PREPARED [signature]	DATE 11/10/80	COMBUSTION ENGINEERING, INC. Chattanooga Nuclear Operations	SHEET 1 OF 26
CHECKED [signature]	DATE 11/19/80	COMPONENT ENGINEERING	CALC. NO. PEC-705
			CONTRACT NO. 71370

TITLE SO. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

1. ABSTRACT

THIS ANALYSIS PRESENTS AN EVALUATION OF THE PRESSURIZER WITH POSTULATED DEFECTS. THE METHOD OF LINEAR ELASTIC FRACTURE MECHANICS IS USED IN ACCORDANCE WITH APPENDIX G OF THE ASME CODE, SECTION III. ALLOWABLE PRESSURE VERSUS FLUID TEMPERATURE CURVES FOR HEATUP AND COOLDOWN ARE CALCULATED.

2. SIGNIFICANT RESULTS

THE ALLOWABLE PRESSURES FOR THE FLUID TEMPERATURES DURING HEATUP AND COOLDOWN ARE CALCULATED FOR EACH LOCATION WITH A HYPOTHETICAL DEFECT. A COMPOSITE CURVE FOR EACH TRANSIENT, SHOWING THE LOWEST ALLOWABLE PRESSURE FOR ALL LOCATIONS, IS GIVEN ON SHEETS 25 AND 26.

3. DISCUSSION OF ANALYSIS

THE METHOD OF LINEAR ELASTIC FRACTURE MECHANICS IS USED IN ACCORDANCE WITH APPENDIX G OF SECTION III OF THE ASME CODE (REF. 1) AND WRCB 175 (REF 2) PROCEDURES. APPENDIX G GIVES THE ALLOWABLE REFERENCE CRITICAL STRESS INTENSITY FACTOR, K_{IR}, AS A FUNCTION OF METAL TEMPERATURE AND NIL-DUCTILITY REFERENCE TEMPERATURE FOR FERRITIC MATERIALS. MODE I STRESS INTENSITY FACTORS, K_I, ARE CALCULATED WITH THE PROCEDURES IN REFS. 1 AND 2. THE FOLLOWING CRITICAL LOCATIONS ARE CONSIDERED:

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TITLE So. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

3. DISCUSSION (CONT'D)

(1) BOTTOM HEAD - SUPPORT SKIRT REGION
(2) SURGE NOZZLE
(3) SPRAY NOZZLE
(4) MANWAY

IN THE BOTTOM HEAD - SUPPORT SKIRT REGION, THE POSTULATED DEFECT IS A 1/4 T, SHARP, SURFACE CRACK WITH LENGTH TO DEPTH RATIO OF 6 LOCATED ON THE PRESSURE BOUNDARY. AT THE PENETRATIONS A CIRCULAR-CONTOUR NOZZLE CORNER CRACK, 1/4 T DEEP, IS ASSUMED.

THE STRESS PARAMETERS REQUIRED TO CALCULATE K_I AT EACH LOCATION WERE OBTAINED FROM THE APPROPRIATE DETAILED STRESS CALCULATIONS - PES-103, PES-202, PES-204 AND PES-207. THERMAL DATA WAS TAKEN FROM THE APPROPRIATE THERMAL CALCULATIONS - PRT-511, PRT-512, PRT-513 AND PRT-515. WHERE THE ACTUAL STRESS AND/OR THERMAL DATA WAS NOT AVAILABLE, THE MOST REPRESENTATIVE, CONSERVATIVE DATA AVAILABLE WAS USED. THE ALLOWABLE PRESSURES, P_{all}, WERE CALCULATED BY EQUATING K_I AND K_{IR} FOR VARYING FLUID TEMPERATURES DURING HEATUP AND COOLDOWN.

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TITLE
SO. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

4. REFERENCES

1. ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, 1980.

2. WELDING RESEARCH COUNCIL BULLETIN 175, PVRC RECOMMENDATIONS ON TOUGHNESS REQUIREMENTS FOR FERRITIC MATERIALS, AUGUST 1972.

3. GENERAL SPECIFICATION FOR A PRESSURIZER ASSEMBLY, SPECIFICATION NO. 00000-PE-130, 6-11-70.

4. PARIS, P.C. AND SIH, G.C., "STRESS ANALYSIS OF CRACKS," FRACTURE TOUGHNESS TESTING AND ITS APPLICATIONS, ASTM STP-381, 1965.

5. CENC-1275, ANALYTICAL REPORT FOR SOUTHERN CALIFORNIA EDISON SAN ONOFRE UNIT NO. 2 PRESSURIZER, SEPTEMBER 1976.

6. CENC-1296, ANALYTICAL REPORT FOR SOUTHERN CALIFORNIA EDISON SAN ONOFRE UNIT NO. 3 PRESSURIZER, SEPTEMBER 1977.

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5. DETAILED ANALYSIS

A. GENERAL REQUIREMENTS

PROTECTION AGAINST BRITTLE FRACTURE IS PROVIDED IN ACCORDANCE WITH PROCEDURES IN APPENDIX G OF THE ASME CODE, SECTION III (REF 1). THE BACKGROUND FOR THESE RULES IS GIVEN IN WRCB 175 (REF 2). THE GUIDELINES THEREIN ARE USED TO SUPPLEMENT THE APPENDIX G PROCEDURES.

APPENDIX G GIVES THE ALLOWABLE STRESS INTENSITY FACTOR, K_{IR}, AS

$$K_{IR} = 26.777 + 1.223\, e^{0.0145(T - RT_{NDT} + 160)} \quad \text{KSI}\sqrt{\text{IN}}$$

WHERE

T IS METAL TEMPERATURE AT THE DEFECT, AND

RT_{NDT} IS THE NIL-DUCTILITY TRANSITION TEMPERATURE FROM TEST RESULTS.

A MAXIMUM VALUE OF $K_{IR} = 200$ KSI$\sqrt{\text{IN}}$ IS USED AS RECOMMENDED IN WRCB-175.

THE POSTULATED DEFECT PER APPENDIX G, PARAGRAPH G-2120, IS A SHARP, SURFACE DEFECT NORMAL TO THE DIRECTION OF THE MAXIMUM STRESS. FOR SECTIONS OVER 4" THICK, THE DEFECT DEPTH IS 1/4 OF THE THICKNESS; FOR SECTIONS THAT ARE LESS THAN 4" THICK, A 1" DEEP DEFECT IS ASSUMED. THE POSTULATED DEFECT HAS A LENGTH-TO-DEPTH RATIO OF 6.

THE MODE I STRESS INTENSITY FACTOR, K_I, PRODUCED BY THE HEATUP AND COOLDOWN CONDITIONS (REF 3) IS CAL-

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TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. DETAILED ANALYSIS

A. GENERAL REQUIREMENTS

CULATED AT EACH LOCATION IN ACCORDANCE WITH PROCEDURES RECOMMENDED BY APPENDIX G AND WRCB-175.

THE REQUIREMENT TO BE SATISFIED AT ANY PRESSURE AND TEMPERATURE AT THE DEFECT IS

$$F(K_{I_m} + K_{I_b})_{PRIMARY} + (K_{I_m} + K_{I_b})_{SECONDARY} \leq K_{IR}$$

WHERE,

F = SAFETY FACTOR = $\begin{cases} 2.0 & \text{FOR NORMAL \& UPSET} \\ 1.5 & \text{FOR TEST} \end{cases}$

$K_{I_m} = K_I$ DUE TO MEMBRANE STRESS

$K_{I_b} = K_I$ DUE TO BENDING STRESS

PARAGRAPH G-2420 IN APPENDIX G RECOMMENDS THAT THE COMPONENT HYDROSTATIC TEST BE RUN AT A TEMPERATURE NO LOWER THAN $RT_{NDT} + 60°F$, WHERE RT_{NDT} IS THE MAXIMUM NIL-DUCTILITY TRANSITION TEMPERATURE. SINCE $RT_{NDT} = 20°F$, THE HYDRO SHOULD BE PERFORMED AT 80°F OR MORE. NO FURTHER EVALUATION OF HYDROTEST IS NECESSARY.

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TITLE S. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

5. DETAILED ANALYSES

B. METHOD

THE PRESSURE STRESS σ_p CAN BE REPRESENTED BY

$$\sigma_p = \sigma_{m_p} + \sigma_{b_p} + \sigma_{s_p}$$

WHERE σ_{m_p}, σ_{b_p}, AND σ_{s_p} ARE THE PRIMARY MEMBRANE, PRIMARY BENDING, AND SECONDARY STRESSES, RESPECTIVELY. FURTHER,

$$\sigma_{mp} = C_m P \qquad \sigma_{bp} = C_b P \qquad \sigma_{sp} = C_s P$$

WHERE P IS PRESSURE AND THE C'S ARE PRESSURE STRESS COEFFICIENTS WHICH CAN BE DETERMINED FROM THE DETAILED STRESS ANALYSES.

THE THERMAL STRESS σ_t IS GIVEN BY

$$\sigma_t = \sigma_{m_t} + \sigma_{b_t}$$

WHERE σ_{m_t} AND σ_{b_t} ARE THE MEMBRANE AND BENDING PORTIONS OF σ_t, RESPECTIVELY. OTHER SOURCES OF STRESS (PRE-LOAD, PIPE LOADS, ETC.) CAN SIMILARLY BE EXPRESSED AS

$$\sigma_c = \sigma_{m_c} + \sigma_{b_c}$$

THE STRESS INTENSITY FACTOR K_I CAN BE EXPRESSED BY

$$K_I = A\sigma_m + B\sigma_b$$

WHERE A AND B ARE GEOMETRY DEPENDENT CONSTANTS.

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TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. Detailed Analysis

B. Method

Substituting the above expressions into the general requirement yields

$$F(AC_m P + BC_b P + A\sigma_{m_c} + B\sigma_{b_c})$$
$$+ (BC_s P + A\sigma_{m_t} + B\sigma_{b_t}) \le K_{IR}$$

Solving for P

$$P \le P_{all} = \frac{K_{IR} - F(A\sigma_{m_c} + B\sigma_{b_c}) - (A\sigma_{m_t} + B\sigma_{b_t})}{FAC_m + FBC_b + BC_s}$$

So for a given condition with known values for σ_{m_t}, σ_{b_t}, σ_{m_c}, σ_{b_c} and T, the metal temperature at the defect, P_{all} can be calculated.

Throughout this calculation it is assumed that

$$\sigma_{m_c} = \sigma_{b_c} = C_b = 0$$

so that

$$P_{all} = \frac{K_{IR} - (A\sigma_{m_t} + B\sigma_{b_t})}{FAC_m + BC_s}$$

Further, in the vessel penetrations

$$\sigma_{m_t} = C_s = 0$$

so

$$P_{all} = \frac{K_{IR} - B\sigma_{b_t}}{FAC_m}$$

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TITLE So. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

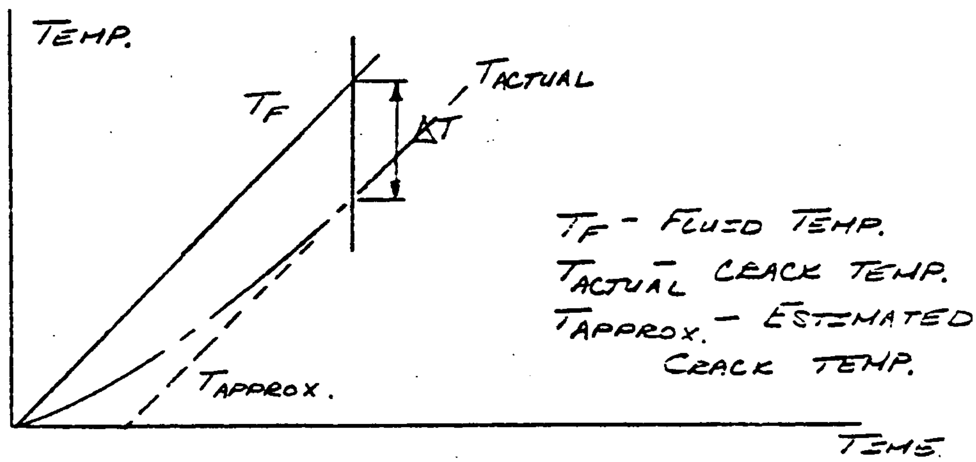
5. DETAILED ANALYSIS

a. METHOD

THE PROCEDURE FOR CALCULATION OF P_{all} IS AS FOLLOWS:

1) ASSUME $T_{REF} = T - RT_{NDT}$
2) CALCULATE K_{IR}
3) CALCULATE THE FLUID TEMPERATURE $T_F = T + \Delta T$ WHERE ΔT IS A CONSERVATIVE ESTIMATE OF THE TEMPERATURE DIFFERENCE BETWEEN THE CRACK AND THE FLUID.
4) CALCULATE P_{all}.

FOR COOLDOWN $T_F \leq T$, SO ΔT CAN BE CONSERVATIVELY TAKEN AS ZERO. FOR HEATUP ΔT WAS CALCULATED AS SHOWN BELOW:



FOR A GIVEN TIME DURING HEATUP, THIS RESULTS IN VALUES FOR T WHICH ARE CONSERVATIVELY LOWER THAN T_{ACTUAL}.

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TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. DETAILED ANALYSES

C. BOTTOM HEAD - SUPPORT SKIRT

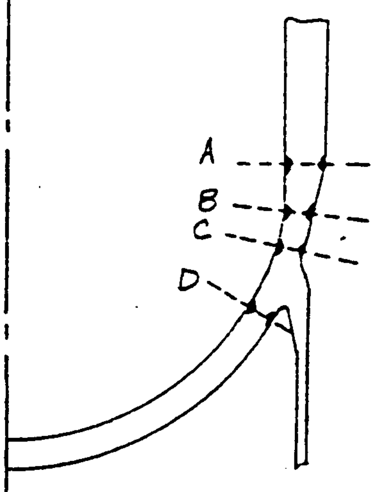
APPENDIX G, PARAGRAPH G-2220, GIVES A METHOD FOR ANALYZING SHELLS AND HEADS NEAR DISCONTINUITIES.

$$K_{I_m} = A\,\sigma_m = M_m\,\sigma_m$$

$$K_{I_b} = B\,\sigma_b = M_b\,\sigma_b$$

WHERE M_m IS GIVEN IN FIG G-2214.1 AND $M_b = 2/3\,M_m$. M_m IS DEPENDENT ON WALL THICKNESS, t, AND IS BASED ON A CRACK DEPTH, $a = T/4$.

THE BOTTOM HEAD - SUPPORT SKIRT IS EVALUATED AT THE EIGHT LOCATIONS SHOWN - THE INSIDE AND OUTSIDE SURFACES OF CUTS A, B, C AND D. PLANT HEATUP AND COOLDOWN ARE EVALUATED AND COVER THE LESS SEVERE TRANSIENTS.



THE BRITTLE FRACTURE PARAMETERS, t, a, A, B ARE SHOWN IN THE TABLE BELOW FOR EACH CUT.

CUT	t	a	A	B
A	4.88	1.22	2.18	1.45
B	3.88	1.00	1.98	1.32
C	3.88	1.00	1.98	1.32
D	3.88	1.00	1.98	1.32

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TITLE So. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

5. DETAILED ANALYSIS

C. BOTTOM HEAD - SUPPORT SKIRT

DUE TO THE COMPLEXITY OF THE STRESS STATE IN THIS REGION, CRACK ORIENTATIONS NORMAL TO BOTH THE HOOP AND LONGITUDINAL STRESSES WERE CONSIDERED. THE RESULTING ALLOWABLE PRESSURES ARE DESIGNATED p_1 FOR THE HOOP STRESS AND p_2 FOR THE LONGITUDINAL STRESS. THE ALLOWABLE PRESSURE p_{all} IS THE LESSER OF p_1 AND p_2.

THE STRESS PARAMETERS - C_m, C_s, σ_{m_t}, AND σ_{b_t} - ARE DERIVED FROM THE PRIMARY PLUS SECONDARY STRESSES AT CUTS A, B, C, AND D GIVEN IN PRS-102 IN REF 5 AND 6. THE APPROPRIATE VALUES FOR EACH LOCATION ARE SHOWN ON THE FOLLOWING TABLES. SINCE VALUES OF THE THERMAL STRESSES, σ_{m_t} AND σ_{b_t}, WERE NOT CALCULATED IN PRS-102 FOR THE TIMES OF INTEREST IN THIS ANALYSIS, THE MOST REPRESENTATIVE VALUES GIVEN IN PRS-102 WERE CONSERVATIVELY USED. THERMAL DATA USED FOR THIS REGION IS TAKEN FROM PRT-511 IN REF. 5 AND 6.

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TITLE: So. Cal. Pressurizer - Brittle Fracture Evaluation

5. Detailed Analysis

C. Bottom Head - Support Skirt

Location: Cut A, Outside

$A = 2.18$

$B = 1.45$

$RT_{NDT} = 20°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$$\Delta T = \begin{cases} 52° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{m\pm} + B\sigma_{b\pm})}{FAC_m + BC_s}$$

$$P_{all} = \text{LESSER OF} \begin{cases} P_1 \\ P_2 \end{cases}$$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	122°	2.07	1.42	1.42	70°	2.68	1.70	1.70
60.	56.5	132°	2.28	1.55	1.55	80°	2.89	1.33	1.33
70.	61.1	142°	2.51	1.70	1.70	90°	3.12	1.93	1.93
80.	66.5	152°	2.85	1.87	1.87	100°	—	2.15	2.15
90.	72.7	162°	3.11	2.07	2.07	110°	—	2.35	2.35
100.	79.8	172°	—	2.30	2.30	120°	—	2.53	2.53
110.	88.1	182°	—	2.57	2.57	130°	—	2.95	2.95
120.	97.7	192°	—	2.88	2.88	140°	—	3.16	3.16
130.	108.7	202°	—	3.24	3.24				
140.	121.5								
150.	136.3								
C_m			4.73	7.28			4.73	7.28	
C_s			-0.73	-0.58			-0.73	-0.58	
$\sigma_{m\pm}$			-0.35	-0.40			0.00	0.00	
$\sigma_{b\pm}$			3.76	6.59			-9.90*	-9.12*	
$A\sigma_{m\pm} + B\sigma_{b\pm}$			11.94	8.68			-14.36*	-10.03*	
$FAC_m + BC_s$			19.56	30.90			19.56	30.70	

* Conservatively assumed to be zero.

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TITLE So. Cal. Pressurizer – Brittle Fracture Evaluation

5. DETAILED ANALYSIS

C. BOTTOM HEAD – SUPPORT SKIRT

LOCATION: CUT A, INSIDE

$A = 2.18$

$B = 1.45$

$RT_{NDT} = 20°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 36° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s}$$

P_{all} = LESSER OF $\begin{cases} P_1 \\ P_2 \end{cases}$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	106°	2.42	1.61	1.61	70°	1.76	1.20	1.20
60.	56.5	116°	2.61	1.73	1.73	80°	1.94	1.32	1.32
70.	61.1	126°	2.82	1.88	1.88	90°	2.16	1.46	1.46
80.	66.5	136°	3.07	2.04	2.04	100°	2.40	1.63	1.63
90.	72.7	146°	—	2.23	2.23	110°	2.69	1.82	1.82
100.	79.8	156°	—	2.45	2.45	120°	3.02	2.04	2.04
110.	88.1	166°	—	2.70	2.70	130°	—	2.29	2.29
120.	97.7	176°	—	3.00	3.00	140°	—	2.58	2.58
130.	108.7					150°	—	2.92	2.92
140.	121.5					160°	—	3.32	3.32
150.	136.3								
C_m			4.73	7.28			4.73	7.28	
C_s			0.73	0.58			0.73	0.58	
σ_{mt}			-0.35*	-0.40*			0.00	0.80	
σ_{bt}			-8.76*	-6.59*			9.90	8.12	
$A\sigma_{mt} + B\sigma_{bt}$			-13.46*	-10.43*			14.36	13.52	
$FAC_m + BC_s$			21.68	32.58			21.68	32.58	

* Conservatively assumed to be zero.

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5. DETAILED ANALYSIS
C. BOTTOM HEAD - SUPPORT SKIRT

LOCATION: CUT 8, INSIDE

$A = 1.98$

$B = 1.32$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 32° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_S}$$

$$P_{all} = \text{LESSER OF} \begin{cases} P_1 \\ P_2 \end{cases}$$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	—	—	—	—	—	—	—	—
60.	56.5	102°	2.35	2.02	2.02	70°	1.45	1.33	1.33
70.	61.1	112°	2.54	2.18	2.18	80°	1.67	1.55	1.55
80.	66.5	122°	2.76	2.37	2.37	90°	1.87	1.74	1.74
90.	72.7	132°	3.02	2.59	2.59	100°	2.12	1.96	1.96
100.	79.8	142°	—	2.85	2.85	110°	2.42	2.22	2.22
110.	88.1	152°	—	3.15	3.15	120°	2.77	2.51	2.51
120.	97.7					130°	3.16	2.85	2.85
130.	108.7					140°	—	3.25	3.25
140.	121.5								
150.	136.3								
C_m			6.03	6.98			6.03	6.98	
C_S			0.13	0.28			0.13	0.28	
σ_{mt}			-0.44*	-1.99*			-0.02*	2.76	
σ_{bt}			-15.03*	-8.09*			16.35	9.32	
$A\sigma_{mt}+B\sigma_{bt}$			-20.71*	-14.62*			21.58	17.77	
$FAC_m + BC_S$			24.05	28.01			24.05	28.01	

* Conservatively assumed to be zero.

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TITLE So. Cal. Pressurizer – Brittle Fracture Evaluation

5. Detailed Analysis

C. Bottom Head – Support Skirt

Location: Cut B, Outside

$A = 1.93$

$B = 1.32$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 46° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s}$$

P_{all} = LESSER OF $\begin{cases} P_1 \\ P_2 \end{cases}$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	—	—	—	—	—	—	—	—
60.	56.5	116.°	1.53	1.82	1.53	70.°	2.38	2.07	2.07
70.	61.1	126.°	1.78	1.99	1.78	80.°	2.53	2.24	2.24
80.	66.5	136.°	2.00	2.19	2.00	90.°	2.80	2.44	2.44
90.	72.7	146.°	2.26	2.42	2.26	100.°	3.06	2.66	2.66
100.	79.8	156.°	2.57	2.68	2.57	110.°	—	2.93	2.93
110.	88.1	166.°	2.92	2.98	2.92	120.°	—	3.23	3.23
120.	97.7	176.°	3.32	3.33	3.32				
130.	108.7								
140.	121.5								
150.	136.3								
C_m			6.03	6.98			6.03	6.98	
C_s			-0.13	-0.28			-0.13	-0.28	
σ_{mt}			-0.44	-1.99			-0.02*	2.76*	
σ_{bt}			15.03	8.09			-16.35*	-9.32*	
$A\sigma_{mt} + B\sigma_{bt}$			18.97	6.74			-21.62*	-6.24*	
$FAC_m + BC_s$			23.71	27.27			23.71	27.27	

*Conservatively assumed to be zero.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

		COMBUSTION ENGINEERING, INC.	
R.E. Johnson PREPARED	10/6/80 DATE	Chattanooga Nuclear Operations	SHEET 15 OF 26
Halliday CHECKED	11/19/80 DATE	COMPONENT ENGINEERING	CALC. NO. PRS-705
			CONTRACT NO. 71370

TITLE So. Cal. Pressurizer – Brittle Fracture Evaluation

5. Detailed Analyses

C. Bottom Head – Support Skirt

Location : Cut C, Outside

$A = 1.98$

$B = 1.32$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 88° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s}$$

P_{all} = LESSER OF $\begin{cases} P_1 \\ P_2 \end{cases}$

Transient		Heatup				Cooldown			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	—	—	—	—	—	—	—	—
60.	56.5	158.	1.65	2.10	1.65	70.°	2.43	2.59	2.43
70.	61.1	168.	1.85	2.31	1.85	80°	2.63	2.81	2.63
80.	66.5	178.	2.08	2.56	2.08	90°	2.86	3.05	2.86
90.	72.7	188.	2.35	2.84	2.35	100°	3.13	—	3.13
100.	79.8	198.	2.65	3.17	2.65				
110.	88.1	208.	3.01	—	3.01				
120.	97.7								
130.	108.7								
140.	121.5								
150.	136.3								
C_m			6.00	5.60			5.00	5.60	
C_s			-0.40	-0.30			-0.40	-0.30	
σ_{mt}			-0.29	-0.21			-0.20*	-0.02*	
σ_{bt}			14.19	8.48			-13.81*	-9.12*	
$A\sigma_{mt} + B\sigma_{bt}$			18.16	10.78			-19.63*	-12.08*	
$FAC_m + BC_s$			23.23	21.78			23.23	21.78	

*Conservatively assumed to be zero.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

5. Detailed Analyses

C. Bottom Head – Support Skirt

Location: Cut C, Inside

$A = 1.98$

$B = 1.32$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 68° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s}$$

P_{all} = LESSER OF $\begin{cases} P_1 \\ P_2 \end{cases}$

Transient		Heatup				Cooldown			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	—	—	—	—	—	—	—	—
60.	56.5	138°	2.32	2.50	2.32	70°	1.59	1.97	1.59
70.	61.1	148°	2.52	2.71	2.52	80°	1.78	2.17	1.78
80.	66.5	158°	2.74	2.95	2.74	90°	2.00	2.41	2.00
90.	72.7	168°	2.99	3.22	2.99	100°	2.26	2.69	2.26
100.	79.8	178°	3.29	—	3.29	110°	2.55	3.00	2.55
110.	88.1					120°	2.89	—	2.89
120.	97.7					130°	3.29	—	3.29
130.	108.7								
140.	121.5								
150.	136.3								

		Heatup P_1	Heatup P_2		Cooldown P_1	Cooldown P_2
C_m		6.00	5.60		6.00	5.60
C_s		0.40	0.30		0.40	0.30
σ_{mt}		-0.29*	-0.21*		-0.20	-0.02
σ_{bt}		-14.19*	-8.48*		13.31	9.12
$A\sigma_{mt} + B\sigma_{bt}$		-19.31*	-11.61*		17.83	12.00
$FAC_m + BC_s$		24.29	22.57		24.29	22.57

*Conservatively assumed to be zero.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. Detailed Analyses

C. Bottom Head - Support Skirt

Location: Cut D, Inside

$A = 1.98$

$B = 1.32$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 51° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{m\pm} + B\sigma_{b\pm})}{FAC_m + BC_s}$$

P_{all} = LESSER OF $\begin{cases} P_1 \\ P_2 \end{cases}$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	—	—	—	—	—	—	—	—
60.	56.5	121°	1.21	2.95	1.21	70°	2.15	1.63	1.63
70.	61.1	131°	1.39	3.19	1.39	80°	2.33	1.87	1.87
80.	66.5	141°	1.59	—	1.59	90°	2.53	2.15	2.15
90.	72.7	151°	1.83	—	1.83	100°	2.77	2.47	2.47
100.	79.8	161°	2.10	—	2.10	110°	3.04	2.35	2.35
110.	88.1	171°	2.42	—	2.42	120°	—	3.28	3.28
120.	97.7	181°	2.78	—	2.78				
130.	108.7	191°	3.20	—	3.20				
140.	121.5								
150.	136.3								
C_m			5.90	4.58			5.90	4.58	
C_s			2.20	0.78			2.20	0.78	
$\sigma_{m\pm}$			-1.30	-12.37*			1.15*	11.17	
$\sigma_{b\pm}$			20.57	0.67*			-15.01*	2.37	
$A\sigma_{m\pm} + B\sigma_{b\pm}$			24.53	-23.61*			-17.54*	25.25	
$FAC_m + BC_s$			26.27	19.17			26.27	19.17	

*Conservatively assumed to be zero.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

R.E. Johnson 10/6/80 PREPARED DATE	COMBUSTION ENGINEERING, INC. Chattanooga Nuclear Operations	SHEET 18 OF 26
Halliday 11/19/80 CHECKED DATE	COMPONENT ENGINEERING	CALC. NO. PRS-705
		CONTRACT NO. 71379

TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. Detailed Analysis

C. Bottom Head - Support Skirt

Location: Cut D, Outside

$A = 1.98$

$B = 1.32$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 81° & \text{HEATUP} \\ 0° & \text{COOLDOWN} \end{cases}$

$$P_{1,2} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s} \qquad P_{all} = \text{LESSER OF} \begin{cases} P_1 \\ P_2 \end{cases}$$

Transient		Heatup				Cooldown			
T_{REF}	K_{IR}	T_F	P_1	P_2	P_{all}	T_F	P_1	P_2	P_{all}
50.	52.5	—	—	—	—	—	—	—	—
60.	56.5	151°	2.76	3.30	2.76	70°	1.68	2.19	1.68
70.	61.1	161°	2.99	—	2.99	80°	1.91	2.46	1.91
80.	66.5	171°	3.25	—	3.25	90°	2.17	2.78	2.17
90.	72.7					100°	2.47	3.14	2.47
100.	79.8					110°	2.82	3.55	2.82
110.	88.1					120°	3.23	—	3.23
120.	97.7								
130.	108.7								
140.	121.5								
150.	136.3								

	Heatup P_1	Heatup P_2	Cooldown P_1	Cooldown P_2
C_m	5.90	4.58	5.90	4.58
C_s	-2.20	-0.78	-2.20	-0.78
σ_{mt}	-1.30*	-12.37*	1.15	11.17
σ_{bt}	-20.57*	-0.67*	15.01	-2.37
$A\sigma_{mt} + B\sigma_{bt}$	-29.73*	-25.38*	22.09	18.99
$FAC_m + BC_s$	20.46	17.11	20.46	17.11

*Conservatively assumed to be zero.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

/80 DATE	**COMBUSTION ENGINEERING, INC.** Chattanooga Nuclear Operations	SHEET	19 OF 26
11/19/80 DATE	COMPONENT ENGINEERING	CALC. NO.	PES-705
		CONTRACT NO.	71370

CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

LED ANALYSES

NETRATIONS

- THREE MAJOR PENETRATIONS ARE EVALUATED TH THE PROCEDURE DESCRIBED IN APPENDIX OF WRCB-175 (REF. 2). THE PROCEDURE IS DIFIED TO ACCOUNT FOR THE EQUI-BIAXIAL RESS FIELD IN THE PRESSURIZER SPHERICAL EADS BY REPLACING THE FACTOR $F(a/r_n)$, BULATED IN REF 2 FOR A 2:1 BIAXIAL TRESS FIELD, WITH AN EQUIVALENT FACTOR OR AN EQUI-AXIAL FIELD. THIS EQUIVALENT AC , DESIGNATED $F_S(a/r_n)$, IS BASED ON HE DATA IN REF. 4.

THE DETAILS OF THE PROCEDURE ARE GIVEN ON THE FOLLOWING SHEET.

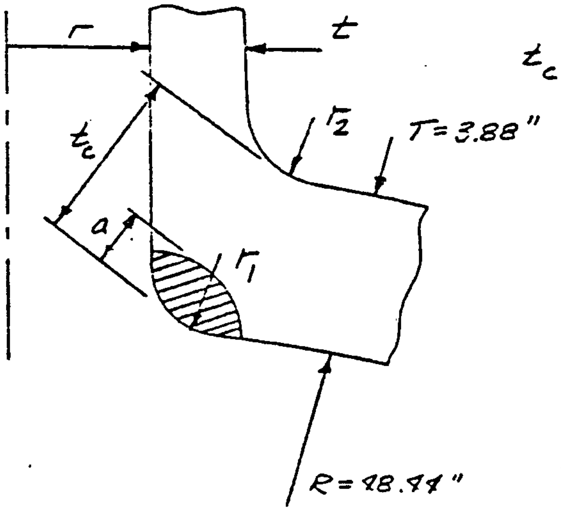
EV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

C. Johnson — PREPARED — 11/10/80 DATE	**COMBUSTION ENGINEERING, INC.** Chattanooga Nuclear Operations	SHEET 20 OF 26
Halliday — CHECKED — 11/19/80 DATE	COMPONENT ENGINEERING	CALC. NO. PES-705
		CONTRACT NO. 71370

TITLE SO. CAL. PRESSURIZER - BRITTLE FRACTURE EVALUATION

5. DETAILED ANALYSIS

D. PENETRATIONS



$$t_c = \sqrt{[(r_2+t)-r_1]^2 + [(r_2+T)-r_1]^2} + r_1 - r_2$$

$$a = .25 t_c$$

$$\sigma_m = C_m P = \frac{R}{2T} P$$

$$C_m = 6.25$$

$$C_s \equiv 0$$

$$\sigma_{bt} = \frac{E\alpha(T_m - T_a)}{1-\nu} = .283(T_m - T_a)$$

FROM REF. 2

$$K_{I_m} = A\sigma_m = \sqrt{\pi a}\, F_s\left(\frac{a}{r_n}\right)\sigma_m$$

$$r_n = r + .29 r_1$$

$$\therefore A = \sqrt{\pi a}\, F_s\left(\frac{a}{r_n}\right)$$

$F_s\left(\frac{a}{r_n}\right)$ FROM REF. 4

$$K_{I_b} = B\sigma_{bt} = M_b\sqrt{\frac{\pi a}{Q}}$$

$$Q = 1.114^2 - .212\left(\frac{\sigma}{\sigma_y}\right)^2 \cong 1.15$$

$$\therefore B = M_b\sqrt{\frac{\pi a}{Q}}$$

$M_b = \frac{2}{3} M_M$ FROM REF. 1

NOZZLE	r	t	r_1	r_2	t_c	a	r_n	$F_s(\frac{a}{r_n})$	M_b	A	B
SURGE	5.94	4.31	1.0	2.5	6.40	1.60	6.23	1.73	1.64	3.88	3.43
SPRAY	2.25	2.00	1.0	1.5	4.54	1.14	2.54	1.53	1.40	2.90	2.47
MANWAY	8.22	3.84	1.5	2.0	5.66	1.42	8.66	1.88	1.57	3.97	3.09

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

P.C. Johnson	10/17/80	COMBUSTION ENGINEERING, INC.	
PREPARED	DATE	Chattanooga Nuclear Operations	SHEET 21 OF 26
Halliday	11/19/80	COMPONENT ENGINEERING	CALC. NO. PRS-705
CHECKED	DATE		CONTRACT NO. 71370

TITLE SO. CAL. PRESSURIZER – BRITTLE FRACTURE EVALUATION

5. DETAILED ANALYSIS

D. PENETRATIONS

LOCATION: SURGE NOZZLE

$A = 3.88$ $\qquad T_F = T_{REF} + RT_{NDT} + \Delta T$

$B = 3.43$ $\qquad \Delta T = \begin{cases} 22^\circ & \text{HEATUP @ 2.9/HR} \\ -45^\circ & \text{COOLDOWN @ 1.5/HR} \end{cases}$

$RT_{NDT} = 10^\circ F$

$$\sigma_{bt} = \frac{E\alpha}{(1-\nu)}(T_m - T_a) = .288(T_m - T_a)$$

$$P_{all} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s} = \frac{K_{IR} - B\sigma_{bt}}{FAC_m}$$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	$T_m - T_a$	σ_{bt}	P_{all}	T_F	$T_m - T_a$	σ_{bt}	P_{all}
60.	56.5	92.	< 0.	0.	1.16	70	0.	0.00	1.16
70.	61.1	102.			1.26	70	20.	5.66	0.86
80.	66.5	112.			1.37	70	34.	9.62	0.69
90.	72.7	122.			1.50	70			0.82
100.	79.8	132.			1.65	70			0.97
110.	88.1	142.			1.82	75			1.14
120.	97.7	152			2.01	85			1.33
130.	108.7	162.			2.24	95.			1.56
140.	121.5	172.			2.51	105.			1.82
150.	136.3	182.			2.81	115			2.13
160.	153.4	192.	< 0.	0.	3.16	125			2.48
170.	173.2					135			2.89
180.	196.0					145.	34.	9.62	3.36
C_m		6.25				6.25			
FAC_m		48.50				48.50			

$$\Delta T = T_F - T_{CL} = T_F - [T_a + .5(T_m - T_a)]$$

T_m, T_a, T_F ARE FROM PRT-513 IN REF. 5 AND 6

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

R.C. Johnson 10/17/80 PREPARED DATE	COMBUSTION ENGINEERING, INC. Chattanooga Nuclear Operations	SHEET 22 OF 26
Halliday 11/19/80 CHECKED DATE	COMPONENT ENGINEERING	CALC. NO. PRS-705 CONTRACT NO. 71370

TITLE So. Cal. Pressurizer – Brittle Fracture Evaluation

5. Detailed Analysis

D. Penetrations

Location: Spray Nozzle

$A = 2.90$

$B = 2.47$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 18° & \text{Heatup @ 2.915 HR} \\ -27° & \text{Cooldown @ 1.592 HR} \end{cases}$

$$\sigma_{b_t} = \frac{E\alpha}{(1-\nu)}(T_m - T_a) = .283(T_m - T_a)$$

$$P_{all} = \frac{K_{IR} - (A\sigma_{m_t} + B\sigma_{b_t})}{FAC_m + BC_s} = \frac{K_{IR} - B\sigma_{b_t}}{FAC_m}$$

Transient		Heatup				Cooldown			
T_{REF}	K_{IR}	T_F	$T_m - T_a$	σ_{bt}	P_{all}	T_F	$T_m - T_a$	σ_{bt}	P_{all}
60.	56.5	88.	< 0.	0.	1.56	70.	0.	0.	1.56
70.	61.1	98.			1.69	70.	20.	5.66	1.30
80.	66.5	108.			1.83	70.	22.	6.23	1.41
90.	72.7	118.			2.01	73.			1.58
100.	79.8	128.			2.20	83.			1.78
110.	88.1	138.			2.43	93.			2.01
120.	97.7	148.			2.70	103.			2.27
130.	108.7	158.	< 0.	0.	3.00	113.			2.57
140.	121.5					123.			2.93
150.	136.3					133.	22.	6.23	3.34
160.	153.4								
170.	173.2								
180.	196.0								
C_m		6.25				6.25			
FAC_m		36.25				36.25			

$$\Delta T = T_F - T_{CL} = T_F - [T_a + .5(T_m - T_a)]$$

T_m, T_a, T_F are from PRT-512 in Ref 5 and 6.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

[signature] 10/17/80 PREPARED DATE	COMBUSTION ENGINEERING, INC. Chattanooga Nuclear Operations		
		SHEET 23 OF 26	
[signature] 11/19/86 CHECKED DATE	COMPONENT ENGINEERING	CALC. NO. PRS-705	
		CONTRACT NO. 71370	

TITLE So. CAL. PRESSURIZER – BRITTLE FRACTURE EVALUATION

5. DETAILED ANALYSIS

D PENETRATIONS

LOCATION: MANWAY

$A = 3.97$

$B = 3.09$

$RT_{NDT} = 10°F$

$T_F = T_{REF} + RT_{NDT} + \Delta T$

$\Delta T = \begin{cases} 13.° & \text{HEATUP @ 2.91 HR} \\ -32° & \text{COOLDOWN 1.5 HR} \end{cases}$

$$\sigma_{bt} = \frac{E\alpha}{(1-\nu)}(T_m - T_a) = .283(T_m - T_a)$$

$$P_{all} = \frac{K_{IR} - (A\sigma_{mt} + B\sigma_{bt})}{FAC_m + BC_s} = \frac{K_{IR} - B\sigma_{bt}}{FAC_m}$$

TRANSIENT		HEATUP				COOLDOWN			
T_{REF}	K_{IR}	T_F	$T_m - T_a$	σ_{bt}	P_{all}	T_F	$T_m - T_a$	σ_{st}	P_{all}
60.	56.5	83.	<0.	0.	1.14	70.	0.	0.	1.14
70.	61.1	93.	↓	↓	1.23	70.	20.	5.66	0.88
80.	66.5	103.			1.34	70.	40.	11.32	0.64
90.	72.7	113.			1.46	70.	44.	12.45	0.69
100.	79.8	123.			1.61	78.	↑	↑	0.83
110.	88.1	133.			1.78	88.			1.00
120.	97.7	143.			1.97	98.			1.19
130.	108.7	153.			2.19	108.			1.41
140.	121.5	163.			2.45	118.			1.67
150.	136.3	173.			2.75	128.			1.97
160.	153.4	183.	<0.	0.	3.09	138.			2.32
170.	173.2					148.	↓	↓	2.71
180.	196.0					158.	44.	12.45	3.17
C_m		6.25				6.25			
FAC_m		49.63				49.63			

$$\Delta T = T_F - T_{CL} = T_F - [T_a + .5(T_m - T_a)]$$

T_m, T_a, T_F ARE FROM PRT-515 IN REF. 5 AND 6.

REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

5. DETAILED ANALYSIS

E. OPERATING LIMITATIONS

OPERATING LIMITATIONS FOR THE PRESSURIZER DURING THE HEATUP AND COOLDOWN TRANSIENTS ARE SHOWN IN THE FIGURES ON SHEETS 25 & 26, RESPECTIVELY. ALL CONDITIONS FALLING BELOW THE CURVES SATISFY THE BRITTLE FRACTURE REQUIREMENTS.

THE DATA POINTS ON THESE FIGURES REPRESENT THE MAXIMUM PRESSURE WHICH, AT THE CORRESPONDING FLUID TEMPERATURE, SATISFIES THE BRITTLE FRACTURE REQUIREMENTS AT A PARTICULAR LOCATION. SINCE EACH CURVE IS A LOWER BOUND TO ALL THE DATA POINTS, OPERATING CONDITIONS BELOW THE CURVES SATISFY ALL THE BRITTLE FRACTURE REQUIREMENTS.

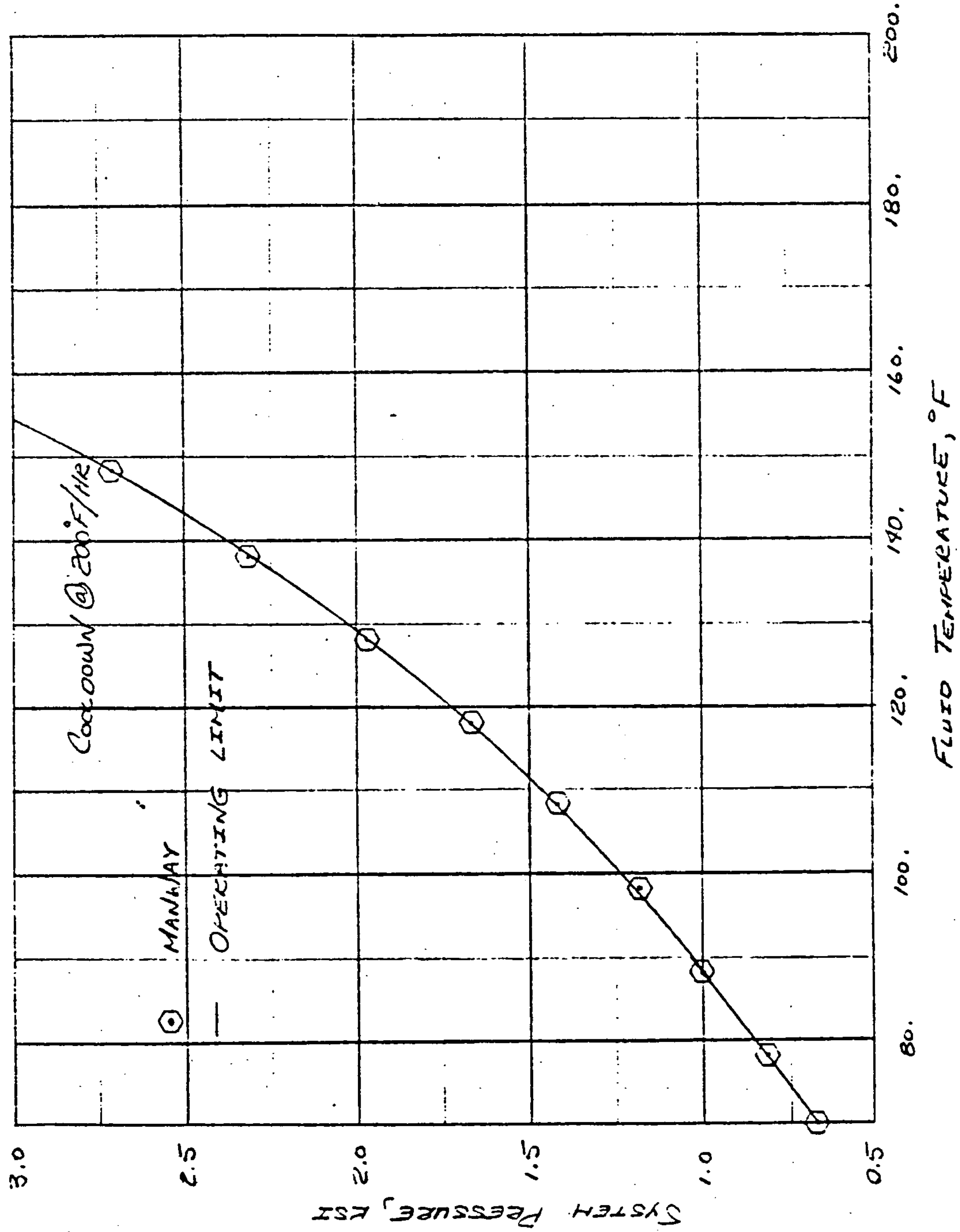
REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

PREPARED R.E. Johnson DATE 11/10/80	COMBUSTION ENGINEERING, INC. Chattanooga Nuclear Operations	SHEET 25 OF 36
CHECKED Halliday DATE 11/19/80	COMPONENT ENGINEERING	CALC. NO. PES-705
		CONTRACT NO. 71370

TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. Detailed Analysis

E. Operating Limitations



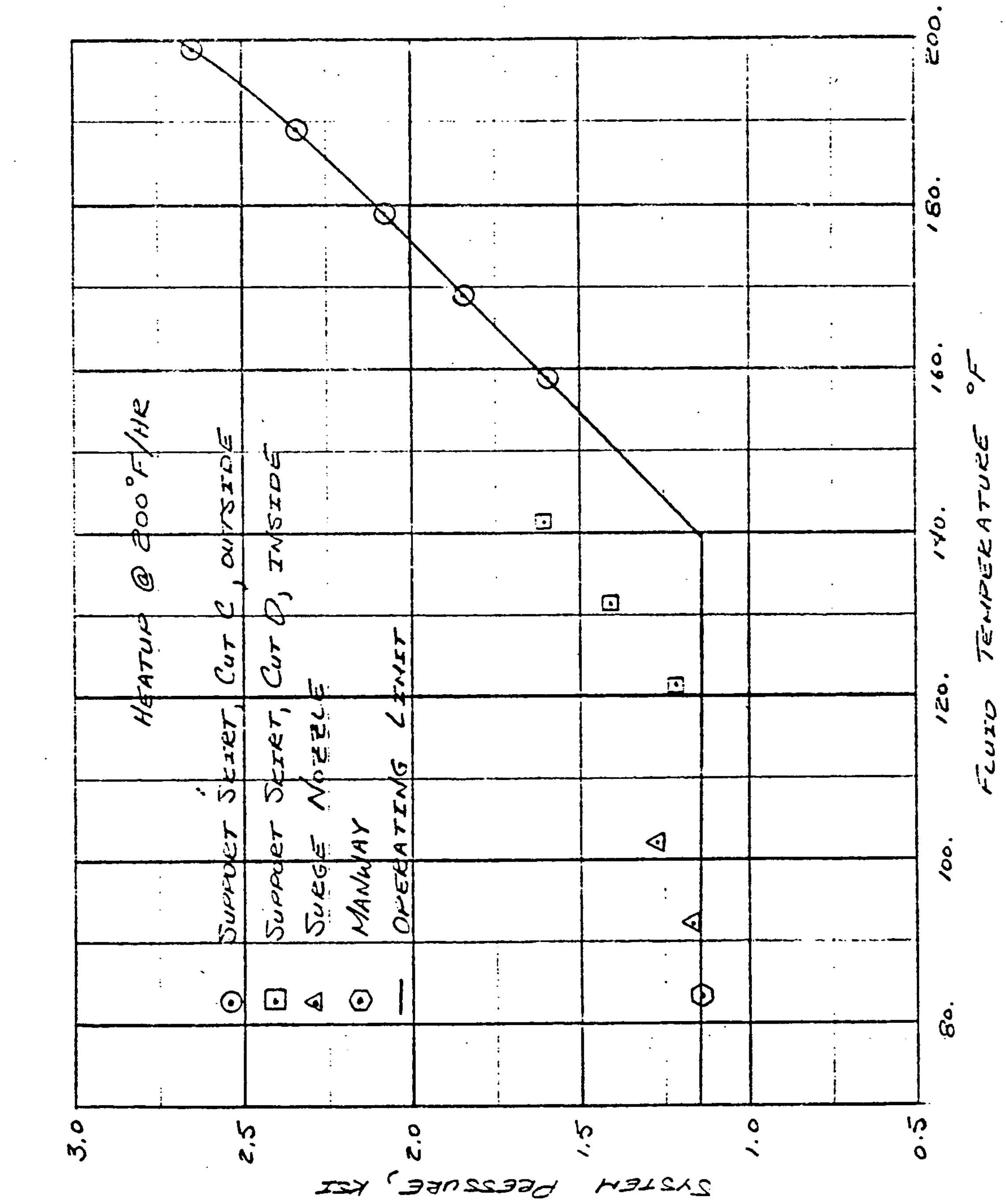
REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

R.C. Johnson 11/10/80 PREPARED DATE	COMBUSTION ENGINEERING, INC. Chattanooga Nuclear Operations		SHEET 26 OF 26
Halliday 11/19/80 CHECKED DATE	COMPONENT ENGINEERING		CALC. NO. PRS-705
			CONTRACT NO. 71370

TITLE So. Cal. Pressurizer - Brittle Fracture Evaluation

5. Detailed Analysis

E. Operating Limitations



REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK	REV.	DATE	BY	CHECK

ENCLOSURE (2)

RESPONSE TO NRC QUESTIONS

Question 001.1

Provide an evaluation which demonstrates that San Onofre 2 and 3 comply with each of the regulations contained in Title 10, Code of Federal Regulations, Parts 20, 50, and 100. Any areas of non-compliance with these regulations should be identified and justified.

Response

The response to NRC Question 001.1 will be provided in an FSAR amendment by January 1981.

Reference

None

Question 010.69

The FSAR does not contain sufficient information to demonstrate that a spent fuel cask drop accident caused by a failure of the cask handling system cannot result in unacceptable conditions because of damages to the spent fuel or excessive spent fuel pool water loss. Utilizing the guidelines in NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" July, 1980, including the analysis methodology in Appendix A, provide the results of an analysis that, along with detailed drawings and sketches as necessary, demonstrates either that such an accident is very unlikely or that the consequences are within allowable limits.

Response

The response to this question will be provided in an FSAR Amendment scheduled for January 1981.

Reference

FSAR subsection 9.1.4. No FSAR change was made.

Question 010.70

In our request 010.13 regarding the adequacy of the component cooling water system (CCWS) and related instrumentation systems to provide assured cooling of the reactor coolant pump (RCP) seals and motor bearings, it was our position that if the CCWS supply and return lines for the RCPs did not meet the single failure criterion, we would require that you demonstrate that the RCPs could operate for about 30 minutes without the loss of function, and that safety grade instrumentation must be provided to detect the loss of CCW to the RCPs and to alarm the operator in the control room. The entire instrumentation system, including audible and visible status indicators for loss of CCW must meet the requirements of IEEE Standard 279-1971/1974. Therefore, demonstrate the adequacy of these instrumentation systems to effect safe shutdown in the event of CCWS failure.

Response

In response to NRC concerns previously expressed in Questions 010.13, 010.29, and 010.59, SCE performed a loss of component cooling water (CCW) test on the reactor coolant pump (RCP) and motor to verify acceptable performance for a minimum of 30 minutes. During this period of time, prior to operator action to either restore CCW flow or to shut off the RCP, there are multiple, diverse alarms in the control room that function to alert the operator to both off-normal RCP and motor parameters and trouble in the CCW system. This alarm instrumentation is detailed in tables 010.31-1 and 010.13-2 and consists of over 40 diverse indications. In addition to these, 1E status indication is provided in the control room for the CCW non-critical loop containment isolation valves.

The instruments and annunciators which display and alarm the output from the above instruments are located in the control room. The annunciators are wired according to a "fail safe" scheme, whereby the annunciator is activated by a loss of signal, whatever the cause. Hence, failure of a non-1E signal would result in annunciation.

The only credible scenario which could result in complete loss of capability to alert the operator to a loss of CCW flow to the RCPs (with the exception of class 1E valve status indication) is loss of "X-Bus" power. Since the RCPs are powered from the "X-Bus," operator action to stop the pumps would not be required.

Based on the above discussion, additional safety grade instrumentation to detect loss of CCW flow to the RCPs is not warranted.

Reference

FSAR Questions 010.13 and 010.29. No FSAR changes were made.

Question 010.71

Provide the design bases and characteristics for the main steam isolation and relief valve enclosure blowout panels, and demonstrate their effectiveness for (1) enclosure overpressure protection and (2) tornado missile protection.

Response

Blowout panels are provided over venting openings in the roof and walls of the reinforced concrete enclosure for the main steam isolation and relief valves. The panels are designed for the following functions:

1. To blow open without becoming dislodged, when a positive pressure of 3 $lb/in.^2g$ is exceeded inside the enclosure.

2. To provide sufficient venting area upon opening of the panels in order to limit the compartment overpressures to the design peak-pressure differentials as specified in FSAR table 010.47-1 (Response to NRC Question 010.47).

3. To remain closed under the postulated tornado transient depressurization of -1.5 $lb/in.^2g$ while retaining blowout capability upon exceeding higher pressures.

4. To provide protection against perforation by tornado-generated missiles. (For definition of tornado event and related missiles refer to FSAR subsection 3.3.2 and paragraph 3.5.1.4.)

The panels fulfill the above functions by means of the following design features:

- The panels are welded steel assemblies fabricated from 1-in. thick plate with 4-in. square and rectangular structural tubing reinforcement on the inside face. The plate thickness provided is sufficient to preclude perforation by tornado-generated missiles.

- The panels swing open by rotating about heavy hinges anchored in the concrete structure. The hinges resist the impact loading upon sudden pressurization and retain the opened panels. The panels are restored by gravity action to their closed position when the internal pressure subsides.

- When initially closed, the panels are restrained to resist the prescribed internal pressure limit by means of the hinges and anchor threaded studs distributed along the three unhinged edges of each panel.

- The anchor studs are 3/8 in. dia. quality class II structural steel bolting material, with a machined-down segment in the stud shank. The reduced shank section is designed such that stud failure and consequent release of panels occurs when the enclosure building internal pressure exceeds 3 $lb/in.^2g$. Therefore, the studs afford sufficient margin to:

(a) resist the tornado depressurization equivalent to an internal pressure of 1.5 $lb/in.^2g$ and (b) fail as designed to ensure that the enclosure building design pressure differentials in Table 010.47-1 are maintained.

- The anchor studs are secured using a double nut arrangement and are easily replaced.

Reference

FSAR subsections 3.3.2 and 3.5.14 and Response to NRC Question 010.47. No FSAR changes were made.

Question 010.73

It is our position that the FSAR contain a statement to the effect that the exhaust air from the fuel pool area be routed through the clean-up filters whenever fuel handling operations are in progress in this area (Reference: Standard Review Plan 9.4.2). Therefore, revise FSAR Section 9.4.3.1, "Fuel Handling Building Ventialtion System" to incorporate this statement.

Response

As discussed in subsection 9.4.3 of the FSAR, the fuel handling building normal ventilation exhaust subsystem includes six pneumatic fail closed, seismic Category I isolation dampers. In the event of a fuel handling accident, a fuel handling isolation signal (FHIS) from redundant airborne radiation monitors located in the exhaust ducts automatically isolates the normal system by closing these isolation dampers within 6 seconds and initiates operation of the emergency recirculation and filtration system. As described in FSAR subsection 15.7.3, the resulting doses are well within the limits of 10CFR 100.

The normal ventilation system which is used during refueling operation is a once-through ventilation system and is designed to maintain the ambient air temperature between 45 and 104F to provide habitable environment for the personnel.

The emergency recirculation and filtration system is designed to remove fission products from the fuel handling building atmosphere following a fuel handling accident rather than to maintain a habitable environment for personnel. Since the normal ventilation system is required to maintain a habitable environment for personnel, the emergency recirculation and filtration system cannot be used in lieu of the normal system during fuel handling operations.

Reference

Refer to FSAR paragraph 9.4.1.3 and subsection 15.7.3.

Question FQ015.45

Your response to Q015.3 was incomplete. Verify that all HVAC wrap and piping insulation have a structural base of noncombustible material (Item 7a) and a potential heat value not exceeding 3500 Btu/lb. in the form in which it is used. Also verify that all interior finishes have a flame-spread rating of not greater than 25 on any surface that would be exposed by cutting through the material on any plane. Identify any materials which do not comply wih these NFPA 220 criteria for limited combustibility material and their estimated weights.

Response

The response to question 015.3 in Amendment 3 to the Fire Hazards Analysis included verification regarding compliance with NFPA 220 criteria for all wrap, insulation, and interior finishes used in the plant. This verification was all inclusive and included the HVAC wrap and piping insulation that are specifically identified in this question. The response in Amendment 3 also identifies the items not in full compliance with NFPA 220 criteria and gives estimated quantities of these items.

Reference

Refer to Fire Hazards Analysis (FHA) Question 015.3 and FHA page I-8. No FSAR or FHA changes were made.

Question FQ015.46

With regard to your response to Q015.4, submit the results of your pressure surge analysis of your fire water system, along with evidence that the valve manufacturer concurs with any finding that there would be no effect on the deluge valve, including the possiblility of a severe water hammer opening fire protection valves not tripped by fire detection systems.

Response

A pressure surge analysis was conducted for the fire water system piping from the containment isolation valve to the automatic sprinkler deluge valves for the reactor coolant pump fire protection spray water systems. The long valve opening time of 25 seconds and the long piping lengths from fire pumps to containment combine to prevent any significant pressure surges. The slow valve opening is sufficient to prevent water hammer type surge and the long piping lengths provide enough flow resistance to prevent high flowrate surges during filling of the empty pipe downstream of the isolation valve. Results of the analysis show the maximum pressure surge is less than 9 $lb/in.^2$, maximum velocity when line is filling is about 14 ft/s, and steady state velocity is about 5.5 ft/s. These values are considered conservative results because no allowance is taken for cushioning effects of air in the line during filling and the breaking up of water in passing through the system so the deluge valve is not hit with a solid slug of water. Therefore, it is concluded that design or operational parameters will not be exceeded as a result of the low pressure and flow transients that are expected when the reactor coolant pump spray systems are operated.

In their letters dated November 6, 1979 and June 25, 1980, Automatic Sprinkler Corporation of America (ASCOA), the supplier of the deluge valve, has confirmed that the deluge valve will unlatch when operated with or without water pressure applied upstream of the valve. In addition, ASCOA has confirmed that the piping installed at San Onofre Units 2&3 meets or exceeds the design and installation requirement as stated in NFPA 13 and is capable of withstanding any anticipated water hammer or oscillation effects.

Reference

Fire Hazards Analysis question 015.4. No FSAR or FHA changes were made.

Question FQ015.47

Provide an automatic water suppression system for Room 425 of auxiliary building, El. 70'-0".

Response

The equivalent fire severity for the general issue room (Room 425) is currently shown as 10 hours in the Fire Hazards Analysis. This severity value was based on using a density value for cotton that is not representative of the density for cotton clothing which is much lower. The combustible loading calculation has been revised using a more reasonable density for cotton clothing and more accurate shelving volume. Based on these changes, the equivalent fire severity is reduced to 1.8 hours. Due to the type of combustible material present, the significantly reduced fire loading, and the absence of safety-related equipment subject to exposure from a fire in the area, an automatic water suppression system is not considered necessary.

The combustible loading summary for Room 425 in the Fire Hazards Analysis, page II-304, is being revised to show the above changes.

Reference

Refer to revised FHA Section II.

Question FQ015.48

It is our position that, in addition to the cable tray deluge system, you must provide an area water suppression system in the cable spreading rooms and cable riser galleries (Zones 12, 29, 30, 41, 42, and 67) to protect against exposure fires. Reference Q015.7a(1).

Response

As stated in the response to question FQ015.7(a), the cable tray deluge system water density rate is based on 0.15 gal/min. per ft^2 of projected surface area of cable trays in accordance with NFPA 15 criteria. Using the flowrate based on the cable tray criteria, the corresponding density varies from approximately 0.52 to 0.69 gal/min. per ft^2 of projected floor surface area for the zones identified (zones 12, 29,30, 41, 42, and 67). This density rate, considered on a zone basis, is approximately double the 0.3 gal/min. per $ft^2$2 of floor area rate that would be provided by an area water suppression system. In addition, redundant safe shutdown cables, separated by less than 20 feet, with the exception of the cable spreading room for the zones identified above, are wrapped with exposure fire barriers having an approximate one-hour fire rating. The above features supplemented with smoke detectors for early warning and fire brigade response with manual hose streams are considered adequate protection for exposure fires in these areas.

Reference

Refer to response to question 015.7(a).
No FSAR changes were made.

Question FQ015.49

Your response to Q015.7a(2) is incomplete. Verify that you have evaluated the hydraulic capability of the water system given simultaneous operation of adjacent fire suppression systems in areas not separated by fire-rated barriers, e.g., cable tunnels and the cable spreading room.

Response

In accordance with requirements of Appendix A of NRC Branch Technical Position Paper 9.5-1, the basic design criteria for the fire protection water supply system requires that the system be capable of delivering the maximum demand of the largest fixed water extinguishing system plus 750 gal/min. These system flow requirements are met with the shortest portion of any one loop main out of service and the highest flow capacity pumps out of service. In applying these criteria, it is assumed that the two electric motor driven pumps combined represent the highest flow capacity pump and both are assumed out of service so the diesel driven pump is operating alone. In addition, flow is assumed in only one flow path to protected areas where more than one parallel flow path exists in the supply piping.

Evaluation of the water system for simultaneous operation of any two adjacent fire suppression systems in areas not separated by fire rated barriers shows design flows and conditions can be satisfied consistent with the basic design criteria requirements defined above for all adjacent system combinations except the two identified below. In order to sustain simultaneous operation of the fire suppression systems in the cable spreading room sections 1 and 2, or the systems in cable tunnel section 7 with any one of the adjacent tunnel sections, it will be necessary to have 2 of the 3 fire pumps operating in any combination, i.e., one motor driven pump and the diesel engine-driven pump or two motor-driven pumps. The most demanding situation is represented by simultaneous operation of the fire suppression systems in cable spreading room sections 1 and 2. When the two motor-driven pumps are operating to supply water to the cable spreading room systems plus the 750 gal/min. hydrant flow, the pumps are operating at approximately 140% of design rated flow of 1500 gal/min. each. Operation of the pumps at this flowrate can be obtained since they must provide 150% of rated capacity to satisfy NFPA 20 requirements. In addition, the pump head flow characteristics are such that the minimum design pressure requirements for the cable spreading room suppression system is available at the increased flow condition identifed.

Reference

FHA Section III, Table III-1, Page III - 34.
Response to FHA question 015.7(a).
No FSAR change was made.

Question FQ015.50

Indicate the size of the fire truck pump and water tank, and the location where the truck will be housed. Provide a diagram showing the existing system layout and proposed modifications, including routings through the plant and locations of all hose connections. Reference Q015.9.

Response

Two fire truck tractors are being provided with one self-powered fire pump mounted on each tractor unit. The fire pumps will be capable of delivering 250 gal/min with the discharge pressure set between 150 and 250 $lb/in.^2$.

Three water tank units with minimum capacity of 6,000 gallons each will be provided to store water for the post-SSE manual fire fighting capability.

The fire trucks will be parked in the railroad access tunnel area of the fuel handling buildings.

A diagram showing the existing fire protection water system layout throughout the plant and the post-SSE seismic upgrade modifications, including location of all fire hose cabinets and hose connections will be available approximately January 1981.

Reference

Refer to NRC Question 015.9. No FSAR or FHA changes were made.

Question FQ015.52

Since your one-hour fire rated walls were tested and found to be acceptable as two-hour fire rated walls and given that one-hour rated fire walls did not have to have fire rated dampers per NFPA 90A, verify that all newly defined two-hour duct penetrations of safety-related area barrier walls are provided with listed fire dampers. Indicate all locations where duct penetrations are not provided with rated fire dampers, or where less than a three-hour rated fire damper is provided in the penetration of a three-hour rated barrier. Also, for those areas where your FHA identified walls as one-hour rated walls, and your subsequent tests have demonstrated a two-hour rating, verify that the one-hour doors will be upgraded to coincide with the two-hour wall ratings.

Indicate the material used for penetration seals and reference a specific design or test method used to qualify the seal for its stated fire rating. Verify that anchored angular steel or other supports will be installed at penetration seals in a manner similar to that used in any tested assemblies. Reference Q015.15.

Response

Listed fire dampers with a fire rating qualified to meet or exceed the fire barrier ratings are provided in all the newly defined two-hour duct penetrations in safety-related area barrier walls. Three-hour rated fire dampers are provided in all duct penetrations through three-hour rated barriers. The fire rating of all doors installed in walls that were upgraded to two-hour rating has also been upgraded to be compatible with the wall rating.

The boundary seal around most penetrating items is made with a silicone foam compound placed between the item and the penetration opening. For high temperature lines, etc., where movement is expected, a flexible boot seal design is used. A glass reinforced silicone fabric is attached to the penetrating item, and to the barrier, to provide a seal on both sides of the barrier. Bulk alumina-silica fiber is packed around the penetrating item through the barrier opening to obtain the required rating for the seal. Reduction systems used to effectively close the barrier opening around the penetration are primarily constructed from alumina-silica in bulk fiber, blanket and refractory fiberboard materials, an expanded perlite high temperature insulation block and necessary steel hardware to hold the materials in place.

All seals are type tested in configurations similar to the intended application and qualified to the required rating through testing in accordance with the requirements of ASTM E119, including hose stream tests. Supporting steel is detailed on drawings included in the approved installation procedure, and installed seals must conform to the typical support arrangement given in the approved drawings in order to be accepted.

Reference

NRC Question 015.15. No FSAR or FHA changes were made.

Question FQ015.53

It is our position that breathing apparatus for fire brigade use be reserved only for fire brigade use. Additional units should be provided for other plant personnel. Verify that a minimum of five self-contained breathing units will be maintained for the exclusive use of fire brigade members during a fire emergency. Reference Q015.17.

Response

At least five self-contained breathing units reserved for exclusive fire brigade use will be located in each of the following areas:

- North Fire Hose House
- South Fire Hose House

Additional self-contained breathing units will be located in the control room area and the radiation protection area outside the containment personnel locks.

This arrangement will enable the fire brigade to approach a fire from up to four locations; breathing units are readily accessible to the fire brigade for use in combating a fire anywhere in the plant.

Reference

Refer to revised FHA section III, table III-1, item D.4 (h) and section III, table III-1, item F.1 (b).

Question FQ015.54

The underground fire water system has insufficient valves to isolate hydrant laterals from essential interior suppression systems. We require that hydrants numbered 1N, 2N, 7N, 8N, 1S, 2S, 3S, and 8S be equipped with isolation valves to avoid the possibility of having important interior fire suppression systems being put out of service becuse of hydrant maintenance. Reference Q015.22.

Response

As stated in the response to question 015.22 in Amendment 3 to the Fire Hazards Analysis, the fire main hydrants are the California break-off type which contains a clapper valve that closes automatically if a hydrant is broken. This feature eliminates the need to isolate a hydrant immediately for maintenance or repair in the event of breakage or failure. This permits time for preparation before maintenance or repair must be done.

Repair procedures will require that the work area be prepared and spare parts be available ahead of time to minimize the time the affected loop section is out of service. In addition, either jumpers from an nonisolated loop section to the standpipes normally served by the isolated section or the post-seismic fire tank trucks will be utilized to assure continuity of fire suppression capability.

Fire watch posting and backup suppression capability are current technical specification requirements.

Due to these design features, repair procedures and technical specification requirements, it is not considered necessary to install shutoff valves in the lateral to each hydrant.

Reference

NRC Question 015.22. No FSAR or FHA changes were made.

Question FQ015.55

It is our position that, because of the potential fire exposure to the control room, an automatic suppression system be provided for the turbine lab area, the instrument repair areas, and the storage areas in the control room support area. In addition, all other control room support areas should be provided with automatic fire detection. Reference Q015.25.

Response

The potential fire exposure to the control room from the control room support areas is minimized by providing automatic fire detection in the control room support areas, the upgrading of 1-hour rated walls to 2-hour rating as a result of tests performed, the regular presence of personnel in the areas, and the negligible combustible fire loadings that are present. In addition, remote shutdown capability, independent of the control room, is provided. Because of these design features, and the existing conditions, an automatic suppression system is not considered necessary for these control room support areas.

Reference

FSAR Question 015.25. No FHA or FSAR changes were made.

Question FQ015.57

It is our position that you provide standpipe hose stations for all areas of the plant, including Zones 28 and 45, in accordance with NFPA 14 requirements. Reference Q015.31.

Response

As stated in the response to Question 015.31, Zones 28 and 45 do not contain any equipment or cabling required for safe shutdown and the equivalent fire severity in these zones is 1 minute. Therefore, portable fire extinguishers are provided in these areas and are considered adequate for the hazards involved.

Reference

Fire Hazards Analysis Question 015.31. No FSAR or FHA changes were made.

Question FQ015.58

It is our position that Zone 30 of the electrical tunnels be provided with standpipe hose stations in accordance with NFPA 14 requirements, considering a maximum of 100 ft. of hose per hose station. Reference Q015.41.

Response

All sections of the electrical cable tunnels (Zone 30) are provided with automatic water spray suppression systems. As a secondary means of suppression, the nearest outside hydrant can be utilized to deliver water for manual fire fighting purposes at any location in the tunnel. Based on the maximum anticipated hose length of 375 ft from any point in the tunnel to the nearest point of water delivery, sufficient pressure will exist at each nozzle to deliver a minimum of two 75 gal/min hose streams.

In addition to the above, as stated in response to Question 015.41 in the Fire Hazard Analysis, electric cable tunnels are included in the areas that will be protected by the Seismic Category I fire protection system. For this purpose, additional hoses are permanently stored near the access into the area, which could be connected to the fire cabinet hoses which receive water supply from the fire truck. Calculations show that based on the maximum anticipated hose length of 375 ft, sufficient pressure will exist at each nozzle to deliver two 75 gal/min hose streams.

Reference

NRC Question 015.41. No Fire Hazards Analysis or FSAR changes were made.

Question FQ015.59

Revise the combustible loading calculations given in the FHA to include the cable loadings which you indicate are in the zone. Reference Q015.43.

Response

The descriptive information in section II of the Fire Hazards Analysis does indicate that cable is present in Zone 3. However, cable in Zone 3 is run inside conduit and therefore is not included in the combustible loading data as described in the analysis approach on page I-3, section B.1.6.

Reference

Fire Hazards Analysis section II. No FSAR or FHA changes were made.

Question FQ015.60

It is our position that all areas which contain redundant safe shutdown systems which are not separated by three-hour fire rated barriers should be provided with an automatic, wet-pipe sprinkler system designed to cover the entire area as well as an early warning smoke detection system. In addition, to allow for possible thermal lag or failure of the suppression system, in those areas where the redundant systems are separated by less than 20 ft. of clear, open air space, an ASTM #E119 rated fire barrier which will completely enclose one of the redundant systems should be provided. The barrier should protect the circuit integrity/equipment availability of that system for one hour under fire test conditions. Areas where such protection is required include the following fire zones:

12 Cable Riser Gallery

13A Emergency HVAC Unit Room 309A

15 Rooms 308A and B, ESF Switchgear Rooms

22 Auxiliary Feedwater Pump Room

23 Spent Fuel Pool Heat Exchanger Room

29 Cable Riser Galleries

30 Electrical Tunnel Elev. 30'-6"

32B Fan Room - 233, 234 - Train B

36 Spent Fuel Pool Pump Room

42 Cable Riser Galleries

44 Intake Structure

48 CCW Heat Exchangers and Piping Rooms, Elev. 8'-0"

63 Corridor, Elev. 50'-0", Control Building

67 Cable Riser Galleries, Radwaste Area, Elev. 63'-6"

72 Corridor 442, Elev. 70'

78 Corridor Room 105

83 Salt Water Cooling Tunnel, Train A, Train B

84 Safety Equipment Building, Elev. 8', A/C Room No. 017

In lieu of the one-hour fire rated barrier, an alternate shutdown system can be provided.

Where safe shutdown capability cannot be assured by barriers, suppression and detection systems, it is our position that an alternate shutdown system should be provided. Such areas include the following fire zones:

5 Cable Riser Gallery

31 Control Room Complex

41 Cable Spreading Room

The alternate shutdown system should be completely independent of the area for which it is being provided such that a fire in either area which damages redundant systems will not affect the shutdown capability from the other area. Reference Q015.44a.

Response

As stated in response to Question FQ15.12, exposure fire barriers are provided for redundant safe shutdown cables separated by less than 20 feet, as required, with the exception of the containment, cable spreading room and control room. This wrapping concept, as a barrier, has been tested to ASTM E-119 temperature profiles and has an approximate 1-hour fire rating.

In addition to the wrapping, automatic suppression systems are provided in those fire zones where fire severity exceeds 1 hour. Automatic suppression systems are not provided for zones with less than a 1-hour fire loading for the following reasons:

1. Wrapping provides an approximate 1-hour protection thereby maintaining integrity of at least one of the safe shutdown trains.

2. Fire barrier ratings exceed the fire severity, which is less than 1-hour, in each zone. Also, as stated in response to Question FQ015.27, the tests showed that the existing construction of the walls provides protection in excess of 2 hours.

3. Manual fire fighting capability is provided in all areas containing redundant safe shutdown systems as required. As stated in response to Question FQ015.9, this capability will exist even after a safe shutdown earthquake.

Smoke detectors are provided for early warning of incipient fires in all areas of high safety-related cable tray concentration outside the containment, as stated in response to Question FQ015.44(d).

As stated in the response to Questions 015.34 and 015.44, alternate shutdown features exist to provide remote safe shutdown capability that is electrically and physically independent of the control room (zone 31) and cable spreading room (zone 41). The cable riser gallery (zone 5), contains only one of the two redundant trains required for safe shutdown of the plant. Thus at least one train will be available for safe shutdown remote from the fire zone, as stated in the revised Fire Hazards Analysis, Zone 5 section.

Reference

See revised Fire Hazards Analysis, Section II, Zone 5, paragraph IIC (2), page II-24.

Question FQ015.61

Your response to Q015.44b is adequate for the concern regarding the control room and cable spreading room separation from the remote shutdown panels. However, you have not addressed remote shutdown for loss of circuits in the areas identified in Question 015.44a. It is our position as stated in Question 015.44a that alternate shutdown systems be provided for areas of the plant in addition to the control room and cable spreading room.

Response

The response to NRC question 015.61 will be provided in an FSAR amendement by January 1981.

Reference

None.

Question 031.1

The following request relates to the environmental qualification information provided for the 600 volt power cables, 480 volt load and motor control centers, diesel driven electrical generating sets and containment building fan motors.

a. Identify the qualified life, for each of the six items, if less than 40 years, provide the documentation method and the reporting plan for replacement after the qualified life.

b. Clearly state the acceptance criteria for the environmental qualification for each of these items.

Response

As discussed with the Equipment Qualification Branch, the response to NRC question 031.1 will be provided in a generic submittal to the NRC as part of the overall environmental qualification review being conducted in accordance with NUREG 0588. Submittal for the 600-volt power cables, containment building fan motors, 480-volt motor control centers and diesel generators, all of which are located in harsh environments, is planned for February 1981. Submittal for the 480-volt load centers, which are located in a benign environment, is planned for November 1981.

Reference

Questions: 031.2, 031.3, 031.4, 031.5, 031.6, 040.69, 040.70, FSAR Table 3.11-1. No FSAR change was made.

Question 031.2

Provide the following information for the 480 volt load centers, 480 volt motor control centers and the diesel driven electrical generating sets.

a. Provide the equiment qualification plans as outlined in Section 5.3 of IEEE Standard 323-1971 (Refer to Table 040.50-1 and Section 3.11-2 of the FSAR). The use of previous operating experience and history may be acceptable for environmental qualification, however, this information must be complete (especially with regard to service conditions and equipment performance) and presented in an auditable form.

b. Provide a date by which the environmental qualification test results will be available for these items. Also, if this date is subsequent to the expected plant operation date provide an interim basis for plant operation.

Response

As discussed with the Equipment Qualification Branch, the response to NRC question 031.2 will be provided in a generic submittal to the NRC as part of the overall environmental qualification review being conducted in accordance with NUREG 0588. Submittal for 480-volt motor control centers and the diesel generator, both of which are located in harsh environments, is planned for February 1981. Submittal for 480-volt load centers, which are located in a benign environment, is planned for November 1981. The basis for interim plant operation will be provided in the February 1981 NUREG 0588 submittal.

Reference

Questions: 031.1, 031.5, 040.69, 040.70,
FSAR Table 3.11-1. No FSAR change was made.

Question 031.5

Provide information which clearly states that the 10^6 Rads documented in the FSAR is enveloped by the qualification plan for the diesel driven electrical generating sets.

Response

As discussed with the Environmental Qualification Branch, the response to NRC question 031.5 will be provided in a generic submittal to the NRC as part of the overall environmental qualification review being conducted in accordance with NUREG 0588. Submittal for equipment located in harsh environments is planned for February 1981.

Reference

Questions 031.1, 031.2,
FSAR Table 3.11-1. No FSAR change was made.

Question 031.6

Provide the following information for the Containment Building Fan Motors.

a. In addition to the qualification parameters (i.e., thermal aging, seismic testing, LOCA testing, etc.) provide the test results of the same type or a similar type motor that uses the insulating materials listed in the Joy Report X-604 subjected to radiation aging (cumulated dose 5 x 10^7 Rads plus margin as stated in the FSAR).

b. Identify the measured motor insulation resistance before the LOCA testing and justify the acceptability of this motor since the motor insulation resistance was zero after testing. Also, state the acceptance criteria for the insulation resistance of this motor and identify the fan motor electrical loading (to include margin) during the LOCA testing.

c. Explicitly identify where the environmental qualification testing was completed considering only LOCA environmental conditions and provide supporting information which demonstrates for any such case that the LOCA environment exceeds or are equivalent to the maximum calculated MSLB conditions.

d. Provide supporting information which clearly indicates that the design and testing conditions for this fan motor envelopes the worst case environmental conditions in the containment.

Response

As discussed with the Equipment Qualification Branch, the response to NRC question 031.6 will be provided in a generic submittal to the NRC as part of the overall environmental qualification review being conducted in accordance with NUREG 0588. Submittal for equipment located in harsh environments is planned for February 1981.

Reference

Questions 031.1, 031.3,
FSAR Table 3.11-1. No FSAR change was made.

Question 031.7

Address the following items which relate to the transmitters.

a. Provide the test report for the transmitters in the balance of plant list that could be subjected to the limiting harsh environmental conditions in the plant. If this transmitter is to be associated with the auxiliary feedwater flow indicator then clearly state that it is environmentally qualified to 10^6 Rads as indicated in the FSAR.

b. State more precisely the installed plant location and define the normal and accident environmental conditions to which the transmitter is to be qualified.

c. Identify the installed and service life of the transmitter and any component part for which the service life is less than the installed life. Also, if the installed and/or service life of this transmitter is less than the 40 year design life, provide the documentation method and the reporting plan for replacement of the transmitter or appropriate component parts after their service life.

Response

As discussed with the Environmental Qualification Branch, the response to NRC question 031.7 will be provided in a generic submittal to the NRC as part of the overall environmental qualification review being conducted in accordance with NUREG 0588. Submittal for equipment located in harsh environments is planned for February 1981.

Reference

FSAR Table 3.11-1. No FSAR change was made.

Question 031.8

For the Electric Motor Valve Actuators, state the acceptance criteria for the valve actuator switch contact chatter and verify that this equipment satisfies this acceptance criteria.

Response

As discussed with the Equipment Qualification Branch, the response to NRC question 031.8 will be provided in a generic submittal to the NRC as part of the overall environmental qualification review being conducted in accordance with NUREG 0588. Submittal for equipment located in harsh environments is planned for February 1981.

Reference

FSAR Table 3.11-1. No FSAR change was made.

Question 032.39

Section 7.3.1 of the FSAR states that the discharge valves of the emergency feedwater system are automatically closed to secure excess feedwater flow when the steam generator water level returns above the low level set point. Provide a detailed description of the operation of these valves, including logic and electrical schematic diagrams. Identify all valves involved in this operation.

Response

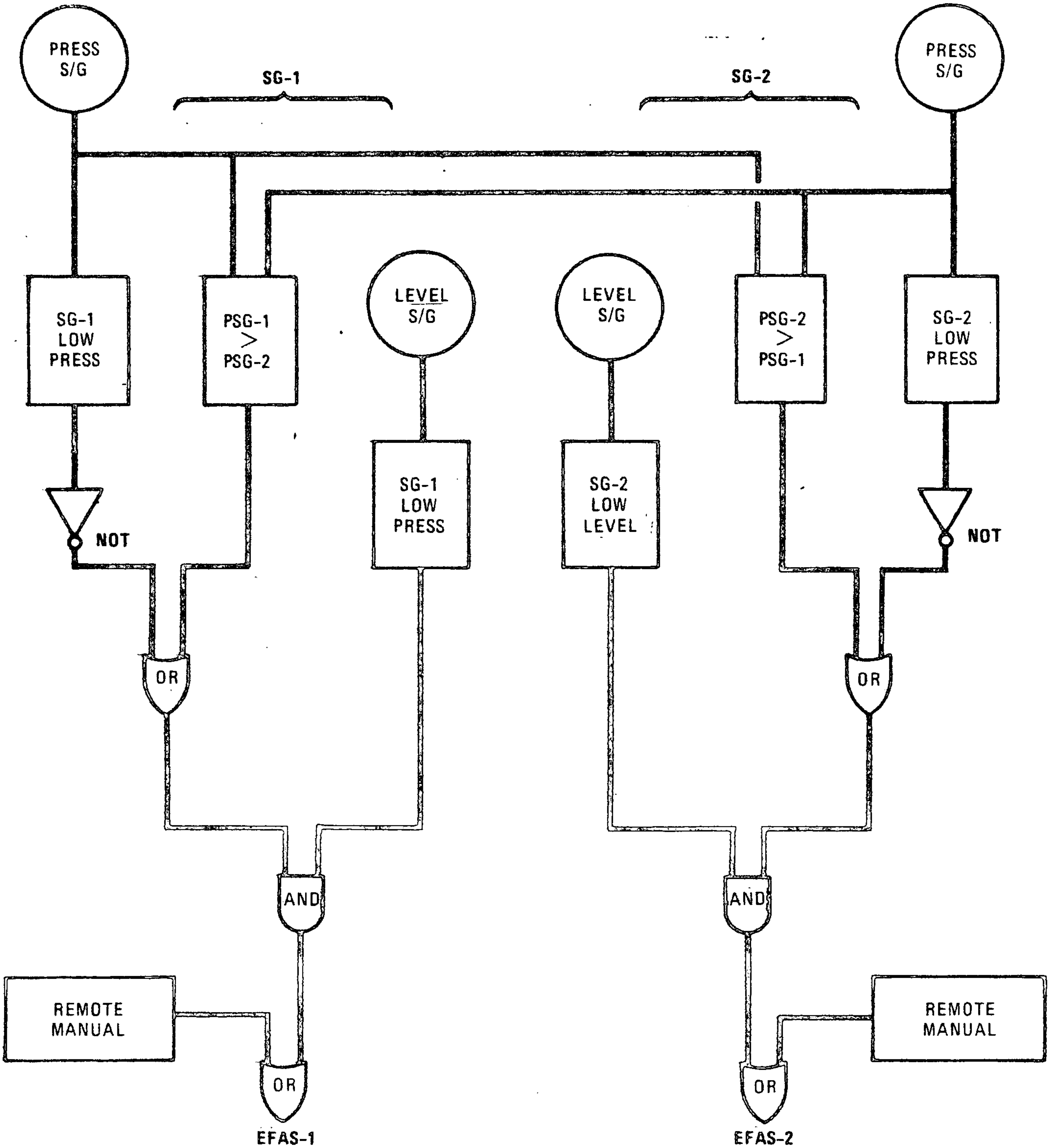
The San Onofre Units 2 and 3 emergency feedwater actuation signal (EFAS) automatically actuates the auxiliary feedwater system by fully opening the isolation and control valves to deliver a minimum feedwater flowrate of 700 gal/min to the intact steam generator(s). The EFAS is initiated for the intact steam generator either by a low steam generator level coincident with no low pressure trip present on the intact unit or by a low steam generator level coincident with a differential pressure between the two steam generators with the higher pressure in the intact unit.

Thw two-out-of four logic is provided independently for each steam generator. When steam generator water level returns to the reset point above the low level setpoint, the auxilliary feedwater system discharge valves will shut automatically as the EFAS is removed to secure excess feedwater flow. The EFAS will continue to function as required to maintain steam generator water level while the plant remains at hot standby or is brought to cold shutdown. Figure 032.39-1 shows the San Onofre Units 2 and 3 EFAS logic and FSAR figure 10.4-9 is the San Onofre Units 2 and 3 auxiliary feedwater system showing the system relationship of the valves, two motor-driven pumps and the turbine driven pump.

The third pump (motor-driven) has recently been added to the system to improve reliability resulting in a redesign of the piping system (FSAR subsection 10.4.9). The EFAS logic was not changed by this action. Subsection 7.3.1 changes to reflect the three pump system will be provided by January, 1980.

Reference

FSAR subsections 7.3.1 and 10.4.9. No FSAR changes were made.



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

SAN ONOFRE UNITS 2 AND 3
EFAS LOGIC

Figure 032.39-1

Question 040.72

Operating experience at certain nuclear power plants which have two cycle turbocharged diesel engines manufactured by the Electromotive Division (EMD) of General Motors driving emergency generators have experienced a significant number of turbocharger mechanical gear drive failures. The failures have occurred as the result of running the emergency diesel generators at no load or light load conditions for extended periods. No load or light load operation could occur during periodic equipment testing or during accident conditions with availability of offsite power. When this equipment is operated under no load conditions insufficient exhaust gas volume is generated to operate the turbocharger. As a result the turbocharger is driven mechanically from a gear drive in order to supply enough combustion air to the engine to maintain rated speed. The turbocharger and mechanical drive gear normally supplied with these engines are not designed for standby service encountered in nuclear power plant application where the equipment may be called upon to operate at no load or light load condition and full rated speed for a prolonged period. The EMD equipment was originally designed for locomotive service where no load speeds for the engine and generator are much lower than full load speeds. The locomotive turbocharged diesel hardly ever runs at full speed except at full load. The EMD has strongly recommended to users of this diesel engine design against operation at no load or light load conditions at full rated speed for extended periods because of the short life expectancy of the turbocharger mechanical gear drive unit normally furnished. No load or light load operation also causes general deterioration in any diesel engine.

To cope with the severe service the equipment is normally subjected to and in the interest of reducing failures and increasing the availability of their equipment EMD has developed a heavy duty turbocharger drive gear unit that can replace existing equipment. This is available as a replacement kit, or engines can be ordered with the heavy duty turbocharger drive gear assembly.

To assure optimum availability of emergency diesel generators on demand, Applicant's who have on order or intend to order emergency generators driven by two cycle diesel engines manufactured by EMD should be provided with the heavy duty turbocharger mechanical drive gear assembly as recommended by EMD for the class of service encountered in nuclear power plants. Confirm your compliance with this requirement.

Response

The San Onofre Nuclear Generating Station Units 2&3 diesel generator units will operate only a few minutes each month in a no-load condition. Plant test procedures will require the diesel generator units be paralleled to the safeguard bus and loaded as quickly as possible (refer to response to Question 040.75). Similiar to emergency operating procedures, the test procedures will also limit the time of no-load operation, and require the operator to shut down the unit if the diesel generator operates more than 30 minutes in a no-load condition.

The engine manufacturer has recommended that the mechanical drive gear assemblies be replaced after 200 cummulative hours of no-load operation or 1000 cummulative hours of operation under a combination of no-load and moderate load operation. These recommendations will be incorporated into the plant maintenance procedures for these diesel units.

The diesel generators are furnished and installed with turbocharger mechanical drive gear assemblies as specified by the engine manufacturer to meet the intended service conditions. EMD of General Motors, the engine manufacturer, has under development a "heavy duty" mechanical drive gear assembly, however, this assembly has not yet completed sufficient testing to qualify it for nuclear service. Subsequent to successful qualification of this new heavy duty drive gear assembly, the replacement of the gear assemblies on the SONGS 2&3 diesel generator units will be considered if there are indications of undue wear on the existing gear assemblies.

Reference

FSAR section 8.3. No FSAR changes were made.

Question 040.73

Several fires have occurred at some operating plants in the area of the diesel engine exhaust manifold and inside the turbocharger housing which have resulted in equipment unavailability. The fires were started from lube oil leaking and accumulating on the engine exhaust manifold and accumulating and igniting inside the turbocharger housing. Accumulation of lube oil in these areas, on some engines, is apparently caused from an excessively long prelube period, generally longer than five minutes, prior to manual starting of a diesel generator. This condition does not occur on an emergency start since the prelube period is minimal.

When manually starting the diesel generators for any reason, to minimize the potential fire hazard and to improve equipment availability, the prelube period should be limited to a maximum of three to five minutes unless otherwise recommended by the diesel engine manufacturer. Confirm your compliance with this requirement or provide your justification for requiring a longer prelube time interval period to manual starting of the diesel generators. Provide the prelube time interval your diesel engine will be exposed to prior to manual start.

Response

The diesel engine manufacturer recommends prelubrication of the engine prior to starting the engine for the first time following a major overhaul or whenever the engine has been shut down for more than 48 hours.

Based upon the engine manufacturer's recommendations, the operating procedures for manual starting of the diesel generators will require that the engines be prelubed for not less than 3 minutes and not more than 5 minutes whenever the above conditions are in effect.

Reference

FSAR section 9.5.7. No FSAR changes were made.

Question 040.76

The availability on demand of an emergency diesel generator is dependent upon, among other things, the proper functioning of its controls and monitoring instrumentation. This equipment is generally panel mounted and in some instances the panels are mounted directly on the diesel generator skid. Major diesel engine damage has occurred at some operating plants from vibration induced wear on skid mounted control and monitoring instrumentation. This sensitive instrumentation is not made to withstand and function accurately for prolonged periods under continuous vibrational stresses normally encountered with internal combustion engines. Operation of sensitive instrumentation under this environment rapidly deteriorates calibration, accuracy and control signal output.

Therefore, except for sensors and other equipment that must be directly mounted on the engine or associated piping, the controls and monitoring instrumentation should be installed on a free standing floor mounted panel separate from the engine skids, and located on a vibration free floor area or equipped with vibration mounts.

Confirm your compliance with the above requirement or provide justification for noncompliance.

Response

To avoid the potential problem of diesel engine damage due to vibrationally induced instrument wear or setpoint drift, the panel-mounted instrumentation will be located on a floor-mounted panel seismically qualified for the service.

Engine-mounted instrumentation subject to vibrational wear or setpoint drift will be periodically tested and recalibrated to assure their continued proper function.

Reference

FSAR paragraph 8.3.1.1.4. No FSAR changes were made.

Question 112.41

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

a. Pre-service Examination

a pre-service examination should be made on all snubbers listed in tables 3.4-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9. This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a minimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

b. Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those system which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

Response

The response to NRC question 112.41 will be provided in an FSAR amendment by January 1981.

Reference

None.

Question 112.42

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action i.e., shutdown or system isolation when the final approved leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance and etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

Response

The response to question 112.42 will be provided in an FSAR amendment by January 1981.

Reference

None.

Question 121.25

According to Section 16.3.9.2.1 of the Technical Specifications, the pressurizer is limited to a maximum heatup and cooldown of 200°F in any one hour period. Paragraph IV.A.2.a, Appendix G, 10 CFR Part 50, requires that the thermal stress intensity factor produced by a heatup and cooldown rate of 200°F/hr plus the membrane stress intensity factor be lower than the reference stress intensity factor by the margins specified in the following equation of Appendix G of the ASME Code:

$$2K_{IM} + K_{It} < K_{IR}$$

To demonstrate compliance with the fracture toughness requirements of Appendix G, 10 CFR Part 50, provide the calculations and analyses used to determine the critical stress intensity factors produced by the membrane tensile stresses and the radial thermal gradient resulting from a heatup and cooldown rate of 200°F/hr. Calculate the reference stress intensity factors, and demonstrate that the margins required by Appendix G, 10 CFR Part 50, are met.

Response

A brittle fracture evaluation of the pressurizer was performed and the results demonstrate clearly that the pressurizer is not the limiting component during plant heatup and cooldown. Performance of this analysis used linear elastic fracture mechanics in accordance with Appendix G of the ASME Code, Section III. Calculated allowable pressure versus fluid temperature curves for heatup and cooldown are shown in the evaluation entitled, "So. Cal. Pressurizer-Brittle Fracture Evaluation, Calc. No. PRS-705," in FSAR table 1.8-7.

The referenced evaluation provides the calculations and analysis used for the San Onofre Units 2 and 3 pressurizer brittle fracture review.

References:

Revised FSAR table 1.8-7 to incorporate reference "So. Cal. Pressurizer-Brittle Fracture Evaluation, Calc. No. PRS-705."

Question 121.27

Information supplied in FSAR Section 16.3/4.4.5 concerning steam generator tube inspection is either incomplete or inadequate. In order to demonstrate compliance with NRC requirements, revise the following areas in this FSAR section to be consistent with NUREG 0212, Revision 1, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors:"

(1) Section 16.4.4.5.1.2.B and .C with regard to first, second, and third sample of tubes at each inspection;

(2) include the additional requirements and acceptance criteria listed in NUREG 0212 regarding eddy current testing in section 16.4.4.5.1.2.B and section 16.4.4.5.1.4.A.1;

(3) in section 16.4.4.5.1.3 add a requirement to increase the inspection frequency of the test results fall into Category C-3;

(4) add a requirement in section 16.4.4.5.1.4.A and Table 16.4-7 for a preservice inspection; and

(5) include the details of the reporting requirements to section 16.4.4.5.1.5 as listed in NUREG 0212.

Response

The Technical Specifications will be amended to conform substantially to NUREG 0212. This specification will be submitted by approximately January 1981.

Reference

None. No FSAR change was made.

Question 121.28

Provide the following information regarding the reactor containment pressure boundary:

a. Identification of the fabrication codes (edition and addenda) and specific paragraphs in these codes that specify the fracture toughness requirements and acceptance criteria (for weldments and base metals). Codes and code paragraphs should be identified for all materials which constitute part of the containment boundary (e.g., piping penetrations, personnel airlocks, equipment hatch).

b. The materials test data that certify that the fracture toughness acceptance standards have been met for each of the identified materials in the containment pressure boundary.

c. Lowest service metal temperature of reactor containment pressure boundary materials.

d. As-built dimensions and materials of construction of flued head of hot line penetration shown in FSAR Figure 3.8-11.

Response

a. The fabrication codes and the specific paragraphs in the codes that specify the fracture toughness and acceptance criteria for weldments and base metals of the reactor containment pressure boundary are as follows:

	Item	Specified Material/Code
1.	Liner Plate	Material and certification of material in accordance with:
	1/4- inch thick plate	ASME SA-285, Grade A
	over 1/4 and up to 5/8-inch thick plate	ASME SA-516, Grade 70
	over 5/8-inch thick plate	ASME SA-516, Grade 70, and compliance with impact test requirements per: ASME B & PV Code, Section III, Subsection NE, Article NE-2320, Impact Test Procedures, at a maximum temperature of 0F, and Article NE-2330, Test Requirements and Acceptance Standards. 1971 Edition, and Addenda thru Winter 1972. Applicable to base metal, heat affected zone, and weld metal.

2.	Penetration Sleeves	Material and certification of material in accordance with:
	Carbon Steel, seamless pipe (30 inches in diameter or less)	ASME SA-333, Grade 1 or 6
	Carbon Steel, welded pipe	ASME SA-155, Grade KCF70, except that plate material comforms with ASME SA-516, Grade 70; and compliance with impact test requirements per:
		ASME B & PV Code, Section III, Subsection NE, Article NE-2320, Impact Test Procedures, at a maximum temperature of 0F, and Article NE-2330, Test Requirements and Acceptance Standards. 1971 Edition, and Addenda thru Winter 1972. Applicable to base metal, heat affected zone, and weld metal.
3.	Personnel Lock and Escape Lock, and their attachments.	Design, materials, fabrication, examination, inspection, testing, construction, installation, and certification in compliance with: ASME B & PV Code, Section III, Subsection NE, Class MC Components.
4.	Equipment Hatch and its attachments.	Design, materials, fabrication, examination, inspection, testing, construction and installation in compliance with: ASME B & PV Code, Section III, Subsection NE, Class MC Components.
5.	Personnel Lock, Escape Lock, and Equipment Hatch and their attachments.	Stress Report in accordance with: ASME B & PV Code, Section III, Subsection NA, Article NA-3350.
		Seismic design in accordance with: ASME B & PV Code, Section III, Subsection NE, Article NE-3130.
		Material and certification of material in accordance with;

5.1	Plate	ASME SA-516, Grade 70, and for plate over 5/8-inch thick compliance with impact test requirements per: ASME B & PV Code, Section III, Subsection NE, Article NE-2320, Impact Test Procedures, at a maximum temperature of OF, and Article NE-2330, Test Requirements and Acceptance Standards. 1971 Edition, and Addenda thru Winter 1972. Applicable to base metal, heat affected zone, and weld metal.
5.2	Forgings	ASME SA-350, Grade LF1 or LF2, and ASME SA-182, Grade F 304.
5.3	Pipe Carbon Steel, seamless pipe (30 inches in diameter or less)	ASME SA-333, Grade 1 or 6
	Carbon Steel, welded pipe	ASME SA-155, Grade KCF70, except that plate material conforms with ASME SA-516, Grade 70; and compliance with impact test requirements per: ASME B & PV Code, Section III, Subsection NE, Article NE-2320, Impact Test Procedures, at a maximum temperature of OF, and Article NE-2330, Test Requirements and Acceptance Standards. 1971 Edition, and Addenda thru Winter 1972. Applicable to base metal, heat affected zone, and weld metal following the final test treatment.
5.4	Castings	ASME SA-216, Grade WCB or ASME SA-351, Grade CF8M.
5.5	Fittings	ASMe SA-420, Grade WPL6 or ASME SA-234, Grade WPB.
5.6	Bolting	ASME SA-193, Grade B7 or B8, with impact tests per ASME B & PV code, Section III, Subsection NE, Class MC.

b. The materials test data that certify that the fracture toughness acceptance standards have been met as required for each of the identified materials in the containment pressure boundary are as follows:

Certified Materials Test Report (CMTR) furnished by the material manufacturer(s), including the following data:

1. Certified reports of the actual results of all required chemical analyses, physical tests, mechanical tests, examinations (including radiographic film), and other tests.

2. A report of repair welds that are required to be radiographed.

3. A statement listing any heat treatments, examinations and other tests required by the materials specifications which have not been performed.

4. A statement giving the manner in which the material is identified, including specific marking.

The corresponding CMTRs are on file at the jobsite office.

c. Lowest service metal temperature of reactor containment pressure boundary materials is as follows:

Liner Plate	80F
Penetration Sleeves	42F
Personnel Lock	55F
Escape Lock and Equipment Hatch	36F

d. As-built dimensions and materials of construction of flued heads are as defined in vendor drawings provided in table 1.8-5. It is noted that the actual flued heads are built to the exacting dimensions and tolerances defined in the drawings, and the only components subject to variation upon erection are the penetration sleeves. As-built dimensions for the penetration sleeves are defined in Drawing 40497, provided in FSAR table 1.8-5.

References

FSAR section 3.8 and revised table 1.8-5.

Question 121.29

Provide the following information for each LP turbine:

a. Turbine type

b. For each disc:

(1) type of material including material specifications
(2) tensile properties data
(3) toughness properties data including Fracture Appearance Transition Temperature and upper energy and temperature
(4) keyway temperatures
(5) critical crack size at operating and design overspeed
(6) crack growth rate
(7) calculated bore and keyway stress at operating and design overspeed
(8) calculated K_{IC} data
(9) minimum yield strength specified for each disc

Response

a. The turbine type has been provided in FSAR paragraph 10.2.2.2.2. Additionally, the turbine is an impulse reaction type.

b. For each disc:

(1) The type of material including material specifications has been provided in FSAR paragraph 10.2.3.1 and table 1.8-3.
(2) Tensile properties have been provided in FSAR paragraph 10.2.3.1.D. Additional data are as follows:
   - 100,000 $lb/in.^2$ min. 0.2% proof stress
   - 120,000 $lb/in.^2$ main ultimate tensile strength
   - 136,000 $lb/in.^2$ max. ultimate tensile strength
(3) The toughness properties data including fracture appearance transition temperature have been provided in FSAR paragraph 10.2.3.1.D. The upper energy is 35 ft-lbs to 96 ft-lbs. The upper temperature is minus 50F to plus 176F.
(4) Keyway temperature: There are no keyways in the discs.
(5) The critical crack size at design overspeed is 2.2 inches radius as provided in FSAR paragraph 10.2.3.2. The critical crack size at operating overspeed is 2.5 inches radius.
(6) The crack growth rate has been provided in FSAR paragraph 10.2.3.2.
(7) Calculated bore and keyway stress at operating and design overspeed:
There are no keyways. The bore stresses at design overspeed have been provided in FSAR Table 10.2-4. At operating overspeed the bore stress is as follows:

Turbine Stage	Bore Stress (lb_f/in^2)
1 & 2	68,300
3 & 4	59,100
5	56,700
6	63,000
7	63,700
8	74,400

(8) The calcualted k_{IC} data has been provided in FSAR paragraph 10.2.3.2.

(9) The minimum yield strength specified for each disc: Material specifications do not stipulate yield strength because stress-strain curves do not show a clear yield for this material. The 0.2% offset proof stress is specified in item (2) above.

Reference

FSAR section 10.2, Revised FSAR Paragraph 10.2.3.1.

Question 121.31

Indicate the turbine discs that will have sufficient moisture in the hub to cause a propensity for stress corrosion cracking.

Response

The disc surfaces likely to be moistened by condensation during steady running are the upstream faces of disc Nos. 3, 4, and 5 in each LP turbine. These faces are expected to be dry in the inner region of the end face, near the bore, but to be moist over the outer region of the end face, near the diaphragm gland, and moist over the panel faces outside the hub.

The turbine vendor (GEC) has experienced stress corrosion cracking only in keyed disc bores.

The elimination of keyways and the use of materials of greater stress corrosion resistance is considered to provide adequate protection against stress corrosion.

Reference

No FSAR changes were made.

Question 121.33

The staff position concerning IWC-1220 exemption criteria, as permitted by the 1974 Edition of Section XI, for Class 2 welds in the emergency core cooling system, the residual heat removal system, and the containment heat removal system, is that a representative sample of welds in these systems must be subjected to inservice volumetric and/or surface examinations. Welds in these safety related systems cannot be completely exempted from volumetric or surface inspection based upon the requirements of 50.55a(b) in 10CFR50, General Design Criteria 36 & 39, and the Summer 1978 Addenda to the 1977 Edition of Section XI. Your ISI program should include a representative sampling of welds and the proposed methods of examination for the ECCS, RHRS, and CHRS welds previously exempted for chemistry control, pressure/temperature conditions, or line size. Identify the lines and welds exampted from examination in the preservice inspection by IWC-1220 criteria.

Response

San Onofre Units 2 and 3 Preservice Examination Program utilized the exemption criteria of ASME Section XI, 1974 Edition through Summer 1975 Addenda, Subarticle IWC-1220 (except for chemistry exemption) since this was approved by the NRC in its adoption of the Code in 10CFR50.55a(b)(2). Because certain lines were exampted from volumetric and/or surface examination by IWC-1220 rules, there was no need to identify each weld in the line. The Preservice Examination Program Plan does identify, however, all exempt lines which receive a visual examination under pressure tests, as required by IWC-2510. These lines are identified as follows:

Line No.	Line No.
143-1"-C-KEO	095-1"-C-KEO
055-12"-C-HEO	071-2"-C-HEO
056-12"-C-HEO	044-4"-C-GEO
096-1"-C-KEO	011-4"-C-GEO
144-1"-C-KEO	013-4"-C-GEO
072-2"-C-HEO	020-4"-C-GEO
073-2"-C-HEO	021-3"-C-FEO
057-12"-C-HEO	025-2"-C-FEO
097-1"-C-KEO	020-2"-C-GEO
141-1"-C-KEO	020-4"-C-GEO
142-1"-C-KEO	018-2"-C-GEO
098-1"-C-KEO	022-2"-C-FEO
074-2"-C-HEO	019-3"-C-FEO
058-12-C-HEO	015-2"-C-GEO
111-12"-C-KEO	012-3"-C-FEO
075-2"-H-KEO	163-3"-C-FEO
077-2"-H-KEO	087-3"-C-FEO
003-24"-C-LLO	016-2"-C-FEO

Line No.	Line No.
004-24"-C-LLO	024-2"-C-FEO
108-24"-C-LLO	017-3"-C-FEO
109-24"-C-LLO	083-1"-C-GEO
002-24"-C-LLO	014-4"-C-GEO
001-24"-C-LLO	021-4"-C-FEO
007-10"-C-LLO	017-4-C-FEO
009-8"-C-LLO	012-2"-C-FEO
008-10"-C-LLO	058-2"-C-FEO
010-8"-C-LLO	146-3"-C-FEO
164-4"-C-FEO	054-8"-C-KEO
052-2"-C-GEO	056-8"-C-KEO
047-2"-C-GEO	139-3"-C-KEI
052-4"-C-GEO	003-8"-C-KEI
049-2"-C-GEO	004-8"-C-KEI
053-2"-C-GEO	048-4"-C-GEO
132-2"-C-GEO	048-1"-C-GEO
048-2"-C-GEO	113-2"-C-LLO
091-2"-C-LLO	119-2"-C-LLO
116-1"-C-LLO	118-2"-C-LLO
107-2"-C-LLO	115-2"-C-LLO
181-1"-C-LLO	117-1"-C-LLO
027-1"-C-LLO	114-2"-C-LLO
122-2"-C-LLO	123-2"-C-LLO
129-2"-C-LLO	

Reference

None

Question 121.34

The preservice inspection program lists Class 1 components exempted from examination by IWB-1220 of Section XI, 1974 Edition including Addenda through Summer 1975. Provide the calculations and assumptions made in determining line sizes exempted under IWB-1220(b)(1) based on reactor coolant makeup capacity.

Response

The exemption criteria of IWB-1220(b)(1) of ASME Section XI, 1974 Edition through Summer 1975 Addenda was not utilized for any Class 1 components at San Onofre Units 2 and 3. Therefore, assumptions and/or calculations were not developed and cannot be provided.

Reference

None

Question 121.35

The San Onofre 2 & 3 PSI program indicates that steam generator and pressurizer nozzle to vessel welds and branch pipe connection welds on lines exceeding 6 inches in diameter will not be examined to the full extent required by the code due to inaccessibility and geometry. Provide the following additional information for our evaluation:

a. The identification of each weld for which this relief request applies.

b. The percentage of the code required examinations performed in the preservice inspection.

c. The construction code examinations performed on these welds.

d. Any supplemental or alternative examinations.

We will require that all areas in the branch pipe connection welds which were not subjected to a volumetric examination be examined by a surface method.

Response

The additional information concerning steam generator and pressurizer nozzle to vessel welds and branch pipe connections exceeding 6 inches in diameter is displayed in tabular form on table 121.35-1. Those areas which were not examined by ultrasonics during the preservice examination were examined during the construction phase (ASME Section III) using both the volumetric and surface methods. These examinations satisfy the preservice requirements under ASME Section XI.

Reference

None

Table-121.35-1 (Sheet 1 of 3)

Nozzle	Weld No.	Exam Cat	Procedure	Extent of Scanning	% of Exam Completed	Code Exam	Supplemental Exam
S/G Inlet	02-003-010	B-D	NIP-747	One Side Only	0^o - 60% 45^o - 84% 60^o - 87%	RT,PT	None
S/G Outlet @ 45^o	02-003-011						
S/G Outlet @ 315^o	02-003-012						
S/G Inlet	02-004-010						
S/G Outlet @ 45^o	02-004-011						
S/G Outlet @ 315^o	02-004-012						
PZR Surge	02-005-009		NIP-742				
PZR Spray	02-005-010						
PZR Safety @ 45^o	02-005-011						
PZR Safety @ 225^o	02-005-012						

Table-121.35-1 (Sheet 2 of 3)

Nozzle	Weld No.	Exam Cat	Procedure	Extent of Scanning	% of Exam Completed	Code Exam	Supplemental Exam
PZR Safety @ 315^{o}	02-005-013	B-D	NIP-742	One Side Only	0^{o} - 60% 45^{o} - 84% 60^{o} - 87%	RT,PT	None
RC Surge	02-006-008	B-J	NIP-755				
RC Drain	02-006-009						
Shutdown Cooling	02-007-009						
RC Drain	02-008-018						
Safety Injection	02-009-009						
RC Spray	02-009-010						
Charging	02-009-011						
RC Drain	02-010-018						
Safety Injection	02-011-009						
RC Spray	02-011-010						
RC Drain	02-012-018						

Table-121.35-1 (Sheet 3 of 3)

Nozzle	Weld No.	Exam Cat	Procedure	Extent of Scanning	% of Exam Completed	Code Exam	Supplemental Exam
Safety Injection	02-013-009	B-J	NIP-755	One Side Only	0° - 60% 45° - 84% 60° - 87%	RT,PT	None
Charging	02-013-010						
RC Drain	02-014-018						
Safety Injection	02-015-009						
S/G Steam	02-042-007	C-B	NIP-764				
Feedwater	02-042-008						
S/G Steam	02-043-007						
Feedwater	02-043-008						

Question 121.36

Standard Review Plan 3.6.1 requires that 100% volumetric examination of high energy fluid system piping welds between containment isolation valves be completed each interval. These augmented inservice inspection requirements exceed Section XI requirements. In order to evalute the degree of compliance with the augmented ISI requirements in SRP 3.6.1, we require the following information:

a. Describe the preservice examinations performed on these welds.

b. Provide a list of the welds in high energy fluid system piping between containment isolation valves that are not being completely examined and a technical justification.

Response

San Onofre Units 2 and 3 Augmented Inservice Inspection requirements are applied to high energy piping between containment isolation valves. The applicable systems are:

- Main Steam (including blowdown)
- Main Feedwater
- Auxiliary Feedwater

Within these systems the welds and piping identified on table 121.36-1 have received a 100% volumetric examination.

Reference

None

Table 121.36-1
(Sheet 1 of 3)

Line No.	Weld No.	Examination Boundary
Main Steam System		
001-40-C-FEO	02-051-031 02-051-032A 02-051-033 02-051-035	Weld 02-051-031 to Penetration 33
	02-053-001 02-053-048 02-053-050 02-053-051A 02-053-070	Penetration 33 to valve HV-8205
363-34-C-HKI	02-053-004 02-053-005A/B 02-053-006 02-053-029 to 02-053-036	Branch connection to relief valves PSV-8401 to 8406
580-26-C-HKI	02-053-053 02-053-055 02-053-056 02-053-057 02-053-058 02-053-059 02-053-061	Branch connection to relief valves PSV-8407 to 8409
595-8-C-HKI	02-053-007 02-053-008 02-053-009 02-053-011 to 02-053-026	Branch connection to valve HV-8419
002-40-C-HKI	02-050-026 02-050-027A/B 02-050-028 02-050-030	Weld 02-050-026 to Penetration 32
	02-052-001 02-052-038A 02-052-040 02-052-041A 02-052-063	Penetration 32 to valve HV-8204

Table 121.36-1
(Sheet 2 of 3)

Line No.	Weld No.	Examination Boundary
309-34-C-HKI	02-052-004 02-052-005A/B 02-052-006 02-052-026 to 02-052-033	Branch connection to relief valves PSV-8410 to 8414
583-26"-C-HKI	02-052-042 02-052-044 to 02-052-049	Branch connections to relief valves PSV-8415 to 8418
582-8"-C-HKI	02-052-007 to 02-05-011 02-052-013 to 02-052-024	Branch connection to valve HV-9421
004-6"-C-HKI	02-052-051 to 02-052-058	Branch connection to 6" pipe to nipple weld 02-052-058
015-6"-C-HKI	02-049-018 to 02-049-022	Penetration 36 to valve HV-4054
016-6"-C-HKI	02-048-037 to 02-048-048	Penetration 37 to valve HV-4054
190-20"-C-GKI	02-044-035 02-044-037	Penetration 28 to valve HV-4052
189-20"-C-GKI	02-045-032 02-045-034	Penetration 29 to valve HV-4048

Auxiliary Feedwater System

Line No.	Weld No.	Examination Boundary
223-6"-C-GKI	02-046-044 to 02-046-057	Penetration 75 to valves HV-4715 & 4731

Table 121.36-1
(Sheet 3 of 3)

Line No.	Weld No.	Examination Boundary
222-6"-C-GKI	02-047-028 to 02-047-032 02-047-035 to 02-047-042	Penetration 78 to valves HV-4714 & 4730

Question 121.37

The PSI program states that ultrasonic examinations of components not covered by Appendix I of the 1974 Edition of the code or Appendix III of the 1977 Edition will have indications greater than 50% of the reference level recorded. The governing specifications for these components is Article 5 of Section V of the ASME Code, which specified that indications greater than 20% must be investigated. Provide the justification to support this deviation from the code in a relief request.

Bolting examination requirements in the 1977 Edition through the Summer 1978 Addenda of the code for your preservice inspection program must meet all of the requirements in the later Edition and Addenda.

Response

The attached request for relief provides the justification to support use of Appendix III, ASME Section XI 1977 Edition through Summer 1978 Addenda.

The bolting examinations of San Onofre Units 2 and 3 bolting meet all the requirements of ASME Section XI, 1977 Edition through Summer 1978 Addenda.

Response

None

Relief Request No. B-6

System: All ASME Class 1 and 2 piping systems

Component: Class 1 piping greater than 1" nominal pipe size.
Class 2 piping greater than 4" nominal pipe size.

Class: . 1 and 2

Function: To provide a pressure boundary to Class 1 and 2 systems.

Examination Requirement: UT examination of Class 1 or Class 2 ferritic steel piping systems shall be conducted in accordance with ASME Section V, Article 5.

Basis for Relief: ASME Section XI, Subarticle IWB-3121 states that inservice nondestructive examination results shall be compared with recorded results of the preservice and prior inservice examinations. In keeping with the interest of the Code, San Onofre's first inservice examination results will be compared to the preservice examination results. Since the 1977 Edition through Summer 1978 Addenda requirements of IWA-2232 only requires recording of reflectors that produce a response greater than 50%, SCE saw no value in recording indications between 20% and 50%.

The present San Onofre Preservice Examination Program for recording of reflectors is verbatim identical to the Code which will be used inservice.

Alternative Examination

None

Question 121.38

Evaluation of examination results is covered by Articles IWC-3000 and ISD-3000 in the code for Class 2 and 3 components respectively. However, both of these Articles are in the course of preparation. Indicate the alternative evaluation procedures you propose to use.

Response

San Onofre Units 2 and 3 Preservice Examination Program utilizes the rules of IWA-3100(a) and (b) which states:

> "Where acceptance standards for a particular component or Examination Category are in course of preparation, evaluation shall be made of any indications detected during any inservice examination that exceed the acceptance standards for materials and welds specified in the Section III edition applicable to the construction of the component in order to determine disposition."
>
> "Alternatively, acceptance standards for Examination Category B-A may be used for Examination Categories B-B, B-C, B-D, C-A and C-B since standards for these categories are in the course of preparation."

Reference

None

Questions 121.39

Supply impact energy data for both the transition and upper shelf energy regions for the following weld seams:
a) 3-203A, 3-203B, 3-203C, and 9-203 of San Onofre Unit No. 3, and
b) 9-203 of San Onofre Unit No. 2.

Response

Table 121.39-1 provides impact energy data for weld seam No. 9-203 of San Onofre Unit 2. Table 121.39-2 provides impact energy data for weld seam Nos. 3-203A, B, and C of San Onofre Unit 3. All three weld seams were fabricated using the same heat of weld wire and lot of flux. Table 121.39-3 provides impact energy data for weld seam No. 9-203 of San Onofre Unit 3.

References

No FSAR change was made.

Table 121.39-1
WELD SEAM 9-203 SAN ONOFRE UNIT 2

IMPACT TESTS

TYPE	TEMP.°F	VALUES			TEMP.°F	VALUES	NDT
CVN		Ft/Lbs	%Shear	MilsLatExp		Drop Weight	
	-60	16	0	9	-60	1 F	-60°F
	-60	15	0	7	-50	2 NF	
	-60	19	0	11	-40	1 NF	
	-40	20	5	11			
	-40	28	10	16			
	-40	32	15	22			
	-20	85	50	53			
	-20	88	50	56			
	-20	76	40	47			
	0	77	40	47			
	0	75	40	45			
	0	99	60	52			
	+20	117	70	74			
	+20	105	60	65			
	+20	114	70	74			
	+60	132	80	77			
	+60	149	100	84			
	+60	123	80	74			
	+100	142	100	82			
	+100	148	100	84			
	+100	140	100	82			

Table 121.39-2

WELD SEAM 3-203A B C IN SAN ONOFRE UNIT 3

IMPACT TESTS

TYPE	TEMP.°F	VALUES			TEMP.°F	VALUES			NDT
CVN		Ft/Lbs	%Shear	MilsLatExp		Ft/Lbs	%Shear	MilsLatExp	
	-104	13	0	7	-10	127	80	70	
	-104	11	0	6	-10	115	70	64	
	-104	20	5	13	-10	117	70	68	
	-80	29	10	22	+10	126	80	78	
	-80	30	10	21	+10	151	100	81	
	-80	24	10	13	+10	156	100	84	
	-40	110	60	66	+50	174	100	86	
	-40	76	40	48	+50	163	100	85	
	-40	114	60	68	+50	162	100	83	
							Drop Weights		
					-70		1 F		-70°F
					-60		2 NF		
					-50		1 NF		

Table 121.39-3
WELD SEAM 9-203 IN SAN ONOFRE UNIT 3

IMPACT TESTS

TYPE	TEMP.°F	VALUES			TEMP.°F	VALUES	NDT
CVN		Ft/Lbs	%Shear	MilsLatExp		Drop Weight	
	-100	13	0	8	-70	1 F	-60°F
	-100	8	0	4	-60	1 F	
	-100	13	0	5	-50	2 NF	
	-80	24	5	15	-40	1 NF	
	-80	43	20	31			
	-80	25	5	17			
	-40	53	25	36			
	-40	69	40	50			
	-40	63	35	44			
	0	83	50	60			
	0	76	40	52			
	0	97	60	67			
	+40	120	90	82			
	+40	118	90	80			
	+40	125	100	82			
	+100	119	100	78			
	+100	117	100	78			
	+100	124	100	83			
	+160	123	100	82			
	+160	121	100	81			
	+160	133	100	82			

Question 121.40

Identify all reactor vessel beltline weld seams and weldment test specimens by the following:

a) weld wire and heat number,
b) flux and lot number, and
c) welding process.

If weldment test specimens were not taken directly from excess vessel shell course materials and welds, identify, in addition to the above, the base metal combinations.

Response

Table 121.40-1 provides weld wire heat number, flux type, and lot numbers for weld seams in the beltline region of San Onofre Unit 2. Similar data is provided for Unit 3 in table 121.40-2. Process, compositional data, and fracture toughness data was supplied in response to NRC Questions 121.11 and 121.12.

References

Responses to NRC Questions 121.11 and 121.12. No FSAR changes were made.

Table 121.40-1
WELD WIRE AND FLUX IDENTIFICATION FOR SAN ONOFRE
UNIT 2 BELTLINE REGION WELBS

Weld Seam No.		Identification
2-203	A	E 8018 C-3 Electrodes, Lot No. EOBC
	B	Same as above
	C	Same as above
3-203	A	Type Mil B-4 Wire, Heat No. 83637 Linde Type 0091 Flux, Lot No. 1122
	B	Same as above
	C	Same as above
9-203		Type Mil B-4 Wire, Heat No. 90130, Linde Type 0091 Flux, Lot No. 0842

Table 121.40-2
WELD WIRE AND FLUX IDENTIFICATION FOR SAN ONOFRE
UNIT 3 BELTLINE REGION WELDS

Weld Seam No.		Identification
2-203	A	Type Mil B-4 Wire, Heat No. 83650, Linde Type 0091 Flux, Lot No. 1122
	B	Same as above
	C	Same as above
3-203	A	Type Mil B-4 Wire, Heat No. 88114, Linde Type 0091 Flux, Lot No. 0145
	B	Same as above
	C	Same as above
9-203		Type Mil B-4 Wire, Heat No. 90069, Linde Type 124 Flux, Lot No. 0951

Question 121.41

Revise Tables 121.24-1, 2, 3, and 4 to include identification of the reactor vessel beltline weld seam that the surveillance program weld metal represents.

Response

The surveillance program weld for San Onofre Units 2 and 3 is represented by weld seam No. 9-203 on both units. A clarifying note has been added to tables 121.24-1 through 121.24-4.

References

NRC Question 121.24 response. No FSAR change was made.

Table 121.24-1

SAN ONOFRE UNIT 2 SURVEILLANCE PROGRAM SHEET 1 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
1	83°	Standby	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal (a) Linde 0091 Lot No. 0842 Mil B-4 Wire Heat No. 90130	12 CVN 3 Tensile	0.03 Cu .003 P 22
				3. HAZ material Plate C-6404-2	12 CVN-T 3 Tensile	0.10 Cu .005 P
2	97°	4	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal (a) Linde 0091 Flux Lot No. 0842 Mil B-4 Weld Wire Heat No. 90130	12 CVN 3 Tensile	0.03 Cu .003 P 22
				3. HAZ material Place C-6404-2	12 CVT-T 3 Tensile	0.10 Cu .005 P

Note: a. Weld metal specimens are fabricated from the same lot of flux and heat of wire as weld seam No. 9-203. 22

Table 121.24-1 (Continued)

SAN ONOFRE UNIT 2 SURVEILLANCE PROGRAM SHEET 2 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
3	104°	17	1.15	1. Plate C-6404-2	12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal (a) 0091 - Heat 0842 B-4 - Heat 90130	12 CVN 3 Tensile	0.30 Cu .003 P
				3. HAZ material Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P
				4. SRM Material HSST Plate 01	12 CVN-L	Ref. C

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Table 121.24-2

SAN ONOFRE UNIT 2 SURVEILLANCE PROGRAM SHEET 1 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
4	284°	30	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal (a) Linde 0091 Flux Lot No. 0842 Mil B-4 Wire Heat No. 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ Material Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P
5	263°	Standby	1.15	1. Plate C-6404-2	12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal (a) Linde 0091 Lot 0842 Mil B-4 Heat 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P
				4. SRM HSST Plate 01	12 CVN	Ref. C

22

22

Note: a. Weld metal specimens are fabricated from the same lot of flux and heat of wire as weld seam No. 9-203. 22

Table 121.24-2 (Continued)

SAN ONOFRE UNIT 2 SURVEILLANCE PROGRAM SHEET 2 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
6	277°	Standby	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal (a) Linde 0091 Lot 0842 Mil B-4 Heat 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ Material Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P

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Table 121.24-3

SAN ONOFRE UNIT 3 SURVEILLANCE PROGRAM SHEET 1 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
1	83°	Standby	1.15	1. Plate C-6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal (a) Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 2 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN-T 3 Tensile	0.05 Cu .008 P
2	97°	7	1.15	1. Plate C6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal (a) Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P

Note: a. Weld metal specimens are fabricated from the same lot of flux and heat of wire as weld seam No. 9-203.

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Table 121.24-3 (Continued)

SAN ONOFRE UNIT 3 SURVEILLANCE PROGRAM SHEET 2 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
3	104°	19	1.15	1. Plate C-6802-1	12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal (a) Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 3 Tensile	0.03 Cu .004 P 22
				3. HAZ Material Plate 6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
				4. SRM Material HSST Plate 01	12 CVN-L	Ref. C

Table 121.24-4

SAN ONOFRE UNIT 3 SURVEILLANCE PROGRAM SHEET 1 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
4	284°	30	1.15	1. Plate C-6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal (a) Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 3 Tensile	0.03 CU .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
5	263°	Standby	1.15	1. Plate C-6802-1	12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal (a) Linde 124 Flux Lot No. 0951 Mil B-4 Heat 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
				4. SRM Material HSST Plate 01	12 CVN	Ref C

Note: a. Weld metal specimens are fabricated from the same lot of flux and heat of wire as weld seam No. 9-203. 22

Table 121.24-4 (Continued)

SAN ONOFRE UNIT 3 SURVEILLANCE PROGRAM SHEET 2 OF 2

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. and Orientation	Chemical[B] Composition
6	277°	Standby	1.15	1. Plate C-6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Lot 0951 Mil B-4 Heat 90069	12 CVN 12 CVN	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P

Question 121.42

Identify the orientation of the Charpy V-notch test specimens (Table 121.26-1) used to establish the upper shelf energy levels of the pump flywheel plate material.

Response

The specimens were oriented in the "Weak" direction of the material. The Charpy notch was oriented in the rolling direction with axis of notch in the thickness direction of the plate material.

References

No FSAR changes were made.

Question 121.43

As required by Paragraph C.1.c of Safety Guide 14, demonstrate that the minimum fracture toughness of the flywheel plate material, ASTM 543, Grade I, Type B, is equivalent to a dynamic stress intensity factor (K_{IC} dynamic) of at least 100 ksi $\sqrt{in}$ at the normal operating temperature of the flywheel by either 1) justifying that the normal operating temperature is 212°F (Table 121.26-1) or 2) that the material has greater than 50 ft-lbs absorbed energy at the normal operating temperature.

Response

The temperature of 212F for the Charpy impact testing was chosen to insure that the upper energy shelf of the material was attained. This was performed to satisfy the criteria specified by Paragraph C.1.b of Safety Guide 14. It is noted that the normal operating temperature of the flywheel is 120F; however, Safety Guide 14, Paragraph C.1.b specifically states that the Charpy impact testing be performed at the upper energy shelf of the material and not at its normal operating temperature.

The minimum fracture toughness of the flywheel plate material was demonstrated to be equivalent to a dynamic stress intensity factor (K_{IC} dynamic) of at least 100 ksi $\sqrt{in}$ by satisfying the requirements of Paragraph C.1.c (3) of Safety Guide 14.

Reference

No FSAR changes were made.

Question 212.160

During our reviews of license applications we have identified concerns related to the containment sump design and its effect on long term cooling following a Loss of Coolant Accident (LOCA).

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or steam which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequte NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

a. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

   This inspection shall be performed at the end of each shutdown as soon as practical before containment isolation.

b. Institute an inspection program according to the requirements of Regulatory Guide 1.82, item 14. This item addresses inspection of the containment sump components including screens and intake structures.

c. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

d. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

   Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

e. Evaluate the extent to which the containment sump(s) in your plant meet the requirements for each of the items previously identified; namely debris, inadequate NPSH, air entrainment, vortex formation, and operator actions.

The following additional guidance is provided for performing this evaluation.

(1) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

(2) Provide a drawing showing the location of the drain sump relative to the containment sumps.

(3) Provide the following information with your evaluation of debris:

(a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump (or torus), the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

(b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

Response

The response to question 212.160 will be provided in an FSAR amendment by January 1981.

Reference

None.

Question 212.161

As the result of our review of your response to our question 212.127 and the "Final Report on Hydraulic Model Studies of Containment Emergency Sump Recirculation Intakes" for SONGS 2&3, we have the following specific questions:

a. What is the influence of north sump operation on south sump performance? Flow straightening by trash racks does not resolve concerns associated with resultant flow stratification.

b. Are there any high pressure pipes in the vicinity of the sumps; if so, how is jet impingement accommodated by the sump design?

c. Are there any drain holes in the ceiling in the vicinity of the sumps; if so, how was the potential for air entrainment accommodated in the design?

d. Address the influence of flow path "C" on the north sump; why isn't the north sump modeled when a failure of pumps in the south sump could lead to counterclockwise rotational patterns from paths B, C and D in the north sump? If this is because of symmetry, show that the tests envelop rotational velocities.

e. Section 5.2 of the sump pump test report indicates that the NPSH required for the spray pump is 24.0 ft. The data you provided in response to our question 212.133 show that the NPSH required for the spray pump is 13.0 ft. Clarify the discrepancy and confirm that all HPSI pumps and spray pumps have sufficient margin in NPSH during the recirculation mode.

Response

The response to question 212.161 will be provided in an FSAR amendment by January 1981.

Reference

None.

Question 212.162

In your response to question 212.157, you have agreed to perform a natural circulation test to demonstrate the capability to cool down to SDCS initiation conditions within 7 hours under minimum cooldown capability. This test will also verify that adequate boron mixing can be achieved using natural circulation. We request that you submit the details of your test procedure for review. We also request that you address the prototypicality of this test to a natural circulation cooldown from full power conditions. In particular, you should address the capability to cooldown to SDCS conditions in 7 hours in light of present knowledge regarding the ST. Lucie cooldown event. (They are presently recommending cooldown rates to SDCS conditions in excess of 7 hours in order to avoid vessel voiding.)

Response

A response to NRC Question 212.162 will be provided in an amendment to the FSAR by February 1981 addressing the San Onofre Units 2 and 3 natural circulation cooldown from full power conditions. The response will address the time required to reach shutdown cooling conditions and the prototypicality of the test to be performed.

Reference

FSAR section 5.4.7. No FSAR change was made.

Question 212.163

At a meeting on August 15, 1980, the staff informed you that your response to question 212.152 was unsatisfactory. The Standard Review Plan (NUREG 75/087) Section 15.4.6 requires that redundant alarms not subject to a single failure be provided to alert the operator of an unplanned dilution event. The staff requests that you describe in detail the redundant alarms which will signal an unplanned dilution during all modes of operation including cooldown.

Response

In addition to the boron dilution alarm provided by the boronometer as discussed in response to Question 212.152, redundant alarms actuated by the source range nuclear instrumentation and annunciated in the control room will be provided to alert the operator to an unplanned boron dilution event in the subcritical operating modes.

Limiting boron dilution events in MODES 3, 4, 5 and 6 were analyzed to determine times to complete loss of shutdown margin, and corresponding neutron flux responses at the startup channel excore detectors. Based on these responses, startup channel alarm setpoints on high neutron flux were established to satisfy the requirements of SRP 15.4.6. This alarm setpoint protection replaces the original procedural response to NRC Question 212.152. In MODES 1 and 2, the operator will be alerted to a boron dilution by one or more of the following alarms: Power Dependent Insertion Limit alarm, high power level alarm or trip, T-average alarm, or high logarithmic power alarm or trip. A detailed discussion of these alarms is given in the response to Question 212.152. In the subcritical modes, the limiting boron dilution event results in the quickest approach to complete loss of shutdown margin, i.e., inadvertent criticality. The limiting dilution event in each subcritical mode was modeled using conservative plant and core parameters. The initial assumed shutdown margin for each event corresponded to the minimum shutdown margin required by the Technical Specifications for the assumed mode of operation. This analysis and the corresponding startup channel alarm setpoint provide protection for the situation when the RCS is partially drained in MODE 5 to permit system maintenance. Because the RCS liquid volume is reduced, and MODE 5 has the smallest required shutdown margin, a dilution event during this plant condition will result in the shortest time to criticality. The reduced RCS volume dilution event was not previously analyzed, but protection is provided by the startup channel alarm.

References

NRC Question 212.152 and its response; FSAR paragraphs 15.4.1.4.2, 15.4.1.4.3, and 7.7.1.1.2 are modified by this question response.

Question 212.164

The staff has reviewed the shutdown cooling system design of San Onofre 2 and 3 for compliance to Reactor Systems Branch Technical Position 5-1 (as to be implemented for Class 2 plants). We have concluded that your present design does not meet that part of BTP 5-1 which requires the operator to be able to bring the plant from normal operating conditions to SDCS entry from the control room. It is our understanding that at least nine (9) valves in the SDCS train need to be manually repositioned from outside of the control room in order to realign from the safety injection to the SDC mode of operation.

It is the staff position that the SDCS design of San Onofre 2 and 3 be revised to comply with the above. We request that you submit the appropriate documentation of your design revision for staff approval prior to installation. Included in your submittal should be an evaluation which demonstrates that the modifications made do not significantly reduce the reliability of ECCS.

Because of the extent of the modifications necessary for compliance, we do not require that compliance be completed prior to your scheduled OL issuance. Rather, we will accept an extended schedule for completing the necessary design revisions. We propose that an acceptable schedule for completing the necessary design revisions is by the end of your first refueling outage.

Your response should acknowledge your acceptance of the staff position and either the acceptability of our proposed implementation schedule or a justifiable alternate schedule.

Response

The SDCS will be redesigned to permit realignment from the safety injection mode to the shutdown cooling system mode from the control room. Because the design changes required to establish this objective have not been finalized no schedule commitment for this change can be made at this time, however, it is the applicants intent to complete the modifications at the first refueling outage. A firm schedule commitment for completion of these modifications will be provided April 1981.

Reference

None.

Question 212.165

Your response to TMI-related requirement item II.B.1 is not sufficient. Provide all necessary information for your proposed Reactor Coolant System Vents including a detail system description, results of analyses, P&IDs, operating procedures and technical specifications as required in the attached clarification for this item.

Response

Descriptions and discussions concerning the San Onofre Units 2 and 3 reactor coolant gas vent system have been provided in the Response to NRC Action Plan NUREG 0660 (item II.B.1) and the FSAR text has been changed in Amendment 21 (subsection 9.3.7) to also include the reactor coolant gas vent system. To further clarify the San Onofre Units 2 and 3 reactor coolant gas vent system compliance with the clarification attached to NRC Question 212.165 an item-by-item discussion of the requirements is provided along with a statement of compliance for the San Onofre system. A technical specification for San Onofre 2 and 3 will be provided based on the NRC Standard Technical Specifications. That discussion is as follows:

1. NRC Position

"Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation."

SONGS Compliance

The current RCGVS design meets the revised NRC position in the following manner:

(1) The sytem is operable from the control room.
(2) The orifices restrict mass flow from a break in the newly installed portion of the system to less than the definition of a LOCA. Thus there is no increase in LOCA probability due to addition of RCGVS.
(3) The design meets the requirements of 10 CFR 50, Appendix A.

(4) The design provides sufficient redundancy action by using locked closed isolation valves in series. These valves are locked closed from the control room through the use of key lock hand switches.

(5) Challenges to containment integrity have not been specifically addressed since the operator can terminate venting at any time to allow hydrógen recombiners to reduce hydrogen concentration.

2. NRC Position

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

SONGS Compliance

The current RCGVS design meets the revised NRC position in the following manner:

(1) The system description is provided in FSAR subsection 9.3.7. A LOCA analysis for the system is not required since orifices limit mass loss in the event of vent line break to less than the LOCA definition.

(2) Procedures and supporting graphs are provided in the Procedural Guidelines document. The use of instrumentation is discussed therein. The Procedural Guidelines document will be submitted for NRC review.

3. NRC Position

The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling.

SONGS Compliance

The SONGS system meets this requirement.

4. NRC Position

Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basis must not result in a violation of the requirements of 10 CFR 50.44 or 10 CFR 50.46.

<u>SONGS Compliance</u>

(1) Procedures provided in the Procedural Guideline document address when to use the vent system. Although when not to use the system is not explicitly stated, by implication, it is not to be used unless there is a bubble to be removed.

(2) The procedures do not differentiate between use for events within or outside the normal design basis. Venting can be terminated at any time by the operator, if the requirements of 10 CFR 50.44 or 46 are approached.

(3) Procedures are directed toward increasing the plant's ability to maintain core cooling.

5. NRC Position

The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with the vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered, is to specify a volume of noncondensible gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside the containment must be considered (e.g., into a decay gas collection and storage system).

<u>SONGS Compliance</u>

(1) The sizing criteria is detailed in the System Description.

(2) The vent system has the capability to discharge either to containment or to the quench tank which in turn can be vented to the waste gas management system outside containment.

6. NRC Position

Where practical, the reactor coolant system vents should be kept smaller than the size corresponding to the definition of LOCA (10 CFR 50, Appendix A). This will minimize the challenges to the emergency core cooling system (ECCS) since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation, although it may result in leakage beyond technical specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve in series with a vent valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed.

SONGS Compliance

The orofices limit mass loss to less than LOCA definition.

7. NRC Position

A positive indication of valve position should be provided in the control room.

The reactor coolant vent system shall be operable from the control room.

SONGS Compliance

Positive indication and controls are provided in the control room.

8. NRC Opsition

Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary, all requirements for the reactor pressure boundary must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures may be a viable option to meet the single-failure criterion. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10 CFR 50.46.

SONGS Compliance

Series valves provide adequate redundancy and RC pressure boundary criteria.

All the requirements of the reactor coolant pressure boundary are met.

9. NRC Position

The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

SONGS Compliance

(1) Parallel fail closed valves are provided. Each path is powered from a separate emergency bus.

(2) Series isolation is provided in the event that a valve does fail open.

10. NRC Position

Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

SONGS Compliance

(1) The vents discharge into upper containment where containment fans assure mixing.

(2) The vent system may also be aligned to the quench (relief) tank and from there to the waste gas management system outside containment.

11. NRC Position

The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE 344-1975 as supplemented by Regulatory Guide 1.100, 1.92 and SEP 3.92, 3.43, and 3.10. Environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

SONGS Compliance

(1) Piping and supports are designed to meet the revised requirements.
(2) The manual valves are capable of withstanding a seismic acceleration of 4.5g in each of the two horizontal directions applied simultaneously with 3.0g in the vertical direction.

(3) The solenoid valves are qualified to IEEE-382-1972 for inside containment, IEEE-344-1975 for seismic and IEEE-323-1974 for environmental qualification. This qualification also includes the position indicator switches.
(4) Cable termination insulation material is environmentally qualified to NUREG 0588 Category II guidelines.

12. NRC Position

Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

SONGS Compliance

Operability testing of the system (i.e. verification of valve operability and flow path verification) has been considered in the design. The system will be included in the scope of testing to subsection IWV of ASME Section IX.

13. NRC Position

It is important that the displays and controls added to the control room as a result of this requirement do not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,
(b) integration into emergency procedures,
(c) integration into operator training, and
(d) other alarms during emergency and need for prioritization of alarms.

SONGS Compliance

(1) A human-factor analysis of the San Onofre control room is in progress and takes into consideration the use of instrumentation by the operator and prioritization of alarms.
(2) Operator training will include use of the RCVGS. System conditions during normal operations as well as operating procedures for emergency conditions will be addressed in the scope of this training.

14. NRC Position

1. Each PWR licensee should provide the capability to vent the reactor vessel head. The reactor vessel head vent should be capable of venting non-condensible gas from the reactor vessel

hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths).

2. Additional venting capability is required for those portions of each hot leg that connot be vented through the reactor vessel head vent or pressurizer. It is impractical to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that assures sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the RCS. Such operating procedures should incorporate this consideration.

3. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

SONGS Compliance

1. Venting capability for the vessel and pressurizer is provided.
2. It is not practicable to provide separate hot leg vents for steam generator U-Tubes.

Reference

No FSAR changes were made.

Question 221.21

Provide a description of the in-core thermocouple system. Include a description of the primary and backup means of monitoring in-core thermocouple temperature and readout/printout capability. State the time required to complete thermocouple mapping.

Response

A description of the San Onofre Units 2 and 3 in-core thermocouple system as discussed in NRC Question 221.21 will be provided by January, 1981.

References

No FSAR change was made.

Question 221.22

Provide complete "Information Required on the Subcooling Meter" defined in the October 30, 1979 letter from H. Denton (NRC) to All Operating Nuclear Power Plants.

Response

Data regarding the sub-cooled margin monitor as discussed in the October 30, 1979 NRC letter from H. Denton to all operating Nuclear Power Plants is presented on table 221.22-1. Additional descriptions of the sub-cooled margin monitor can be found in FSAR paragraphs 7.1.1.7 and 7.5.1.9 and in the Response to NRC Action Plan NUREG 0660, item II.F.2.

References

FSAR paragraphs 7.1.1.7 and 7.5.1.9; Response to NRC Action Plan NUREG 0660. No FSAR change was made.

Table 221.22-1

SUBCOOLED MARGIN MONITOR DATA
(Sheet 1 of 3)

Display Information Displayed (T-Tsat, Tsat, Press, etc.)	Selectable
	1. Pressure or Temperature margin 2. Tsat or Psat
Display Type (Analog, Digital, CRT)	Digital Meter
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Main Control Board
Alarms (include setpoints)	30F reset at 35F
Overall uncertainty (^{O}F, PSI)	Not Available
Range of Display	0-3000 lb/in.2 0-710F
Qualifications (seismic, environmental, IEEE 323)	IEEE 323-1975 Seismic IEEE 323-1974 Environment
Calculator	
Type (process computer, dedicated digital or analog calc.)	Dedicated digital Microprocessor
If process computer is used specify availability. (% of time)	NA
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	Highest Temp. RCS hot leg Lowest Pressurizer Pressurizer

Table 221.22-1

SUBCOOLED MARGIN MONITOR DATA
(Sheet 2 of 3)

Display	
Information Displayed (T-Tsat, Tsat, Press, Etc.)	Selectable
Qualifications (seismic, environmental, IEEE 323)	IEEE 344-1975 Seismic IEEE 323-1974 Environment
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables
Input	
Temperature (RTD's or T/C's)	RTD's
Temperature (number of sensors and locations)	T_{cold}: TE-0915-2 TE-0911Y1 TE-0925-1 TE-0921Y2 T_{hot}: TE-0911X1 TE-0921X2
Range of temperature sensors	0-710^{o}F
Uncertainty of temperature sensors (^{o}F at 1)	Not Available
Qualifications (seismic, environmental, IEEE 323)	IEEE 344-1975 Seismic IEEE 323-1971 Environmental
Pressure (specify instrument used)	Diaphragm Type Electronic Transmitter
Pressure (number of sensors and locations)	2 Pressurizer Press. Sensors PT-0102-1, PT-0102-2
Range of Pressure sensors	0-3000 lb/in.2
Uncertainty* of pressure sensors (PSI at 1)	Not Available
Qualifications (seismic, environmental, IEEE 323)	IEEE-344-1971 Seismic IEEE-323-1971 Environmental

Table 221.22-1

SUBCOOLED MARGIN MONITOR DATA
(Sheet 3 of 3)

Display Information Displayed (T-Tsat, Tsat, Press, etc.)	Selectable
Backup Capability Availability of Temp & Press	Temperatures - all hot and cold legs indicated, both hot logs recorded one cold leg per steam generator recorded. Pressure - four channels of pressurizer pressure indication in addition to indication of the two channels which provide input to the SMM.
Availability of Steam Tables etc.	*__________
Training of operators	*__________
Procedures	*__________

*These items will be completed prior to fuel load

Question 221.23

Provide your schedule for the procurement, testing and installation of reactor vessel water level instrumentation at San Onofre 2 and 3.

Response

The San Onofre Units 2 and 3 reactor vessel water level instrumentation equipment schedule for procurement, testing and installation is as follows:

Reactor Vessel Internals Changes (Installation of instrument and detector holders, and upper guide structure modifications)	- Complete by Fuel Load
Heated Junction Thermocouple Instrumentation (including detectors)	- Procurement - 11/80 - 1/82 Testing - first refueling Installation- first refueling

Reference

Response to NRC Action Plan NUREG 0660 item II.F.2. No FSAR change was made.

Question 231.34

The NRC staff has been generically evaluating three materials models that are used in ECCS evaluations. Those models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have (a) discussed our evaluation with vendors and other industry representatives (Reference 1), (b) published NUREG 0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Reference 2), and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG 0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (References 3 and 4).

Until we have completed our generic review and implemented new acceptance criteria for cladding models, we will require that the ECCS analyses in your FSAR be accompanied by supplemental calculations to be performed with the materials models of NUREG 0630. For these supplemental calculations only, we will accept other compensatory model changes that may not yet be approved by the NRC, but are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above.

Please provide the supplemental calculations described above.

References

1. Memorandun from R. P. Denise, NRC, to R. J Mattson, "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," November 20, 1979. Available in NRC for inspection and copying for a fee.

2. D.A. Powers and R. O Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG 0630, April 1980. Available from the NRC Division of Technical Information and Docket Control.

3. Letter from D. G. Eisenhut, NRC, to all Operating Light Water Reactors, dated November 9, 1979. Available in NRC PDR for inspection and copying for a fee.

4. Memorandum from H. R. Denton, NRC, to Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979. Available in NRC PDR for inspection and copying for a fee.

Response

A response stating the San Onofre Units 2 and 3 schedule for supplemental calculations with the materials models of NUREG 0630 will be provided by January 1981.

Reference

No FSAR change was made.

Question 312.44

The post-accident air cleanup system for the fuel handling area is designed as a full flow recirculation system with redundant filter units. The system is not designed to produce and maintain a negative pressure in the building. Your model for the analysis of the radiological consequences from a fuel handling accident in the fuel building includes the following two assumptions: (1) The activity released from the fuel pool surface diffuses instantaneously to uniformly occupy the fuel building volume; and (2) There is no unfiltered leakage from the building to the environment.

a. With respect to the first assumption we note that the openings in the air intake and return ducts of the system are located approximately at the 110 feet elevation close to the roof of the building and approximately 50 feet above the surface of the spent fuel pool. The return duct openings are located as close as seven feet from the intake openings. The current design and operation of the system potentially can short-circuit the intended airflow and mixing of the atmosphere and therefore may not provide for an effective air cleanup, i.e., removal of radioiodine released from the pool surface during the accident.

   We request that you provide an analysis of the air flow characteristics in the building that demonstrates the effectiveness of the system. Such analysis should take into consideration potential temperature gradients in the building that would inhibit natural convection flow. If your analysis shows that the existing system cannot assure the required mixing of the building atmosphere the relocation of the air intakes to within close proximity of the spent fuel pool would be an acceptable approach for providing an increased sweep action over the pool. Because such relocation would be limited by the required travel of the fuel handling bridge over the pool, you should consider a location of the intakes at the wall of the fuel building.

b. With respect to the second assumption, in your analysis the post-accident cleanup system is modeled as a once-through ventilation and filter system discharging directly to the environment as described in your response to our earlier question 312.38. While this model maximizes the offsite doses with respect to filtered leakage it does not consider the contribution from "actual exfiltration" which should be assumed to be unfiltered leakage. Such exfiltration could arise as a result of a pressure difference between the building internal pressure and the outside barometric pressure. Although the staff finds that the fuel handling building, in comparison with such buildings at other facilities, has been designed and constructed to greatly reduce such leakage we cannot conclude that it is a zero leakage building. We therefore request that you provide an analysis that defines the actual exfiltration rate under a slight overpressure (about 0.1 inches water gauge) in the fuel building. An acceptable approach would be the determination, by test, of the necessary air flow into the building that would produce and maintain the slight overpressure. Such test should be performed with the post-accident air cleanup system in full operation.

Response

a. In order to provide adequate air circulation and mixing characteristics during the operation of the post accident cleanup system, four-way diffuser type air registers will be replaced with new high throw supply air outlets which will have a minimum of 50'-0" throw at zero angle with a straight downwards discharge. It was verified that sufficient air pressure is available in the ducts at the supply air outlets to obtain 50'-0" throw and thus deliver the supply air at the fuel pool surface level. This will result in adequate circulation and mixing at the fuel pool surface. The existing location of the return air intakes will ensure a good mixing pattern within the building. Thus, as a result of this modification, it is not considered necessary to lower the return air intakes.

An analysis is being performed to verify that an effective mixing of the air flow within the spent fuel pool area will occur. This analysis takes into account the potential temperature gradients, performance characteristics of the new supply air registers and the air intakes at their present location.

b. In response to Question 312.38, it was stated that during preoperational testing, a negative pressure differential in the spent fuel pool area will be verified during operation of the normal ventilation system. During this testing, sufficient data will be obtained to determine the infiltration rate at the test differential pressure. This data could be utilized to determine the exfiltration rate when slight overpressure (about 0.1 inches water gauge) exists in the spent fuel pool area. Therefore an additional test to determine exfiltration rate is not considered necessary.

As discussed in response to Question 312.38 and FSAR subsection 15.7.3, the iodine removal efficiencies assumed for the fuel handling post-accident cleanup units are less than the efficiencies allowed to be assumed by Regulatory Guide 1.52. Since some of the filtered leakage is assumed to be unfiltered, this results in a conservative model.

Reference

Refer to response to Question No. 312.38 and FSAR subsection 15.7.3.

Question 312.45

Your analysis of the radiological consequences resulting from continuous post-LOCA leakage from ESF components located outside containment is based on the leakage sources listed in Table 15.6-19. These leakage sources include the valve stems and pump seals of the high and low pressure injection pumps and the containment spray pumps. It is our understanding that the valves listed in the table are located in various rooms within the ESF building. We request the following information:

(1) Provide a listing and identify the location, by room, of the valves in each of the ESF systems.

(2) Describe the potential leakage path(s) to the outside environment from each of the locations in (1) above.

(3) Provide the bases for the leak rates from valve stems and seals as listed in Table 15.6-19 that were used in your analysis.

(4) Propose technical specifications and surveillance requirements for the valves and seals listed in Table 15.6-9 above to assure that the leak rates listed will not be exceeded.

Response

a. Valves which contain post-LOCA recirculating fluid are listed by room in table 312.45-1.

The detailed analysis from table 312.45-1 results in an actual maximum expected leakage of 540 cm^3/hr which verifies that the maximum expected leakages from valve stems of 843 cm^3/hr as stated in table 15.6-19 are conservative.

b. While each of the above rooms has a slightly different but highly tortuous release path to the environment, these leakage paths are conservatively considered when evaluating the radiological consequences of a postulated LOCA. To calculate the resultant activity which is eventually released to the environment, the airborne activity from all the above rooms is assumed to be instantaneously and directly released at the site boundary. No credit is taken for ground disposition or radioactive decay during transit to the exclusion area boundary or LPZ outer boundary.

The leakage paths from the pump rooms generally are through piping penetrations in the pump room walls since personal access is provided through water tight doors.

From the safety equipment building pump room area, there are three major paths to the outside environs. First is a 160 ft pipe tunnel with no forced air ventilation which connects the pump rooms to the tankage building. Second is a pipe hose which leads from the pump rooms to the penetration area. Leakage paths from the penetration area include doorways and the rubber seals used for the seismic gap. The third major path involves leakage from the unsealed piping penetration in the LPSI and HPSI pump rooms. The activity released proceeds via the stairway and equipment hatches to the upper elevations of the safety equipment building. From the safety equipment building, the leakage can continue to the atmosphere through piping penetrations, and equipment hatches.

Since these leakage paths are tortuous and would include a long delay time, the assumption of instantaneous release is highly conservative.

c. The maximum leakage rates for the HPSI and LPSI pump seals are listed in FSAR table 15.6-19 as 50 cm^3/hr/seal. This leakage is based on a criteria in the specification for these pumps which states that leakage under operating conditions shall not exceed 50 cc/hr/seal. In addition, response to NRC question 312.27 presented an analysis of a gross seal failure for a leak rate of 500 cm^3/min with acceptable offsite dire consequences.

Table 15.6-19 also specifies the maximum stem leakage as 10 cc/hr/inch stem diameter. These valves were purchased under several specifications. Those purchased by the NSSS vendor have in their specification a test requirement which states that stem leakage shall not exceed 10 cc/hr/inch stem diameter. The other specifications require that there be no visible leakage from the stem. If leakage is observed, the packing gland will be tightened. All of the gate and globe valves are backseated, and all of the Non-NSSS supplied valves are double packed.

In addition to the specification requirements, operating experience with valves of similar design at San Onofre Unit 1 indicates that no valve stem leakage is expected. This conclusion is supported by visual inspection during operation and by the absence of crystalized boric acid in the valve bodies.

It is considered incredible that all valves and pump seals will be leaking simultaneously at the maximum rate continuously during post-LOCA recirculation. Nevertheless, for additional conservatism, a leak rate of twice the assumed maximum expected leak rate (2486 cm^3/hr) was used in the analysis and this leak rate was assumed constant for the duration of the accident.

In addition, SCE is establishing a leak reduction program which will be fully implemented prior to operation above full power. This program will help ensure that the potential sources of leakage discussed above are minimized. This program is described in the response to NRC Question 321.10.

SONGS 2 and 3 will provide a section for the plant Technical Specifications requiring verification on a periodic basis that the ESF components in FSAR Table 15.6-19 do not exceed the specified maximum leakages. The surveillance requirements for these ESF components will be consistent with the general requirements in the leakage reduction program described in the response to Question 321.10.

Reference

FSAR section 15.6 and response to NRC Question 321.10. No FSAR change was made.

Table 312.45-1
VALVES CONTAINING POST-LOCA RECIRCULATING FLUIDS (Sheet 1 of 3)

Valve (Size)	Normally Open/Closed	Type	Stem Diameter (in.)	Maximum Expected Leakage (cc/hr)
Sump Line Area				
2HV-9303 (24")	Normally Open	Butterfly	2	20
2HV-9302 (24")	Normally Open	Butterfly	2	20
2"-048-C-376	Locked Closed	Packless Globe (1)	N/A	-
2"-047-C-376	Locked Closed	Packless Globe (1)	N/A	-
24"-004-C-724	N/A	Split Disk Check	N/A	-
24"-003-C-724	N/A	Split Disk Check	N/A	-
LPSI Pump #2				
10"-008-0-675	N/A	Check	N/A	-
2"-035-C-329	Locked Open	Stop Check	1	10
8"-009-C-212	Locked Open	Gate	1.25	12.5
4"-015-C-358	Locked Open	Stop Check	1.5	15
16"-087-C-675	N/A	Check	N/A	-
8"-014-C-406	Locked Open	Stop Check	2	20
2"-011-C-329	Locked Open	Stop Check	1	10
16"-005-C-212	Locked Open	Gate	1.75	17.5
LPSI Pump #1				
2"-034-C-329	Locked Open	Stop Check	1	10
4"-012-C-358	Locked Open	Stop Check	1.5	15
4"-013-C-075	Locked Closed	Gate	1.25	1.25
16"-062-C-212	Locked Open	Gate	1.75	17.5
16"-088-C-675	N/A	Check	N/A	-
8"-012-C-406	Locked Open	Stop Check	2	20
2"-010-C-329	Locked Open	Stop Check	1	10

(1) Packless valves have diaphram seals which prevent any stem leakage.

(2) Although these drain valves are packless globe valves and, therefore stem leakage is not credible, they are process drains which are routed to the applicable drain system. For these valves, that leakage is credible and is specified to be no more than 10cc/hr/inch of nominal valve size. All other vents and drain lines are capped, plugged or block flanged and use packless metal diaphram valves and, therefore, are not considered credible leak paths.

Table 312.45-1
VALVES CONTAINING POST-LOCA RECIRCULATING FLUIDS (Sheet 2 of 3)

Valve (Size)	Normally Open/Closed	Type	Stem Diameter (in.)	Maximum Expected Leakage (cc/hr)
Corridor Adjacent to LPSI Pump #1				
8"-010-C-212	Locked Closed	Gate	1.25	12.5
HPSI Pump #3				
8"-011-C-212	Locked Closed	Gate	1.25	12.5
4"-014-C-075	Locked Closed	Gate	1.25	12.5
4"-016-C-355	Locked Open	Stop Check	1.5	15
2"-036-C-358	Locked Closed	Stop Check	1	10
Shutdown Hx #2				
8"-005-C-173	Locked Open	Gate	1.375	13.75
12"-002-C-173	Locked Closed	Gate	1.625	16.25
3/4"-019-C-376	Normally Closed	Packless Globe	N/A	7.5(2)
3/4"-020-C-376	Normally Closed	Packless Globe	N/A	7.5(2)
3/4"-021-C-376	Normally Closed	Packless Globe	N/A	7.5(2)
Shutdown Hx #1				
8"-003-C-173	Locked Open	Gate	1.375	13.75
12"-001-C-17	Locked Closed	Gate	1.625	16.25
3/4"-023-C-37	Normally Closed	Packless Globe	N/A	7.5(2)
3/4"-024-C-376	Normally Closed	Packless Globe	N/A	7.5(2)
3/4"-022-C-376	Normally Closed	Packless Globe	N/A	7.5(2)
Penetration Area				
3/4"-025-C-376	Normally Closed	Packless Globe	N/A	-
2HV-9367 (8")	Normally Closed	Gate	1.5	15
4"-008-C-174	Locked Closed	Gate	1	10
3/4"-064-C-334	Normally Closed	Packless Globe	N/A	-
2HV-9420 (3")	Normally Closed	Globe	1	10
2HV-9434 (3")	Normally Closed	Globe	1	10
3/4"-045-C-334	Normally Closed	Packless Globe	N/A	-
3/4"-044-C-334	Normally Closed	Packless Globe	N/A	-
2HV-9368 (8")	Normally Closed	Gate	1.5	15
3/4"-017-C-376	Normally Closed	Packless Globe	N/A	-
2HV-9329 (2.0")	Normally Closed	Globe	1	10
2HV-9330 (2.0")	Normally Closed	Globe	1	10

Table 312.45-1
VALVES CONTAINING POST-LOCA RECIRCULATING FLUIDS (Sheet 3 of 3)

Valve (Size)	Normally Open/Closed	Type	Stem Diameter (in.)	Maximum Expected Leakage (cc/hr)
Penetration Area	(Con't)			
3"-155-C-551	N/A	Check	N/A	-
2"-005-C-334	Normally Closed	Packless Globe	N/A	-
2"-154-C-334	Normally Closed	Packless Globe	N/A	-
2HV-9326 (2.0")	Normally Closed	Globe	1	10
2HV-9327 (2.0")	Normally Closed	Globe	1	10
2HV-9324 (2.0")	Normally Closed	Globe	1	10
2HV-9323 (2.0")	Normally Closed	Globe	1	10
2HV-9333 (2.0")	Normally Closed	Globe	1	10
2HV-9332 (2.0")	Normally Closed	Globe	1	10
3/4"-065-C-334	Normally Closed	Packless Globe	N/A	-
4"-009-C-174	Locked Closed	Gate	1	10
Pipe Chase				
4" [illegible]C-553	N/A	Check	N/A	-
HPSI Pump #1				
8"-007-C-212	Locked Open	Gate	1.25	12.5
10"-006-C-675	N/A	Check	N/A	-
			Total	540
		Maximum Expected Leakage		

Question 312.46

Our review of your response to Q312.42 concludes that you have not shown that the explosion risks associated with transportation of hazardous materials past the site are sufficiently low to be acceptable. Therefore, it is our position that you should consider some mitigative measures which would provide a demonstrable and significant reduction of the explosion risk. For example, we believe the following considerations should be evaluated for their effectiveness in risk reduction:

a. Moving the rairoad switch, which is currently situated near SONGS Unit 2, outside the exclusion boundary and well to the south of it.

b. Continuous and visual monitoring of the I-5 highway and ATSF railway within the exclusion boundary. Timely detection of traffic accidents or other hazardous events, followed by an appropriate emergency response, should be considered. A contingency plan, and accident response capability (e.g., fire fighting personnel and equipment, traffic control under accident conditions) should be developed.

c. The ATSF railway should be monitored periodically and necessary corrective steps implemented whenever track conditions are found to be defective or degraded.

d. The effectiveness of a barrier between the ATSF railway and the plant should be considered with respect to heavier than air vapor diversion, overpressure intensity reduction, and minimizing the potential for derailed cars approaching the plant structures.

Alternatively, you may wish to consider other possible mitigative steps beside the above suggested items. Upon receipt of this type of information we will review it and evaluate its potential for risk reduction.

Response

The response to question 312.46 will be provided in an FSAR amendement by January 1981.

References

None.

Question 312.47

With respect to your analysis of toxic gas hazards from transportation accidents, we are unable to verify the motor carrier accident rate which is presented in Section 6.4 of the FSAR. The value of 2×10^{-10} accidents per mile used in Section 6.4 is about four orders of magnitude less than the truck accident rate based on nationally averaged statistics used in FSAR Section 2.2 analyses. Thus, the estimated need for control room operator protection may have to extend beyond the selected gases (chlorine, butane, and anhydrous ammonia). Our position is that you should substantiate the truck accident rate used in the toxic gas analysis or revise it accordingly.

Response

The response to question 312.47 will be provided in an FSAR amendment by January 1981.

Reference

None.

Question 321.10

Primary Coolant Sources Outside Containment
Action Plan III.D.1.1

We have reviewed your response to Section 21.6a of NUREG 0578. In addition to what you plan to implement prior to the issuance of full power operating license, you should complete to following requirement at this time:

a) Provide a description of the practical leak reduction measures you will implement immediately to reduce leakages from all systems outside the containment that could carry radioactive fluids. Your description should include measuring values of actual leakage rates with the systems in operation and a summary report of your test results.

b) Provide a description of the continuing leak reduction program you propose to establish and implement. This description should include the preventive maintenance program to reduce leakages to as-low-as practical limits, the leak rate test method and summary of procedures for each system or subsystem, the test frequencies and acceptance criteria. You should include the steps you will take to minimize occupational radiation exposures and assure system completeness. Your description should also specify the staffing and training requirements and the quality assurance aspects of your program. For further information, see NUREG 0694.

Response

a) In order to help minimize the potential leakage of primary coolant sources outside containment, SCE is currently in the process of establishing a leak reduction program, which will be fully implemented prior to full power operation. The program will include the following items:

1. Component Identification

(a) The program will address pump seals, valves and where applicable, flanged connections in all systems that are expected to be used to carry fluid with a potentially high source term in the post-accident mode outside the containment. A preliminary list of program components is shown in table 321.10-1.
(b) Detailed justification is provided for excluding any systems that are normally used but whose use will be prohibited in the post accident mode (e.g., letdown system.) A summary of this evaluation is provided in Response to NRC Action Plan NUREG 0660, part II.B.2.
(c) A complete component list, including locations, will be prepared.
(d) The locations where leakages from each component are collected will be defined.

2. Initial Program

(a) Baseline leakage rates will be established for each component identified in Part 1. Prior to establishing these leak rates, undesirable leakages will be minimized by tightening flanges, re-packing valves, etc.
(b) The baseline leakage rates for each component in Part 1 will be documented.
(c) The measurements of leakages will be performed while each system is under operating conditions. Where justifiable, an integrated leakage may be used, but a component-level measurement is preferred.
(d) Specific methods will be identified for measuring the leakages from each of the Part 1 components. In particular, for gaseous systems such as the containment air emergency sample system, methods for performing Helium leak testing or its equivalent will be identified.
(e) Results of the above baseline measurement program will be submitted to NRC prior to full power operation.
(f) A procedural requirement to verify acceptable component leakages using the baseline data above on a periodic basis will be incorporated into the Plant Procedures. Portions of the leakage verification may be included in the procedures for implementing Section XI testing of pumps and valves. Longer intervals than those initially specified may be justifiable in many cases where leakage trends show no appreciable changes. Methods used can include:
  (1) Measurement of pump seal leakage during the periodic pump performance testing.
  (2) Measurement of valve stem leakage during the Section XI testing. In addition, indications from this type of leakage as well as flange leakage can be visually inspected during operator walk down.
(g) Maintenance procedures will be reviewed to insure proper component testing is conducted to minimuze leakage after a Part 1 component is restored to operability.

b) A continuing leak reduction program will also be established to keep potential leakage as low as practical. The following steps will be considered and incorporated where applicable:

1. Existing procedures for maintenance inspection and repair of valves, pumps, etc. should be reviewed to identify those steps where an increased level of inspection might minimize potential leakages. This may include reduced intervals for valve packing change-out, periodic tightening of nuts, and re-evaluation of acceptance criteria.

2. Techniques should be described for reducing any potential exposure to personnel during access or maintenance of Part 1 equipment.

3. Procedures for control of radioactive sump levels should be evaluated including when and where leakage fluid will be pumped or stored.

4. Any staffing, quality control or training changes that may be necessary to implement the initial and long term leak reduction programs should be identified.

Reference

FSAR Section 15.6.
No FSAR change was made.

Question 321.11

Post Accident Sampling
Action Plan II.B.3

We have reviewed your response to Section 2.1.8a of NUREG-0578. In addition we need the following information:

a) Submit a descriptive summary of the interim provisions and procedures for sampling and analyzing the reactor coolant and containment atmosphere. Your summary should include the interim modifications, you will need to conduct the physical, chemical and radiological analysis steps.

b) Provide a description of the final system design of the sample handling and counting facilities. Your final system description should include addition of new sampling station equipment and/or final modifications to existing sample handling and counting facilities to achieve analysis within the time specified in Item 2.1.8a given in the November 9, 1979 letter.

For further information, see NUREG 0694

Response

a) Interim provisions and procedures for sampling and analyzing the reactor coolant and containment atmosphere will not be required since the in-line post accident sampling system described in our Response to NUREG 0660 part II.B.3 will be operational prior to exceeding 5% power.

b) A description of the final system design for the Post-Accident Sampling System (PASS) is provided in subsection 9.3.6.

Reference

Revised FSAR subsection 9.3.6, Post-Accident Sampling System.

Question 321.14

Will the area radiation monitors that you propose to install to monitor steam dump/safety valve releases provide a dose rate range equivalent to Xe-133 equivalent concentration range of 10^{-1} to 10^{3} uCi/cc in the discharge? How will you correct the readings of these external monitors for low energy gammas? Describe the procedures and calculational methods you will employ to convert the dose rate to concentrations and release rates.

Response

The response to Question 321.14 will be provided in an FSAR amendment by January 1981.

Reference

FSAR section 11.5.

Question 321.16

Indicate how you will correct instrument readings for background effects when applicable.

Response

The response to Question 321.16 will be provided in an FSAR amendment by January 1981.

Reference

FSAR section 11.5.

Question 321.21 (III.D.1.1)

We require that you leak test in the immediate future (a) containment spray and safety injection systems which you have recognized may contain highly radioactive fluids following a postulated accident (b) post-accident reactor coolant and containment air sample lines (containment air return sample line up to stop valve that will be added to the waste gas header), and (c) other applicable systems that are unique to San Onofre, Unit Nos. 2 and 3. You should provide a summary description, together with the initial leak test results at least 4 months prior to issuance of full power operating license.

Response

Leak tests will be conducted on components of the containment spray and safety injection systems which may contain highly radioactive fluids post accident. Leakages from the post-accident sample lines and other applicable systems will be included in the test. An outline of the initial leakage test is included in the leak reduction program provided in the response to question 321.10. A summary of the initial test results will be provided to NRC 4 months prior to operation at full power.

Reference

Response to NRC Question 321.10. No FSAR change was made.

Question 321.22

You should leak test the CVCS and waste gas systems since they may get contaminated with highly radioactive fluids prior to their isolation and/or may be used during the accident.

Response

The response to Question 321.22 will be provided in an FSAR amendment by January 1981.

Reference

None

Question 321.23 (III.D.1.1)

Provide the details of immediate leak reduction measures you plan to implement.

Response

The details of the immediate leak reduction measures to be implemented at San Onofre Nuclear Generating Station Units 2&3 are provided in the response to question 321.10.

Reference

Response to NRC Question 321.10. No FSAR change was made.

Question 321.24 (III.D.1.1)

The statement in your August 1980 TMI response that you are evaluating leakage of systems located outside the containment to determine whether a leak reduction program is necessary is unsatisfactory. Provide information on the continuing leak reduction program you are required to implement. This information should include (a) frequency of the integrated leak tests, (b) method and summary of procedures for testing each system or subsystem, (c) steps that you will take for minimizing occupational exposures, and (d) details on the preventive maintenance steps to reduce leakage to as-low-as practical levels.

Response

The description of the continuing leak reduction program to be implemented at San Onofre Nuclear Generating Station Units 2&3 is provided in the response to question 321.10, and addresses the four items in this question.

Reference

Response to NRC Question 321.10. No FSAR change was made.

Question 321.25

Provide assurance that reactor coolant and containment atmosphere sampling during post-accident situations will not require an isolated auxiliary system to be placed in operation in order to use the sampling system.

Response

The sample inlets to the post-accident sampling system come directly off the normal sample lines for reactor coolant and containment atmosphere sampling. These normal sample lines exit directly off the reactor coolant loop or out of the containment atmosphere respectively. No auxiliary systems, such as the letdown system, need be nonisolated in order to obtain representative coolant or atmospheric samples.

Reference

FSAR Section 9.3. No FSAR changes were made.

Question 321.26

Clarify what you mean by the statement that you have included provisions to measure total dissolved gas concentrations up to approximately 2,000 cc/KG.

Response

As a result of the possible generation of H_2 within the core during an accident situation, there is a possiblility that H_2, and therefore total gas, concentrations may approach 2000 cc/Kg. The post accident sampling system is designed to isolate and measure remotely reactor coolant total gas concentrations up to 2000 cc/Kg in order to more realistically determine the extent of the zirc hydriding reaction. This is accomplished by measuring the level change in a burette upon depressurization of a pressurized sample.

Reference

FSAR Section 9.3.
No FSAR changes were made.

Question 321.27

Describe the sample room exhaust filters referred to in your August 1980 TMI submittal. Your description should include filter efficiencies for all forms of gaseous iodine and particulates.

Response

The post accident sampling system includes a charcoal filter which has an efficiency of 99.99% (DF=10^4) for all forms of iodine. The charcoal used is of the activated, impregnated type. A HEPA filter is provided downstream of the PASS charcoal filter. This HEPA filter has an efficiency of 99.97% on 0.3 micron particles.

Reference

FSAR Section 9.3.
No FSAR changes are made.

Question 321.28

Provide assurance that backup sampling through grab sampling will be provided for systems using in-line monitoring for samples. Give the frequency of such grab sampling.

Response

Backup grab sampling capability exists in the post accident sampling system (PASS) for boron concentration and coolant and containment atmospheric activity levels using a diluted fluid sample. Total gas and coolant hydrogen are measured using separate instruments, one a hydrogen analyzer and the second a level instrument. The level instrument indicates the volume of gas coming out of solution from a pressurized coolant sample upon depressurization. Grab sampling is available on an as-needed basis. Provisions (i.e., valves and quick disconnect couplings) for additional undiluted grab sampling have also been incorporated in the PASS.

Reference

FSAR Section 9.3.
No FSAR changes were made.

Question 423.31

It has come to our attention that some applicants do not intend to conduct confirmatory tests of some distribution systems and transformers supplying power to vital buses as required by Position 3 of Regulatory Guide 1.68, and more specifically by Part 4 of the staff position on degraded grid voltage (applied to all plants in licensing review by the Power Systems Branch since 1976). Part 4 of the degraded grid voltage position states as follows:

"4. The voltage levels at the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Provide a description of the method for making this verification; before initial reactor power operation, provide the documentation required to establish that this verification has been accomplished."

Your test description in FSAR Chapter 14 does not contain sufficient detail for us to determine if you intend to conduct such a test. It is our position that confirmatory tests of all vital buses must be conducted including all sources of power supplies to the buses. Modify your test description to indicate that this testing will be conducted in accordance with Regulatory Guide 1.68 and the above cited position.

Response

Voltage levels at the safety-related buses have been optimized for the full load and minimum load conditions by the use of calculations which assume the most adverse cases for the expected ranges of load conditions and anticipated deviations of the offsite power source. The worst case condition assumes a load contribution from the opposite unit.

The anticipated minimum and maximum voltages of the offsite power system will be established on one of the 220Kv buses in the San Onofre switchyard. Then, using the same line up of safety related equipment used to perform the load sequencing tests on the diesel generators, the safety-related loads will be sequenced through the various possible supplies to the safety-related buses.

The test will be run using only Unit 2 loads, since loads from Unit 3 will not be available. Proof of the adequacy of the design of the power distribution system at the maximum load condition will be by comparison of test results with predictions for the tested condition.

Reference

See response to NRC Question 040.48 and revised FSAR paragraph 14.2.12.72.U.

Question 432.19

It is not clear whether the staffing level on p. 5-1 is for one unit or for all three units. Please clarify this and justify any differences between your staffing level and the requirements in Table B-1 of NUREG 0654.

Response

A revised table indicating minimum shift crew for Units 2 and 3 is as follows:

One (1)	Watch Engineer
One (1)	Shift Technical Advisor
Two (2)	Operating Foremen
Three (3)	Control Room Operators
Three (3)	Assistant Control Room Operators
Six (6)	Plant Equipment Operators
One (1)	Health Physics Technician
*One (1)	Radiation-Chemical Technician

These staffing levels are consistent with Table B-1 of NUREG 0654.

Reference

None.

*Shared between Unit No. 1 and Units 2 and 3.

Question 432.23

Identify radiological laboratories and their capabilities and expected response times.

Response

The response to NRC Question 432.23 will be provided in an FSAR amendment by January 1981.

Reference

Emergency Plan. No FSAR change was made.

Question 432.35

Indicate which natural phenomena monitors listed in table 7-3 are to be placed offsite.

Response

All of the natural phenomena monitors listed in table 7-3 are to be placed onsite.

Reference

None.

Question 432.39

Identify plant system and effluent parameter values characteristic of a spectrum of off-normal conditions and all the example initiating conditions in NUREG 0610.

Response

The response to question 432.39 will be provided in an FSAR amendement by February 1981.

Reference

Emergency plan.

Question 432.41

Provide methods and techniques to determine the source term of release of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.

Response

The response to question 432.41 will be provided in an FSAR amendment by February 1981.

Reference

Emergency plan.

Question 432.42

Establish the relationship between effluent monitor readings and on-site and offsite exposures and contamination for various meteorological conditions.

Response

The response to question 432.42 will be provided in an FSAR amendment by February 1981.

Reference

Emergency plan.

Question 432.43

(See H.8.) Also, there shall be provisions for access to meteorological information by emergency response centers.

Response

The meteorological data from the site monitoring station is continuously displayed in the control room area. Wind direction, wind speed, and stability parameters will be communicated to the technical support center (TSC) when requested by telephone. An individual in the technical support center will be assigned the duty of maintaining the current status of meteorological conditions and communicating this and forecasted data to the primary emergency operations center (interim, EOE) by telephone under the condition of a declared "general emergency." State personnel will be present at this location.

NRC personnel in the TSC will be responsible for communicating meteorological data to offsite NRC centers by telephone.

Reference

None.

Question 432.44

Establish the methodology for determining the release rate/projected doses if the instrumentation for such assessment are offscale or inoperable.

Response

The response to Question 432.44 will be provided in an FSAR amendment by February 1981.

Reference

Emergency plan.

Question 432.45

Provide the sensitivities of your air samples.

Response

Offsite air samples will be obtained utilizing portable air samplers. These samples will be analyzed by onsite equipment to measure activity levels. The lower limit of detection (LLD) for this equipment for I-131 is 1×10^{-12} $\mu c/ml$, for noble gases is 1×10^{-6} $\mu c/ml$, and for particulates is 1×10^{-11} $\mu c/ml$. Similar equiment is also available at Unit 1 for analyses.

Reference

None.

Question 432.49

Provide decontamination capability for evacuated personnel.

Response

The response to NRC Question 432.49 will be provided in an FSAR amendment by January 1981.

Reference

Emergency Plan. No FSAR change was made.

Question 432.53

Provide an onsite radiation protection program to be implemented during emergencies. It shall identify indiviudals, by position or title, who can authorize emergency workers to receive doses in excess of 10 CFR 20 limits.

Response

A radiation protection program for San Onofre Units 2 and 3 is under development and will be in effect by the time of the first delivery of new fuel. This program will meet the requirements of 10 CFR 20 and the criteria of EPA 520/1 - 75/001 and Draft ANSI 13.12.

The supervisor of chemistry and radiation protection can authorize emergency workers to receive doses in excess of 10 CFR 20 limits. In the absence of the supervisor of chemistry and radiation protection, the watch engineer or plant manager shall authorize emergency workers to receive doses in excess of 10 CFR 20 limits. Guidelines utilized by the above individuals shall include but not necessarily be limited to:

1. Emergency personnel should be volunteers.

2. Emergency personnel shall be familiar with the consequences of exposures.

3. Women capable of reproduction will not take part in these actions.

4. Other considerations being equal, volunteers above the age of 45 shall be selected.

5. Internal exposure shall be minimized by the use of the best available respiratory protection, and contamination shall be controlled by the use of available protective clothing.

Emergency procedures will be prepared to have three clearly distinguished objectives. The first is to restrict exposures to ALARA; the second is to bring the situation back under control; and the third is to obtain information for assessing the causes and consequences of the event.

Reference

None.

Question 432.54

Specify the criteria for determining the need for personnel decontamination.

Response

Personnel will be decontaminated when contamination levels are equal to or exceed the following limits:

Beta, Gamma	$-220\ dpm/cm^2$
Alpha	-1/10 of Beta, Gamma limits.

Reference

None.

Question 432.55

Provide your criteria for permitting return of areas and items to normal use after their contamination (Expand Section 9.1).

Response

Equipment and areas will be permitted to be used in a normal manner when contamination levels are less than the following limits:

Beta, Gamma	-220 dpm/100 cm^2
Alpha	-1/10 of Beta, Gamma limits.

Reference

None.

Question 432.58

Provide an index which covers any State and local plans, and a cross reference between your plan and each criteria in NUREG 0654.

Response

The resonse to NRC Question 432.58 will be provided in an FSAR amendment by January 1981.

Reference

Emergency Plan. No FSAR change was made.