WM. H. ZIMMER POWER STATION

INSTRUCTIONS FOR UPDATING YOUR DESIGN ASSESSMENT REPORT

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conservatively based on existing steam condensation data. Because condensation oscillation occurs over a wide range of blowdown conditions, two CO loads were defined. The first is a high mass flux CO load which would correspond to the early portion of a large break LOCA. The main components of this load are defined as: 14

a. Sinusoidal Pressure Fluctuations

± 4.5 psi @ 2-7 Hz 14

± 2.2 psi @ 11-13 Hz

b. Random Pressure Fluctuations

Steam Bubble Collapse: 15-50 Hz 14

The 2 to 7 hertz component specified represents an increase of about 20% over the DFFR/NUREG-0487 load. The 11 to 13 hertz component is an additional load to account for any vent acoustic effects. The higher frequency portion of the load is added to bound random high frequencies which may appear in test data.

At lower mass fluxes there may be a possibility of a higher contribution from the vent acoustic effect with a corresponding decrease in the low frequency component. The main components of this load are defined as: 14

a. Sinusoidal Pressure Fluctuations

± 2.2 psi 2-7 Hz

± 3.8 psi 11-13 Hz

b. Random Pressure Fluctuations

Steam Bubble Collapse: 15-50 Hz

The 2 to 7 hertz component here is 50% of the low frequency component used in the high mass flux load while the vent acoustic amplitude has been conservatively assumed to be even higher than the amplitude specified in the lower 2 to 7 hertz range in the DFFR. The higher frequency load is defined as described above. The Zimmer Empirical Condensation Oscillation Load bounds the requirements of the NRC Lead Plant Acceptance Criteria (NUREG-0487), as demonstrated by Figure 2.1-1.

2.1.4 Chugging

Chugging loads are divided into two areas. The chugging lateral load is the self loading of the downcomer vent during chugging and affects the design of the downcomers, bracing, and drywell floor. The chugging event also generates a hydrodynamic load which loads the submerged boundaries of the suppression pool.

5.2 SAFETY/RELIEF VALVE (SRV) LOADS - PRESENT DESIGN LOADS (T-QUENCHERS)

Actuation of safety/relief valves (SRV) produces direct transient loads on components and structures in the suppression chamber region and the associated structural response produces transient loadings on piping systems and equipment in the containment region and reactor building. These transient SRV loadings are discussed in the following subsections.

Prior to actuation, the discharge piping of an SRV line contains atmospheric air and a column of water corresponding to the line submergence. Following SRV actuation, pressure builds up inside the piping as steam compresses the air in the line. The resulting high-pressure air bubble that enters the pool oscillates in the pool as it goes through cycles of overexpansion and recompression. The bubble oscillations resulting from SRV actuation and discharge cause oscillating pressures throughout the 14
pool, resulting in dynamic loads on pool boundaries and submerged structures. These dynamic loads cause a dynamic structural response sufficient to affect piping systems and equipment in the containment and reactor buildings. The assessment of the affected systems for these responses is discussed in Chapter 7.0.

Steam condensation vibration phenomena can occur if high-pressure, high-temperature steam is continuously discharged at high-mass velocity from rams head devices into the pool, when the pool is at elevated temperatures. This phenomena is 14
mitigated by maintaining a low pool temperature as discussed in Chapter 8.0 and by installing quencher discharge devices.

The characteristics of the SRV actuation load vary depending on the piping configuration and the discharge device (rams head or quencher) located at the exit of the SRV line. Typically, the quencher device produces lower dynamic loads. Zimmer Power Station used a bounding load calculated for a rams head device as an original design basis for structures, equipment, and piping systems. A bounding quencher load is now used. To provide increased plant safety margins for containment SRV loads and to increase the threshold temperature limit for steam condensation vibration, SRV quencher devices are installed in the plant.

Pool temperature transients for several postulated cases involving a stuck-open safety/relief valve are presented in Section 8.2. The calculated maximum pool temperature was calculated to be a few degrees below the threshold temperature limit for steam condensation instability for a rams head discharge device.

In order to increase the margin between the calculated maximum temperature and this threshold temperature limit, it was decided to install a quencher device having a higher suppression pool

TABLE 5.4-1

ZIMMER POSITION ON NRC LEAD PLANT ACCEPTANCE CRITERIA (NUREG-0487 AND NUREG-0487, SUPPLEMENT NO. 1)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
I. LOCA-Related Hydrodynamic Loads			
A. Submerged Boundary Loads During Vent Clearing	24 psi over-pressure added to local hydrostatic below vent exit (walls and basemat) - linear attenuation to pool surface. Based on 4T results, General Electric letter MFN-080-79, Mr. Sobon to Mr. Stolz, March 20, 1979.	Acceptable (2)*	Acceptable
B. Pool Swell Loads 1. Pool Swell Analytical Model			
a) Air Bubble Pressure	Calculated by the Pool Swell Analytical Model (PSAM) used in calculation of submerged boundary loads.	Acceptable (1)	Acceptable
b) Pool Swell Elevation	Use PSAM with polytropic exponent of 1.2 to a maximum swell height which is the greater of 1.5 vent submergence or the elevation corresponding to the drywell floor uplift ΔP per NUREG-0487 criteria I.A.4. The associated maximum wetwell air compression, per NUREG-0487 criteria I.A.4, is used for design assessment.	Acceptable (2)	Acceptable
c) Pool Swell Velocity	Velocity history vs. pool elevation predicted by the PSAM used to compute impact loading on small structures and drag on gratings between initial pool surface and maximum pool elevation and steady-state drag between vent exit and maximum pool elevation. Analytical velocity variation used up to maximum velocity. Maximum velocity applies thereafter up to maximum pool swell.	NRC Criteria I.A.2 (1)	Acceptable The impact of a 10% increase in pool swell velocity has been assessed and it is concluded that the design is adequate.

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*Numbers in parentheses refer to notes at the end of this table.

TABLE 5.4-1 (Cont'd)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
d) Pool Swell Acceleration	Acceleration predicted by the PSAM. Pool acceleration is utilized in the calculation of acceleration drag during pool swell.	Acceptable (1)	Acceptable
e) Wetwell Air Compression	Wetwell air compression is calculated by the PSAM. Defines the pressure loading on the wetwell boundary above the pool surface during pool swell.	Acceptable (1)	Acceptable
f) Drywell Pressure History	Plant unique. Utilized by PSAM to calculate pool swell loads.	Acceptable if based on NEDM-10320. Otherwise plant unique reviews required. (1)	Acceptable
2. Loads on Sugmerged Boundaries	Maximum bubble pressure predicted by the PSAM added uniformly to local hydrostatic below vent exit (wells and basemat) linear attenuation to pool surface. Applied to walls up to maximum pool swell elevation.	Acceptable (1)	Acceptable
3. Impact Loads			
a) Small Structures	1.5 x Pressure-Velocity correlation for pipes and I beams. Constant duration pulse.	NRC criteria I.A.6 (1)	Acceptable
b) Large Structures	None - Plant unique load where applicable.	Plant unique review where applicable.	Acceptable. Zimmer has no large structures in the pool swell zone.
c) Grating	No impact load specified. P_{drag} vs. open area correlation and velocity vs. elevation history from the PSAM.	NRC Criteria I.A.3 (1)	Acceptable. Zimmer has no grating in pool swell area.
4. Wetwell Air Compression			
a) Wall Loads	Direct application of the PSAM calculated pressure due to wetwell compression.	Acceptable (1)	Acceptable
b) Diaphragm Upward Loads	2.5 psid	NRC Criteria I.A.4 (1)	Acceptable

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TABLE 5.4-1 (Cont'd)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
5. Asymmetric Load	Use 10% of the maximum bubble pressure statically applied to ½ of the submerged boundary (GE letter MFN-076-79, March 16, 1979).	NRC Criteria in Section II.A.3 of Supplement 1 to NUREG-0487 (2)	Acceptable
C. Steam Condensation and Chugging Loads			
1. Downcomer Lateral Loads			
a) Single Vent Loads	8.8 kip static	NRC Criteria I.B.1 (1)	Acceptable
b) Multiple Vent Loads	Prescribes variation of load per downcomer vs. number of downcomers.	NRC Criteria I.B.2 (1)	Acceptable
2. Submerged Boundary Loads			
a) High Steam Flux Loads	Sinusoidal pressure fluctuation added to local hydrostatic. Amplitude uniform below vent exit-linear attentuation to pool surface. 4.4 psi peak-to-peak amplitude. 2 to 7 Hz frequencies.	Acceptable (1)	Acceptable. As described in the discussion of the Zimmer Empirical Loads approach in Chapter 2.0, more conservative loads have been used for assessment and redesign.
b) Medium Steam Flux Loads	Sinusoidal pressure fluctuation added to local hydrostatic. Amplitude uniform below vent exit-linear attenuation to pool surface. 7.5 psi peak-to-peak amplitude. 2 to 7 Hz frequencies.	Acceptable (1)	Acceptable
c) Chugging Loads	Representative pressure fluctuation taken from 4T test added to local hydrostatic.	Acceptable pending resolution of FSI concerns.	Acceptable. Conservatism inherent in the use of 4T data is discussed in Section 3.3.1.1.6.
- uniform loading condition	Maximum amplitude uniform below vent exit-linear attenuation to pool surface, +4.8 psi maximum overpressure, -4.0 psi maximum underpressure, 20 to 30 Hz frequency. Zimmer specific (Lead Plant) chugging load criteria (based on 4T test results) submitted July 1980. Revision 1 to the Lead Plant chugging report was submitted September 1980.		

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TABLE 5.4-1 (Cont'd)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
- asymmetric loading condition	Maximum amplitude uniform below vent exit-linear attenuation to pool surface. 20 psi maximum overpressure, -14 psi maximum underpressure, 20-30 Hz frequency, peripheral variation of amplitude follows observed statistical distribution with maximum and minimum diametrically opposed.		
II. SRV-Related Hydrodynamic Loads			
A. Pool Temperature Limits for KWU and GE four-arm quencher	None specified	NRC Criteria II.1 and II.3 (1)	Acceptable
Quencher Air Cleaning Loads	The methodology documented in the Susquehanna DAR is used for the T-quencher load definition.	NRC Criteria in Section II.B.5 of Supplement 1 to NUREG-0487 (2)	The Zimmer station is being assessed for the T-quencher loads. These loads are considered to be conservative and demonstrate the adequacy of the Zimmer design. A presentation on the impact of modifications to the SRV frequency range was given in the February 13, 1979 meeting. Results of an assessment of the SRV T-quencher frequency range was presented at the July 26, 1979 meeting. The conservatism of the T-quencher load in both amplitude and frequency is described in Subsection 5.2.2.1. A further demonstration of the conservatism of the lead plant approach has been documented by Long Island Lighting Co. (SNRC-374, March 30, 1979, Mr. Novarro [LILCO] to Mr. S. A. Varga [NRC] transmitting a report entitled "Justification of Mark II Lead Plant SRV Load Definition.") In-plant tests will be run to demonstrate the adequacy and conservatism of the design loads.
B. Quencher Tie-Down Loads			
1. Quencher Arm Loads			
a) Four-Arm Quencher	Vertical and lateral arm loads developed on the basis of bounding assumptions for air/water discharge from the quencher and conservative combinations of maximum/minimum bubble pressure acting on the quencher.	Acceptable	Acceptable
b) KWU T-Quencher	T-quencher arm loads, as specified for SSES. (3)	Acceptable	Acceptable. These loads have been calculated using the methodology and assumptions described in DFFR for four-arm quenchers, as recommended in the Acceptance Criteria. KWU T-quencher methods were used to verify conservatism.

TABLE 5.4-1 (Cont'd)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
2. Quencher Tie-Down Loads			
a) Four-Arm Quencher	Includes vertical and lateral arm load transmitted to the basemat via the tie-downs. See II.C.1.a above plus vertical transient wave and thrust loads. Thrust load calculated using a standard momentum balance. Vertical and lateral moments for air or water clearing are calculated based on conservative clearing assumptions.	Acceptable	Acceptable
b) KWU "T"-Quencher	T-quencher tie-down loads, as specified for SSES. (3)	Acceptable	Acceptable. These loads have been calculated using the methodology and assumptions described in DFFR for four-arm quenchers, as recommended in the Acceptance Criteria. KWU T-quencher methods were used to verify conservatism. 14
III. LOCA/SRV Submerged Structure Loads			
A. LOCA/SRV Jet Loads			
1. LOCA/Rams Head SRV Jet Loads	Methodology based on a quasi-one-dimensional model.	NRC Criteria in Section II.C.1 of Supplement 1 to NUREG-0487. (2)	Acceptable. Rams head - N/A. For LOCA jet see Subsection 5.3.2.1.
2. SRV-Quencher Jet Loads	No loads specified for lead plants. Model under development in long-term program.	NRC Criteria III.A.2 (1)	Acceptable. The spherical zone of influence defined in the Acceptance Criteria is not appropriate for the two-arm quencher. A zone of influence for each arm will be defined as a cylinder with an axis coincidental with the quencher arm. The length of the cylinder will be equal to the length of the quencher arm plus 10 end cap hole diameters. The radius of the cylinder is expected to be quite small. However, because no structures are within 5 feet of the quencher arm, 5 feet will be assumed.
B. LOCA/SRV Air Bubble Drag Loads			
1. LOCA Air Bubble Loads	Details of methodology are included in Appendix G.	NRC Criteria in Section II.C.2 of Supplement 1 to NUREG-0487. (2)	Acceptable
2. SRV-Rams Head Air Bubble Loads	The methodology is based on an analytical model of the bubble charging process including bubble rise and oscillation. Acceleration drag alone is considered.	NRC Criteria III.B.2 (1)	Acceptable. Used in Item III.B.3. Standard drag is calculated and included for all submerged structure load calculations. Other submerged structure concerns in NUREG-0487 are addressed in Appendix G.

TABLE 5.4-1 (Cont'd)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
3. SRV-Quencher Air Bubble Loads	No quencher drag model provided for lead plants. Lead plants propose interim use of rams head model (See III.B.2 above). Model will be developed in long-term program.	NRC Criteria III.B.3 (1)	The bubble location and radius will be defined appropriately for T-quenchers. Bubbles are located near the arms. The bubble size is predicted from the line air volume.
C. Steam Condensation Drag Loads	No generic load methodology provided. Generic model under development in long-term program.	Lead plant load specification and NRC review.	Described in Subsections 5.3.1.3.5 and 5.3.1.3.6. 14
IV. Secondary Loads			
A. Sonic Wave Load	Negligible Load - none specified	Acceptable	---------------
B. Compressive Wave Load	Negligible Load - none specified	Acceptable	---------------
C. Post Swell Wave Load	No generic load provided	Plant unique load specification and NRC review.	Described in Zimmer Closure Report
D. Seismic Slosh Load	No generic load provided	Plant unique load specification and NRC review.	Described in Zimmer Closure Report
E. Fallback Load on Submerged Boundary	Negligible load - none specified	Acceptable	---------------
F. Thrust Loads	Momentum balance	Acceptable	---------------
G. Friction Drag Loads on Vents	Standard friction drag calculations	Acceptable	---------------
H. Vent Clearing Loads	Negligible Load - none specified	Acceptable	---------------

NOTES: (1) Per NUREG-0487 (NRC Mark II Lead Plant Acceptance Criteria)
(2) Per NUREG-0487, Supplement 1
(3) Per Susquehanna Steam Electric Station (SSES) Design Assessment Report (DAR).

TABLE 5.4-1 (Cont'd)

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
FUNCTIONAL CAPABILITY		Interim technical position (7/19/78)	Acceptable, Rodabaugh criteria may be used in some cases if NRC finds acceptable.
MASS-ENERGY RELEASE FOR ANNULUS PRESS.		Verify using RELAP4/MOD5	Acceptable
QUESTIONS MEB-2, MEB-5		15% peak broadening to be used.	Acceptable
MEB-3, MEB-5		Closely spaced modes combined Per 1.92	Acceptable. NSSS scope uses modified summation per approved GESSAR.
MEB-1		Dynamic analysis methods acceptable	Acceptable
MEB-2		OBE Damping - Level A or B SSE Damping - Level C or D	Acceptable
MEB-6		Seismic slosh-plant unique review	Acceptable
MEB-7a and b		Load Combinations: AP+SSE OBE+SRV	Acceptable. See load combination table for Case #2 and #7.
MEB-8		Functional capability and piping acceptance criteria	See load combination table.

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TABLE 5.4-1 (Cont'd)

LOAD PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS	ZIMMER POSITION ON ACCEPTANCE CRITERIA
1.		N+SRV$_x$ To B	Acceptable
2.		N+SRV$_x$+OBE to B	Acceptable Approved GESSAR approach used for NSSS.
3.		N+SRV$_{all}$+SSE to C	Acceptable
4.		N+SRV$_{ads}$+OBE+IBA to C	Acceptable
5.		N+SRV$_{ads}$+OBE+IBA to C	Acceptable
6.		N+SRV$_{ads}$+SSE+IBA to C	Acceptable
7.		N+SSE+DBA to C	Acceptable
8.		N to A	Acceptable
9.		N+OBE to B	Acceptable
10.		N+SRV$_a$+SSE+DBA to C	Applied to containment structure only (See M 020.22 and DFFR 5.2.4.)

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TABLE 6.1-1

DESIGN LOAD COMBINATIONS*

EQN	LOAD COND	D	L	F_O	P_O	T_O	R_O	E_O	E_{SS}	P_B	P_A	T_A	R_A	R_R	SRV	ADS	ALL	ASYMMET-RICAL	SINGLE
1	Normal w/o Temp	1.4	1.7	1.0	1.0	-	-	-	-	-	-	-	-	-	1.5	0	X	X	
2	Normal w/Temp	1.0	1.3	1.0	1.0	1.0	1.0	-	-	-	-	-	-	-	1.3	0	X	X	
3	Normal Sev. Env.	1.0	1.0	1.0	1.0	1.0	1.0	1.25	-	-	-	-	-	-	1.25	0	X	X	
4	Abnormal	1.0	1.0	1.0	-	-	-	-	-	1.25	-	1.0	1.0	-	1.25	X	0	X	
4a		1.0	1.0	1.0	-	-	-	-	-	-	1.25	1.0	1.0	-	1.0	0	0	0	X
5	Abnormal Sev. Env.	1.0	1.0	1.0	-	-	-	1.1	-	1.1	-	1.0	1.0	-	1.1	X	0	X	
5a		1.0	1.0	1.0	-	-	-	1.1	-	-	1.1	1.0	1.0	-	1.0	0	0	0	X
6	Normal Ext. Env.	1.0	1.0	1.0	1.0	1.0	1.0	-	1.0	-	-	-	-	-	1.0	0	X	X	
7	Abnormal Ext. Env.	1.0	1.0	1.0	-	-	-	-	1.0	1.0	-	1.0	1.0	1.0	1.0	X	0	X	
7a		1.0	1.0	1.0	-	-	-	-	1.0	-	1.0	1.0	1.0	1.0	1.0	0	0	0	X

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LOAD DESCRIPTION

D = Dead Loads

L = Live Loads

F = Prestressing Loads

T_O = Operating Temperature Loads

R_O = Operating Pipe Reactions

P_O = Operating Pressure Loads

SRV = Safety/Relief Valve Loads

E_O = Operating Basis Earthquake

SBA = Small Break Accident

E_{SS} = Safe Shutdown Earthquake

P_B = SBA and IBA Pressure Load

T_A = Pipe Break Temperature Load

R_A = Pipe Break Temperature Reactions Load

P_A = DBA Pressure Loads (including all pool hydrodynamic loadings)

R_R = Reactions and Jet Forces Due to Pipe Break

IBA = Intermediate Break Accident

*In any load combinations, if the effect of any load other than D reduces the design forces, it will be deleted from the combination.

CHUG - Chugging load defined between 20 to 30 hertz; and

$CO_{(EMPIRICAL)}$ - Condensation oscillation load including higher frequency contributions (up to 50 hertz). Includes the envelope of two independent CO empirical loads (see Chapter 2.0). 14

6.4.2.1.2 Piping Systems Outside the Reactor Building

All essential and safety-related piping systems outside the reactor building considered the effects of seismic loads. The load combinations considered included the following:

LOAD COMBINATION	ACCEPTANCE CRITERIA
N + OBE	Service Level B
N + DBE	Service Level C

where the loads are as defined in Subsection 6.4.2.1.1.

6.4.3 Balance-of-Plant Equipment

6.4.3.1 Loading Combinations

The table below defines the combinations of the normal, seismic, and pool dynamic loads considered in the equipment qualification:

a. N + OBE + Envelope ($SRV_{ALL}TQ$ and $SRV_{ASY}TQ$)

b. N + Envelope (OBE and SSE) + Envelope ($SRV_{ADS}TQ$ and $SRV_{ASY}TQ$) + CO (Zimmer empirical medium mass flux)*

c. N + Envelope (OBE and SSE) + Envelope ($SRV_{ADS}TQ$ and $SRV_{ASY}TQ$) + Chugging 14

d. N + Envelope (OBE and SSE) + Envelope (0.6 SRV_{ADS-TQ} and SRV_{ASY-TQ}) + CO (Zimmer empirical high mass flux)*

e. N + $\sqrt{(AP)^2 + (SSE)^2}$

*See Chapter 2.0 for explanation of condensation oscillation.

6.4.3.2 Acceptance Criteria

a. Allowable Stress Limits

1. ASME Equipment

3. The transfer functions of the response were obtained by the computer program FAST.

From Equation (1)

$$T_k(\omega) = \frac{R_k(\omega)}{F(\omega)}$$

in which $R_k(\omega)$ was the Fourier transform of the responses saved in step (2) and $F(\omega)$ is the Fourier transform of the white noise load used in step (1) of the above.

4. For steady-state solution of the harmonic load, by definition from Equation (1), the transfer function itself was the response.

For SRV loads with variable frequency, the transfer functions were scanned in the frequency range of the loading. The maximum response could be obtained as the product of the transfer functions and the Fourier transforms of the load, using the FAST program. Response acceleration time histories were also generated to obtain response spectra using the RSG program.

In order to consider a conservative frequency content, three KWU time history traces reported in the SSES DAR were expanded into longer and shorter time history durations by multiplying the time scales by a factor of 2.0 and 0.9, respectively. In addition, the pressure scale were multiplied by a factor of 1.5 for each of the three traces. 14

The resulting structural responses to the various SRV T-quencher loads were combined with the other appropriate loads as per the load combinations shown in Table 6.1-1. The margin factors from these load combinations are presented in Tables 7.1-17 through 7.1-24.

7.1.2 Structural Analysis of LOCA Loads

The analysis of the structure for the LOCA loads was performed as a set of analyses covering each LOCA related phenomenon

Magnitude: ± 3.75 psi

Frequency: 2 to 7 hertz

The spatial distributions of the condensation oscillation loads are shown in Figure 7.1-13 for rams head design basis and in Figure 7.1-14 for T-quencher design basis, as ± 3.75 psig acting at a frequency 2 to 7 hertz on the basemat, containment, and reactor pedestal.

The structural model described in Subsection 7.1.2.1 was used for the rams head design basis, and the one described in Subsection 7.1.1 was used for the T-quencher design basis.

The load was assumed to be harmonic in time, and only the steady-state response was considered as being of interest. For this purpose, frequency response variations were determined for all response components of interest using the computer program FAST, Appendix A, which obtained the complex frequency response by calculation of the discrete Fourier transform of both load and response. The frequency range of 2 to 7 hertz on the frequency response was considered relevant in evaluating the structural response.

The resulting structural responses to the condensation oscillation loads were combined with the other appropriate loads as per the load combinations shown in Table 6.1-1. The margin factors from these combinations are presented in Tables 7.1-2 through 7.1-24.

In addition to the above CO load (2-7 hertz), an empirical limiting CO load was considered in combination with the T-quencher design basis of the ZPS-1 containment. This load was intended to be a best estimate of the conservative load specification which resulted from the full-scale condensation oscillation test to be conducted in the 4T facility. All the details for this load are described in Chapter 2.0. 14

This ZPS-1 empirical CO load was incorporated for the T-quencher design basis. The spatial distributions of this load are shown in Figure 7.1-14.

The resulting structural responses to this empirical CO load were combined with the other appropriate loads as per the load combination shown in Table 6.1-1. The margin factors from the load combinations are presented in Tables 7.1-25 through 7.1-28.

7.1.2.4 Chugging Analysis

The chugging loads used in the analysis are described in Section 5.3 and presented in Figure 7.1-15. The finite-element model used in the analysis is described in Subsection 7.1.2.

The forces of reactor support margin factor were obtained by analysis using the model described in Subsection 7.1.2.1.

Margins shown in Table 7.1-14 for loading conditions 4a, 5a, and 7a on the drywell floor are for the LOCA effects, including the lateral loads on the downcomers. As per DFFR Subsection 4.4.6.6, a net upward load of 9 psid acting on the drywell floor has been considered.

Margins shown in Table 7.1-14 for loading conditions 1, 2, 3, and 6 on the drywell floor are for all the valves discharge loading which clearly governs the design of the drywell floor rather than the asymmetric two valve discharge loading.

Loading conditions 4, 5, and 7 in Table 7.1-14 include all loads resulting from a small pipe break combined with the loads due to the discharge of all 13 SRV's. This was done for reasons of analytical expediency, since the discharge of all 13 SRV's transmits significantly more energy to the drywell floor than the 6 valve ADS discharge. Since ZPS-1 can take this higher loading case, the actual loading from the ADS valves was not considered. For the drag loads on the downcomer, the maximum load described in Section 5.2 was used for all loading combinations which include SRV loads irrespective of the discharge mode (ALL, ASYMMETRIC, or ADS). 14

T-QUENCHER DESIGN BASIS

LOAD COMBINATION WITH NRC CO LOAD (DFFR)

a. Basemat — Tables 7.1-17 through 7.1-20

b. Containment wall — Tables 7.1-21 through 7.1-24

LOAD COMBINATION WITH EMPIRICAL LIMITING CO LOAD

a. Basemat — Tables 7.1-25 and 7.1-26

b. Containment wall — Tables 7.1-27 and 7.1-28

The margin factors were calculated as results of the assessment based on the NRC acceptance criteria (modified for the T-quencher).

All the margin factors were greater than 1.0 except the following cases:

7.3 OTHER STRUCTURAL COMPONENTS

A reassessment for the additional effects of pool dynamic loads was made for steel and concrete structures in the reactor building including steel framing and galleries, cable pan hangers, conduit hangers, HVAC duct hangers, and concrete slabs, beams and shear walls. The design and analysis procedure for the reassessment was the same as described in Subsections 3.8.3 and 3.8.4 of the ZPS-1 FSAR for the original design of the plant. Load combinations described in Section 6.3 of this report were used for the assessment.

7.3.1 Downcomers and Downcomer Bracing

7.3.1.1 General Description

There are 88 downcomers anchored in the drywell floor. The downcomers are also connected to the containment structure by horizontal bracing at elevation 496 feet. Figure 7.3-1 shows the downcomers in the suppression pool, and Figure 7.3-2 shows the layout of the horizontal bracing.

The bracing members are attached to embedment plates which are anchored to the containment wall by a system of bolts and backup cap devices. A typical detail of this attachment is shown in Figure 7.3-4.

As shown in the Figure 7.3-4, the length of the bolt is such that it does not interfere with the vertical or hoop tendons in place. Leak tightness is restored by providing stainless steel pipe caps over the bolts and stainless steel plate inserts around the embedment plate. A typical detail of connection of bracing to downcomer is shown in Figure 7.3-3. 14

The downcomers and downcomer bracing are subjected to static and dynamic loads due to normal, upset, emergency, and faulted plant operating conditions. The loadings cases were obtained from the DFFR and are identified in detail in Subsection 7.3.1.2. The loading combinations are explained in Subsection 7.3.1.3. The design limits are identified in Subsection 7.3.1.4, and the analytical methods are presented in Subsection 7.3.1.5.

7.3.1.1.1 Downcomer Properties

The following are the properties of the downcomers:

a. outside diameter - 25.00 inches;

b. wall thickness - 0.500 inch;

c. weight per unit length - 131 lb/ft;

d. material - SA-516, Grade 60; and

e. damping coefficient - 3%.

7.3.1.1.2 <u>Bracing Properties</u>

The following are the properties of the bracing:

a. outside diameter - 8.625 inches;

b. wall thickness - 0.875 inch;

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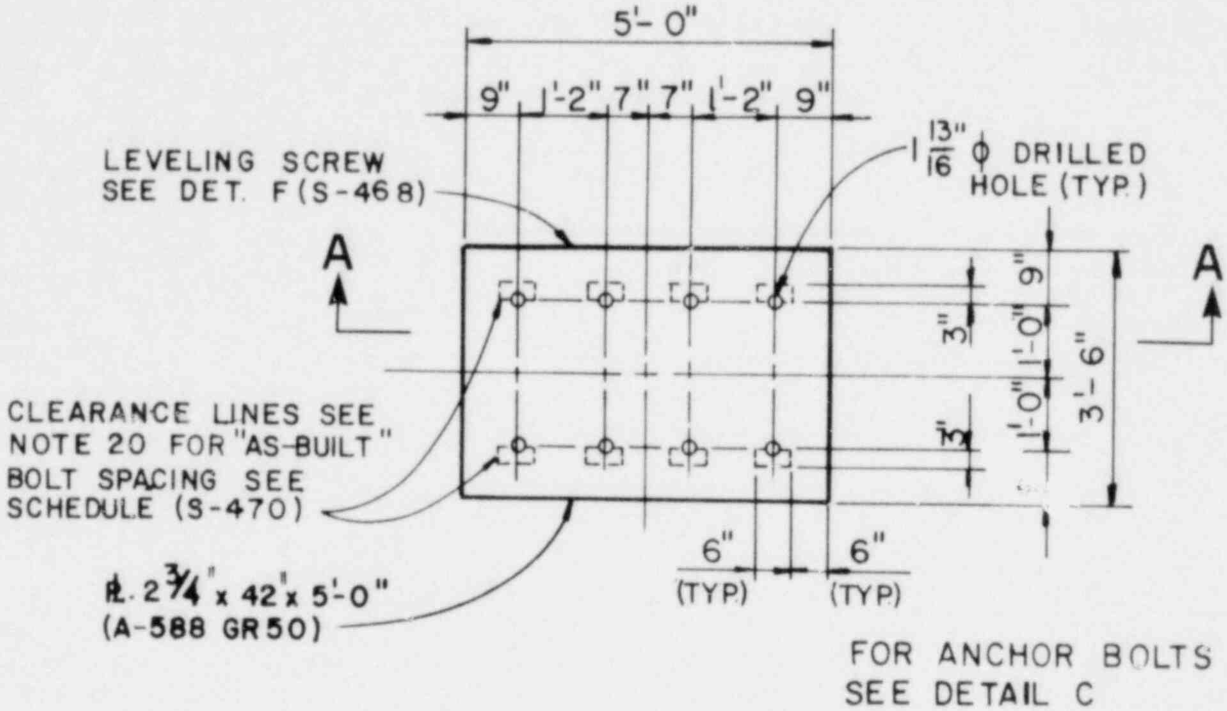
TABLE 7.3-1

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR DOWNCOMER AND DOWNCOMER BRACING

LOAD CASE	NRC LOAD COMBINATION (NUREG-0487)	T-QUENCHER DESIGN-BASIS	ASME STRESS CRITERIA
1	$N+SRV_X$	N+SRV	B (UPSET)
2	$N+SRV_X+OBE$	N+SRV +OBE	B (UPSET)
3	$N+SRV_X+SSE$	N+SRV +SSE	C (EMERGENCY)
4	$N+SRV_{ADS}+IBA(SBA)$	N+SRV +CHUG	C (EMERGENCY)
5	$N+SRV_{ADS}+OBE+IBA(SBA)$	N+SRV +OBE+CHUG	C (EMERGENCY)
6	$N+SRV_{ADS}+SSE+IBA(SBA)$	N+SRV +SSE+CHUG	C (EMERGENCY)
7	N+SSE+DBA	N+SSE+CO	C (EMERGENCY)
8	N	N	A (NORMAL)
9	N+OBE	N+OBE	B (UPSET)
10	$N+SRV_X+SSE+DBA$	- CONTAINMENT STRUCTURE ONLY JUSTIFICATION PROVIDED BY GE.	

14



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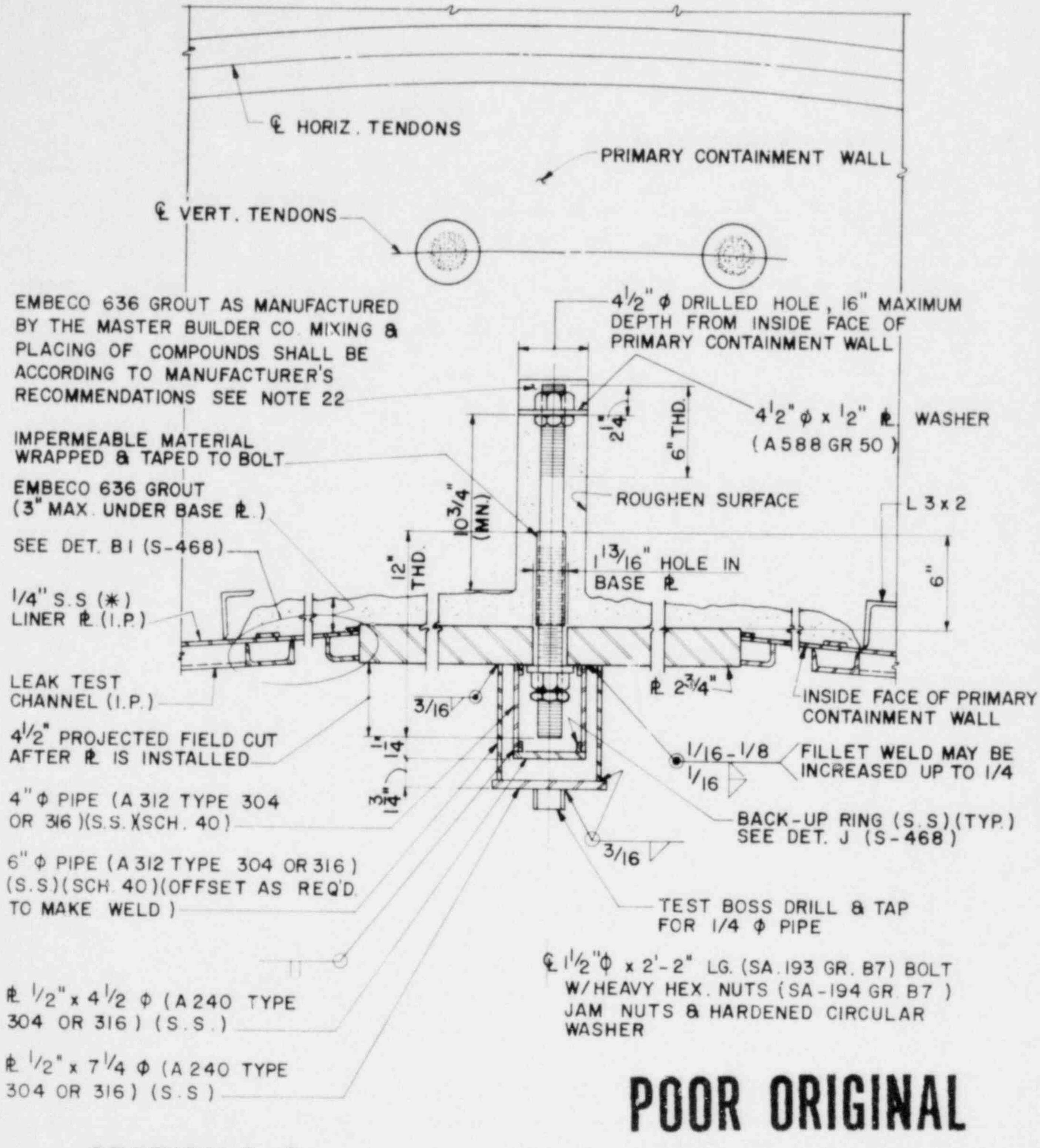
WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1

MARK II DESIGN ASSESSMENT REPORT

FIGURE 7.3-4

EMBEDMENT PLATE

(SHEET 1 of 2)



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WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1

MARK II DESIGN ASSESSMENT REPORT

FIGURE 7.3-4

EMBEDMENT PLATE

(SHEET 2 of 2)

CHAPTER 8.0 - SUPPRESSION POOL WATER TEMPERATURE MONITORING SYSTEM

8.1 SYSTEM DESIGN

8.1.1 Safety Design Basis

The safety design basis for setting the temperature limits for the suppression pool temperature monitoring system are based on providing the operator with adequate time to take the necessary action required to ensure that the suppression pool temperature will always remain below the pool temperature limit established by the NRC. An analysis of suppression pool temperature transients can be found in Section 8.2. The system design also provides the operator with necessary information regarding localized heatup of the pool water while the reactor vessel is being depressurized. If relief valves are selected for actuation, they may be chosen to ensure mixing and uniformity of heat energy injection to the pool.

8.1.2 General System Description

The suppression pool temperature monitoring system monitors the pool water temperature in order to prevent the local pool water temperature from exceeding the pool temperature limit during SRV discharge and provides the operator with the information necessary to prevent excessive pool temperatures during a transient or accident. Temperatures in the pool are recorded and alarmed in the main control room. The instrumentation arrangement in the suppression pool consists of 18 local temperature sensors in individual guide tubes mounted off the pool walls. 14

The local temperature sensors consist of 18 dual-element, copper constantan thermocouples located 1 foot below the low water level.

Twelve of the sensors are located off the outer suppression pool wall at azimuths 28°, 45°, 86°, 117°, 147°, 183°, 217°, 240° 263°, 277°, 325°, and 344°. The other six are located off the pedestal at azimuths 55°, 142°, 202°, 246°, 298°, and 344°.

The sensors and readout devices are assigned to ESS-1 and ESS-2 divisions and local discharge areas are monitored by two sensors, one from each division. This represents a conservative measurement of local pool water heatup. All instrumentation will be qualified Seismic Category I. The time constant of the thermocouple installation will be no greater than 15 seconds. The difference between measurement reading and actual temperature will be within ± 2° F.

The display techniques for monitoring the pool temperature are:

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a. to continuously input to the computer system the measurement made by Element 1 of each of the nine thermocouples in ESS-1 which can be displayed individually or averaged by the computer to display the bulk temperature;

b. to sequentially record on a multipoint recorder the measurement made by Element 1 of each of the nine thermocouples in ESS-2 at a rate of 5 sec/point when all nine are below the alarm level, and at a rate of 1 sec/point when any of the nine are above the alarm level;

c. to continously record on a strip-chart recorder the bulk temperature obtained by electrically averaging the nine Element 2 thermocouples of ESS-1; and

d. to continuously input to the computer system and display on a hardwired indicator the bulk temperature as obtained by electrically averaging the nine Element 2 thermocouples of ESS-2.

Each instrumentation division has the capability of alarming both local and bulk high temperature. The computer system provides temperature readout via CRT/data logger on demand. The above configuration provides the maximum flexibility for providing redundant pool temperature information to the operator. 14

The quenching of the steam at the quencher discharge forms jets that heat the water and generate convection currents in the suppression pool. These currents eventually rise and displace cooler water near the pool surface.

During an extended blowdown, a large temperature gradient is expected initially near the quencher. After a short time the pool gradients will stabilize with a bulk to local temperature difference of about 10° F. The adequacy of the temperature monitoring system will be confirmed by the in-plant SRV testing.

8.1.3 Normal Plant Operation

The temperature monitoring system is utilized during normal plant operation to ensure that the pool temperature will remain low enough to condense all quantities of steam that may be released in any anticipated transient or postulated accident. When rams head devices were specified for design, there was an NRC concern that high pool temperature might result in high pool dynamic loads during SRV discharge because of unstable steam condensation. Installation of T-quenchers has eliminated this concern. During normal plant operation, the system is in continuous operation recording the suppression pool water temperature with a readout in the main control room. If the pool temperature rises above normal operating temperatures, an alarm

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is actuated in the control room allowing the operator to take actions as required to maintain pool temperature below the pool temperature limit as described in Section 8.2.

8.1.4 Abnormal Plant Operation

BWR plants take advantage of the large thermal capacity of the suppression pool during plant transients which require relief valve actuation. The discharge of each relief valve is piped to the suppression pool, where the steam is condensed. This results in a pool water temperature increase but a negligible increase in containment pressure. However, certain events have the potential for substantial energy addition to the suppression pool and could result in a high local pool temperature if timely corrective action is not taken.

When rams head discharge devices are used, test results and operating experience indicate that high magnitude oscillatory loads may occur when a high steam mass flux is injected into a pool with local temperature above 170° F. Although analysis demonstrates that the pool temperature will remain below 150° F, when the steam mass flux is high enough to cause these loads, T-quenchers have been installed instead of the rams heads to provide additional margin to the pool temperature limits. 14

8.1.4.1 Plant Transients

This subsection discusses various plant transients which result in SRV discharges to the suppression pool and which could possibly lead to high pool temperature.

8.1.4.2 Abnormal Events

Various events that result in energy being discharged to the suppression pool via the safety/relief valves are discussed in the following paragraphs. Most of these transients are of short duration and have little effect on the suppression pool temperature. However, three events have the potential for substantially high energy release to the pool that could result in undesirably high pool temperatures if timely corrective action is not taken. These events are: (1) events that result in the isolation of the plant from the main condenser, (2) stuck-open relief valve, and (3) automatic depressurization system (ADS) operation. A brief description of each of these events is given in the following subsections.

8.1.4.3 Primary System Isolation

When the primary system is isolated from the main condenser, the reactor is scrammed automatically and the stored energy in the vessel internals, fuel relaxation energy, and decay heat is rejected to the suppression pool. The amount of heat rejected to the pool depends on reactor size, power level, and primary system heat removal capability. This includes condensing type heat exchangers which remove steam directly from the RPV.

8.1.4.4 Stuck-Open Relief Valve

The steam flow rate through a safety-relief valve (SRV) is proportional to reactor pressure. One method to terminate energy input to the pool is to scram the reactor and depressurize the RPV in the event the relief valve cannot be closed. During the energy dump, the pool temperature will increase at a rate determined by the RPV pressure, flow capacity of the SRV, primary system heat removal system capability, and suppression pool water heat removal capability.

8.1.4.5 Automatic Depressurization System (ADS)

Activation of ADS results in rapid depressurization of the RPV by the opening of a designated number of safety-relief valves. During this transient, the bulk suppression pool temperature rises. In a typical case, the RPV is depressurized below 150 psia in about 10 minutes.

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8.1.5 Transients of Concern

There are seven plant depressurization transients that were considered as limiting events (with rams heads) for energy released to the suppression pool. These events are numbered 1 through 7 for ease of reference and are described in the following paragraphs:

Event 1 is a stuck-open relief valve with the reactor at full power. The plant is scrammed and depressurization begun via the stuck-open valve. The initial pool temperature is the maximum pool temperature allowed for continuous operation. This is the only event that is not truly limiting, since all RHR heat exchanger equipment is considered operational.

Event 2 is identical to Event 1, except that one RHR heat exchanger is considered unavailable. The remaining heat exchanger equipment is placed in the suppression pool cooling mode.

Event 3 is a stuck-open relief valve with the reactor isolated from the main condenser and maintained at operating pressure and temperature. All RHR heat exchanger equipment is considered operational in this case, because if one heat exchanger is not available, the reactor may not remain isolated and pressurized. The bulk pool temperature assumed in this event is the highest allowed while the reactor is maintaining pressure.

Event 4 is a controlled depressurization from the plant condition described in Event 3. This event demonstrates the ability of the plant to safely depressurize through a controlled transient from the limiting conditions.

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Event 5 is a controlled depressurization several minutes after reactor isolation described in Event 3. One heat exchanger is considered unavailable because power operation in that condition is not prohibited. This event demonstates the ability to safely depressurize the plant through a controlled transient with only partial RHR heat exchanger equipment availability.

Event 6 is a rapid depressurization of the reactor resulting from actuation of the automatic depressurization system (ADS). ADS actuation takes place at the time the plant is scrammed. Because this event takes place quickly, no heat exchangers are brought into operation. This event represents the most rapid energy release to the pool and demonstrates that the plant may be safely depressurized in this manner without use of heat exchangers. 14

Event 7 is also a rapid depressurization via ADS, but with the reactor isolated and pressurized and with a limiting pool temperature condition. All heat exchanger equipment is placed in the pool cooling mode 0.5 hour after scram but is considered unavailable when depressurization begins.

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8.2 SUPPRESSION POOL TEMPERATURE RESPONSE

8.2.1 Introduction

This submittal is in response to an NRC request for suppression pool temperature calculations for specified transients. CG&E has worked with the Mark II Owners Group and the Mass/Energy Subcommittee to develop a generic approach to this response. This analysis was performed in accordance with the Mass/Energy "White Paper" (revision 1) and incorporates a degree of conservatism that causes the Zimmer pool temperature calculations to slightly exceed the bulk pool temperature goal of 190° F in two cases. CG&E has identified conservatisms and performed a Zimmer plant unique analysis and sensitivity study. This analysis and study indicates that ZPS-1 will not exceed the desired goal of 190° F for the postulated events.

The Wm. H. Zimmer Station Unit 1 (ZPS-1) takes advantage of the large thermal capacitance of the suppression pool during plant transients requiring safety/relief valve (SRV) actuation. The discharged steam is piped from the reactor pressure vessel (RPV) to the suppression pool where it condenses, resulting in a temperature increase of the pool water, but negligible increase in the containment pressure. Most transients that result in relief valve actuations are of very short duration and have a small effect on the suppression pool temperature. However, certain postulated events with conservative assumptions present the potential for substantial energy additions to the suppression pool that could result in high pool temperature. See Figure 8.2-1 for a graphical representation. 14

8.2.1.1 System Description

ZPS-1 uses the residual heat removal (RHR) system to remove heat from the suppression pool. The RHR system consists of three independent loops. Two of these loops are equipped with 100% capacity heat exchangers that are available in the pool cooling, shutdown cooling, containment spray, and steam condensing modes of operation. The RHR heat exchangers are U-tube, vertical head-down units which use Ohio River water for tube side coolant. All three RHR loops are available for the low pressure coolant injection (LPCI) mode of operation. See Figures 8.2-2 through 8.2-6 for a graphical representation of the RHR system.

In the pool cooling mode of operation, RHR pump suction is taken from the suppression pool, cycled through the heat exchanger, and returned to the pool. In shutdown cooling, pump suction is taken from the reactor recirculation (RR) piping, cycled through the heat exchanger, and returned to the reactor pressure vessel (RPV). The steam condensing mode removes steam from the RPV, condenses it in the heat exchanger, and returns it to the RPV via the reactor core isolation cooling (RCIC) system. The steam condensing mode is not used in this analysis. The LPCI mode of operation is automatically initiated on high drywell pressure or on low RPV water level (-146 inches). In the LPCI mode, pump suction is taken from the suppression pool and cycled back to the suppression pool through the minimum flow bypass line until the RPV reaches pressure permissive for injection into the RPV. See Subsection 5.5.7 of the Zimmer FSAR for the RHR system description.

ZPS-1 is equipped with two steam-driven feedwater pump turbines (Figure 8.2-7) which receive their steam supply from the main steamlines downstream from the main steam isolation valves (MSIV's). When MSIV closure occurs, the steam driving force to the feedwater turbines is eliminated, causing a rapid decrease in the amount of feedwater entering the RPV. This treatment of the feedwater system is not used in this analysis. ZPS-1 feedwater system is described in Subsection 10.4.7 of the ZPS-1 FSAR.

8.2.1.2 Background, Response to NRC Request

Cincinnati Gas & Electric Company was asked to demonstrate, for several postulated transients at Zimmer plant, that the NRC pool temperature limit would not be exceeded.

The NRC requested figures showing reactor pressure and suppression pool temperature versus time for the following events: 14

a. stuck open SRV during power operation assuming reactor scram at 10 minutes after pool temperature reached 110° F and all RHR loops available;

b. same as above except only one RHR train available;

c. stuck open SRV during hot standby condition assuming 120° F pool temperature initially and only one RHR train available;

d. automatic depressurization system (ADS) activated during a small line break assuming an initial pool temperature of 120° F and only one RHR train available; and

e. the primary system is isolated and depressurized at a rate of 100° F/hour with an initial pool temperature of 120° F and only one RHR train available.

In addition, CG&E was asked to provide important plant parameters, such as, service water temperature, RHR heat exchanger capability, and initial pool mass for the analysis.

This submittal of pool temperature transients answers the NRC request. The Mark II Owners Group "White Paper" (Revision 1)

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identified the six cases to be analyzed by the Mark II plants. These are:

1. Stuck-Open Relief Valve

   a. from power operation with loss of one RHR heat exchanger, and

   b. from power operation with spurious closure of MSIV's.

2. Isolation/Scram (SRV Discharge)

   a. loss of one RHR heat exchanger, and

   b. stuck-open relief valve.

3. Small break accident

   a. loss of one RHR heat exchanger, and

   b. loss of shutdown cooling. 14

Results of the analysis are listed in Table 8.2-1, and shown in Figures 8.2-8 through 8.2-13. A list of important plant parameters is in Table 8.2-2.

8.2.1.3 Conservatisms

Subsection 8.2.2.2 lists all the general assumptions and initial conditions. Many of the assumptions used in the analysis maximize heat addition to the suppression pool. For example, with the exception of case 1a, MSIV's are assumed to close 3.5 seconds after scram. This eliminates the condenser as a heat sink and maximizes heat addition to the suppression pool. Justification for maintaining condenser availability in case 1a can be found in Attachment 8B.

This analysis further maximizes heat addition to the RPV by assuming hot feedwater flows into the RPV throughout the transients. In reality, however, MSIV closure eliminates steam supply to the turbine-driven feedwater pumps which causes termination of feedwater flow into the RPV within seconds. The ECCS pumps would be the source for providing liquid inventory to maintain RPV level on isolation events. Since hot feedwater has a much higher enthalpy than ECCS suction from the condensate storage tanks or the suppression pool, the obvious difference in heat addition to the RPV is a conservatism which has a significant impact on pool temperature. A sensitivity study on feedwater flow is in Attachment 8A.

Using design fouling values for the RHR heat exchangers, coupled with the assumption that 5% of the tubes are plugged, plus use of

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design temperature for the service water system greatly reduce the effectiveness of the RHR heat exchangers. A clean heat exchanger is almost twice as effective as one with design fouling. Design temperature of the service water system is 95° F; however, the average recorded temperature for the month of August is 84° F. A record high river temperature of 88° F was recorded only twice; first occurring in August 1975, later in July 1977. CG&E has performed an in-depth study of RHR heat exchanger effectiveness, which can be found in Attachment 8A.

The technical specification values that maximize heat addition to the suppression pool are: The events begin with the minimum suppression pool volume allowed by the technical specifications. Initial pool temperature is assumed to the maximum allowed prior to reaching alarm setpoints. The operation manually scrams the reactor at 110° F pool temperature, which is the maximum allowed by the technical specifications. A maximum technical specification temperature of 120° F was chosen for the initiation of manual depressurization of the RPV. Justification for a manual scram of the reactor at 110° F pool temperature is provided in Attachment 8C.

No credit is taken for heat lost to the surroundings. The amount of energy required to heat up the submerged structures in the wetwell is neglected. In the small break case, the amount of energy required to heat up the drywell is all assumed to be directed into the suppression pool. Also, the energy "held up" in the drywell is ignored. Furthermore, the inventory in the feedwater system is assumed to remain at a constant temperature throughout the transient; i.e., no credit is taken for heat lost through the system boundary. 14

These cases bound all of the Chapter 15.0 accident analyses in the FSAR, primarily because only one heat exchanger is used in cases 1a, 2a, and 3a. Also, stored energy dumped into the the condenser is minimized due to spurious MSIV closure. See Attachment 8E a comparison to Chapter 15.0 events.

Using the conservative, nonmechanistic assumptions enumerated in this subsection, two of the Zimmer cases submitted slightly exceed the bulk pool temperature goal of 190° F (see Table 8.2-1).

CG&E has evaluated the options available which could be incorporated into the analysis in order to reduce the bulk pool temperatures. Options include: rapid depressurization of the RPV, mechanistic treatment of the feedwater system, or elimination of some of the extra conservatisms used in the analysis. In this evaluation, CG&E has performed sensitivity studies on feedwater coastdown rates, initial suppression pool temperature, service water temperature, suppression pool volume, RHR heat exchanger tube fouling, and the number of heat exchangers in service. The results of this study are in

Attachment 8A. CG&E has chosen not to implement any unnecessary rapid depressurizations by opening additional SRV's, because it violates technical specification maximum cooldown rates, increases challenges to the safety relief valves, induces a load transient on primary containment structures, and is not consistent with accepted operating procedures. It is not believed that rapid depressurization is a desirable method of reducing bulk pool temperature.

The final values listed in Table 8.2-1 are in accordance with "White Paper" (revision 1) developed under the direction of the Mass Energy Subcommittee, which was financed by the Mark II Owners Group. CG&E has attempted to work in accordance with the generic "White Paper" assumptions; however, in this analysis, feedwater provides makeup to the RPV, and the HPCS/RCIC is not utilized.

Results of the Zimmer specific cases which were performed by GC&E are also included in Attachment 8A for your information. Attachment 8A illustrates the degree of conservatism that was used in the Zimmer pool temperature calculations and further shows that the Zimmer results are within the limits specified by the NRC. 14

8.2.2 Temperature Response Analysis

This analysis was performed for the quencher SRV discharge device. Pool temperatures were calculated until a peak pool temperature was reached.

8.2.2.1 Model Description

Non-LOCA Events

To solve the transient response of the reactor vessel and supression pool temperature due to the postulated events, a coupled reactor vessel and suppression pool thermodynamic model was used. The model is based on the principles of conservation of mass and energy and accounts for any possible flow to and from the reactor vessel and the suppression pool.

The model incorporates a control volume approach for the reactor pressure vessel and suppression pool. It is capable of tracking a collapsed reactor vessel water level and having a rate of change of temperature or pressure imposed on it. The various modes of operation of the residual heat removal (RHR) system can be simulated, as well as the relief valves, HPCS, RCIC, and feedwater functions. The model also simulates system setpoints (automatic and manual) and operator actions and accepts as input the specific plant geometry and equiment capability.

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Small Break Accident Model

In the small break accident analysis, the mass and energy conservation laws are applied to a control volume which includes all of the reactor vessel contents and its walls. This control volume is subjected to the boundary conditions of decay heat input. The break and the safety/relief valve flow rates and the associated fluid enthalpies are derived from the state of fluid in the control volume undergoing the transient and the specified flow areas and locations.

The time-dependent break and safety/relief valve mass and energy flows are then input to another control volume containing the suppression pool. The pool temperature transient is obtained using the energy and mass balance equations on the suppression pool.

8.2.2.2 General Assumptions and Initial Conditions ("White Paper," Revision 1, Except As Noted By *)

The following common assumptions were used throughout the analysis of the ZPS-1 suppression pool temperature response:

a. Decay heat per ANS 5-20/10. 14

b. Design fouling of RHR heat exchangers.

c. Wetwell air temperature equal to the suppression pool water temperature.

*d. Feedwater in excess of instantaneous pool temperature is assumed to maintain level rather than condensate storage tank inventory via RCIC and HPCS. This assumption maximizes heat addition to the pool.

e. In calculating the overall heat transfer coefficient of the vessel wall and internal structures, it is assumed that the heat transfer is dominated by conduction. The heat transfer area of the reactor internals is obtained by assuming that the average metal thickness is 0.166 feet (2 inches).

f. The control volume of the reactor includes the reactor vessel, the recirculation lines, the feedwater lines from the vessel to the nearest feedwater heaters, and the steamlines from the vessel to the inboard main isolation valves (MSIV).

g. The initial water level in the reactor vessel is calculated based on the assumption that the voids in the two-phase region collapse. Therefore, the ECCS ON/OFF volumes are based on the total liquid volume of the reactor vessel, the feedwater lines, and the recirculation lines combined.

h. The specific heat of the reactor vessel and the internal is assumed to be 0.123 Btu/lbm/°F. The metal density is assumed to be 490 lbm/ft³.

i. A stuck-open relief valve can be detected and the corresponding quencher within the suppression chamber identified.

j. Additional safety/relief valves are manually opened as necessary to depressurize the reactor.

k. Minimum technical specification suppression pool water level.

l. Maximum suppression pool initial temperature which was 95° F during power operation and 120° F at hot standby.

m. ASME safety/relief valve flow rate rated at 122.5%.

n. No credit is taken for heat lost to the surroundings; i.e., all energy discharged from vessel is added to the suppression pool. 14

8.2.2.3 Description of Non-LOCA Events

This subsection describes the safety/relief valve discharges for non-LOCA events (Subsection 8.2.2.4 describes the LOCA event). A complete description of the sequence of events for all of the cases, i.e., Events 1a, 1b, 2a, 2b, 3a, and 3b is given in Tables 8.2-3 through 8.2-8.

8.2.2.3.1 SORV at Power

a. The SORV is the initiating event and two single failures are considered separately:

1. loss of one RHR HX, and

2. MSIV isolation signal at t = 0 minutes.

b. In accordance with the Technical Specifications, manual scram occurs at 110° F. Manual scram is accomplished by arming and depressing the four manual scram buttons and transferring the mode switch from "run" to "shutdown", in accordance with OP.EOP.01, Revision 5, in Attachment 8C.

c. Pool cooling initiated at t = 10 minutes.

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d. For a.1., main condenser remains available. See Attachment 8B.

e. For a.1., the operable RHR HX is placed in shutdown cooling mode. For a.2., two RHR HX are available and no shutdown cooling is used in the analysis.

8.2.2.3.2 Isolation/Scram

a. Isolation/scram is the initiating event and two single failures are considered separately:

1. loss of one RHR HX, and

2. spurious failure of a safety/relief valve in the open position (SORV).

b. Pool cooling initiated at t = 10 minutes.

c. A reactor depressurization is initiated at 120° F.

d. For a.2., the SORV is assumed to occur at t = 0 minutes.

e. For a.1., the operable RHR HX is placed in shutdown cooling mode. For a.2., two RHR HX are available and no shutdown cooling is used. 14

8.2.2.4 Small Break Accident

a. Two single failures considered separately:

1. loss of one RHR HX, and

2. loss of shutdown cooling mode.

b. SCRAM on high drywell pressure and MSIV closure signal assumed at t = 0 minutes.

c. At t = 10 minutes, pool cooling is initiated.

d. A reactor depressurization is initiated at 120° F.

e. For a.1., the operable RHR HX is placed in shutdown cooling mode. For a.2., two RHR HX are available and no shutdown cooling is used.

8.2.3 Results/Conculsions

The results obtained from the ZPS-1 suppression pool temperature analyses are depicted in Figures 8.2-8 through 8.2-13. Summary results are presented in Table 8.2-1. Conservative assumptions were used for transient events presented in this report. For example, maximum initial pool temperature, minimum initial pool mass, design fouling of RHR heat exchanger, continued addition of feedwater energy into the reactor vessel, and the initial reactor power corresponding up 102% of rated steam flow are conservative parameters that will affect the pool temperature.

For the case of a stuck-open-relief valve from power (Figures 8.2-8 and 8.2-9), the peak bulk pool temperatures are 179° F (loss of one RHR) and 179° F (spurious isolation).

The cases of isolation/scram are given in Figures 8.2-10 and 8.2-11. The peak bulk pool temperature for these cases are 192° F (loss of one RHR) and 186° F (SCRV).

To maximize pool temperature, SBA analyses were performed without actuation of ADS, as shown in Figures 8.2-12 and 8.2-13. The peak pool temperatures for these cases are 196° F (loss of one RHR) and 190° F (loss of shutdown cooling).

8.2.4 Summary

The initial Zimmer pool temperature calculations were performed in accordance with "Assumptions for Use in Analyzing Mark II BWR Suppression Pool Temperature Response to Plant Transients Involving Safety/Relief Valve Discharge-Revision 1," transmitted from Mr. Bucholz (GE), to Mr. Kniel (NRC), on January 9, 1981. These assumptions are commonly referred to as the Mass/Energy Subcommittee "White Paper" (Revision 1). 14

The calculations indicate that the bulk pool temperature is acceptable for all events considered, even though some very conservative nonmechanistic assumptions were used. Item d. of Subsection 8.2.2.2 states that, "Feedwater in excess of instantaneous pool temperature is assumed to maintain level rather than condensate storage tank inventory via RCIC and HPCS. This assumption maximizes heat addition to the pool." In this analysis, all the hot feedwater in the system, including feedwater heaters, is injected, as required, into the RPV. Furthermore, no credit is taken for heat escaping from the feedwater through the pipe walls in these transients, which may require 5 to 6 hours to reach maximum pool temperature.

The Zimmer station has two steam-driven turbine feedwater pumps. If treated mechanistically, these pumps would rapidly coast down to zero flow after MSIV closure. With feedwater no longer available for maintaining water level in the vessel, the ECCS pumps, taking suction from the condensate storage tanks or the suppression pool, would automatically initiate on level 2 and inject water into the RPV until vessel water level reaches level 8. This is considered mechanistic treatment of the feedwater system.

The obvious difference in heat addition between feedwater makeup (nonmechanistic) and HPCS (mechanistic) is very significant in the cases analyzed. Both CG&E and GE have run Zimmer cases utilizing mechanistic feedwater treatment, and the resulting bulk pool temperatures are well below the goal of 190° F in all cases analyzed.

In this analysis, no credit was taken for heat sinks in the drywell and the wetwell. In the SBA case, for example, all of the heat generated by the inventory leaving the small break is assumed to be discharged, via the LOCA downcomers, into the suppression pool. It is believed that it can be demonstrated that significant heat "hold up" in the drywell would occur in this type of accident. Also, the submerged structures in the wetwell, i.e., the downcomer bracing, pool liner, etc., would absorb some of the heat energy entering the wetwell. These items could significantly reduce the bulk pool temperature.

The Zimmer SRV discharge devices are T-quenchers. They are submerged 18.5 feet at minimum pool volume (LWL), which increases the saturation temperature at the quencher device to approximately 235° F. Test data indicates that the quencher demonstrates stability at mass fluxes corresponding to the Zimmer peak bulk pool temperatures. See Attachment 8F for the ZPS-1 saturation temperature at the quencher, and the corresponding increased local pool temperature limits. 14

CG&E has performed extensive studies on the Zimmer RHR heat exchangers. It is our intention to have regular maintenance performed on the RHR heat exchanger tube I.D to remove corrosion causing agents and to ensure that fouling is kept to a minimum. Discussions with various tube cleaning companies are in the preliminary stages. Our studies indicate that a relatively low fouling factor (10% of design) coupled with realistic but conservative service water temperature can reduce bulk pool temperature as much as 30° F. We have determined that regular cleaning of heat exchanger tubes will ensure a higher degree of heat exchanger efficiency. A clean heat exchanger is almost twice as effective in removing heat as one that has reached design fouling conditions.

Plant modifications are being made on ZPS-1 to ensure that no single failure will result in the loss of shutdown cooling and an RHR heat exchanger loop in pool cooling.

CG&E engineering and operating personnel are working together to ensure that emergency operating procedures are consistent with the assumptions used in this analysis. For your information, please find Reactor Scram Operating procedure OP.EOP.01 and Stuck Open Relief Valve operating procedure OP.EOP.30 in Attachment 8D.

The Zimmer in-plant tests will confirm the bulk to local temperature difference. The results described herein demonstrate conformance with acceptable quencher pool temperature limits. 14

In summary, CG&E is taking steps to increase RHR system availability and to ensure that operator actions and procedures are consistent with the assumptions used in this analysis. The ZPS-1 suppression pool temperature response for all cases analyzed is acceptable.

TABLE 8.2-1

POOL TEMPERATURE ANALYSIS RESULTS

	NRC CASE NUMBER	WHITE PAPER CASE NUMBER	WHITE PAPER ZIMMER BULK POOL PEAK TEMPERATURE	ATTACHMENT 8A ZIMMER UNIQUE TEMPERATURES
1. SORV at Power - Loss of 1 RHR Hx	b	1a	179° F	N/A
2. SORV at Power - Spurious Isolation	a	1b	179° F	N/A
3. Isolation/Scram - Loss of 1 RHR Hx	e	2a	192° F	184° F*
4. Isolation/Scram - SORV	e	2b	186° F	N/A
5. SBA-Loss of 1 RHR Hx	d	3a	196° F	181° F*
6. SBA-Shutdown Cooling Not Available	d	3b	190° F	N/A

* Assuming Feedwater coastdown to 0 flow occurs 30 seconds after SCRAM.

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TABLE 8.2-2

IMPORTANT SYSTEM CHARACTERISTICS

Initial Pool Mass	5.8×10^6 lbm	
Initial Pool Temperature	95° F	
Initial RPV Liquid Mass	457,300 lbm	
Initial RPV Steam Mass	17,800 lbm	
RPV and Internals Mass	2,479,000 lbm	
Initial Vessel Pressure	1,020 psia	
Initial Core Power (102% Rated)	2.35×10^6 Btu/sec	
Initial Steam Flow (102% Rated)	2,968 lbm/sec	
Initial CRD Flow	8.33 lbm/sec	
CRD Flow After Scram ($P_{RPV}=0_{psig}$)	23.6 lbm/sec	
CRD Enthalpy (From CSD)	68 Btu/lbm	
HPCS On Volume	8,566 ft^3	
HPCS Off Volume	10,265 ft^3	
Vessel MAX P For Shutdown Cooling	150 psia	
RHR K In Shutdown Cooling	190 Btu/sec°F	
RHR In Pool Cooling	190 Btu/sec°F	
RHR Flow Rate In Pool Cooling	702 lbm/sec	
RHR Flow Rate In Shutdown Cooling	702 lbm/sec	
S/RV Flow (122.5% ASME)	P psia	FLOW lbm/sec
	0	0
	1500	368
Feedwater	MASS (lbm)	ENTHALPY (pipe and fluid) (Btu/lbm)
	226,030	360
	423,130	268
	110,340	221

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TABLE 8.2-2 (Cont'd)

REACTOR POWER DECAY

TIME (seconds)	POWER DECAY (fraction of 1)
0	1.084
2	0.5026
6	0.6271
10	0.5249
20	0.2309
30	0.1372
31	0.1370
60	0.0492
100	0.0427
120	0.0400
121	0.0390
200	0.0358
600	0.0279
1000	0.0245
1001	0.0244
2000	0.0192
6000	0.0138
10,000	0.0120
20,000	0.0101
6×10^4	.00739
1×10^5	.00624
2×10^5	.00512
1×10^6	.00298
4×10^6	.00180

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TABLE 8.2-3

POOL TEMPERATURE CONDITIONS - CASE 1a

NRC Case b

SORV at full power, 1 RHR available

Manual Scram at Tpool = 110° F.

Mechanistic closure of the turbine stop and bypass valves (Product Line Unique - BWR-4 or 5).

One RHR in pool cooling 10 minutes after high temperature alarm. 14

Main condenser reestablished through bypass system 20 minutes after scram using plant specified bypass capacity.

Main condenser available using full bypass capacity until reactor vessel permissive for RHR shutdown cooling.

RHR out-of-pool cooling when pressure permissive for RHR shutdown cooling is reached. Sixteen-minute delay for RHR transfer to shutdown cooling. (Additional SRV's opened, as required, during switchover to ensure no repressurization during switchover.)

TABLE 8.2-4

POOL TEMPERATURE CONDITIONS - CASE 1b

NRC Case a

SORV at full power, 2 RHR's available

Manual Scram at Tpool = 110° F. 14

Nonmechanistic isolation at scram, with 3.5 seconds main isolation valve closure.

Two RHR's in pool cooling 10 minutes after high pool temperature alarm.

No manual depressurization is required. Depressurization rate is controlled by SORV only.

RHR shutdown cooling not initiated.

TABLE 8.2-5

POOL TEMPERATURE CONDITIONS - CASE 2a

NRC Case e

Isolation-Scram (nonmechanistic), 1 RHR available

Isolation Scram at t = o, nonmechanistic, with 3.5 seconds main isolation valve closure.

One RHR in pool cooling 10 minutes after the event. 14

When Tpool = 120° F, begin manual depressurization by opening additional valves as needed. Depressurize at 100° F/hr.

RHR out-of-pool cooling when pressure permissive for RHR shutdown cooling is reached. Sixteen-minute delay for RHR transfer to shutdown cooling. (Additional SRV's opened, as required, during switchover to ensure no repressurization during switchover.)

TABLE 8.2-6

POOL TEMPERATURE CONDITIONS - CASE 2b

NRC Case e

Isolation Scram (nonmechanistic), 2 RHR's available

Isolation Scram at t = o nonmechanistic, with 3.5 seconds main isolation valve closure. 14

SORV at t = o.

Two RHR's in Pool Cooling at 10 minutes after the event.

When Tpool = 120° F, begin manual depressurization by opening additional valves. Depressurize at 100° F.

RHR shutdown cooling not initiated.

TABLE 8.2-7

POOL TEMPERATURE CONDITIONS - CASE 3a

NRC Case d

SBA Event Mode, 1 RHR Available

Scram at t = o on high drywell pressure.

Isolation at t = o (nonmechanistic), with 3.5 seconds main isolation valve closure.

One RHR in pool cooling 10 minutes after high pool temperature alarm.

When Tpool = 120° F, begin manual depressurization by opening additional SRV's as needed. Depressurize at 100° F/hr. 14

RHR out-of-pool cooling when pressure permissive for RHR shutdown cooling is reached. Sixteen-minute delay for RHR transfer to shutdown cooling. (SRV's opened, as required, during switchover to ensure no repressurization during switchover.)

Automatic RHR switchover to the LPCI mode occurs when the high drywell pressure is reached. Therefore, RHR pool cooling is assumed to be unavailable for the first 10 minutes after scram.

TABLE 8.2-8

POOL TEMPERATURE CONDITIONS - CASE 3b

NRC Case d

SBA Event, 2 RHR's Available

Scram at t = o on high drywell pressure.

Isolation at t = o (nonmechanistic), with 3.5 seconds main isolation valve closure. 14

Two RHR's in pool cooling 10 minutes after high pool temperature alarm.

When Tpool = 120° F, begin manual depressurization by opening SRV's as needed. Depressurize at 100° F/hr.

RHR shutdown cooling not initiated.

Automatic RHR switchover to the LPCI mode occurs when the high drywell pressure is reached. Therefore, RHR pool cooling is assumed to be unavailable for the first 10 minutes after scram.

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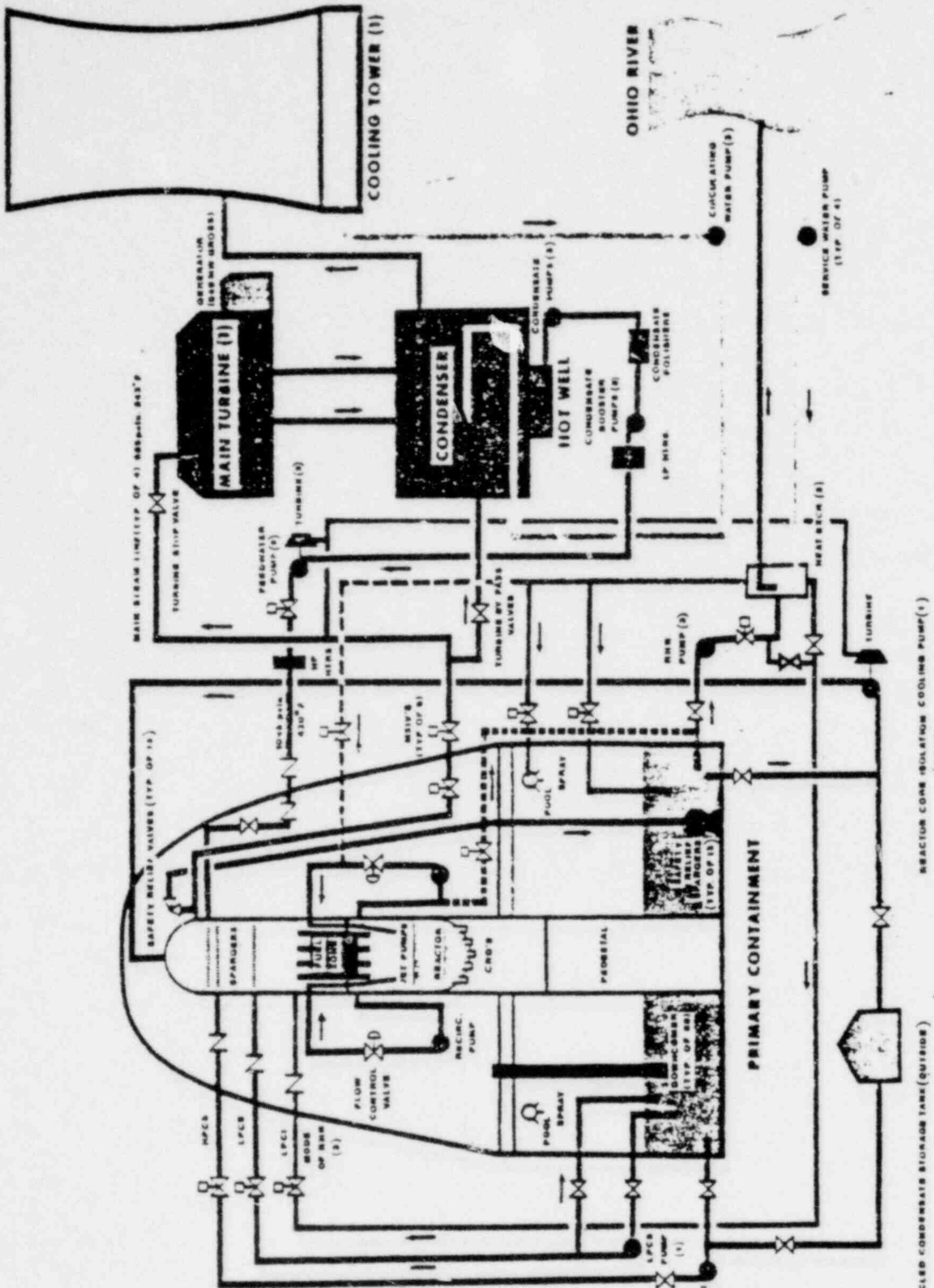

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FIGURE 8.2-1

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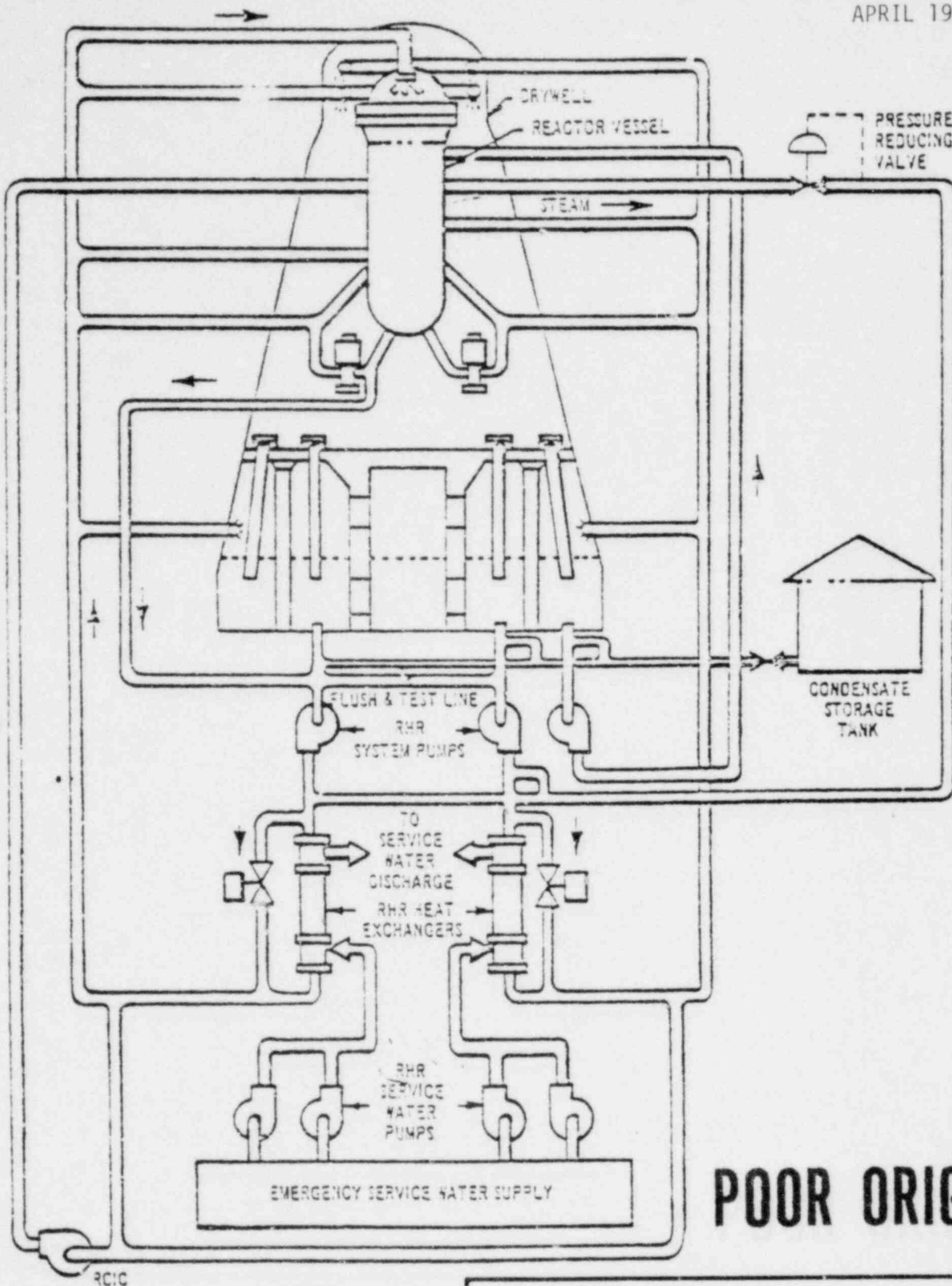
POOR ORIGINAL



POOR ORIGINAL

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FIGURE 8.2-2

RESIDUAL HEAT REMOVAL SYSTEM

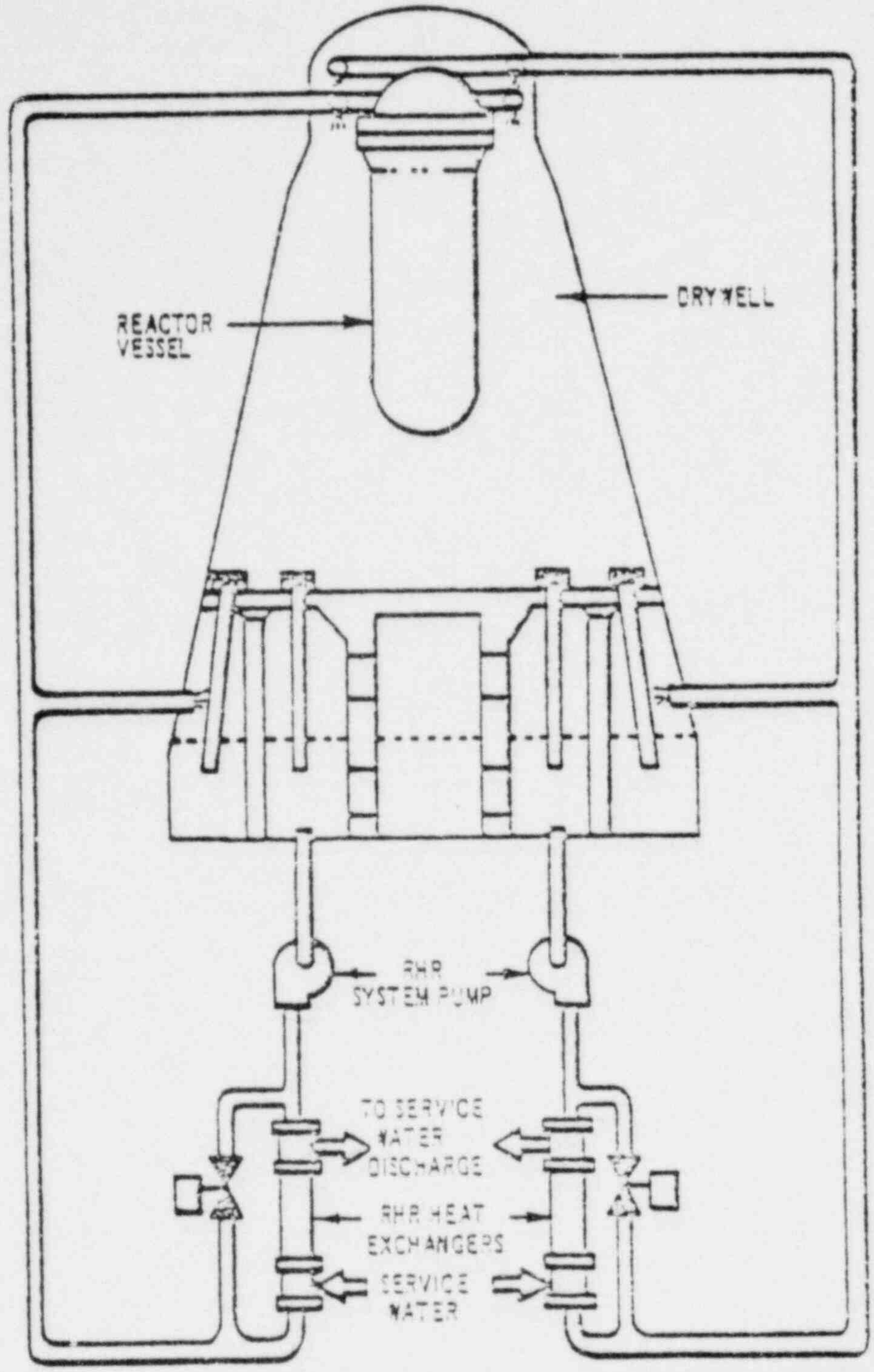

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FIGURE 8.2-3

RESIDUAL HEAT REMOVAL SYSTEM
CONTAINMENT COOLING MODE

POOR ORIGINAL

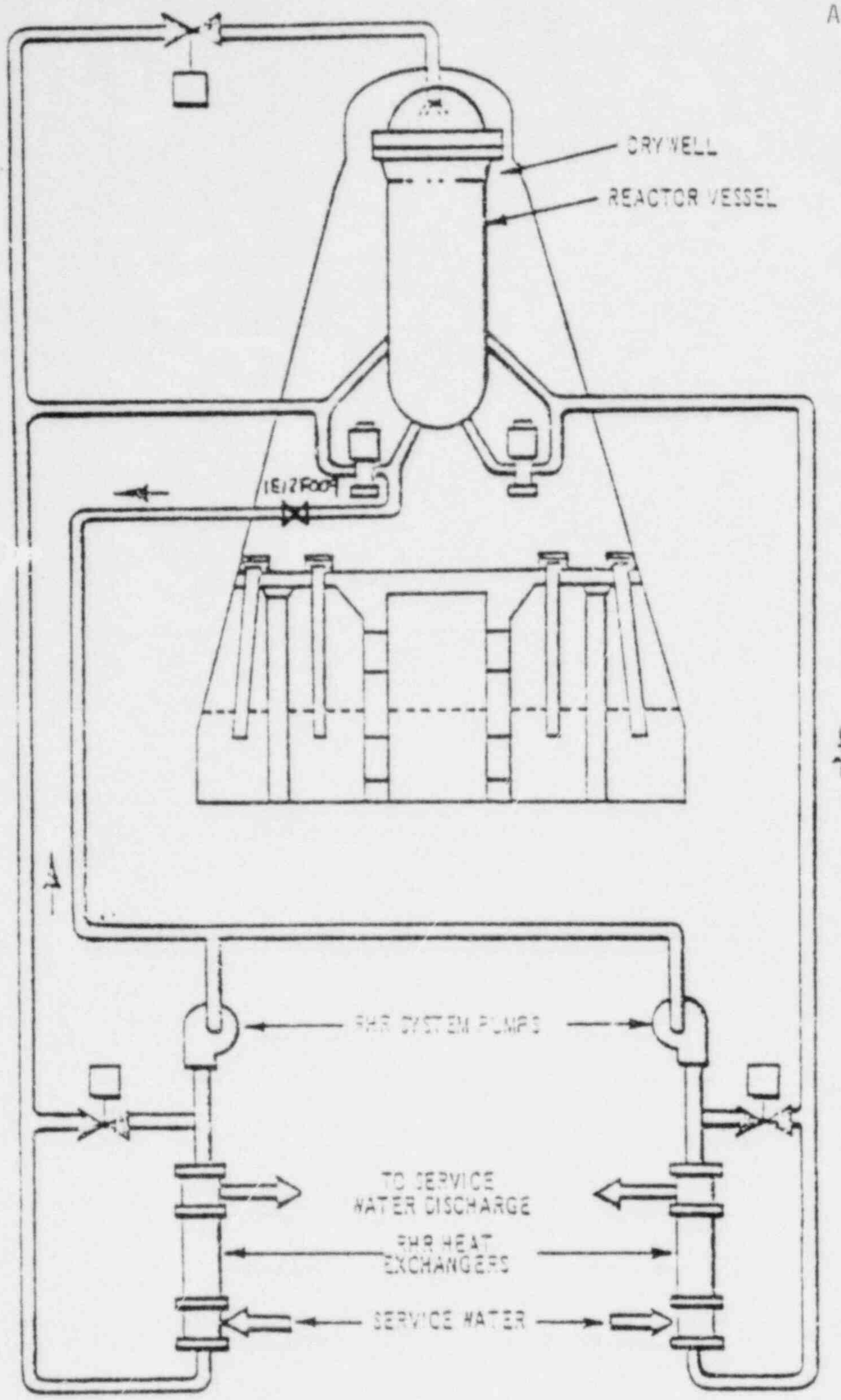

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FIGURE 8.2-4

RESIDUAL HEAT REMOVAL SYSTEM
SHUTDOWN COOLING MODE

POOR ORIGINAL

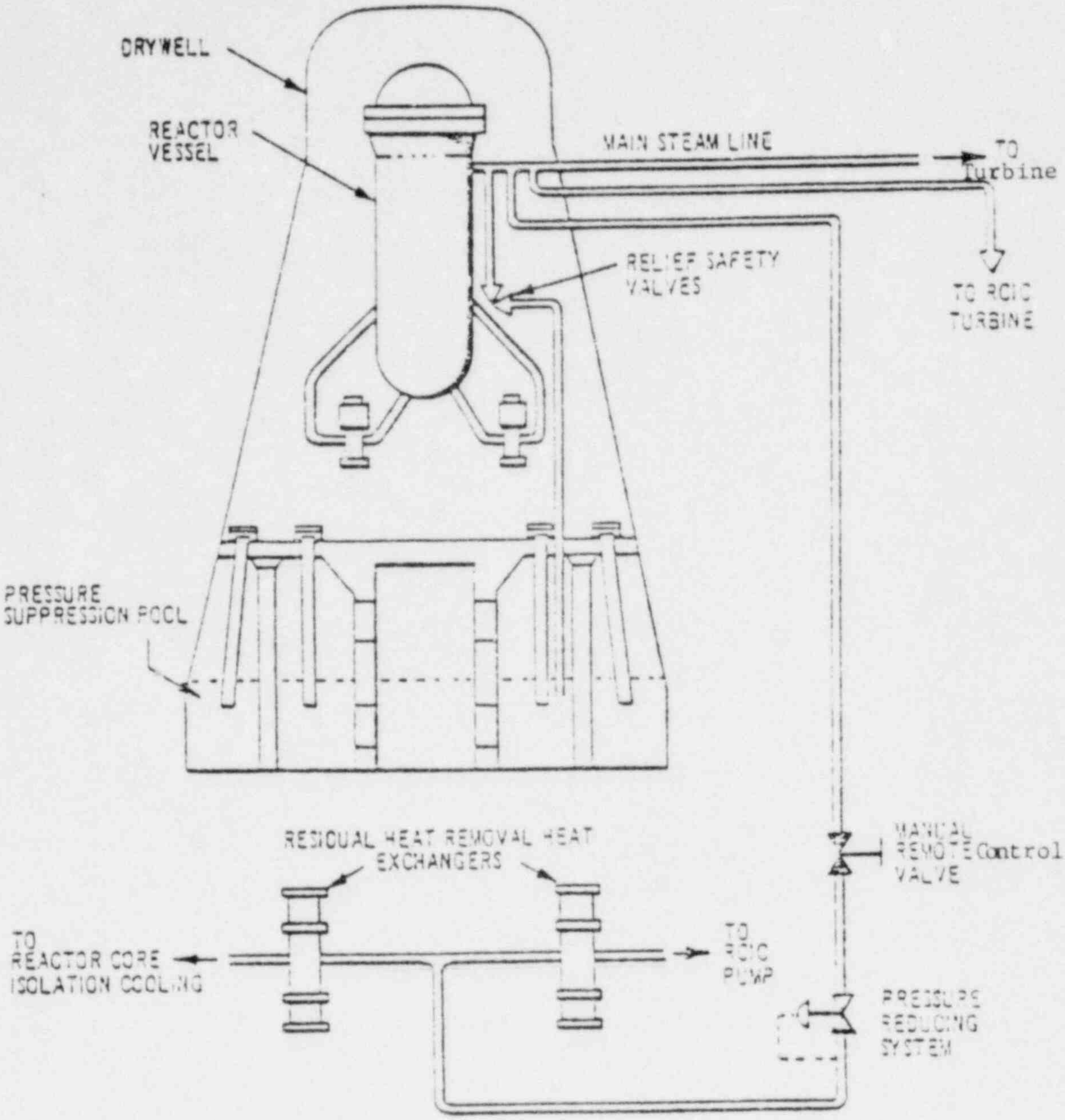

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FIGURE 8.2-5 RESIDUAL HEAT REMOVAL SYSTEM HOT STANDBY MODE

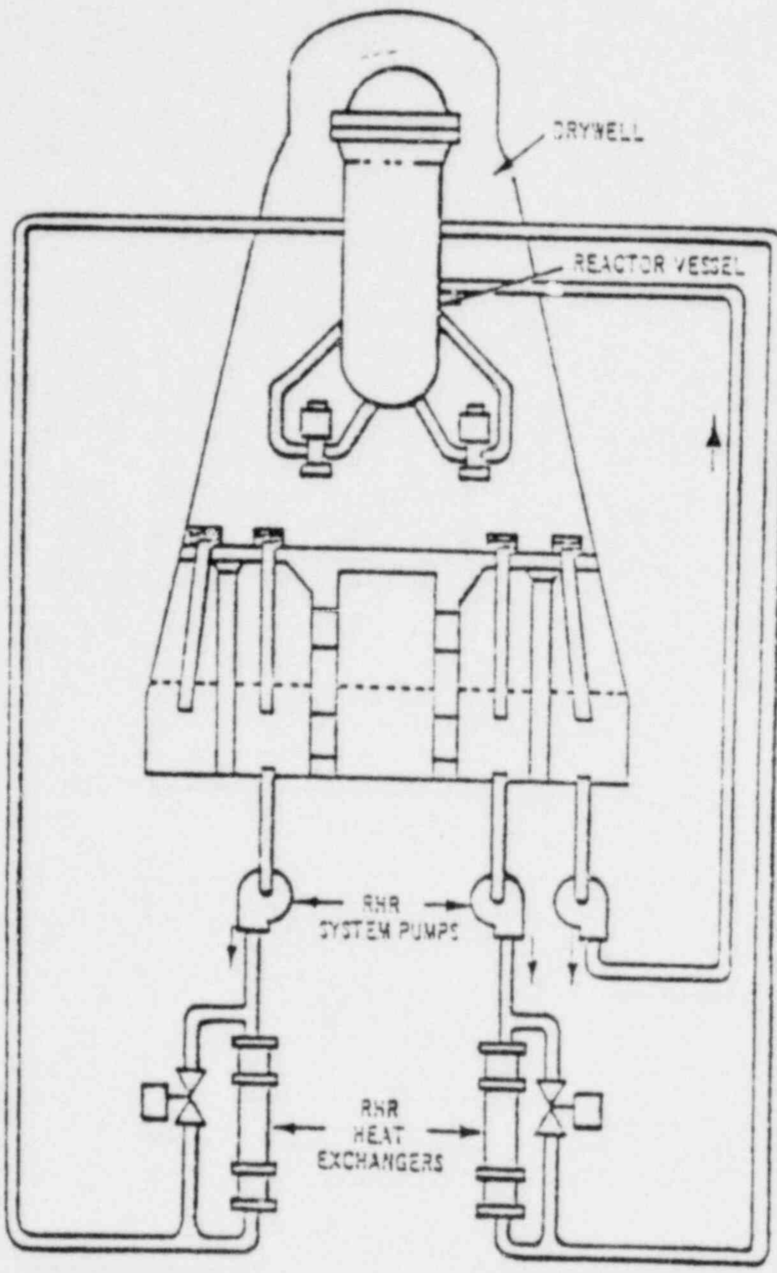

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FIGURE 8.2-6

RESIDUAL HEAT REMOVAL SYSTEM
LOW PRESSURE COOLANT INJECTION MODE

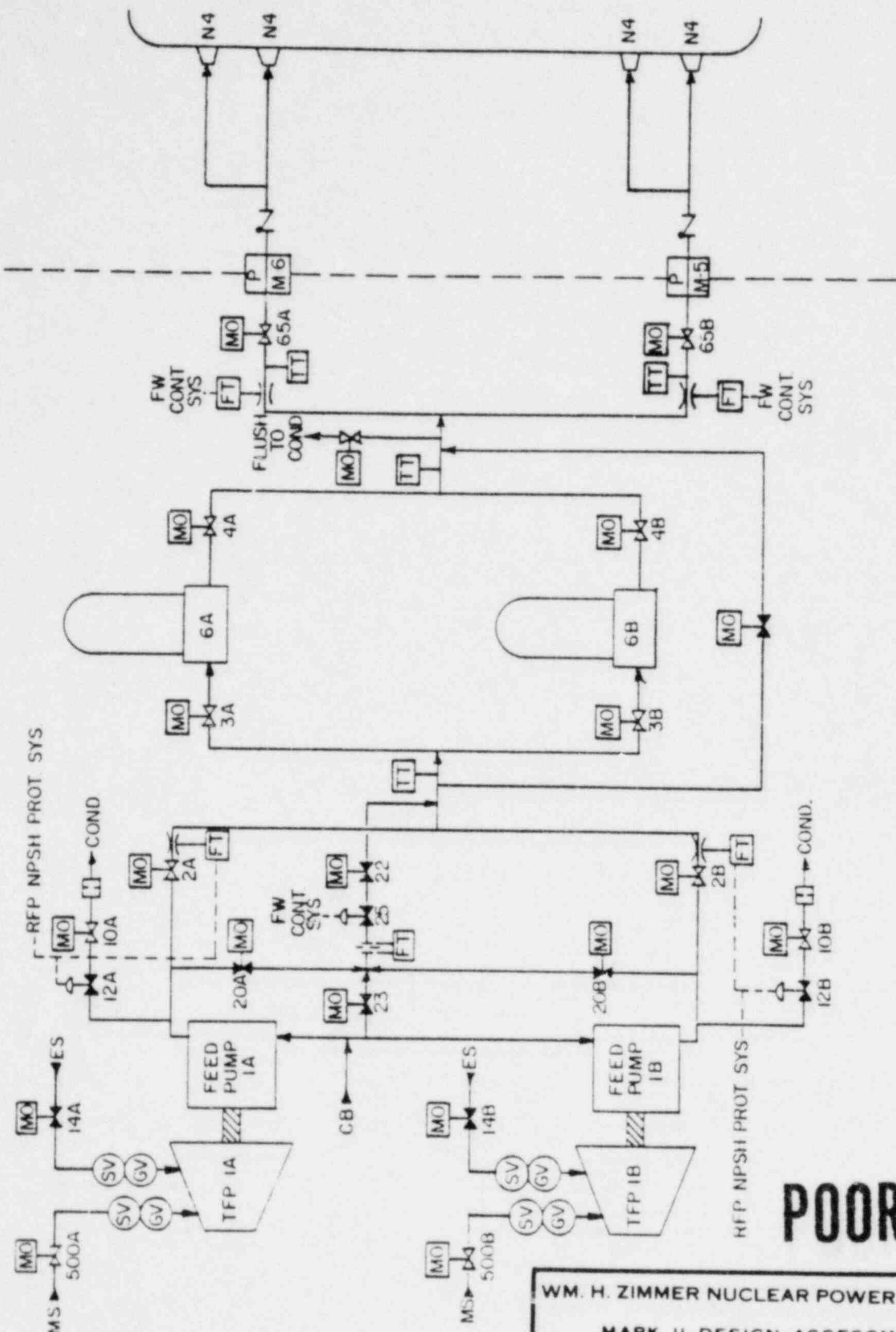

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FIGURE 8.2-7

FEEDWATER SYSTEM (FW)

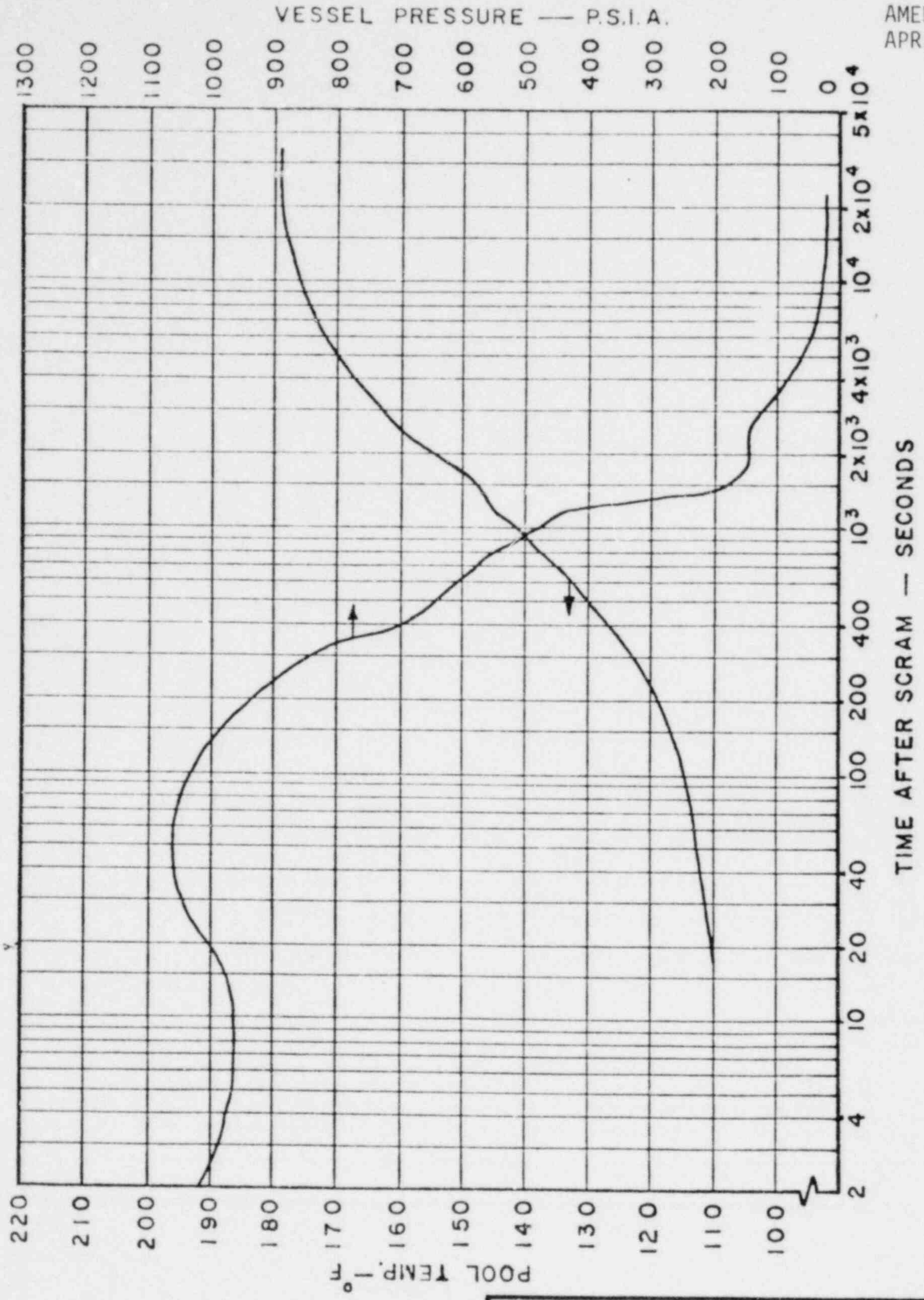

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FIGURE 8.2-8

POOL TEMPERATURE RESPONSE - CASE 1A SORV AT FULL POWER, 1 RHR AVAILABLE

POOR ORIGINAL

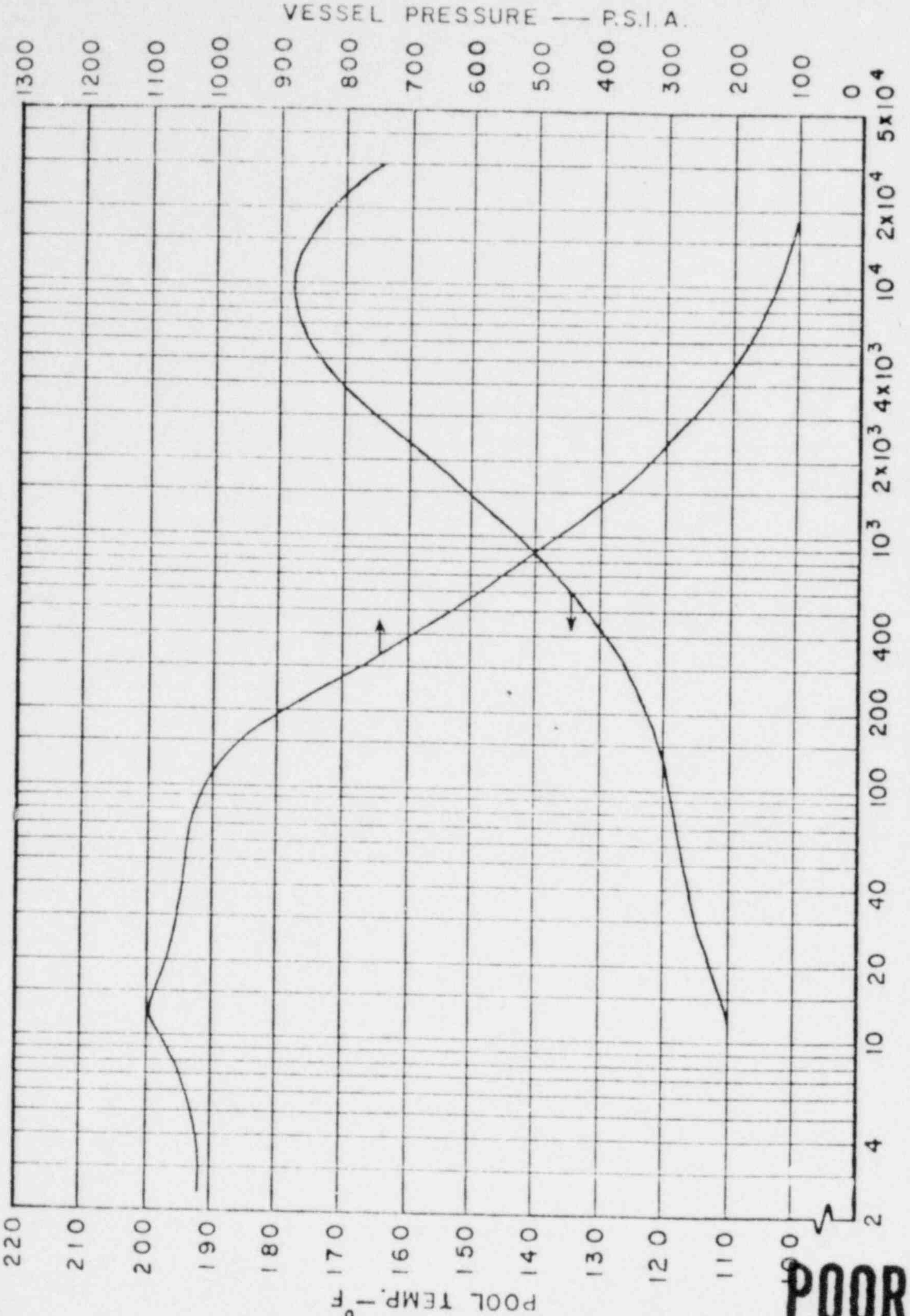

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FIGURE 8.2-9

POOL TEMPERATURE RESPONSE -
CASE 1B SORV AT FULL POWER,
2 RHR'S AVAILABLE

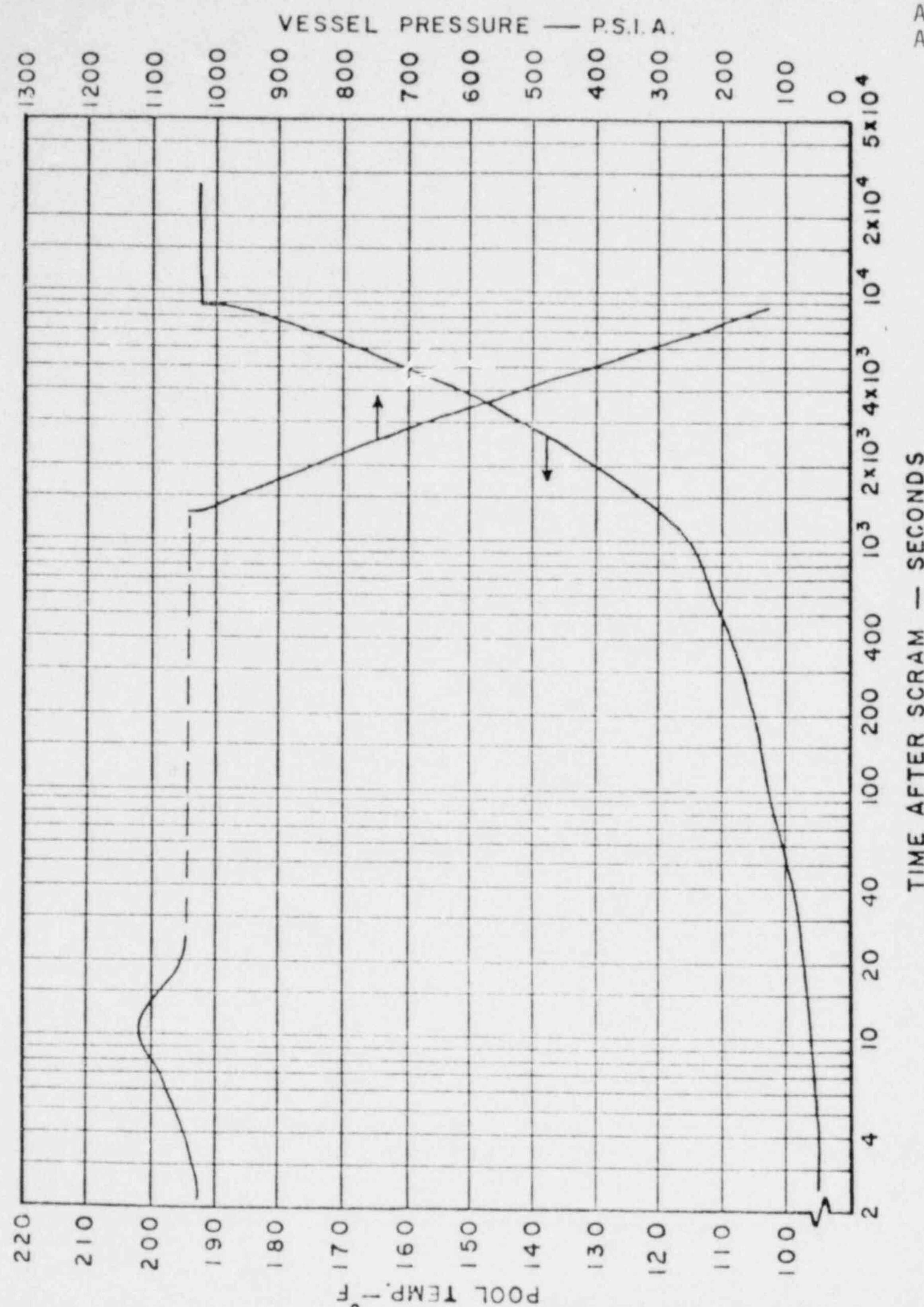

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FIGURE 8.2-10

POOL TEMPERATURE RESPONSE -
CASE 2A ISOLATION/SCRAM,
1 RHR AVAILABLE

POOR ORIGINAL

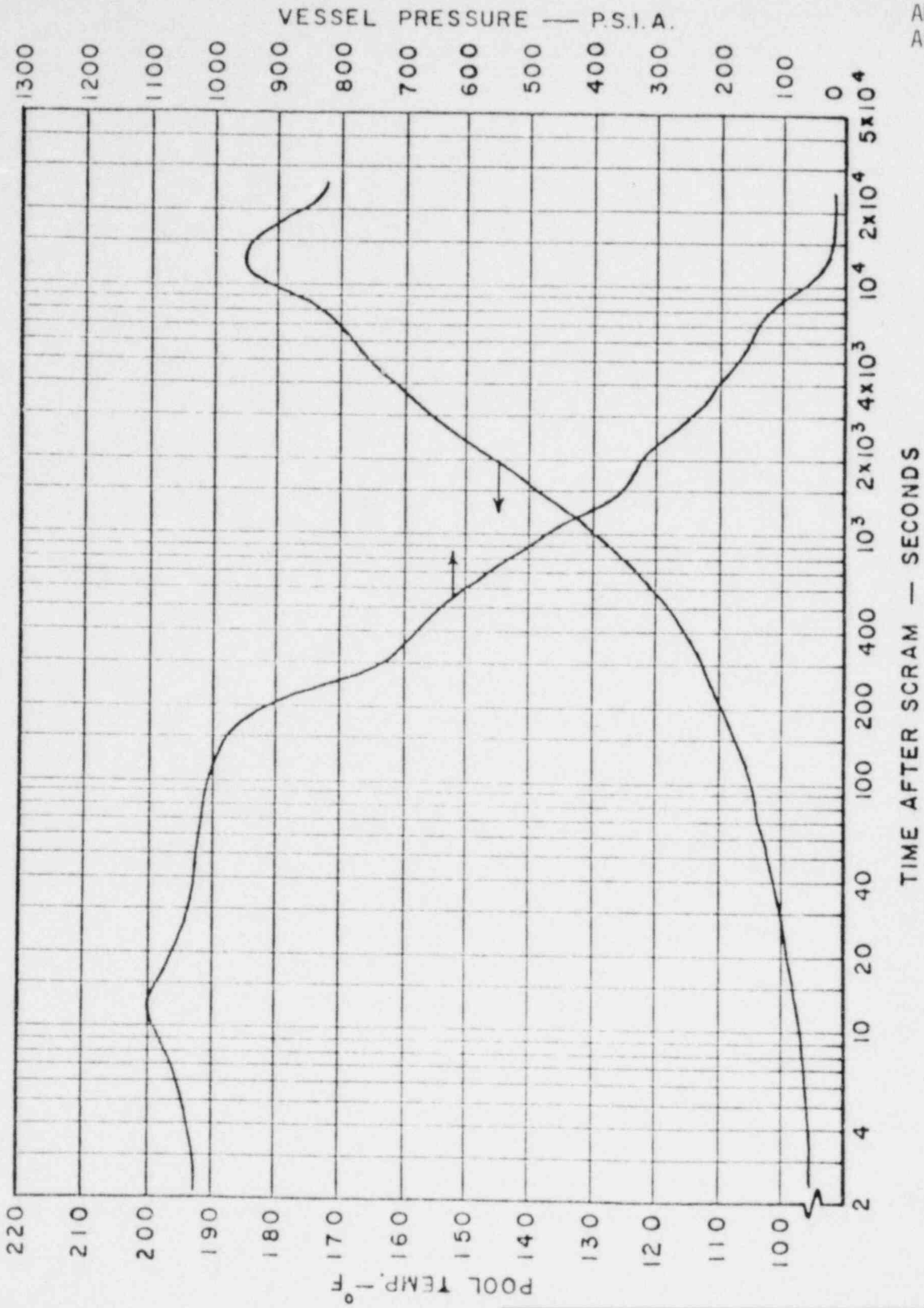

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FIGURE 8.2-11 POOL TEMPERATURE RESPONSE - CASE 2B ISOLATION/SCRAM, 2 RHR'S AVAILABLE

POOR ORIGINAL

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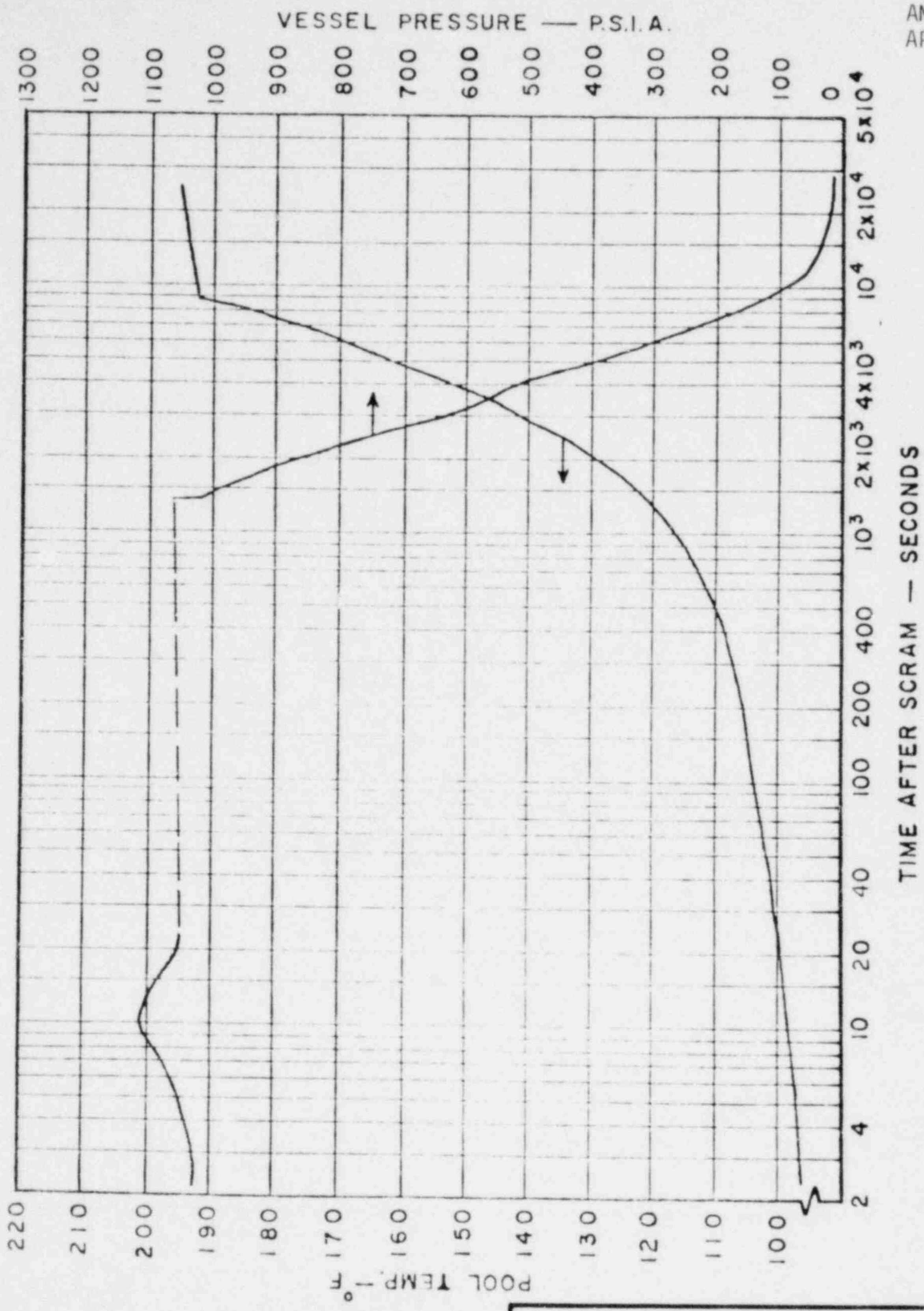

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FIGURE 8.2-12

POOL TEMPERATURE RESPONSE -
CASE 3A SBA,
1 RHR AVAILABLE

POOR ORIGINAL

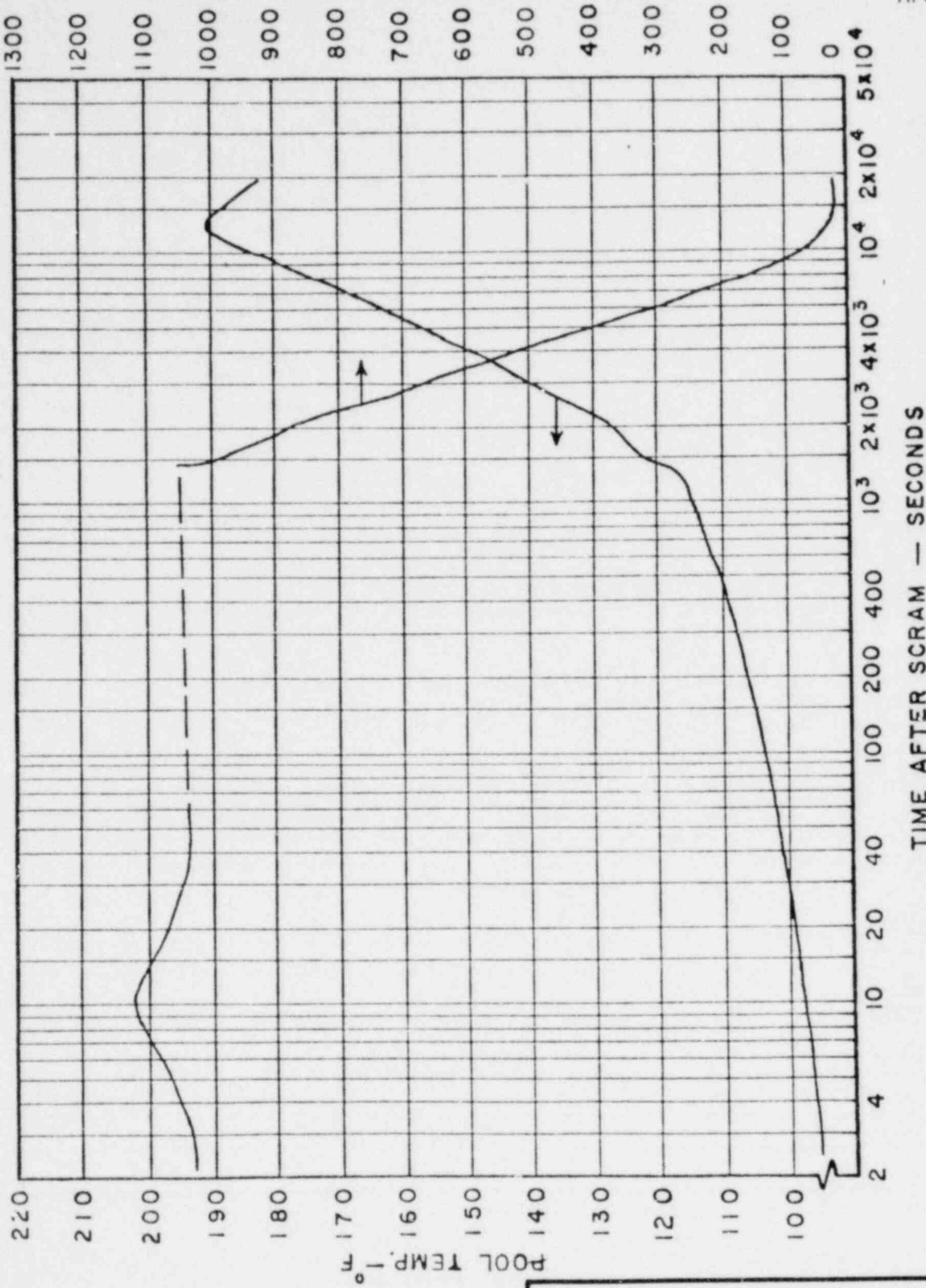

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FIGURE 8.2-13

POOL TEMPERATURE RESPONSE -
CASE 3A SBA,
2 RHR'S AVAILABLE

ATTACHMENT 8A

ATTACHMENT 8A

The purpose of Attachment 8A is to demonstrate the contribution made by various conservatisms used in the Zimmer pool temperature submittal, based on an independent study by CG&E Company. The two primary items examined in this study are the feedwater system and RHR heat exchanger performance.

Feedwater

Original cases 2a - Isolation Scram, and 3a - Small Break, were chosen for the feedwater sensitivity study because these two cases slightly exceed the maximum desired bulk pool temperature of 190° F. This sensitivity study concludes that if the feedwater system is treated mechanistically, bulk pool temperatures are well below the 190° F goal. See Table 8A-1 for results of this sensitivity study. The mechanistic assumptions used for this study are consistent with Zimmer plant operating procedures, system performance, and Mass Energy "White Paper", revision 1, transmitted from Mr. Buchholz (G.E.) to Mr. Kniel (NRC) on January 9, 1981. 14

Heat Exchanger Performance

Original case 3b - Small Break, was chosen for this sensitivity study because two heat exchangers are in service in the pool cooling mode throughout the event. This study demonstrates that bulk pool temperature is very sensitive to the degree of tube fouling used, and the number of heat exchangers in service. It also demonstrates that bulk pool temperature for this event is well below the goal of 190° F using one heat exchanger only in the pool cooling mode, provided the tube fouling does not exceed 10% of design fouling, and the service water is reduced to the maximum recorded temperature. See Table 8A-2 for results of this sensitivity study.

TABLE 8A-1

FEEDWATER SENSITIVITY STUDY CASES 2a and 3a

CASE	DESCRIPTION	FEEDWATER COASTDOWN RATE (Seconds)	BULK POOL TEMPERATURE °F	GOAL °F	COMMENTS
2a	Isolation Scram, 1 Hx, 100° F/Hr Cooldown	*N/A (Nonmechanistic)	192	190	White Paper Case 2a
2a	Isolation Scram, 1 Hx, 100° F/Hr Cooldown	7 Mechanistic	183.25	190	See Figure 8A-1
2a	Isolation Scram, 1 Hx, 100° F/Hr Cooldown	15 Mechanistic	183.5	190	See Figure 8A-2
2a	Isolation Scram, 1 Hx, 100° F/Hr Cooldown	30 Mechanistic	183.96	190	See Figure 8A-3
2a	Isolation Scram, 1 Hx, 100° F/Hr Cooldown	60 Mechanistic	184.8	190	See Figure 8A-4
3a	Small Break, 1 Hx, 100° F/Hr Cooldown	*N/A (Nonmechanistic)	196	190	White Paper Case 3a
3a	Small Break, 1 Hx, 100° F/Hr Cooldown	7 Mechanistic	181.7	190	See Figure 8A-5
3a	Small Break, 1 Hx, 100° F/Hr Cooldown	15 Mechanistic	182.94	190	See Figure 8A-6
3a	Small Break, 1 Hx, 100° F/Hr Cooldown	30 Mechanistic	181.18	190	See Figure 8A-7
3a	Small Break, 1 Hx, 100° F/Hr Cooldown	60 Mechanistic	178.9	190	See Figure 8A-8

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* Feedwater is assumed to be available throughout the transient and maintains vessel water level rather than HPCS or RCIC.

TABLE 8A-2

HEAT EXCHANGER SENSITIVITY STUDY - CASE 3b
(Pool Cooling Mode)

Case	Suppression Pool Mass lbm	Service Water Temperature °F	Hx "K" BTU/sec °F	Hx In Service	Bulk Pool Temperature °F*	°F Change From Base Case	Variance
3b base	5.80E6	95 (design)	190	2	190	White Paper Case 3b	
1	5.80E6	95 (design)	198 (Design Fouling w/5% Plugged)	1	207.8	Base case in sensitivity study	
2	5.97E6	95 (design)	198	1	206	-1.8	Pool Level
3	5.80E6	88 (hi record)	198	1	204.5	-3.3	S.W. Temperature
4	5.80E6	84 (avg. max. Aug.)	198	1	202.7	-5.1	S.W. Temperature
5	5.80E6	60 (avg. yearly)	198	1	192	-15.8	S.W. Temperature
6	5.80E6	95 (design)	208 (Design Fouling)	1	205.7	-2.1	No Plugged Tubes
7	5.80E6	95 (design)	248 (50% Design Fouling)	1	198.2	-9.6	50% Design Fouling
8	5.80E6	95 (design)	295 (10% Design Fouling)	1	190.9	-16.9	10% Design Fouling
9	5.80E6	95 (design)	341 (No Fouling)	1	184.6	-23.2	No Fouling
10	5.80E6	95 (design)	323 (No Fouling, 5% Plugged)	1	187.0	-20.8	5% Plugged

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TABLE 8A-2 (Cont'd)

Case	Suppression Pool Mass lbm	Service Water Temperature °F	Hx "K" BTU/sec °F	Hx In Service	Bulk Pool Temperature °F*	°F Change From Base Case	Variance
11	5.80E6	95 (design)	280 (10% Design Fouling, 5% Plugged)	1	193.1	-14.7	10% Fouling
12	5.80E6	95 (design)	236 (50% Design Fouling, 5% Plugged)	1	200.3	-7.5	50% Fouling
13	5.97E6	88 (hi record)	295 (10% Design Fouling)	1	185.7	-22.1	Realistic case 1**
14	5.97E6	84 (avg. max. Aug.)	295 (10% Design Fouling)	1	183.6	-24.2	Realistic case 2**
15	5.97E6	60 (avg. yearly)	295 (10% Design Fouling)	1	172.4	-35.2	Realistic case 3**
16	5.80E6	95 (design)	198 (Design Fouling w/5% Plugged)	2	179.5	-28.3	Add 1 Hx
17	5.80E6	95 (design)	208 (Design Fouling)	2	177.9	-29.9	No Plugged Tubes
18	5.80E6	95 (design)	248 (50% Design Fouling)	2	171.9	-35.9	50% Design Fouling
19	5.80E6	95 (design)	295 (10% Design Fouling)	2	165.8	-42	10% Design Fouling
20	5.80E6	95 (design)	341 (No Fouling)	2	160.5	-47.3	No Fouling
21	5.80E6	95 (design)	323 (No Fouling, 5% Plugged)	2	162.5	-45.3	No Fouling, 5% Plugged

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TABLE 8A-2 (Cont'd)

Case	Suppression Pool Mass lbm	Service Water Temperature °F	Hx "K" BTU/sec °F	Hx In Service	Bulk Pool Temperature °F*	°F Change From Base Case	Variance
22	5.80E6	95 (design)	280 (10% Fouling, 5% Plugged)	2	167.6	-40.2	10% Fouling, 5% Plugged
23	5.80E6	95 (design)	236 (50% Fouling, 5% Plugged)	2	173.6	-34.2	50% Fouling, 5% Plugged

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* Desired maximum bulk pool temperature is 190° F.

** Cases 13, 14, and 15 use estimates of heat exchanger fouling after 1 year of intermittent operation, assuming units are flushed and laid up with demineralized water after each usage.

CONCLUSIONS

1. Pool volume changes do not significantly change bulk pool temperatures.
2. Changes in service water design temperature do not significantly affect bulk pool temperature.
3. Heat exchanger fouling is a very significant factor.
4. The number of heat exchangers available is very significant.

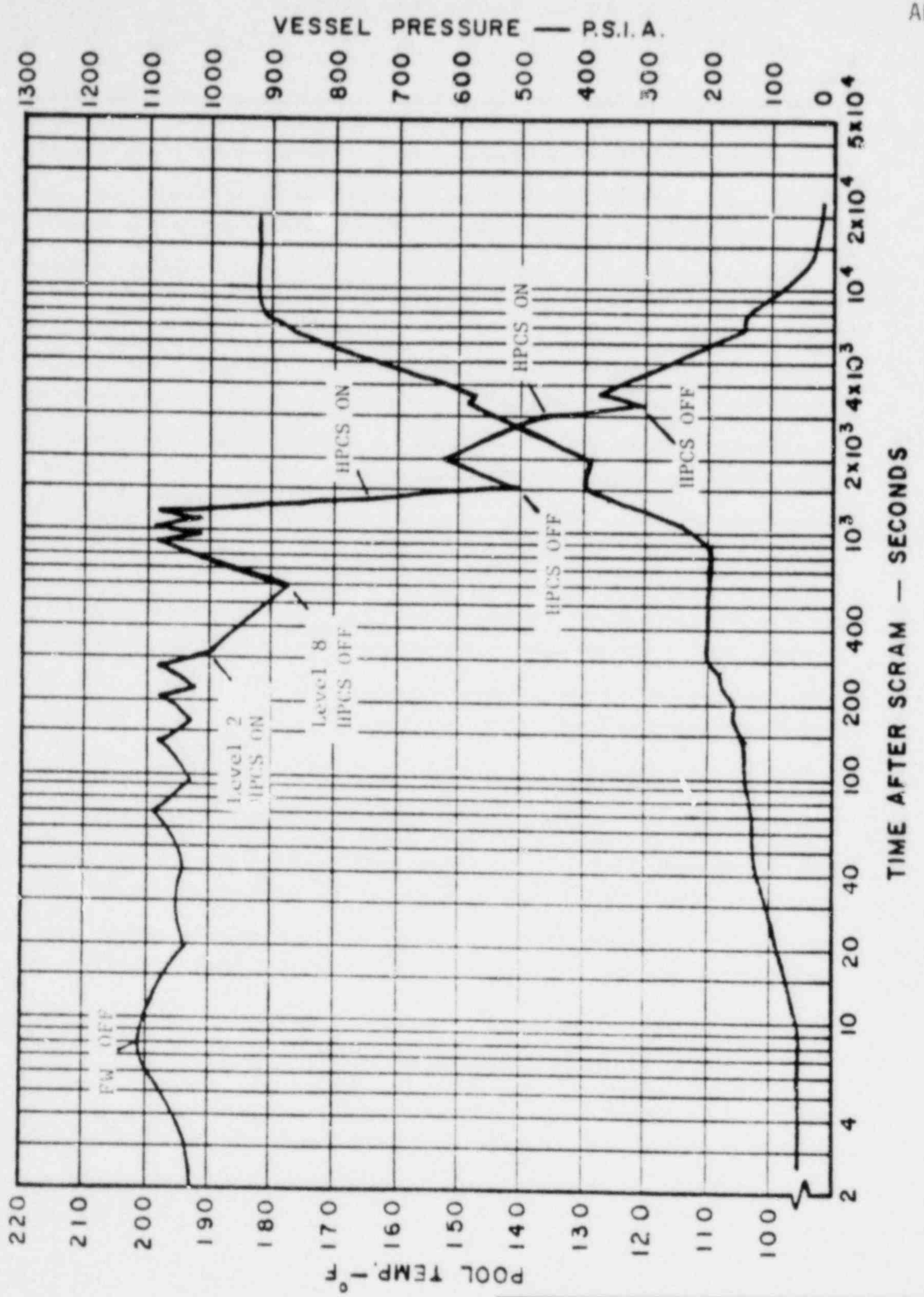

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FIGURE 8A-1

FEEDWATER SENSITIVITY STUDY
7 SEC. COASTDOWN CASE 2a,
1 RHR AVAILABLE

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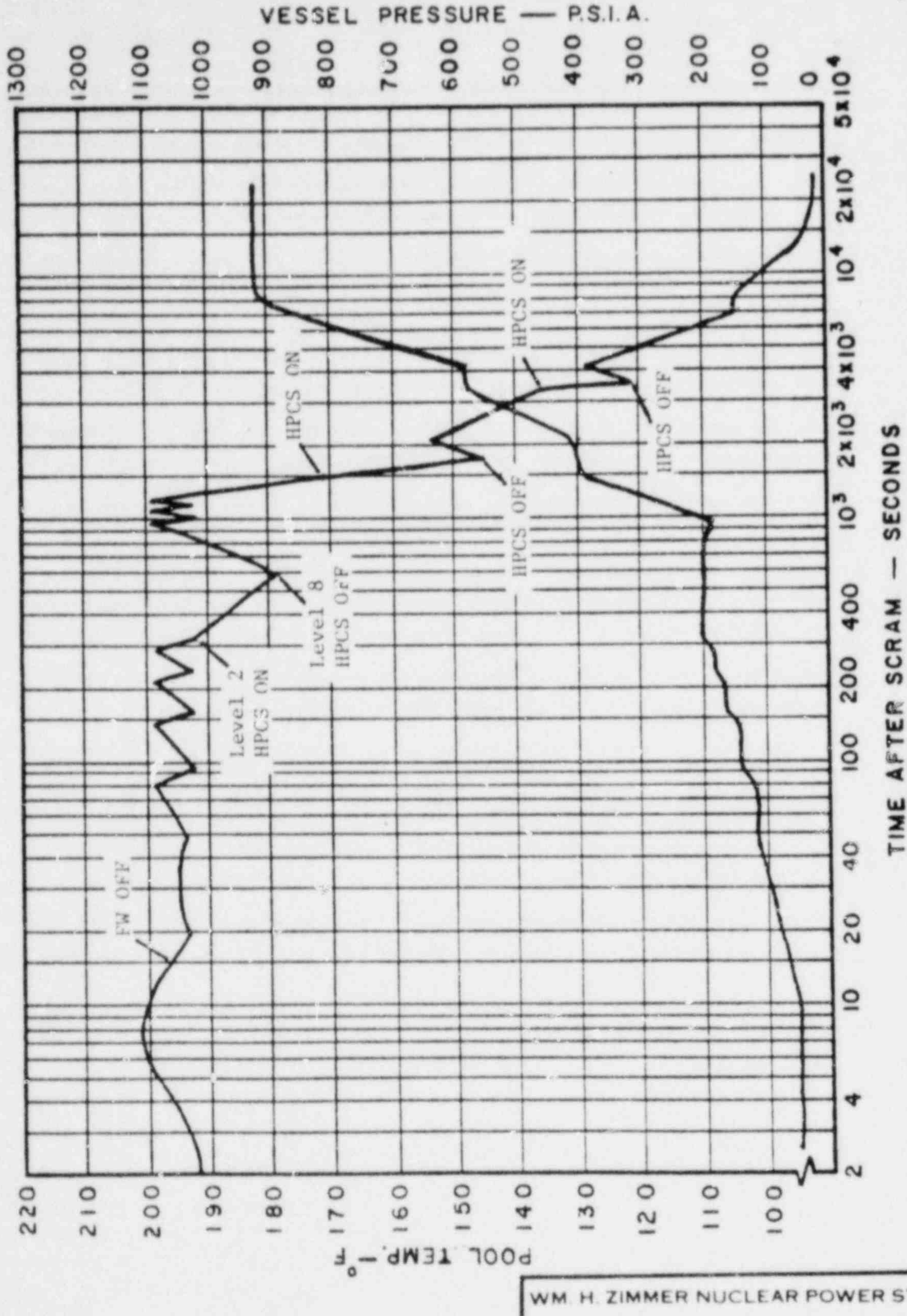

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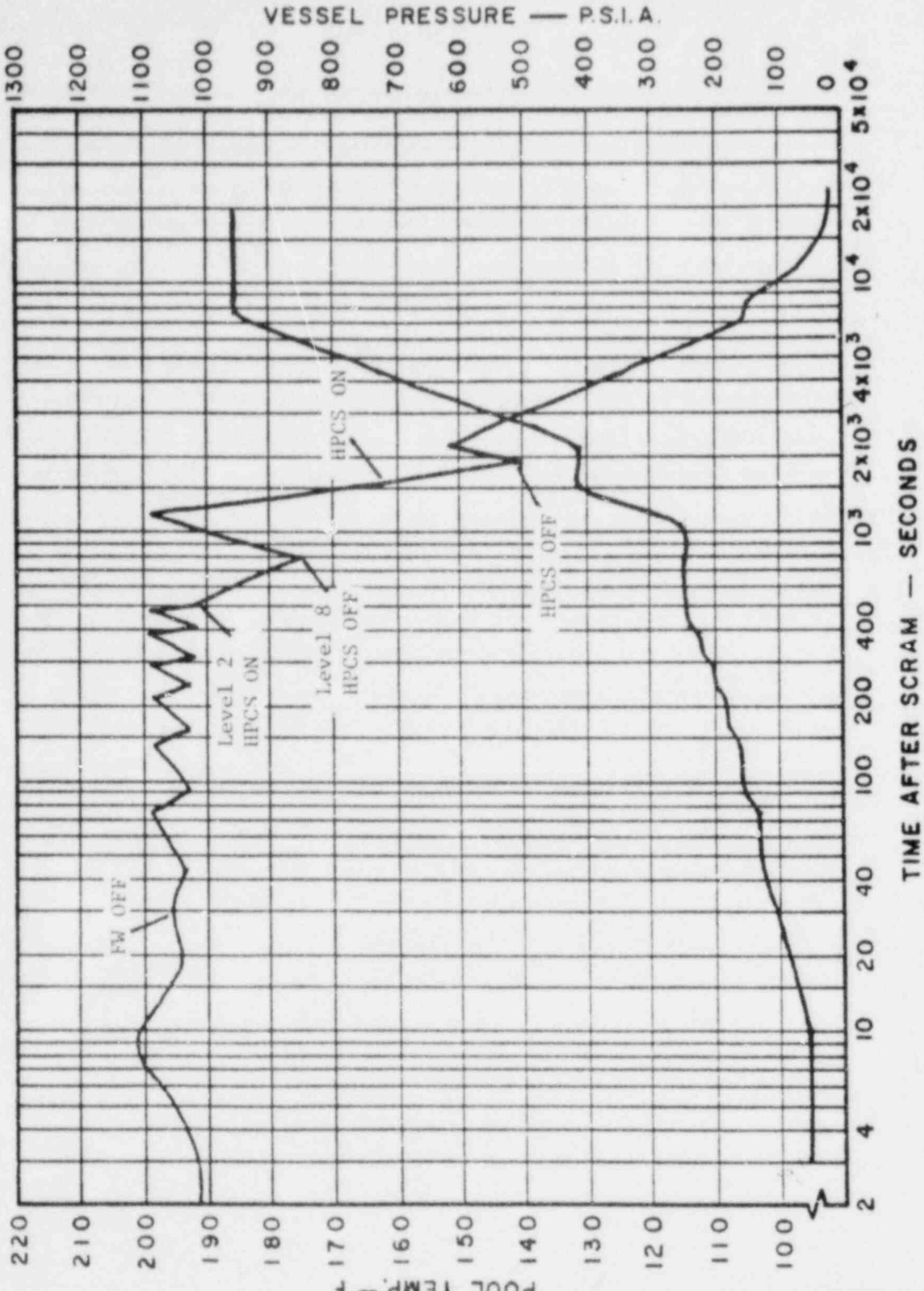
FIGURE 8A-2

FEEDWATER SENSITIVITY STUDY
15 SEC. COASTDOWN CASE 2a,
1 RHR AVAILABLE

POOR ORIGINAL

AMENDMENT 14
APRIL 1981



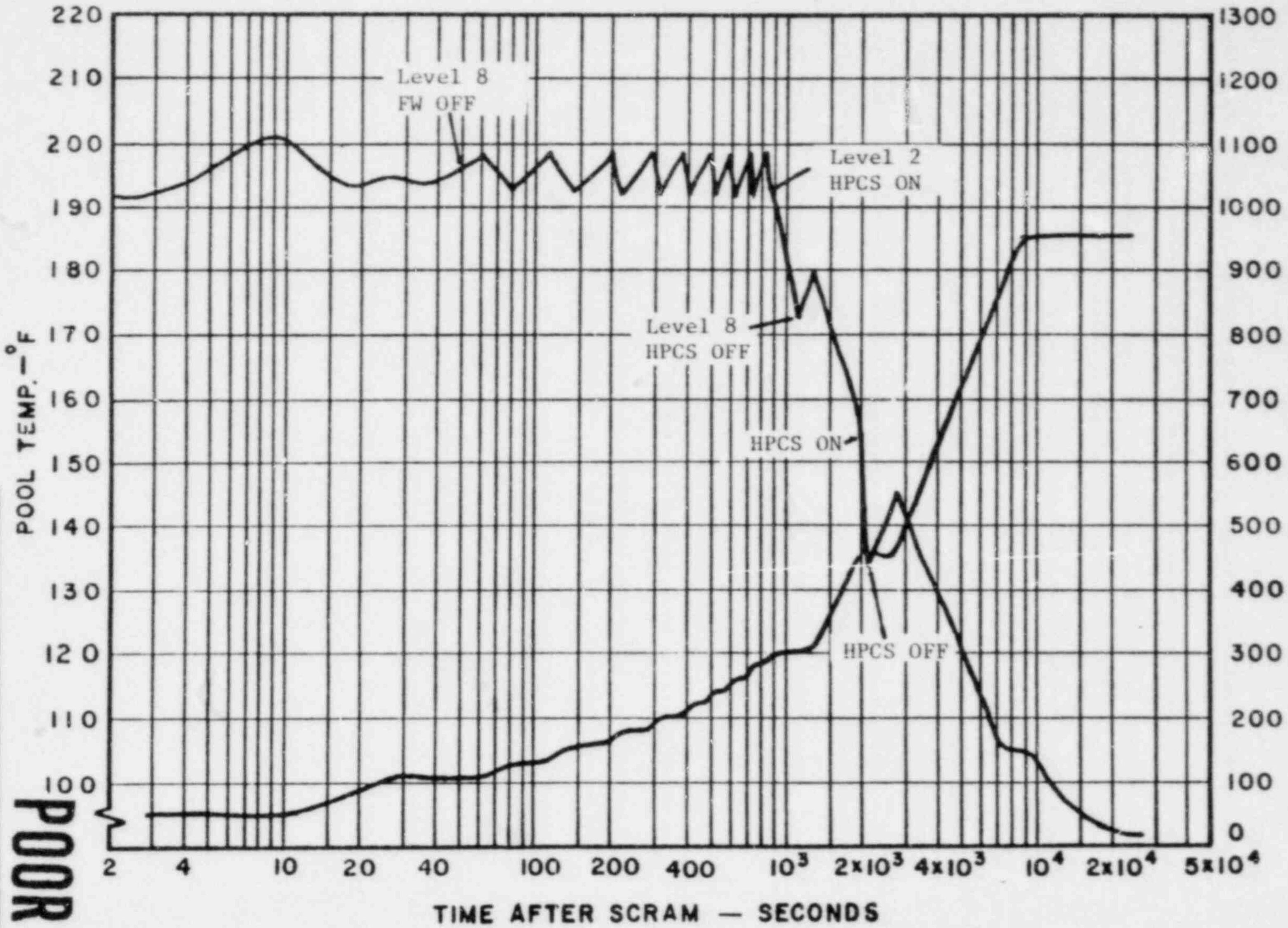
WM. H. ZIMMER NUCLEAR POWER STATION. UNIT 1

MARK II DESIGN ASSESSMENT REPORT

FIGURE 8A-3

FEEDWATER SENSITIVITY STUDY
30 SEC. COASTDOWN CASE 2a,
1 RHR AVAILABLE

POOR ORIGINAL



POOR ORIGINA

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FIGURE 8A-4

FEEDWATER SENSITIVITY STUDY
60 SEC. COASTDOWN CASE 2a,
1 RHR AVAILABLE

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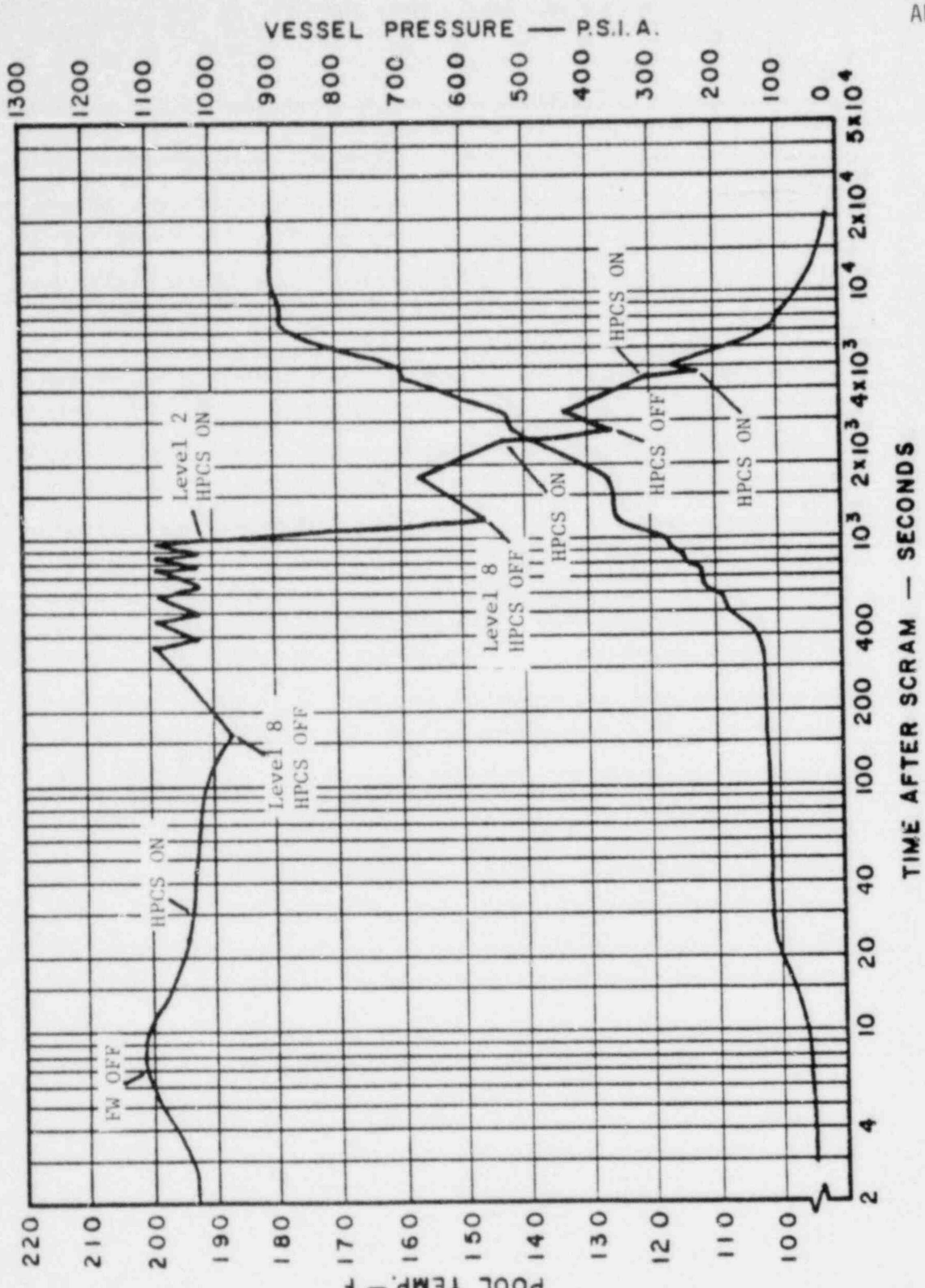

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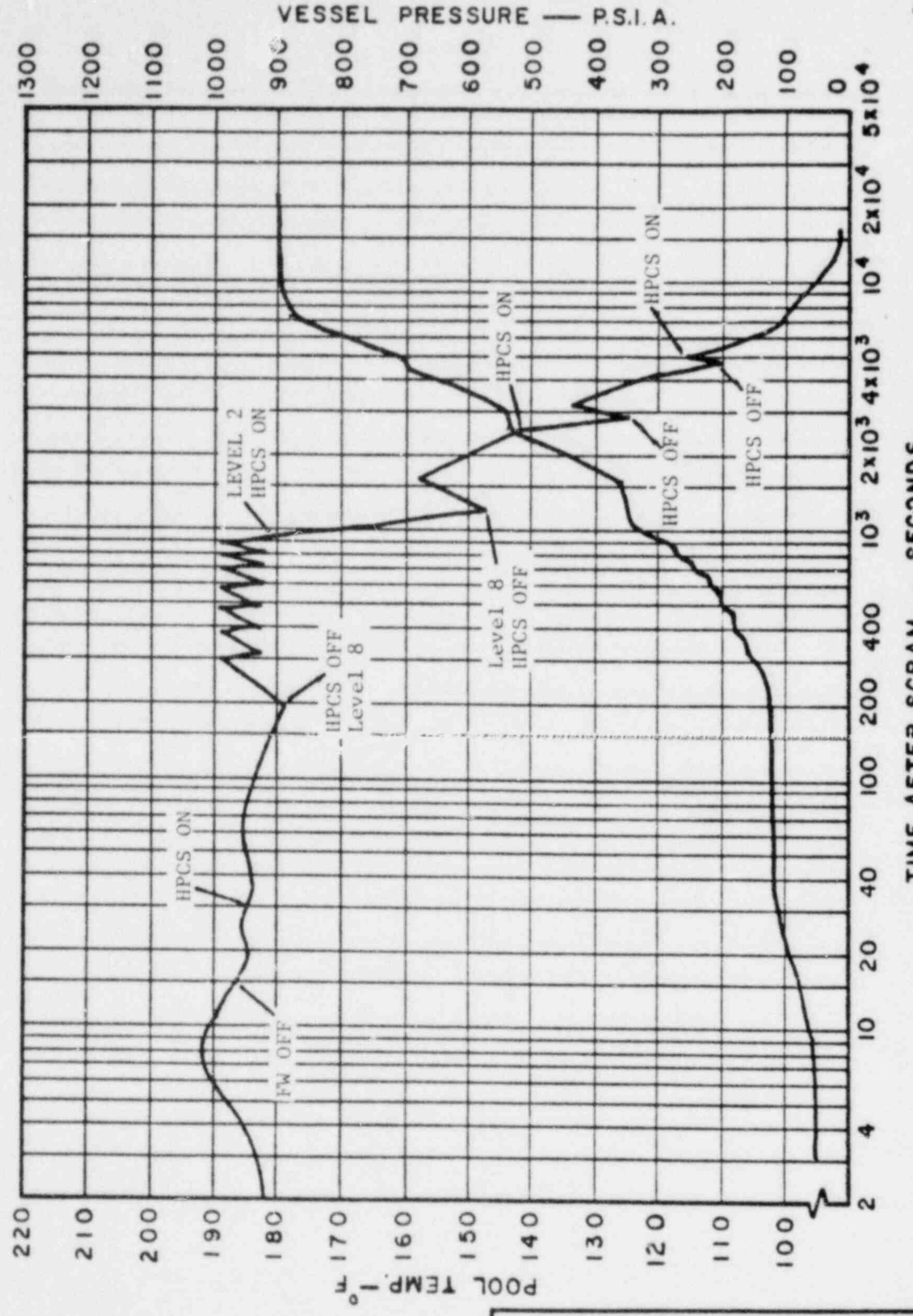
MARK II DESIGN ASSESSMENT REPORT

FIGURE 8A-5

FEEDWATER SENSITIVITY STUDY
7 SEC. COASTDOWN CASE 3a,
1 RHR AVAILABLE

POOR ORIGINAL



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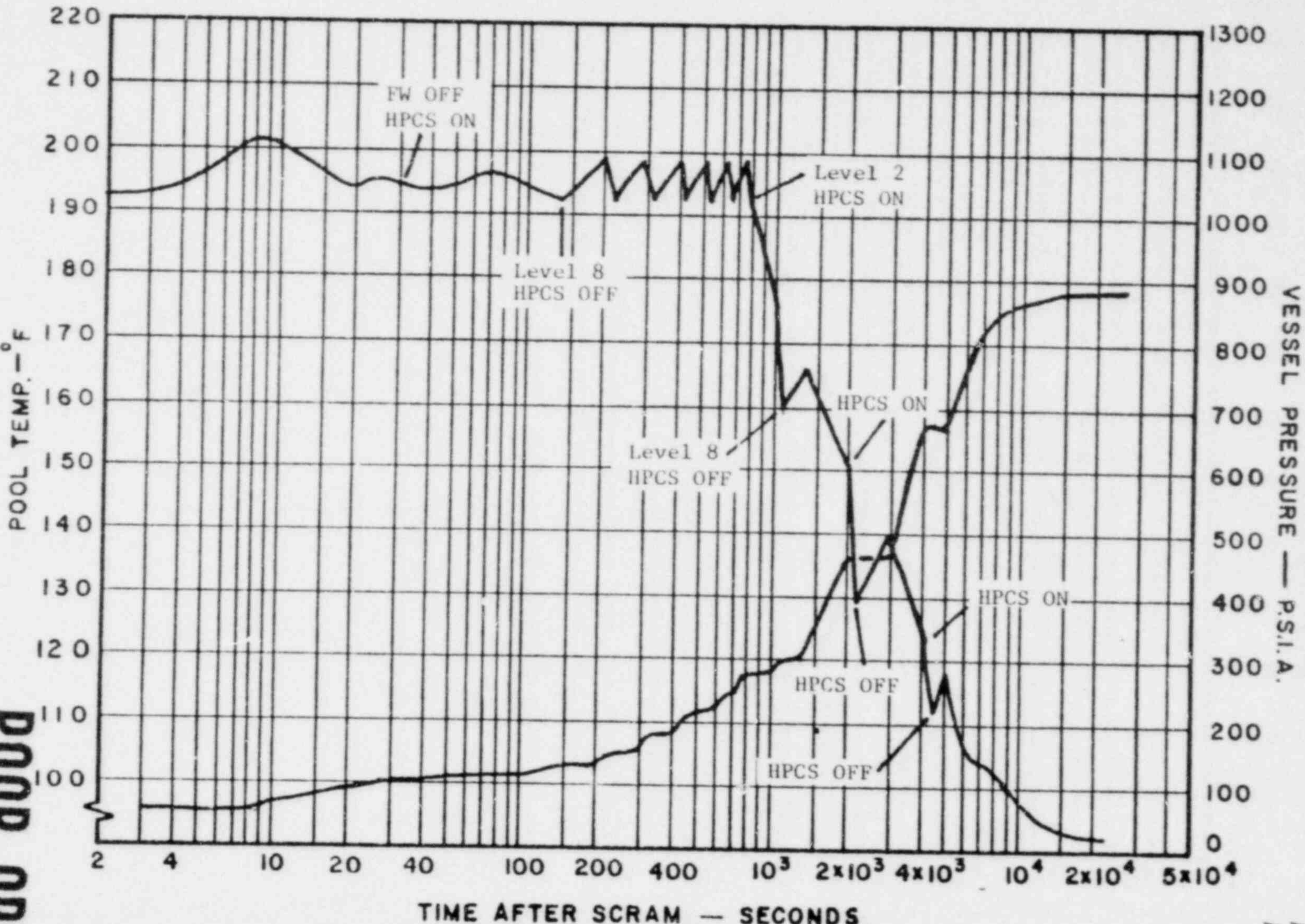
MARK II DESIGN ASSESSMENT REPORT

FIGURE 8A-6

FEEDWATER SENSITIVITY STUDY
15 SEC. COASTDOWN CASE 3a,
1 RHR AVAILABLE

POOR ORIGINAL



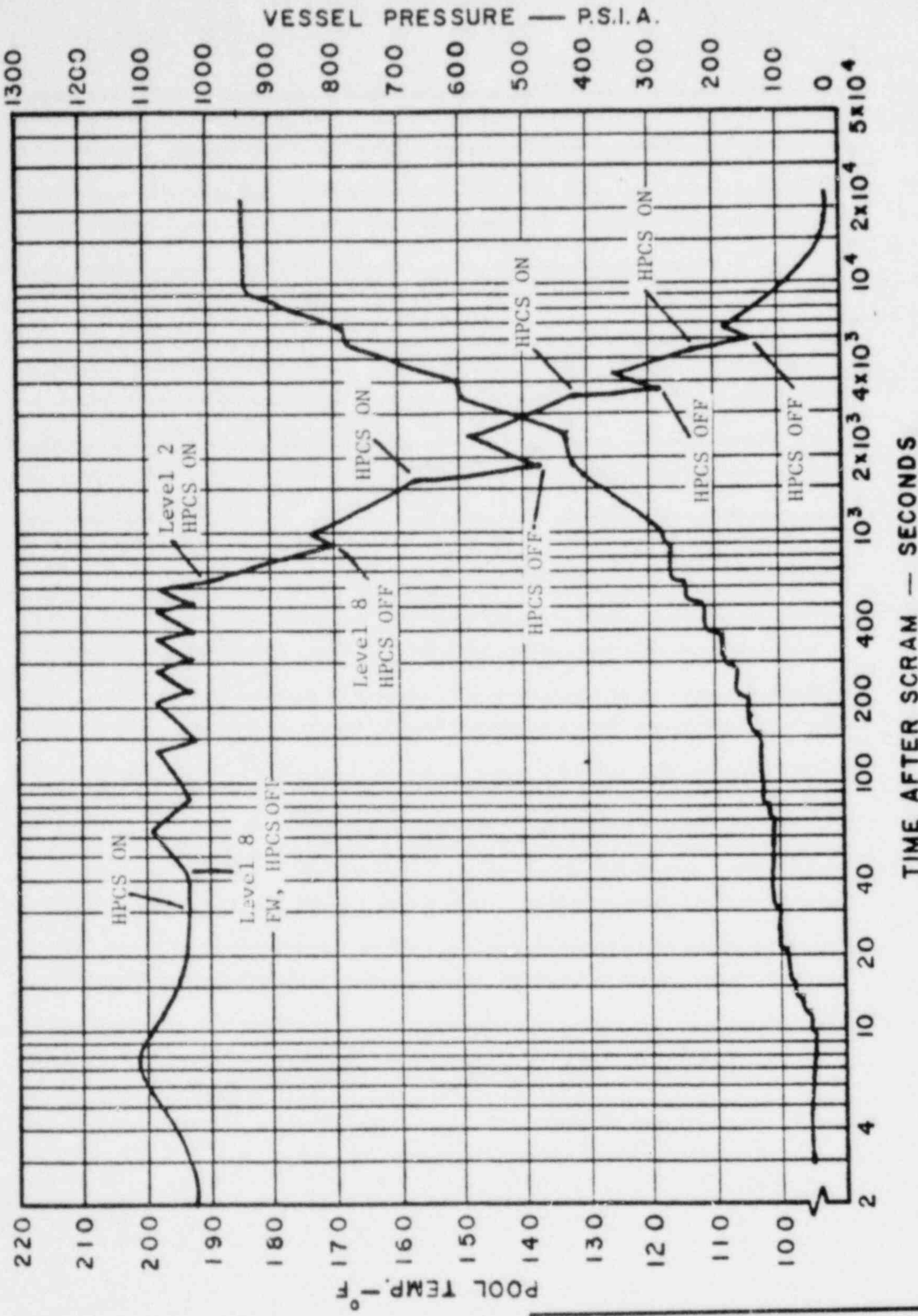
POOR ORIGINAL

WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1
MARK II DESIGN ASSESSMENT REPORT

FIGURE 8A-7

FEEDWATER SENSITIVITY STUDY
30 SEC. COASTDOWN CASE 3a,
1 RHR AVAILABLE

AMENDMENT 14
APRIL 1981



WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1
MARK II DESIGN ASSESSMENT REPORT
FIGURE 8A-8 FEEDWATER SENSITIVITY STUDY 60 SEC. COASTDOWN CASE 3a, 1 RHR

POOR ORIGINAL

ATTACHMENT 8B

ATTACHMENT 8B

MAIN CONDENSER

The Mark II Owners Group (Reference 1) utilizes the main condenser in the transient analysis of a stuck-open relief valve (SORV) at power with one residual heat removal system available.

Both Mark I operating experience and Zimmer transient analyses have been evaulated and it was concluded that the main condenser will be available in the unlikely event that a relief valve sticks open during reactor full-power operation. Therefore, CG&E Company concurs with the Mark II Owners Group position.

The use of the main condenser as a heat sink requires that the bypass system be available, the circulating water system function, and that the main steam isolation valves (MSIV) remain open.

These three requirements are addressed as follows:

The only active components of the turbine bypass system are three hydraulically-operated control valves which are capable of being opened by remote manual operation. 14

Decay heat immediately following scram is less than 10% of the initial core power. Only one out of the three valves is required to meet the decay heat load (total bypass capability [three valves] is 25% of the reactor heated steam flow).

The bypass system, as described in FSAR Subsection 10.4.4, is designed to control reactor pressure:

a. during the reactor heatup to rated pressure, while the turbine generator is being brought up to speed and synchronized,

b. during power operation when the reactor steam generation exceeds the transient turbine steam requirements, and

c. during reactor cooldown.

Both a and b provide online operability checks to ensure that the bypass system is operable.

Online ensurance of operability and redundancy of components (one required of three) ensures the availability of the bypass system in the unlikely event that a relief valve sticks open at reactor full-power operation.

The circulating water system function is to remove heat from the main condenser. It does so by taking water from the cooling tower, passing it through the main condenser, and returning it to the cooling tower.

The only primary active components used to achieve the above function are the three 1/3-capacity circulating water pumps and two 100%-capacity cooling tower makeup pumps.

All three circulating water pumps and one cooling tower makeup pump are in use when the reactor is at full power and, therefore, all pumps must be in operation when the SORV is postulated to occur. The failure of one circulating water pump or one cooling tower makup pump will not degrade the circulating water system below what is needed to supply a sufficient amount of water for the removal of the decay heat that will be bypassed (through the bypass valves) to the main condenser.

The ensured operability of the pumps/system and the redundancy provided by the pumps ensures the availability of circulating water to the condenser in the unlikely event that a relief valve sticks open at reactor full-power operation.

In order to determine whether the main steam isolation valves remain open, the event sequence must be evaluated. The following event sequence for a SORV occurring at reactor full power has been chosen to maximize the severity of the transient. The initial conditions are the same as those utilized by the Mark II Owners Group (Reference 1). 14

A safety/relief valve (SRV) spuriously opens and sticks open.

As described in the event sequence for manually scramming the reactor with a SORV at reactor full power, (See Attachment 8D OP.EOP.01 and OP.EOP.30) the alarms sound and automatic controls adjust the generator load to the decrease in steam flow. Prior to the pool temperature reaching TS3 (110° F) the operator scrams the reactor by placing the reactor mode switch into "shutdown". This procedure maintains the MSIV's open when the reactor pressure falls below the low pressure MSIV closure setpoint.

Following the scram, the feedwater system continues providing makeup to the RPV thereby maintaining reactor vessel water level, the turbine control valves (TCV) close as the RPV pressure drops, and the stuck open relief valve continues to depressurize the RPV. After scramming the reactor and stabilizing the water level, the operator places the RHR system into suppression pool cooling. At this time, no MSIV isolation signals have been generated.

As the SORV continued to depressurize the vessel to the "low steamline pressure" main steam isolation signal, the MSIV's do not close. This isolation signal is bypassed when the reactor mode switch is in shutdown.

Twenty minutes after the initiation of the transient, the operator opens the bypass valves thereby utilizing the main condenser as a heat sink. The bypass valves do not open automatically due to the decreasing reactor pressure caused by the SORV. Condenser vacuum is maintained by the steam jet air ejectors (SJAE).

The main condenser and SORV continue cooling and depressurizing the reactor. When the shutdown cooling pressure permissive is reached, the main condenser is no longer used as a heat sink and the RHR system is transferred to shutdown cooling.

From the above event description one can see that no mechanistic MSIV closure signals are generated and, therefore, the MSIV's remain open. 14

The redundancy provided in the bypass and circulating water systems and the use of the systems during plant operations ensure their availability. The availability of the bypass and circulating water systems along with the main steamline isolation valves remaining open allow the operator to utilize the main condenser as a heat sink. Therefore, CG&E Company has justified that the main condenser is available for the analysis of the postulated pool temperature transient involving a stuck open relief valve at reactor full-power operation coincident with the unavailability of one of the two redundant residual heat removal systems.

REFERENCES

1. Letter plus enclosure dated January 9, 1981, from Mr. Buchholz (GE) to Mr. Kniel (NRC) on Mark II Containment Program, "Assumptions for Use in Analyzing Mark II BWR Suppression Pool Temperature Response to Plant Transients Involving Safety/Relief Valve Discharge-revision 1."

ATTACHEMENT 8C

ATTACHMENT 8C

MANUAL SCRAM

As identified by the Mark II Owners Group (Reference 1), it is assumed that a manual scram of the reactor occurs when the suppression pool temperature reaches TS3 for the transient analysis of a stuck open relief valve at power. TS3 is defined as the maximum allowable suppression pool temperature while maintaining the reactor critical.

CG&E Company concurs with this assumption for Zimmer. TS3 for Zimmer is 110° F.

In the unlikely event that a relief valve sticks open at Zimmer, the operator would be alerted to this by both primary and secondary alarms and plant parameter displays.

The following primary alarms/displays would indicate an open valve immediately after the valve opened:

a. the SRV LEAK DETECTOR alarm,

b. the ADS/SRV OPEN position alarm, and 14

c. the ADS/SRV OPEN position indication.

The following secondary alarms/displays would identify changes in the plant due to steam discharging through the open valve:

a. the continuous display Pool-Temperature recorder/indicator would show an increasing pool temperature,

b. the pool temperature monitoring system would alarm at TS1 (95° F),

c. power meters would indicate a generator load decrease with no change in reactor power, and

d. steam flow, feed flow mismatch could occur.

The above primary and secondary alarms/displays are those that are representative of a stuck-open relief valve (SORV). These alarms/displays provide the operator both immediate and unambiguous indications of a stuck-oper relief valve and high pool temperature.

In order to clarify what information the operator sees and actions he must perform, the following event sequence is provided. This event sequence has been chosen to maximize the severity of the transient. Initial conditions are the same as those utilized by the Mark II Owners Group (Reference 1).

A safety/relief valve (SRV) spuriously opens and sticks open. Immediately the ADS/SRV OPEN alarm sounds, the ADS/SRV OPEN indication light for the specific valve turns on, and the SRV LEAK DETECTOR alarm sounds.[1] Since it is assumed that the pool temperature is just below TS1, at the outset of the transient, the pool temperature monitoring system will alarm immediately after the valve opens and the displayed temperature will begin to increase.

Since steam is being diverted from the Turbine-Generator, the generator load will decrease while a constant reactor power is maintained. The operator at this time (<1 minute after initiation of event) has a clear set of "symptoms" that indicate a stuck-open relief valve has occurred. Operating procedures require the following operator actions:

a. Attempt to close the relief valve remote manually by placing the control switch to open, than back to auto two times, in accordance with OP.EOP.30. See Attachment 8D.

b. SCRAM the reactor and follow OP.EOP.01. See Attachment 8D. 14

c. Initiate pool cooling.

d. Open bypass valves fully.

After scram, the pool temperature monitor would continue to show an increasing temperature and an alarm would sound when TS3 was reached.

From the above description, it can be seen that the operator has sufficient information via alarms and displays to immediately identify a SORV.

Due to clear identification of a SORV and the minimal operator action required, the reactor will be scrammed prior to the bulk pool temperature reaching 110° F.

CG&E Company has provided diverse primary and secondary alarms/ displays and an extensive pool temperature monitoring system that will provide an immediate identification of a SORV. CG&E Company plant operating procedures provide for and require specific actions be taken in the event of a SORV. CG&E Company

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(1) The ADS/SRV OPEN alarm and ADS/SRV OPEN indication light are triggered by position switches which provide a positive valve position indication. The SRV LEAK DETECTOR alarm is triggered by discharge pipe thermocouples.

has, therefore, justified that a manual scram at TS3 (110° F) is a conservative assumption for the evaluation of pool temperature transients involving a stuck-open relief valve at power.

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REFERENCES

1. Letter plus enclosure dated January 9, 1981, from Mr. Buchholz (GE) to Mr. Kniel (NRC) on Mark II Containment Program, "Assumptions for Use in Analyzing Mark II BWR Suppression Pool Temperature Response to Plant Transients Involving Safety/Relief Valve Discharge-revision 1."

ATTACHMENT 8D

ATTACHMENT 8D

THE CINCINNATI GAS & ELECTRIC COMPANY

WM. H. ZIMMER NUCLEAR POWER STATION

UNIT 1

EMERGENCY OPERATING PROCEDURE

STUCK OPEN RELIEF VALVE

PROCEDURE NUMBER: OP.EOP.30

REVISION: 00

PREPARED BY: [signature] DATE: 2/4/81

REVIEWED BY: [signature] DATE: 2/4/81

OPERATIONS SUPERVISOR: [signature] DATE: 2/4/81

STATION SUPERINTENDENT: J. Schott (PEK) DATE: 2/5/81

POOR ORIGINAL

STUCK OPEN RELIEF VALVE (SORV)

1.0 CONDITIONS

SAFETY RELIEF VALVE INDICATES OPEN. 14

2.0 AUTOMATIC ACTIONS

NONE

POOR ORIGINAL

3.0 IMMEDIATE ACTIONS

3.1 CONTROL OPERATOR

3.1.1 CLOSE THE SAFETY RELIEF VALVE. ATTEMPT THIS TWICE.

3.1.2 IF... THE SORV CAN NOT BE CLOSED

THEN...INSERT A MANUAL SCRAM AND ENTER OP.EOP.01. REACTOR SCRAM CONCURRENTLY WITH THIS PROCEDURE.

3.2 SENIOR CONTROL OPERATOR

3.2.1 IF... THE SAFETY RELIEF VALVE IS STUCK OPEN,

THEN...ANNOUNCE TWICE ON THE PA. "STUCK OPEN RELIEF VALVE, INSERTING REACTOR SCRAM."

3.2.2 INITIATE SUPPRESSION POOL COOLING WITH RH LOOPS A AND B.

3.3 LICENSED PLANT OPERATOR

3.3.1 REPORT TO THE CONTROL ROOM. 14

CAUTION

THE 100 DEGREE F/HR COOLDOWN RATE MUST BE EXCEEDED FOR A SORV.

3.3.2 IF...THE SAFETY RELIEF VALVE IS STUCK OPEN,

THEN...OPEN ALL TURBINE BYPASS VALVES FULLY.

3.5 SHIFT TECHNICAL ADVISOR (STA)

REPORT TO THE CONTROL ROOM.

3.6 RAD/CHEM TECH

3.6.1 RETURN TO RADWASTE CONTROL ROOM.

3.6.2 SECURE RADWASTE PROCESSES AS NECESSARY.

3.6.3 NOTIFY THE MAIN CONTROL ROOM OF ACTIONS TAKEN AND PROCEED AS DIRECTED BY THE SHIFT SUPERVISOR.

POOR ORIGINAL

3.7 SHIFT SUPERVISOR

3.7.1 REPORT TO THE CONTROL ROOM.

3.7.2 ASSUME EMERGENCY DUTY SUPERVISOR.

3.7.3 ASSESS THE SITUATION.

3.8 CAS OPERATOR

3.8.1 RESTRICT CONTROL ROOM ACCESS.

3.8.2 PREPARE FOR NOTIFICATION.

4.0 SUPPLEMENTARY ACTION

4.1 SENIOR CONTROL OPERATOR

4.1.1 ENTER OP.EPG.02, "CONTAINMENT CONTROL" CONCURRENTLY WITH THIS PROCEDURE. 14

4.1.2 WHEN...THE SHUTDOWN COOLING INTERLOCKS CLEAR,

THEN...PLACE SHUTDOWN COOLING IN SERVICE, OMITTING THE WARMUP AND FLUSHING REQUIREMENTS.

4.2 SHIFT SUPERVISOR

4.2.1 DECLARE AN UNUSUAL EVENT.

4.2.2 NOTIFY THE OPERATING ENGINEER.

4.2.3 ENTER OP.IPOP.03, "PLANT SHUTDOWN" CONCURRENTLY WITH THIS PROCEDURE.

END

POOR ORIGINAL

THE CINCINNATI GAS & ELECTRIC COMPANY

WM. H. ZIMMER NUCLEAR POWER STATION

UNIT 1

EMERGENCY OPERATING PROCEDURE

REACTOR SCRAM

PROCEDURE NUMBER: OP.EOP.01

REVISION: 05

PREPARED BY: Jeffrey W. Godsey DATE: 2/4/81

REVIEWED BY: [illegible] DATE: 2-4-81

OPERATIONS SUPERVISOR: [illegible] DATE: 2-4-81

STATION SUPERINTENDENT: [illegible] DATE: 2/4/81

POOR ORIGINAL

REACTOR SCRAM

1.0 CONDITION

1.1 REACTOR SCRAM OCCURS PER ATTACHMENT 1.

2.0 AUTOMATIC ACTIONS

2.1 CONTROL RODS ARE INSERTED.

2.2 REACTOR PRESSURE CONTROLLED BY DEHC AND/OR SRV. 14

2.3 REACTOR LEVEL CONTROLLED AT 18.75".

2.4 RR PUMPS MAY SHIFT FROM FAST TO SLOW SPEED OR TRIP.

2.5 EMERGENCY CORE COOLING SYSTEMS, AND THE DIESEL GENERATORS MAY INITIATE.

POOR ORIGINAL

3.0 IMMEDIATE ACTIONS

3.1 CONTROL OPERATOR

3.1.1 ARM AND DEPRESS THE 4 MANUAL SCRAM BUTTONS.

3.1.2 PLACE REACTOR MODE SWITCH TO SHUTDOWN

3.1.3 INSERT IRM'S AND SRM'S

3.1.4 SHIFT APRM/IRM RECORDERS TO THE IRM POSITIONS TO MONITOR AND VERIFY REACTOR POWER DECREASE.

3.1.5 VERIFY AUTOMATIC ACTIONS.

3.1.6 MONITOR RPV PRESSURE.

3.1.7 RE-ESTABLISH RPV WATER LEVEL ABOVE THE SCRAM SETPOINT.

3.2 SENIOR CONTROL OPERATOR

3.2.1 ANNOUNCE OVER PA SYSTEM, "REACTOR SCRAM, REACTOR SCRAM".

3.2.2 MONITOR THE PRIMARY CONTAINMENT PRESSURES, TEMPERATURES, AND SUPPRESSION POOL LEVEL. 14

3.2.3 ANNOUNCE ANY CONTAINMENT ISOLATIONS TO CONTROL ROOM PERSONNEL.

3.3 LICENSED PLANT OPERATOR

3.3.1 VERIFY THE TURBINE GENERATOR HAS TRIPPED.

3.3.2 VERIFY TRANSFER OF THE AUXILIARY ELECTRICAL LOADS.

3.3.3 VERIFY AUTO START OF DIESEL GENERATORS, IF APPLICABLE.

3.4 PLANT OPERATORS

REPORT TO THE CONTROL ROOM.

3.5 SHIFT TECHNICAL ADVISOR

REPORT TO THE CONTROL ROOM.

3.6 RAD/CHEM TECH

3.6.1 RETURN TO THE RADWASTE CONTROL ROOM.

3.6.2 SECURE RADWASTE PROCESSES AS NECESSARY.

POOR ORIGINAL

3.6.3 NOTIFY THE MAIN CONTROL ROOM OF ACTIONS TAKEN, AND PROCEED AS DIRECTED BY THE SHIFT SUPERVISOR.

3.7 SHIFT SUPERVISOR

3.7.1 REPORT TO AND REMAIN IN THE CONTROL ROOM.

3.7.2 ASSUME EMERGENCY DUTY SUPERVISOR.

3.7.3 ASSESS THE SITUATION.

3.8 CAS OPERATOR

3.8.1 RESTRICT CONTROL ROOM ACCESS

3.8.2 PREPARE FOR NOTIFICATION

4.0 SUPPLEMENTARY ACTIONS

4.1 CONTROL OPERATOR

4.1.1 ESTABLISH REACTOR WATER LEVEL BETWEEN +12.5" AND +58" WITH THE RFP'S.

```
*****************************************
*   IF...UNABLE TO RESTORE VESSEL LEVEL,  *
*                                       *
*   THEN...PROCEED TO OP.EPG.01.          *
*****************************************
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14

4.1.2 PLACE FEEDWATER CONTROL IN SINGLE ELEMENT CONTROL.

4.1.3 RESET THE SCRAM SETDOWN FEATURE WHEN DIRECTED BY SS.

4.1.4 RETURN RT TO SERVICE.

4.1.5 DETERMINE THE CAUSE OF THE SCRAM.

4.1.6 WHEN THE SCRAM SIGNAL HAS CLEARED, BYPASS THE SCRAM DISCHARGE VOLUME AND RESET THE SCRAM VALVES.

4.1.7 NOTE ON CHART TRACES THE TIME & EVENT.

4.1.8 COMPLETE THE SCRAM REPORT (ATTACHMENT 2).

4.2 SENIOR CONTROL OPERATOR

4.2.1 IF...SUPPRESSION POOL TEMPERATURE EXCEEDS 95 DEG F, OR DRYWELL TEMPERATURE EXCEEDS 135 DEG F, OR DRYWELL PRESSURE EXCEEDS 1.69 PSIG, OR

POOR ORIGINAL

SUPPRESSION POOL WATER LEVEL EXCEEDS 22 FT. 6 IN..

THEN...ENTER OP.EPG.02, CONTAINMENT CONTROL AND EXECUTE IT CONCURRENTLY WITH THIS PROCEDURE.

4.2.2 CLOSE ANY SORV

A. IF AN SORV CANNOT BE CLOSED AFTER 2 ATTEMPTS, VERIFY THE REACTOR IS SCRAMMED.

4.2.3 IF SRV'S ARE CYCLING, MANUALLY OPEN ONE SRV AND REDUCE RPV PRESSURE TO BELOW 900 PSIG.

4.2.4 RESET GROUP ISOLATIONS AS DIRECTED BY THE SS, IN ACCORDANCE WITH OP.PC.02, PRIMARY CONTAINMENT ISOLATION RESET.

4.3 <u>LICENSED PLANT OPERATOR</u>

NOTE

HP AUTO INITIATION SIGNALS WILL CAUSE A LOADSHED ON SWITCHGROUP 1C, 1D, AND 1E.

4.3.1 IF A LOADSHED OCCURRED, INITIATE OP.EOP.21, LOSS OF AUXILIARY ELECTRICAL POWER. 14

4.3.2 VERIFY THAT THE TURBINE SUPPORT SYSTEMS & PARAMETERS ARE WITHIN LIMITS.

4.3.3 MAINTAIN AND/OR RE-ESTABLISH MAIN CONDENSER VACUUM IN ACCORDANCE WITH OP.CA.01 AS DIRECTED BY THE SHIFT SUPERVISOR.

4.4 <u>SHIFT SUPERVISOR</u>

4.4.1 CHECK THAT THE IMMEDIATE ACTIONS OF THE REACTOR SCRAM PROCEDURE HAVE BEEN COMPLETED. DIRECT SHIFT CREW MEMBERS ON APPLICABLE SUPPLEMENTARY ACTIONS.

4.4.2 DETERMINE THE CAUSE OF THE SCRAM AND CLEAR THE CONDITION, IF POSSIBLE. COORDINATE THE IMMEDIATE AND SUPPLEMENTARY ACTIONS OF THE APPROPRIATE PROCEDURES.

4.4.3 DIRECT THE CONTROL OPERATOR TO RESET SCRAM.

4.4.4 IF CONDITIONS EXIST WHICH ARE DESCRIBED IN TABLE F.4-1 OF THE EMERGENCY PLAN, DECLARE:

A. UNUSUAL EVENT

POOR ORIGINAL

B. ALERT

C. SITE EMERGENCY

D. GENERAL EMERGENCY

4.4.5 HAVE CAS OPERATOR INITIATE NOTIFICATIONS, AS APPLICABLE.

4.4.6 HAVE OP.EPP.08, AIRBORNE RELEASE CALCULATIONS PERFORMED IF AN INADVERTENT RELEASE OF AIRBORNE RADIOACTIVE MATERIAL HAS OCCURRED AS VERIFIED BY ANY OF THE FOLLOWING CONTROL ROOM PANEL INDICATIONS:

A. AUDIBLE ALARMS AND SUSTAINED INCREASED READOUTS ON:

MAIN PLANT VENT STACK RADIATION MONITOR; AND/OR

OFFGAS POST TREATMENT RADIATION MONITOR; AND/OR

PLANT VENT PLENUM RADIATION MONITOR; AND/OR

REFUELING FLOOR VENT PLENUM RADIATION MONITOR 14

B. AUTOMATIC INITIATION OF THE VG SYSTEM.

4.4.7 HAVE THE COMPUTER ANNUNCIATOR LOG REVIEWED.

4.4.9 NOTIFY THE POWER SUPERVISOR.

4.4.10 NOTIFY OPERATING ENGINEER OF REACTOR SCRAM AND CAUSE. IF CAUSE IS CORRECTED, OBTAIN PERMISSION TO RESTART UNIT OR SHUTDOWN IN ACCORDANCE WITH OP.IPOP.03, SHUTDOWN, OR OP.EPG.03, COOLDOWN GUIDELINE.

4.4.11 REQUEST THAT RAD/CHEM TECH OBTAIN A SAMPLE OF REACTOR WATER AND ANALYZE FOR DOSE EQUIVALENT IODINE ACTIVITY, AT THE POST LOCA SAMPLE MONITOR.

5.0 DISCUSSION

THE CONDITIONS AND SETPOINTS WHICH COULD CAUSE A SCRAM ARE LISTED IN ATTACHMENT 1. IF ANY OF THESE CONDITIONS ARE DISCOVERED TO EXIST, OR IF IT IS EVIDENT THAT AN AUTOMATIC SCRAM IS UNAVOIDABLE, THE REACTOR SHOULD BE MANUALLY SCRAMMED.

THE OPERATOR SHOULD BE AWARE THAT ANY SCRAM MAY CAUSE A TEMPORARY LOW WATER LEVEL CONDITION IN THE REACTOR. THIS LEVEL MAY INITIATE EMERGENCY CORE COOLING SYSTEMS. THE OPERATOR SHALL CONTINUOUSLY MONITOR RPV WATER LEVEL USING ALL AVAILABLE INSTRUMENTATION AS FOLLOWS:

TYPE RANGE	INSTRUMENT NO.	PANEL
NARROW RANGE (0" TO +60")	1LR-1C34-R608	1H13-P603
NARROW RANGE (0" TO +60")	1LI-1C34-R606A,B,C	1H13-P603
WIDE RANGE (+60" TO -150")	1LR-1B21-R623A	1H13-P601
WIDE RANGE (+60" TO -150")	1LR-1B21-R623B	1H13-P601
WIDE RANGE (+60" TO -150")	1LI-1B21-R604	1H13-P601
SHUTDOWN RANGE (0" TO 400")	1LI-1B21-R605	1H13-P602
FUEL ZONE RANGE (+50" TO -150")	1LI-1B21-R610	1H13-P601
FUEL ZONE RANGE (+50" TO -150")	1LR-1B21-R615	1H13-P601

14

END

OP.EOP.01-7
REV.05

POOR ORIGINAL

AUTOMATIC REACTOR SCRAM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	REACTOR MODE SWITCH
1. IRM HI FLUX	120/125 OF FULL SCALE	SU
2. APRM:		
A. HI FLUX	15%	SU
B. FLOW BIAS FLUX	(0.66W + 51)% OR	RUN
	113.5% MAX	RUN
C. HI FLUX (FIXED)	118%	RUN
3. RX VESSEL PRESSURE HI	1043 PSIG	SU/RUN
4. RX VESSEL LEVEL LOW	12.5 INCHES	SU/RUN
5. MSIV CLOSURE	6% CLOSURE EITHER MSIV IN 3 OF 4 STM LINES	RUN
6. MSL RADIATION HI	3X FULL POWER BACKGROUND	SU/RUN
7. PRI CONT PRESS HI	1.69 PSIG	SU/RUN
8. SCRAM DISCH VOL LVL HI	76 GALS	SU/RUN
9. TSV CLOSURE	3 OF 4 TSV CLOSED 10% (IF >30% POWER)	RUN
10. TCV FAST CLOSURE	850 PSIG OIL PRESSURE (IF > 30% POWER)	RUN
11. MODE SW IN SHUTDOWN	N/A	N/A
12. MANUAL SCRAM	N/A	SU/RUN

14

MANUAL SCRAMS

1. 110 DEG F SUPPRESSION POOL TEMPERATURE
2. SORV. (AFTER 2 ATTEMPTS TO RECLOSE).
3. TRIP OF BOTH RR PUMPS TO ZERO SPEED.
4. LOSS OF IA AND ROD DRIFT.
5. OG PRETREATMENT MONITOR >5 CI/SEC.

ATTACHMENT 1
OP.EOP.01-8
REV.05

POOR ORIGINAL

SCRAM REPORT

1. SCRAM NUMBER: __________

2. SCRAM TIME: __________ DATE: __________

3. PRESCRAM CONDITIONS:

   A. REACTOR: CRITICAL ______________ SUBCRITICAL ________________

   B. PLANT EVOLUTION: START UP ____ SHUTTING DOWN ____ STEADY OP ____

   C. POWER LEVEL: ____________________% ______________________MWT

   D. MODE SWITCH: RUN __________________ SU/STBY ________________

   E. TURBINE-GEN: SYNC ___________ LOAD _________ MWE ____________RPM

   F. RECIRC FLOW ________________________ NO OF PUMPS ______________

   CONTROLLER POSITION: ____________ LOCAL MAN. ________________

   G. TURB 1ST STAGE PRESS.: ____________________________________PSIG

   H. REACTOR STEAM FLOW: _____________________________________LB/HR

   I. FEEDWATER FLOW: ____LB/HR

   J. CORE DIFF PRESS AND FLOW: _________________PSI________________%

   K. RPV LEVEL: ____IN

   L. RPV PRESS: ____PSIG 14

4. RELIEF VALVE OPERATION SUMMARY:

SRV	OPERATED YES/NO	#OPENINGS	MANUAL	AUTO	COMMENTS
A					
B					
C					
D					
E					
F					
G					
H					
K					
L					
P					
R					
S					

ATTACHMENT 2
OP.EOP.01-9
REV.05

POOR ORIGINAL

5. MAXIMUM RPV PRESSURE __________ PSIG

6. ITEMS ATTACHED:

   A. CONTROL ROOM LOG (COPY)

   B. COMPUTER LOG

   C. SEQUENCE OF EVENTS (COMPUTER)

   D. ANNUNCIATOR LOG

   E. RECORDER CHARTS - APPLICABLE SECTIONS OF:

      MAIN PLANT VENT STACK RADIATION MONITOR
      OFFGAS POST TREATMENT RADIATION MONITOR
      PLANT VENT PLENUM RADIATION MONITOR
      REACTOR LEVEL
      FEEDWATER FLOW
      STEAM FLOW
      REACTOR PRESSURE

7. <u>CAUSE OF AND DESCRIPTION OF SCRAM</u>

DESCRIBE THE SEQUENCE OF EVENTS LEADING UP TO AND FOLLOWING THE SCRAM. DESCRIBE OR SKETCH ANY ABNORMAL OBSERVED TRANSIENTS. DETERMINE THE PROBABLE CAUSE AND ATTACH THE APPROPRIATE BCP AND NSS POST TRIP LOGS AND RECORDER CHARTS TO THE SCRAM REPORT. 14

TIME	DATE	SHIFT SUPERVISOR

ATTACHMENT 2
OP.EOP.01-10
REV.05

POOR ORIGINAL

ATTACHMENT 8E

TABLE E-1

FSAR CHAPTER 15 EVENTS VERSUS

LONG TERM POOL TEMPERATURE EVENTS

SIMILAR LONG TERM POOL TEMPERATURE EVENTS

FSAR CHAPTER 15 EVENTS		CASES 1a,1b SORV AT POWER	CASES 2a,2b ISOLATION/ SCRAM	CASES 3a,3b SMALL BREAK ACCIDENT	NO POOL TEMPERATURE INCREASE
15.1.1	Loss of Feedwater Heating				X
2	Feedwater Controller Failure - Maximum Demand		X		
3	Pressure Regulator Failure - Open		X		
4	Inadvertent Safety Relief Valve Opening	X			
5	PWR Steam Piping Break				N/A
6	Inadvertent RHR Shutdown Cooling Operation				X
15.2.1	Pressure Regulator Failure - Closed				X
2	Generator Load Reject		X		
3	Turbine Trip		X		
4	MSIV Closures		X		
5	Loss of Condenser Vacuum		X		
6	Loss of AC Power		X		
7	Loss of Feedwater Flow		X		
8	Feedwater Line Break		X		
9	Failure of RHR Shutdown Cooling		X		
15.3.1	Recirculation Pump Trip		X		
2	Recirculation Flow Control Failure - Decreasing Flow		X		
3	Recirculation Pump Seizure		X		
4	Recirculation Pump Shaft Break		X		

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AMENDMENT 14
APRIL 1981

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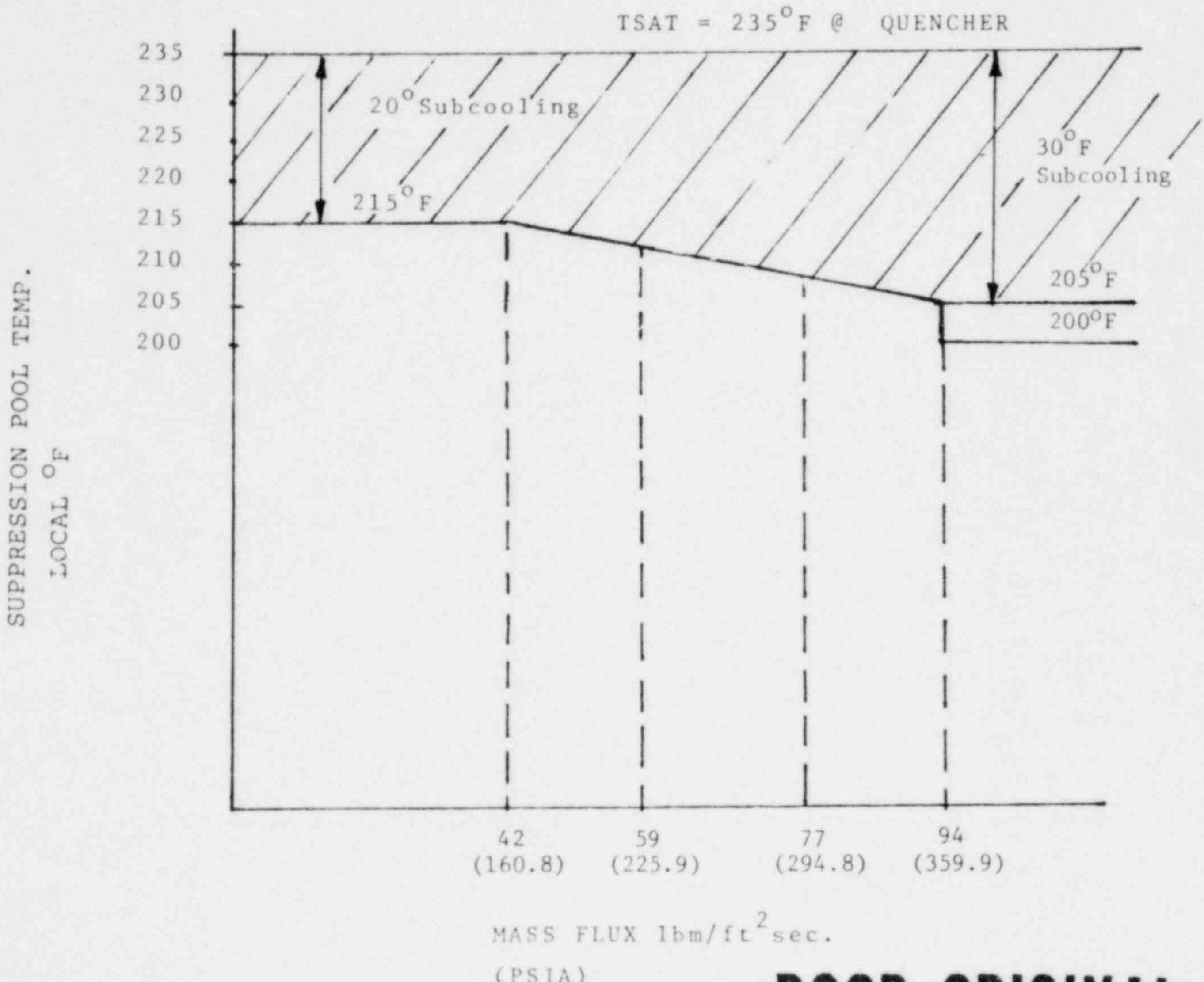
TABLE E-1 (CONTINUED)

FSAR CHAPTER 15 EVENTS VERSUS

LONG TERM POOL TEMPERATURE EVENTS

		SIMILAR LONG TERM POOL TEMPERATURE EVENTS			
FSAR CHAPTER 15 EVENTS		CASES 1a, 1b SORV AT POWER	CASES 2a,2b ISOLATION/ SCRAM	CASES 3a,3b SMALL BREAK ACCIDENT	NO POOL TEMPERATURE INCREASE
15.4.1	Rod Withdrawal Error - Low Power				X
2	Rod Withdrawal Error - At Power				X
3	Control Rod Maloperation				X
4	Abnormal Startup of Idle Recirculation Pump				X
5	Recirculation Flow Control Failure with Increasing Flow				X
6	Chemical and Volume Control System Malfunctions				N/A
7	Misplaced Bundle Accident				X
8	Spectrum of Rod Ejection Assemblies				N/A
9	Control Rod Drop Accident				X
15.5.1	Inadvertent HPCI/HPCS Startup				X
2	Chemical Volume Control System Malfunction				N/A
15.6.1	Inadvertent Safety Relief Valve Opening	X			
2	Instrument Line Break				X
3	Steam Generator Tube Failure				N/A
4	Steam System Piping Break Outside Containment		X		
5	Loss-of-Coolant Accidents			X	
6	Feedwater Line Break Outside Containment		X		

ATTACHMENT 8F



POOR ORIGINAL

WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1

MARK II DESIGN ASSESSMENT REPORT

FIGURE 8F-1

SUPPRESSION POOL TEMPERATURE vs. MASS FLUX

INDEX TO NRC QUESTIONS (Cont'd)

QUESTION 020.63

"For limiting the suppression pool temperature, provide the following additional information:

"(1) Present the temperature transient of the suppression pool starting from the specified temperature limits for the following transients:

(a) Stuck open relief valve

(b) Primary system isolation

(c) Initiation of auto depressurization system

"(2) Describe the instrumentation which will alert the operator to take action to prevent the pool temperature limit to be exceeded.

"(3) Describe the operator actions and operational sequence for those transients stated in Item 1 above. Provide and justify the assumptions of time for initiating each action and the corresponding pool temperature."

RESPONSE 14

The information requested is included in Chapter 8.0 of the Design Assessment Report.

G.5 FLOW BLOCKAGE EFFECTS OF DOWNCOMER BRACING

G.5.1 Correction Factor For Blockage

The downcomer bracing is a flow restriction which increases the fluid velocity and acceleration during pool swell. As a result, the standard drag, acceleration drag and lift loads on structures in the pool swell zone are higher than those which would exist if no downcomer bracing was present. Although the pool swell loads are not design-controlling criteria, the load calculations were adjusted for blockage effects by introducing a multiplicative factor to the fluid velocity and acceleration.

A method of correction has been developed based on References 1 and 2. A multiplicative factor has been determined based upon Maskell's paper (Reference 2):

$$\frac{C_{D_m}}{C_{D_f}} = \left[1 + n\, C_{D_f} \frac{S}{C}\right] \qquad (1)$$

where:

C_{D_m} is the modified drag coefficient of structures in the pool swell zone

C_{D_f} is the steady flow, free stream drag coefficient | 14

n is the blockage factor

S is the total blocked area

C is the unrestricted flow area.

The value of the blockage factor, n, depends upon the structure's geometry. The blockage ratio varies from 0.96 to 2.77 for structures with aspect ratios, AR, from ∞ to 1.0, respectively. Maskell (1963) recommends a blockage factor of 2.5 for bluff bodies and the value is considered conservative for the suppression pool bracing system.

The unrestricted flow area, C, is the pool surface area minus the area of the columns, downcomers and MSRV lines. The blocked area, S, includes all the flanges and members of the bracing system. The drag coefficient used on the right side of equation (1) is the steady flow, free stream drag coefficient of the particular structure being analyzed.

Equation (1) can be rearranged to obtain

$$C_{D_m} = \left(1 + n\, C_{D_F} \frac{S}{C}\right) C_{D_f} \qquad (2)$$

$$= f\, C_{D_f}$$

where f is defined as the blockage correction factor. Since f is proportional to the drag coefficients, and hence to the square of the velocity, the fluid velocity and acceleration are multiplied by the square root of f. For consistency, the same multiplicative factor is used for the velocity and acceleration.

As a result, the pool swell loads are calculated by the following equations:

Standard Drag Load: $F_X = \frac{1}{2} \rho \, (\sqrt{f}\ V)^2 C_D\, A$

Acceleration Drag Load: $F_A = \rho \, (\sqrt{f}\, a) C_m\, V_s$

Lift Load: $F_L = \frac{1}{2} \rho \left[\sqrt{f}\ V\right]^2 C_L A$ 14

where:

- ρ is the fluid density
- f is the blockage correction factor
- V is the fluid velocity
- C_D is standard drag coefficient which accounts for any interference effects
- A is the projected cross-sectional area of the structure
- C_L is the lift coefficient
- a is the fluid acceleration
- C_m is the inertial coefficient
- V_S is the structure's volume.