Public Service Company of Colorado

12015 East 46th Avenue, Suite 440; Denver, CO 80239

June 17, 1981
Fort St. Vrain
Unit No. 1
P-81168

Mr. George Kuzmycz, Project Manager
Special Projects Branch
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket No. 50-267

Subject: Environmental Qualification of Class 1E Equipment IE Bulletin 79-01B

Reference: P-80350, 10/3/80

Dear Mr. Kuzmycz:

The following information is being submitted to assist you in the review of PSC's responses to IE Bulletin 79-01B:

1. Summary of FSAR and Other Documentation Pertaining to High Energy Line Breaks and Design Basis Accidents 1 and 2.

   A. Design Basis Accident Number 1

      "Permanent Loss of Forced Circulation (LOFC)"

      The analysis of this accident is contained in section 14.10 and Appendix D of the FSAR.

   B. Design Basis Accident Number 2

      "Rapid Depressurization/Blowdown"

      The analysis of this accident is in section 14.11 of the FSAR.

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C. Response to DRL Question 6.1

(Supplied as Attachment 1 to this letter)

D. Qualification of Fort St. Vrain Safe Shutdown Equipment for Steam Environment Resulting from Pipe Ruptures

Gulf General Atomic Report - GA-A12405 published May 30, 1972.

E. Evaluation of the Consequences of Postulated Pipe Failures Outside of the Reactor Building for Fort St. Vrain Unit No. 1

This was provided as Attachment C to Amendment 26 of the FSAR.

2. Summary of DBA1 and DBA2 and Environments

The opening discussion pertaining to DBA1 and DBA2 are quoted from the FSAR and supplied as Attachment 2 to this letter.

Temperature Environments

Comparison of the DBA and steam line rupture temperature transients are provided by Figure 6.1-1 of Attachment 1 to this letter. It can be seen from this figure that the Steam Line Rupture Temperature environment envelopes the DBA Temperature Environment.

Steam line rupture temperature transient curves for distances of 20 feet and greater from the leak source are provided by Figures 6.1-2 and 6.1-3 of Attachment 1 to the letter.

Steam Line Rupture Temperature Transient curves for distances of less than 20 feet were provided by figures 3 and 4 of Attachment D to our letter P-80350 dated October 3, 1980. These curves are provided as Attachment 3 to this letter for ease of reference.

Radiation Environments

The following is excerpted from pages 7 and 8 of our letter P-80350.

"Radiation:

There are no radiological concerns directly associated with a high energy line break at FSV. That is, the process fluid (steam or feed-water is not contaminated. To postulate a radiological incident DBA #1 "Permanent Loss of Forced Circulation" and DBA #2 "Rapid Depressurization/Blowdown" were considered. DBA #1 provides the worst case radiological conditions, but the overall radiological concerns are minimal.

Complete details of this accident may be found in Section 2.1.6.b (Attachment C) of P-79312 (Swart to Varga) dated December 1979.

In summary, the peak doses in the Reactor Building following DBA #1 are as follows:

Location	Peak Gamma Cose Rate	Time of Peak	180 Day Accumulated Dose (REM)
Reactor Building	1.4R/hr	24 hours	400

In conclusion, the reactor building will be accessible for short-term operations following such an accident. The accumulated doses indicated above would have no operational effect on the Reactor Building equipment."

As indicated during our phone call of June 9, 1981, the above doses represent the worst case in the reactor building. The 400 REM dose includes the contribution from the gas borne activity in the reactor building along with the contribution from the activity of the gas in the refueling penetrations.

Equipment at other locations in the reactor building would be subject to radiation fields from gas borne activity only. Therefore, the resultant accumulated dose would be less than 400 REM.

3. How to Determine Temperature Transient for Each Equipment Item.

   The following fields from Enclosure 3 (P-80350, October 3, 1980) define the time temperature qualifications for components.

Field	Description
LOC (35)	Defines general environmental location. That is, either turbine building (TB2) or reactor building (RX2) steam line break environments
TEST-DIST(10)	Defines the time-temperature curve utilized for qualification of equipment

NOTE: Components at a distance of 20 feet or greater from a steam line were qualified to the 20 foot curve. Items between 15 and 20 feet were qualified to the 15 foot curve. The same logic holds for the remaining curves.

4. Basis For Not Qualifying Equipment to Category 1 of NUREG-0588

As indicated during our phone call of June 9, 1981, NUREG-0588 was written for LWR technology and postulated accidents.

It is PSC's position that the only items in NUREG-0588 that apply to the Fort St. Vrain HTGR are the environmental conditions outside of containment. It does not seem prudent to commit to a document in which the vast majority of the requirements do not apply to Fort St. Vrain.

PSC is not taking exception to qualifying equipment environmentally, we are stating that it is our intent to continue qualifying equipment to the accident environmental conditions applicable to Fort St. Vrain. These accident conditions and our environmental qualification programs have been presented to, and reviewed by, the NRC Staff many times. It is PSC's position that thepresent program is adequate for providing environmentally qualified equipment.

Very truly yours,

Frederic E. Swart

Frederic E. Swart
Nuclear Project Manager

FES/MEN:pa

Attachments

6.0 ENGINEERED SAFETY FEATURES

6.1 Question: Identify all Reactor Protection and Engineered Safety Feature equipment and components (e.g. motors, switchgear, cables, filters, pump, seal) located in the reactor or turbine buildings which are required to be operable during and subsequent to a loss-of-coolant (depressurization), feedwater line break, or a steam line break accident. Describe the qualification tests which have been or will be performed on each of these items to ensure their availability in the resulting environments (i.e. helium, temperature, pressure, humidity).

Answer: Environmental steam tests have been performed to verify that all Safe Shutdown Cooling Equipment located in the reactor and turbine buildings will operate during and following a design basis accident (DBA), feedwater line break or steam pipe rupture.

A listing of the Safe Shutdown Cooling Equipment is presented in Table 6.1-1. The test requirements and description of the test facility, as well as the complete results of the environmental steam tests are presented in Gulf General Atomic Report GULF-GA-A12045, "Qualification of Fort St. Vrain Safe Shutdown Equipment for Steam Environment Resulting from Pipe Ruptures," dated May 1972. The results of the tests show that all components tested have been qualified either unprotected or protected. The components which required some protection or minor design change in order to survive the temperature transient following a steam pipe rupture accident are listed in Table 6.1-2. A more detailed description of the required changes is given in GULF-GA-A12045.
25 With the planned incorporation of these changes, it is concluded that the environmental steam test program has successfully demonstrated that the Safe Shutdown Cooling Equipment has sufficient high-temperature endurance to survive any anticipated steam pipe rupture accident in the reactor or turbine buildings.

The temperature transients used for the environmental steam tests were analytically determined. The results of that analysis led to the conclusion that a cold reheat steam pipe rupture in the reactor building and a hot reheat steam pipe rupture in the turbine building are the worst postulated accident conditions. Although the reactor building atmosphere temperature is considerably higher for a DBA (rapid depressurization of the primary cooling system) than for a steam leak accident, the reverse is true for the surface temperatures of the components. The reason is that the heat transfer coefficient for steam is much larger than for helium. Because the surface temperature of a component and not the building atmosphere temperature determines the temperature endurance limit of the component, the cold reheat steam pipe rupture is representative of the worst case and was selected for the environmental test program. The results of the comparative analysis between the temperatures of a DBA and a cold reheat steam pipe rupture are shown in Figure 6.1-1.

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25 CONTEMPT-G computer code was used to analyze the reactor and turbine building atmosphere response following a steam pipe rupture accident. This is a modified version of the original CONTEMPT code which has been adapted for high-temperature gas-cooled reactors. For the analysis the building pressure was assumed constant because the building pressure relieves through the louvers which open at three inches water gage pressure. The results of the analysis are shown by a family of curves in Figure 6.1-2 and 6.1-3 for the reactor and turbine buildings, respectively. The complete analysis and the rationale for selecting these two sets of curves for the environmental steam tests are given in the referenced report. Each curve represents a temperature transient for a specific distance from the postulated steam leak. In both the reactor and turbine buildings the minimum distance from the steam leak of any Safe Shutdown component is approximately 20 ft.

Table 6.1-1

SAFE S[illegible]TDOWN COOLING EQUIPMENT LIST

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System or Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Buffer He System	Two loops-equipment for only one loop noted	Buffer He recirculator bypass valve	PDIC-2143 PDV-2143	Reactor Bldg Reactor Bldg
		Recirculator inlet block valve	HV-2193 HS-2193	Reactor Bldg Control Room
		Recirculator inlet block valve	HV-21213 HS-21213	Reactor Bldg Control Room
Bearing Water System	Two loops-equipment for only one loop noted	Bearing water pumps	P-2101 P-2101S P-2106	Reactor Bldg Reactor Bldg Reactor Bldg
		Surge tank emergency drain control	LC-21245 LV-21245	Reactor Bldg Reactor Bldg
		Normal surge tank level controls	LC-2135 LV-2135-1&2	Reactor Bldg Reactor Bldg
		Surge tank level switch & isolation valve	LSL-2137 LV-2137	Reactor Bldg Reactor Bldg
		Bearing water pump switches	HS-2131-1,2&3	Control Room
		Bearing water make-up pumps & switches	P-2105 P-2108 HS-21331 HS-21394	Reactor Bldg Reactor Bldg Control Room Reactor Bldg
		Low flow bypass bearing water make-up pump	FS-21333 FV-21333	Reactor Bldg Reactor Bldg
		Low flow bypass (bearing water pumps)	FSL-21297 FV-21297	Reactor Bldg Reactor Bldg
Turbine Water Removal		Turbine water removal low-pressure separator backup drain controls (to turbine water drain tank)	LC-21119 LV-21119	Reactor Bldg Reactor Bldg
		Drain controls (to reactor building sump)	LSH-21118 LV-21118 LT-21118	Reactor Bldg Reactor Bldg

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Table 6.1-1 (continued)

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System or Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Turbine Water Removal (continued)		Pressure relief turbine water drain header	PS-21120 PV-21120	Reactor Bldg Reactor Bldg
		Turbine water drain tank equalization	HV-21277-1,2&3	Reactor Bldg
		Standby pump start (turbine water removal pump)	LSH-21132	Reactor Bldg
		Normal pump start (turbine water removal pump)	LT-21129 LSH-21129	Reactor Bldg Reactor Bldg
		Normal turbine water removal tank level controls	LC-21114 LV-21114	Reactor Bldg Reactor Bldg
		Backup turbine water removal tank level controls.	LS-21130 LV-21130	Reactor Bldg Reactor Bldg
		Turbine water removal pumps & switches	P-2103,35 HS-21111 HS-21112	Reactor Bldg Control Room Control Room
25		Equalization line valve to turbine water drain tank	HV-21277-4,5, 6&7	Reactor Bldg
Circulator bearing water control	Four circulators -equipment for only one circulator noted-typical of four circulators	Bearing water supply isolation	FS-2183	Reactor Bldg
		Main drain control	PDT-2175-1 PDC-2175 PDV-2175	Reactor Bldg
		Steam/water drain control	PDT-2179-1 PDC-2179 PDV-2179 PM-2179	Reactor Bldg

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Table 6.1-1 (continued)

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System or Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Circulator Water Turbine	Four circulators -equipment for only one circulator noted-typical of four circulators	Circulator	C-2101	Reactor Bldg
		Circulator water turbine block valves	HS-2109 HV-2109-1,2	Reactor Bldg Reactor Bldg
		Water turbine control valve	SC-2109 HC-2109 SV-2109	Control Room Reactor Bldg Reactor Bldg
		High-pressure separator level control	LC-21303 LV-21303	Reactor Bldg Reactor Bldg
		Circulator service system isolation valves, circulator brake and static seal	HS-2187-1 thru -6&-8 HS-21192, & -21203 HS-2187-1 thru -9 HV-21192-1&-2 -21203-1&-2	Control Room Control Room Reactor Bldg Reactor Bldg.
		Emergency condensate admission valve to circulator water turbine	HS-21257 HV-21257	Reactor Bldg Reactor Bldg
		Plant Protective System (PPS) circulator trip inputs	PDIS-21149 PDIS-21151 PDIS-21153 PDIS-21173 PDIS-21175 PDIS-21177 PDIS-21319 PDIS-21321 PDIS-21323	Reactor Bldg Reactor Bldg Reactor Bldg Reactor Bldg Reactor Bldg Reactor Bldg Reactor Bldg Reactor Bldg Reactor Bldg
Emergency Condensate Header	Two loops-equipment for only one loop noted	Emergency condensate header admission valve to steam generator	HS-2237 HV-2237	Control Room Turbine Bldg
Feedwater Control	Two loops-equipment for only one loop noted	Feedwater measurement control	FT-2205 FM-2205-1 FC-2205 FM-2205-2 FM-2205-7 FV-2205	Turbine Bldg Control Room Control Room Control Room Control Room Turbine Bldg

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Table 6.1-1 (continued)

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System or Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Steam Generator Control	Two loops-equipment for only one loop noted	Superheater temperature indicator	TE-2225-1 to-6	Reactor Bldg
			TT-2225-1 to-6	Control Room
			TM-2225-1, -2	Control Room
			TI-2225	Control Room
		Steam generator pressure control	PT-22129	Turbine Bldg
			PC-2229	Control Room
			PV-2229	Turbine Bldg
			PT-22129-1	Turbine Bldg
			PI-22129-1	Control Room
			PM-2229	Control Room
			PI-22129	Control Room
		Bypass block valve	HV-2293	Turbine Bldg
			HS-2293	Control Room
		Superheater outlet stop check valve	HV-2223	Turbine Bldg
			HS-2223	Control Room
Emergency Condensate	Two loops-equipment for only one loop noted	Emergency condensate flow control	FT-2239	Reactor Bldg
			FM-2239	Control Room
			FC-2239	Control Room
			FV-2239	Reactor Bldg
		Emergency condensate block valve	HS-2291	Control Room
			HV-2291	Reactor Bldg
Steam Generator Control (Reheater)		Cold reheat isolation (inlet)	HS-2249	Control Room
			HV-2249	Reactor Bldg
		Cold reheat isolation (inlet)	HS-2251	Control Room
			HV-2251	Reactor Bldg
		Cold reheat isolation (inlet)	PV-2243	Reactor Bldg
			PM-2243-1	Control Room
			PC-2243	Control Room
Steam Generator Control (Reheater)	Two loops-equipment for only one loop noted	Reheat bypass pressure control	PT-2267	Turbine Bldg
			PC-2267	Control Room
			PV-2267	Turbine Bldg
Steam Generator Control (Reheater)	Two loops-equipment for only one loop noted	Reheater bypass pressure control	HV-22131	Turbine Bldg
			HS-22131	Control Room
		Reheater (outlet) stop check valve	HV-2253	Turbine Bldg
			HS-2253	Control Room

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Table 6.1-1 (continued)

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System of Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Reactor Plant Cooling Water		PCRV Cooling Pumps	P-4601,1S,2,2S	Reactor Bldg
		Cooling water outlet temperature	TE-4637-1,38-1	Reactor Bldg
			TE-4637-3,38-3	Reactor Bldg
			TT-4637-1,38-1	Control Room
			TI-4637-1,38-1	Control Room
Hydraulic System		Hydraulic pumps	P-9101X, SX	Reactor Bldg
		Hydraulic pump switches	HS-9101-1	Control Room
			HS-9103-1	Control Room
Instrument Air		Instrument Air Compressors and aftercoolers	HS-8211-1,-2,-3	Control Room
			C8201	Turbine Bldg
			C8201S	Turbine Bldg
			C8203	Turbine Bldg
Fire Water		Fire water pump	P-4501	Fire Water Pump House
			HS-4504-2	Control Room
			PS-4504	Turbine Bldg
			LS-4504-1	Turbine Bldg
			LS-4504-2	Turbine Bldg
		Engine driven fire pump and controls	P-4501-S	Fire Water Pump House
		Control panel for engine driven pump	HS-4504-1	Control Room

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Table 6.1-1(continued)

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System or Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Diesel Generator Coolers		Fire water backup to service water for diesel generator coolers	PIC-4256 PCV-4256 PS-4266 PIC-4266 PCV-4266	Turbine Bldg Turbine Bldg Turbine Bldg Turbine Bldg Turbine Bldg
Diesel Generator Air Cooler		Fire water backup to instrument air coolers	None	
Circulating Water Makeup			Pumps A,B,C	Circulating Water Makeup Pump House
		Circulating water makeup pumps & switches	HS-4102-1 HS-4102-2	Control Room Control Room
Service Water		Service Water pumps and controls	P-4201,2,2S HS- 211-1,2,3	Service W. er Pump House Control Room
Service Water		Isolation to decay heat removal exchanger	HV-4225 HS-4225	Turbine Bldg Control Room
		Isolation non-essential turbine water service	HV-4257 HS-4257	Turbine Bldg Control Room
		Isolation to instrument air coolers	FSV-8211-1 FSV-8211-2 FSV-8211-3	Turbine Bldg Turbine Bldg Turbine Bldg
		Temperature control valve for instrument air compressors	TCV, TV, TE-4234, 4235, 4274	Turbine Bldg Turbine Bldg
		Diesel generator coolers temperature instrumentation and control	TIC-4266 TIC-4267 TIC-4269 TIC-4270	Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms
		Diesel generator coolers temperature control valve	TCV-4266 TCV-4267 TCV-4269 TCV-4270	Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms

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Table 6.1-1(continued)

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System or Function	Notes	Instrumentation or Equipment	Instrumentation or Equipment Number	Location
Service Water (continued)		Diesel generator coolers temperature outlet sensors	TE-4266 TE-4267 TE-4269 TE-4270	Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms
		Isolation from emergency circulating water supply	HS-4221 HV-4221-2	Control Room Turbine Bldg
Standby Generator Oil & Air		Diesel oil transfer pumps and controls	P-9201X,SX HS-9299 HS-92100	Diesel Gen. Rooms Diesel Gen. Rooms Diesel Gen. Rooms
		Diesel generator air compressors and controls	C-9201,1S,2,2S HS-92101-1,-2,02-1,-2	Diesel Gen. Rooms Diesel Gen. Rooms
		Auxiliary boiler fuel oil feed pumps and switches	P-8404X P-8404SX HS-8427 HS-8428	Turbine Bldg Turbine Bldg Turbine Bldg Turbine Bldg

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Table 6.1-2

SUMMARY OF CORRECTIVE ACTION PLANNED FOR SAFE SHUTDOWN COOLING EQUIPMENT TO WITHSTAND STEAM PIPE RUPTURE ACCIDENT

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Test Tag No.	Component	Planned Corrective Action	Affected Component
PIC-4256	Pneumatic Pressure Controller	apply insulation	PIC-4256 PIC-4266
TCV-4234	Temperature Controlled Valve	none, because failure is in safe direction	TCV-4234 TCV-4235 TCV-4274
None	Class A31 A-C Combination Motor Starter	design modification (provide for remote by-pass of overload circuit if needed)	HV-2253 HV-2254 HV-22131 HV-22132 HV-4225 HV-4257
HV-2312	Motor Driven Valve Operator	design modification (short out overload element in valves - overload protection provided elsewhere)	HV-2290 HV-2291 HV-2253 HV-2254 HV-22131 HV-22132
LV-21114 TV-2227	Control Valve Actuator	design modification (remove pressure gage)	PV-2267 PV-2268

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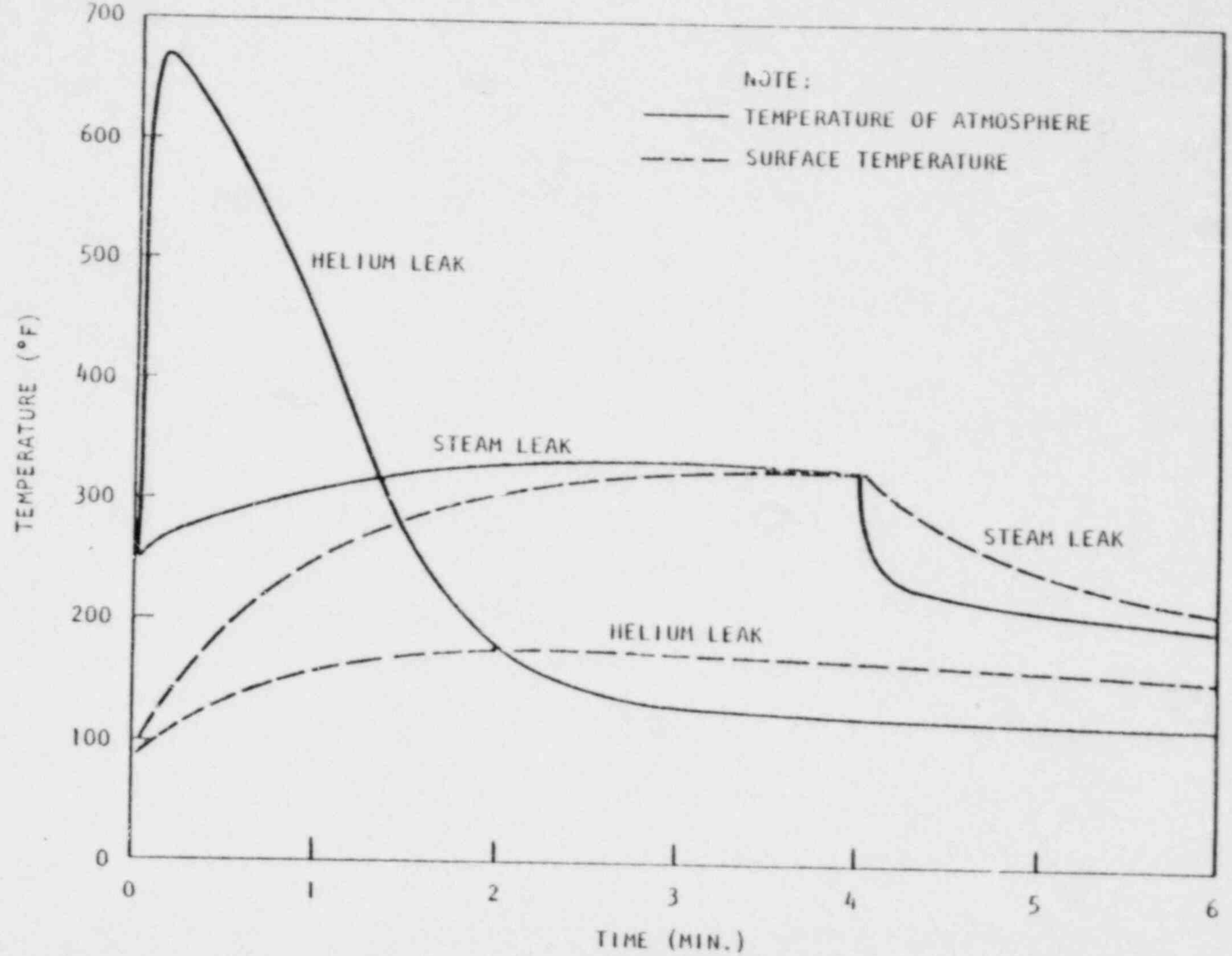

Fig. 6.1-1--Temperatures in the Fort St. Vrain reactor building following a Design Basis Accident or a cold reheat steam pipe rupture inside PCRV support ring at 30 ft from leak

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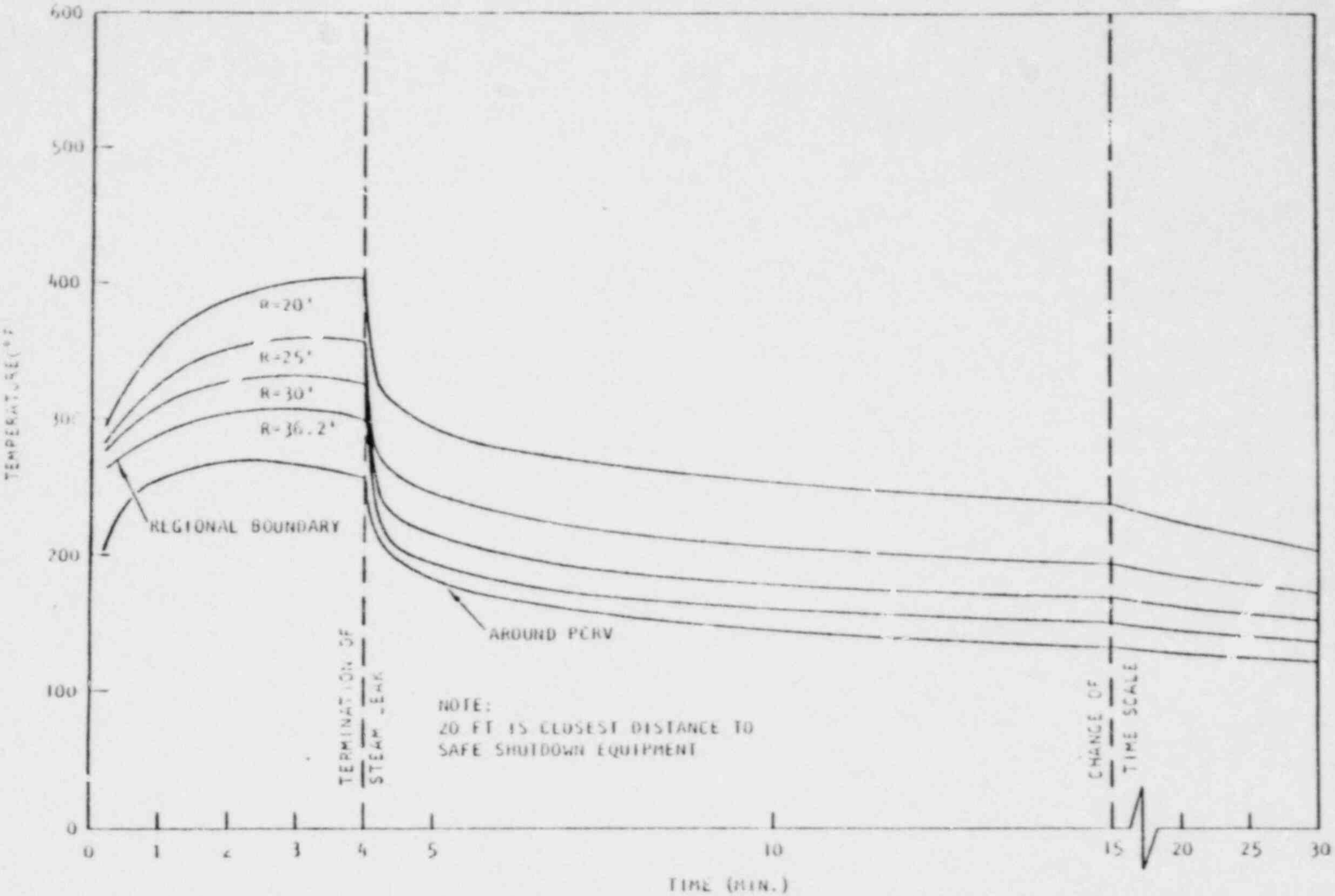

Fig. 6.1-2--Temperature of reactor building atmosphere during postulated cold reheat steam pipe rupture accident as a function of distance "R" from possible source of steam leak

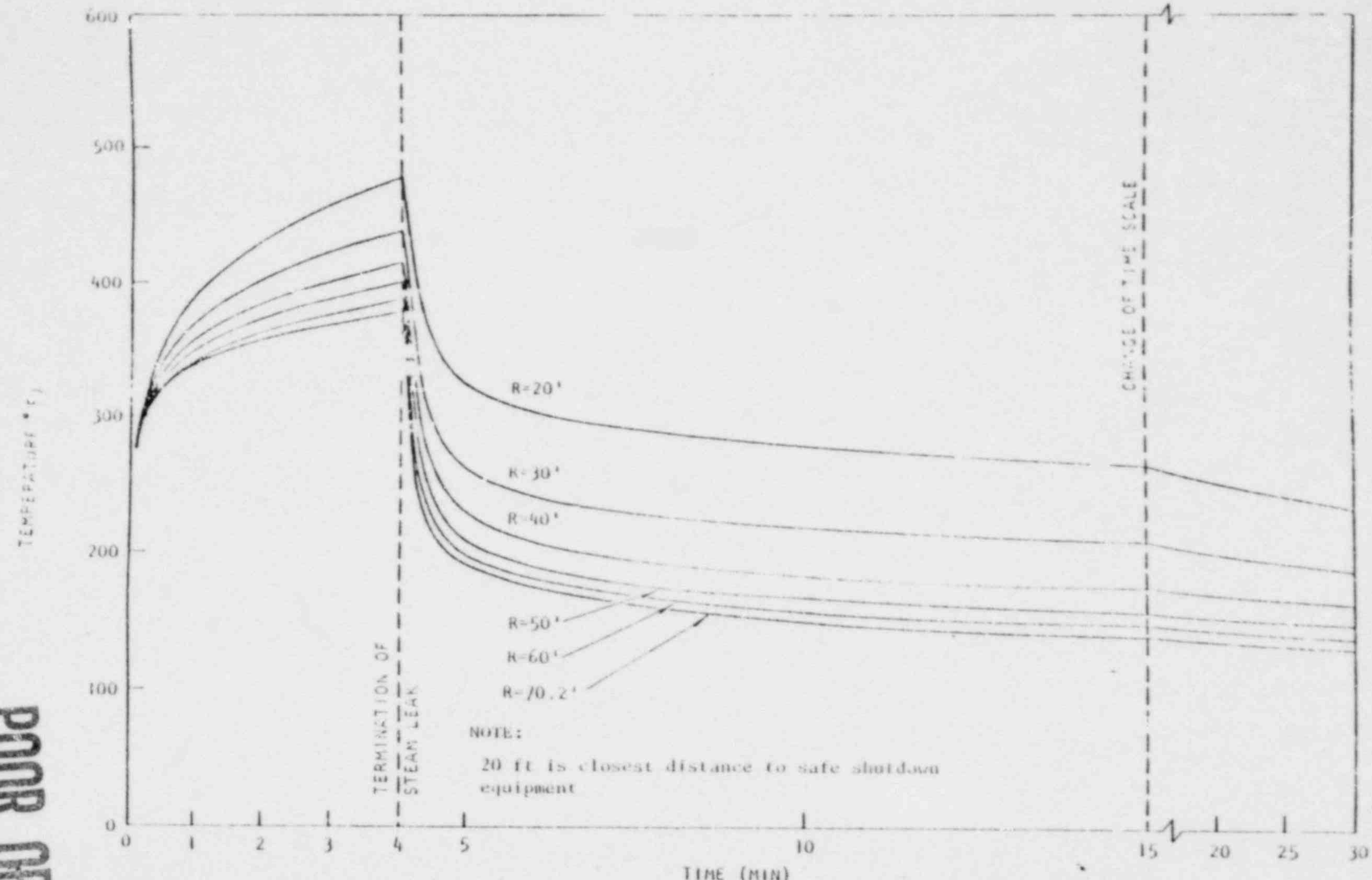

Fig. 6.1-3--Temperature of turbine building atmosphere during postulated hot reheat steam pipe rupture accident as a function of distance "R" from possible source of steam leak

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Amendment 25

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14.10. DESIGN BASIS ACCIDENT NO. 1 "PERMANENT LOSS OF FORCED CIRCULATION (LOFC)" (Power Level 879 Mw(t))

14.10.1 Introduction

A hypothetical permanent loss of forced circulation of primary coolant helium would require the extended failure of all four helium circulators, their steam and water drives or their multiple sources of motive power, or failure of both the main steam and reheat steam sections of both steam generators. This condition is an extension of the 30 min temporary loss of normal shutdown cooling accident described in Section 14.4.

The detailed description, consequences and supplementary information pertaining to this accident and supplementary accidents are given in Appendix D, detailed and supplementary information pertaining to the permanent loss of forced circulation (LOFC) for the Fort St. Vrain HTGR. The contents of Appendix D include:

D.1. Detailed Description of Design Basis Accident No. 1
D.2. Supplemental LOFC Accident Results
D.3. LOFC Accident Experimental Data and Analytical Methods

14.10.2. Conditions of the Accident

At the time of this hypothetical loss of forced circulation, the reactor would have scrammed, most probably on "two-loop-trouble" as defined in Table 7.1-3. Loss of forced circulation in one loop causes isolation of that loop while subsequent loss of circulation in the second loop constitutes two-loop trouble.

When it becomes apparent to the plant operator that the loss of forced circulation is permanent, e.g., after about 5 hr when resumption of cooling would cause steam generator damage (see Appendix D, Section D.2.5), the primary coolant system would be depressurized to storage (in a few hours) in the normal manner through the helium purification system as described in Section 9.4.3.5. The reserve shutdown system would be operated after this initial period to assure an adequate shutdown margin.

The PCRV cooling water system would continue in operation and would be closely monitored since its operation is vital to the PCRV integrity during the accident. This system is described in Section 9.7 as a Class I system connected to the essential electric ' bus. Two separate identical closed loops supply cooling water to three separate zones of the PCRV: the top head penetrations; the core support floor; PCRV liner on the side wall and top head; and the PCRV bottom head and bottom penetrations. Half capacity cooling (one of two identical loops operating) is assumed as the conservative limiting case.

The reactor plant ventilation system would continue to operate normally during the accident in order to provide filtration and elevated release for any fission product activity escaping from the PCRV during the course of the accident. This system is described in Section 6.1.3.2.

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No other reactor plant equipment is required to function during this accident. Continued operation of non-essential equipment, instruments and controls normally operating during reactor shutdown is assumed for purposes of monitoring plant conditions, but this equipment has no effect on the course or consequences of the accident.

The following operator actions have been determined to be either necessary or desirable to mitigate the consequences of this accident:

1. Normal post scram operations.
2. Actions required to re-establish helium circulation (assumed to be unsuccessful for this hypothetical accident).
3. Primary coolant system depressurization.
4. Operation of the reserve shutdown system.
5. Adjustment of the PCRV cooling system water flow rates and cover pressure to increase cooling ability in areas affected.

None of the above items require rapid operator response and thus these actions could be carried out in a logical and thoughtful manner.

14.10.3. Accident Results and Consequences

Summary. This accident involves both core damage and fission product release causing off-site doses. The core hot regions slowly heat up to about 5400°F, maximum, occurring after 83 hr. Approximately 95% of the fuel particles in the core will suffer failed coatings, resulting in release of about 28% of the core fission product inventory. Of this 28% of the inventory, less than 5% remains gas borne in the PCRV. This 5% is essentially all noble gas with a small amount of iodine.

Other than melting of the steel components of the control rod assembly and some local failure of the PCRV liner insulation, no other failures in the core or PCRV internals will occur. The core will remain subcritical during all periods of the accident due to control rod and reserve shutdown system poisons.

The doses resulting from this accident are orders of magnitude lower than the guidelines of 10 CFR 100. The total duration (6 month) doses at the low population zone boundary (16,000 meters) based on initial operation of the plant at 879 Mw(t), are listed below.

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Whole body gamma	0.37	mrem
Thyroid	36.	mrem
Bone	1.0	mrem

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14.11. DESIGN BASIS ACCIDENT NO. 2 "RAPID DEPRESSURIZATION/BLOWDOWN" (Power Level 879 Mw(t))

In this section the Design Basis Accident No. 2 (Rapid Depressurization) is analyzed and evaluated. This accident consists of a hypothetical sudden failure of both closures in a PCRV penetration so that the primary coolant system is rapidly depressurized, and any potential for air ingress is developed.

This accident was originally presented in the FSAR as a "Maximum Hypothetical Accident" to illustrate that the essentially instantaneous release of the reactor circulating fission product inventory would not exceed 10 CFR 100 limits. The doses are at least an order of magnitude below the 10 CFR 100 limits. It assumed that the coolant with its activity is allowed to escape directly from the building into the atmosphere at ground level without any credit for holdup or filtration by the ventilation system. The off-site doses resulting from this accident, based on "design" activities, are given in Table 14.11-1. These doses would be even lower if based on "expected" primary coolant activities.

Meteorological conditions assumed for the MHA are assumed to be 1 m/sec wind speed and stability condition G with the release initiating as an area source.

Table 14.11-1

DOSES FROM MAXIMUM HYPOTHETICAL ACCIDENT
(Power Level 879 Mw(t))

(Release of "Design" primary coolant gas-borne activity only)

	Total Duration Dose (rem)	
Type of Dose	At Exclusion Area Boundary	At Low Population Zone Boundary
Whole body gamma (WBG)	2.5	0.073
Thyroid	5.0	0.30
Bone	0.075	0.006

A mechanical basis was not considered to exist for the Maximum Hypothetical Accident, and therefore only the hypothetical radiological consequences were considered. However, at the request of the DRL the hypothetical sudden failure of both closures in any penetration was analyzed. The resultant primary system depressurization rates were analyzed mechanistically with respect to the free flow area presumed to develop in each penetration as limited by the "Flow Restrictor" engineered safeguards. The resulting analyses showed that even with failure of both closures in a penetration:

a. The integrity of the fuel and core configuration are not disturbed by any thrust forces developed by the primary coolant, and imposed upon reactor internals, during the depressurization.

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14.11-2

2. Adequate primary circuit cooling would be maintained following the depressurization by use of the circulators on either steam or water-turbine drive with the reduced coolant density. No damage to the circulators would occur during the depressurization.

3. The continuation of the normal return-flow of clean helium to the PCRV would almost totally exclude ingress of air into the PCRV. However, even if all return flow to the PCRV were eliminated, the air ingress would be insignificant from both a heat generation and graphite combustion viewpoint.

4. Although there are a number of actions which the operator could perform to further ensure or improve the safe shutdown of the plant there are no immediate or necessary actions which are required of him.

5. The pressure differentials and jet forces due to the depressurizing primary coolant will not damage or overpressurize any required components of the PCRV or the reactor building

6. The radiological consequences of the rapid depressurization, although increased in severity with respect to the Maximum Hypothetical Accident, would be well within the limits prescribed by 10 CFR 100.

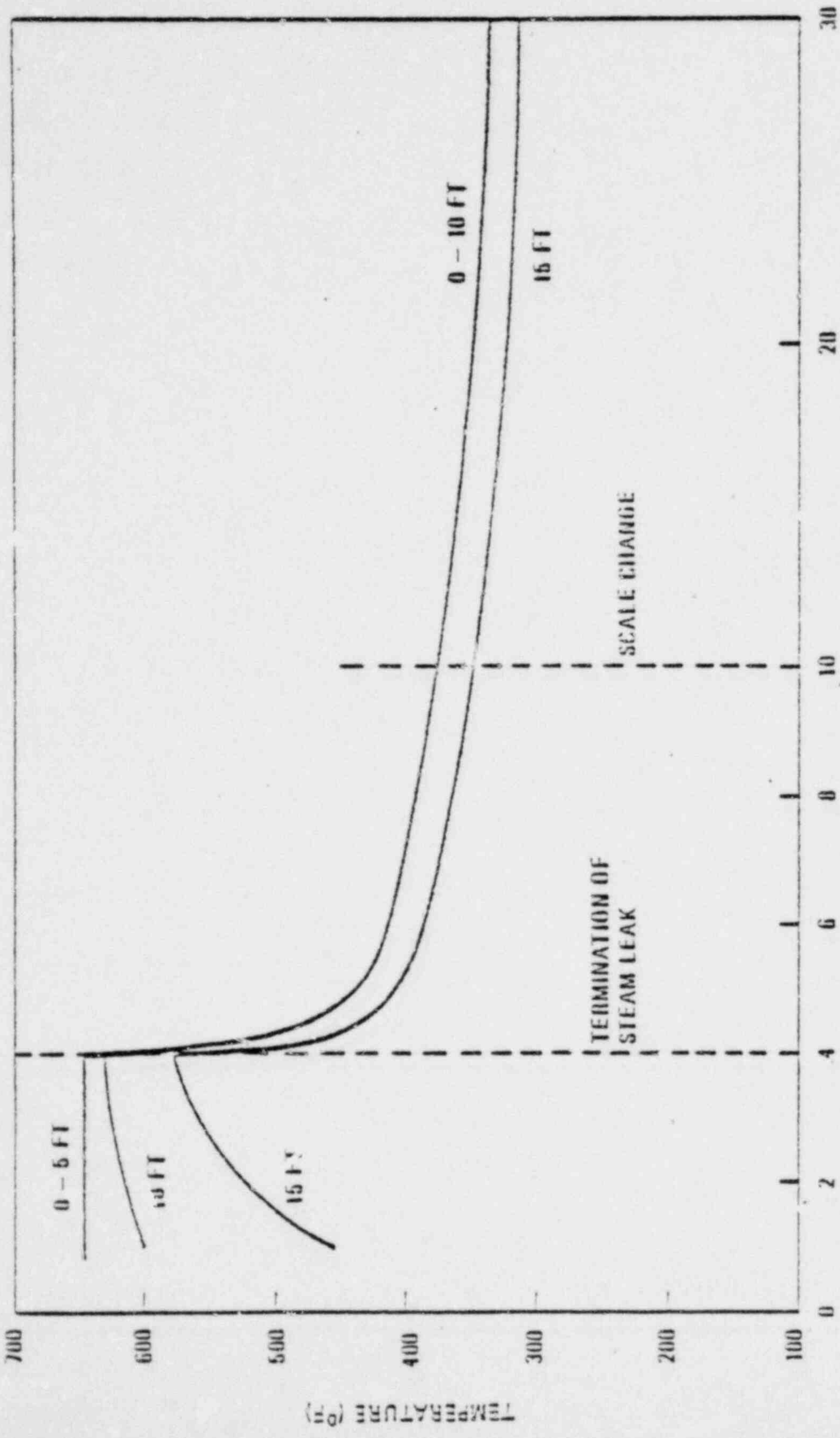

Fig. 3. Temperature response of the environment near the rupture for a reactor building cold reheat pipe rupture

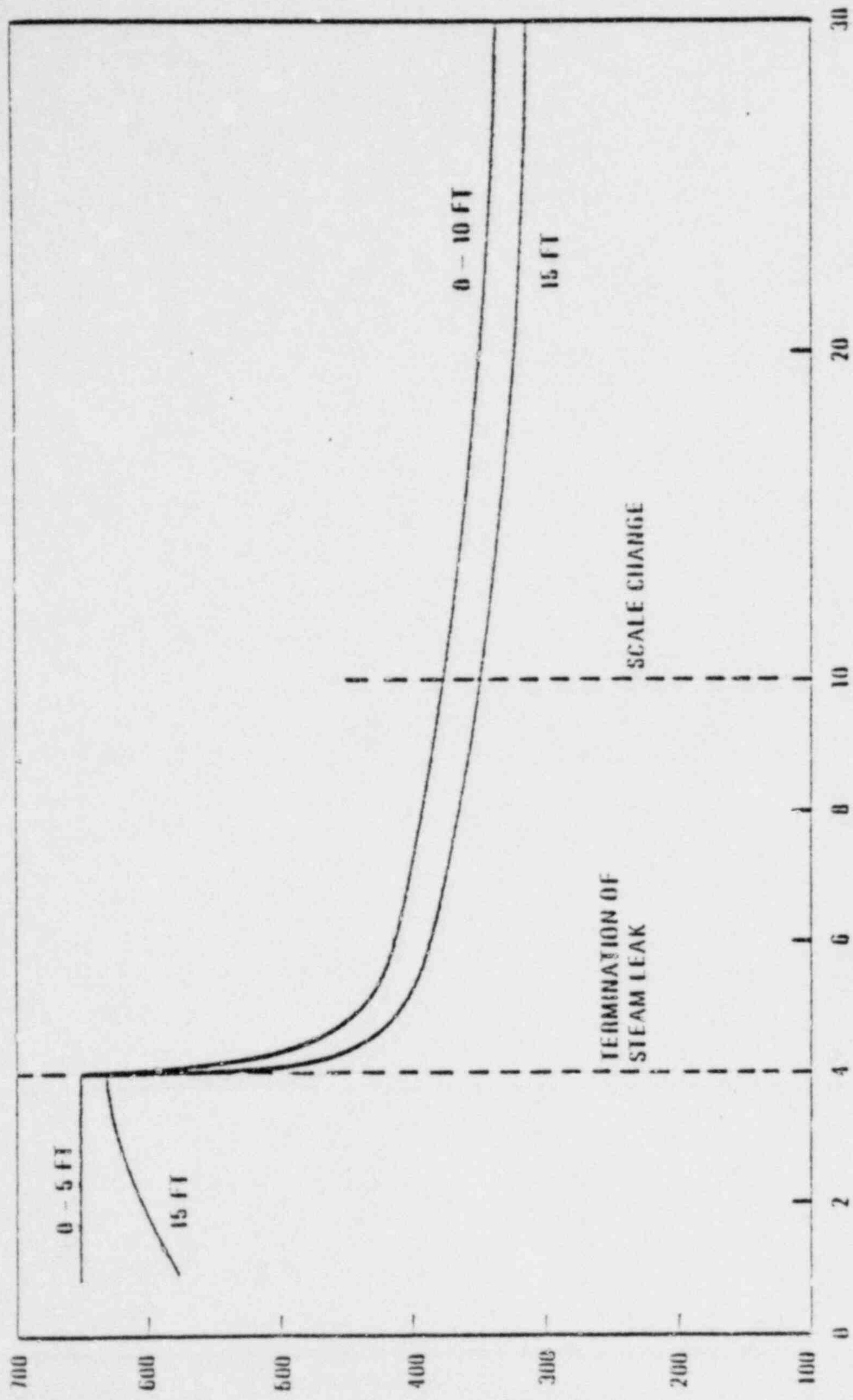

Fig. 4. Temperature response of the environment near the rupture for a turbine building hot reheat pipe rupture