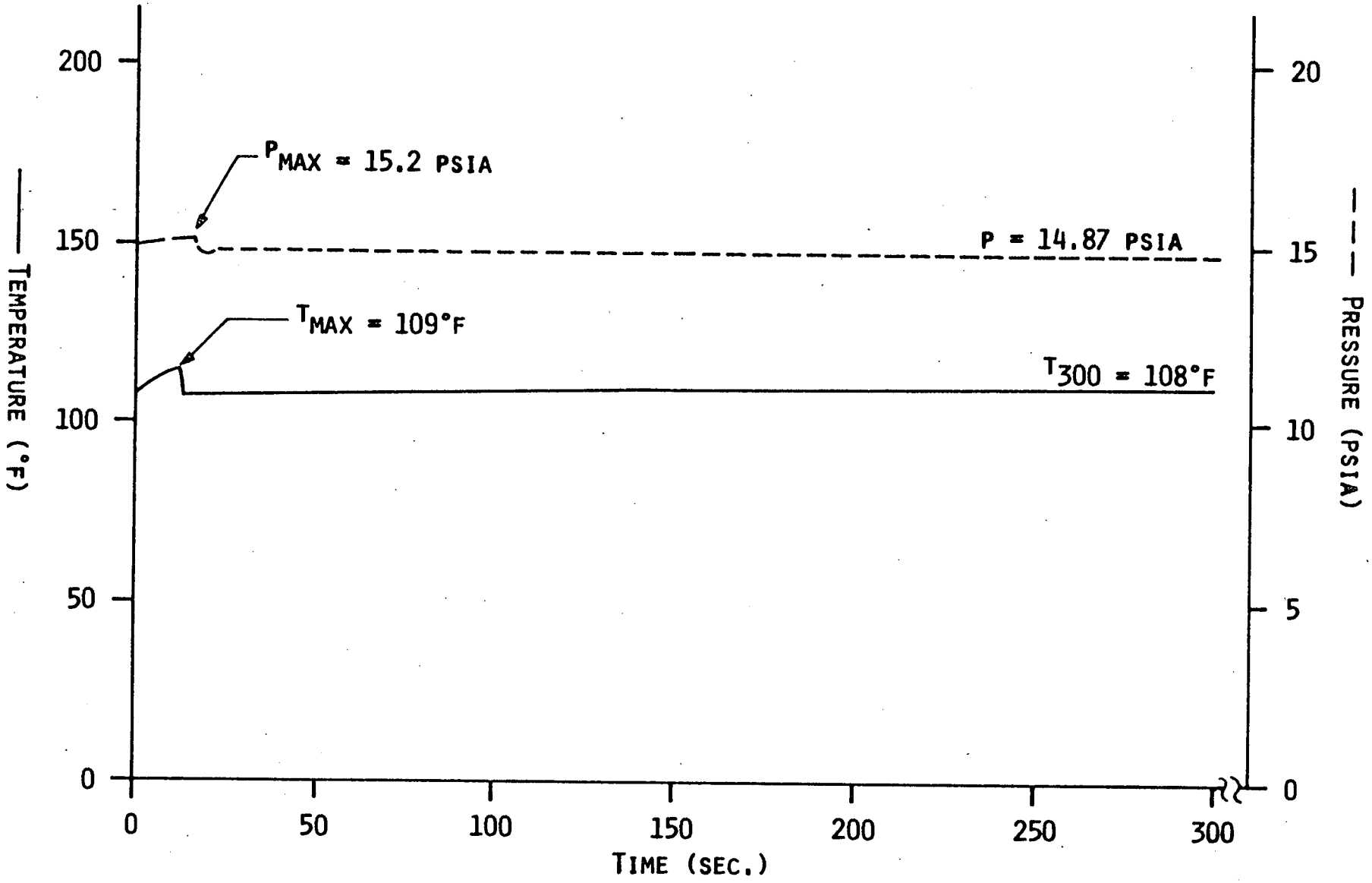


FIGURE B. 10 - RWCU LINE BREAK IN WEST SIDE REACTOR BLDG.
TEMPERATURE AND PRESSURE TRANSIENT FOR LPCI INJECTION VALVE AREA
EL. 935'-0"

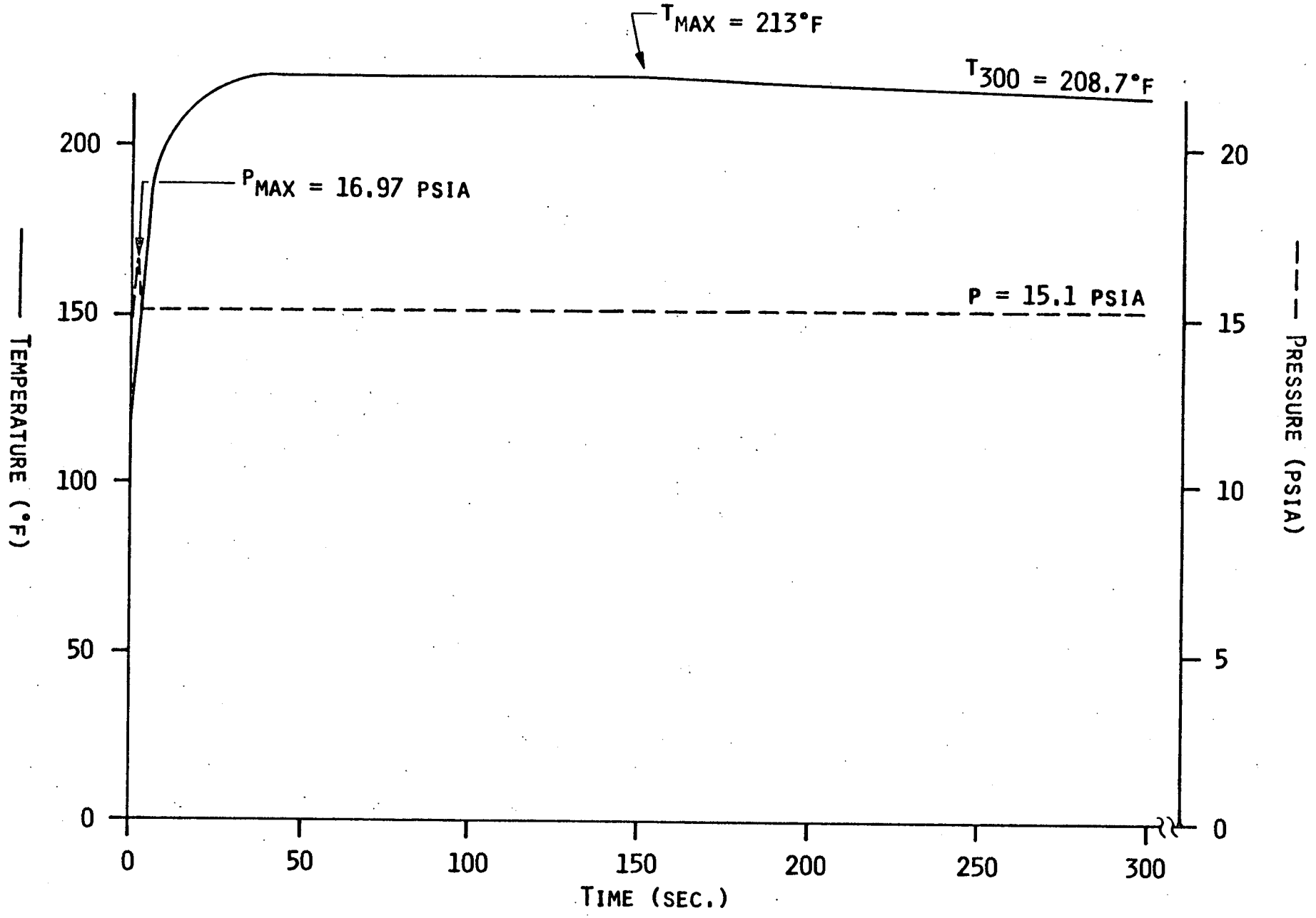


FIGURE B. 11 - RWCU LINE BREAK IN RWCU PUMP ROOM
TEMPERATURE AND PRESSURE TRANSIENT FOR RWCU PUMP ROOM
EL. 962'-6"

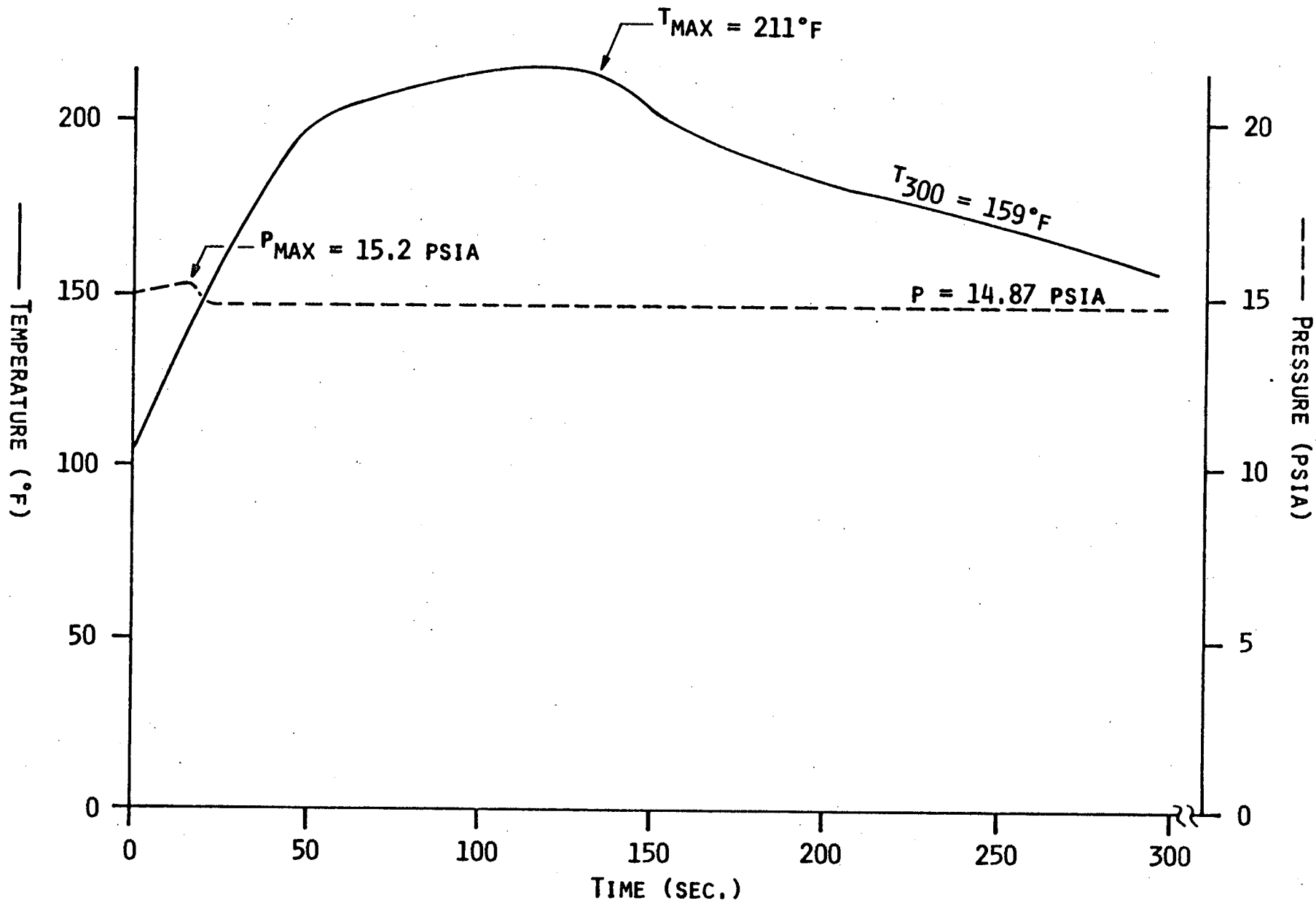


FIGURE B.12 - RWCU LINE BREAK IN WEST SIDE REACTOR BLDG. TEMPERATURE AND PRESSURE TRANSIENT FOR EAST SIDE REACTOR BLDG. EL. 962'-6"

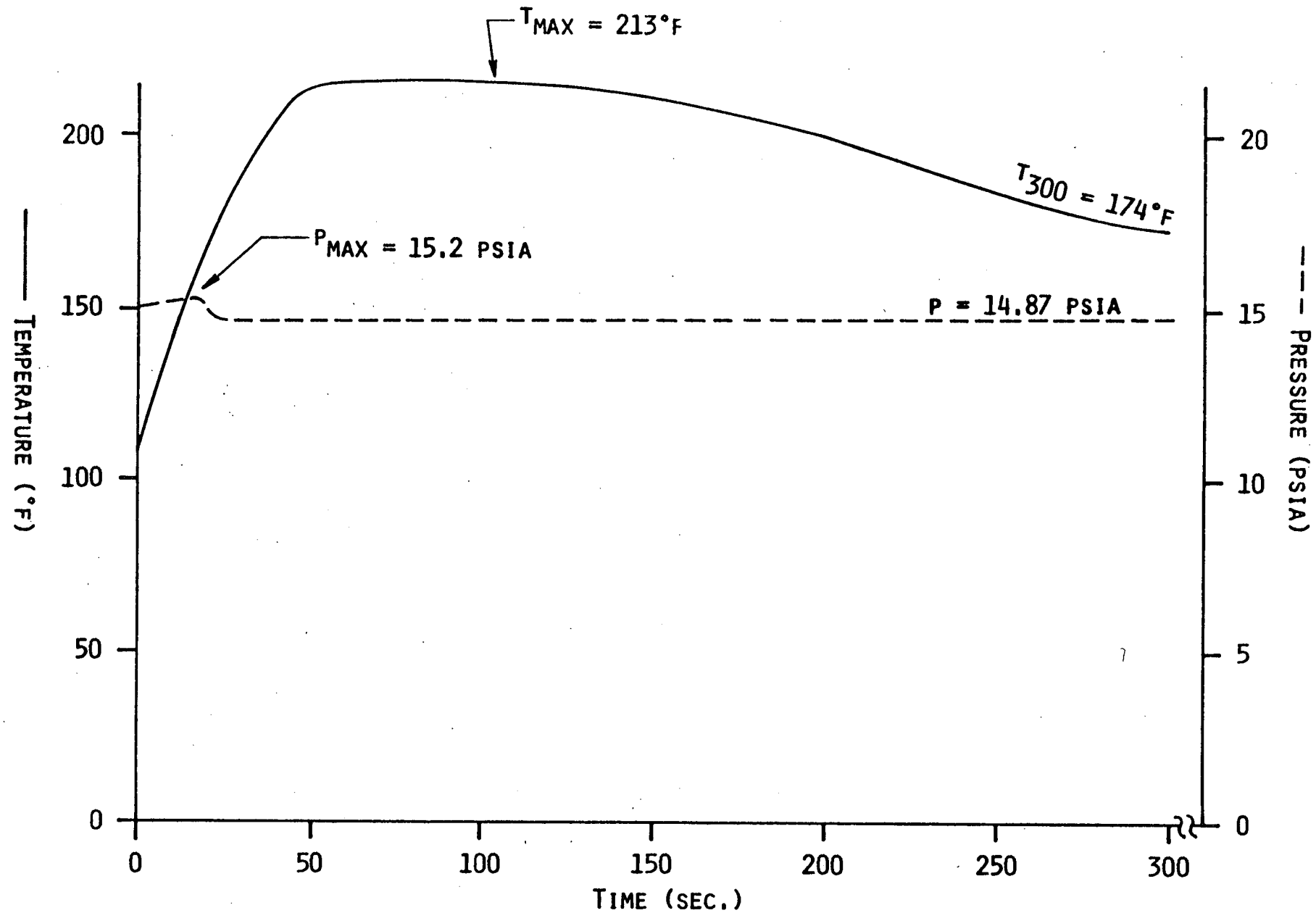


FIGURE B.13 - RWCU LINE BREAK IN WEST SIDE REACTOR BLDG. TEMPERATURE AND PRESSURE TRANSIENT FOR SOUTH SIDE REACTOR BLDG. EL. 962'-6"

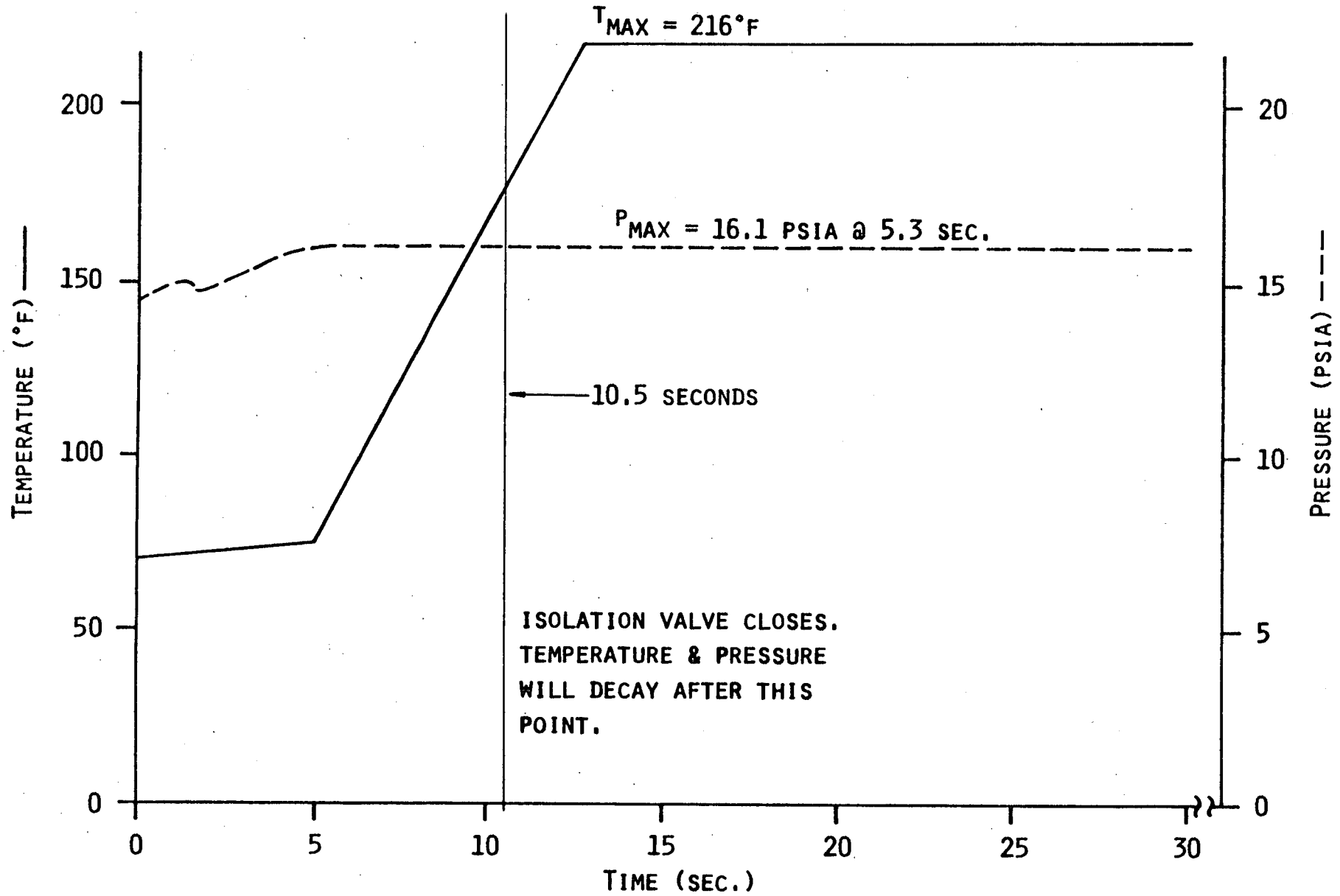


FIGURE B.14 - MAIN STEAM LINE BREAK IN TURBINE BLDG.
TEMPERATURE AND PRESSURE TRANSIENT FOR TURBINE BLDG.

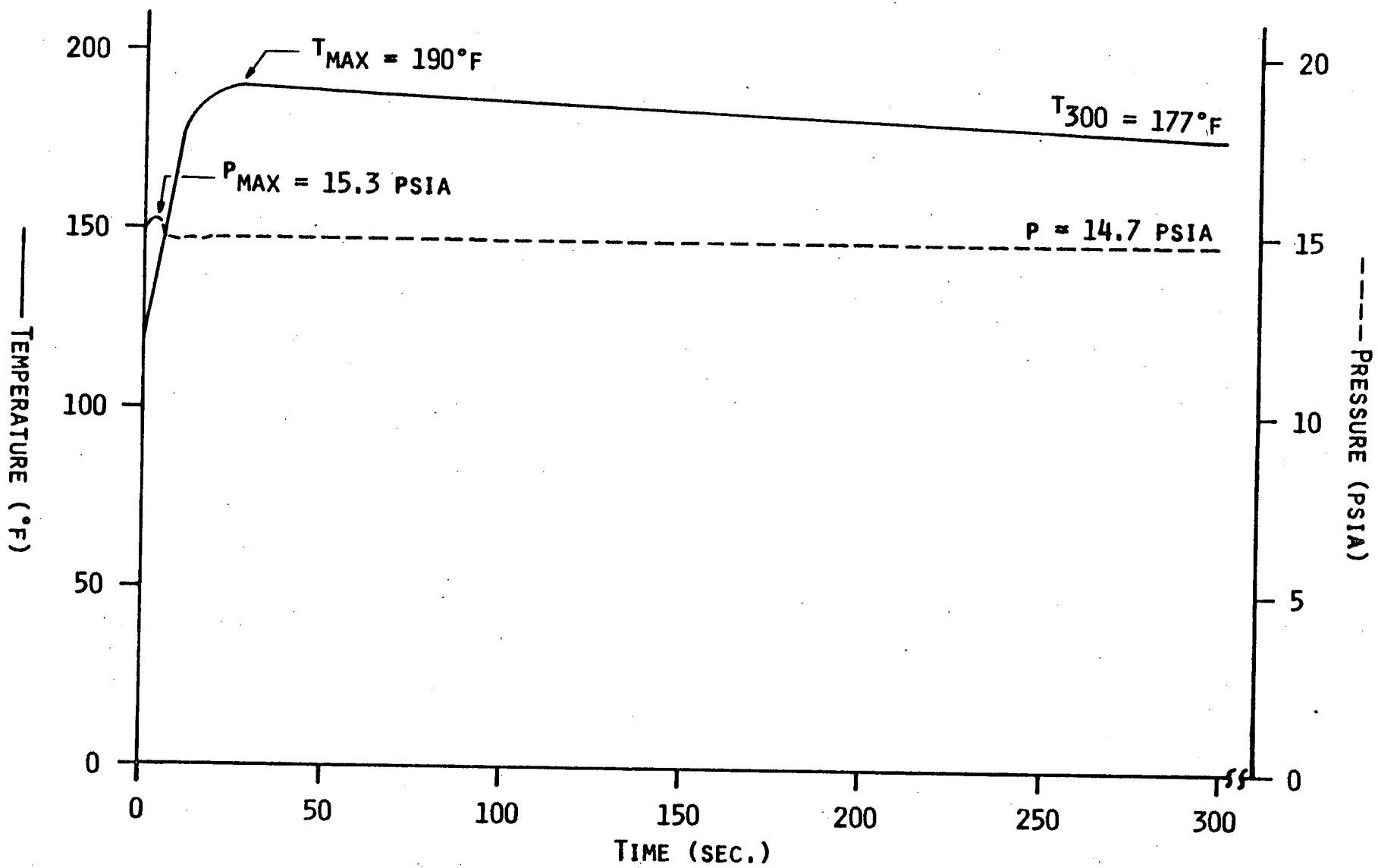


FIGURE B.15 - MAIN STEAM LINE BREAK IN MAIN STEAM TUNNEL
TEMPERATURE AND PRESSURE TRANSIENT FOR SOUTH-WEST CORNER REACTOR BLDG.
EL. 935'-0"

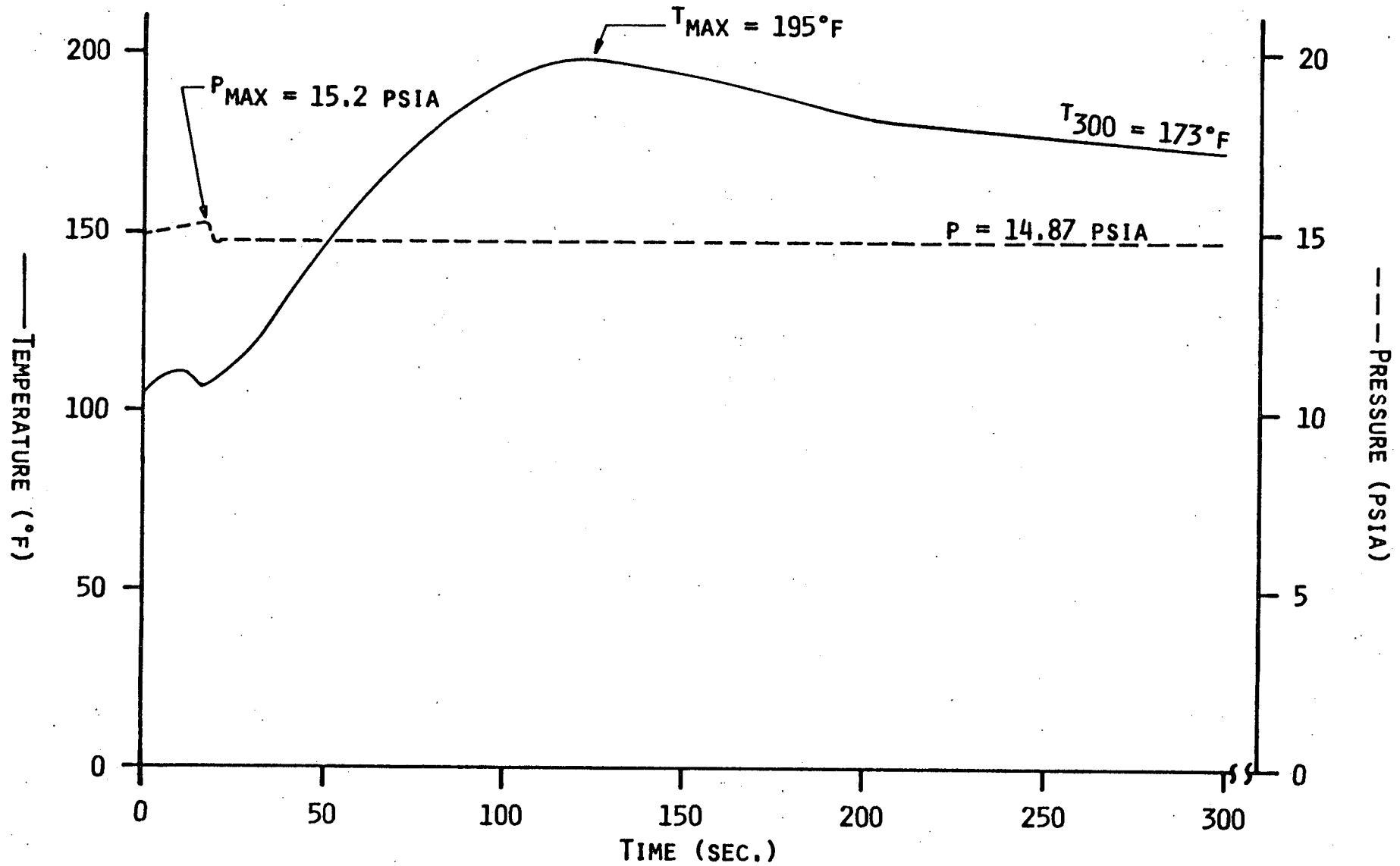


FIGURE B.16 - RWCU LINE BREAK IN WEST SIDE REACOTR BLDG.
TEMPERATURE AND PRESSURE TRANSIENT FOR SOUTH-EAST CORNER REACTOR BLDG.
EL. 935'-0"

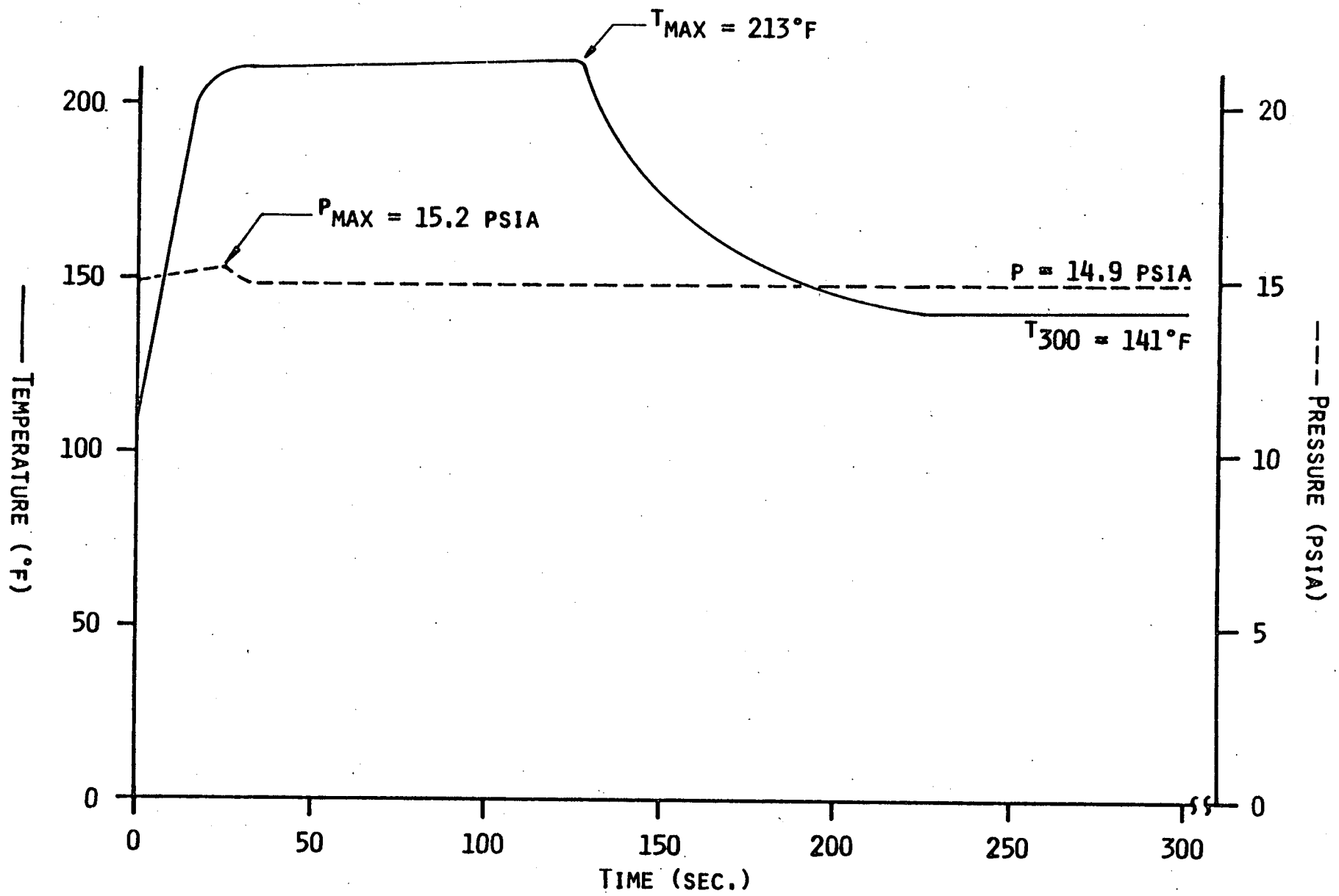
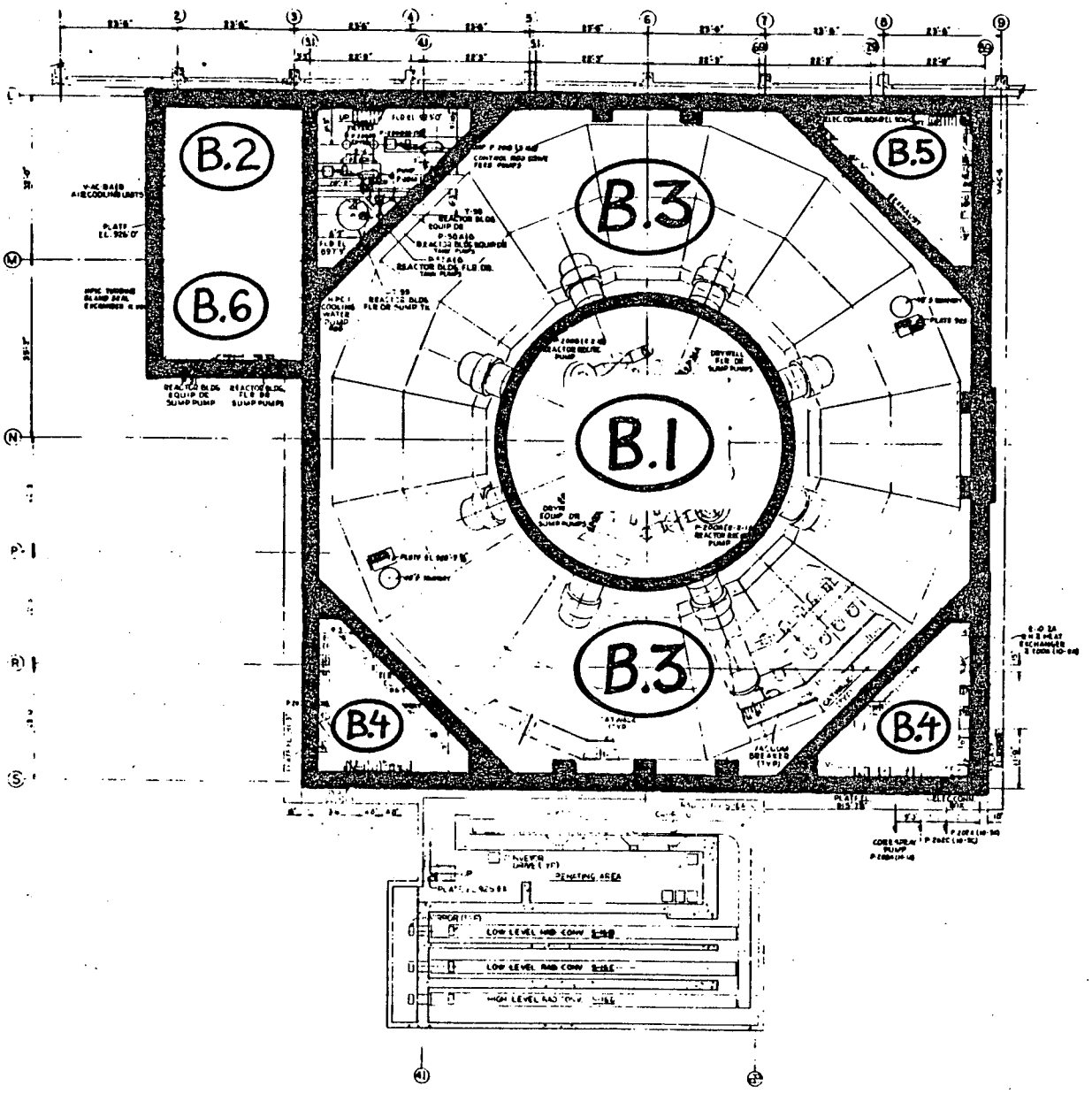


FIGURE B. 17 - RWCU LINE BREAK IN WEST SIDE REACTOR BLDG.
TEMPERATURE AND PRESSURE TRANSIENT FOR NORTH-WEST CORNER REACTOR BLDG.
EL. 962'-6"



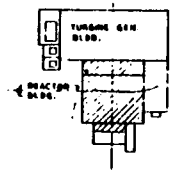


Figure B.18 - Reactor Bldg. Plans at El. 896 Ft-3 In., 911 Ft-6 In., and 922 Ft-6 In.

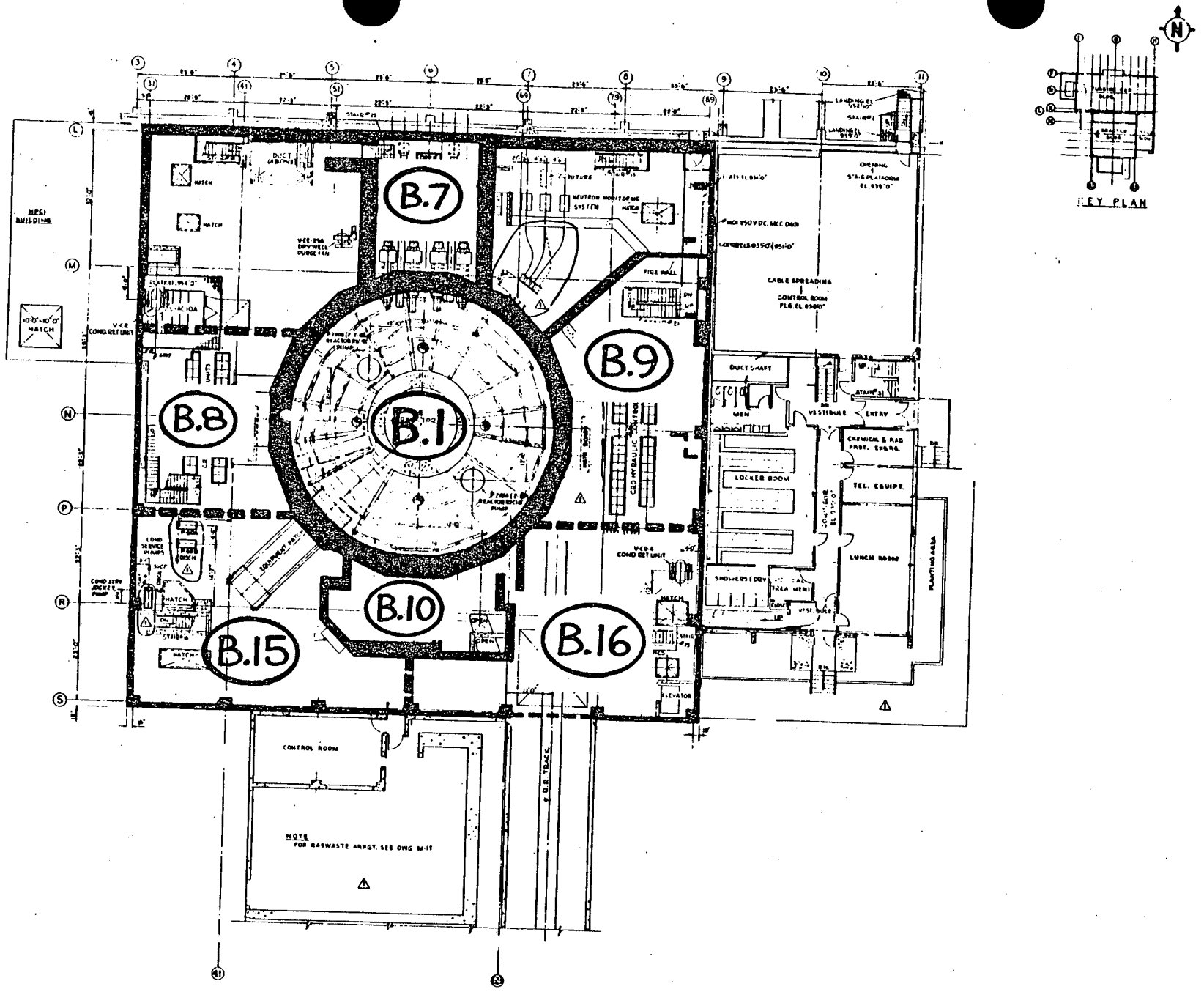
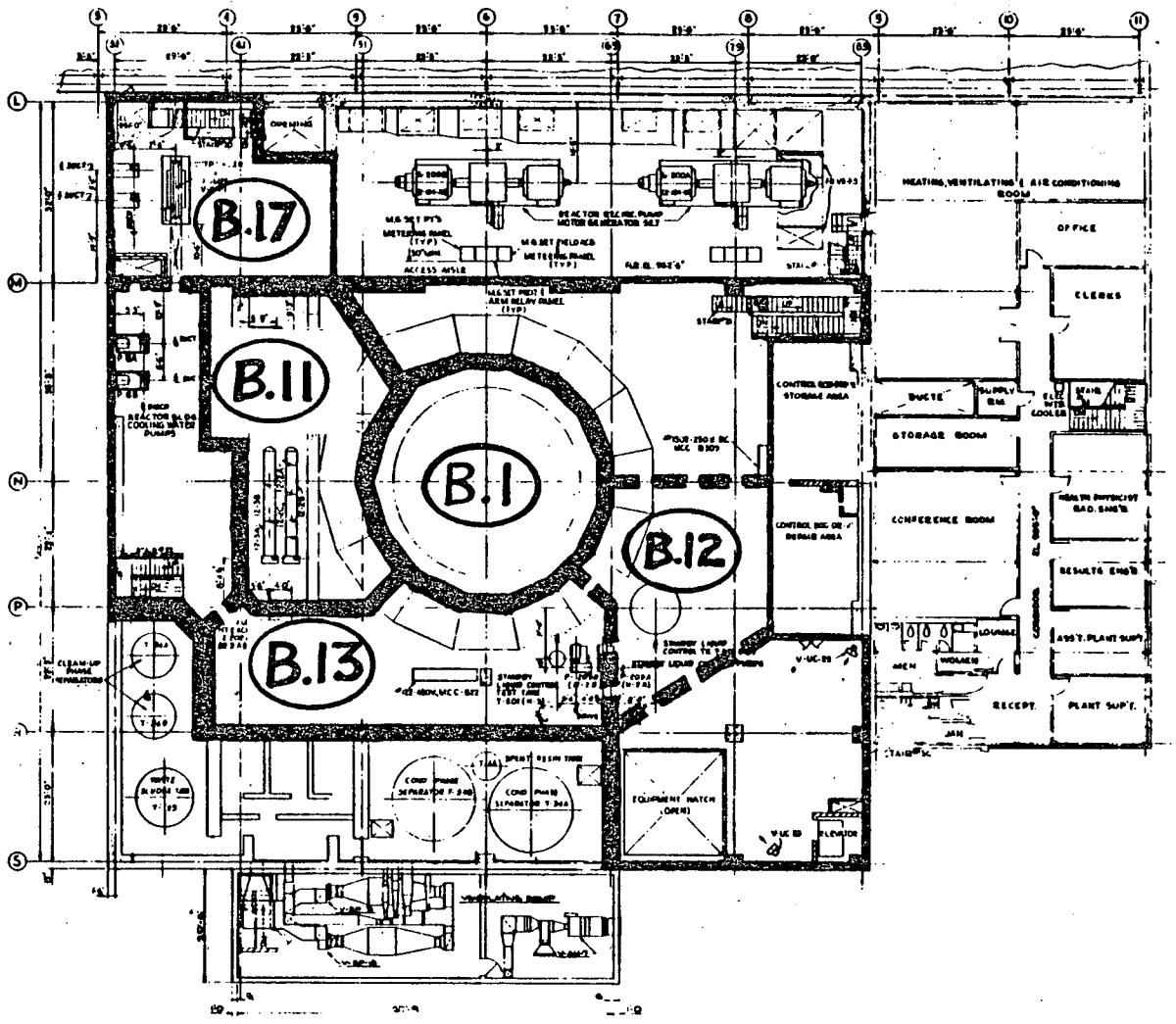


Figure B.19 - Reactor Bldg. Plans at El. 935 Ft-0 In.



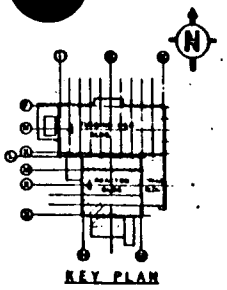


Figure B. 20 - Reactor Bldg. Plan at El. 962 Ft-6 In.

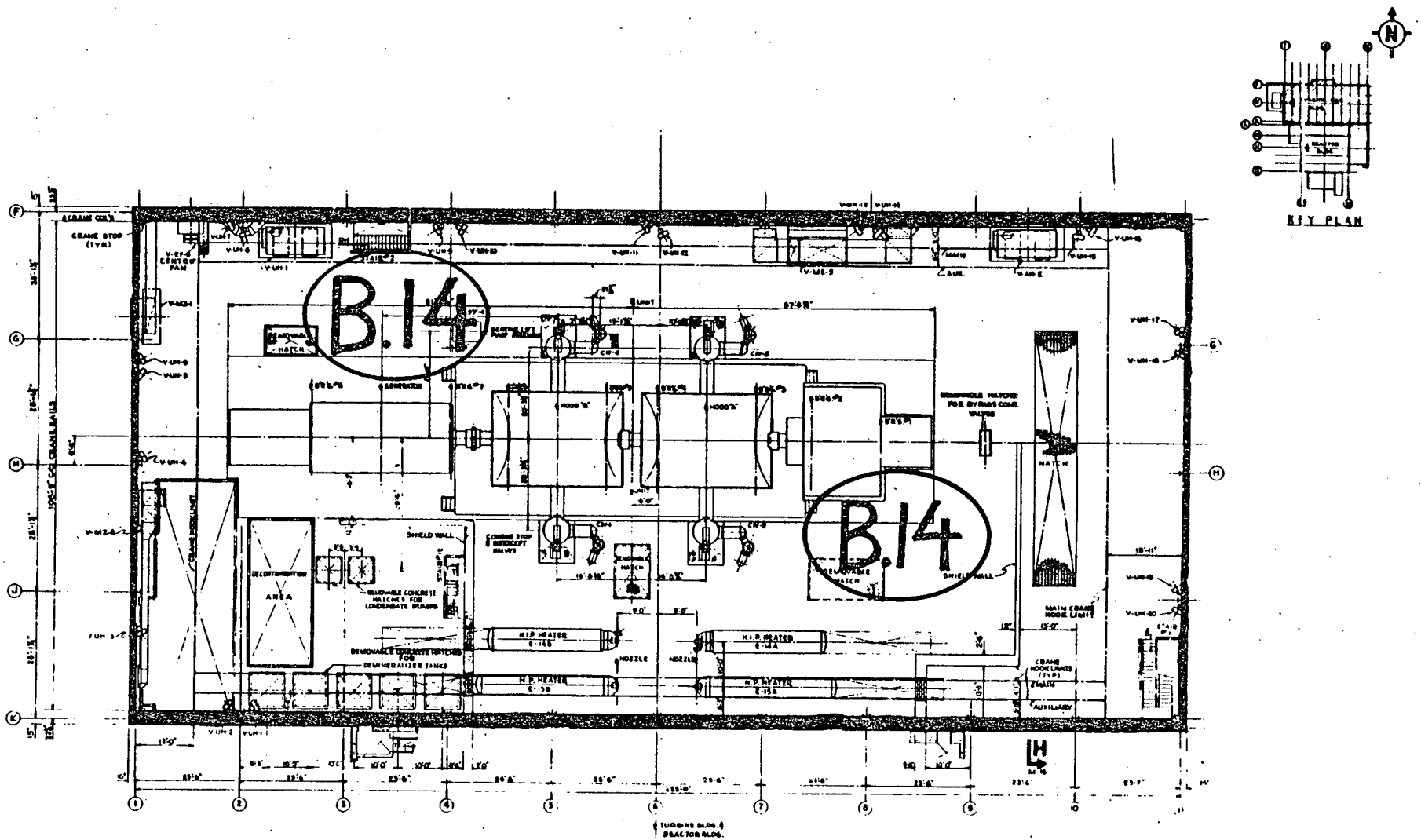


Figure B.21 - Turbine Building Operating Floor Plan at El. 931 Ft-0 In.

APPENDIX C

SYSTEM COMPONENT EVALUATION

WORKSHEETS

Component Evaluation Worksheets are currently being typed and will be included in a subsequent response.