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CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

FIRST SUPPLEMENT TO: PRELIMINARY SAFETY ANALYSIS REPORT

March 31, 1966

INTRODUCTION

This exhibit contains answers to the questions raised by the AEC Regulatory Staff in its letter to Applicant dated February 28, 1966.

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

ADDITIONAL INFORMATION REQUIRED

1. A list of signifiant differences between the proposed Indian Point Unit No. 2 facility design and the Brookwood and Connecticut-Yankee plants was supplied in Section 2 of the Preliminary Safety Analysis Report. Please provide the purpose, justification and a critical evaluation of the effects of each change on reactor safety.

2. The Commission published on November 22, 1965, "General Design Criteria for Nuclear Power Plant Construction Permits". Please provide a discussion, together with data and analysis, sufficient to show how each of the criteria applicable to your facility will be fulfilled.

3. Your attention is directed to a letter from the ACRS to the Chairman, AEC, dated November 24, 1965, concerning reactor pressure vessels. Please discuss the consideration which has been given in the design of your facility to the recommendations contained in numbered paragraphs 1 and 2 of the ACRS letter. For your guidance in providing a complete answer to this question the following are some of the detailed areas of concern explored recently by an ACRS subcommittee on the proposed Rochester Gas and Electric Corporation Brookwood facility. Your reply should incorporate answers to these questions.

   a. Please give details on the best prediction of maximum fast neutron flux dose in the pressure vessel, including uncertianties in prediction.

   b. Please give details on the method of measuring NDT for base plate and quantitatively describe uncertainties therein.

   c. Please give details on method of prediction of NDT shift with fast neutron dose and quantitatively describe uncertainties therein, including considerations of weld regions and heat affected zones.

   d. Please describe in detail the stress considerations to be allowable below NDT plus 60 and below NDT. State assumptions and give reasons, allowing for flaw size in particular.

   e. Please give rationale for relationship between NDT and allowed stress emphasizing in particular the degree of conservatism which it is felt the circumstances require, and why.

   f. Define the flaw size and type in the pressure vessel which is accepted in the specifications. What flaws larger in size or of special significance might not be detected, particularly in zones of irregular geometry?

g. What flaw size is accepted in the studs of the pressure vessel? What frequencies of stud inspection or replacement is planned? How many studs can fail without threatening the integrity of the closure?

h. Please describe requirements concerning the support structure for the pressure vessel, including the degree of levelness over reactor life, which are needed to insure no problems due to local overstressing of the pressure vessel.

i. Describe how small leaks in the pressure vessel would be detected and the action to be taken, should such occur. How is adequate response assured in the event of a previous existence of small leaks in other parts of the system?

j. Describe the surveillance program for the pressure vessel in some detail. What uncertainties would be expected from the experimental results?

k. Are you considering procedures for detecting the propagation of cracks within the pressure vessel wall, i.e., acoustic emission?

l. State and justify the energy required to initiate failure of the primary system boundary. Can a control cluster ejection or any other credible mechanism provide this amount of energy by reactivity insertion?

4. The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

   a. Relate the available energy sources by showing the total energy that could be provided by (1) the primary coolant, (2) a 100% metal-water reaction, (3) the hydrogen-air reaction, and (4) the core decay heat at 10, 20, and 30 minutes. The relative energy sources should be provided on a percentage basis that totals 100% for each case. What is the total energy available from secondary sources (e.g., steam generators)?

   b. Plot in graphical form up through one hour for your assumed model of post-accident conditions, (1) the ratio of decay heat energy in the containment atmosphere to primary coolant energy, (2) the ratio of metal-water energy in the containment atmosphere to primary coolant energy, (3) the ratio of H_2 recombination energy in the containment atmosphere to the primary coolant energy, (4) The ratio of total energy in the containment atmosphere to total available energy, and (5) the ratio of total energy in heat sinks to total available energy.

c. Indicate in graphical form the per cent of zirconium in the core available for reaction by providing a family of curves indicating the per cent of core clad at or above given temperatures and the zirconium assumed reacted, as a function of time with (1) no safety injection, and (2) full safety injection.

d. Provide the containment pressure transient curves following the MCA, assuming no further energy is added to the containment after the initial blowdown, for the cases wherein (1) all engineered safeguards function and (2) no engineered safeguards function and the containment and structures act as the only heat sink. Also show the per cent of total primary system energy lost as a function oftime.

e. Provide the information in part (d) showing the increase in containment pressure resulting by (1) adding additional energy by the mechanism of steam generation equal to 50% and 100% of the original primary coolant energy, linearly with time in 1000 seconds and (2) adding additional energy stepwise equivalent to 20% of the primary coolant energy at 500 and 1000 seconds by superheating the atmosphere. Also show the pound moles of air, steam, and hydrogen in the containment as a function of time.

f. Provide the containment pressure trnasient curves following the MCA for the cases in which the only engineered safeguards assumed to function are: (1) one high head and one low head safety injection pump, and (2) one containment spray pump, and (3) four containment air recirculation coolers.

5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

a. Summarize the peak heat flux factor (F_q), peak enthalpy rise factor ($F_{\Delta H}$), and the peak axial flux factor (F_z) for the following situations:

1) Nominal conditions for worst time in core life (using worst expected rod conditions).
2) Design conditions for worst time in core life (no engineering factors).
3) Hot channel conditions for worst time in life (with engineering factors).

b. Supply a distribution curve showing the fraction of the core operating above various power levels with their corresponding DNB ratios for condition (a-1).

c. For condition (a-2), provide the total number of fuel rods that are within 90% of the design peak power level and the corresponding DNB ratios (include the effects of instrument errors).

d. Repeat part (b) and (c) for the overpower condition. If any channel has bulk boiling, or would require less than 5% additional power to cause boiling, tabulate these results indicating at what distance from the core top boiling ensues.

e. For a hypothetical 125% overpower condition, estimate whether any fuel rods approach design limits (e.g. DNB or center fuel melting).

f. For the hot channel positions, provide the DNB, exit quality, and center fuel temperature at 100%, and 125% for the worst design conditions. In addition, arbitrarily raise the $F_{\Delta H}$ and F_z factors by 10% and tabulate as above for each condition.

g. For the engineering hot channel factors, indicate the 2 and 3 values of the various statistical components before they are combined into one overall factor.

6. Provide a diagram of your conceptual layout of the internal air recirculation and iodine filtration systems showing the relative location of the input and exhaust ducts, fans, heating and cooling units, demisters, anc charcoal filters. State and justify the estimated temperature and relative humidity of the containment atmosphere at each of the above locations for the anticipated conditions following the double-ended rupture of a primary coolant pipe. Describe the systems (including redundancy) provided to prevent ignition of the charcoal filters, and discuss the potential effects on containment pressure and off-site doses if total ignition of the filters is assumed. What experimental evidence can be given to justify the elemental and organic iodine removal efficiencies assumed in the PSAR? What is the basis for the selection of the fraction of organic iodine initially present, and for its growth rate throughout the accident? Also, what fraction of the total gaseous activity is assumed to be present in the fuel gaps?

7. The containment penetration pressurization system has been designed to limit leakage fromthe containment under accident conditions and to provide continuous verification of containment integrity during normal operation. To obtain a better understanding of how these goals will be achieved, provide a description of system operation under normal, abnormal, and accident conditions. Discuss the capacity ofthe gas supply systems, the sensitivity of the leakage monitors, and analyze system operation with various component failures. Discuss the magnitude and potential effects on containment pressure of inleakage from this system that can be tolerated during normal operation.

8. The containment spray system is provided as an independent backup to the air recirculation and iodine filtration system. Discuss the experimental basis for the design of the containment spray system and indicate how the pressure reduction and iodine removal values were derived. What is the flow rate of the system under normal and emergency power availability? Discuss the redundancy provided by this system.

9. The operation of some engineered safeguards systems will require that large quantities of radioactive liquid be pumped outside the containment under accident conditions thereby extending the effective containment boundary. Estimate the amount of leakage of radioactivity (liquid and gaseous) from lines, valves, pumps, etc. outside of the containment under accident conditions and discuss how leakage will be controlled to limit potential off-site doses under accident conditions. State and justify the maximum leakage that can be tolerated from these systems before off-site doses exceed Part 100 values. In consideration of the importance of achieving low leakage of radioactive materials from your facility under accident conditions, discuss the advisability of installing some or all of this equipment inside the containment vessel.

10. Discuss the operation of the emergency diesel power supply system under accident conditions with no normal power sources available. Indicate how the proper equipment is selected for operation (assume failure of one bus or diesel) and how unnecessary loads are dropped to prevent overloading and possible tripping of the remaining diesels.

11. Provide preliminary accident evaluations to support the results reported in the Preliminary Safety Analysis Report for the startup accident, steam line rupture, refueling, and control rod cluster ejection accident. For example, show such parameters as core reactivity, core temperature, and system pressures plotted against time for the worst condition during core lifetime. Consider the possible generation of curves that relate minimum reactor period to (a) integrated excursion energy and (b) average fuel temperature. These curves should consider cases of hypothetical reactivity insertions considerably greater than that resulting from the ejection of a single control cluster. For each accident, state the potential off-site doses.

12. The borated safety injection water may be diluted by the non-borated primary coolant or secondary system water following a major pipe failure. Analyze the consequences of adding diluted safety injection water to the reactor assuming several dilution factors, and provide the corresponding periods and energy releases if the control rods are and are not assumed to be inserted.

13. The steam generators provide the primary mechanism for dissipation of primary system heat in the event of complete loss of power. Indicate the water sources and capacity available to the steam generators under these conditions. Discuss how this water can be delivered, and how long the reactor can be safely cooled by these sources.

14. Provide the following information regarding the proposed instrumentation system:

    a. Discuss the instrumentation provided to prevent low water levels in the steam generators. Is this instrumentation redundant?

    b. Discuss how the cluster control system has been designed so that rod insertion time is not delayed as a result of pressure gradients generated by potential blowdown forces.

c. Discuss how the position of critical isolation valves will be indicated in the control room.

d. A rupture of the tap feeding two of the three pressurizer-low-level channels can remove the intended automatic protection provided by this circuit. Please justify this proposed design.

e. Provide a list of all monitors that will be provided to indicate the reactivity status of the reactor, and the pressure, temperature and humidity conditions inside the containment after the MCA. Discuss the design lifetime of the critical components associated with this equipment when operated in the post-MCA containment environment.

f. Provide experimental evidence to indicate the sensitivity of the external ion chambers to changes in the axial and radial flux distribution. Relate this information to internal monitor readings, if possible.

15. Provide the distance and location of the Chelsea intake for New York City and discuss the possibility of transport of activity from the plant to this point.

16. The inversion frequency assumed for the 30 day meteorology does not appear to be conservative since it is near the average value for two years. Please justify the selection of this value.

17. Discuss the relation to nuclear safety of any system or equipment at the Indian Point site that will be shared by both reactors. Provide the infinite and maximum 8 hour thyroid and whole body doses for both control rooms following the potential MCA at either facility.

18. Provide the anticipated pressure-flow characteristics for the safety injection and the charging pumps.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

a. For missile protection of the containment and other engineered safeguards during an assumed instantaneous rupture of the largest pipe of the primary system.

b. For pipe motion resulting from an assumed instantaneous rupture of the largest pipe of the primary system.

c. For steam generators with respect to tube or tube sheet failure due to rupture of either the primary or secondary piping.

d. For the maximum permissible primary and secondary coolant activity during unrestricted power operation.

e. For containment vessel penetrations. Provide a list of all penetrations and the type of isolation planned.

f. For the radioactive gas hold-up tank. What is the maximum radioactive fission product inventory that can be stored in the gas storage tanks? Describe the environmental consequences of a storage tank leak. Describe provisions to monitor gaseous releases for iodine.

g. For the liquid waste collection tanks considering necessary capacity during the accidental release.

h. For the fuel hold-down fixtures considering the uplift forces during a major loss of coolant accident.

i. To limit core drop if the upper support fails. Explain how the "in-core" instrumentation structure will be designed to limit the core drop. What are the consequences of the maximum potential reactivity insertion under these conditions?

j. For the operational reliability of reactor safety, containment isolation, and engineered safeguards systems. Important equipment such as sensors, valves, solenoids, breakers, switches, pumps, cooling water, injection water, etc. should be considered. The "fail safe" and/or redundant features or lack of some should be discussed and, where important to an evaluation of system adequacy, test provisions and the criteria relative to frequency of tests should be stated and justified.

Containment Design Information

1. In load criteria on p. 5-8 (Exh. B, Vol. 2, Part A), what values of parameters correspond to the values of T, TLT', TL'? What are the temperatures in the interior and at the steel locations?

2. How is the lateral force (shear) in the structure carried? Is there shear reinforcement? For the liner, elastic stability provisions and load capacity based on yield are noted on p. 5-9. How does the steel liner participate in carrying the shear and other loads? What anchorage means is contemplated?

3. How will the splicing of the large 14S and 18S bars be handled? A general sketch of the contemplated reinforcing bar patterns is desirable.

4. What special provisions or special studies will be made to insure the adequacy of the penetrations (large and small) in terms of retaining strength and ductility while preventing leakage. Details of the concept of reinforcement around penetrations are desirable.

5. A tabulation of sources of stress, along with the appropriate allowable stress (or permissible resisting load and load factor) values, would help clarify the design approach. Also, a discussion of allowable ductility and provisions for obtaining same, is desirable.

6. What magnitude of vertical acceleration in earthquakes will be considered?

7. In citation (ii) it is noted that "...any vertical acceleration would be counteracted by the weight of the building". This statement is not correct. Also, vertical seismic motion should be assumed to act simultaneously with horizontal excitation. A more scientifically valid criterion for the earthquake analysis is required.

8. In the table of damping values given on p. 5-16, the damping factors for thy containment structure is shown as 7.0 per cent of critical and for the concrete support structure, item 2, 5 per cent. Similar values are shown in item 5. On what basis were these selected? Such large values correspond to rather heavily cracked concrete sections, stressed well into the yielding range. Lower values would be much more reasonable.

9. A description of the actual analysis techniques that will be employed in arriving at the design would be helpful. Only indirect statements about the procedures to be followed are given in the report on p. 5-16. What rigorous and acceptable procedures will be followed? How will the response spectra be employed in the procedures?

10. What criteria exist for adding the stresses arising from the different loadings, in contrast to combining loads? Since the loads act in different directions in many cases, a stress (or load resistance) combination approach would appear to be more rational. Discussion and comment is needed.

11. What wind loads will be assumed in the design?

1. A list of significant differences between the proposed Indian Point Unit No. 2 facility design and the Brookwood and Connecticut-Yankee plants was supplied in Section 2 of the Preliminary Safety Analysis Report. Please provide the purpose, justification and a critical evaluation of the effects of each change on reactor safety.

Answer

The comparison table of reactor design parameters for the Indian Point Unit #2 and the Brookwood and Connecticut-Yankee plants is attached with an evaluation of the differences as they relate to reactor safety. The parameters for the Brookwood facility are different from those presented in Chapter 12 of the Preliminary Safety Analysis Report and reflect revised design information presented in the First Supplement to the Brookwood application.

COMPARISON OF DESIGN PARAMETERS

	INDIAN POINT #2 PRELIMINARY REPORT	BROOKWOOD PRELIMINARY REPORT	CONNECTICUT YANKEE PRELIMINARY REPORT	REFERENCE LINE NO.
HYDRAULIC AND THERMAL DESIGN PARAMETERS				
Total Heat Output, MWt	2758	1300	1473	1
Total Heat Output, Btu/hr	9413×10^6	4437×10^6	5027×10^6	2
Heat Generated in Fuel, %	97.4	97.4	97.4	3
Maximum Overpower	12%	12%	18%	4
System Pressure, Nominal, psia	2250	2250	2065	5
System Pressure, Minimum Steady State, psia	2220	2220	2040	6
Hot Channel Factors				
Heat Flux, F_q	3.25	3.41	3.25	7
Enthalpy Rise, $F_{\Delta H}$	1.88	1.88	2.11	8
DNB Ratio at Nominal Conditions	1.81 (W-3)	1.90 (W-3)	2.00 (W-2)	9
Minimum DNBR for Designs Transients	1.30 (W-3)	1.30 (W-3)	1.25 (W-2)	10
Coolant Flow				
Total Flow Rate, lb/hr	136.2×10^6	67.1×10^6	93.6×10^6	11
Effective Flow Rate for Heat Transfer, lb/hr	124.1×10^6	61.1×10^6	84.2×10^6	12
Effective Flow Area for Heat Transfer, ft^2	48.4	25.1	40.5	13
Average Velocity Along Fuel Rods, ft/sec	16.1	15.1	12.1	14
Average Mass Velocity, lb/hr-ft^2	2.56×10^6	2.43×10^6	2.31×10^6	15
Coolant Temperatures, °F				
Nominal Inlet	543	556	546	16
Maximum Inlet Due to Instrumentation Error and Deadband, °F	547	560	550	17
Average Rise in Vessel, °F	53.0	49	41	18
Average Rise in Core	57.0	54	46	19
Average in Core	572.7	584.0	566.5	20
Average in Vessel	570.0	581.0	569.0	21
Nominal Outlet of Hot Channel	643.0	644.0	633.0	22
Average Film Coefficient, Btu/hr-ft^2-F	5900	5830	4750	23
Average Film Temperature Difference, °F	30.0	26.0	29.0	24
Heat Transfer at 100% Power				
Active Heat Transfer Surface Area, ft^2	52,200	28,500	36,400	25
Average Heat Flux, Btu/hr-ft^2	175,600	151,800	134,500	26
Maximum Heat Flux, Btu/hr-ft^2	570,800	517,500	437,200	27
Average Thermal Output, kw/ft	5.7	4.90	4.25	28
Maximum Thermal Output, kw/ft	18.5	16.7	13.8	29
Maximum Clad Surface Temperature at Nominal Pressure, °F	659	659	645	30
Fuel Central Temperature, °F				
Maximum at 100% Power	∿ 4200	3920	∿ 3900	31
Maximum at Overpower	∿ 4400	4150	∿ 4100	32
Thermal Output, kw/ft at Maximum Overpower	20.7	18.7	16.3	33

	INDIAN POINT #2 PRELIMINARY REPORT	BROOKWOOD PRELIMINARY REPORT	CONNECTICUT YANKEE PRELIMINARY REPORT	REFERENCE LINE NO.
CORE MECHANICAL DESIGN PARAMETERS				
Fuel Assemblies				
Design	RCC Canless 15 x 15	RCC Canless 14 x 14	RCC Can 15 x 15	34
Rod Pitch, in.	0.563	0.556	0.553	35
Overall Dimensions, In.	8.426 x 8.426	7.763 x 7.763	8.426 x 8.426	36
Fuel Weight (as UO_2), pounds	215,319	117,527	170,000	37
Total Weight, pounds	273,408	151,632	211,300	38
Number of Grids per Assembly	8	8	7	39
Fuel Rods				
Number	39,372	21,480	34,605	40
Outside Diameter, in.	0.422	0.422	0.4115	41
Diametral Gap, in.	0.0065	0.0065	0.0045	42
Clad Thickness, in.	0.0243	0.0243	0.016	43
Clad Material	Zircaloy	Zircaloy	Stainless Steel	44
Fuel Pellets				
Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	45
Density (% of Theoretical)	94-93	94-93	94	46
Diameter, in.	0.3669	0.3669	0.375	47
Length, in.	0.6000	0.6000	0.550	48
Rod Cluster Control Assemblies				
Neutron Absorber	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	49
Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	50
Clad Thickness, in.	0.019	0.019	0.019	51
Number of Clusters	53	32	45	52
Number of Control Rods per Cluster	20	16	16	53
Core Structure				
Core Barrel I.D./O.D., in.	148.5/152.5	109.0/112.5	131/134.6	54
Thermal Shield I.D./O.D., in.	158.5/164.0	114.5/122.5	138.5/146.5	55
PRELIMINARY NUCLEAR DESIGN DATA				
Structural Characteristics				
Fuel Weight (As UO_2), lbs.	215,319	117,527	170,000	56
Clad Weight, lbs.	43,785	24,208	41,300	57
Core Diameter, in. (Equivalent)	133.7	96.5	119	58
Core Height, in. (Active Fuel)	144	144	120	59
Reflector Thickness and Composition				
Top - Water plus Steel	10 in.	10 in.	10 in.	60
Bottom - Water plus Steel	10 in.	10 in.	10 in.	61
Side - Water plus Steel	15 in.	15 in.	15 in.	62
H_2O/U, Unit Cell (Cold Volume Ratio)	3.48	3.32	3.14	63
Number of Fuel Assemblies	193	120	157	64
UO_2 Rods per Assembly	204	179	225 or 209	65

	INDIAN POINT #2 PRELIMINARY REPORT	BROOKWOOD PRELIMINARY REPORT	CONNECTICUT YANKEE PRELIMINARY REPORT	REFERENCE LINE NO.
Performance Characteristics				
Loading Technique	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform	66
Fuel Discharge Burnup, MWD/MTU				
Average First Cycle	12,000	13,500	12,000	67
First Core Average	21,800	21,800	21,800	68
Feed Enrichments, w/o				
Region 1	2.23	2.35	3.4	69
Region 2	2.38	2.50	3.8	70
Region 3	2.68	2.80	4.2	71
Equilibrium	2.92	3.05	-	
Control Characteristics				
Effective Multiplication (Beginning of life)				
Cold, No Power, Clean	1.275	1.255	1.257	72
Hot, No Power, Clean	1.225	1.210	1.205	73
Hot, Full Power, Xe and Sm Equilibrium	1.170	1.160	1.155	74
Rod Cluster Control Assemblies				
Material	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	75
Number of RCC Assemblies	53	32	45	76
Number of Absorber Rods per RCC Assembly	20	16	16	77
Total Rod Worth	7%	∿7%	∿7-1/2%	78
Boron Concentrations				
To shut reactor down with no rods Inserted, clean (k_{eff} = .99) Cold/Hot	3400 ppm/3500 ppm	2200 ppm/2350 ppm	3210 ppm/3220 ppm	79
To control at power with no rods inserted, clean/equilibrium xenon and samarium	2800 ppm/2300 ppm	2030 ppm/1700 ppm	2800 ppm/2300 ppm	80
Boron worth, Hot	1% δk/k / 150 ppm	1% δk/k / 125 ppm	1% δk/k / 150 ppm	81
Boron worth, Cold	1% δk/k / 120 ppm	1% δk/k / 95 ppm	1% δk/k / 120 ppm	82
Kinetic Characteristics				
Moderator Temperature Coefficient	$+1.0 \times 10^{-4}$ to $-3.0 - \times 10^{-4}$ δk/k / °F	$+1\times10^{-4}$ to -3×10^{-4} δk/k / °F	0 to -2.4×10^{-4} δk/k / °F	83
Moderator Pressure Coefficient	-1.0×10^{-6} to $+3.0 \times 10^{-6}$ δk/k / psi	-1×10^{-6} to -3×10^{-6} δk/k / psi	0 to $+2.4\times10^{-6}$ δk/k / psi	84
Moderator Void Coefficient	$+1.0 \times 10^{-3}$ to -3×10^{-3} δk/k / % void	$+1\times10^{-3}$ to -3×10^{-3} δk/k / % void	0 to -2×10^{-3} δk/k / % void	85
Doppler Coefficient	$-1 \times$ to^{-5} to -2.0×10^{-5} δk/k / °F	-1×10^{-5} to -2×10^{-5} δk/k / °F	-0.5×10^{-5} to -2.1×10^{-5} δk/k / °F	86

	INDIAN POINT #2 PRELIMINARY REPORT	BROOKWOOD PRELIMINARY REPORT	CONNECTICUT YANKEE PRELIMINARY REPORT	REFERENCE LINE NO.
REACTOR COOLANT SYSTEM - CODE REQUIREMENTS				
Component				
Reactor Vessel	ASME III Class A	ASME III Class A	ASME VIII, 1270N, 1273N	87
Steam Generator				
Tube Side	ASME III Class A	ASME III Class A	ASME VIII, 1270N, 1273N	88
Shell Side	ASME III Class A	ASME III Class C	ASME VIII, 1270N, 1273N	89
Pressurizer	ASME III Class A	ASME III Class A	ASME VIII, 1270N, 1273N	90
Pressurizer Relief Tank	ASME III Class C	ASME III Class C	ASME VIII, Par. UW-2	91
Pressurizer Safety Valves	ASME III	ASME III	ASME I, Case 1271N	92
Reactor Coolant Piping	ASA B31.1	ASA B31.1	ASA B31.1	93
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM				
Reactor Heat Output, MWt	2758	1300	1473	94
Reactor Heat Output, Btu/hr	9412×10^6	4437×10^6	5027×10^6	95
Operating Pressure, psig	2235	2235	2050	96
Reactor Inlet Temperature	543	556	546	97
Reactor Outlet Temperature	596	605.4	587.0	98
Number of Loops	4	2	4	99
Design Pressure, psig	2485	2485	2285 & 2485	100
Design Temperature, °F	650	650	650	101
Hydrostatic Test Pressure (Cold), psig	3110	3110	3735	102
Coolant Volume, including pressurizer,cu.ft.	12,209	6238	8635	103
Total Reactor Flow, gpm	358,800	180,000	248,400	104
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL				
Material	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	105
Design Pressure, psig	2485	2485	2485	106
Design Temperature, °F	650	650	650	107
Operating Pressure, psig	2235	2235	2050	108
Inside Diameter of Shell, in.	173	132	154	109
Outside Diameter Across Nozzles, in.	245	220	234	110
Overall Height of Vessel & Enclosure Heat,ft-in.	42-4	39-0	41-6	111
Minimum Clad Thickness, in.	5/32	5/32	5/32	112

	INDIAN POINT #2 PRELIMINARY REPORT	BROOKWOOD PRELIMINARY REPORT	CONNECTICUT YANKEE PRELIMINARY REPORT	REFERENCE LINE NO.
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS				
Number of Units	4	2	4	113
Type	Vertical U-Tube with integral-moisture separator	Vertical U-tube with integral-moisture separator	Vertical U-tube with integral-moisture separator	114
Tube Material	Inconel	Inconel	Inconel	115
Shell Material	Carbon Steel	Carbon Steel	Carbon Steel	116
Tube Side Design Pressure, psig	2485	2485	2485	117
Tube Side Design Temperature, °F	650	650	650	118
Tube Side Design Flow, lb/hr	34.05×10^6	38.55×10^6	23.4×10^6	119
Shell Side Design Pressure, psig	1085	1085	985	120
Shell Side Design Temperature, °F	600	600	600	121
Operating Pressure, Tube Side, Nominal, psig	2235	2235	2050	122
Operating Pressure, Shell Side, Maximum, psig	1005	1005	910	123
Maximum Moisture at Outlet at Full Load, %	1/4	1/4	1/4	124
Hydrostatic Test Pressure, Tube Side (cold),psig	3110	3110	3735	125
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS				
Number of Units	4	2	4	126
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	127
Design Pressure, psig	2485	2485	2485	128
Design Temperature, °F	650	650	650	129
Operating Pressure, Nominal, psig	2235	2235	2050	130
Suction Temperature, °F	543	557	546	131
Design Capacity, gpm	89,700	90,000	62,100	132
Design Head, ft.	272	252	240	133
Hydrostatic Test Pressure (cold), psig	3110	3110	3735	134
Motor Type	A-C Induction single speed	A-C Induction single speed	A-C Induction single speed	135
Motor Rating	6000 HP	5500 HP	4000 HP	136
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING				
Material	Austenitic SS	Austenitic SS	Austenitic SS	137
Hot Leg - I.D., in.	29	29	27-1/2	138
Cold Leg - I.D., in.	27-1/2	27-1/2	27-1/2	139
Between Pump and Steam Generator - I.D. in.	31	31	29	140
Design Pressure	2485	2485	2285	141

LINE ITEM COMPARISON

INDIAN POINT #2 - BROOKWOOD - CONNECTICUT YANKEE

Line Item	Notes
1.	Nominal reactor power level higher. Power related to nuclear safety only in the ability to produce and remove the power in the core as designed.
2.	Directly related to Item 1 by conversion.
3.	No change in the fraction of the total heat generated in the core.
4.	See writeup #1.
5.	See writeup #2.
6.	See writeup #2.
7.	See writeup #3.
8.	See writeup #3.
9.	See writeup #3.
10.	See writeup #3.
11.	Total coolant flow rate is higher but rate is consistent with the heat removal requirements and other thermal hydraulic limits.

Line Item	Notes
12.	No change. Bypass flow of 9% is the same for all plants.
13.	No safety significance. Effective flow area for heat transfer determined by mechanical design of fuel assemblies and core.
14.	Velocity along fuel rods is determined by the total flow rate and the core mechanical design. The design value is higher than Connecticut Yankee and Brookwood but is only slightly higher than the Yankee plant which operates at 15.2 ft/sec.
15.	Mass velocity determined by flow area and flow rate. Related to reactor safety in that it must be within the range covered by the DNB correlation. (0.5×10^6 to 5.0×10^6 $lb/hr\text{-}ft^2$ for W-3 correlation)
16.	No significant change in core inlet temperature. Core inlet temperature related to reactor safety in that the core conditions produced must be within the design limits.
17.	No change in instrumentation error and deadband.
18. 19. 20. 21. 22.	No significant change in these core coolant temperatures. They are related to reactor safety in that the core conditions produced must be within the thermal-hydraulic design limits and in that the containment must be able to accommodate the loss of coolant accident for these coolant energy conditions.

Line Item	Notes
23.	The film coefficient is higher primarily because the mass velocity has been increased. The correlation used to determine the film coefficient is the same for all three plants.
24.	The average film temperature difference is determined by the film coefficient and the average heat flux.
25.	The active heat transfer area is higher and is related to reactor safety in that the heat to be transferred through this area must be considered in the fuel design and must be within the fuel capability.
26. 27. 28. 29.	The heat transfer parameters are determined by the required heat output, the heat transfer surface area and the design peaking factors for the core. They are related to clad integrity in that these conditions must be within the capability of the fuel and must also meet the thermal-hydraulic design criteria of DNB and fuel temperature. See writeup #4.
30.	The maximum clad surface temperature is the same as Brookwood and only slightly higher than Connecticut Yankee due to the increased nominal operating pressure.
31. 32.	The fuel central temperatures are higher than those for the other plants. The temperatures are well below the UO_2 melting temperature of 5000°F.
33.	The overpower linear power density is higher than for the other plants but is still well within the fuel capability. See writeup #4.

Line Item	Notes
34. 35. 36.	The fuel assembly design is not significantly changed with respect to type, rod pitch and overall dimensions.
37.	The total amount of fuel utilized has been increased. The ratio of fuel weight to power has been reduced due to use of Zircaloy clad instead of stainless steel and higher power density.
38.	The total weight of each fuel assembly includes the weight of the fuel, clad, grids, RCC guide tubes, and top and bottom nozzles. The total weight for the Connecticut Yankee plant also includes the weight of the fuel assembly can.
39.	The number of grids per assembly has been increased because of the longer core design. The additional grid provides approximately the same support spacing as in the shorter Connecticut Yankee fuel assembly.
40.	The total number of fuel rods is consistent with the fuel assembly design and number of fuel assemblies.
41.	There is no significant change in the fuel rod outside diameter.
42.	The diametral gap has been increased to allow for the greater differential expansion between the fuel and the Zircaloy clad.
43.	The clad thickness has increased so that the Zircaloy clad will meet the design ground rules for the stainless steel clad design. See writeup #4.

Line Item	Notes
44.	The cladding material has been changed to Zircaloy. See writeup #4.
45. 46. 47. 48.	The design of the fuel pellets has not substantially changed.
49. 50. 51. 52. 53.	The rod cluster control design is the same for all three plants. The number of RCC assemblies for each plant is determined based upon the control requirements.
54. 55.	The core barrel and thermal shield diameters are consistent with the large core diameter of the Indian Point #2 core.
56. 57.	See Line Items 37 and 38.
58.	The core equivalent diameter has been increased due to the use of more fuel assemblies. See writeup #5.
59.	The core length has been increased by 2 feet. See writeup #5.
60. 61. 62.	The calculated characteristics of the reflector have not changed.

Line Item	Notes
63.	The water to uranium ratio has been increased to reflect the optimization of the core design because of the use of Zircaloy clad.
64.	The number of fuel assemblies required has been increased consistent with core design and improved core performance.
65.	The number of fuel rods per assembly is consistent with the 20 rod RCC control cluster and the incorporation of bottom mounted in-core instrumentation so that any fuel assembly can be used with in-core instrumentation probes.
66.	The core loading procedures have not changed.
67. 68.	Thy average first cycle and first core burnups have not changed significantly.
69. 70. 71.	The core enrichment requirements have been reduced due to the use of Zircaloy clad and the accompanying improvement in neutron economy.
72. 73. 74.	The excess reactivity initially installed in the core has not changed significantly.
75. 76. 77.	Previously discussed in Line Item 49, 50 and 51.
78.	The total control rod worth has not changed significantly.

Line Item	Notes
79. 80.	The boron requirements for reactor shutdown and control have been reduced because of the increased boron worth (see items 81 and 82).
81. 82.	The boron in the coolant is worth more per ppm because of the increased water to uranium ratio so there is more weight of boron in the core for a given concentration.
83.	The moderator coefficient is now slightly positive at the beginning of life. This is related to the evaluation of the stability of the core. See writeup #5.
84.	The pressure coefficient has changed to correspond to the new moderator coefficient. The correlation of the coefficient has not changed from the other plants.
85. 86.	The moderator void and Doppler coefficient have not changed significantly.
87. 88. 89. 90. 91. 92. 93.	The change in code design requirements are discussed in writeup #6.
94. 95.	See Line Items 1 and 2.
96.	The nominal operating pressure has been increased. See writeup #2.

Line Item	Notes
97. 98.	The reactor coolant temperatures have been discussed under Line Items 16-22.
99.	The number of coolant loops is the same as for Connecticut Yankee. The total power removed in each loop is larger due to increase in pump capacity and improved heat transfer from the core. The total power per loop is approximately the same as for Brookwood.
100.	The reactor coolant system design pressure has not changed.
101.	The reactor coolant system design temperature has not changed.
102.	The hydro test pressure is consistent with the change in code requirements.
103.	The reactor coolant system volume has been increased compared to Brookwood due to the use of more loops and a more compact arrangement of components. The increase over Connecticut Yankee is due to the use of larger components.
104.	Previously discussed under Line Item 11.
105.	The reactor vessel material has not changed.
106.	No change in design pressure.
107.	No change in design temperature.
108.	Nominal operating pressure increase discussed under Line Item 96.

Line Item	Notes
109. 110. 111.	The physical dimensions of the reactor vessel have changed and are consistent with the larger core and new code requirements.
112.	The reactor vessel clad thickness has not changed.
113. 114. 115. 116. 117. 118.	The steam generator design bases have not changed. The number of generators is consistent with the number of coolant loops.
119.	The design flow rate is based on the appropriate fraction of the total flow rate which passes through each steam generator.
120.	The shell side design pressure has increased compared to Connecticut Yankee because of the increase in the maximum shell side operating pressure. This increase permits an improvement in the heat rate of the turbine plant.
121.	No change in the shell side design temperature.
122.	Operating pressure of the tube side has been increased because of the increase in the operating pressure of the reactor. See Writeup #2.
123.	The shell side maximum operating pressure has been increased to provide an improved heat rate for the turbine plant. The increase is not substantial and the margin to the design pressure is the same (∿85 psi) for the three plants.

Line Item	Notes
124.	No change in the maximum moisture at the outlet at full load.
125.	The hydro test pressure is consistent with the code requirements.
126. 127. 128. 129.	The type of reactor coolant pump (shaft seal) and the design conditions have not changed. The number of pumps is consistent with the number of reactor coolant loops.
130.	The nominal operating pressure has changed to correspond to the increased system operating pressure.
131.	There has been no significant difference in the suction temperature to the pumps.
132.	The design capacity of the reactor coolant pumps has been increased compared to Connecticut Yankee to accommodate the required total flow rate in the two loops. The increase is obtained through engineering extrapolation of the pump designs for Connecticut Yankee and San Onofre.
133.	The design head of the pumps has been increased due to the particular combination of components and system pressure drops for the Indian Point #2 design.
134.	The hydro test pressure is consistent with code requirements.
135. 136.	The type and design of the pump motors has not changed. The horsepower rating has increased to accommodate the increased flow and head requirements of the pump. The increase is obtained by engineering extrapolation of the previous motor designs.

Line Item	Notes
137. 138. 139. 140.	The reactor coolant piping has not changed significantly. The hot leg pipe has increased in diameter to maintain the same flow velocity limitation (<50 ft/sec) as used in the other plants. The pipe between the steam generator and the pump to meet the allowable velocity limits at the pump inlet which are less than the pipe requirements.
141.	The piping design pressure is the same as the other components of the reactor coolant system.

INDIAN POINT #2 COMPARISON WRITEUP #1
MAXIMUM OVERPOWER CONDITIONS

The primary consideration in overpower protection is not the actual value of the trip set point but rather the allowances that make up the margin to trip. The set point is selected so that a minimum DNB ratio of 1.3 is maintained at the condition of the maximum overpower (112% in the Indian Point #2 Plant) when all instrumentation errors are taken in the adverse direction and with the most adverse pressure and temperature allowed by the variable low pressure trip (again including errors in the adverse direction). The combination of these two protection channels (low pressure and overpower) limit the range of allowable plant conditions to a region of temperature, pressure and power which preclude DNB or core damage for credible accidents. For hypothetical accidents, such as control rod ejection, resulting in very rapid power excursion, the overpower set point has little or no effect on either the time to trip or the integrated power of the excursion.

The allowances tabulated in Table 1 are subject to verification by performance tests of the installed system. The errors due to drift and set point reproducibility for the Indian Point #2 Plant are errors quoted by many instrumentation manufacturers and are demonstrated in actual performance tests on the equipment before shipment. The improved performance is attributable to the use of a solid state system in the Indian Point #2 Plant.

TABLE 1

	Previous Designs (Point Detectors)		Indian Point #2 Plant (Long Ion Chambers)	
	Set Point	Maximum Overpower	Set Point	Maximum Overpower
Full Power	100	100	100	100
Calorimetric error	0	3	0	2
Transient Overshoot Allowance	+3	3	+3	3
Errors due to Rod Motion	+4	4	±1.5	3
Allowance for Drift and Set Point Reproducibility	±4	8	±2	4
	111%	118%	106.5%	112%

The errors due to rod motion result from variations in axial flux distribution with rod motion. Because of this variation, ion chamber readings at a given axial location may differ for the same core average power level. These errors are reduced by the use of long ion chambers with top and bottom detectors, each equal in length to about one-half the core height. The detectors yield an average reading over one-half the axial length. The 4 per cent error shown for the point detectors is not doubled for the maximum overpower since for that system either the top or the bottom detectors would show a positive error whereas for the long ion chambers a positive or negative error must be considered.

The reduction in calorimetric error is simply a recognition of the accuracy obtainable in current practice at conventional stations using the techniques and equipment which will be employed in the Indian Point #2 Plant.

Comparison of the two systems shows that the maximum overpower of 118% was required in previous design to prevent a spurious trip because of larger errors in the channel. The previous design did not offer any greater safety margin at the trip point, since the probability of getting a maximum overpower of 118% with the old system is as great as the probability of getting 112% with the new system and the margin to DNB would be the same in both cases.

INDIAN POINT #2 COMPARISON WRITEUP #2
INCREASED NOMINAL OPERATING PRESSURE

The reactor coolant system design pressure for the Indian Point #2 plant is 2500 psia, which is the same as for previous Westinghouse designs; e.g., San Onofre and Connecticut Yankee. For all conditions the system pressure is limited by code safety valves set to open at design pressure and sized to prevent system pressure from exceeding code limitations. Equipment capabilities for over-pressure protection are established by the complete loss of load without an immediate reactor trip. The maximum overpressure for this transient is therefore a function of the safety valve capacity and the maximum pressurizer surge rate and is not dependent on the value of the nominal operating pressure.

The operating pressure is selected to insure that desired thermal conditions are maintained in the core: for example, to limit the exit void fraction in the hot channels and to prevent DNB. Variable low pressure and fixed high pressure reactor trip circuits are provided to safely shut down the reactor in the event that operating pressure reaches the allowable limits. It is also desirable to limit pressure increases during normal operating transients sufficiently below the safety valve set point to preclude any safety valve leakage. A pressure control system consisting of heaters, spray and power operated relief valves is designed to suppress pressure surges during operating transients. Experience in operating pressurized water reactors has shown these methods of pressure control to be adequate and reliable.

The operating pressure is established and maintained between the upper and lower trip limits to permit transient variations in either direction with the assistance of the pressure control system.

The normal operating pressure for the Indian Point #2 Plant (2250 psia) has been increased over the previous designs; e.g., San Onofre (2100 psia) and Connecticut Yankee (2065 psia) based on preliminary design parameters and provides sufficient operating margin between the upper and lower pressure limits. Increasing the operating pressure allows higher reactor coolant operating temperatures and hence improves the secondary plant performance. This increase in operating pressure over previous designs is obtained in part because the difference in plant characteristics which increases the upper pressure limit to 2350 psia and by reducing the margin to the upper pressure limit to 100 psi. This reduction is based on the satisfactory operating experience obtained with the spray and relief valve pressure control system to be provided for the Indian Point #2 Plant and will present no operational difficulty.

In summary, therefore, the change to a higher nominal operating pressure in the Indian Point #2 design neither increases the probability of reaching the design pressure in operation, nor reduces the margin of safety in effect when design pressure is reached.

INDIAN POINT #2 COMPARISON WRITEUP #3
THERMAL-HYDRAULIC CORE DESIGN

The main core design limitation is departure from nucleate boiling (DNB). Present design methods are to use the W-3 DNB correlation in evaluating DNB and to design the core for a DNB ratio equal to or greater than 1.30 during steady state and operational transient conditions. The W-3 DNB correlation has been released in report WCAP-5584, "DNB Prediction for an Axially Non-Uniform Heat Flux Distribution" by L. S. Tong, September, 1965. Previous to using the W-3 correlation, the W-2 correlations were used. These were presented in "Nucleonics, May, 1963, "New Correlations Predict DNB Condition," L. S. Tong, H. B. Currin, A. G. Thorp, II. The minimum allowable design DNB ratio was 1.25 by this W-2 correlation. The minimum allowable design DNB ratio has been defined as that ratio at which it can be stated with a probability of 95% and a confidence level of 95% that DNB will not occur. The new minimum ratio of 1.3 by the W-3 is larger because the spread of the data on which the correlation is based is a little greater. This is largely due to the W-3 correlation covering the subcooled and quality regimes, from -15% to +15% quality. The W-2 correlations were one, the q'' correlation, for the subcooled region and a second, the H correlation, for the quality region.

The W-3 DNBR in the Indian Point #2 Plant is 1.81 at nominal operating conditions. The Brookwood design has 1.90 DNBR at nominal operating conditions and the Connecticut Yankee Design is 2.0 (by W-2). The differences are small and are caused by the different operating conditions of the plants, particularly the heat flux. As an indication of this, the peak heat fluxes are 517,500 and 437,200 Btu/hr-ft^2 respectively for the Brookwood and Connecticut Yankee Plants.

Since the two correlations are intrinsically different, the difference in the ratios for Indian Point #2 and Connecticut Yankee is also in the manner in which the various parameters affect the correlation.

The minimum DNB ratio in the core occurs in the hot channel, which is defined as the local unit cell at the nuclear flux peak in the core. At this location, certain engineering effects are presumed to adversely affect the thermal and hydraulic conditions. These factors are combined into the engineering hot channel factor. The total hot channel factors are the products of the nuclear and engineering factors. Two hot channel factors are considered, one for coolant enthalpy rise ($F^E_{\Delta H}$) and one for the heat flux (F^E_q).

The current engineering factor for coolant enthalpy rise is 1.075. It is a product of subfactors which consider the fuel pellet diameter, density, and enrichment; the fuel rod diameter, pitch and bowing; the effect of the inlet flow maldistribution; the core flow redistribution; and the coolant mixing in the core. The previously used engineering factor was 1.22. The following table shows the comparison:

$F^E_{\Delta H}$ ENTHALPY RISE HOT CHANNEL FACTOR

	Past	Current
Pellet Diameter, Density, Enrichment Rod Diameter, Pitch and Bowing	1.14	1.08
Inlet Flow Maldistribution	1.07	1.03
Flow Redistribution	1.05	1.05
Flow Mixing	0.95	0.92
Total	1.22	1.075

The change in the fuel pellet and fuel rod subfactor is due to an evaluation of actual measurements made on the fuel pellets and fuel rod spacing in the assemblies for the SELNI, Indian Point Core B, and two test grid type assemblies for Yankee Core II. All of these are grid type assemblies. Measurements taken to date on the SENA fuel assemblies indicate about the same deviation from the nominal conditions. On the basis of the above measurements, the subfactor has been reduced from 1.14 to 1.08.

The inlet flow maldistribution has been studied in one-seventh scale models of the San Onofre and Connecticut Yankee vessels. In both cases, a review of the flow distribution and a core power distribution indicated that the inlet flow maldistribution factor was 1.03. The Indian Point #2 vessel and internal designs are similar to these two plants. The 1.07 factor was verified in another 1/7 scale model of the SELNI vessel in which the lower plenum had a different design than presently used in Indian Point #2 or Brookwood.

Mixing tests have been performed on a grid type fuel assembly having mixing vanes on the grids to induce cross flow between the unit cells of the assembly. The tests demonstrated that the factor obtained with the mixing vane grid design is 0.92. The past factor 0.95 resulted from mixing test in a fuel assembly that did not have mixing vanes.

Measurements and testing will continue to be performed in order to further verify and reduce, if possible, the engineering hot channel factor, F_H^E.

The engineering hot channel factor for heat flux, F_q^E, is essentially the same as in the past, 1.04 and 1.045 are the present and past values respectively. This factor is based solely on the fuel pellet diameter, density and enrichment variations.

The following table shows the constituent parts of the total hot channel factor for the Brookwood Plant and factors used in previous designs.

	Indian Point #2		Past Design	
	$F_{\Delta H}$	F_q	$F_{\Delta H}$	F_q
Nuclear, Radial	1.75	1.75	1.73	1.73
Axial	-	1.78	-	1.80
Total Nuclear		3.12		3.11
Engineering	1.075	1.04	1.22	1.045
Total	1.88	3.25	2.11	3.25

Another thermal-hydraulic design criterion is a limit on the exit void fraction of the coolant at the outlet of the hydrodynamic hot array (non-statistical hot channels). This is imposed to protect against flow pattern instability in the core that may cause premature DNB. The non-statistical hot channels have been defined as the nuclear peak channels with only the flow subfactor of $I^E_{\Delta H}$ added. In other words, the fuel pellet and fuel rod considerations, which are of a statistical nature, are omitted. In the past, the limit was about 0 to 2% quality. Boiling heat transfer tests and DNB tests have shown that this quality can be definitely exceeded without deleterious effects. An investigation of the literature concerning flow instability indicated that void fraction is a better guide of instability than quality. From rod bundle DNB tests with boiling, stable operation was achieved at void fractions in excess of 32%. Thus the void fraction at the outlet of the non-statistical hot channels will be limited to 32% or less to avoid the possibility of flow instability. At 2250 psi, 32% void fraction is equivalent to 7.2% quality. Analyses are being initiated to study flow instability in an open core with the expectation of relaxing this limitation.

The physical parameters, other than the design limits, for the plant operation are selected and/or determined so that they satisfy the requirements for power production requirements and, as long as the basic thermal and hydraulic design criteria are satisfied, operation with the selected parameters is permitted. These are such items as system pressure, inlet temperature of coolant, coolant temperature rise, and total coolant flow rate.

INDIAN POINT #2 COMPARISON WRITEUP #4
FUEL DESIGN CONSIDERATIONS

The average heat flux of 175,600 Btu/hr ft^2 and the maximum heat flux of 570,800 Btu/hr ft^2 specified for the Indian Point #2 Plant will raise the temperature requirements of the fuel. The maximum fuel central temperature will be about 4200° F at 100% power and about 4400° F at overpower conditions. These temperatures are greater than those for the Brookwood and Connecticut Yankee plants but are well below the UO_2 melting point of about 5000 F°. Therefore the given values are acceptable and pose no limitations on the fuel. The temperature requirements for the clad material associated with these heat fluxes are also taken into consideration in the fuel element design. The cladding thickness of 0.0243 in. is chosen so that the stresses in the cladding material will be below the yield strength of the material throughout the life of the core for normal operating conditions and design transient conditions. The maximum cladding strains will be limited to 1/2 to 1 per cent throughout life and will be calculated considering internal fission gas pressure, reactor coolant pressure, fuel thermal expansion and swelling and clad creep. The diametral gap of 0.0065 inches is sufficient to accommodate the fuel thermal expansion.

The specified thermal output figures are: average, 5.70 kw/ft; maximum, 18.5 kw/ft; and maximum at overpower, 20.7 kw/ft. The delete experience outlined in Appendix A indicates that no problem exists at these stated thermal outputs.

The total hydrogen content, assuming an initial concentration of 25 ppm, to be expected in the cladding is as follows:

a.	At an average heat flux of	175,600 Btu/hr-ft^2
	After 1 year	∿ 72 ppm
	2 years	∿133 ppm
	3 years	∿202 ppm

b. At hot spot of 571,000 $Btu/hr\text{-}ft^2$

After 1 year	∿ 93 ppm
2 years	∿182 ppm
3 years	∿290 ppm

c. For combined 1 year at 571,000 $Btu/hr\text{-}ft^2$ + 2 years at 175,600 $Btu/hr\text{-}ft^2$ 230 ppm H_2.

Present experimental indications are that at least 500 ppm of hydrogen absorption can be tolerated in Zircaloy before change in mechanical properties becomes significant. Therefore, hydrogen absorption and hydriding problems will not arise in the Indian Point #2 plant fuel cladding and the corrosion weight gains associated with this hydriding are not expected to cause any problems.

The fuel element density of 93-94% is consistent with the expected burnups of 27,000 MWD/MTU (equilibrium core average) and 45,000 MWD/MTU (peak) so that there is sufficient porosity in the fuel and void space in the fuel elements to accommodate the expected fuel swelling. The plenum in each fuel rod will be sized to accommodate the fission gas release associated with these burnups so that the internal fission gas pressure will never exceed the reactor coolant normal operating pressure.

INDIAN POINT COMPARISON WRITEUP #5

SPATIAL STABILITY

I. DIFFERENCES

The only significant difference in the core physics design between Indian Point Unit #2 and Brookwood is the core diameter. Indian Point Unit #2 has an equivalent core diameter of 11.2 feet. The importance of the larger diameter relates to the sensitivity to potential spatial redistribution of power and the magnitude of the moderator temperature coefficient.

II. XENON INDUCED POWER REDISTRIBUTION

In Appendix C of the Preliminary Safety Analysis Report it was concluded that the oscillations due to xenon redistribution were damped both axially and azimuthally. This conclusion was based upon an axial calculation which demonstrated that the axial oscillation was damped and threshold analyses which indicated that azimuthal redistribution was less probable than axial.

At this time, the assertion that oscillations due to xenon redistribution will be damped cannot be proved. This is due primarily to the present inability to accurately estimate the damping effects of the power coefficient for a Zircaloy clad core. This is primarily because most of the experimental data has been obtained for steel clad cores.

In the event oscillations due to xenon occur, a number of possibilities exist to control the axial power redistributions. Specific calculations have been performed for one of these in which it was assumed that part length absorber rods were used to suppress the power peak. Two cases were examined using values for the power coefficient. In the first case a value for the power coefficient samewhat less than nominal was used and in the second case the power coefficient was assumed to be zero. In both cases, the results showed that a procedure can be easily provided to control the peak power without unduly restricting the motion of the power control group.

While the engineering design calculations to investigate azimuthal power redistribution due to xenon are less advanced than the axial studies because of the complexity of the calculations, preliminary results are available for an extreme case investigated to characterize the potential modes of azimuthal redistributions. In this case, the damping effect of the power coefficient was removed by setting the coefficient to zero and the oscillation was excited. The azimuthal oscillation obtained was essentially a tilt along the core diameter on which the excitation was introduced and showed little or no tendency toward precession. These results also suggest that the previous conclusions that the azimuthal redistributions are more damped than axial redistributions may not be correct.

In the event that induced azimuthal oscillations are not damped, it is judged by the success of the axial studies that they can be controlled. Engineering studies are being performed to demonstrate this fact. As opposed to the axial case in which the power control group motion provides a significant axial excitation, there are no intentional variations which excite the azimuthal redistribution. Other distinct differences between the two are recognized and are being considered in the azimuthal control studies as follows:

(1) The major azimuthal excitation is very likely to be the corrective action itself.

(2) Even though an uncorrected azimuthal redistribution will not precess, the control mode selected to suppress it must be sufficiently well distributed azimuthally that control induced precession is insignificant or can be compensated.

(3) Full length azimuthal control is desirable, although not necessary, since it retains separability between azimuthal and axial redistribution and simplifies corrective procedures.

The main conclusion is that azimuthal redistribution can be controlled. If required, this control will probably be accomplished by means of small reactivity insertion broadly dispersed over the core cross section. Design studies are under way which will determine the requirements for the control of azimuthal redistribution if any, in Indian Point Unit #2.

III. MODERATOR COEFFICIENT INDUCED REDISTRIBUTION

Appendix C of the Preliminary Safety Analysis Report presented an analysis of the potential for a moderator temperature induced power redistribution. This analysis* is conservative by some unknown but probably substantial amount because

(1) The power is assumed to rise in a channel directly according to the increase in core reactivity which results when the channel undergoes an increase in temperature or reduction in density. This is in direct opposition to results from Figure 2.2-5 in WCAP-2858 which demonstrates that the power level drops even though reactivity increases.

(2) A thermal-hydraulic amplification factor (M_H) is applied to the moderator coefficient on the assumption that all channels are at the worst possible thermal-hydraulic conditions. In Appendix C M_H was incorporated in the effective temperature rise to compute ρ_c even though it should not be applied to an axial calculation.

It was concluded in Appendix C that the moderator coefficient could be increased by about 150% before the conservative threshold for instability was reached (a calculated margin of 0.90% was obtained which could be added to the value ρ_c* equal to 0.61). The 150% margin in the moderator coefficient must account for any uncertainty in the calculated values. If conservative limits on the moderator and power coefficient are applied as in the Third Supplement to the Brookwood Preliminary Safety Analysis Report, the 150% margin is reduced to about 70%. This latter value is consistent with the presentation in the Third Supplement for Brookwood but is based upon calculations available for the Preliminary Safety Analysis Report for Indian Point Unit #2.

It is indicated by present design work that there will be some changes in the Indian Point Unit #2 constants which relate to instability induced by the moderator coefficient.

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*See Table I of Appendix C, conservative rather than nominal entries.

As indicated earlier, uncertainties exist in the effectiveness of the power coefficient. In addition, the azimuthal xenon calculations suggest that the azimuthal redistribution has somewhat less margin to instability than the previous threshold analysis predicted. It has been assumed that the margin to an instability induced by positive moderator coefficients is also reduced accordingly. Table I presents a revised estimate of margin to azimuthal instability for Indian Point Unit #2 with the conservative equation considering these factors and including allowances for analytic uncertainty on a basis consistent with Supplement III for Brookwood. The uncertainty in temperature coefficient is taken to be +0.25 x 10^{-4} $(°F)^{-1}$ or 25% and the uncertainty in power coefficient is taken to be 20%. It is concluded that there is an adequate margin in the moderator coefficient considering the conservative threshold analysis which has hidden in its assumptions, perhaps an additional 100% margin in the moderator coefficient.

Table I

Factor	Symbol	Reactivity
Leakage increment	$m^2 \; \Delta B^2$	+ 0.38
Power defect	$-\rho_F^*$	+ 0.50
Coolant reactivity including hydraulic multiplier of 2.0	$-\rho_c^*$	- 0.76
Conservative Threshold Margin		0.12

To provide a better understanding of the significance of the margin, Figure 1 presents the variation in (ρ_c - $M^2 \Delta B^2$) and in $\dot{\rho}_F$ as a function of the fraction of core full power. Instability is possible when these curves cross. The solid lines indicate results which have margin in calculated values of input parameters and employ a conservative hydraulic interaction factor of 2.0. At full power (P/P_o = 1.0) the values are consistent with Table I. From this set of curves, on a very conservative basis, the power must be increased by 45% above full power to work an unstable situation. The dashed lines give the best estimate of the actual situation. With the best estimate, there is no power level which will result in an unstable situation. Clearly, the core would have to remain at the overpower at which instability is expected until the moderator heated up to a nearly steady state condition and an overpower transient (normally limited to 12% overpower) would not be of sufficient duration to excite the instability.

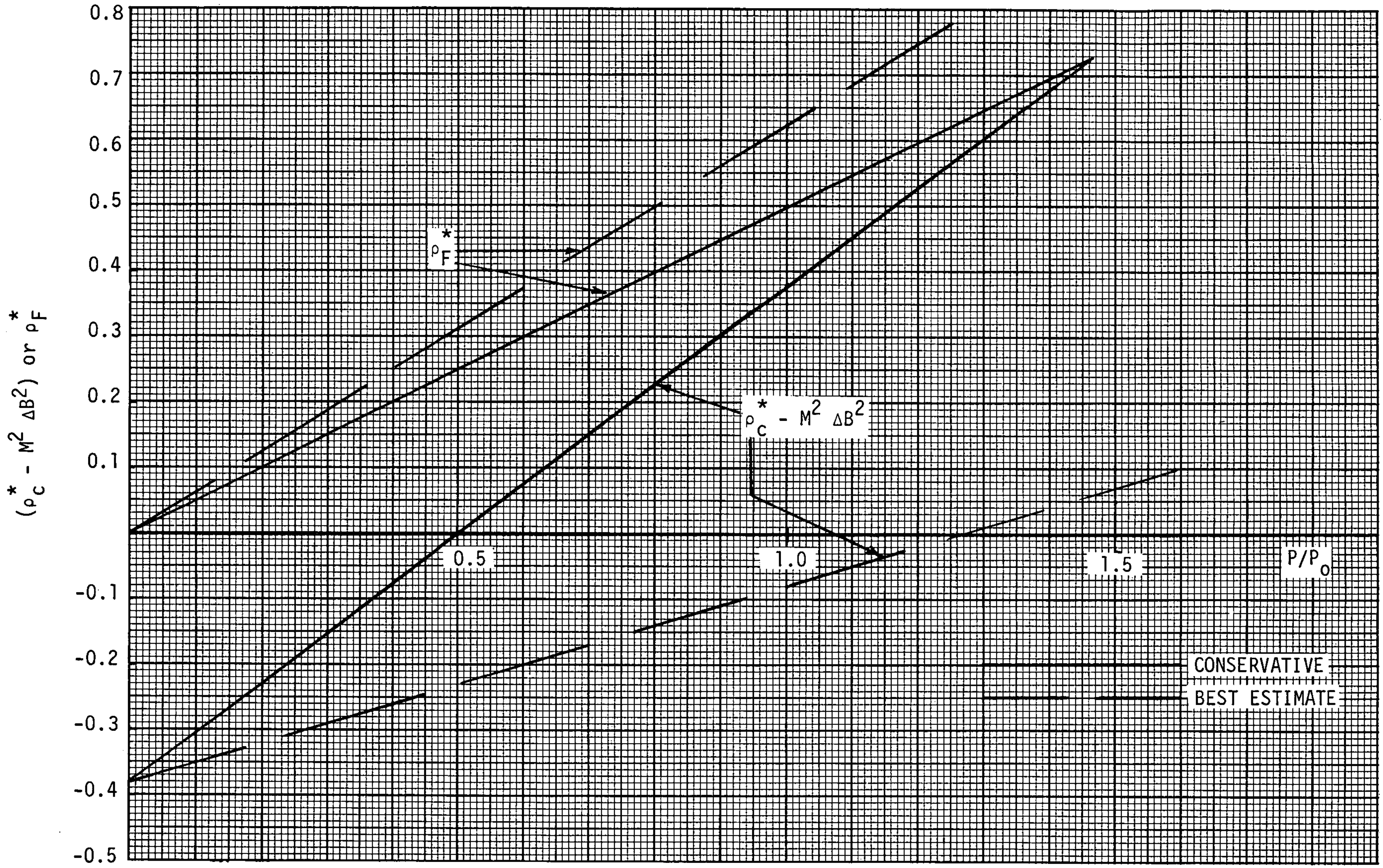

FIGURE 1

STABILITY MARGIN AS A FUNCTION
OF POWER LEVEL

INDIAN POINT #2 COMPARISON WRITEUP #6

ASME BOILER AND PRESSURE VESSEL CODE

The Indian Point #2 plant vessels will be designed and built to Section III, "Nuclear Vessels," of the ASME Boiler and Pressure Vessel Code instead of Section VIII, "Unfired Pressure Vessels," of the code with its Nuclear Code Cases for the following reasons. When Section III was published, the Nuclear Code Cases were annuled as of December 31, 1964, which precedes the contract date of this plant. In addition, the ASME code committee decreed that no nuclear vessels contracted for after that date can be considered as complying with the ASME Boiler and Pressure Vessel Code unless compliance with all the requirements of Section III has been demonstrated.

In addition, Section III is considered to be a better design guide because it has significantly upgraded Section VIII and its associated Nuclear Code Cases. It presents in a new and complete package, the latest skills in the analytical techniques of pressure vessel design and improved knowledge of the failure patterns in pressure vessels.

The differences between Section III and Section VIII of the code apply primarily to those vessels designated as Class A vessels in Section III. For Class C vessels, the subsection of Section III that deals with these vessels explicitly states that Section VIII regulations apply with a few additional requirements. These additional requirements are that (1) paragraph U-1(g) of Section VIII which waives specific inspection requirements of certain vessels shall not apply to Class C vessels, (2) weld joints of category A and B shall meet requirements of Class A vessels, and (3) the Section VIII code stamp for these vessels shall include the letter "N".

Section III of the code, as it pertains to the Class A vessels, is a better code than Section VIII from the design standpoint for the following reasons. It requires the detailed calculation and classification of all stresses and the application of different stress limits to different classes of stress, whereas Section VIII only gives formulas for minimum allowable wall thickness. Section III requires calculation of thermal stresses and gives allowable values for them whereas Section VIII does not. Section III considers the possibility of fatigue failure and gives rules for its prevention, whereas Section VIII does not.

In summary, Section III is an upgrading and compilation of Section VIII and the Nuclear Code Cases into one document. Section VIII with the special rulings for nuclear vessels (Nuclear Code Cases) was comparable to Section III as far as the quality of material, fabrication, and inspection was concerned. The design analysis techniques were comparable to PB 151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components," which is the basis for the design and analysis criteria of Section III.

The hydrostatic test pressure for the primary (tube) side of the steam generator and the reactor coolant pump have changed from 3735 psig for Connecticut Yankee to 3110 psig for Indian Point #2 and Brookwood because of the change from Section VIII of the ASME Boiler and Pressure Vessel Code as the basis for vessel design to Section III which covers nuclear vessels. Section VIII requires a hydrostatic test of 1.5 times the design pressure while Section III requires a hydrostatic test of 1.25 times the design pressure. This lower test pressure for Section III vessels is in recognition of the fact that the higher allowable design stresses permitted by Section III require lower test pressures to keep primary membrane stresses during hydrostatic test below 90% of the yield strength of the material.

Considering the design basis, the 3110 psig hydrostatic test on a Section III vessel is equivalent to a 3750 psig hydrostatic test on a Section VIII vessel and in conjunction with other non-destructive tests required by Section III provides adequate assurance against design or fabrication defects which would affect vessel reliability.

2. The Commssion published on November 22, 1965, "General Design Criteria for Nuclear Power Plant Construction Permits." Please provide a discussion, together with data and analysis, sufficient to show how each of the criteria applicable to your facility will be fulfilled.

Answer

The 27 "General Design Criteria for Nuclear Power Plant Construction Permits" as published for public comment on November 22, 1965, are still in the process of review. Until such review is completed and a final version of these criteria is published, the present set of criteria will be considered preliminary in nature and subject to interpretation when applying the criteria to a specific reactor plant.

The Indian Point Unit #2 design has been evaluated against these criteria and the details of the evaluation are presented in the following pages. We believe that these criteria, as interpreted to apply to pressurized water reactors, are satisfied.

CRITERION 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for non-nuclear applications may not be adequate.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice and other natural phenomena anticipated at the proposed site.

Features of the facility essential to accident prevention and mitigation are the fuel, reactor coolant and containment barriers; the controls and emergency cooling systems whose function is to maintain the integrity of these three barriers; systems which depressurize and reduce the contamination level of the containment; power supplies and essential services to the above features; and the components employed to safely convey and store radioactive materials including spent reactor fuel.

Quality standards of material selection, design, fabrication, and inspection governing the above features will conform to the applicable provisions of recognized codes and good nuclear practice. Vessels will comply with Section III of the ASME Boiler and Pressure Vessel Code under the specific classification dictated by their use. The principles of this Code, or equivalent guidelines, will be employed where the Code is not strictly applicable but where the safety function calls for an equivalent assurance of quality.

In the same manner, piping will conform to the requirements of ASA standard B 31.1, and the reinforced concrete structure of the reactor containment will conform to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is given in Chapter 5.0 of the Preliminary Safety Analysis Report.

In the case of the reactor vessel, additional quality assurance is gained through measurement of physical and chemical properties as well as nil ductility transition temperature (NDTT) to ascertain that these are in accordance with the Code and acceptable for the application. Westinghouse verifies that these measurements are made by the vessel and material vendors, and develops further checks on material quality by pre-irradiation tests associated with the vessel surveillance program. As will be shown by the design analysis performed for the reactor vessel, the fatigue usage factor, derived from an assumed number of thermal cycles which is more than four times the probable number of such cycles for this plant, will be less than that at which propagation of material defects would occur. Design margin and material surveillance ensure that the vessel will be operated well within the ductile range of temperatures when stressed above 20% of the material yield point. The reactor vessel size is within the range of previous experience of the manufacturer and of the nuclear plant designer.

The reactor vessel and other components of the Reactor Coolant System are designed for a temperature of 650°F and pressure of 2485 psig. The normal operating conditions of 2235 psig and hot leg temperature of 596°F provide adequate margins for normal maneuvering and operating transients.

All piping, components and supporting structures of the reactor and safety related systems are designed in accordance with Class I seismic criteria. These criteria specify that there will be no loss of function of such equipment in the event of a ground acceleration of 0.15 g, acting in the horizontal direction and 0.10 g in the vertical direction simultaneously. The dynamic response of the structure to ground acceleration, based on appropriate spectral characteristics of the site foundation and on the damping of the structure, is included in the design analysis.

The reactor containment is similarly defined as a Class I structure. Its structural members will have sufficient capacity to accept without exceeding yield stresses a combination of normal operating loads, functional loads due to the 47 psig design pressure acting simultaneously with a 0.1 g seismic ground acceleration in the horizontal and .05 g acceleration acting in the vertical direction. Loadings imposed by the maximum wind velocity specified by local construction codes will be compared with those due to earthquake, and the larger of the two will be used in the design of each member.

(Revised 6-1-66)

The emergency on-site power sources are designed for maximum earthquake loading and are not subject to interruption due to windstorm, ice, or to disturbances on the external power grid. Power cabling, motors and other equipment required for operation of the engineered safeguards are suitably protected against the effects of the accident, or of severe external weather conditions as applicable, to obtain a high degree of confidence in the operability of these systems in the event they should be required.

CRITERION 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or material augment the release of significant amounts of fission products from the facility.

The most severe chemical reaction is the zirconium-water reaction. Such a reaction could result if core cooling were lost for a significant time because of an uncontrolled loss of coolant from the reactor coolant system.

The consequences of the zirconium-water reaction are two additional energy sources: (1) the exothermic heat of reaction stored in the core which would be released when the core is quenched, and (2) the H_2-O_2 reaction energy released to the containment. The zirconium-water reaction is limited by the same safeguard that cools the core following a loss-of-coolant accident; i.e., the safety injection system.

The safety injection system consists of three high head, low flow safety injection pumps (design conditions each: 400 gpm at 2500 ft), and two high flow, low head pumps of the residual heat removal system (design conditions each: 3000 gpm at 280 ft). The safety injection pumps inject into each reactor coolant cold leg, while the residual heat removal pumps deluge the core by nozzles that are connected to each reactor coolant hot leg. The flow and head of the pumps and the supply of water are sized to give adequate core protection for the full range of break sizes. In addition, the charging pumps of the chemical and volume control system are normally available but are not required to augment the flow of the safety injection system.

For the hypothetical accident (doubled-ended break of reactor coolant pipe), with full operation of the safety injection system, the core is completely reflooded in 285 seconds and the Zr-H_2O reaction is approximately one per cent. For the same accident, but with operation of two out of three diesels and minimum safeguards components (one high head pump and one residual heat removal pump), the core is completely reflooded in 430 seconds and the $Zr-H_2O$ reaction is limited to approximately five per cent. In either case described above, the containment design pressure is not exceeded.

Although the engineered safeguards loads are arranged to operate from electrical buses supplied from normal outside AC power which should not fail as a result of reactor trip, reliable on-site emergency power is provided. Thus, if normal AC power to the station is lost concurrent with a loss-of-coolant accident, power is available for the engineered safeguards.

In the event that one of the three diesel generators fails to start immediately, any of the engineered safeguards equipment normally supplied by that diesel is automatically transferred, if required, to one of the other two diesel supplied buses. In this manner, any engineered safeguard component can be used to meet the minimum starting requirements listed below. The minimum safeguards load started under these conditions is:

1 Residual Heat Removal Pump (Core Deluge; Hot Leg Injection)
1 High Head Safety Injection Pump (Cold Leg Injection)
4 Containment Fan-Cooler-Filter Units
1 Containment Spray Pump
1 Service Water Pump

CRITERION 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

Missile protection for the Indian Point Unit #2 will be provided to comply with the following criteria:

a) The containment and liner shall be protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.

b) The engineered safeguards systems and components required to maintain containment integrity and to meet the site criteria of 10 CFR 100 shall be protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria will be developed and implemented using the following considerations:

a) The reactor coolant system, including the pressurizer and steam generators will be surrounded by reinforced concrete and/or steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe and designed to stop the missiles.

b) The structural design of the missile shielding will take into account both static and impact loads and will be based upon the state of the art missile penetration data.

c) Missile velocities will be calculated considering both fluid and mechanical driving forces which can act during missile generation.

d) Components of the reactor coolant system will be examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.

The types of missiles for which missile protection will be provided are:

a) All valve stems up to and including the largest size to be used
b) All valves up to and including the largest size to be used
c) Massive chunks of metal up to 6 inches thick
d) All valve bonnets
e) All instrument thimbles
f) Various type and sizes of nuts and bolts
g) Pieces of pipe up to 10-inch diameter striking broadside or end on
h) Complete control rod drive mechanisms
i) Reactor vessel head bolts

CRITERION 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine-generator and loss of power to the reactor recirculation system pumps.

The reactor protection system is designed to actuate a reactor trip for any credible combination of plant conditions which can cause DNB and possible fuel failure or coolant system damage.

In the unlikely event of a complete loss of load from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer will result in a high pressurizer pressure trip without fuel damage. A loss of load of 50% of full power will be controlled by rod cluster insertion and steam dump to prevent a large temperature and pressure increase and thus prevent a reactor trip. In this case, the variable low pressure trip would guard against any combination of pressure, temperature and power which could result in DNB during the transient.

The controlled leakage pumps provided for the Indian Point Unit 2 Plant will be supplied with sufficient rotational inertia to maintain an adequate flow coastdown in the event of simultaneous loss of power to all pumps. The amount of required inertia is established when detailed system design parameters are available. The flow coastdown of the pumps will provide enough flow to prevent core damage following the low flow reactor trip. Subsequent removal of heat from the core is covered by the procedures outlined in Criterion 10.

In neither the loss-of-load nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion. This result is a consequence of the large system thermal inertia (slow reactivity feedback due to coolant temperature changes) and small void fraction (small reactivity feedback due to small void fraction change). Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

CRITERION 5

The reactor must be designed so power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

The reactor control and protection systems are designed to safely shut down the plant without fuel damage in the event of credible transients which approach protection limits. Those considerations which may lead to spatial instabilities will be examined in detail during the design and either it will be demonstrated that such instabilities will not occur or their effects will be included in the design of the control and protection system.

Two types of oscillations are of interest: first, variation of the coolant average temperature within the control deadband; and second, spatial oscillation of power distribution in the core.

Throughout core life, rod cluster motion under either automatic or manual control is required to follow load changes in order to maintain coolant average temperature in accordance with a pre-determined load program. During the portion of core life in which the moderator coefficient is positive, control rod motion will serve the additional function of periodically compensating for the slight positive feedback effect of moderator temperature. The temperature oscillation associated with this condition has been calculated to be within $\pm 2°$ F of the programmed value, and the shortest period will be approximately two to four minutes. (This oscillation corresponds to a core power change of about 2 per cent per minute.) The control and protection capabilities demanded by design station load transients of 5 per cent per minute are more restrictive than those associated with this oscillation; hence no adverse effect on primary system or fuel integrity is foreseen.

In Appendix C of the Preliminary Safety Analysis Report it was concluded that the oscillations due to xenon redistribution were damped both axially and azimuthally. This conclusion was based upon an axial calculation which demonstrated that the axial oscillation was damped and threshold analyses which indicated that azimuthal redistribution was less probable than axial.

At this time, the assertion that oscillations due to xenon redistribution will be damped cannot be proved. This is due primarily to the present inability to accurately estimate the damping effects of the power coefficient for a Zircaloy clad core. This is primarily because most of the experimental data has been obtained for steel clad cores.

In the event oscillations due to xenon occur, a number of possibilities exist to control the axial power redistributions. Specific calculations have been performed for one of these in which it was assumed that part length absorber rods were used to suppress the power peak. Two cases were examined using values for the power coefficient. In the first case a value for the power coefficient samewhat less than nominal was used and in the second case the power coefficient was assumed to be zero. In both cases, the results showed that a procedure can be easily provided to control the peak power without unduly restricting the motion of the power control group.

While the engineering design calculations to investigate azimuthal power redistribution due to xenon are less advanced than the axial studies because of the complexity of the calculations, preliminary results are available for an extreme case investigated to characterize the potential modes of azimuthal redistributions. In this case, the damping effect of the power coefficient was removed by setting the coefficient to zero and the oscillation was excited. The azimuthal oscillation obtained was essentially a tilt along the core diameter on which the excitation was introduced and showed little or no tendency toward precession. These results also suggest that the previous conclusions that the azimuthal redistributions are more damped than axial redistributions may not be correct.

In the event that induced azimuthal oscillations are not damped, it is judged by the success of the axial studies that they can be controlled. Engineering studies are being performed to demonstrate this fact. As opposed to the axial case in which the power control group motion provides a significant axial excitation, there are no intentional variations which excite the azimuthal redistribution. Other distinct differences between the two are recognized and are being considered in the azimuthal control studies as follows:

(1) The major azimuthal excitation is very likely to be the corrective action itself.

(2) Even though an uncorrected azimuthal redistribution will not precess, the control mode selected to suppress it must be sufficiently well distributed azimuthally that control induced precession is insignificant or can be compensated.

(3) Full length azimuthal control is desirable, although not necessary, since it retains separability between azimuthal and axial redistribution and simplifies corrective procedures.

The main conclusion is that azimuthal redistribution can be controlled. If required, this control will probably be accomplished by means of small reactivity insertion broadly dispersed over the core cross section. Design studies are under way which will determine the requirements for the control of azimuthal redistribution if any, in Indian Point Unit #2.

Appendix C of the Preliminary Safety Analysis Report presented an analysis of the potential for a moderator temperature induced power redistribution. This analysis* is conservative by some unknown but probably substantial amount because

(1) The power is assumed to rise in a channel directly according to the increase in <u>core</u> reactivity which results when the channel undergoes an increase in temperature or reduction in density. This is in direct opposition to results from Figure 2.2-5 in WCAP-2858 which demonstrates that the power level drops even though reactivity increases.

(2) A thermal-hydraulic amplification factor (M_H) is applied to the moderator coefficient on the assumption that <u>all</u> channels are at the worst possible thermal-hydraulic conditions. In Appendix C M_H was incorporated in the effective temperature rise to compute ρ_c even though it should not be applied to an axial calculation.

It was concluded in Appendix C that the moderator coefficient could be increased by about 150% before the conservative threshold for instability was reached (a calculated margin of 0.90% was obtained which could be added to the value ρ_c* equal to 0.61). The 150% margin in the moderator coefficient must account for any uncertainty in the calculated values. If conservative limits on the

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*See Table I of Appendix C, conservative rather than nominal entries.

moderator and power coefficient are applied as in the Third Supplement to the Brookwood Preliminary Safety Analysis Report, the 150% margin is reduced to about 70%. This latter value is consistent with the presentation in the Third Supplement for Brookwood but is based upon calculations available for the Preliminary Safety Analysis Report for Indian Point Unit #2.

It is indicated by present design work that there will be some changes in the Indian Point Unit #2 constants which relate to instability induced by the moderator coefficient.

As indicated earlier, uncertainties exist in the effectiveness of the power coefficient. In addition, the azimuthal xenon calculations suggest that the azimuthal redistribution has somewhat less margin to instability than the previous threshold analysis predicted. It has been assumed that the margin to an instability induced by positive moderator coefficients is also reduced accordingly. Table I presents a revised estimate of margin to azimuthal instability for Indian Point Unit #2 with the conservative equation considering these factors and including allowances for analytic uncertainty on a basis consistent with Supplement III for Brookwood. The uncertainty in temperature coefficient is taken to be $+0.25 \times 10^{-4}$ $(°F)^{-1}$ or 25% and the uncertainty in power coefficient is taken to be 20%. It is concluded that there is an adequate margin in the moderator coefficient considering the conservative threshold analysis which has hidden in its assumptions, perhaps an additional 100% margin in the moderator coefficient.

Table I

Factor	Symbol	Reactivity
Leakage increment	$m^2 \ \Delta B^2$	+ 0.38
Power defect	$-\rho_F{}^*$	+ 0.50
Coolant reactivity including hydraulic multiplier of 2.0	$-\rho_c{}^*$	- 0.76
Conservative Threshold Margin		0.12

To provide a better understanding of the significance of the margin, Figure 1 presents the variation in ($\rho_c - M^2 \Delta B^2$) and in ρ_F as a function of the fraction of core full power. Instability is possible when these curves cross. The solid lines indicate results which have margin in calculated values of input parameters and employ a conservative hydraulic interaction factor of 2.0. At full power ($P/P_o = 1.0$) the values are consistent with Table I. From this set of curves, on a very conservative basis, the power must be increased by 45% above full power to work an unstable situation. The dashed lines give the best estimate of the actual situation. With the best estimate, there is no power level which will result in an unstable situation. Clearly, the core would have to remain at the overpower at which instability is expected until the moderator heated up to a nearly steady state condition and an overpower transient (normally limited to 12% overpower) would not be of sufficient duration to excite the instability.

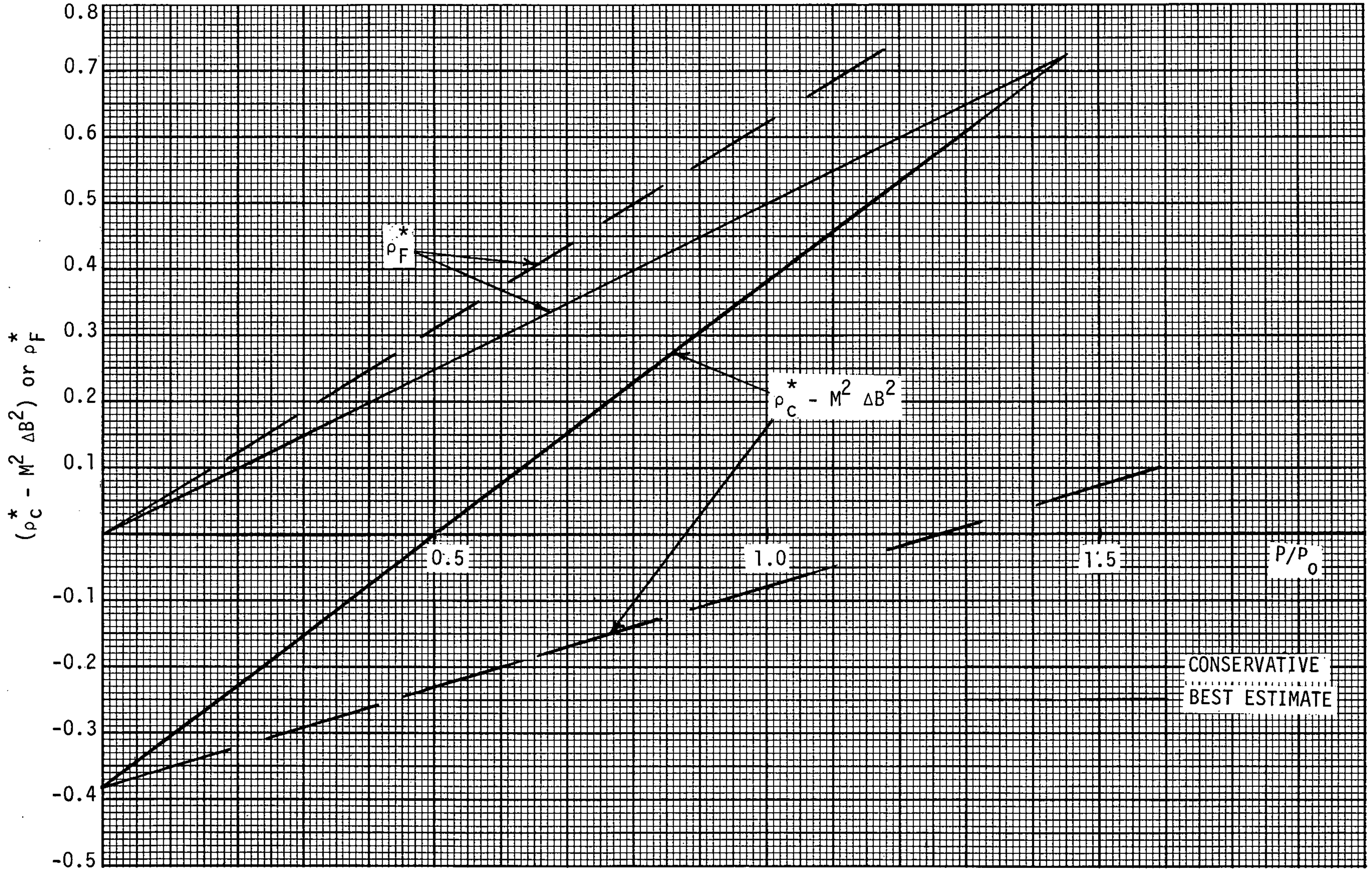


FIGURE 1

STABILITY MARGIN AS A FUNCTION
OF POWER LEVEL

CRITERION 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

The reactor fuel elements will be 12 feet in length and contain UO_2 fuel. Fuel elements of three enrichments, i.e., 2.23, 2.38 and 2.68 weight per cent U-235, will be present in the first core. The anticipated average burnup for the first core will be 21,800 MWD/MTU. The equilibrium core feed enrichment is about 2.92 weight per cent which will yield an average equilibrium core burnup of 27,000 MWD/MTU.

The fuel rods will be clad with Zircaloy having an outside diameter of 0.422 inch and wall thickness of 0.0243 inch. No unclad or vented fuels will be used.

The integrity of the fuel cladding is ensured by preventing excessive fuel swelling, excessive clad overheating, and excessive cladding stress. This is achieved by designing the fuel elements so that the following limits are not exceeded during anticipated normal or abnormal operating conditions (including an accidental overpower condition of 112%):

1) Minimum DNB ratio equal to or greater than 1.3
2) Fuel center temperature below melting point of UO_2
3) Clad stresses less than the Zircaloy yield strength

For the total loss of flow a DNB ratio of lower than 1.3 may occur but clad damage (excessive clad temperatures) will not occur and limits (2) and (3) will be met.

The fuel rod cladding will be designed such that the internal gas pressure will be less than the nominal external pressure (2250 psia), even at the end of life.

CRITERION 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

The reactor control system employs 53 control rod clusters, approximately half of which are fully withdrawn during power operation, serving as shutdown rods. The remaining rods comprise the controlling groups, and are used to control load and reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and these group configurations are not altered during core life. The rods are therefore prevented from withdrawing in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced by a fixed speed motor driven cycler which provides a limit on the maximum rod withdrawal speed. The maximum reactivity insertion rate occurs when the maximum worth group is withdrawn at the maximum speed while in the region of maximum incremental worth. This maximum reactivity insertion rate is of the order of 2×10^{-4} Δk/sec, which is well within the capability of the overpower and variable low pressure protection circuits to prevent core damage.

No credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than its drive speed. The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not a credible accident.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, there are only a few rods (and these rods are only partially inserted) in the core at full power. By utilizing the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the final design will limit the maximum fuel temperature for the highest worth ejected rod to a value which will preclude any consequential damage to the primary system, i.e. gross fuel dispersion in the coolant and possible pressure surge. Fuel damage would occur for this accident. The results to be obtained in the final design studies are not expected to differ significantly from those obtained for the San Onofre reactor, in which these criteria were met (See Question 11).

CRITERION 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

The maximum excess reactivity expected for the Indian Point Unit 2 core is 0.275 and occurs for the cold, clean condition at the beginning of life of the initial core. This excess reactivity will be controlled by a combination of control rods and soluble neutron absorber (boron). A total of 53 Rod Cluster Control (RCC) Assemblies are provided with a total worth of 0.07. These RCC assemblies are divided into two categories, a control group and a shutdown group.

The control group, used in combination with chemical shim control, provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup.

The shutdown group is provided to supplement the control group of RCC assemblies to make the reactor at least one per cent subcritical ($k_{eff} = 0.99$) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position. Boric acid solution will be used to supplement the RCC assemblies in maintaining the shutdown margin for the long term conditions of xenon decay or plant cooldown.

CRITERION 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating conditions.

Reactor shutdown with rods is completely independent of the normal control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals. (Refer to Criterion 12.)

Normal reactivity shutdown capability will be provided by control rods with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. Any time that the plant is at power, the quantity of boric acid retained in the boric acid tanks is ready for injection and will always exceed that quantity required for a normal cold shutdown. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay (k_{eff} = 0.99 after xenon decay).

Boric acid will be pumped from the boric acid tanks by either one of two boric acid pumps to the suction of either one of two charging pumps which will inject boric acid into the reactor coolant. Boric acid can be injected by one pump at a rate which will shut the reactor down with no rods inserted in less than fifteen minutes. In fifteen additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level will not begin until approximately 15 hours after shutdown. Also, assuming no control rod motion following shutdown, additional boric acid injection will be employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

CRITERION 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

Redundant heat removal systems are provided that are capable of removing core decay heat following all anticipated abnormal and credible accident conditions.

The sequence of events following the loss of all external power is as follows:

1) Turbine and reactor trip
2) Loss of feedwater flow
3) Loss of the main condenser as a heat sink following loss of circulating water flow

The loss-of-power signal opens the steam admission valve to the emergency turbine-driven feedwater pump.

The capacity of the emergency feedwater pump driven by steam from the steam generators is sufficient to restore the normal water level in the steam generators without uncovering the steam generator tube sheet and to maintain the plant in the hot shutdown condition. Heat is transferred from the core to the steam generators by natural circulation of the reactor coolant. Continued removal of residual heat is accomplished by the discharge of secondary steam to the atmosphere. The supply of stored condensate is adequate to dissipate decay heat in this manner for about 24 hours. Alternate supplies of water for continued long term cooling will be provided from either the primary plant make up supply or from the 1.5 million gallon plant storage tank for city water.

Following a loss-of-coolant accident, heat removal from the reactor core and from the containment atmosphere is accomplished by the engineered safeguard systems.

The Safety Injection System initially supplements the reactor coolant inventory with cool, borated water from the refueling water storage tank. Heat is transferred to the water surrounding the core, resulting in the formation of some steam and raising the temperature of the injected water, some of which subsequently spills to the containment sump. Before the supply of water in the refueling water storage tank is depleted, water from the sump is recirculated to the reactor coolant system through the residual heat exchangers. At a minimum, this recirculated water flow is sufficient to assure continued cooling of the core by evaporation, using one residual heat removal pump and residual heat exchanger and one safety injection pump. This evaporation will release some or all of the residual heat to the containment.

Two means of removing heat from the containment atmosphere are provided:

1) The air recirculation units cooled by service water, and

2) The containment spray and water recirculated to the core by a residual heat removal pump, a safety injection pump, and cooled by component cooling water circulated through the residual heat exchanger.

Each of these systems provide sufficient steam-condensing capacity to assure against containment overstress and to remove that portion of the residual heat released to the containment.

Two of the three emergency diesel generator sets can power the residual heat removal pump, component cooling water pump, service water pump, and safety injection pump which are needed to assure recirculation and cooling of water from the sump to the reactor core. In addition, the equipment for either of the cooling systems needed to assure sufficient containment heat removal can be powered without exceeding the capability of two of the three diesels. These additional loads are either:

1) For operation of the ventilation air cooling units:
   a. One service water pump
   b. Four of five containment cooling unit fans, or

2) For recirculation spray and core cooling:

   a. One service water pump

   b. One component cooling water pump

   c. One containment spray pump

This partial operation of either cooling system provides enough capacity for the continued removal of residual heat from the containment atmosphere.

CRITERION 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are NDT + 60°F and NDT + 30°F, respectively.

The design transition temperature (DTT) for the reactor vessel material before irradiation will be specified after the actual notch ductility properties of the materials have been determined. The specified DTT for each plate will be a minimum of nil ductility transition temperature (NDTT) plus 60°F at all times and will dictate the procedures followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of DTT will be increased during the life of the plant as required by the expected shift in the NDT temperature, confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime.

A DTT shift of 275°F has been selected as the value which will allow plant operation on the basis of design plant heatup and cooldown rates and yet provide an adequate margin between the maximum DTT and the normal reactor operating temperature. The DTT shift of 275°F corresponds to an integrated fast flux (E 1 Mev) of approximately 3.7×10^{19} n/cm^2 which is well in excess of the expected exposure of 0.85×10^{19} n/cm^2 for this vessel throughout the plant lifetime.

To define permissible operating conditions below DDT, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit of vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. As the normal operating temperature of the reactor vessel will be well above the maximum expected DTT, brittle fracture during normal operation is not considered credible.

(Revised 6-1-66)

The reinforced concrete vapor container is not susceptible to a low temperature brittle fracture for the following reasons:

1) The structure is non-homogeneous and therefore crack propagation is confined to individual bars.

2) The loadings are not impact or cyclical.

3) The carrying capacity of a bar would increase from zero at a break in proportion to the bond shear between the steel and concrete. Therefore, the loss of capacity in the bar will be localized and not effect the full length of a bar.

4) Triaxial stresses which inhibit ductility do not arise because the bars are loaded uniaxially.

The containment liner is enclosed within the containment and thus will not be exposed to the temperature extremes of the environs. The containment ambient temperature during plant operation will be between 50°F and 120°F which is well above the NDTT + 30°F for the liner material.

CRITERION 12

Capability for control rod insertion under abnormal conditions must be provided.

The Indian Point Unit 2 reactor will use the Westinghouse magnetic latch-type control rod drive mechanisms which are the same type as those used in the San Onofre and Connecticut-Yankee plants. Upon a loss of power to the coils, the rod cluster control assembly will be released and fall by gravity into the core.

The reactor internals, fuel assemblies, RCC assemblies and drive system components will be considered Class I for seismic design purposes. The RCC assemblies will be fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube above. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Due to this and the flexibility designed into the RCC assemblies, normal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

The RCC control rod guide system throughout its length is locked together with dowels to insure against misalignments which would impair control rod movement under normal operating conditions and credible accident conditions. An analogous system has successfully undergone 2539 hours of testing during which 527,000 steps and 1000 trips were accomplished, in the Westinghouse Reactor Evaluation Channel with test alignments in excess of the maximum possible misalignment that may be experienced in plant. Reports describing these tests are given as Reference 2, page 3.2-65 of the Preliminary Safety Analysis Report.

CRITERION 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

The Indian Point Unit #2 facility will be equipped with a control room which will contain all controls and instrumentation necessary for operation of the reactor and turbine generator under normal or accident conditons.

Personnel remaining in the control room for an indefinite period of time following an accident of the type described in Criterion 17 in either Unit #1 or Unit #2 would receive radiation doses less than 1.5 rem (whole body) and 3 rem (thyroid). The derivation of these results, as discussed in the answer to Question 17, considers only the containment shielding and the physical separation of the control room from the respective containments and potential leakage points. No credit is taken for the protection afforded by the control room structure itself.

These radiation levels will allow access to and egress from the site and permit operators to make excursions out of the control room closer to the containment. The permissible time out of the control room will depend on the time elapsed following the accident and the distance from the containment, and will be limited so that the maximum potential dose for the course of an accident will not exceed 25 rem.

CRITERION 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

When the reactor is critical, means for showing the relative reactivity status of the reactor will be provided by an individual position indicator for each RCC assembly displayed on a panel in the control room. The position of the control group of RCC assemblies is directly related to the reactivity status of the reactor when at power and any unexpected change in the position of the control group of RCC assemblies under automatic control or a change in the coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. Periodic samples of coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

During refueling operations, the boron concentration in the refueling water is perodically sampled. This concentration is sufficient, with the control rods, to maintain a reactivity shutdown margin of approximately ten per cent. Continuous mixing will be maintained through the reactor vessel by the residual heat removal flow. Neutron sources installed in the core and separate BF_3 monitoring with audible count rate provide immediate indication of a change in reactivity status of the core. In the event of dilution, the count rate would increase approximately in proportion to the multiplication factor. A reduction in shutdown margin from ten per cent to five per cent would result in an increase in count rate by a factor of more than two.

Any appreciable increase in the neutron source multiplication, induced by the maximum physical dilution rate (approximately 580 ppm per hour), is slow enough to give ample time to effect corrective action (terminating dilution by tripping the makeup water pumps which are the only source of unborated water and initiating boration).

The maximum dilution rate is based upon the abnormal condition of both charging pumps operating at full speed, delivering unborated makeup water to the reactor coolant system at a particular time during the refueling operation when boron concentration is at the maximum value and the water volume in the system is at a minimum.

CRITERION 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

Every automatic protection channel is at least duplicated (a one-out-of-two trip mode) in all cases for protection during startup and power operation. The startup rate trip channel is one-out-of-two and the nuclear overpower trip channel is two-out-of-four. In many cases the coincident trip philosophy (2 of 3 or 2 of 4) is carried out to provide a safe and reliable system such that failure or removal from service of a single component or circuit will not defeat the function of the channel and will not cause a spurious plant trip. This design also provides the capability for channel calibration and test at any time. Channel independence is carried throughout the system extending from the sensor through the relays providing redundant coincident logic for actuation of the reactor trip breakers.

Two reactor trip breakers are provided to interrupt power to the rod drive mechanisms. The breakers are located in series with the rod drive mechanism coils and each breaker has a main contact in both the positive and negative lines feeding the mechanism coils. Opening either breaker interrupts DC power to all mechanisms causing them to release all rods to fall by gravity into the core. Each breaker is actuated separately by a shunt trip coil and an undervoltage trip coil. A two-out-of-three protection channel feeds three relays wired to provide three two-out-of-three relay matrices, one matrix for the shunt trip coil of each breaker and one matrix for the parallel operation of the undervoltage trip coils of the two breakers.

Each protection channel in service at power is capable of being checked and its trip points calibrated independently by simulated test signals to verify its operation. This includes checking through the final relay which forms a part of the two-out-of-three logic. Thus the operability of each trip channel can be determined without ambiguity.

There are no stringent requirements for detection of spatial variations during accident conditions since the control rods are not normally moved individually but rather in prewired groups. Should an individual rod move, its maximum reactivity insertion rate is well within the capability of the control system. In the event of the hypothetical rod ejection, the flux increase is rapid enough in all quadrants of the core to be sensed on all out of core channels and effect an immediate reactor trip. (See Criterion 7.)

The initiation signal for the safety injection system provided for loss-of-coolant accidents is accomplished from redundant signals derived from reactor coolant system instrumentation. Each of three pressurizer pressure instruments and each of three pressurizer water level instruments sends a signal to relay matrices which develop actuation signals when two-out-of-three low pressure signals are received in coincidence with two-out-of-three low water level signals. Channel independence is carried throughout the system from the sensors to the signal output relays including the power supplies for the channels. The coincidence circuit allows checking of the operability and calibration of each channel at any time.

The signal for containment isolation; i.e., the isolation valves trip signal is derived from a coincidence of two-out-of-three containment high pressure signals. For this circuit also, the channels are independent from sensor to output relay. Calibration and operability checks may be performed on the individual channels at any time since a two-out-of-three coincidence circuit is provided for containment isolation.

The initiation signal for the containment air recirculation filtration is accomplished on a high containment pressure signal and initiation of containment spray is accomplished from coincident high containment pressure and safety injection signals.

Automatic starting of the emergency diesel-generators is initiated by an undervoltage relay on the 480 volt bus to which the diesel-generator is to be connected. Engine cranking is accomplished by a stored energy system supplied solely for the associated diesel-generator. The undervoltage relay scheme will de-energize to actuate so that loss of 480 volt power will not prevent the relay scheme from functioning properly.

Redundancy is provided in that there are three diesel-generator sets capable of supplying separate 480 volt buses. One complete set of safeguards equipment is supplied from any two out of three diesel generators.

In the event that a diesel-generator fails to start, the 480 volt bus that it supplies is automatically tied to a bus energized by a running diesel. This would then allow a duplicate safeguards component from the bus associated with a failed diesel-generator to be fed from the energized bus. In the event of a bus fault on either bus, closing of the tie breaker will be blocked.

"On the line" testing of the diesel-generator starting scheme will be possible by opening the potential transformer circuit for each undervoltage relay scheme. The generator breaker will not be closed automatically after starting unless there is a coincident requirement for safeguards equipment operation. Complete "on the line" testing of the starting of either diesel-generator could be accomplished by tripping the associated 480 volt bus supply breaker. Blocking the closing of the diesel generator breaker will cause the bus tie breaker to close after a time delay sufficient for normal diesel-generator starting on the other bus.

CRITERION 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can interfere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

In general, each reactor trip channel is designed so that trip occurs when the circuit is de-energized; an open circuit or loss of channel power therefore would cause the system to go into its safety state. Reliability is obtained by redundancy. In a two-out-of-three circuit, for example, the three channels are equipped with separate primary sensors. Failure to de-energize or failure of a de-energized relay to drop out when required would be a mode of malfunction that would affect only one channel; the trip signal furnished by the two remaining channels would be unimpaired in this event.

Control rod cluster insertion is itself a fail-safe function. Reactor trip is implemented by interrupting DC power to the magnetic latch mechanisms on each drive, allowing the rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of DC power. The initiation signal for the engineered safeguards systems is developed from a two-out-of-three coincidence circuit. With the two-out-of-three coincidence circuit, failure of one channel to operate when required will not negate the safety action.

The signal for containment isolation is also developed from a two-out-of-three circuit in which each channel is separate and independent and which signals for containment isolation upon loss of power. The failure of any channel to de-energize when required will not interfere with the proper functioning of the isolation circuit.

Automatic starting of the three emergency diesel-generators is initiated by an undervoltage relay on the 480 volt bus to which the diesel-generator is to be connected. The undervoltage scheme will de-energize to actuate so in that loss of 480 volt power will not prevent the relay scheme from functioning properly.

The components of the reactor protection and safeguards systems are designed and laid out so that the environment accompanying any emergency situation in which the components are required to function will not interfere with that function.

CRITERION 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

The containment structure which houses the entire primary system (and any components or systems exterior to the containment structure which are in effect part of the containment system) will be designed to withstand pressure loads and temperature gradients resulting from the most severe loss of coolant accident, acting in combination with dead loads and maximum seismic loads. (Loads resulting from the maximum wind forces characteristic of the site are substituted for the seismic load in the design of any member where the wind load would be limiting).

The detail of the methods used in deriving the component loads is given in Chapter 5.0 of the Preliminary Safety Analysis Report. The following general criteria are followed to assure conservatism in computing the required structural load capacity:

1. In calculating the containment pressure, rupture sizes up to and including a double-ended main loop severance are considered. The highest pressure rise in the containment from any of these ruptures is greater than that calculated for an instantaneous release of the fluid contents of the reactor coolant system.

2. In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated, including inability to actively quench the zirconium-water reaction and to limit core meltdown. Contingent mechanical or electrical failures are assumed to disable one of the five fan-cooler units and one of the two containment spray pumps. The remaining means of containment heat removal are capable of simultaneous operation on power supplied by two of the three on-site emergency generators.

3. The pressure and temperature loadings obtained by analyzing these cases, when combined with operating loads and maximum seismic forces, do not exceed the load-carrying capacity of the structure, its access openings or penetrations.

Specific results obtained in these analyses are summarized below:

Case 1. Theoretical instantaneous release of reactor coolant into the free containment volume with thermal equilibrium and conservation of internal energy in the liquid and vapor phases.

Result: Calculated containment pressure is 38 psig.

Case 2. Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by normal functioning of safety injection, containment spray and fan coolers. Hydrogen burns as it is liberated.

Result: Containment pressure rises to 40 psig during the blowdown period of about 12 seconds. The pressure remains nearly constant at the peak value, decreasing slightly, until 290 seconds. Containment pressure continuously decreases thereafter.

Case 3. Same as Case 2, except that only the engineered safeguards operate which can run simultaneously with power from two of the three emergency on-site diesel generators. This includes one high head and one low head safety injection pump, four of five fan-cooler units, and one containment spray pump.

Result: In this case a secondary pressure peak occurs at approximately 420 seconds and is no higher than the first peak of 40 psig.

Case 4. Same as Case 3, except that safety injection is delayed indefinitely; i.e., no credit is taken for active quenching of the zirconium-water reaction. Dissipation of core heat occurs by natural means only. The only safeguards that are assumed to operate are four of five fan-cooler units plus one containment spray pump on power supplied by two of the three on-site emergency generators. Hydrogen is assumed to burn as it is generated in the zirconium-water reaction.

Result: Containment pressure rises to 40 psig during the blowdown period of about 12 seconds. A second pressure peak of 35.8 psig is reached in 440 seconds. This pressure peak is caused by the boil off of all the water remaining in the vessel below the core due to the slumping of a molten portion of the core. After boiling this water and melting the vessel bottom heatd 37.5 minutes after the rupture, all of the core material and other hot metal falls into the spilled coolant in the containment sump and is cooled to saturated containment conditions. The boil off of sump water causes a final pressure peak of 43.4 psig. The total zirconium reacted on a very conservative basis for this case is about 44 per cent of the total core cladding mass.

CRITERION 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e.g., water and power systems) to such engineered safeguards must also be redundant.

To assure integrity of the containment following the hypothetical loss-of-coolant accident with no active quenching systems (safety injection), any four of the five installed fan cooler units must be placed in operation for long term removal of residual heat. The heat sink for the fan coolers is river water. Operation of the service water system will provide sufficient cooling water for the four fan cooler units. The service water pumps are located at the circulating water intake structure and take suction directly from the river and pump water to the containment fan coolers.

The containment spray system is an independent backup to the fan cooler units. The heat removal capacity of one spray pump is equivalent to four fan cooler units. The containment spray pumps take suction from the refueling water storage tank. Before exhaustion of the refueling water storage tank, the spray water will be provided by recirculating water from the containment sump through the residual heat removal heat exchangers. Cooling for the residual heat exchangers is provided by the component cooling system which in turn is cooled by the service water system.

Electrical power for the fan motors, the service water pumps and spray pumps is provided from the normal 480 V station outside auxiliary supply. If outside auxiliary power is not available, three on-site diesel engine-generator units supply power. Any two of the three engine-generator sets will power four fan motors and one service water pump in addition to the active quenching systems consisting of one safety injection pump, one residual heat removal pump and one containment spray pump. The emergency bus electrical power arrangement power arrangement and logic network permits failure of one engine-generator unit with coincident failure of any engineered safeguards load on the bus supplied by the active diesels. For example, if 480 volt buses A, B and C are supplied by the diesels and if the diesel on bus A fails to start and a residual heat removal pump on bus B fails to start, automatic transfer is accomplished so that the residual heat removal pump on bus A will be powered by either of the diesels supplying bus B or C.

CRITERION 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

The design leak rate of the containment is 0.1 per cent of the contained volume in 24 hours at 47 psig. With good quality control during erection, this is a reasonable requirement. With a containment leak rate of 0.1 per cent per day and four fan-filter units operating, the off-site exposures to the public will be a factor of two to three below 10 CFR 100 limits for the hypothetical loss-of-coolant accident with no credit for the Safety Injection System in limiting core meltdown.

The basis of the leak rate test is the reference volume method. In addition to the usual calculation of leak rate as a function of pressure differential, air is returned to the reactor containment at the conclusion of the test through a precision gas meter until the differential pressure is returned to its original condition. This provides a check on the calculated leak rate. Reactor containment ambient temperature and humidity are also measured during the course of the test to provide further backup information.

The initial leak rate test consists of establishing the leak rate at 47 psig and at one other lower pressure. Because the containment is a thick-walled concrete structure, short term temperature or meteorological variations should not have any appreciable effect on the containment ambient temperature and pressure. It should, therefore, be possible to establish meaningful leak rates in a shorter term test than might be required in a bare steel vessel. The containment will be held at each test pressure for a minimum of 24 hours.

A leak rate test at any suitable pressure up to the design pressure using the same method as the initial leak rate test described above can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

CRITERION 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leaktightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

A permanently piped monitoring system will be provided such that all penetrations are checked continuously for leaktight integrity during plant operation.

Penetrations are designed with double seals and are continously pressurized during plant operation to prevent outleakage in the event of a loss-of-coolant accident. The large access openings such as the equipment hatch and personnel air locks are equipped with double gasket seals with the space between the gaskets connected to the pressurized system. The system utilizes a supply of clean, dry, compressed air which places all the penetrations under an internal pressure slightly above the containment design pressure. The plant air supply has backup supplies of nitrogen gas capable of 24 hours of service.

Leakage from the pressurized penetrations is checked by continuous measurement of the integrated makeup air flow. In the event that excessive leakage is discovered, each penetration can then be checked separately at any time.

CRITERION 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

The Indian Point Unit #2 is supplied with normal, standby and emergency power. There are available four separate and independent sources as follows:

1. Normal source of auxiliary power during plant operation is the generator. Power is supplied via an auxiliary transformer that is connected to the main leads of the generator.

2. Standby power required during plant startup, shutdown and after reactor trip is supplied from Consolidated Edison's 138 KV system.

3. Emergency power will be available from three diesel-generator sets.

4. An emergency supply for vital instruments and control will be from the station 125 V dc batteries.

The Indian Point 138/6.9 KV station startup transformer will be supplied from the 138 KV bus at Buchanan substation. Buchanan has connections to Indian Point No. 1 generator, the Lovett station of the Orange and Rockland system and the Consolidated Edison 138 KV transmission system via two overhead lines to Millwood East.

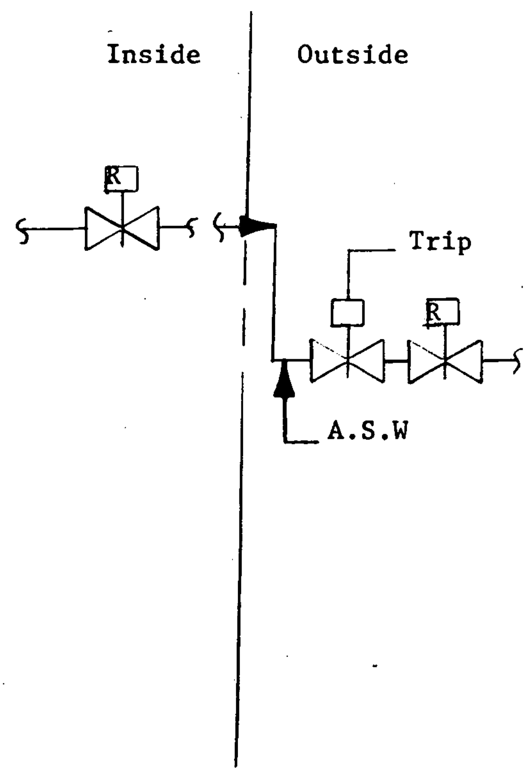
The diesel-generator sets will be located on the plant site and will be connected to separate 480 volt auxiliary system buses. Each set will be started automatically and placed on the line upon the loss of the supply to all the 480 volt auxiliary buses. The diesels will also be capable of manual starting from the control room for periodic test purposes. Two diesels are adequate to supply the engineered safeguards equipment for the hypothetical loss-of-coolant accident concurrent with loss of outside power. This capacity is adequate to provide a safe and orderly plant shutdown in the event of loss of outside electrical power.

CRITERION 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occurred and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent. Containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

Piping penetrating the containment is designed for pressures at least equal to the containment design pressure. Containment isolation valves are provided as necessary in lines penetrating the containment to prevent release of radioactivity. The six classes of penetrations listed below describe the ways of providing at least two barriers between the containment atmosphere and the environs outside the containment, and to preventing the escape of the isolation valve seal water through the line, away from the containment. This design is such that failure of one valve to close will not prevent isolation. No manual operation is required for immediate isolation of the containment.

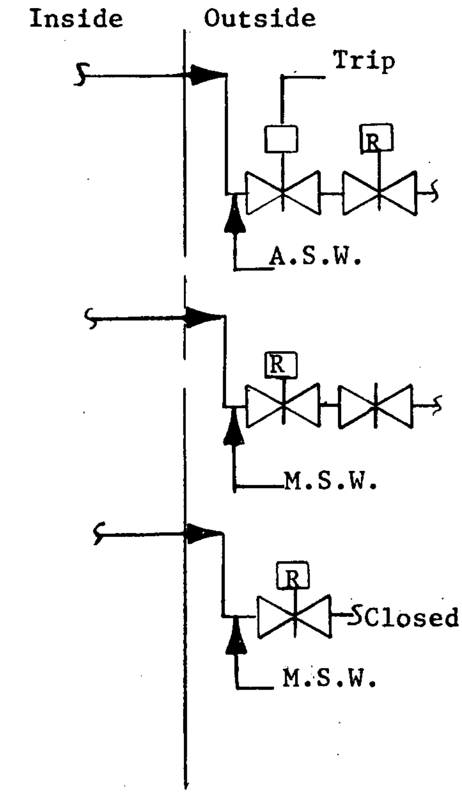
Class 1 (Outgoing Lines, Reactor Coolant System)



Normally operating outgoing lines connected to the reactor coolant system are provided with at least one automatically operated trip valve and one remote operated isolation valve in series located outside the containment. In addition to the isolation valves, each line connected to the Reactor Coolant System is provided with a remote operated root valve located near its connection to the Reactor Coolant System. This class of penetration applies to lines with automatic seal water (ASW) injection.

Class 2 (Outgoing Lines)

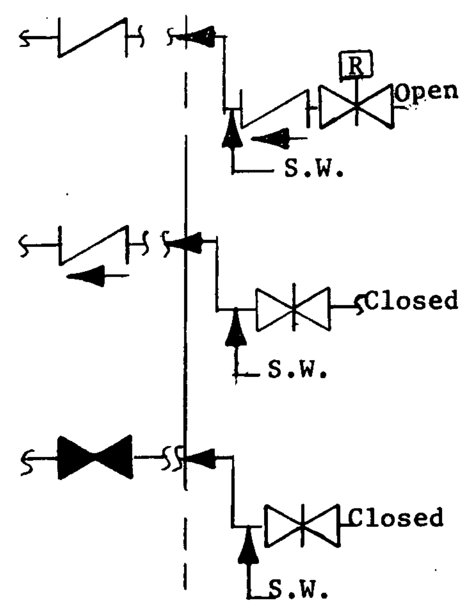
Normally operating outgoing lines not connected to the reactor coolant system and not protected from missiles throughout their length are provided with at least one remotely operated stop valve located outside the containment. All lines with automatic seal water injection (ASW) have a trip valve in series with the remote valve. Open system lines with manual seal water (MSW) injection have a locally operated manual valve as the second isolation barrier. In lines connecting to closed systems, the closed piping system produces the necessary isolation redundancy.



Class 3 (Incoming Lines)

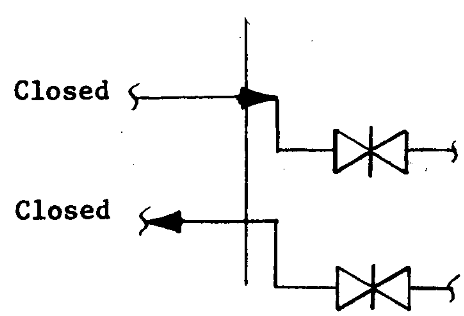
Incoming lines connected to open systems outside the containment are provided with one remote operated valve and two check valves in series, one located inside and one outside the containment.

Incoming lines connected to closed systems outside the containment are provided at a minimum, with one check valve or normally closed isolation valve located inside the containment and one manually operated valve outside the containment. This class of penetration is equally applicable to automatic or manual seal water injection.



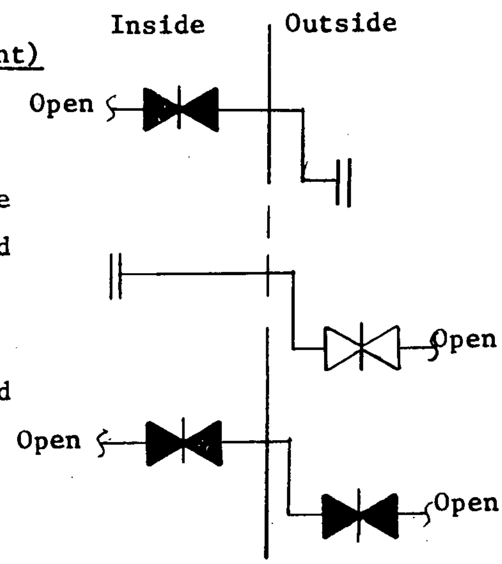
Class 4 (Missle Protected)

Normally operating incoming and outgoing lines which penetrate the containment are connected to closed systems inside the containment and protected from missiles throughout their length,



are provided with at least one manual isolation valve located outside the containment. Seal water injection is not required for the class of penetration.

Class 5 (Normally Closed Lines Open to the Containment)



Lines which penetrate the containment and which can be opened to the containment atmosphere but which are normally closed during reactor operation are provided with two isolation valves in series or one isolation valve and one blind flange. One valve or flange is located inside and the second valve or flange located outside the containment. Gas filled lines will be provided with automatic seal water injection.

Class 6 (Special Service)

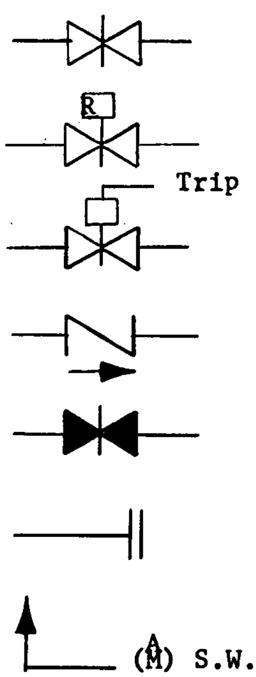
The ventilation purge duct penetrations, the containment access openings at the fuel transfer tube are special cases.

Each ventilation purge duct penetration is provided with two tight-closing butterfly valves which are closed automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration. The space between valves is pressurized by the penetration pressurization system whenever they are closed during plant operation.

The equipment access closure is a bolted, gasketed closure which is sealed during reactor operation. The personnel air locks consist of two doors in sevies with mechanical interlocks to assure that one door is closed at all times. Each airlock door and the equipment closure are provided with double gaskets to permit pressurization between the gaskets by the penetration pressurization system.

The fuel transfer tube is shown in Figure 5-4 in the Preliminary Safety Analysis Report. The penetration inside the containment is designed to present a missle protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the the containment. The penetration closure is treated in a manner similar to the equipment access hatch. The inside closure is a blind flange which contains two gaskets. A positive pressure is maintained between these gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

Symbols



Normally open, local manual

Normally open, remotely operated manual

Normally open, automatically tripped closed

Check

Normally closed, local manual

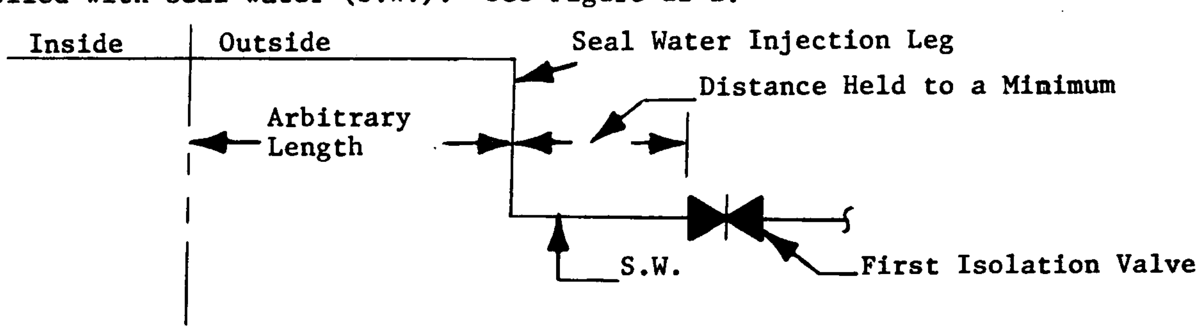
Blind flange

Seal water injection point (Automatic/Manual)

All remote operated and trip isolation valves are provided with control switches and position indication in the main control room. Valves which are designated as automatic trip isolation valves are designed to fail in the closed position upon loss of control air or electric services. The trip valves will be closed automatically upon receipt of the containment isolation signal.

An isolation valve seal water system is used to assure the effectiveness of the containment isolation valves in the event of a loss-of-coolant accident by providing a water seal between the containment and an isolation valve in any line which can communicate with the inside of the containment atmosphere. The seal water system functions after a loss-of-coolant accident to establish a water leg between the potential source of radioactivity in the containment and the closed isolation valve or closed piping system outside the containment. The water leg blocks leakage of the containment atomosphere through valve seats and stem packings. The system is arranged to allow the water leg to be established manually or automatically. Manual seal water injection is provided for long term leakage makeup to all lines which penetrate the containment except those that cannot communicate with the containment atmosphere. Automatic seal water injection is provided for piping that communicates with containment atmosphere and can be void of water in the event of a loss-of-coolant accident. The lines without the automatic injection feature are those which will have a water leg established by virture of there function or operation. The containment isolation signal will be derived from redundant channels monitoring containment pressure.

The isolation valves need not be adjacent to the containment but are located adjacent to the vertical seal water leg required on those lines supplied with seal water (S.W.). See Figure 22-1.



Isolation Valve Layout Criteria

Figure 22-1

CRITERION 23

In determining the suitability of a facility for a proposed site the acceptance of the inherent and engineered safety afforded by the systems, materials, and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components and engineered safeguards can be tested and inspected during the life of the plant.

A comprehensive program of plant testing has been formulated for all equipment vital to the functioning of engineered safeguards. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, an integrated test of the system as a whole, and periodic tests of the activation circuitry and mechanical components to assure reliable performance upon demand throughout the plant lifetime.

Initial Performance Tests

The initial tests of individual components and the integrated test of the system as a whole complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry.

For example, pumps will be tested in the manufacturer's shops to establish conformance with design conditions. Also, the filter units will be tested in the manufacturer's shop to determine filter efficiency. The cooling units and demisters will also be tested prior to installation to demonstrate conformance to design conditions.

The initial test of the Safety Injection System will be conducted during hot functional testing of the Reactor Coolant System before the initial plant startup. This test will complement the shop tests of individual components. No attempt will be made to achieve flows approaching the maximum values which were demonstrated for pumps in the shop tests. The purpose of the integrated system test will be to demonstrate proper functioning of instrumentation and actuation circuits, to evaluate the dynamics of placing the system in operation, and to expose all members in the system to pressure conditions representative of those which can be expected for a loss-of-coolant accident. Flow will not be introduced into the Reactor Coolant System or through the containment spray or filter dousing nozzles during this test, but will be established in all parts of the system up to the final remote operated isolation valves in the safety injection, core deluge, and containment spray loops. Flow is maintained in each loop in the following manner: 1) safety injection loop - temporary test piping is installed between the main injection header and the refueling water storage tank, 2) core deluge loop - the minimum flow recirculation line is used for the test, and 3) containment spray loop -permanent test piping is installed between a point upstream of the final isolation valve (this valve must be unlocked and manually closed) to the minimum flow recirculation line. The remote operated valves will be tested separately. Spray and filter dousing nozzle clearance will be verified by introducing air into the spray header.

During this test, Safety Injection System operation will be initiated by the installed instrumentation and controls. Both pressurizer level and pressure will be varied to provide the required coincidence of low level and pressure to initiate injection, or set points of pressure and level bistable units will be varied to produce an automatic injection signal. Containment spray operation is initiated automatically by coincidence of safety injection initiation and high containment pressure signal. Valve operating times, system flows, and system pressures will be measured. In addition, pump acceleration times and associated auxiliary electrical system voltage dips will be measured.

After installation at the site, the diesel-generator sets will be tested for conformance to design requirements and the tests will include the necessary electrical tests to assure that the automatic bus transfer, load sequencing and load transfer operations can be completed as required to place the engineered safeguards in operation from the diesel-generators.

Periodic Testing

The following series of periodic tests and checks provide continued assurance that the systems can perform their design functions whenever they should be called on during the plant lifetime.

1) Integrated Test of Actuation Circuits and Motor-Operated Valves

The automatic actuation circuitry, valves and pump circuit breakers can be checked during integrated system tests performed during each planned cooldown of the Reactor Coolant System for refueling.

The integrated system test can be performed during the late stages of plant cooldown when the residual heat removal loop is in service. This test would not introduce flow into the Reactor Coolant System, but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

During the plant life, a complete integrated test which introduces flow into the reactor vessel can be performed. This test can be performed during either a heatup or cooldown of the Reactor Coolant System. Upon actuation of the safety injection signal, complete flow paths would be aligned, and flow would be introduced into the Reactor Coolant System. Rising water level in the pressurizer will verify flow into the system.

2) Safety Injection, Core Deluge, Containment Spray and Charging Pumps Test

The safety injection and containment spray pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The core deluge pumps are used every time the residual heat removal loops are put into operation. All remote operated valves can be exercised and actuation circuits can be tested periodically during plant operation. As in the initial testing program, an air purge can be employed to check the spray piping and nozzles.

3) Air Recirculation, Cooling and Filtration Unit

The air recirculation and cooling units and the service water pumps that supply the cooling units are in operation on a relatively continuous schedule during plant operation, and no additional periodic test is required. The filters are bypassed during normal operation by bubble-tight butterfly valves. These valves can be periodically tested by actuating the controls and verifying deflection by instruments in the control room. The filter units can be periodically tested in place during shutdown by aerosol injection to determine integrity of the flow path. The filters can be periodically removed and tested to verify their continued efficiency.

4) Boric Acid Concentration in the Injection Lines

The safety injection piping up to the final isolation valve is maintained full of borated water at refueling water concentration while the plant is in operation. This concentration will be checked periodically by sampling. The injection lines will be refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small bypass line and a return line are provided for this purpose in each injection flow path.

5) Sodium Thiosulfate Concentration in the Thiosulfate Tank

The concentration of $Na_2S_2O_3$ solution in this tank will be checked periodically by local sampling. Additional solution can be added through a connection provided for this purpose.

6) Diesel Generators

The three diesel generator sets can be tested at any time during plant operation.

7) Pressurized Penetrations and Weld Channels

The containment penetrations and weld channels are provided with pressure and makeup flow instrumentation which continuously monitor the effectiveness of these features, as described in Question 7.

CRITERION 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

All fuel storage and waste handling facilities are contained in the reactor auxiliary building, fuel storage building and in the waste holdup tank pit. The facilities and equipment are designed to prevent any unmonitored accidental releases of radioactivity directly to the environment.

All spent fuel is stored underwater at low temperature in a pit having reinforced concrete walls lined with stainless steel plate. Water removed from the pit must be pumped out as there are no gravity drains. Spillage or leakage of any liquids from waste handling facilities will be retained within the auxiliary building by accumulation in floor drains which flow into a drain tank or in the waste holdup tank pit which is designed to prevent uncontrolled drain-off of spilled liquid waste.

A controlled ventilation system removes gaseous radioactivity from the atmosphere in fuel storage and waste treating areas and discharges it to the atmosphere via the plant vent. Radiation monitors will be in continual service in these areas to actuate high-activity alarms on the control board annunciator. Radiation monitors will also sample the plant vent gas effluent stream and actuate alarms on the control board annunciator and terminate discharge of ventilating air or waste gases on a high activity signal.

CRITERION 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

During the refueling of the reactor, all operations are carried out with the spent fuel under water which provides visual control of the operation at all times and also maintains low radiation levels (less than 50 mr/hr) throughout the operation. The borated refueling water assures subcriticality at all times and also provides adequate cooling for the spent fuel during transfer. Spent fuel is taken from the reactor and transferred to the refueling canal and placed in the fuel transfer system. RCC transfer from a spent fuel assembly to a new fuel assembly is accomplished prior to transferring the spent fuel to the spent fuel storage pit.

The spent fuel storage pit is provided to permit radioactive decay of spent fuel prior to shipment from the site. The spent fuel pit is designed to accommodate the storage of a total of one and one-thirds cores plus the spent fuel shipping cask. The storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The spent fuel is stored in a vertical array with a 21-inch center-to-center distance between assemblies. This provides 12 inches of water flooded space between assemblies which offers nuclear isolation. The multiplication of each assembly is sufficiently low that the multiplication in the rack is less than 0.9. This arrangement will insure subcriticality even if the pit were filled with unborated water. The storage racks will be designed so that it will be impossible to insert assemblies in other than the prescribed locations. The water level maintained in the pit will provide sufficient shielding to permit normal occupancy of the area by operating personnel. The spent fuel pit is provided with systems to maintain water cleanliness and to indicate pit water level and to provide for heat removal from the pit water.

The reactor cavity, refueling canal and the spent fuel storage pit are reinforced concrete structures lined with seam welded stainless steel plate. These structures will be designed to withstand the anticipated earthquake loadings as Class I structures so that the stainless steel liner should prevent leakage even in the event the reinforced concrete develops cracks. The transfer tube which connects the refueling canal and spent fuel pit forms part of the reactor containment. It is provided with a gate valve in the spent fuel pit and blind flange in the refueling canal which effectively close off the transfer tube when not in use. The flange is double gasketed, and pressure is maintained between the gaskets to monitor the leak-tightness of the seal. The space between the transfer tube and a surrounding liner sleeve is also pressurized to check the leaktightness of the transfer tube system.

New fuel is brought into the containment through the equipment hatch or through the spent fuel pit and fuel transfer tube. The new fuel is stored dry in the new fuel storage area. The fuel is stored vertically in racks designed to prevent criticality even in the event the vault should become flooded. This storage plan provides 12 inches of space between assemblies which offers nuclear isolation. The multiplication of each assembly is sufficiently low that the multiplication in the rack is less than 0.9. The storage area is at an elevation above the top of the spent fuel storage pit and is provided with ample drains to prevent water accumulation.

CRITERION 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate holdup capacity must be provided for retention of gaseous, liquid or solid effluents.

Unfavorable environmental conditions are not expected to place any restrictions on the normal release of operational radioactive effluents to the environment. Radioactive fluids entering the Waste Disposal System are collected in sumps and/or tanks until the course of subsequent treatment is determined by analysis.

Radioactive gases are pumped by one of two compressors through a manifold to one of four storage tanks where they are held a suitable period of time for decay. Three tanks are provided for normal operation - one filling, one in isolation for decay and one being discharged. The fourth is provided to accommodate unexpected plant operations such as cold or hot shutdown which might produce additional waste gases. During normal operation, gases will be discharged at a controlled rate from these tanks through the monitored plant vent.

All radioactive liquid wastes are collected in tanks or sumps prior to processing. Low level liquid wastes can be discharged from the plant through a monitored line into the condenser water canal. The Waste Disposal System is designed so that, if desired, liquid wastes can be processed for reuse in the primary plant. The concentrates from the evaporation process contain most of the radioactive material from the liquid wastes and can be drummed and stored on-site until shipment off-site for permanent disposal. Alternatively, the concentrates can be processed for reuse (in the Chemical and Volume Control System) as concentrated boric acid solution. The condensate from the evaporation process is collected and monitored to determine its ultimate disposition, reuse or discharge to the condenser discharge canal. Two valves must be opened deliberately prior to discharging gases or liquids to the environment.

All solid wastes will be placed in suitable containers and stored on-site until shipment off-site for ultimate disposal.

In all cases it is possible to hold the solid, liquid or gaseous waste within plant confines should circumstances require their retention.

CRITERION 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

Under conditions attending a loss-of-coolant or other reactor accident, monitored systems will prevent release of radioactivity by way of leakage paths through the containment and its penetrating lines. As described in Section 5.1.2.2 of the Preliminary Safety Analysis Report, a system of double penetration seals and liner joint channels, each pressurized above the containment pressure, prevents outward leakage through the containment boundary at those points where defects might be expected. The Isolation Valve Seal Water System, described in Section 5.2.2, augments the leaktightness of penetrating pipelines by providing a water seal in the path of any leakage caused by imperfect isolation. The effectiveness of these leak preventive systems is monitored: pressure in the penetration and liner joint protection zones is indicated in the control room and alarmed if pressure falls below an adequate value. Liquid level in the penetrating lines protected by the Isolation Valve Seal Water System is indicated locally in areas accessible following the accident.

Should an accidental release of radioactivity occur which circumvents these lines of protection, or which originates in a plant system outside the reactor containment, either the permanent area radiation monitors or the plant ventilation or liquid effluent monitors, supplemented by portable survey monitors, if required, would provide indication of the extent and location of the release.

3. Your attention is directed to a letter from the ACRS to the Chairman, AEC, dated November 24, 1965, concerning reactor pressure vessels. Please discuss the consideration which has been given in the design of your facility to the recommendations contained in numbered paragraphs 1 and 2 of the ACRS letter. For your guidance in providing a complete answer to this question the following are some of the detailed areas of concern explored recently by an ACRS subcommittee on the proposed Rochester Gas and Electric Corporation Brookwood facility. Your reply should incorporate answers to these questions.

   a. Please give details on the best prediction of maximum fast neutron flux dose in the pressure vessel, including uncertainties in prediction.

   b. Please give details on the method of measuring NDT for base plate and quantitatively describe uncertainties therein.

   c. Please give details on method of prediction of NDT shift with fast neutron dose and quantitatively describe uncertainties therein, including considerations of weld regions and heat affected zones.

   d. Please describe in detail the stress considerations to be allowable below NDT plus 60 and below NDT. State assumptions and give reasons, allowing for flaw size in particular.

   e. Please give rationale for relationship between NDT and allowed stress emphasizing in particular the degree of conservatism which it is felt the circumstances require, and why.

   f. Define the flow size and type in the pressure vessel which is accepted in the specifications. What flaws larger in size or of special significance might not be detected, particularly in zones of irregular geometry?

   g. What flaw size is accepted in the studs of the pressure vessel? What frequencies of stud inspection or replacement is planned? How many studs can fail without threatening the integrity of the closure?

   h. Please describe requirements concerning the support structure for the pressure vessel, including the degree of levelness over reactor life, which are needed to insure no problems due to local overstressing of the pressure vessel.

   i. Describe how small leaks in the pressure vessel would be detected and the action to be taken, should such occur. How is adequate response assured in the event of a previous existence of small leaks in other parts of the system?

j. Describe the surveillance program for the pressure vessel in some detail. What uncertainties would be expected from the experimental results?

k. Are you considering procedures for detecting the propagation of cracks within the pressure vessel wall, i.e., acoustic emission.?

l. State and justify the energy required to initiate failure of the primary system boundary. Can a control cluster ejection or any other credible mechanism provide this amount of energy by reactivity insertion?

Answer

The considerations given in the design of this facility to the design, fabrication, evaluation, and operation of the reactor vessel are given in the following pages. The criteria and provisions for missile protection for this facility are presented in Qeustion 2, Criterion 3.

The basic modes of possible failure for vessels as covered in "Material Requirements for Long-Life Pressure Vessels" by B. F. Langer (ASME Paper 63-WA-194) are Plastic Deformation and Bursting, Fatigue, Creep, Corrosion, and Brittle Fracture. The general preventative design, inspection and operational controls which are exercised to prevent such failures are as follows:

Bursting -

A design margin of 3 between membrane stresses and bursting stresses is maintained by ASME code.

A hydrostatic test at 1.25 design pressure is applied.

The pressurizer pressure control system and redundant safety valve protection are provided against vessel overpressurization.

Fatigue -

The ASME Code requirements provide design margins of 2 on stress and 20 on cycles, between design and failure conditions.

The supplier performs detailed stress analysis covering all forecast plant operating transients (both normal and emergency).

WAPD stress analysts make independent detailed reviews of these calculations.

Internal and external surfaces, where stresses are highest, are readily accessible for inspection before and during fabrication for the accurate detection and control of flaws by magnaflux and/or liquid penetrate inspection.

Creep -

Materials are used in the vessel which do not present a creep problem at the maximum design temperature of 650°F.

Corrosion -

All surfaces of the vessel in contact with the primary coolant are made of material compatible with the primary coolant.

Control of oxygen and halides is maintained to prevent stress corrosion cracking of austenitic stainless steel.

Ductile failure of the vessel under any of these modes would, of course, be serious. However, excluding bursting, the failure would result initially in a relatively small breach in the vessel with leakage of the primary coolant which would be detected and allow corrective action. In any event, failure would not produce a catastrophic large scale rupture - nor would any missiles be generated. As discussed in NRL report 5920, even deliberate hydrostatic bursting of the vessel would only result in limited ductile failure with no fragmentation.

Small leaks from the primary system will probably occur during the normal course of operation. These are usually from valves and bolted flanges. No method will be provided to determine the source of these leaks while the plant is at power.

Leakage of coolant through the thick-walled reactor pressure vessel is assumed to have a nil probability, considering the design, fabrication and inspection techniques which have been stated and the fact that the vessel material is ductile at operating pressure and temperature. Leakage through the vessel-to-head closure joint is allowed for by installation of double gaskets with leakoff connections. Experience to date has shown little, if any, leakage occurs through the vessel-to-head closure.

The existence of leakage from the reactor coolant system to the containment, regardless of the source of leakage, will be detected by one or more of the following conditions:

a. An increase in the amount of coolant makeup water required to maintain normal level in the pressurizer.

b. An increase in the containment atmosphere radiogas or particulate activity indicated on instrumentation in the control room.

c. An increase in containment sump level which is indicated on instrumentation in the control room.

Periodically during hot shutdown, inspections will be made for significant sources of leakage and necessary corrective action taken. If a significant leak in the reactor coolant system were found, the plant would be taken to the cold shutdown condition and an extensive examination of the problem would be made.

<u>Brittle Fracture</u> -

Brittle fracture is a function of three items; stress, flaw size, and temperature. It is important to remember that these items are always interrelated; must be considered jointly. The relationship between them is graphically expressed by the Fracture Analysis Diagram (Attachment 1) by Pellini in NRL Report 5920. This diagram accounts for the two phases of brittle fracture; crack initiation; crack propagation. The diagram relates the conditions at which cracks will initiate to the conditions which will stop the propagation for these cracks.

For a flaw-free material at about -200 degrees F, the tensile strength equals the yield strength and ductility is nil. This temperature may be considered the NDT in the absence of a flaw. With a small sharp flaw the stress required to initiate a crack equals the yield stress at the NDT temperature (as shown on Attachment 1). As larger flaws are considered, fracture will result at lower stress levels, as established by vessel failure tests conducted by Battelle (Welding Journal, May 1957) and actual service failure data. This is shown in the family of dotted curves on Attachment 1.

Various tests and actual service failures (as detailed in NRL Report 5920) have established the conditions at which the propagation of brittle fractures is stopped. This is represented by the CAT or Crack Arrest Temperature Curve, also shown on Attachment 1. The data show that the crack arrest temperature for loading at the yield stress is NDT plus 60, and at ultimate tensile stress loading, NDT plus 120. The lower shelf of 5000 to 8000 psi shown on Attachment 1 is the point below which the minimum amount of strain energy required for continuous propagation of brittle fracture is not attained. It is apparent that initiation or propagation of brittle failures is not possible in the zone to the right of the Crack Arrest Temperature Curve, even with gross flaws as big as two feet long.

The inspection of vessel material and welds by ultrasonic test, radiography, magnaflux and liquid penetrant is detailed in Attachment 2. In summary these inspections determine not only the size of possible flaws, but the orientation. It should be noted that improvement beyond existing code requirements will be made. Longitudinal wave ultrasonic inspection of plates covers 100% of the volume instead of the 9" grid sampling per Code. In addition, 100% volumetric shear wave UT inspection will be performed in two directions.

These inspections limit the size of laminar type defects to any area equivalent to a 3 inch diameter circle or a circle whose diameter is one half the plate thickness, whichever is larger. Since this type of defect invariably occurs in the center of plates where the stress is not only low, but parallel to the direction of the stress, there is little if any tendency for either fatigue or brittle propagation. The limitation on linear flaws (cracks, laps, seams) is one inch long by 3% of the plate thickness. They are most likely to occur at the surface as a result of heat treating and working of the material in the rolling, forging, forming, fabricating processes. As Attachment 1 shows, this range of flaw size requires general stress levels approaching the yield stress to initiate brittle fracture at temperatures at or below NDT, and stresses approaching the ultimate tensile strength at temperatures up to NDT plus 120F. The maximum design allowable stresses of 2/3 yield is considerably below the levels required to initiate brittle failure.

Initiation of brittle failures must still be evaluated because local areas of higher stress can exist (as permitted by Code). Where local stresses are high enough to exceed the critical stress-flaw size combination, a brittle failure can be started; however, as established in the test data, and actual service failures, it will stop as soon as it reaches the area of lower general stress. Although the service history of thousands of pressure vessels designed, manufactured, and inspected to previous ASME provisions which were less stringent than current Code Requirements would strongly question the point, it is possible that flaw sizes could increase over the life of the plant; therefore, propagation of defects is considered. Here again brittle failures might be postulated at some point where a critical flaw size - general stress level combination was reached at temperatures below NDT. However, as shown by Attachment 1(curve C) plant operation is controlled to assure that operation is always maintained in the area to the right of the Crack Arrest Temperature Curve where brittle failure cannot initiate. This curve represents the minimum pressure (stress) vs temperature relation which must be maintained during the most limiting condition of startup - shutdown of the plant. (It is based on minimum pump operating requirement to assure proper shaft seal operation and to prevent cavitation.) It includes a 60 psi instrument-control deviation on pressure and a 10 degree temperature deviation allowance.

While residual stresses resulting from construction tend to be additive to the pressure stresses, the analysis of actual service failures in NRL 5920 shows that the stress relief required by ASME code effectively reduce these stresses so they do not contribute to failure. The residual stresses are localized, concentrating mainly at the welds, and even if brittle failure did initiate across the weld it would stop in the adjacent area of reduced general stress.

Because the NDT temperature is the anchor point of the CAT curve, this naturally leads to the question of the uncertainties associated with establishing the NDT and the shift caused by fast neutron irradiation. The vessel wall in the area of the core is limiting and is the basis for the following.

(1) Data Scatter

Test data and actual failure correlations show that the drop weight determination is highly reproducible and has little scatter. It is also insensitive to orientation with respect to material working. The details and uncertainties of NDT tests are discussed in Attachment 3. The location of the test specimen must also be considered. Specimens are taken at 1/4 of plate thickness (1/4 T). Data show that in thick plate an increase in NDT as high as 85 degrees may exist from 1/4 T to center properties. Curve A on Attachment 1 shows this allowance.

(2) Irradiation

The shift due to fast neutron exposure is calculated as 160°F at the inner surface of the wall at $.85 \times 10^{19}$ nvt. At the center (where impact properties are lowest as shown above) the shift is 85°F. Curve B of Attachment 1 shows this shift. The basis and uncertainties of determining the shift by calculation is given in Attachment 4. Actual NDT shift will be monitored during plant life by test of samples of the vessel material fixed inside the vessel opposite the core center. The details of this surveillance program are given in Attachment 5.

It should be noted that in addition to the tensile and Charpy V-notch specimens wedge opening loading (WOL) type specimens are included. These specimens are part of the effort to provide additional information for interpreting the irradiated vessel material properties which can be used in applying fracture mechanics approach to acceptable stress-flaw combinations.

As attachment 1 shows, even a total NDT shift of 170°F (85° due to reduced impact properties at plate center and 85°F for irradiation) leaves a margin of well over 200°F between limiting plant operating conditions (Curve C) and the nearest point on the Crack Arrest Curve, which is the adjusted NDT temperature. Therefore, brittle fracture will be prevented by control of the stress-flaw size-temperature interrelation during plant life.

The vessel closure studs are not primary pressure boundary materials, but are essentially equivalent. The control of flaw size in the studs is included in Attachment 2, which shows that the maximum flaw size expected is 1/2". The simultaneous failure of all 54 studs is not considered credible, particularly in view of the regular inspections to be performed on them (visual inspection of all studs at each refueling and magnaflux inspection of 1/2). Fourteen of the studs distributed in approximate symmetry around the flange are sufficient to withstand the full hydrostatic end load without exceeding yield strength. In addition, if it is postulated that one adjacent stud after another fails, it is estimated that after six have failed, the flange will start to leak in the unclamped segment leading to detection and correction. Some 13 studs would have to fail in "zipper" fashion to result in failure of the remaining studs. This is not considered a credible condition.

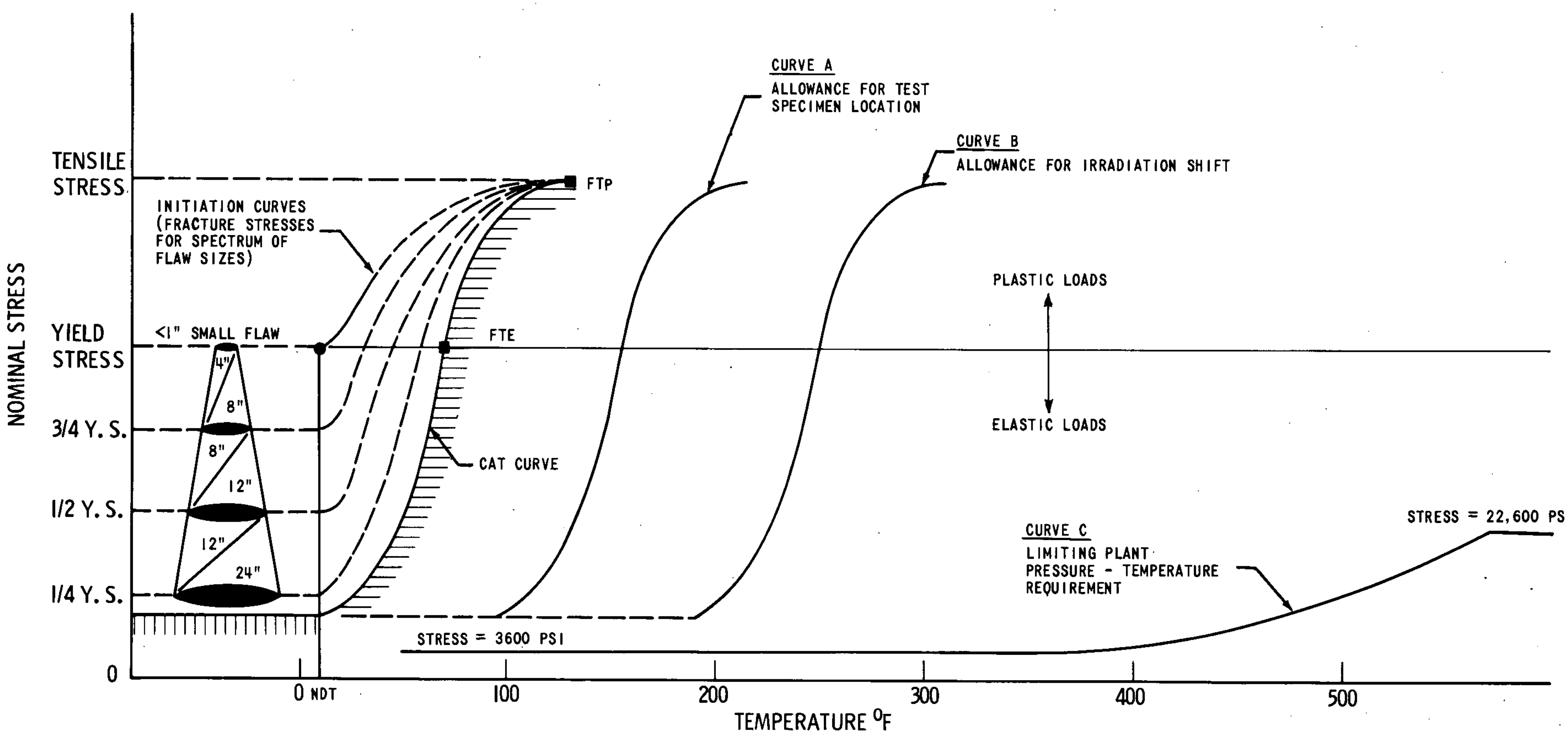
The aspect of vessel support design needed to insure no problems due to the loads they impose on the vessel have been considered. The requirements for the vessel supports are developed considering static and dynamic loads, earthquake loading factors and piping reactions. Although a levelness of 0.0005 inches per foot of flange diameter is held to facilitate proper assembly and operation of internals, changes in levelness will not produce local overstressing in the vessel. Should any gross change in foundation occur, local yielding of the connecting piping might result until the support loading redistributes.

In summary, it is considered that improvements are being provided beyond current ASME design, inspection and irradiation surveillance requirement to secure optimum stress analysis, minimize flaw size and propagation and monitor NDT over plant life.

The energy required to initiate failure of the primary system is very much a function of the manner in which energy is dissipated e.g., as a sharp impulse as could occur with a pressure shock or as a relatively sustained pressure occurring uniformly in the system. The control rod ejection accident is the only means whereby the potential exists for large and rapid energy release in the reactor coolant system. An energy release sufficiently rapid to jeopardize the integrity of the primary system boundary further requires a rapid dispersal of molten UO_2 in the coolant since the inherent heat transfer resistance at the cladding surface while intact precludes a rapid enough heat addition to the coolant to create any potential shock wave generation.

Since the rod ejection accident results in very localized peaking effects, the number of affected fuel rods is small. Also, the operation of pressurized water reactors using chemical shem is such that the consequences of a rod ejection are limited since the amount of control rod cluster insertion at power throughout core life is minimized. The reactor will be designed and operated so that this accident will not cause further failure of the primary system.

A dynamic and static overpressure analysis, performed for a typical pressurized water reactor shows that even if a considerable amount of fuel were dispersed to the coolant, stresses in the reactor coolant system components would not exceed yield. Specifically, UO_2 dispersal from 25 fuel rods resulted in a maximum stress increase less than 10 per cent of the margin to yield. This result is considered very conservative because it is based on a high density of 10 mil fuel particles in the water, no energy dissipation in the axial direction, no energy dissipation in the core or internal structures, no initial voids in the core, and a doubling effect on overpressure due to wave reflection. Thus it appears that a considerably greater amount of fuel could be dispersed in the manner assumed without reaching yield in any of the reactor coolant system components.



FRACTURE ANALYSIS DIAGRAM

ATTACHMENT I - QUESTION 3

ATTACHMENT 2

INSPECTION REQUIREMENTS AND STANDARDS

The detection of flaws during fabrication of the reactor vessel will be accomplished by the following non-destructive testing techniques.

1. Radiographic Examination - all pressure containing welds.

2. Ultrasonic Examination - all plates, forgings, closure studs, pipes and tubes.

3. Magnetic Particle Examination - all surfaces to be clad or welded. All unclad surfaces and welds after hydro test. Closure stud surfaces before and after threading.

4. Liquid Penetrant Examination - all cladding after final stress relief. All weld overlay in weld regions after stress relief.

The acceptance standards for all above inspections are based upon Section III of the ASME Code as follows:

In radiographic examination of welds, acceptable defect sizes are as follows:

A. Parallel to the external surface:
   1. Slag inclusions no longer than 3/4 in.
   2. Porosity not exceeding 0.240 sq. in. total area in any 6 in. of weld length and located within the space limits given in Appendix IV of Section III.

B. Perpendicular to the external surface:
   1. Defects not greater in dimension than 2% of the wall thickness up to 6 inches and 1% of the wall thickness for those exceeding 6 inches.

Cracks, lack of weld fusion or lack of weld penetration will not be tolerated.

In ultrasonic examinations, acceptable defect sizes are as follows:

A. Plate - Longitudinal Wave
   1. A defect in which the area does not exceed a 3 inch diameter circle or a circle whose diameter is one half the plate thickness, whichever is larger.

(Revised 6-1-66)

B. Plate - Shear Wave

1. Any defect which does not exceed a depth of 3% of the plate thickness by 1 in. long.

C. Forgings - Longitudinal Wave

1. Any defect which does not cause a loss of back reflection greater than the loss of back reflection from a 3/4 inch diameter flat-bottomed hole for sections 4 to 9 inches thick and a 1 inch diameter hole for section 9 to 16 inches thick.

D. Forgings - Shear Wave

1. Any defect which does not cause a loss of back reflection greater than the loss of back reflection from a notch 3% of the plate thickness in depth but not to exceed 3/8 in. by 1 in. long.

In Magnetic Particle and Liquid Penetrant Inspections, acceptable defect sizes are as follows:

Acceptable defects are linear inclusions not over 3/4 in. in length and non-linear defects are exceeding 3/32 in. in any dimensions. No cracks are permitted.

It is not expected that flaws exceeding those described above will be encountered in vessels manufactured utilizing the non-destructive techniques described above which will be used in the Indian Point vessel as described above.

Procedures for detecting defects in the pressure vessel continuously or periodically during plant life have been investigated. No equipment has been found available with demonstrated capability to monitor the configurations, materials and thicknesses represented in the vessel.

The flaw size on the stud forgings will be determined by two ultrasonic inspections. A radial longitudinal beam inspection will be performed. The rejection standard will be 100% loss of back reflection greater than that from a 1/2 inch diameter flat bottom hole or an indication in excess of 20% of the adjusted back reflection. A radial inspection will be made using the angle beam technique. This inspection will carry the same rejection standards as for forgings.

A wet Magnaglow inspection will be performed on the finished studs. Axial indications revealed by the wet Magnaglow inspection with a depth greater than the thread depth and non-axial defects are unacceptable.

ATTACHMENT 3

DETERMINATION OF NIL DUCTILITY TEMPERATURE

The measurement of the nil ductility transition temperature (NDTT) of the base plate material is based on two separate testing techniques, namely, the drop weight test per ASTM E208 and the notched bar impact tests using Charpy V-notch impact specimens (Type A) per ASTM E23. The drop weight test gives a break or no-break temperature for the NDTT and is defined in ASTM E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no-break performance occurs at a 10°F higher temperature." The NDTT as determined using the Charpy V-notch tests is defined as the temperature at which the energy required to break the specimen is 30 ft-lbs. This 30 ft-lb "fix" is based on a correlation with drop weight tests as stated in ASME Section III, Table N-332. A curve of energy versus temperature is plotted as a means through the data. The intersection of this curve with a 30 ft-lb ordinate is defined to be NDTT. At least 15 test results, which include three tests at five different temperatures, are done to provide the data.

Regarding the use of 2" x 4" x 5/8" specimens to measure NDTT, evaluation of service failures (references in ASTM E208) has shown that these failures correlate with NDTT established by these specimens.

The test material for these tests are obtained from each plate or ring forging used in lieu of plate used in the reactor vessel. The thermal history of the test material is representative of the bar material in the final vessel condition. All tests are taken at a distance of 1/4 T (1/4 thickness) from the quenched surfaces and at a distance T from the quenched edges. The tests described above are performed by the vessel fabricator and by Westinghouse as a part of the reactor vessel surveillance program. Additional tests are also done by the fabricator. The specific test specimen locations for tests are reviewed and approved by WAPD to assure compliance with specifications. Similarly, Charpy V-impact tests are performed on the base metal, weld metal, and heat-affected zone material of the vessel test plates.

As part of the Westinghouse surveillance program, Charpy V-impact tests, tensile tests, and fracture mechanics specimens are taken from the core region plates, and core region weldments including heat-affected zone material.

The test locations are similar to those used in the tests by the fabricator.

The uncertainties of measurement of the NDTT of base plate are:

1. Differences in Charpy V-notch foot pound values at a given temperature between specimens.

2. Variation of impact properties through plate thickness.

The fracture toughness technology for pressure vessels and correlation with service failures based on Charpy V-notch impact data are based on the averaging of data. The Charpy V-notch 30 ft-lb "fix" temperature is based on multiple tests at the fabricator and by Westinghouse as part of the surveillance program. The average of sets of three specimens at each test temperature is used in determining each of five data points (total of 15 specimens). In the review of available data, differences of 0°F to approximately 40°F have been observed in comparing curves plotted through the minimum and average values respectively. The value of NDTT derived from the average curve is judged to be representative of the material because of the averaging of at least 15 data points, consistent with the specified procedures of ASTM E23. In the case of the assessment of NDTT shift due to fast neutron flux, the displacement of transition curves is measured. The selection of maximum, minimum or average curves for this assessment is not significant since like curves would be used.

There are quantitative differences between the NDTT measurements at the surface, 1/4 thickness or the center of a plate. Differences in NDTT between 1/4 T and the center in heavy plates have been observed to vary from improvement in the NDTT to increases up to 85°F. The NDTT at the surface has been measured to be as much as 85°F lower than at 1/4 T.

The 1/4 T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDTT. In addition, the limiting NDTT for the reactor vessel after operation will be based on the NDTT shift due to irradiation. Since the fast neutron dose is highest at the inner surface, it is considered that using the 1/4 T NDTT criterion is conservative.

Data is being accumulated on the variation of NDTT across heavy section steels at WAPD. Similarly, the Pressure Vessel Research Committee is sponsoring an evaluation of properties of pressure vessel steels in plates 6 to 12 inches thick. Preliminary data has shown NDTT differences between 1/4 T and center of less than 20°F. The present criteria of NDTT +60°F at the 1/4 T location without taking advantage of the enhanced properties at the surfaces of reactor vessel plates is considered conservative.

ATTACHMENT 4

PREDICTION OF FAST NEUTRON FLUX AND NDT SHIFT

FAST NEUTRON FLUX CALCULATION

The maximum time-integrated fast neutron flux (E 1 Mev) incident of the Indian Point Unit #2 pressure vessel was calculated to be .85 x 10^{19} n/cm^2 and was obtained using a corrected P1MG one-dimensional 55 group diffusion code. Because of the conservative methods and assumptions employed in the calculation the actual exposure is expected to be lower than the calculated value by as much as a factor of 1.7.

The correction factors applied to the P1MG include the following:

1. F_a: An axial peaking factor. For the Indian Point plant, this factor was set equal to 1.5 although a time averaged value of 1.4 is expected.
2. F_c: A corner effect factor which accounts for the irregular shape of the core. This factor was set equal to 1.25.
3. F_s: A spatial correction factor. For the Indian Point Unit #2 this factor was calculated to be 2.7 and was obtained by comparison of P1MG results with experimental fast neutron data in all water medium because of the lack of sufficient experimental data on neutron energy spectra in non-hydrogenous media at this time. The spatial correction factor determined as described above results in a more conservative value than for other methods presently available. For example, experimental results indicate that using experimental thermal neutrons fluxes results in a correction factor which is approximately 40% lower than that obtained using fast neutron data.

A further conservatism results from the assumption in the calculation that all correction factors occur at a given location on the reactor vessel for the entire life of the plant. With the combination of conservative methods and assumptions the calculation is expected to over predict the neutron exposure by as much as 1.7.

The Pertinent Indian Point plant design parameters used are:

1. Core thermal power — 2758 MW
2. Plant design life — 40 years
3. Load factor — 0.8
4. Active core height — 12 ft.
5. Effective core diameter — 133.7 in.
6. Pressure vessel I.D. — 173.0 in.

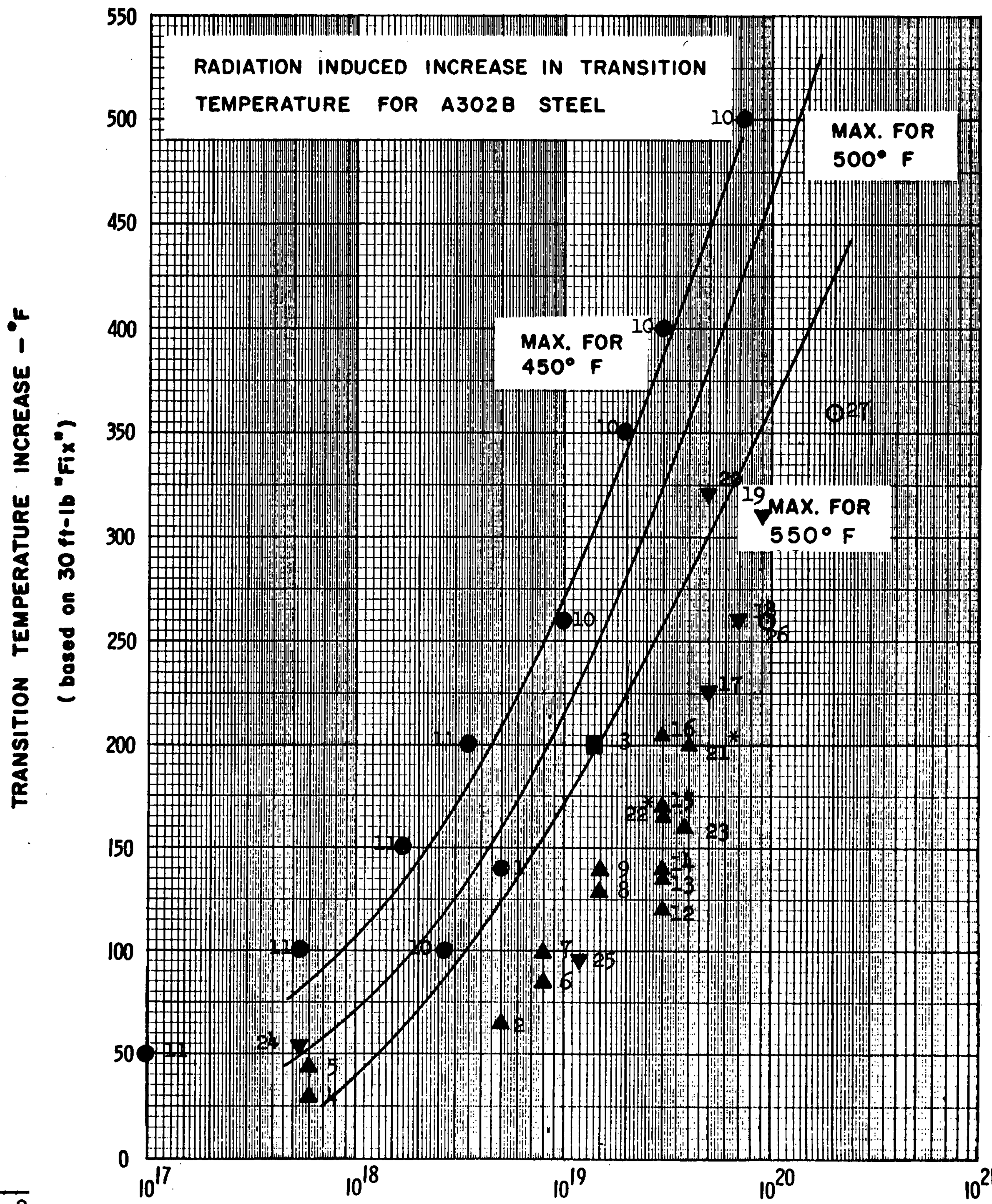
In addition to the conservatism in predicting the integrated fast neutron exposure of $.85 \times 10^{19} n/cm^2$, an exposure of $3.7 \times 10^{19} n/cm^2$ can be tolerated before the design plant heatup and cooldown limit are reached and an exposure of 7×10^{19} can be tolerated before significant operating restrictions would have to be imposed.

PREDICTION OF NDT SHIFT

The reference design curves used for the prediction of NDT shift with fast neutron exposure are presented by Figure 3-1. For the Indian Point Unit #2 reactor vessel, the 550°F maximum curve is being used because the vessel is exposed to inlet coolant of about 543°F. The radiation effects data at 550°F show that less NDTT shift is expected than the design value predicted by the 550°F curve.

The irradiation data on weld metal and heat affected zone metal have shown similar NDTT shifts versus fast neutron exposure. However, most of the data accumulated has been on base metal materials. To assess any possible uncertainties in the consideration of NDTT shift for welds and heat affected zone as well as base metal test specimens of these three "material types" have been included in the reactor vessel surveillance program.

Using a best fit curve for the 550°F data points a margin of about $2.0 \times 10^{19} n/cm^2$ above the assumed design point of $.85 \times 10^{19} n/cm^2$ would be obtained with a corresponding NDT shift of 160°F at the inner surface of the vessel.



Code

Temp. °F	
●	450
■	490
▲	550
▼	475 to 540
⊙	600

Numbers 1 through 27 (see attached sheets)

References for

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR A302B STEEL

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (> 1 Mev)	NDT °F
1.	NRL Report 6160 Page 12	SA302B	450	5×10^{18}	140
2.	NRL Report 6160 Page 12	SA302B	550	5×10^{18}	65
3.	NRL Report 6160 Page 13	SA302B	490	1.4×10^{19}	200
4.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	30**
5.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	45
6.	ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	85**
7.	ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	100
8.	ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	130**
9.	ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	140
10.	NRL Report 6160 Page 6	All Steels	<450	Various	Various
11.	Nuclear Science & Engineering 19:18-38 (1964)	SA302B	<450	Various	Various
12.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	120
13.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	135

**Transverse Specimens

E.L.

References (Continued)

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (> 1 Mev)	NDT °F
14.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	140
15.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	170
16.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	205
17.	NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	225
18.	NRL Report 6179 Page 9	SA302B	475-540	7×10^{19}	260
19.	NRL Report 6179 Page 9	SA302B	475-540	9×10^{19}	310
20.	NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	320
21.	NRL Report 6160 Page 15	SA302B	540*	4×10^{19}	200
22.	NRL Report 6160 Page 15	SA302B	540*	3×10^{19}	165
23.	Private Communication with NRL	SA302B	550	3.8×10^{19}	160
24.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	≈525	5.4×10^{18}	54

E.L.

References (Continued)

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (> 1 Mev)	NDT °F
25.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	≈525	1.2×10^{19}	96
26.	"	SA302B	≈600	9.5×10^{19}	260
27.	"	SA302B	≈600	2×10^{20}	360

E.L.

ATTACHMENT 5

In the surveillance programs, the evaluation of the radiation damage is based on pre- and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors".

The Indian Point Unit #2 reactor vessel surveillance programs utilize eight specimen capsules which are located about 3 inches from the vessel wall directly opposite the center portion of the core.

The capsules can be removed and replaced when the vessel head is removed. The capsules contain reactor vessel steel specimens from the shell plates located in the core region of the reactor and associated weld metal and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM Committee E10 Radioisotopes and Radiation Effects are inserted in the capsules. The eight capsules will contain approximately 64 tensile specimens, 384 Charpy V-notch specimens (which will include weld metal and heat affected zone material) and 96 WOL specimens. Dosimeters including pure Ni, Al-Co, (0.15%), Cd shielded Al-Cl, Cd shielded Np-237 and Cd shielded U-238 are placed in impact specimens, tensile specimens or filler blocks drilled to contain the dosimeters. The dosimeters will permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion.

The tentative schedule for removal of capsules is as follows:

Capsule	Estimated Exposure Time
1	Replacement of 1st region
2	Replacement of 2nd region
3	Replacement of 4th region
4	10 years

5	20 years
6	30 years
7	40 years
8	Extra capsule for complementary testing

Irradiation of the specimens will be higher than the irradiation of the vessel because the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and they are actual samples from the materials used in the vessel, the NDTT from the specimens will be representative of the vessel at a later time in life.

The methods and uncertainties of determining NDTT of the vessel material are discussed in the previous sections. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

4. The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

   a. Relate the available energy sources by showing the total energy that could be provided by (1) the primary coolant, (2) a 100% metal-water reaction, (3) the hydrogen-air reaction, and (4) the core decay heat at 10, 20, and 30 minutes. The relative energy sources should be provided on a percentage basis that totals 100% for each case. What is the total energy available from secondary sources (e.g., steam generators)?

Answer

The total energy which could be provided by each of the following sources is:

1. The primary coolant

   The total available internal energy is 300×10^6 Btu. All of this energy is included in the containment transient analysis.

2. A 100% metal-water reaction

   Reaction of 100% of the Zr mass would produce 113×10^6 Btu. This energy is greater than that which has been conservatively established by the analysis in Section 12.2 of the PSAR as the upper limit for no safety injection. As shown in Figure 4c-1, 45% of the Zr mass could react.

3. The hydrogen-air reaction

   It was assumed that the basis for establishing this energy should be that associated with 100% Zr reaction, to be consistent with part (2) of this question. For this case, 88.2×10^6 Btu would be produced.

4. Core decay heat at 10, 20, and 30 minutes

   Reactor operation for infinite time was assumed in establishing the core decay heat for the containment transient analyses. This is conservative, since finite time operation reduces the decay heat production term. The integrated decay heat values are:

a.	10 minutes	67.1×10^6 Btu
b.	20 minutes	111.5×10^6 Btu
c.	30 minutes	149.5×10^6 Btu

5. Secondary sources

The total energy from secondary sources (such as steam generators) includes

a.	Hot reactor coolant system metal	18.1×10^6 Btu
b.	Steam generator tubes (4 units)	15.75×10^6 Btu
c.	Steam generator secondary water internal energy (4 units)	175×10^6 Btu

The mechanism by which heat from the hot reactor coolant system metal and steam generator tubes is added in the transient analysis is discussed under Question 4b below. Concurrent energy addition from a loss-of-coolant accident and from the four steam generator units is not included in the containment transient analysis because the design provisions for the steam generators and their supports make such an occurrence incredible.

4. The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

   b. Plot in graphical form up through one hour for your assumed model of post-accident conditions, (1) the ratio of decay heat energy in the containment atmosphere to primary coolant energy, (2) the ratio of metal-water energy in the containment atmosphere to primary coolant energy, (3) the ratio of H_2 recombination energy in the containment atmosphere to the primary coolant energy, (4) the ratio of total energy in the containment atmosphere to total available energy, and (5) the ratio of total energy in heat sinks to total available energy.

Answer

The mathematical models for the reactor coolant system blowdown, core Thermal Transient, and containment pressure transient analyses simulate the physical phenonema occurring as a consequence of a hypothetical loss-of-coolant accident. In these models, it is sometines not possible to detail the exact source of specific energy terms. For example, the boil-off of safety injection water as an energy source to the containment is a result of the total core stored energy. It is not possible to state the fraction of the energy entering the containment that is due only to residual heat, or metal-water reaction energy, or initial core stored energy. Furthermore, it is unclear if "containment atmosphere" means only the steam-air phase. The containment analysis treats the steam-air mixture and the containment sump water as two interacting systems. In the overall energy balance, and the containment pressure calculation, both systems must be considered.

The intent of this question is to obtain an understanding of the contribution of various energy sources and sink terms in the containment pressure transient analyses. Therefore, the integrated energy source and sink terms versus time have been provided in graphical form.

The double ended rupture of a reactor coolant pipe is the accident chosen for presentation. It is assumed that one component fails in each engineered safeguards system. The operating safeguards include one high head and one low head safety injection pumps, four fan cooler units, one containment spray pump, and one recirculation heat exchanger for long term recirculation core cooling.

All safeguards systems in operation at any one time can be powered by two of the three emergency diesel-generators provided. The containment pressure transient curve is shown on Figure 4b-1.

Following the reactor coolant system blowdown, the safety injection pumps deliver 505 lb/sec. of 100°F borated water from the refueling water tank starting at 20 seconds. Part of the flow is boiled off when it comes in contact with the hot core, and passes through the break into the containment. The addition of safety injection water results in only 4.7 per cent total metal water reaction by the time the core is cooled and covered at 430 seconds. The residual heat generation rate after that time is insufficient to cause any net steam formation in the safety injection water. The water level rises to the level of the break at 542 seconds, and the heated injection water spills out into the containment after this time. At 45 seconds, one internal spray pump begins delivering 2600 gpm of 100°F water. At 2700 seconds, spray and safety injection exhausts the supply of water in the refueling water storage tank. Spray is shut off and recirculation is started. Hot containment sump water is withdrawn at the rate of 3000 gpm, passed through a residual heat exchanger and returned to the core via the core deluge and safety injection lines. Because the recirculation water leaving the heat exchanger is warmer than the safety injection water, some steam is formed by core decay heat. However, heat removal by the static heat sinks and fan coolers is able to prevent a large change in containment pressure.

Figure 4b-2 shows the integrated core residual heat generation and the reactor vessel metal energy release curves. For conservatism, the decay heat was calculated on the basis of infinite operating lifetime prior to the accident. The hot metal energy is released as the safety injection water contacts the walls of the reactor vessel. The total energy available to boil or heat injection water consists of the above two energy terms plus the metal-water reaction energy shown on Figure 4b-3 and the initial core stored energy of 42.0×10^6 Btu. The next energy source for the core and vessel is the enthalpy of the 100°F safety injection water entering the vessel. This curve is also shown on Figure 4b-2 and included in the energy balance for the containment. At 2700 seconds, the supply of safety injection and spray water in the refueling water storage tank has been exhausted. Recirculation cooling of the containment sump water is then started. The enthalpy of the recirculation water entering the vessel, after being cooled in the residual heat exchanger, is also shown on Figure 4b-2.

There are several containment energy sources The largest is the integrated enthalpy flow of 324.4×10^{6} Btu. resulting from reactor coolant blowdown. The reactor coolant enthalpy, rather than internal energy, is the correct thermodynanic property to describe a flow process. This term also contains the core and thin metal stored energy and core decay heat transferred to the coolant during the blowdown period. The hydrogen-oxygen recombination energy is shown on Figure 4b-3. Hydrogen formed by the zirconium-water reaction is assumed to flow to the containment, burning as it enters the containment air-steam mixture. Another energy source for the containment is the stored energy in hot metal walls above the level of the break and in the steam generators. This energy release is plotted on Figure 4b-4.

The analysis assumes that the hot walls and steam generator tubes are in direct contact with the containment steam-air mixture. This is an extremely conservative assumption as these surfaces are in stagnent flow regions within the reactor coolant system. The actual rate of energy release by the surfaces will be much smaller than assumed.

The final two containment energy sources result from mass entering the containment volume and are also shown on Figure 4b-4. The first of these sources is the combined enthalpy flow of internal spray and spilling safety injection water which does not reach the core because it is injected into the severed reactor coolant loop. The second enthalpy source term carries the vessel and core energies into the containment. From 20 to 430 seconds, it is a flow of steam as safety injection water cools the core. From 430 to 542 seconds, there is no flow as the vessel refills to the break level. From 542 to 2700 seconds, heated but subcooled safety injection water is spilling out the break. After 2700 seconds, recirculation of containment sump water is used in place of injection of the refueling water.

The containment heat sinks are shown in Figure 4b-5. The net heat sinks include cold structural walls and the four fan coolers. The heat removal by the containment spray is also shown on Figure 4b-5. The containment spray does not remove energy from the containment volume, but only transfers energy from the steam-air mixture into the sump water. At 2700 seconds, the spray is stopped because the refueling water storage tank is empty. Water is then drawn from the containment sump, passed through a residual heat exchanger, and injected back into the vessel to keep the core covered. The enthalpy of the sump water being removed from the containment by the recirculation system is shown on Figure 4b-5.

The internal energies of the containment steam-air mixture, sump water, and reactor vessel water are presented on Figure 4b-6.

To aid in understanding the various source and sink curves presented, a sample calculation was prepared to show the energy balance at 1000 seconds.

SAMPLE ENERGY BALANCE AT 1000 SECONDS

All energies have units of 10^6 Btu.

1. Core and Vessel Energy Terms

Decay Heat (Figure 4b-2)	94.0	
Metal-Water Reaction (Figure 4b-3)	5.4	
Vessel Metal Energy (Figure 4b-2)	3.5	
Initial Core Stored Energy	42.0	
Safety Injection Enthalpy	33.6	
	177.5	
Steam Flow to Containment (Figure 4b-4)		126.0
Vessel Water Stored Energy (Figure 4b-6)		48.0
		174.0

Error = 3.5 (∿2.0%)

2. Containment Energy Terms

Steam Flow to Containment (Figure 4b-4)	126.0	
Spray and Spilled Safety Injection (Figure 4b-4)	32.4	
Hot Walls and Steam Generators (Figure 4b-4)	22.0	
Hydrogen-Oxygen Recombination Energy (Figure 4b-3)	4.2	
Reactor Coolant System Blowdown	324.4	
	509.0	
Structural Heat Sinks (Figure 4b-5)		65.0
Four Fan Coolers (Figure 4b-5)		51.0
Sump Water Stored Energy (Figure 4b-6)		223.0
Steam-Air Stored Energy (Figure 4b-6)		166.0
		505.0

Estimated Error = 4 (∿0.6%)

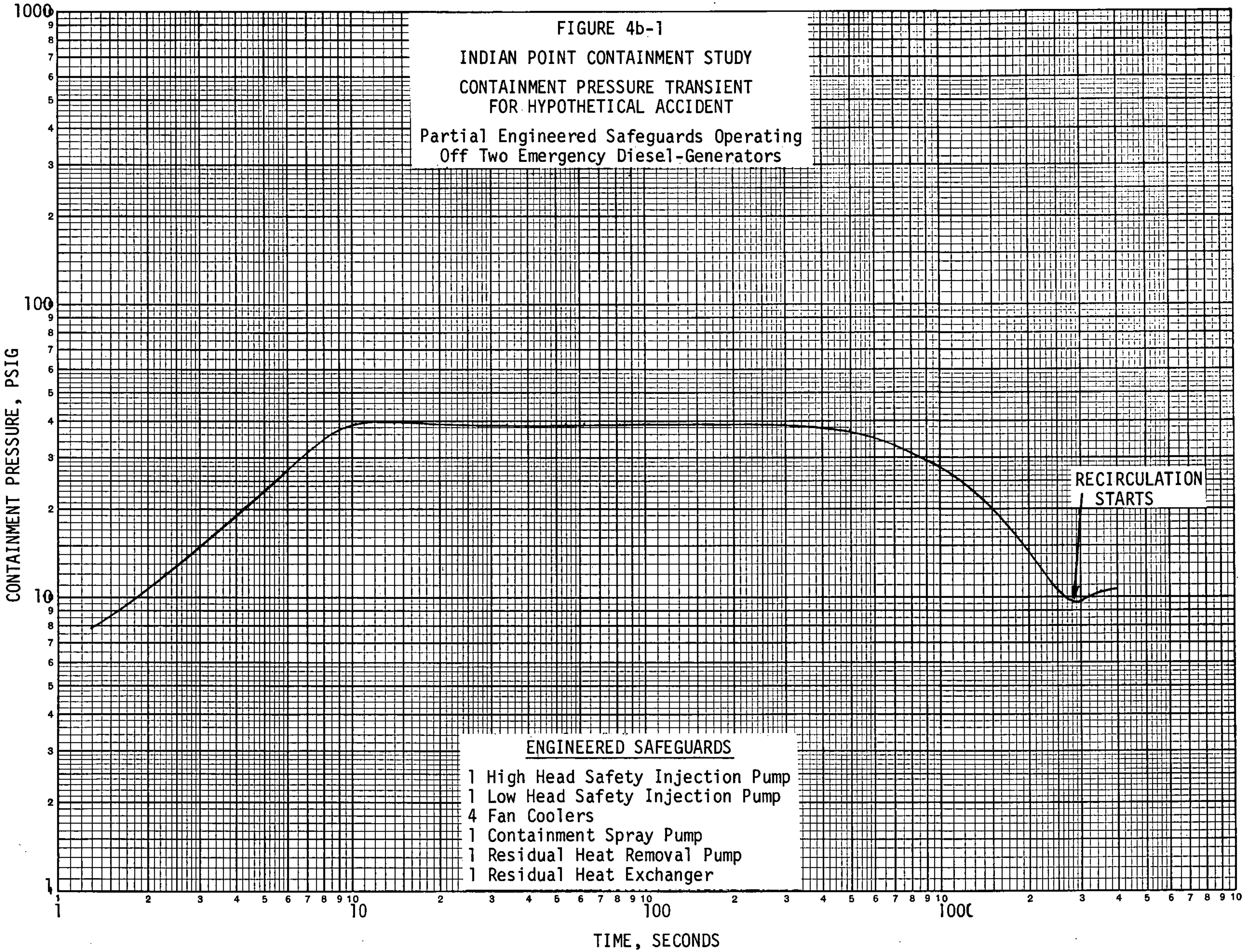


FIGURE 4b-1
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENT
FOR HYPOTHETICAL ACCIDENT
Partial Engineered Safeguards Operating
Off Two Emergency Diesel-Generators
RECIRCULATION
STARTS
ENGINEERED SAFEGUARDS
1 High Head Safety Injection Pump
1 Low Head Safety Injection Pump
4 Fan Coolers
1 Containment Spray Pump
1 Residual Heat Removal Pump
1 Residual Heat Exchanger
CONTAINMENT PRESSURE, PSIG
1000
100
10
1
10
100
1000
TIME, SECONDS

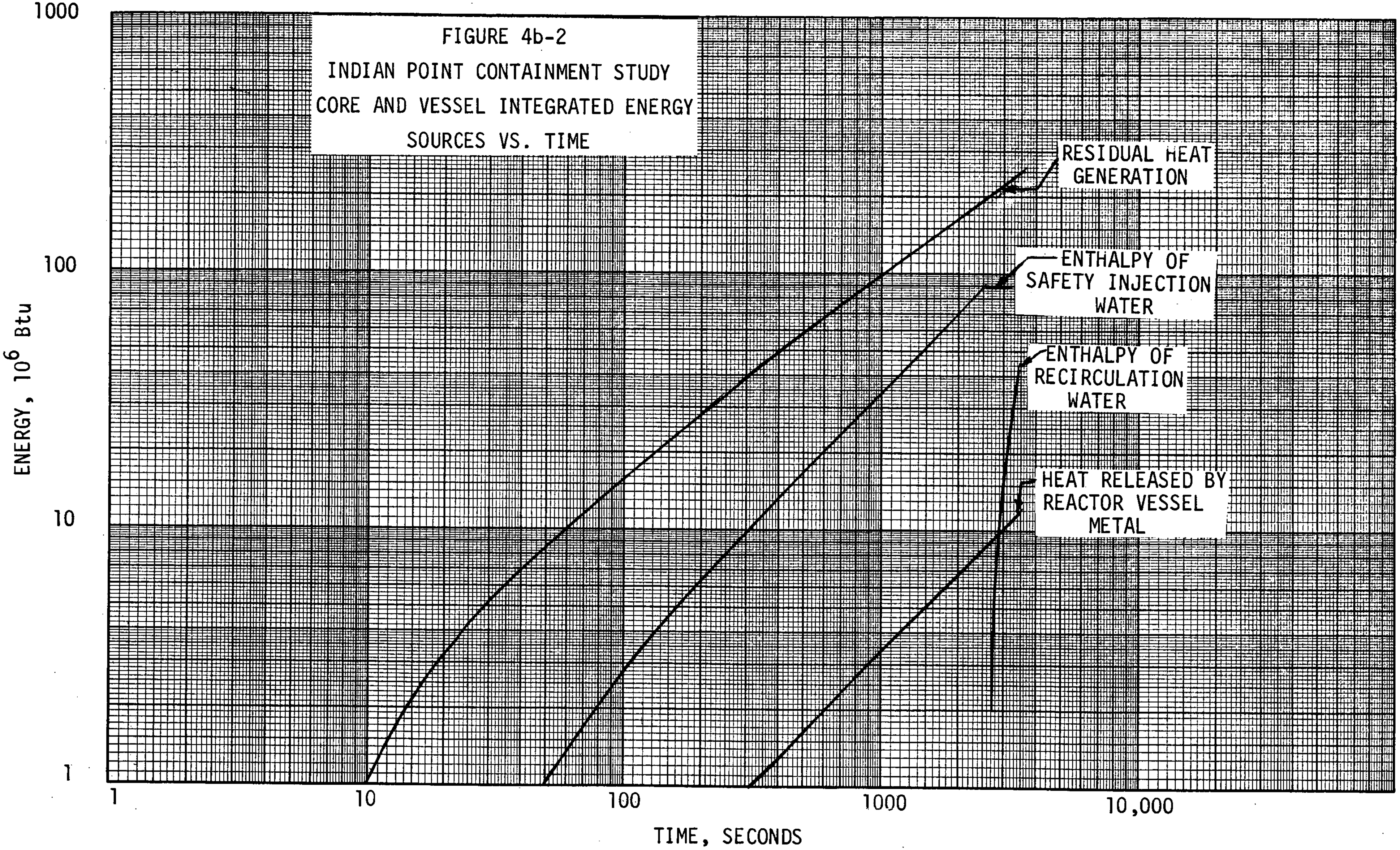


FIGURE 4b-2
INDIAN POINT CONTAINMENT STUDY
CORE AND VESSEL INTEGRATED ENERGY
SOURCES VS. TIME
RESIDUAL HEAT GENERATION
ENTHALPY OF SAFETY INJECTION WATER
ENTHALPY OF RECIRCULATION WATER
HEAT RELEASED BY REACTOR VESSEL METAL
1000
100
10
1
ENERGY, 10^6 Btu
1
10
100
1000
10,000
TIME, SECONDS

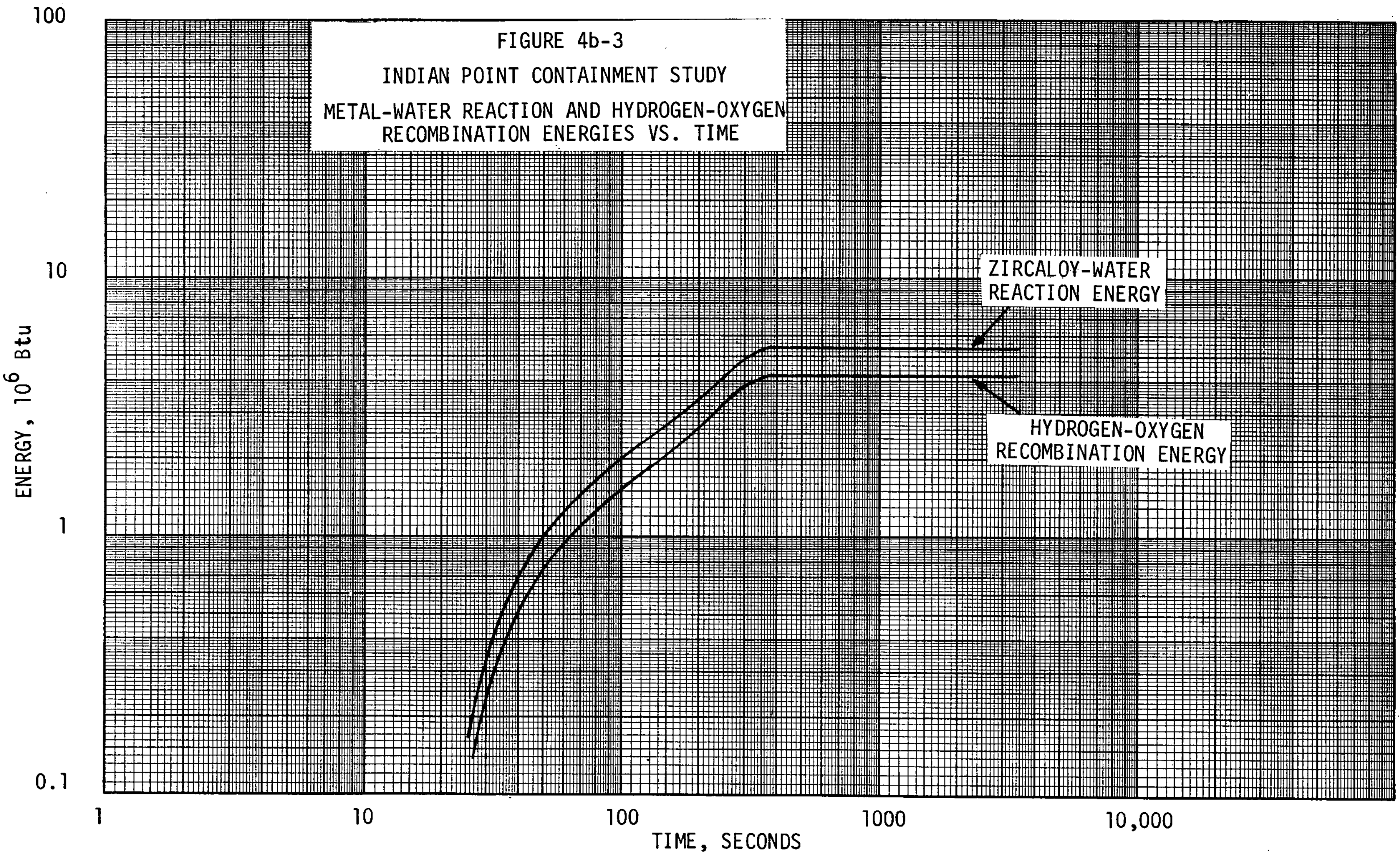


FIGURE 4b-3
INDIAN POINT CONTAINMENT STUDY
METAL-WATER REACTION AND HYDROGEN-OXYGEN RECOMBINATION ENERGIES VS. TIME
ZIRCALOY-WATER REACTION ENERGY
HYDROGEN-OXYGEN RECOMBINATION ENERGY
ENERGY, 10^6 Btu
100
10
1
0.1
1
10
100
1000
10,000
TIME, SECONDS

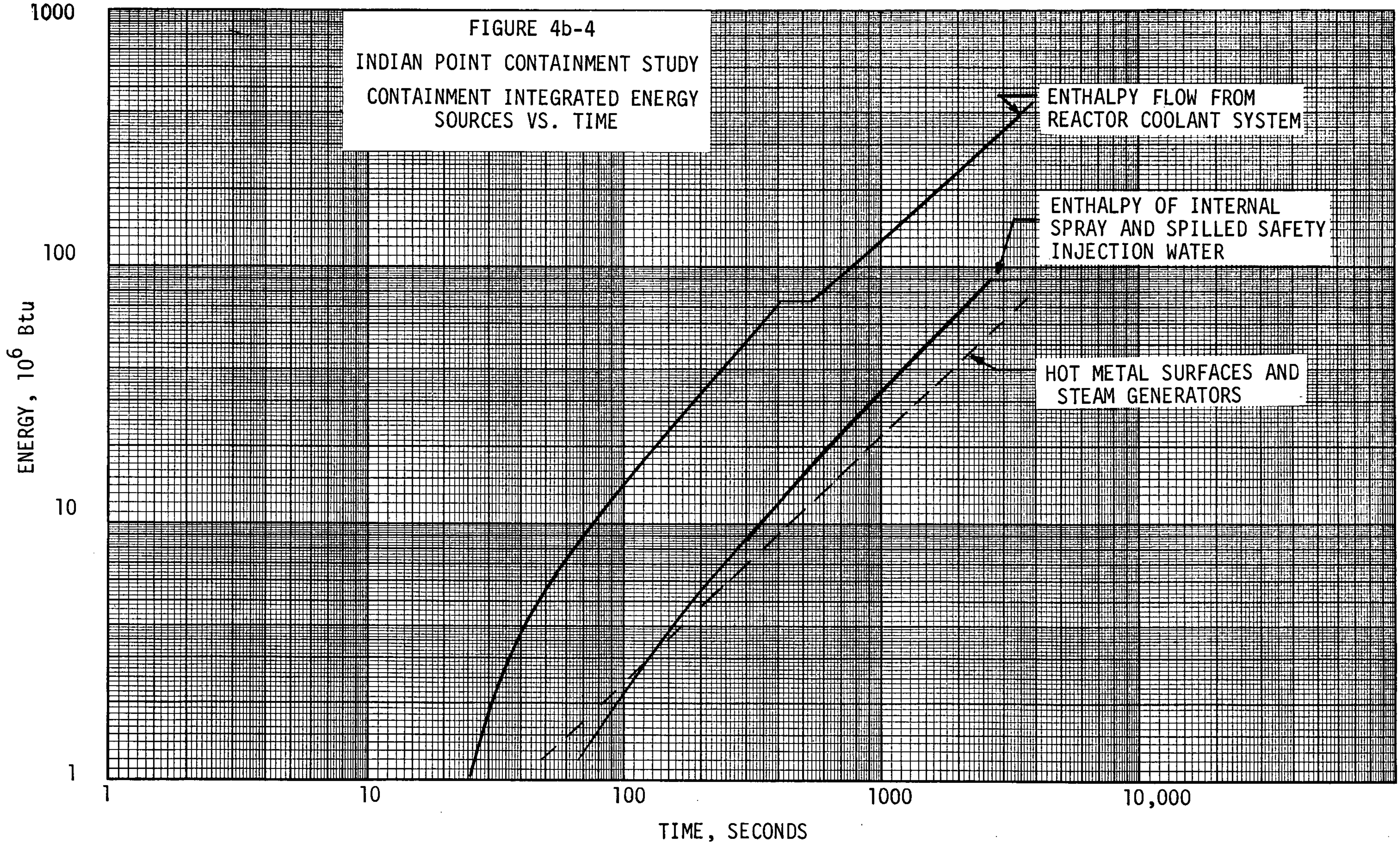


FIGURE 4b-4
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT INTEGRATED ENERGY SOURCES VS. TIME
ENTHALPY FLOW FROM REACTOR COOLANT SYSTEM
ENTHALPY OF INTERNAL SPRAY AND SPILLED SAFETY INJECTION WATER
HOT METAL SURFACES AND STEAM GENERATORS
1000
100
10
1
ENERGY, 10^6 Btu
1
10
100
1000
10,000
TIME, SECONDS

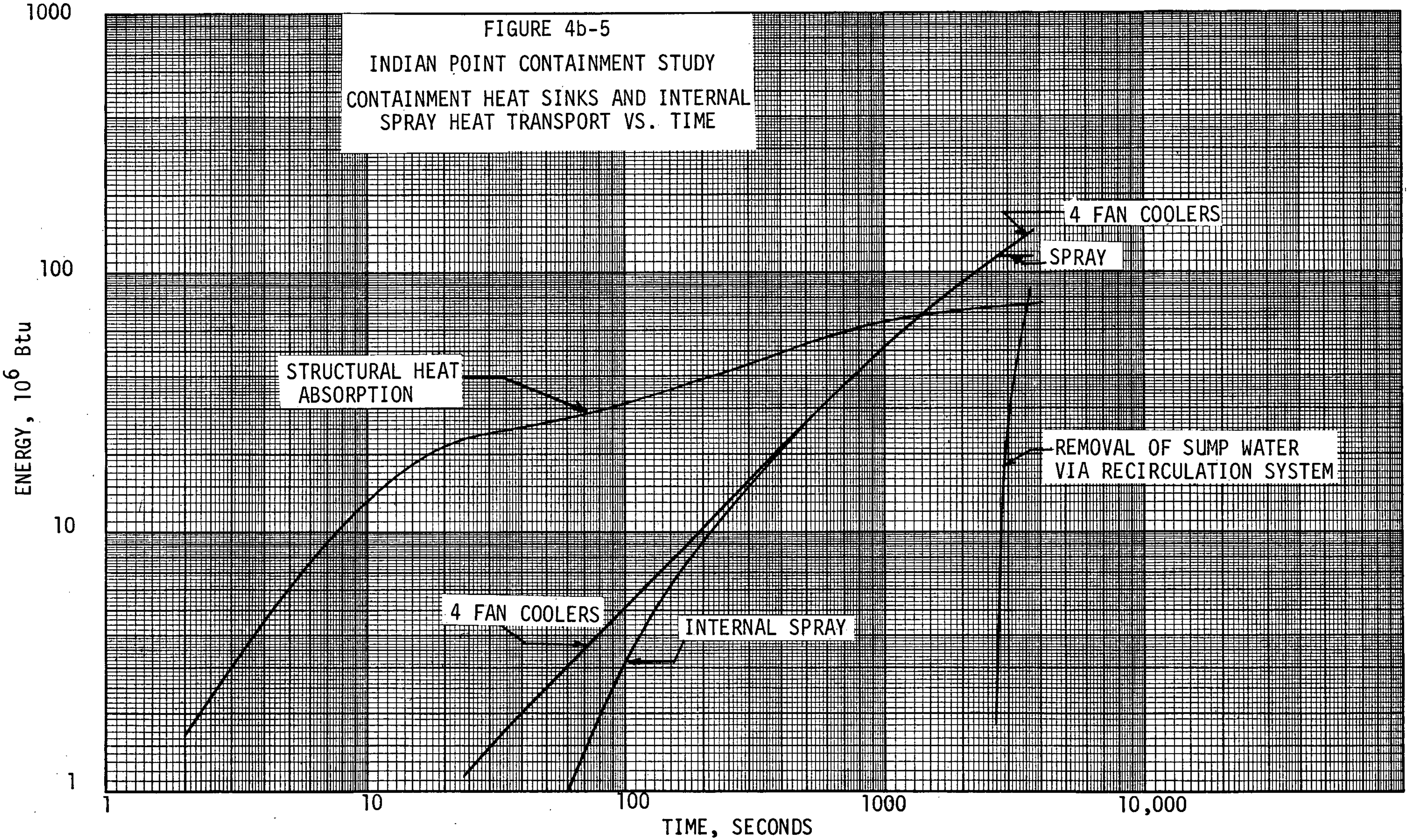


FIGURE 4b-5
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT HEAT SINKS AND INTERNAL SPRAY HEAT TRANSPORT VS. TIME
1000
100
10
1
ENERGY, 10^6 Btu
1
10
100
1000
10,000
TIME, SECONDS
4 FAN COOLERS
SPRAY
STRUCTURAL HEAT ABSORPTION
REMOVAL OF SUMP WATER VIA RECIRCULATION SYSTEM
4 FAN COOLERS
INTERNAL SPRAY

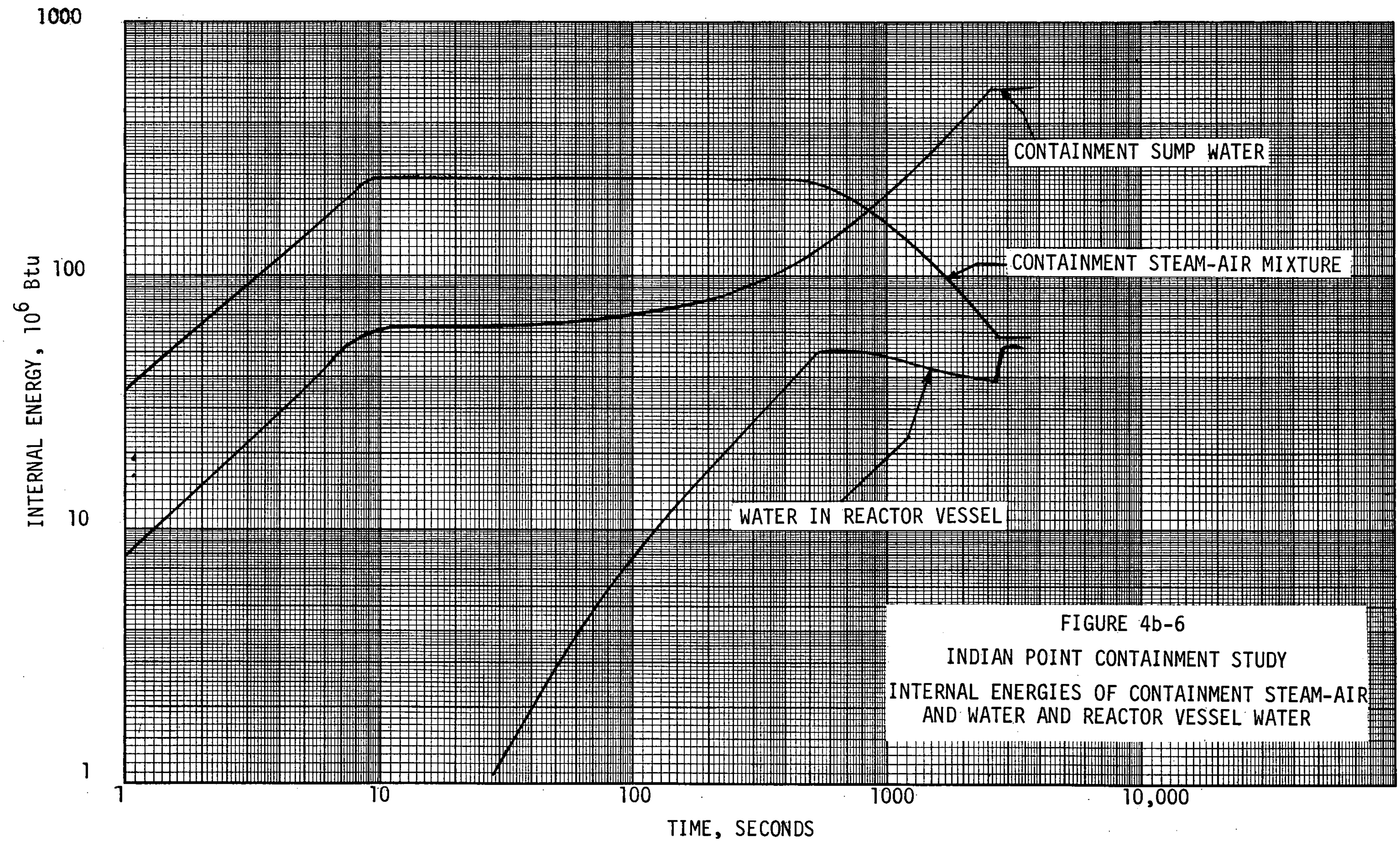


FIGURE 4b-6

INDIAN POINT CONTAINMENT STUDY

INTERNAL ENERGIES OF CONTAINMENT STEAM-AIR AND WATER AND REACTOR VESSEL WATER

4. The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

   c. Indicate in graphical form the per cent of zirconium in the core available for reaction by providing a family of curves indicating the per cent of core clad at or above given temperatures and the zirconium assumed reacted, as a function of time with (1) no safety injection, and (2) full safety injection.

Answer

Figures 4c-1 and 4c-2 present the per cent of zirconium reacted and per cent of core cladding above given temperatures, respectively, for the case of a double-ended rupture of a reactor coolant pipe with no safety injection. Three temperatures of interest are shown on Figure 4c-2, 1800°F - the temperature at which any significant reaction can occur, 3375°F - the temperature at which Zr metal melts, and 4800°F - the temperature at which ZrO_2 melts. In Figure 4c-1, which shows % reaction vs time, it should be noted that the more conservative assumption that the cladding remains intact until reaching the ZrO_2 melting temperature is used.

Figure 4c-1 presents in graphical form the maximum per cent zirconium reacted as a function of time. The assumed reaction limit (45%) used for the containment analysis is well above the predicted limit of less than 40%.

Curves of cladding temperature and per cent zirconium reacted as a function of time with full safety injection operation are given in Figures 4c-3 and 4c-4, respectively. The per cent reaction and fraction of cladding above a given temperature are drastically reduced.

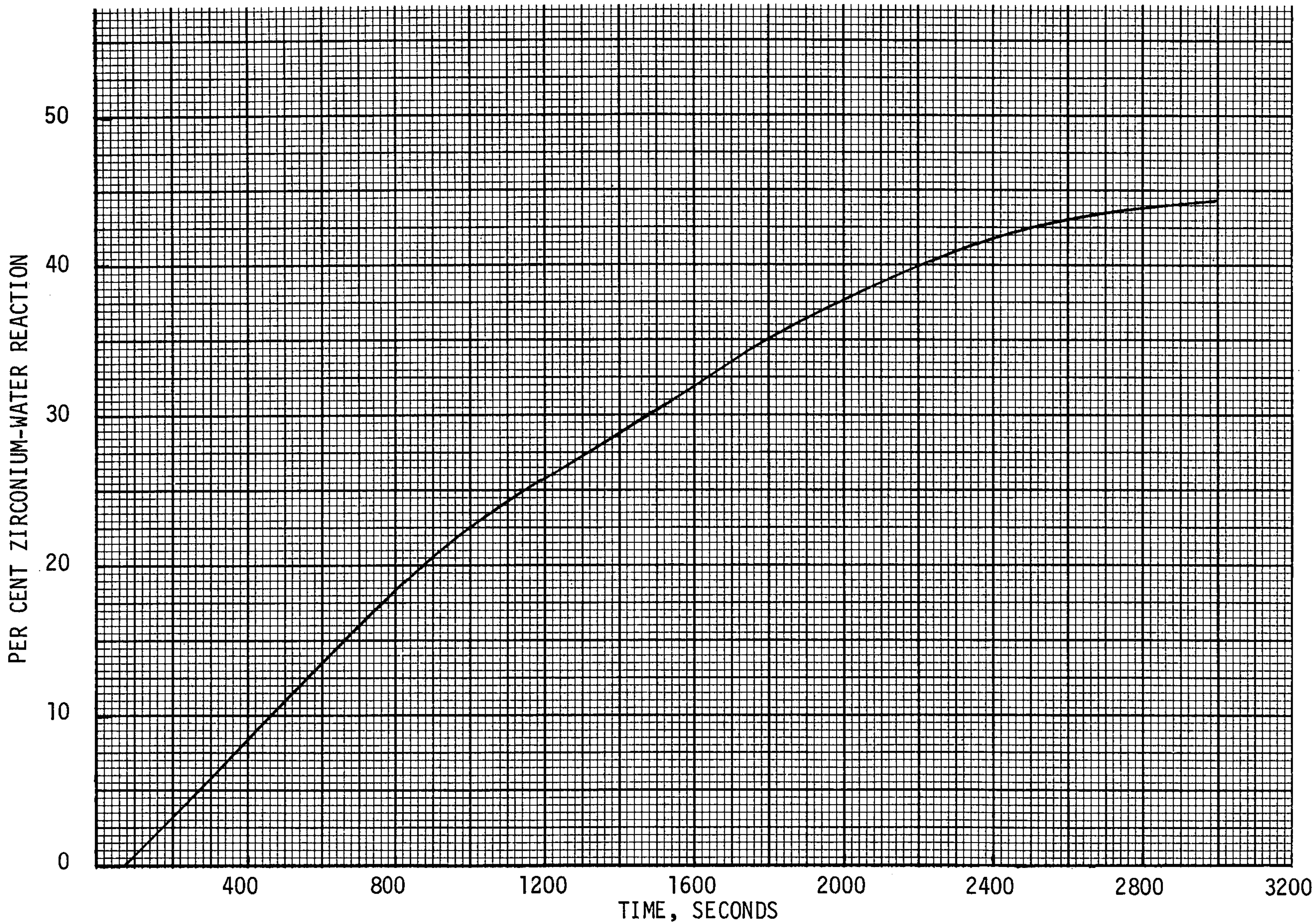


FIGURE 4c-1 DOUBLE ENDED RUPTURE MAIN COOLANT PIPE PER CENT ZIRCONIUM REACTED VS. TIME - NO SAFETY INJECTION

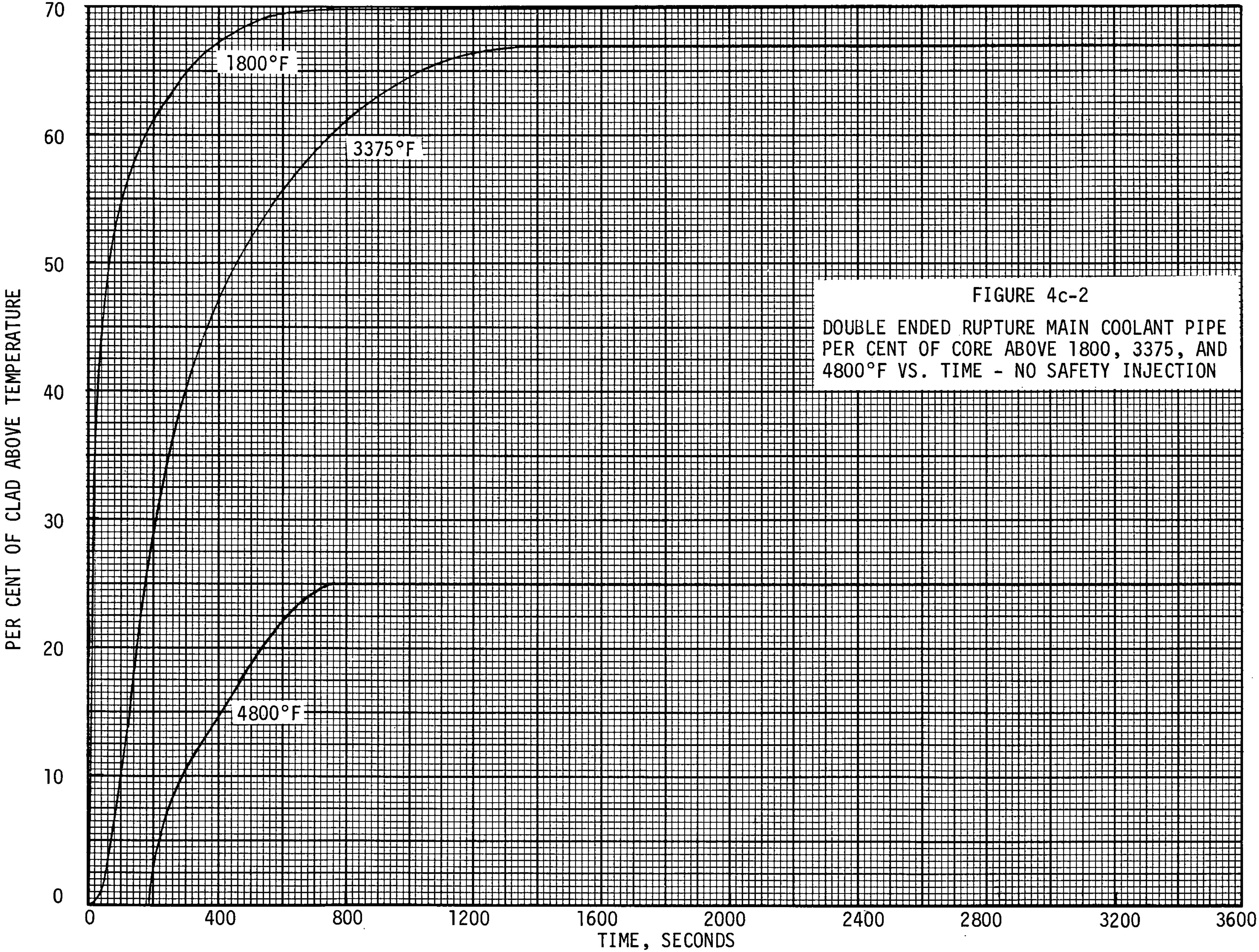


FIGURE 4c-2
DOUBLE ENDED RUPTURE MAIN COOLANT PIPE
PER CENT OF CORE ABOVE 1800, 3375, AND
4800°F VS. TIME - NO SAFETY INJECTION
1800°F
3375°F
4800°F
PER CENT OF CLAD ABOVE TEMPERATURE
TIME, SECONDS
0
10
20
30
40
50
60
70
0
400
800
1200
1600
2000
2400
2800
3200
3600

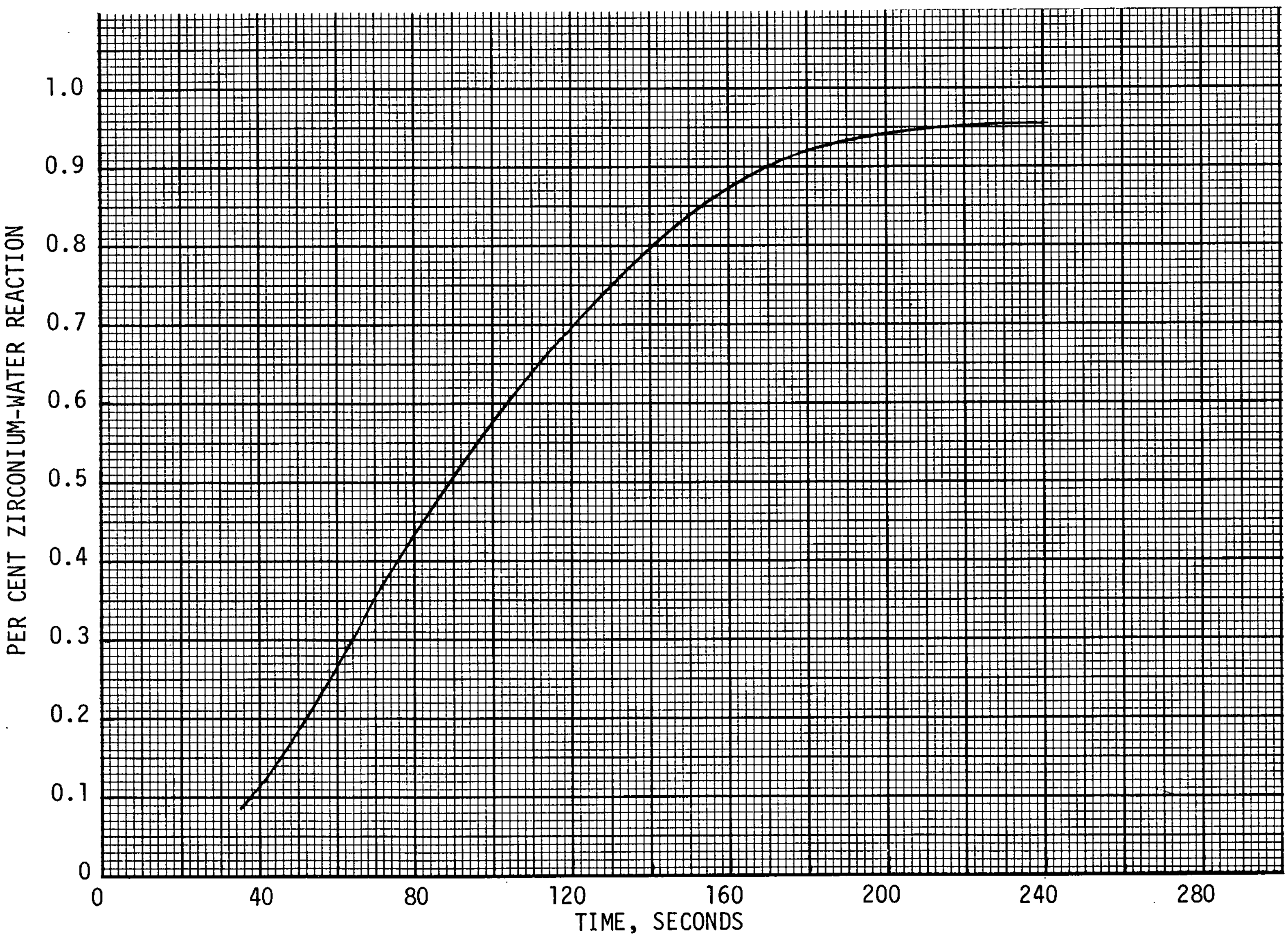


FIGURE 4c-3 DOUBLE ENDED RUPTURE MAIN COOLANT PIPE PER CENT ZIRCONIUM REACTED VS. TIME - FULL SAFETY INJECTION

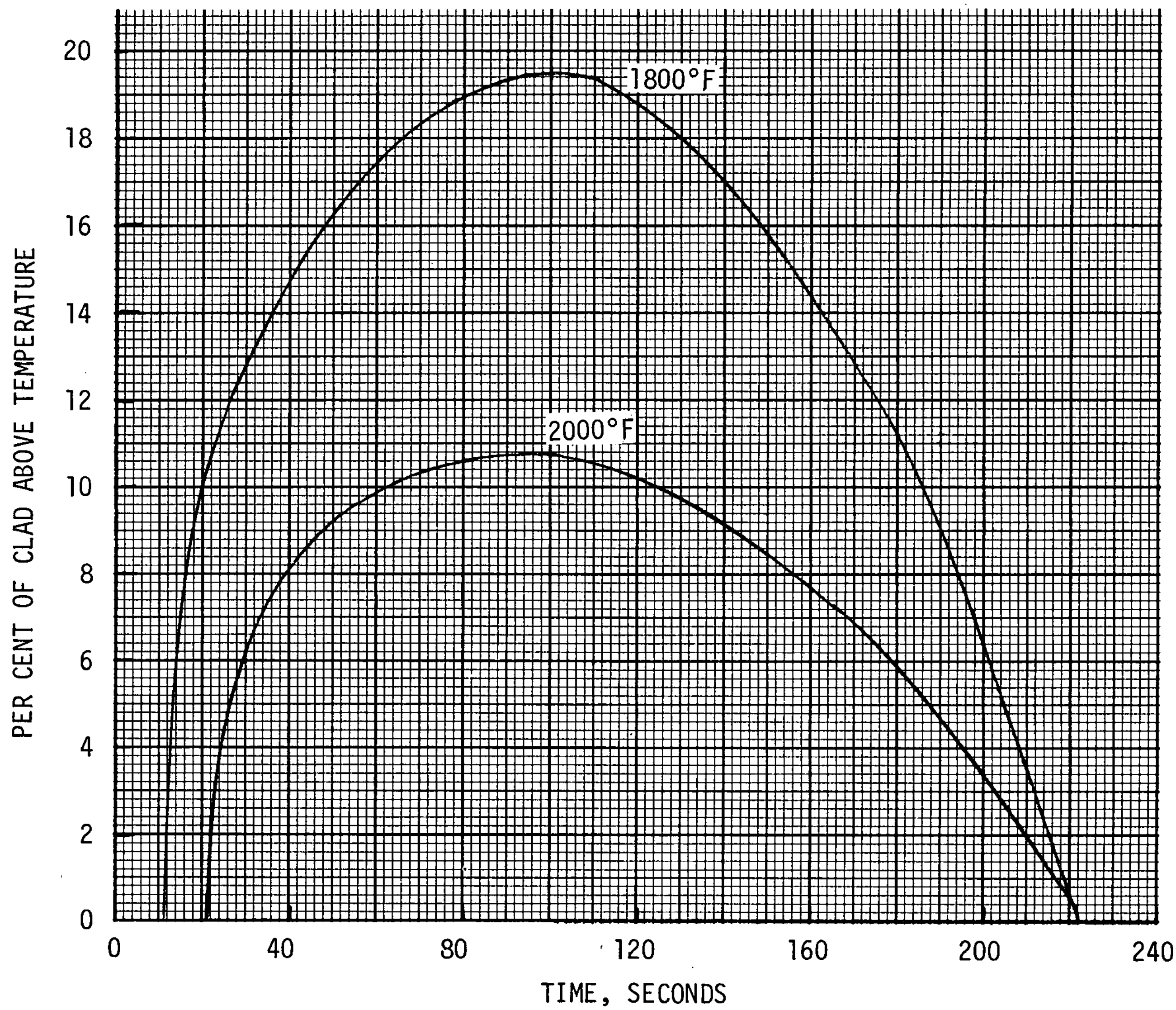


FIGURE 4c-4

DOUBLE ENDED RUPTURE MAIN COOLANT PIPE PER CENT
OF CLAD ABOVE 1800 and 2000°F VS. TIME - FULL SAFETY INJECTION

4. The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

   d. Provide the containment pressure transient curves following the MCA, assuming no further energy is added to the containment after the initial blowdown, for the cases wherein (1) all engineered safeguards function and (2) no engineered safeguards function and the containment and structures act as the only heat sink. Also show the per cent of total primary system energy lost as a function of time.

   e. Provide the information in part (d) showing the increase in containment pressure resulting by (1) adding additional energy by the mechanism of steam generation equal to 50% and 100% of the original primary coolant energy, linearly with time in 1000 seconds and (2) adding additional energy stepwise equivalent to 20% of the primary coolant energy at 500 and 1000 seconds by superheating the atmosphere. Also show the pound moles of air, steam, and hydrogen in the containment as a function of time.

Answer

The following ground rules were employed to obtain the various containment pressure transients:

First, the specifications included in these questions as to how energy is to be added to the containment make this a sensitivity study. None of the results presented can be interpreted as being credible accident cases.

Second, five fan coolers and two spray pumps were used in cases where all containment engineered safeguards are assumed to function. Four fans are assumed to be operating at the time of the accident. The fifth fan and the two containment spray pumps are started at 45 seconds. Safety injection was not used in this analysis because the question requires that no further energy is added to the containment after the initial blowdown. After the refueling water storage tank is emptied, spray is continued using sump water that is cooled in the two recirculation heat exchangers.

Third, the percentage energy additions are based on the integrated blowdown enthalpy of 324.4×10^6 Btu which includes part of the core and vessel stored energy plus residual heat generated during the blowdown phase.

Fourth, the energy of steam generation is added to the containment at a uniform rate between 10 and 1010 seconds. The superheat energy source is simulated by using a heat source which goes directly into the containment steam-air phase over a 10 second period at the specified times.

The pressure transient curves are presented for the following cases of energy addition both with and without engineered safeguards functioning:

1) Initial Reactor Coolant System Blowdown Only

The pressure transient curves for no engineered safeguards and all safeguards are shown on Figure 4d-1. The only anomaly occurs in the all engineered safeguards case at 3700 seconds when the spray is changed to the recirculation mode. The pressure is so low at this time that the steam-air temperature is lower than the temperature of the water leaving the recirculation heat exchangers. Recirculation spray warms the steam-air mixture causing a slight increase in containment pressure. The energies removed from the containment steam-air mixture by the walls, fans, and spray are shown on Figure 4d-2. Note that the rapid depressurization caused by fans, coolers and spray cause heat absorbed in the structure to be released back into the containment.

2) Blowdown Energy Plus An Additional 35 Per Cent of the Coolant Energy as Steam Generation

Additional energy was added to the containment by steam generation over a 1000 second period. The total energy added is 35 per cent of the reactor coolant blowdown enthalpy flow into the containment. The two pressure transient curves are shown on Figure 4d-3, while the energies removed from the containment steam-air mixture are shown on Figure 4d-4.

3) Blowdown Energy Plus 70 Per Cent Steam Generation

This case is similar to case 2 above except the percentage energy addition is increased to 70%. The two pressure transient curves are shown on Figure 4d-5, while the energies removed are shown on Figure 4d-6.

4) Blowdown Energy Plus 20 Per Cent of the Coolant Energy as Superheat at 500 Seconds

The energy addition is now assumed to go only into the containment steam-air mixture as a heat source. The total added energy is 20 per cent of the reactor coolant blowdown enthalpy flow into the containment,and goes into the steam-air mixture over a 10 second period starting at 500 seconds. As shown by Figure 4d-7, this form of energy addition causes a large pressure spike due to superheating of the containment steam. The pressure peak is 56.0 psig with all engineered safeguards operating and 73.0 psig with no safeguards. The energy removal breakdown is shown on Figure 4d-8. For the time period after 500 seconds when the steam is highly superheated, spray water entering the containment will flash to steam. Although the spray does reduce pressure by adding mass, there is no spray heat removal from the steam phase because spray flow is vaporized to steam. Hence, there is a flat spot in spray heat removal curve until the degree of superheating is reduced.

5) Blowdown Energy Plus 20 Per Cent as Superheat at 1000 Seconds

This case is similar to case 4 above except the energy addition is made at 1000 seconds. As shown on Figure 4d-9, the pressure peak is 46.0 psig with all engineered safeguards operating and 70.0 psig with no safeguards. The various energy removal terms are shown on Figure 4d-10.

The only hydrogen in the containment is that amount which was dissolved in the reactor coolant previous to the accident. A maximum of about 1 lb. mole of hydrogen could be obtained if all the dissolved gas were to come out of solution during the blowdown. This is a negligible amount of hydrogen in the containment. The mass of air in the containment is constant with time at value of 6,450 lb. moles (187,409 lbs.).

If the energy addition specified in the question as 20 per cent of the primary coolant energy were converted into a hydrogen equivalent, 654.0 lb. moles (1308 lbs.) of hydrogen would be required. Since in all the discussion above, transient phenomena in the core are ignored, there is no way to determine the hydrogen equivalent as a function of time and maintain consistancy with the analytical model as requested.

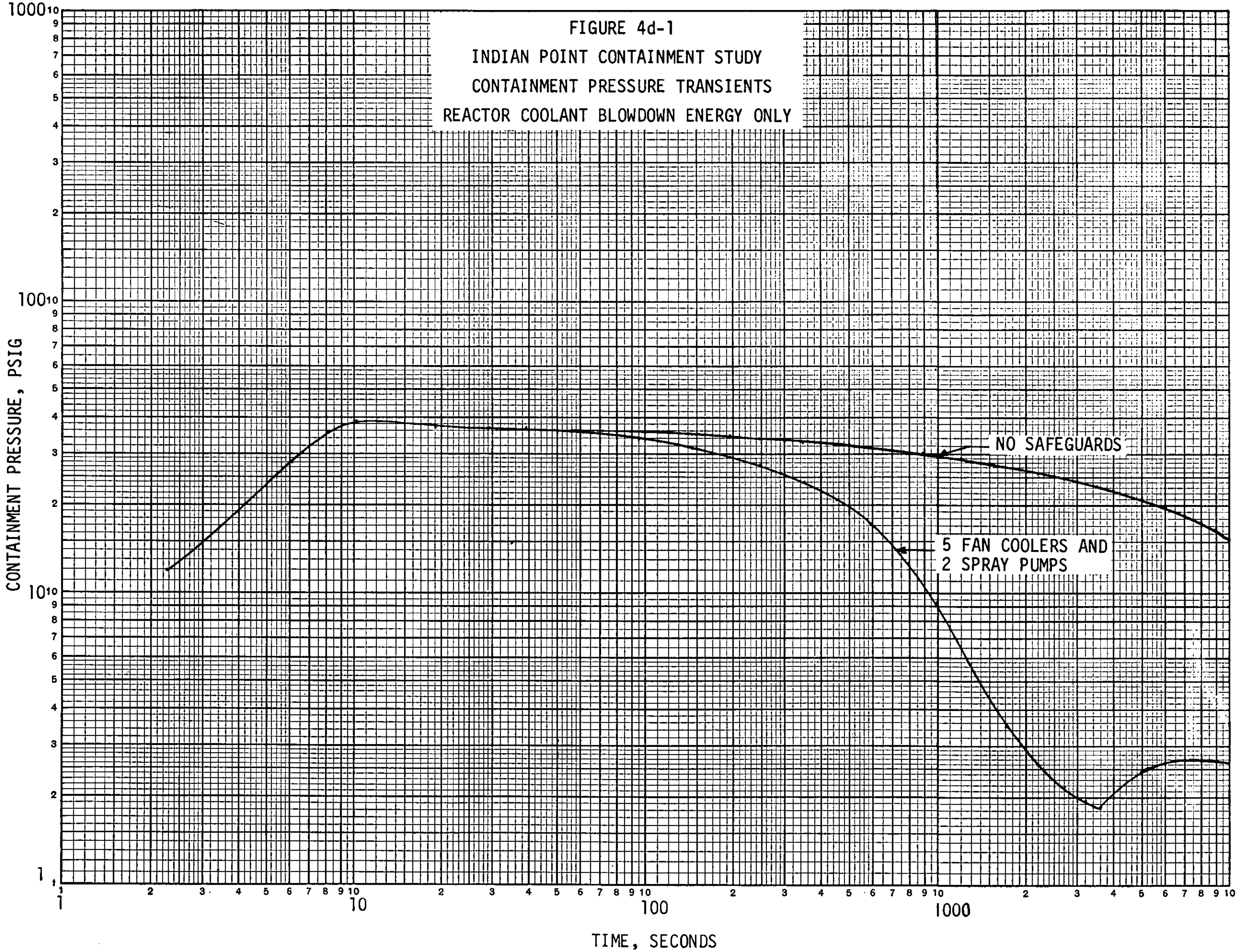


FIGURE 4d-1
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENTS
REACTOR COOLANT BLOWDOWN ENERGY ONLY
NO SAFEGUARDS
5 FAN COOLERS AND
2 SPRAY PUMPS
CONTAINMENT PRESSURE, PSIG
TIME, SECONDS
1000
100
10
1
1
10
100
1000

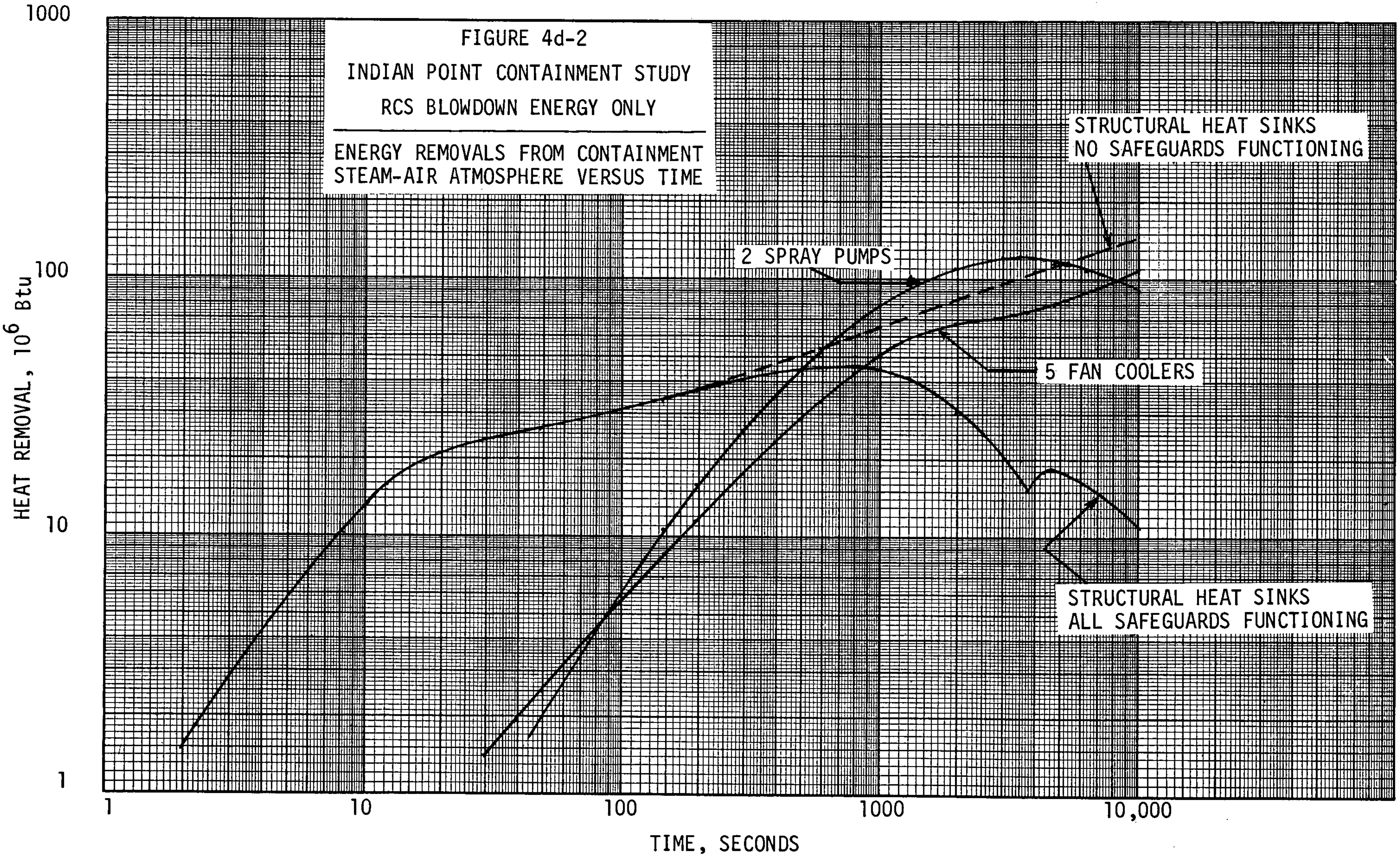
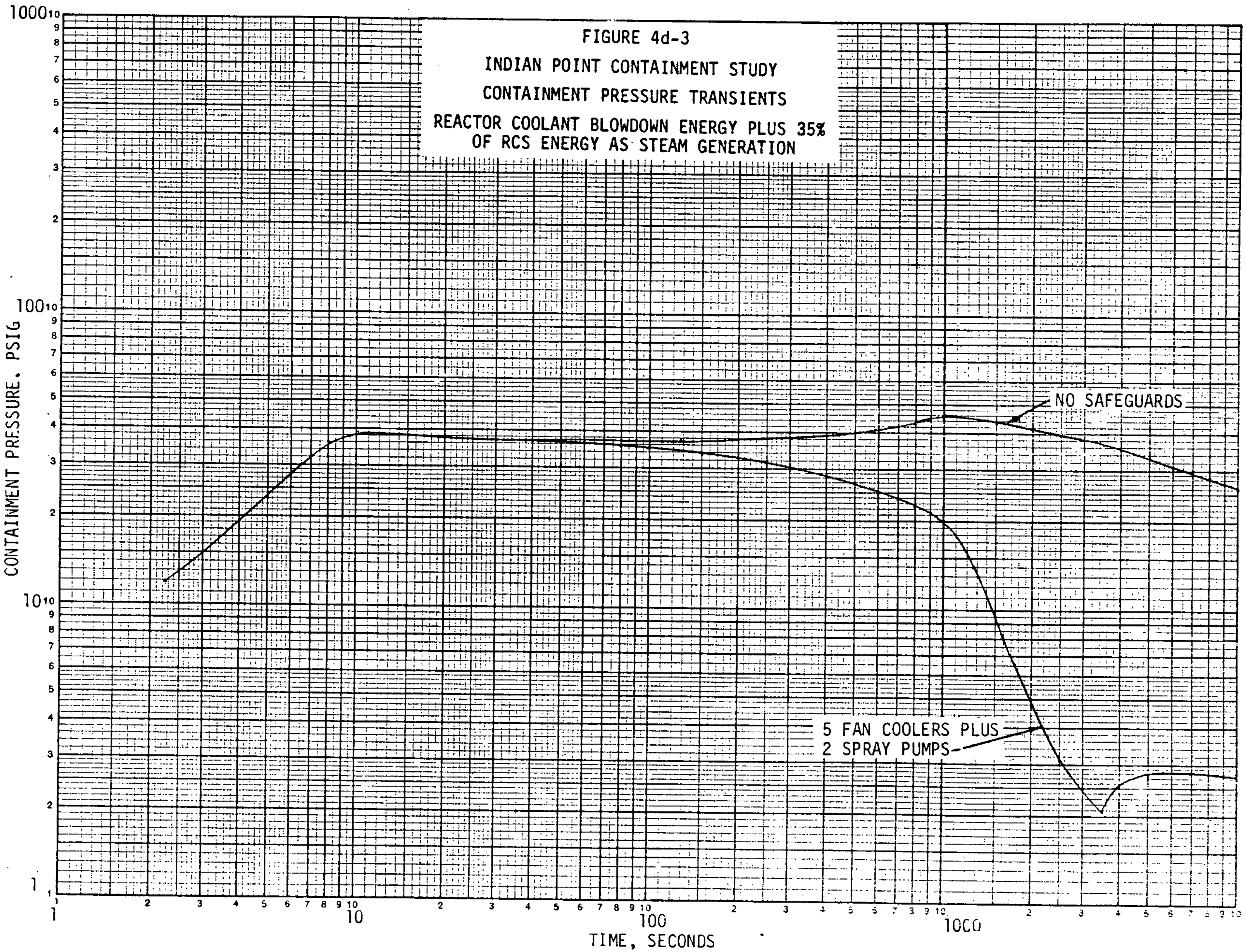
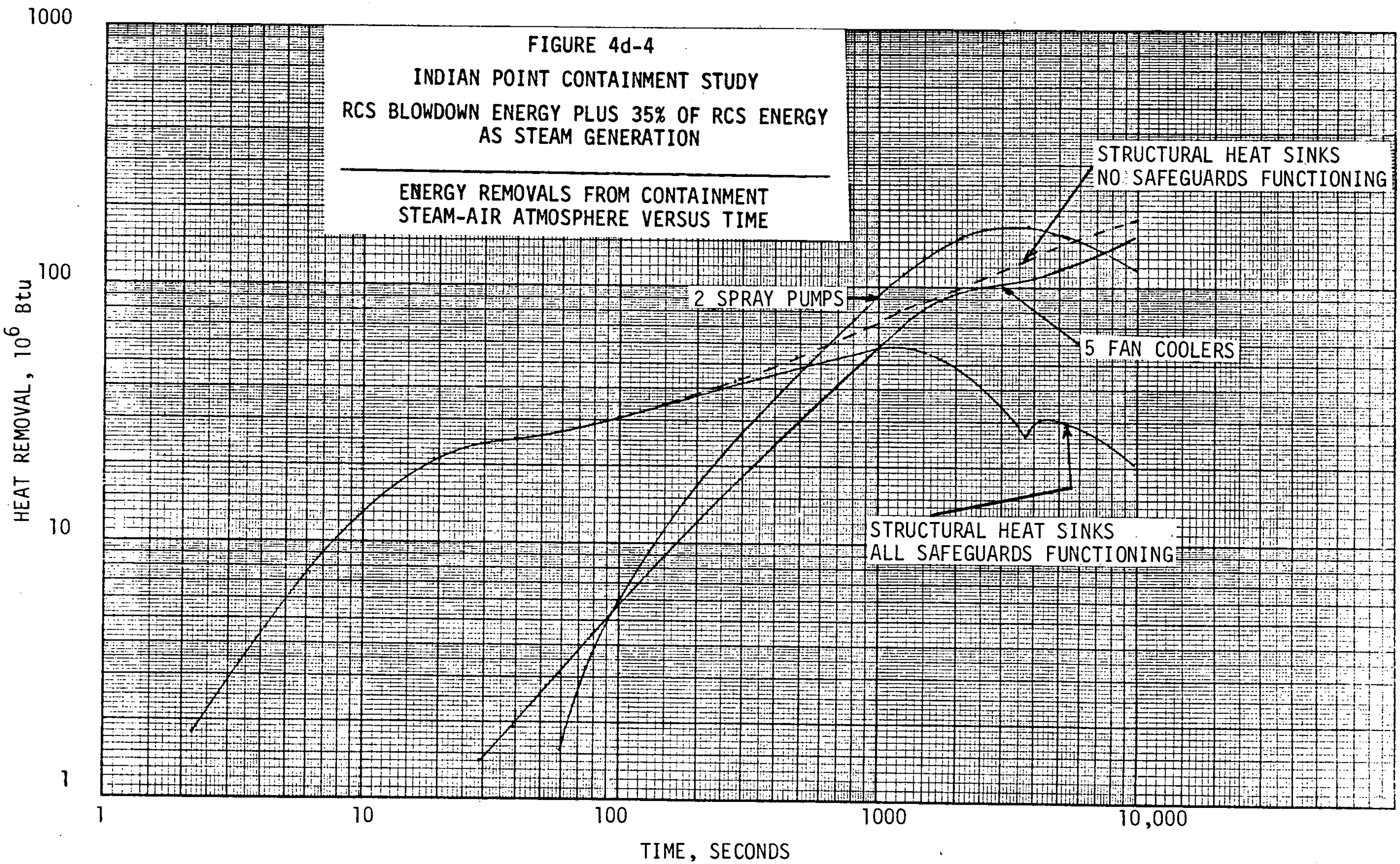


FIGURE 4d-2
INDIAN POINT CONTAINMENT STUDY
RCS BLOWDOWN ENERGY ONLY
ENERGY REMOVALS FROM CONTAINMENT
STEAM-AIR ATMOSPHERE VERSUS TIME
STRUCTURAL HEAT SINKS
NO SAFEGUARDS FUNCTIONING
2 SPRAY PUMPS
5 FAN COOLERS
STRUCTURAL HEAT SINKS
ALL SAFEGUARDS FUNCTIONING
HEAT REMOVAL, 10^6 Btu
1000
100
10
1
1
10
100
1000
10,000
TIME, SECONDS



(Revised 6-1-66)



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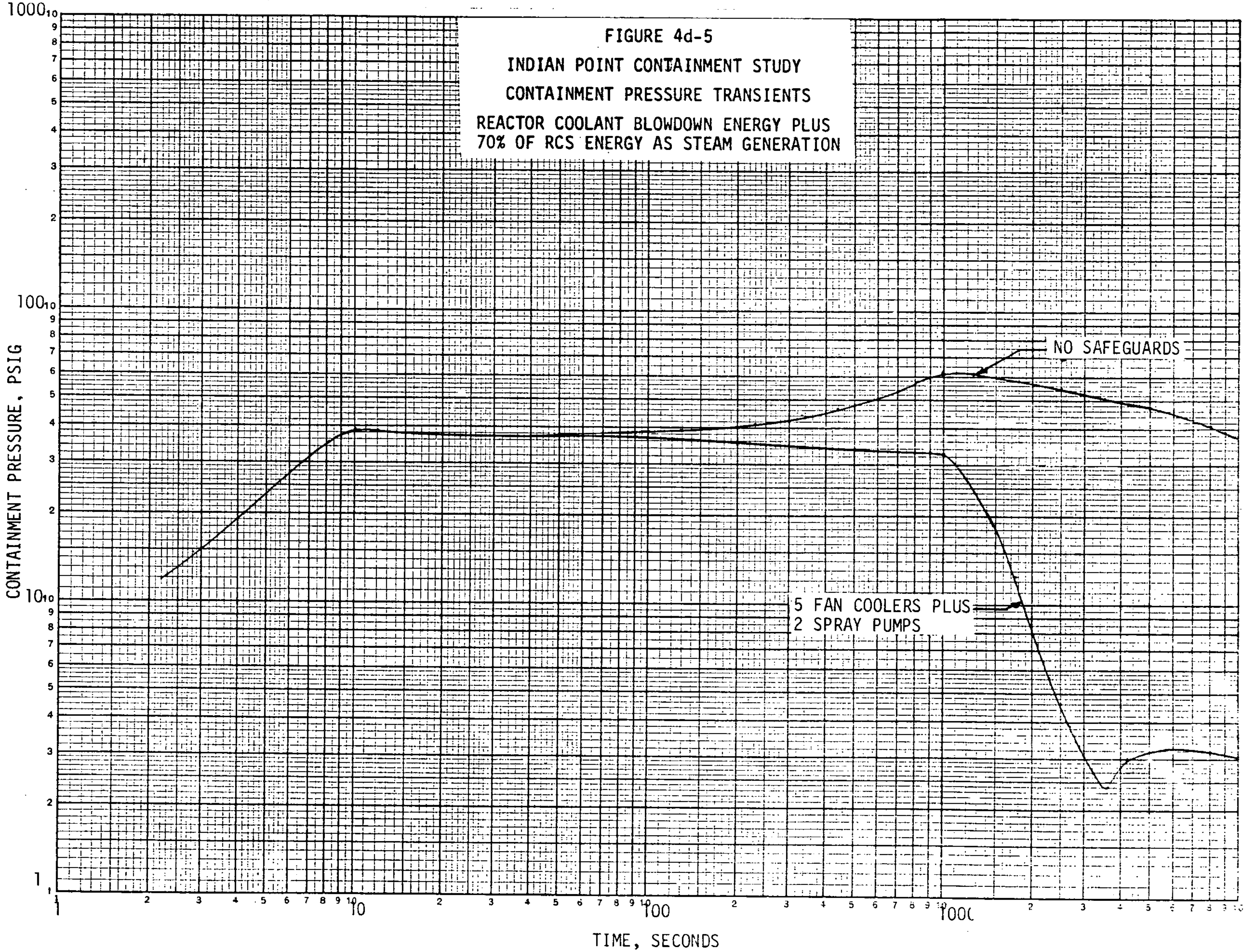
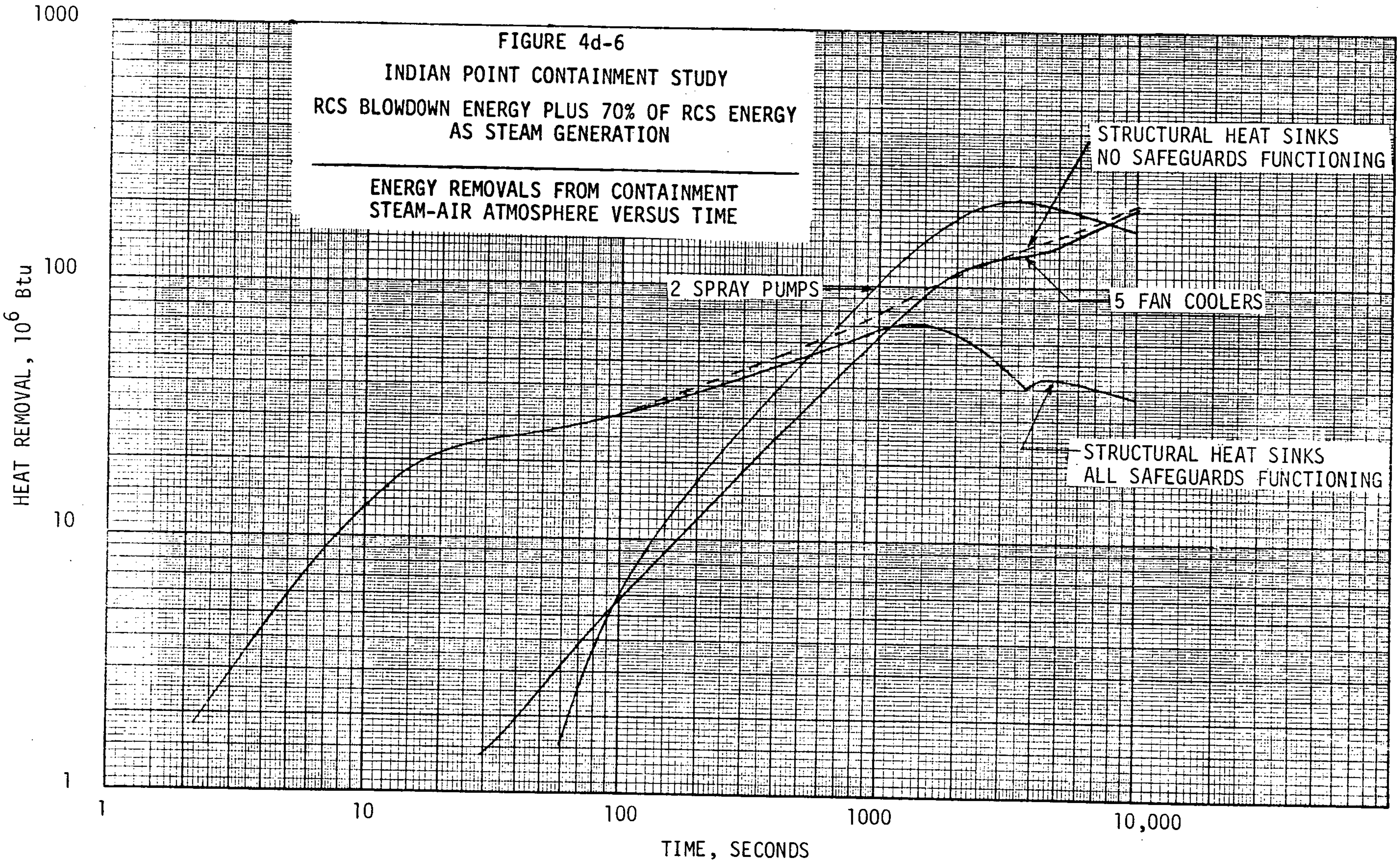


FIGURE 4d-5
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENTS
REACTOR COOLANT BLOWDOWN ENERGY PLUS
70% OF RCS ENERGY AS STEAM GENERATION

(Revised 6-1-66)



(Revised 6-1-66)

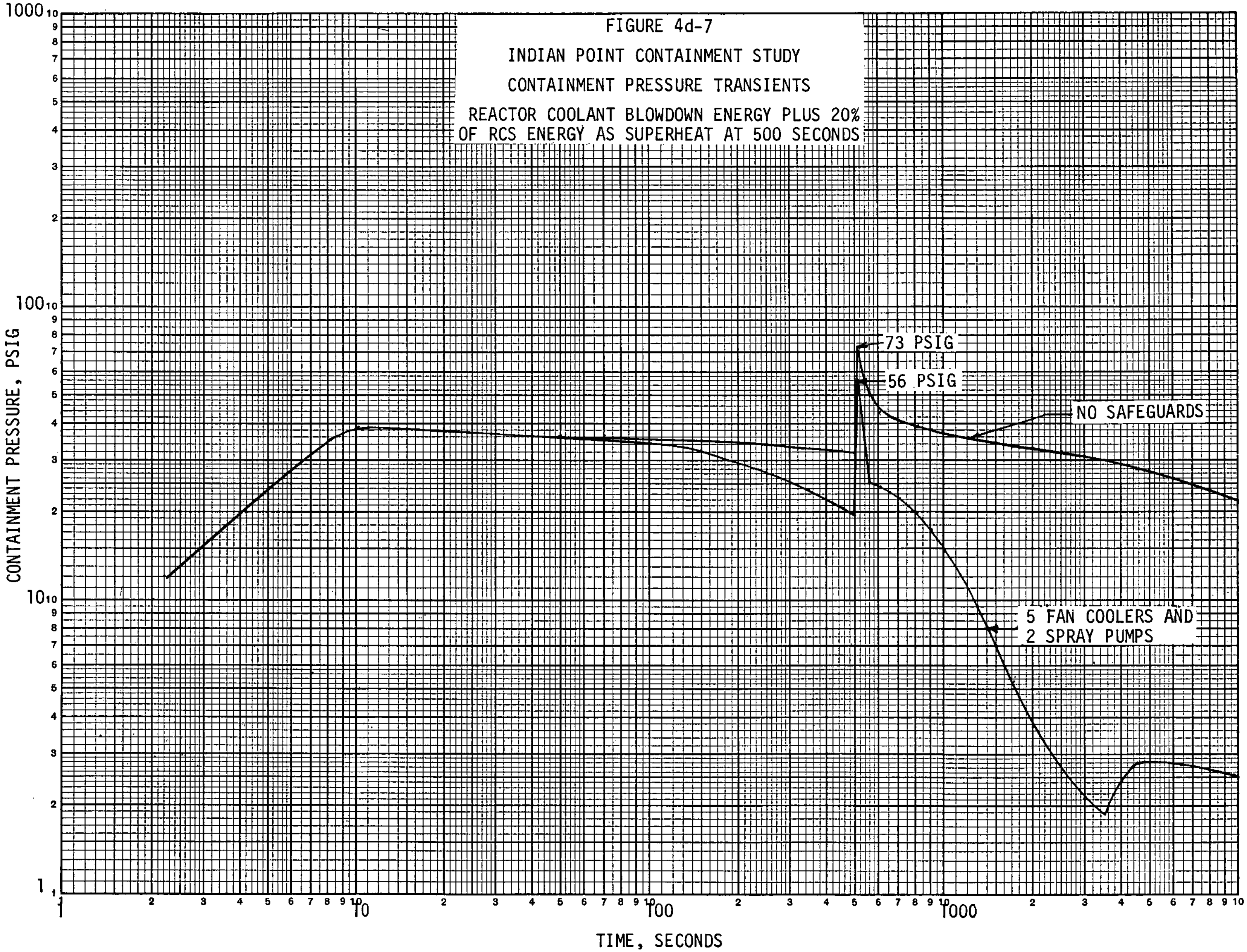


FIGURE 4d-7
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENTS
REACTOR COOLANT BLOWDOWN ENERGY PLUS 20%
OF RCS ENERGY AS SUPERHEAT AT 500 SECONDS
73 PSIG
56 PSIG
NO SAFEGUARDS
5 FAN COOLERS AND
2 SPRAY PUMPS
CONTAINMENT PRESSURE, PSIG
TIME, SECONDS
1000
100
10
1
1
10
100
1000

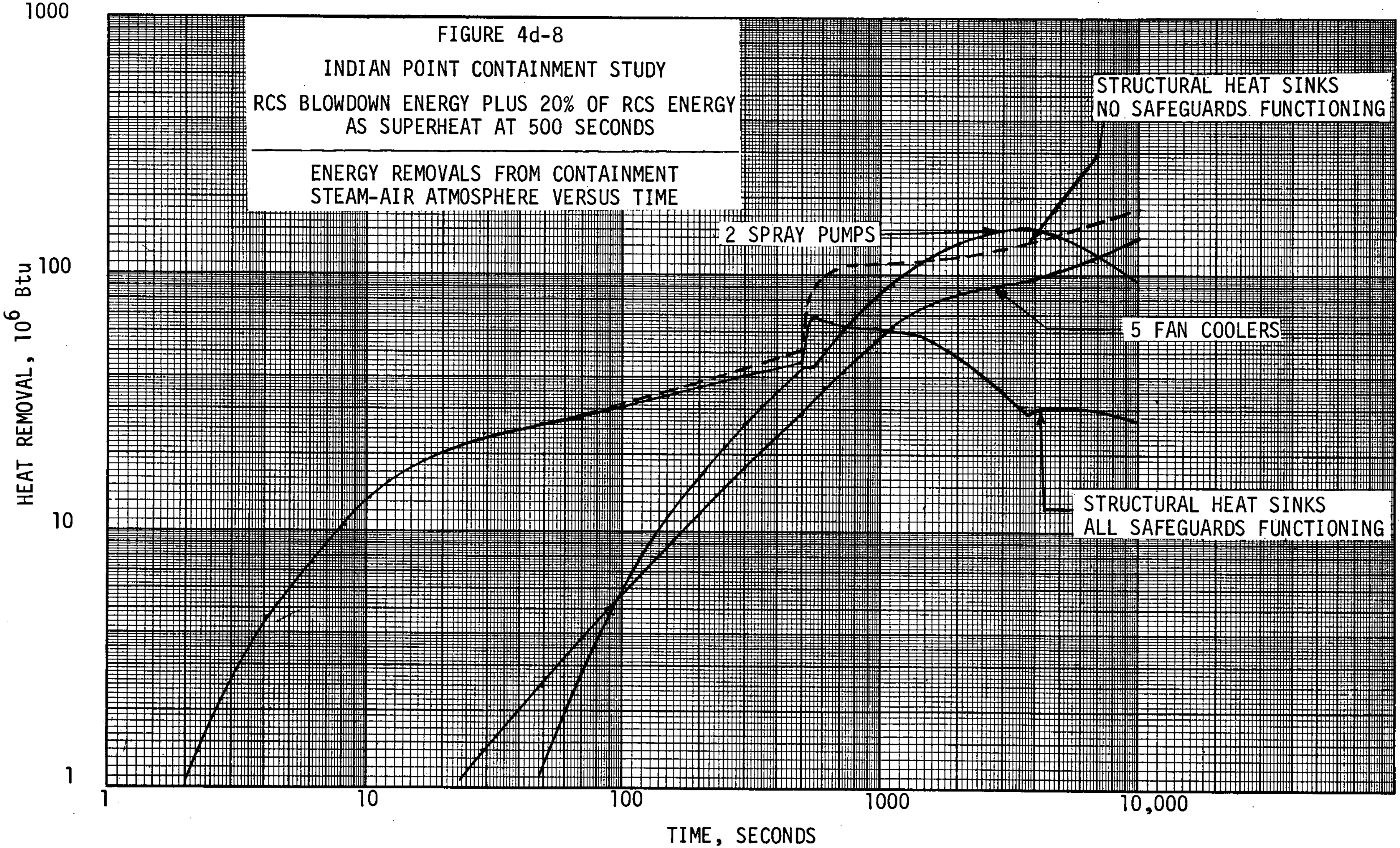


FIGURE 4d-8
INDIAN POINT CONTAINMENT STUDY
RCS BLOWDOWN ENERGY PLUS 20% OF RCS ENERGY AS SUPERHEAT AT 500 SECONDS
ENERGY REMOVALS FROM CONTAINMENT STEAM-AIR ATMOSPHERE VERSUS TIME
STRUCTURAL HEAT SINKS NO SAFEGUARDS FUNCTIONING
2 SPRAY PUMPS
5 FAN COOLERS
STRUCTURAL HEAT SINKS ALL SAFEGUARDS FUNCTIONING
HEAT REMOVAL, 10^6 Btu
1000
100
10
1
1
10
100
1000
10,000
TIME, SECONDS

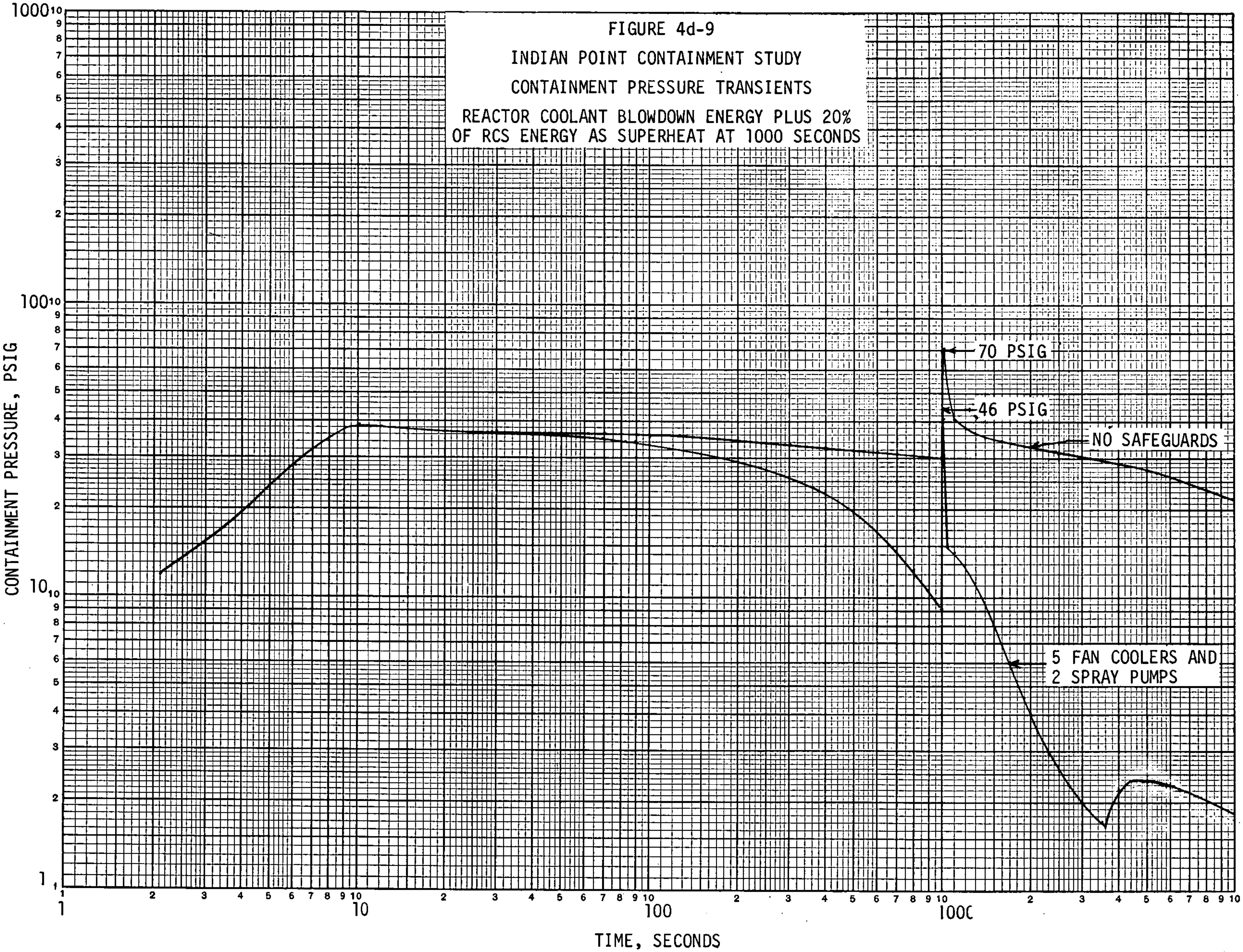


FIGURE 4d-9
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENTS
REACTOR COOLANT BLOWDOWN ENERGY PLUS 20%
OF RCS ENERGY AS SUPERHEAT AT 1000 SECONDS
70 PSIG
46 PSIG
NO SAFEGUARDS
5 FAN COOLERS AND
2 SPRAY PUMPS
CONTAINMENT PRESSURE, PSIG
TIME, SECONDS
1000
100
10
1
10
100
1000

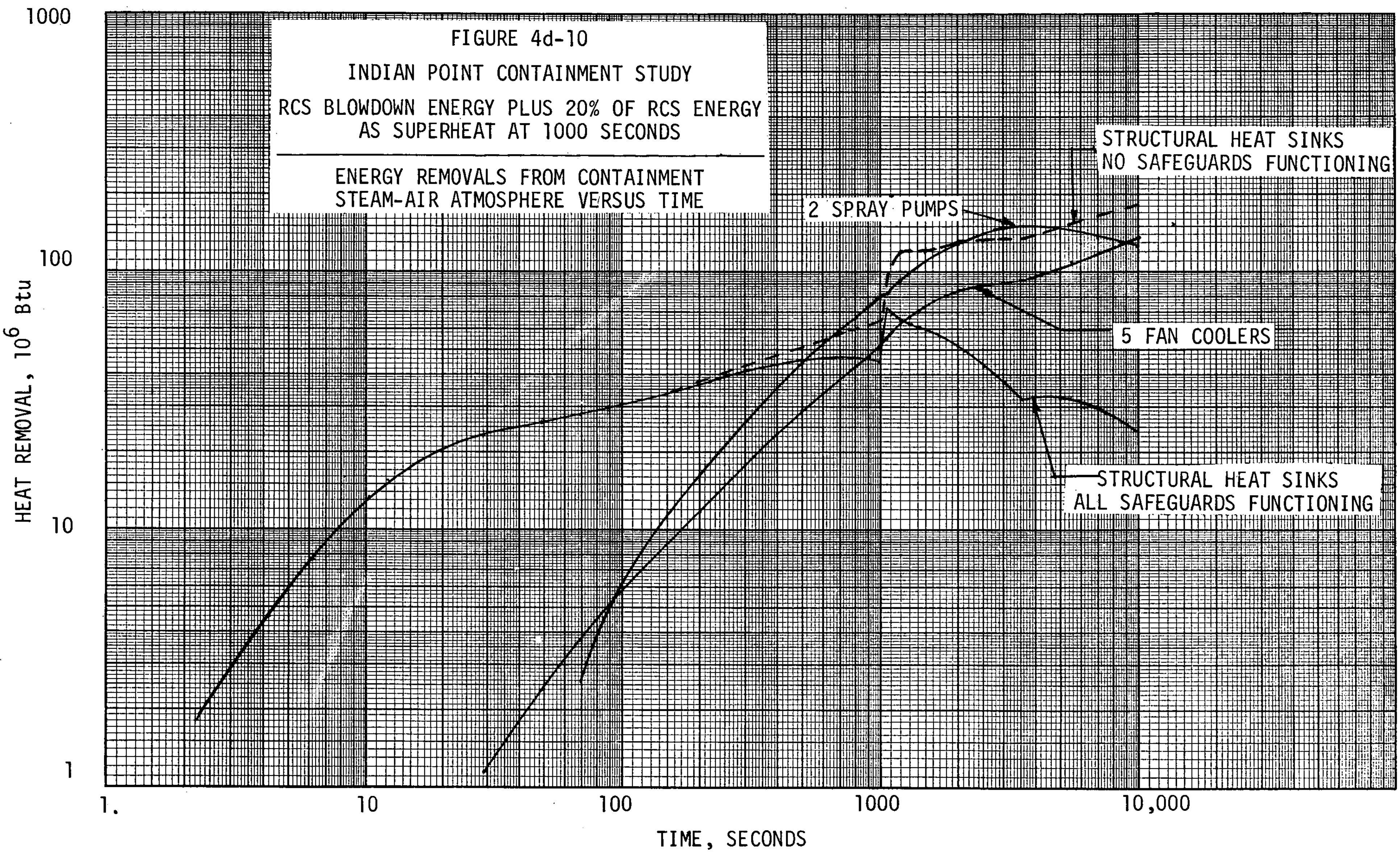


FIGURE 4d-10
INDIAN POINT CONTAINMENT STUDY
RCS BLOWDOWN ENERGY PLUS 20% OF RCS ENERGY AS SUPERHEAT AT 1000 SECONDS
ENERGY REMOVALS FROM CONTAINMENT STEAM-AIR ATMOSPHERE VERSUS TIME
STRUCTURAL HEAT SINKS NO SAFEGUARDS FUNCTIONING
2 SPRAY PUMPS
5 FAN COOLERS
STRUCTURAL HEAT SINKS ALL SAFEGUARDS FUNCTIONING
1000
100
10
1
HEAT REMOVAL, 10^6 Btu
1.
10
100
1000
10,000
TIME, SECONDS

4. The PSAR contains pressure transient curves following the double-ended rupture of a primary coolant pipe for conditions wherein all engineered safeguards function, no engineered safeguards function, and the engineered safeguards function on emergency power only. In order that we may assess the margin of safety provided by these systems in the containment design and the relative effectiveness of each engineered safeguard, provide the following information:

   f. Provide the containment pressure transient curves following the MCA for the cases in which the only engineered safeguards assumed to function are: (1) one high head and one low head safety injection pump, (2) one containment spray pump, and (3) four containment air recirculation coolers.

Answer

1) Only One High Head and One Low Head Safety Injection Pump Operate

The pressure transient for this case is shown on Figure 4f-1. During the blowdown of the reactor coolant system, the containment pressure rises to 39.2 psig at 10 seconds. Following blowdown, the containment pressure slowly rises to a peak of 44.8 psig at 280 seconds as part of the safety injection water is boiled away during the core cooldown transient. Once the core is cooled and covered with water, the safety injection water continues to remove the core decay heat, but is not heated sufficiently to flash when it spills into the containment through the break. The containment structure continues to condense steam, and the pressure slowly falls. At 4150 seconds, the refueling water storage tank is emptied. Water is then drawn from the mixture of reactor coolant and spilled safety injection water in the containment sump. This recirculation water is cooled in a single recirculation heat exchanger, then pumped back to the core at the 3000 gpm capacity of the residual heat removal pump via the core deluge and safety injection lines. The recirculation heat removal is less than the core decay heat for much of the transient shown on Figure 4f-1, but additional steam is not formed because the containment sump water is still subcooled. Structural heat sinks and the recirculation through the residual heat exchanger continue to reduce containment pressure until the analysis is stopped at about 3 hours.

2) <u>One Containment Spray Pump Operates</u>

The pressure transient for this case in shown in Figure 4f-2. The blowdown portion of the transient is identical with case 4f-1 above. At 45 seconds, one containment spray pump is started and pressure decreases. However, the hotter regions of the core begin melting at 145 seconds due to decay heat and metal-water reaction energy. It is assumed in this analysis that the vessel is filled with saturated water to the bottom of the core. The resulting steam generation as the core sections drop causes containment pressure to rise to 40.5 psig at 575 seconds. At this time, all the water in the vessel has been boiled away, and the hot core begins melting through the reactor vessel lower head. Simultaneously, while no residual heat generated in the core is being transferred into the containment, spray is reducing the pressure. At 2300 seconds, the core melts through the vessel and drops into the containment sump water. It is assumed that the core and vessel stored heat is removed in 100 seconds by the formation of steam. This results in a pressure peak of 57.6 psig. After the core is cooled to equilibrium with the water in the bottom of the containment, the further residual heat is transferred directly to the sump water as it is generated. The sump water does not begin to boil until 3800 seconds because of the sub-cooling effect provided by the air partial pressure. At 7450 seconds, the supply of refueling water is exhausted. Starting at that time, sump water is drawn through a single recirculation heat exchanger, then sprayed back into the containment steam-air phase by the spray pump. The effect of spraying recirculation water rather than cold refueling water is to reduce the rate of containment depressurization. If the recirculation spray were not initiated at this time, containment pressure would rise again as core residual heat boils away sump water.

3) <u>Only Four Containment Fan Coolers Operate</u>

The heat removal capacity of four fan coolers is equivalent to that of one spray pump, therefore, the pressure transient for this case is nearly identical with that shown on Figure 4f-2 with one spray operating. The deviation starts only after the pressure peak of 57.6 psig at 2400 seconds. Neither safety injection nor spray has been running, and the reactor coolant is the only water in the containment sump. Boiling of the sump water by core decay heat occurs earlier, hence the pressure does not drop as rapidly as in the spray case. With neither a spray nor safety injection pump available, no credit is taken for recirculation cooling of the sump water. However, the fans continue to reduce containment pressure, condensing the vapor formed by boil-off of sump water by core residual heat generation.

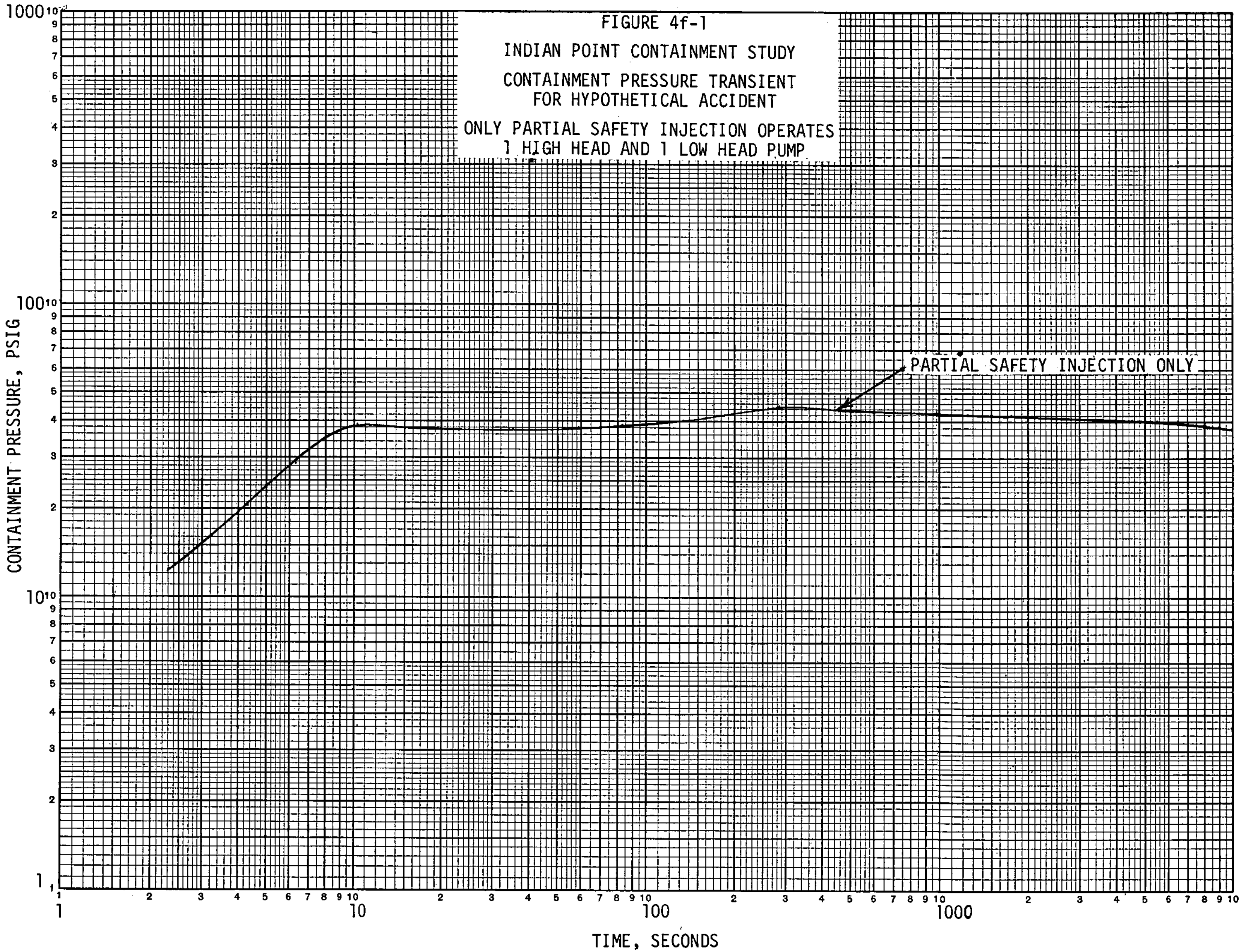


FIGURE 4f-1
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENT
FOR HYPOTHETICAL ACCIDENT
ONLY PARTIAL SAFETY INJECTION OPERATES
1 HIGH HEAD AND 1 LOW HEAD PUMP
PARTIAL SAFETY INJECTION ONLY
CONTAINMENT PRESSURE, PSIG
1000
100
10
1
1
10
100
1000
TIME, SECONDS

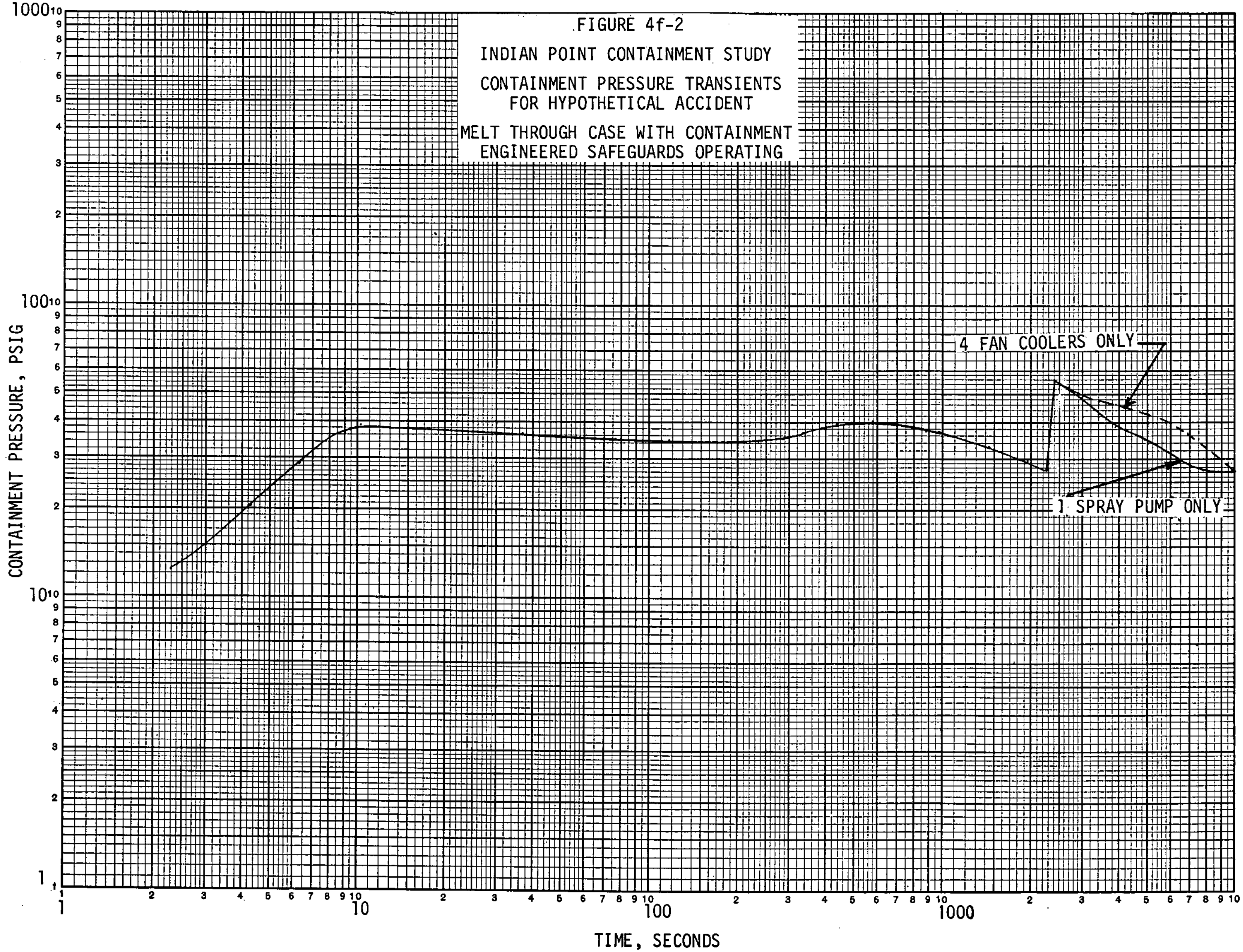


FIGURE 4f-2
INDIAN POINT CONTAINMENT STUDY
CONTAINMENT PRESSURE TRANSIENTS
FOR HYPOTHETICAL ACCIDENT
MELT THROUGH CASE WITH CONTAINMENT
ENGINEERED SAFEGUARDS OPERATING
4 FAN COOLERS ONLY
1 SPRAY PUMP ONLY
CONTAINMENT PRESSURE, PSIG
TIME, SECONDS
1000
100
10
1
1
10
100
1000

5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   a. Summarize the peak heat flux factor (F_q), peak enthalpy rise factor ($F_{\Delta H}$), and the peak axial flux factor (F_z) for the following situations:

      (1) Nominal conditions for worst time in core life (using worst expected rod conditions).

      (2) Design conditions for worst time in core life (no engineering factors).

      (3) Hot channel conditions for worst time in life (with engineering factors).

<u>Answer</u>

(1) Nominal conditions for worst time in core life (using worst expected rod conditions)

$$F_{\Delta H}^{T} = F_{\Delta H}^{N} F_{\Delta H}^{HHA} = (1.45)(1.04) = 1.51$$

$$F_q^T = F_q^N = 2.30$$

$$F_z = 2.30/1.45 = 1.59$$

(2) Design conditions for worst time in core life (no engineering factors)

$$F_{\Delta H}^{T} = F_{\Delta H}^{N} F_{\Delta H}^{HHA} = (1.75)(1.04) = 1.82$$

$$F_q^T = F_q^N = 3.12$$

$$F_z = 3.12/1.75 = 1.78$$

(3) Hot channel conditions for worst time in life (with engineering factors)

$$F_{\Delta H}^{T} = F_{\Delta H}^{N} \; F_{\Delta H}^{E} = (1.75)\,(1.075) = 1.88$$

$$F_{q}^{T} = F_{q}^{N} \; F_{q}^{E} = (3.12)\,(1.04) = 3.25$$

$$F_{z} = 3.12/1.75 = 1.78$$

<u>Definitions</u>

$F_{\Delta H}^{T}$ = Total enthalpy rise factor

$F_{\Delta H}^{N}$ = Nuclear radial peak to average factor

$F_{\Delta H}^{HHA}$ = Hydrodynamic hot array enthalpy rise factor

F_{q}^{T} = Total heat flux factor

F_{z} = Nuclear axial peak to average factor

F_{q}^{N} = $(F_{z})\,(F_{\Delta H}^{N})$

$F_{\Delta H}^{E}$ = Enthalpy rise engineering factor

F_{q}^{E} = Heat flux engineering factor

5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   b. Supply a distribution curve showing the fraction of the core operating above various power levels with their corresponding DNB ratios for condition (a-1).

Answer

The distribution curve with the corresponding DNB ratios is illustrated in Figure 5-1. The following conditions were employed:

Power = 100%

T_{in} = 543°F

P = 2250 psia

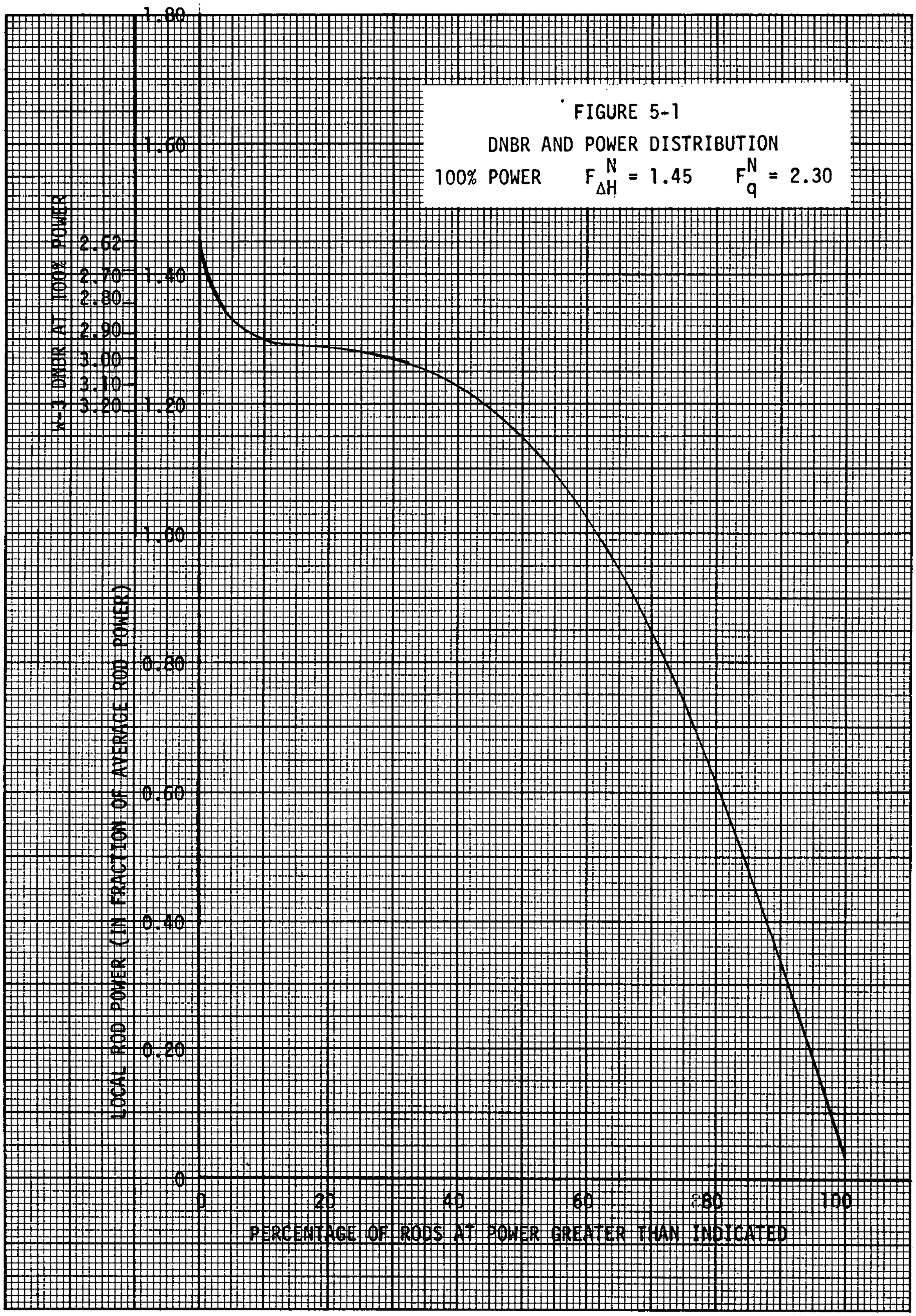
(a-1) conditions

The above conditions yielded local enthalpies at the location of minimum DNB ratio which were outside of the lower limit of the W-3 correlation (-15% quality) for local to average rod powers less than 1.45. Therefore, the DNB ratios for local to average rod powers less than 1.45 were conservatively calculated using the burnout heat flux obtained at -15% quality.

FIGURE 5-1

DNBR AND POWER DISTRIBUTION

100% POWER $F^N_{\Delta H} = 1.45$ $F^N_q = 2.30$



5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   c. For condition (a-2), provide the total number of fuel rods that are within 90% of the design peak power level and the corresponding DNB ratios (include the effects of instrument errors).

Answer

As demonstrated in Figure 5-2, there areapproximately 800 fuel rods out of a total of 39,372 fuel rods that are within 90% of the peak power level. DNB ratios are shown in Figure 5-2. The following conditions were used:

Power = 102%

T_{in} = 547°F

P = 2220 psia

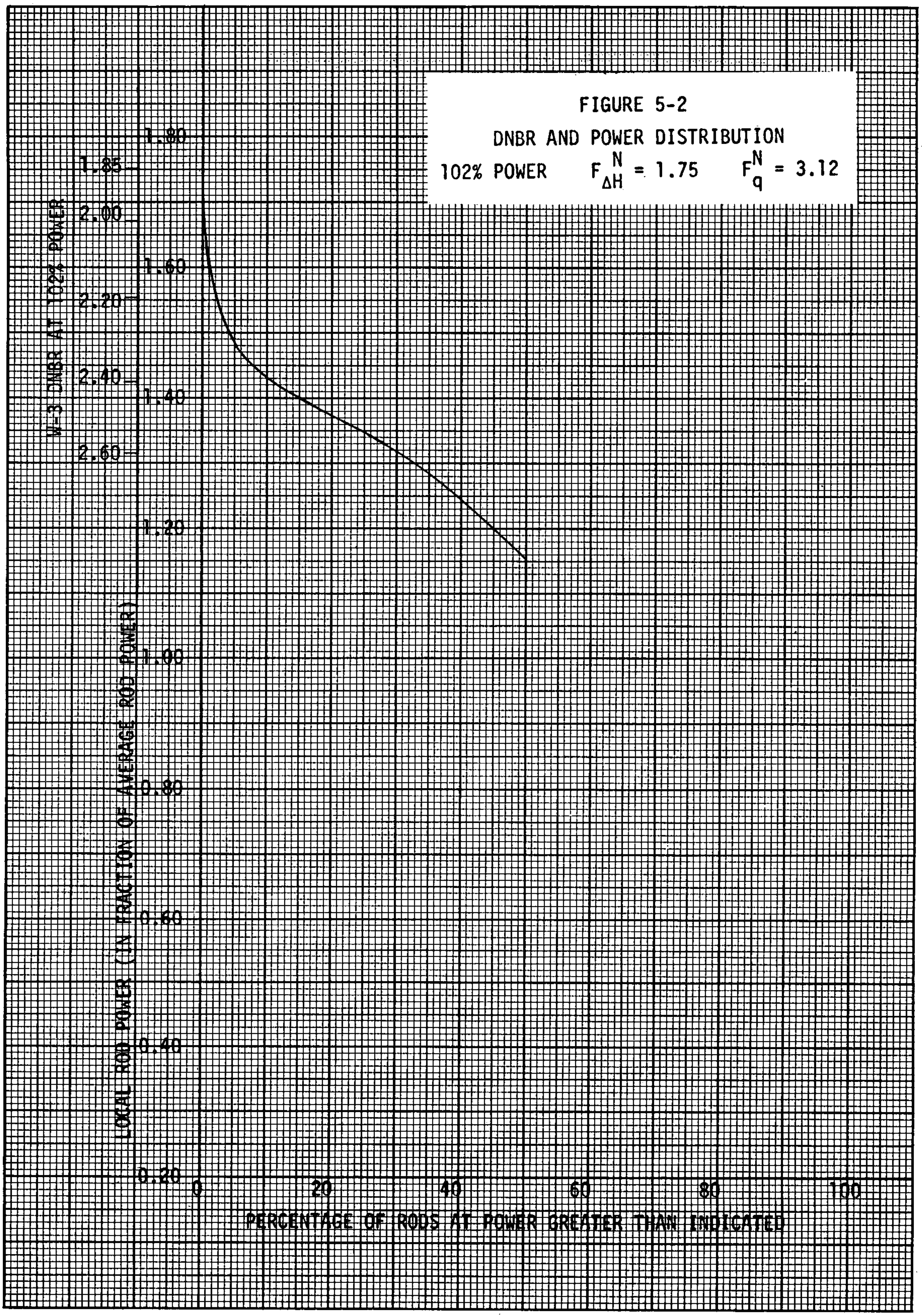
(a-2) conditions

The above conditions yielded local enthalpies at the location of minimum DNB ratio which were outside of the lower limit of the W-3 correlation (-15% quality) for local to average rod powers less than 1.62. Therefore, the DNB ratios for local to average rod powers less than 1.62 were conservatively using the burnout heat flux obtained at - 15% quality.

FIGURE 5-2

DNBR AND POWER DISTRIBUTION

102% POWER $F^N_{\Delta H} = 1.75$ $F^N_q = 3.12$



5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   d) Repeat parts (b) and (c) for the overpower condition. If any channel has bulk boiling, or would require less than 5% additional power to cause boiling, tabulate these results indicating at what distance from the core top boiling ensues.

Answer

Figure 5-3 specifies the distribution curve with the corresponding DNB ratios, for the following conditions:

$$\begin{aligned} \text{Power} &= 106.5\% \\ T_{in} &= 543°\text{F} \\ P &= 2250 \text{ psia} \end{aligned}$$

(a-1) conditions

The above conditions yielded local enthalpies at the location of minimum DNB ratio which were outside of the lower limit of the W-3 correlation (-15% quality) for local to average rod powers less than 1.45. Therefore, the DNB ratios for local to average rod powers less than 1.45 were conservatively calculated using the burnout heat flux obtained at 15% quality.

Bulk boiling will not occur in the hottest channel for either the 106.5% or the 112% overpower condition when evaluated at the above conditions.

Figure 5-4 identifies the distribution curve with the corresponding DNB ratios for the following conditions:

$$\begin{aligned} \text{Power} &= 112\% \\ T_{in} &= 559°\text{F} \\ P &= 2350 \text{ psia} \end{aligned}$$

(a-2) conditions

The above conditions yielded local enthalpies at the location of DNB which were outside of the lower limit of the W-3 correlation (-15% quality) for local to average rod powers less than 1.51. Therefore, the DNB ratios for local to average rod powers less than 1.51 were conservatively calculated using the burnout heat flux obtained at 15% quality.

Bulk boiling will not occur in the hottest channel for the 112% overpower condition when evaluated at the above conditions. Bulk boiling commences at 10.2 feet from the bottom of the core for the 118% overpower condition.

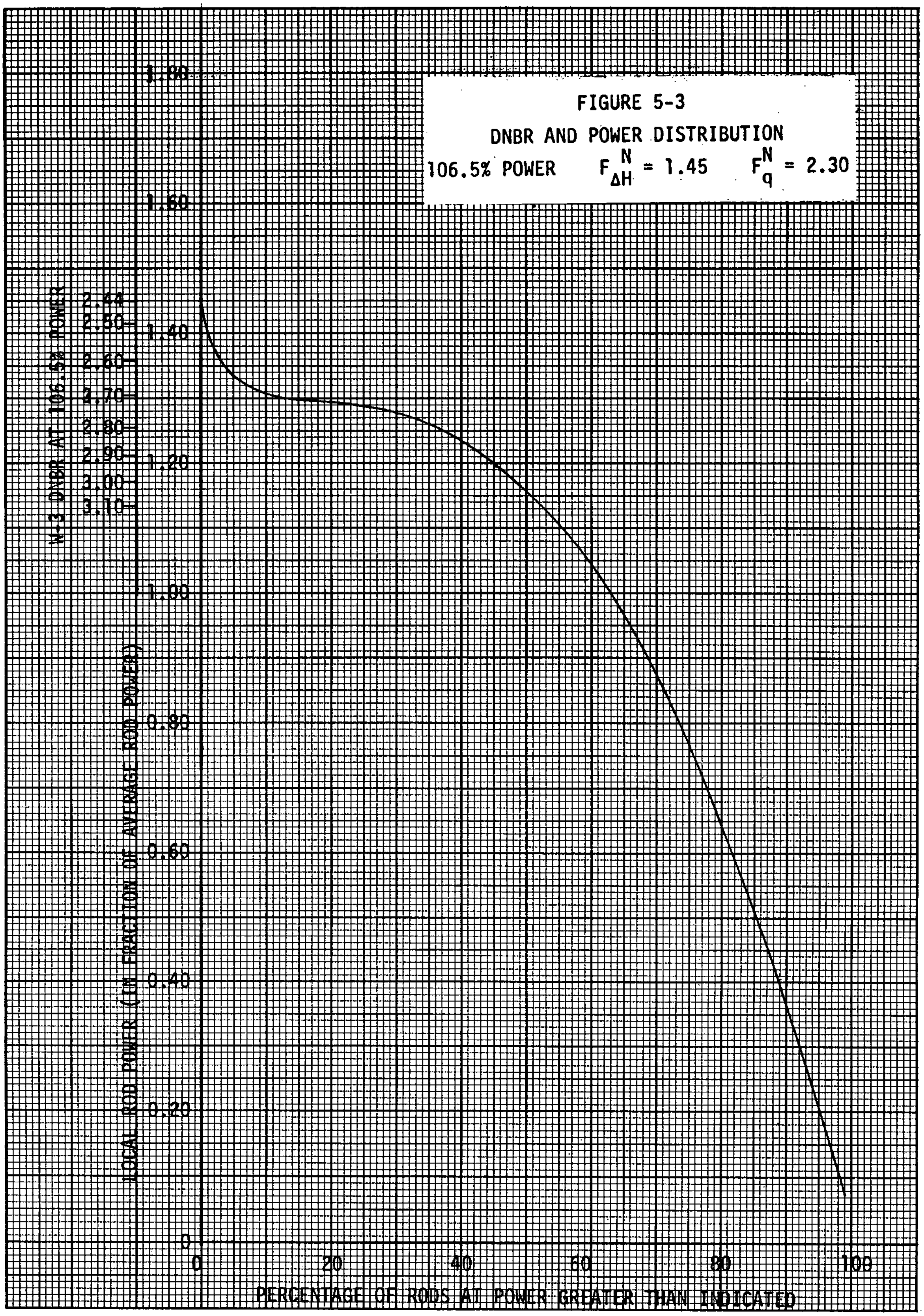


FIGURE 5-3
DNBR AND POWER DISTRIBUTION
106.5% POWER $F_{\Delta H}^{N} = 1.45$ $F_{q}^{N} = 2.30$
N-3 DNBR AT 106.5% POWER
2.44
2.50
2.60
2.70
2.80
2.90
3.00
3.10
LOCAL ROD POWER (IN FRACTION OF AVERAGE ROD POWER)
1.80
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
0
0
20
40
60
80
100
PERCENTAGE OF RODS AT POWER GREATER THAN INDICATED

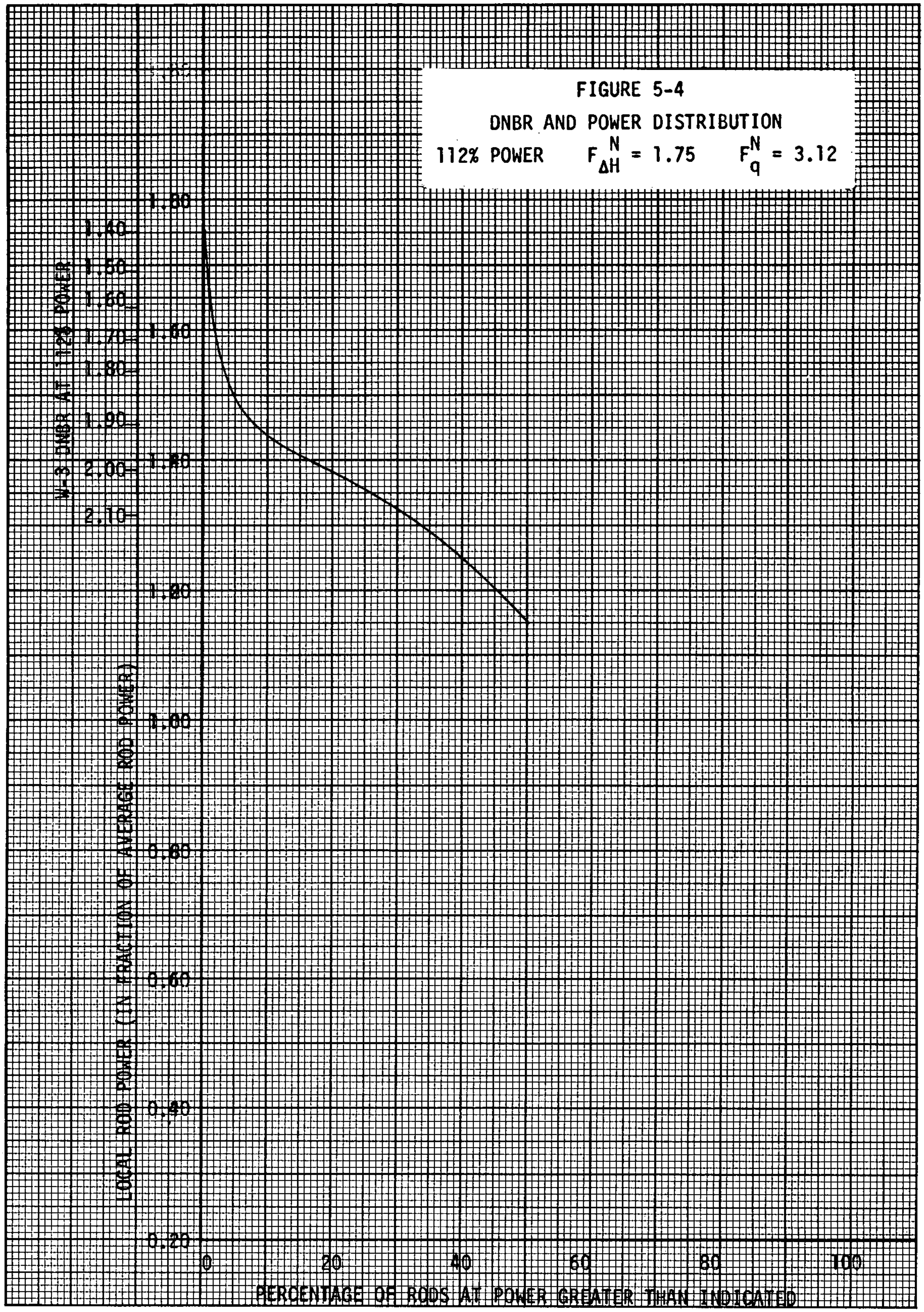


FIGURE 5-4
DNBR AND POWER DISTRIBUTION
112% POWER $F^N_{\Delta H}$ = 1.75 F^N_q = 3.12
W-3 DNBR AT 112% POWER
1.40
1.50
1.60
1.70
1.80
1.90
2.00
2.10
LOCAL ROD POWER (IN FRACTION OF AVERAGE ROD POWER)
1.80
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
0
20
40
60
80
100
PERCENTAGE OF RODS AT POWER GREATER THAN INDICATED

5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   e. For a hypothetical 125% overpower condition, estimate whether any fuel rods approach design limits (e.g., DNB or center fuel melting).

Answer

For a hypothetical 125% overpower condition, the center fuel temperature is 4600°F, and the minimum DNBR is 1.22. The following conditions were utilized:

$$T_{in} = 547°F$$
$$P = 2220 \text{ psia}$$
$$F_H^N = 1.75$$
$$F_H^T = 1.82$$
$$F_q^N = 3.11$$
$$F_q^T = 3.11$$

It should be noted, however, that the reactor protection system limits the reactor overpower to 112%.

To estimate the number of rods which might reach DNB, a convolution procedure is employed. In this procedure the product of the number of rods experiencing a given minimum DNBR, and the probability of reaching DNB for the above given DNBR, is summed over the entire core. The number of rods that might possibly reach DNB at the above hypothetical conditions is 36.5 or effectively, 37 out of 39,372- the total number of rods.

5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   f. For the hot channel positions, provide the DNB, exit quality, and center fuel temperature at 100%, 110% and 125% for the worst design conditions. In addition, arbitrarily raise the $F_{\Delta H}$ and F_z factors by 10% and tabulate as above for each condition.

<u>Answer</u>

Case	Power %	$F_{\Delta H}^T$	F_q^T	DNBR	Exit Quality %	Center Fuel Temperature °F
1	100	1.88	3.25	1.80	0	4150
2	110	1.88	3.25	1.50	0	4350
3	125	1.88	3.25	1.11	5.4	4650
4	100	1.88	3.57	1.64	0	4350
5	110	1.88	3.57	1.37	0	4600
6	125	1.88	3.57	1.01	5.4	4950
7	100	2.07	3.57	1.51	0	4350
8	110	2.07	3.57	1.21	3.9	4600
9	125	2.07	3.57	0.83	10.1	4950

The following conditions were employed for all cases:

$$T_{in} = 547°F$$
$$P = 2220 \text{ psia}$$

It should be noted that the present reactor protection system limits the overpower to 112%. In addition, the design values of F_q^T and $F_{\Delta H}^T$ were raised arbitrarily 10% to answer this question. Exceeding these design values is not consistent with the present design.

5. The maximum specific power for the proposed fuel rods is higher than in any currently licensed reactor. In order to assess the conservatism of the proposed design, please provide the following information:

   g. For the engineering hot channel factors, indicate the 2σ and 3σ values of the various statistical components before they are combined into one overall factor.

Answer

The heat flux engineering factor, F_q^E, is 1.04, which represents the 3σ value. The 2σ value is 1.027.

The enthalpy rise engineering factor, $F_{\Delta H}^E$, is 1.075. This factor is composed of the following subfactors:

Pellet Diameter, Density, Enrichment Rod Diameter, Pitch and Bowing	1.08
Inlet Flow Maldistribution	1.03
Flow Redistribution	1.05
Flow Mixing	0.92
Resulting $F_{\Delta H}^E$	1.075

The subfactor relating to the fuel pellet is statistical in nature and has a 2σ value of 1.053 and a 3σ value of 1.08. However, the other subfactors are non-statistical in nature; therefore, there is no standard deviation associated with these subfactors.

6. Provide a diagram of your conceptual layout of the internal air recirculation and iodine filtration systems showing the relative location of the input and exhaust ducts, fans, heating and cooling units, demisters, and charcoal filters. State and justify the estimated temperature and relative humidity of the containment atmosphere at each of the above locations for the aniticipated conditions following the double-ended rupture of a primary coolant pipe. Describe the systems (including redundancy) provided to prevent ignition of the charcoal filters, and discuss the potential effects on containment pressure and off-site doses if total ignition of the filters is assumed. What experimental evidence can be given to justify the elemental and organic iodine removal efficiencies assumed in the PSAR? What is the basis for the selection of the fraction of organic iodine initially present, and for its growth rate throughout the accident? Also, what fraction of the total gaseous activity is assumed to be present in the fuel gaps?

Answer

The recirculation ventilating system consists of five (5) air handling units (four of the five units will be required to operate during an accident) each supplying air to a common distribution header. Branch ducts form this header proportion and direct air to areas below and above the operating floor as required.

Each air handling unit consists of the following components arranged so that the ventilating air, during normal operation, flows through the unit in the following sequence: demister, cooling coil, roughing filter, absolute filter, fan and distribution header. In the event of an accident, the air will continue this same flow path except that, after passing through the fan, it will be diverted to a compartment containing charcoal filters before entering the distribution header. During normal operation, this compartment is isolated from the rest of the unit by "bubble" tight dampers designed to prevent air leakage into the compartment and possible deterioration of the charcoal filter beds. These dampers operate on a high containment pressure signal and are spring loaded for "fail-safe" operation.

All of the air handling units are located on the intermediate floor between the Containment Building outer wall and the primary compartment shield wall. The distribution header and service water cooling piping are also located outside the shield wall. This arrangement provides missile protection for all components.

The location of the distribution ductwork outlets, with reference to the location of the ventilating unit return inlets, ensures that the air will be directed to areas requiring ventilation before returning to the units. A conceptual sketch of the ventilation system layout is shown on Figure 6.

In addition to ventilating areas inside the periphery of the shield wall, the distribution system also includes two branch ducts located at opposite extremes of the containment wall for ventilating the dome portion of the containment. These ducts will be provided with nozzles and will extend upward along the containment wall as required to permit throw of air from nozzles to reach highest point in containment and assure that the discharge air will mix with the atmosphere.

The air discharge inside the periphery of the shield wall will circulate and rise above the operating floor through openings around the steam generators where it will mix with air displaced from the dome area. This mixture will return to the ventilating units through floor grating located at the operating floor directly above each ventilating unit inlet. The temperature of this air will be essentially the ambient existing in the containment vessel.

The steam-air mixture from the containment entering the demister will be at approximately 271°F and have a density of 0.175 pounds per cubic foot. The demister will remove all entrained moisture or fog but not vapor. The fluid will therefore leave the demister and enter the cooling coil at approximately 271°F and saturated (100% R.H.) condition. Part of the watervapor will condense in the cooling coil, and the air leaving the unit will be saturated at a temperature somewhat below 271°F.

The fluid will remain in this condition as it flows through the roughing filter, absolute filter and into the fan. At this point it will pick up some sensible heat from the fan and fan motor before flowing through the charcoal filters and then into the distribution header. This sensible heat will increase the dry-bulb temperature slightly above 271°F and will decrease the relative humidity slightly below 100 percent.

In addition to the above air handling units, two small recirculating fan systems each complete with roughing filters, absolute filters and charcoal filters are provided for iodine removal if required for access during normal operation. One fan system will operate as required, the other will serve as a standby unit.

DESIGN CRITERIA FOR THE SYSTEM

The containment ventilation system is designed to be capable of operating after a loss of coolant accident which results in a 47 psig containment pressure. In addition, every component of these systems is capable of withstanding without impairing operability, a pressure of 1.5 times the design pressure and the associated temperature of the air-vapor mixture (298 F) for a period of 1 hour.

The following design criteria are common and applicable to charcoal filter assemblies, absolute and roughing filter assemblies, moisture eliminators and cooling coils for each of the five air handling assemblies.

1. Mimimum design flow rate per assembly 65,000 cfm
   Number of units required to operate four out of five
3. Fluid description; saturated steam and air mixture, 271 F at 47 psig, density 0.175 lb/cu. ft.
4. All components and their supports shall meet the requirement for Class I (Seismic) structures.
5. All components shall be capable of withstanding or shall be protected from differential pressures which may occur during the rapid pressure rise to 47 psig in ten (10) seconds.
6. Each component shall be mounted to isolate if from fan vibration.

In addition to the design criteria common the the components stated above, additional design criteria applicable to specific components are as follows:

Charcoal Filters

1. Mimimum efficiency for absorption of elemental iodine shall be 99.0%.
2. Filters shall be activated and impregnated coconut shell charcoal.
3. Filter units shall be supported on vibration isolation mountings and duct connections shall be flexible to prevent transmission to duce vibration to the filter units.
4. Filter plenums shall be provided with high temperature detectors (redundant in each plenum) and associated alarms in the control room.

5. Each filter unit shall be provided with a spray system for dousing upon signal of high temperature. The borated spray water shall be provided from the containment spray supply header. Initiation of dousing shall be manual from the control room. The dousing system will be testable as part of the containment spray system.

6. Filter gaskets shall be capable of withstanding a temperature of 300 F.

7. Charcoal filter unit cells shall be removable for testing.

8. Filter bank shall be capable of periodic leak tightness testing using a suitable aerosol.

9. The velocity for each unit shall be 250 FPM.

10. Filters shall be installed in stainless steel frames securely sealed against the gasket material.

Absolute Filters

1. Minimum efficiency with particles 0.3 microns and larger shall be 99.97%.

2. Filter media shall be of the self-extinguishing type.

3. Filter cells shall be installed in stainless steel frames securely sealed against a gasket.

4. The gasket shall be capable of withstanding a temperature of 300 F.

Moisture Eliminators (Demister)

With air entrained moisture content of 0.35 lb/1000 cfm, the leaving fluid shall have essentially zero moisture content.

Cooling Coils

1. Cooling duty, 46,8000,000 Btu/hr per air handling assembly at saturation conditions (47 psig, 271 F).

2. Design pressure of coil, 150 psig.

3. Coils shall be provided with adequate drain pans and drain piping to prevent flooding. Water will be directed to sump.

4. Coils shall withstand an external pressure of 70.5 psig without damage.

Fans

1. Characteristic curve for fan shall provide a minimum flow rate of 65,000 cfm when operating against the system resistance existing during the accident condition.

2. Fan shall be centrifugal, non-overloading, direct drive type.

3. Shaft bearings and motor coupling shall be suitable for operation in the temperature and pressure environment existing during accident.

4. Fan assembly and support shall be designed as a Class I (Seismic) structure.

5. Fan shall be capable of withstanding or shall be protected from differential pressures which may occur during a rapid pressure rise to 47 psig in 10 seconds.

6. Fan rotating assembly shall be statically and dynamically balanced.

7. All parts in contact with containment fluid shall be suitably protected against corrosion.

8. The fan assembly shall be supported by vibration isolators.

Motors

1. Rating of motor shall be suitable to match the power requirements of the fan during the accident condition described below.

2. Electrical insulation and bearings shall be suitable for the temperature and humidity conditions during the accident as described below.

3. Enclosure of motor shall be of a special design which will withstand the pressure and temperature condition during the accident without impairing operation.

4. Motors shall be statically and dynamically balanced.

5. Accident conditions will be described in the specification for the motor as follows:

   a. Motor shall run for 48 hours at the load required by the fan in an atmosphere consisting of an air water vapor mixture at 47 psig and 271 F.

   b. The load on the motor will gradually decrease from the peak condition of (a.) during the first two hours to a lower load equivalent to operation of the fan in an atmosphere with 5 to 10 psi water vapor pressure.

5. The motor shall operate for ten days in the atmosphere with 5 to 10 psi water vapor pressure.

Ducts

1. Ducts shall be designed, constructed and supported to meet the requirements of Class I (Seismic) structures.

2. Ducts shall be capable of withstanding the differential pressure developed during the hypothetical loss-of-coolant accident, or shall be provided with pressure devices to prevent an excessive differential pressure.

3. Ducts shall be constructed of corrosion resistant material.

4. Ducts shall be capable of withstanding the maximum temperature and shall be supported to accommodate expansion due to temperature changes occurring during an accident.

5. Ducts shall be of welded construction except where flanged joints are used near components. Joints shall be provided with gaskets suitable for temperatures to 300 F.

Louver Damper

1. Shall be provided with gasketed closures.

2. Louver materials in contact with containment fluid shall be corrosion resistant.

3. Gaskets and bearings shall be suitable for a temperature of 300 F.

Butterfly Valves

1. Shut off shall be bubble tight.

2. Seats shall be renewable soft material suitable for temperature of 300 F.

3. Shaft and disc seating surface shall be corrosion resistant material.

Louver & Valve Operators

1. Operators shall be suitable for operation during the accident under the conditions of 47 psig pressure and 271 F temperature. The operator shall not be adversely affected by a pressure of 70.5 psi and a temperature of 300 F.

2. Upon loss of electric or pneumatic power the operator shall fail in the position required for post-accident cooling and filtration.

3. The operator shall be provided with position indicating devices which will indicate in the control room.

4. The operator shall be capable of remote operation from the control roon.

Instrumentation and Control

1. Local flow and temperature indication, outside containment, for service water to each cooling unit shall be provided. Abnormal flow and temperature alarms shall be provided in the control room.

2. Vane switches to monitor existence of flow from air handling unit and through charcoal filter plenum shall be provided. Abnormal flow alarms shall be provided in control room. These channels shall be redundant and operable during the accident.

3. High temperature detectors and alarms are provided for each charcoal filter assembly. These channels shall be redundant.

4. Upon receipt of high containment pressure signal, dampers and valves in ventilation will be tripped to the accident position.

5. The service water discharge from the containment shall be monitored for radioactivity and the monitor shall function during the loss of coolant accident.

Each of the five charcoal filter assemblies is equipped with two independent temperature sensing devices. High temperature in either of the two sensors produces an alarm in the control room. Lights are provided on the control panel to indicate which filter assembly has a high temperature. The sensors will be arranged in the assemblies to detect the highest assembly temperature with or without the fans operating. The temperature alarm initiation set point will be well below the carbon kindling temperature and above the calculated post loss-of-coolant containment atmosphere conditions.

A charcoal dousing spray unit is provided for each filter assembly as part of the containment spray system. The filter dousing spray system is shown on Figure 6-1, Revision 1 attached to Question 9 of this supplement. There are two separate containment spray headers, and either one can supply water to the filter dousing units. A missile shielded cross connection between the two spray headers supplies water to the filter dousing units. Check valves in the cross over maintain independence of the two spray headers by permitting flow from either containment spray header to the filter dousing system while preventing flow to the other containment spray header.

Containment spray is automatically actuated and will be running in the event of a loss-of-coolant accident. In the event of a high temperature alarm in a filter unit, the operator initiates filter dousing by actuating the parallel-connected isolation valves for each filter assembly. Because of the piping arrangement described above, either of the two spray pumps can be started to feed the dousing lines. The flow to the dousing unit is sized to compensate for the decay heat of the absorbed iodine and the heat of combustion of the carbon filter. The system is designed so containment spray at reduced flow can continue simultaneously with filter dousing. This has the secondary effect of dissipating any energy that could be released to the containment by the filters.

The potential effect on containment pressure assuming total combustion of one of the five charcoal filter units has been analyzed by treating the charcoal as a fuel bed. A filter unit contains 65 filter sections, each holding about 140 pounds of coconut charcoal. Although the filter will not ignite with the fan running, it was assumed that combustion could not occur at a higher rate than is possible to supply oxygen in the air-steam mixture flowing at 65,000 cfm at accident conditions. At 20°C and atmospheric pressure, the heat of combustion of water-free charcoal is 8.1 Kg. cal per gram. Complete combustion of the charcoal in one filter unit would produce a maximum heat input of 132×10^6 Btu. in forming CO_2.

Assuming a containment pressure of 47 psig when the charcoal ignites, the initial oxygen flow rate is 34.9 lb-mols per minute which would produce heat at 102,000 Btu. per second assuming complete combustion of the charcoal to carbon dioxide. As the combustion process decreases the oxygen content of the containment air-steam mixture, the oxygen flow to the filter bed would decrease with a corresponding decrease in rate of heat generation. The entire filter bed would be burned in approximately 1890 seconds.

A peak containment pressure of 60 psig was calculated for the hypothetical loss of coolant accident assuming operation of the safety injection system on diesel generated power, no fan cooling (the fans are assumed to run to load the filter units with iodine), and no spray, with complete combustion of hydrogen from the zirconium-water reaction as it is released plus combustion of one filter bed. This pressure is 15 psi above the 45 psig peak calculated for the same accident and the same post-accident system operation, but with no filter bed combustion.

For the above case, the containment pressure remained an average of about 5 psig above the penetration and weld channel pressurization system pressure (∿50 psig) for 2 1/2 hours. The maximum leak rate from the system for this period would be about 0.1% x $\frac{5}{47}$ or 0.01% per day. With the three remaining fan filter units in operation to remove the iodine released from the burned filter 25% of the core inventory is assumed to be on the 4 units. The potential exposure at the site boundary is about 1.35 rem to the thyroid.

Experimental evidence of the trapping behavior of iodine as summarized in the following paragraphs, supports the assumptions of 90% and 70% effective removal of elemental and organic iodines, respectively.

1. Elemental Iodine:

Tests at Oak Ridge National Laboratory reported in CF-60-11-39, yielded results with air and with steam (up to 101.9% relative humidity). Removal was consistently better than 99.9%. Only minor variations in performace were noted as conditions were varied: e.g. velocity from 24 to 75 fpm; temperature from 170 to 240°F; atmosphere from air to saturated and superheated steam.

Experiments were conducted at Savannah River and reported in TID-17548, in which elemental iodine trapping was measured in air, steam, and fog at temperatures for 160 to 270°F and atmospheric pressure. The charcoal test beds were pre-saturated with water at the test condition. Results were consistently better than 99.99% removal.

Iodine released from fuel specimens melted in the Nuclear Safety Pilot Plant at Oak Ridge was largely present as elemental iodine when the melt was perforned under oxidizing condition. Operation of a 10 in. thick charcoal filter in a recirculation loop resulted in iodine removal with an observed efficiency of 99%. Conditions were initially 25 psi saturated steam (∿240°F) during release, decreasing to 1 atm (<212°F) but with continued saturation humidity during filter operation.

It is concluded from these experiments and from other work summarized in the available literature (1), that failure to achieve >99% removal of elemental iodine with ordinary activated charcoal could only be caused by:

a. Mechanical defects (e.g. voids or bypass leaks in the filter)
b. Gross flooding waterlogging
c. Temperatures high enough to cause accelerated oxidation of the charcoal grain surface.

As discussed in the PSAR, periodic testing will detect significant voids and bypass leaks should they occur in normal service. The units will be protected against missile damage flooding and waterlogging in the accident. While operating, charcoal temperatures will not exceed those under which environmental tests have demonstrated adequate endurance and sensitivity. On the basis, the efficiency for 90% assumed for elemental iodine is believed adequately justified.

Organic Iodine

Experiments have been conducted at ORNL, and reported on ORNL-3864, in which methyl iodine (CH_3I) was exchanged with commercially impregnated charcoals. The test data, obtained at 168°F and 100% relative humidity predict an efficiency of ∿98% for a filter bed of the reference dimensions and gas velocity at these conditions.

A series of confirmatory tests is continuing at ORNL to extend the conditions of filter operation to saturated air-steam mixtures at 260°F. Preliminary results of these tests indicate that the exchange of methyl iodide with impregnated charcoal will be at least 70% effective at the accident conditions, as assumed in the PSAR. To further substantiate this analysis and to develop additional engineering data for the design of filters, a testing program utilizing full-size filter modules in a recirculating steam-air system is planned. This program will be carried out by Westinghouse during the early stages of the design and construction of Indian Point Unit #2. Life testing of filter modules after prolonged storage at containment ambient conditions will be performed to ascertain the effects (if any), of aging of the impregnated charcoal.

It is not expected that organic iodine would be liberated from the fuel at meltdown. This conclusion is based on the absence of indications of such release in in-pile fuel meltdown experiments conducted by Oak Ridge National Laboratory. Nevertheless, it was conservatively assumed for analysis that a fraction of the fuel inventory of iodine is in the organic form. The fraction assumed (0.2%) was justified on page 12-36 of the PSAR, on the basis that in the region of the fuel rod where conditions would be most favorable for the existence of organic iodine, the rates of thermal and radiolytic decomposition would exceed the rate of replenishment.

The more plausible mechanism for organic iodine formation is by reaction of elemental iodine in an absorbed state on organic-contaminated surfaces. Whether limited by diffusion to the surface or by the reaction rate of absorbed iodine, the resulting fractional conversion of airborne iodine per unit time is proportional to the surface to volume ratio of the enclosure. Therefore, observed yields of organic iodine as a function of aging time in various test enclosures were extrapolated to the Indian Point Unit #2 containment in proportion to the surface/volume ratio. These results, referenced on page 12-36 of the PSAR in no case exceeded a calculated conversion rate of 0.0035% of the atmospheric iodine per hour. A higher value of 0.05% per hour was arbitrarily assumed in the PSAR for the accident condition.

At this rate, the formation of organic iodine has a negligible effect on the consequences of containment leakage. In short, the mechanisms which are believed to have produced significant amounts of organic iodine in test facilities would be so diminished in effect by the vastly reduced relative surface to volume ratio of the plant containment, that the organic iodine component will be of minor importance.

The principal gaseous fission products are listed below, with the calculated inventory of each in the total core and in the fuel gap:

	Inventory, Curies*		
Isotope	Total Core	Fuel Gaps	Fraction in Gaps
Kr-85	0.72×10^6	0.50×10^6	0.70
Kr-85m	30×10^6	0.45c10^6	0.015
Kr-87	58×10^6	0.026×10^6	0.00045
Kr-88	84×10^6	0.075	0.00089
Xe-133	154×10^6	7.2×10^6	0.047
Xe-135	66×10^6	1.45×10^6	0.022
Total	390×10^6	9.7×10^6	0.025

*Based on 1.6 years continuous operation at 2758 Mwt.

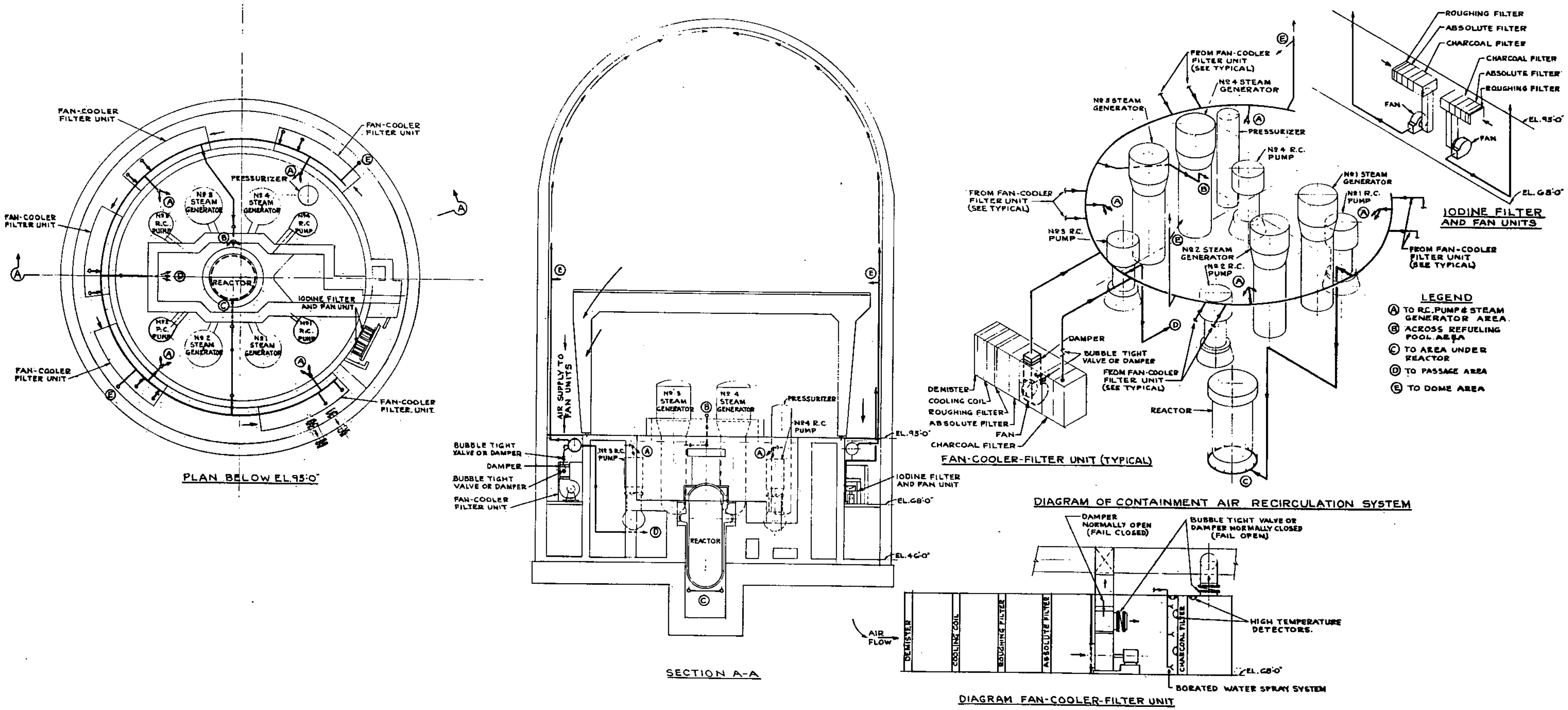


DIAGRAM OF CONTAINMENT RECIRCULATION AND IODINE FILTRATION SYSTEMS FIG. 6

7. Describe the containment penetration pressurization system under normal, abnormal, and accident conditions. Discuss the capacity of the gas supply systems, the sensitivity of the leakage monitors, and analyze system operation with various component failures. Discuss the magnitude and potential effects on containment pressure of inleakage from this system that can be tolerated during normal operation.

Answer

The function of the containment penetration pressurization system is to prevent leakage of containment air through penetrations and weld channel joints under all conditions by supplying air above containment maximum incident pressure to the positive pressure zones incorporated in the penetration and weld joint design. The system is designed to operate in this manner under all conditions - normal, abnormal and accident. A flow diagram of this system is shown on Figure 7-1 attached.

The system is supplied continuously with dried and filtered air from the 100 psig instrument air supply, and is operated at approximately 50 psig. There are two sources of air within the instrument air supply. With loss of both of these sources, each of the four pressurization system zones will continue to be supplied with air from its respective air receiver. Each air receiver will be sized to provide make up air to its respective pressurized zone for a period of four hours, based on a leakage rate of 0.2% of the containment volume per day. Should the receivers become exhausted before air service is restored, nitrogen will be supplied to each of the four pressurization zones by a bank of nitrogen cylinders. Each of these banks is sized to supply nitrogen to its zone for 24 hours, based on a total leakage rate from the entire pressurization system of 0.2% of the containment volume in 24 hours.

The preoperational leak rate test program consists of two integrated leak tests at 47 psig and a repeat of each of these tests at a lower pressure. All of these tests use the reference volume technique.

The first phase of the 47 psig test will be performed with all the pressurized zones in the penetrations and weld seam channels opened to the containment atmosphere. This test must show that the total integrated leakage from the containment through the liner weld seams and through the penetration outer containment plates is no greater than 0.1% of the containment volume per day. The second phase of this test will be essentially a repeat of the first phase but will have all pressurized zones vented to the atmosphere outside the containment. This portion of the test must demonstrate that the total leakage through the welded seams between the liner and the liner weld channels and through the penetration unit containment plates is no greater than 0.1% of the containment volume per day. Thus the air consumption of the pressurization system should not exceed 0.2% of the containment volume per day, as measured at pressurization system operating pressure.

A variable area flow sensing device will be located in each of the four headers supplying make up air to the four pressurization zones, with the integrating recorder and a high flow alarm located in the control room. The flow measurement accuracy will be within $\pm$ 1% and the reproducibility 0.3%. Since a flow of 0.2% of the containment volume per day at 47 psig is approximately 3.6 ft^3/minute, the sensitivity of the flowmeters is well within the maximum leakage of the pressurization system. The full scale pre- and post-operational integrated leak rate tests will indicate the true leak rates from the penetrations, liner welds, and liner weld joint channels.

The make up air flow to the penetrations and liner weld joint channels during normal operation is recognized to be only an indication of the potential leakage from the containment. However, it does indicate the leakage from the pressurization system, and the degree of accuracy will be increased when correlated with the results of the full scale containment leak rate tests. The criteria for selection of the operating limits of air consumption of the pressurization system are based upon the integrated containment leak rate test acceptance criterion and upon the maintenance of suitable reserves for the air supplies in the static reserviors comprised of the air receivers and nitrogen cylinders. A summary of these operating limits is as follows:

1. A base-line air consumption rate shall be established for each of the four pressurization headers at the time of successful completion of the integrated containment leak rate tests. Unexplained increases from this consumption rate shall be considered as reason for concern and normal practice will require routine investigation and location of the point of leakage.

2. The upper limit for long-term uncorrected air consumption for the pressurization system shall be 0.2% of the containment volume per day (sum of four headers). This is consistent with maintenance of a minimum of 24 hours supply in the reserve nitrogen cylinders.

3. The upper limit for short-term air consumption for the pressurization system shall be 0.5% of the containment volume per day, contingent on the following:

    a) Pressure in all pressurization zones is maintained above incident pressure.

    b) Air supply is maintained from the compressed air systems with compressors running.

    c) The full complement of standby nitrogen cylinders is charged, assuring an aggregate gas supply equivalent to at least six hours of operation.

The modes of system operation with loss of the various sources of pressurization gas are discussed above. The piping system itself is designed and tested to high standards of quality and for minimum leakage. Careful attention is paid to layout to ensure freedom from accidental mechanical damage. It is inconceivable that the piping system and valves would fail from overpressure for the following reasons. A pressure relief valve protects the system from failure of the pressure reducing valve in the line from each of the groups of nitrogen storage cylinders. Each zone of piping will also be protected by a rupture disc. From a strength standpoint the piping is overdesigned by more than an order of magnitude. Should an air receiver fail, the pressurization gas load will be automatically picked up by the bank of nitrogen bottles in that particular piping zone. Pressure control valves, shut off valves and check valves will be overdesigned from the temperature and pressure standpoint and will be located outside the containment for ease of inspection and maintenance. Failure of any of these components except those shut-off valves which are locked-open manual valves at each penetration and weld joint channel will not lead to loss of pressurizing gas, since another source will automatically take over on loss of pressure in the supply source. Pressure and flow measuring instruments and sensing devices can be removed, checked, and maintained during scheduled station shutdowns.

In order to ensure that the station operators are aware at all times that all penetrations and liner weld seam channels are pressurized, the following instrumentation will be provided. Each piping penetration sleeve, each electrical penetration, each of the two ventilation purge duct penetrations, and the double gasketed space on the outside hatch of each of the personnel air locks will have a locally mounted pressure gage on the outside of the containment, available for regular reading and located for ready accessibility. The accuracy of these gages will be within 0.5% of the full scale reading.

The pressurized zones located entirely inside the containment, and those penetrations which are located in inaccessible or unsheltered areas will be equipped to actuate remote low pressure indication in the central control room. Examples of the zones so equipped are:

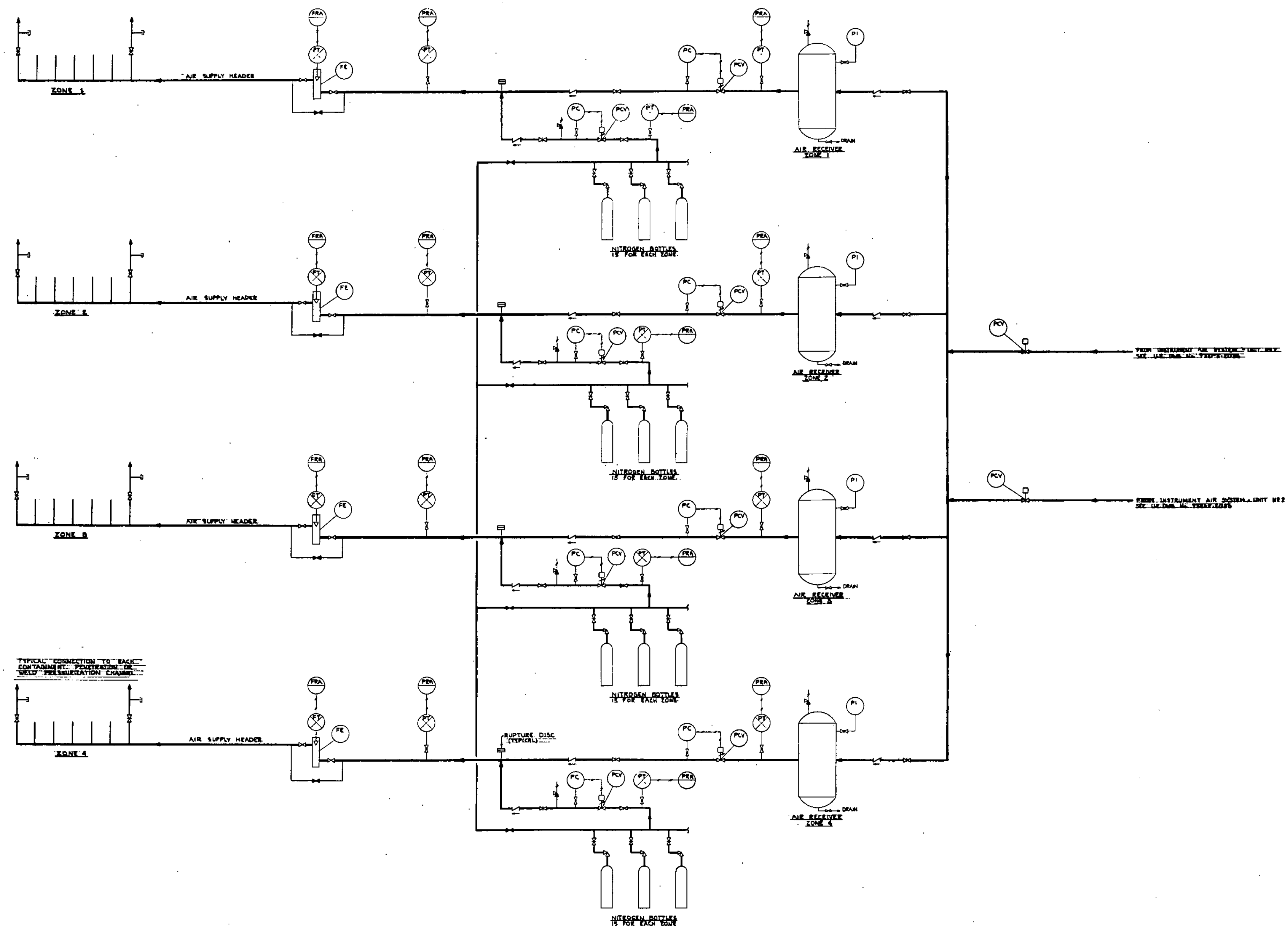
a) Each liner seam weld channel
b) The double-gasketed space on each hatch of the personnel air lock.
c) The double-gasketed space on the equipment door flange, and
d) The pressurized zones in the spent fuel tranfer tube, etc.

The pressure sensing device is a pressure switch, set just above incident pressure and just below the nitrogen supply regulator setting. Should pressure in any of these zones fall below the pressure switch set point, a light and an alarm will be activated. Each penetration and each section of liner weld joint channel so alarmed will be represented by a separate light and identified.

Continuous pressurization of air lock door double-gasketed barriers and the protection of the pressurization header against air loss is assured by a set of interlocks. One interlock on each air-lock door prevents opening of the door until the pressurization line is isolated and pressure in the double-gasketed closure is bled to atmosphere. This prevents excessive leakage from the pressurization system. The pressurization line to this pressurized zone is also equipped with a restricting orifice to assure that air consumption, even upon failure of the interlock, will be within the capacity of the pressurization system, and will not result in loss of pressure in other zones connected to the same pressurization header. Another set of interlocks prevents opening of one air lock door until the double-gasketed zone on the other door is re-pressurized.

The containment ventilation purge penetration valves are also interlocked to prevent the opening of either valve until the pressurization connection has been isolated. Isolation of the pressurization line to each purge penetration pressurized zone can be accomplished remotely from the central control room. Alarm lights, prominently displayed on a panel indicating the isolation status of the containment, remain lit identifying an open purge duct penetration isolation valve or a low pressurization zone pressure in these penetrations. Restricting orifices are installed in each pressurization line to the ventilation purge penetrations to assure that air consumption, even on failure of an interlock, will not result in loss of pressure to the other zones connected to the same pressurization header.

With a continuous inleakage to the containment from the penetration and liner weld joint channel pressurization system of 0.1% of the containment volume per day, the calculated time for the containment pressure to rise to 1 psi is approximately 14 days and therefore is not considered to be an operating or safety problem. From the standpoint of allowable pressure, a much greater inleakage would be permitted. With the ability to limit the activity of the air in the containment during normal operation through the use of the two small recirculation fans and filters described in Question 6, containment overpressure can be relieved as required through the purge duct and exhaust fan, passing through the absolute filter and up the discharge duct, along with the exhaust air from the Auxiliary Building. The containment pressure will be limited to maximum of 1 psig.



PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM FIGURE 7-1

8. The containment spray system is provided as an independent backup to the air recirculation and iodine filtration system. Discuss the experimental basis for the design of the containment spray system and indicate how the pressure reduction and iodine removal values were derived.

Answer

The heat transfer model used in predicting the effect of spray on containment pressure is in agreement with the experimental work of Hasson, et al[(1)] and of Brown[(2)], who investigated steam condensation on laminar water sheets and water drops generated by spray nozzles. These studies indicate that the bulk temperature of the liquid drop responds very quickly to changes in the surface temperature. In other words, conduction and convection of heat to the interior of the drop will not be a significant resistance to heat transfer.

When the gas film resistance is calculated, with allowance for the effect of a non-condensible component (air), it is shown that the surface temperature of the drop approaches the bulk containment atmospheric temperature during free fall. Since the interior of the drop also reaches this temperature, the heat removal by the drop is determined by a simple heat balance to be the sensible heat required to raise the average drop temperature from that of the spray inlet to that of the surroundings. This is the model employed to predict spray effect in the PSAR.

The basis used in the PSAR for predicting iodine removal by the sodium thiosulfate spray follows the method described by Griffiths[(3)]. The fundamental assumption is that elemental iodine (I_2) absorbed by the

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(1) D. Hasson, D. Luss, and U. Navon, "An Experimental Study of Steam Condensation on a Laminar Water Sheet," International Journal of Heat and Mass Transfer, Vol. 7, pp. 983-1001.

(2) G. Brown, "Heat Transmission by Condensation of Steam on a Spray of Water Drops," Inst. of Mech. Eng. and ASME Proceedings of the General Discussion on Heat Transfer, Sept. 1951, p. 49.

(3) V. Griffiths, "The Removal of Iodine from the Atmosphere by Sprays," AHSB (S) R 45.

liquid drop is rapidly reduced to the highly soluble iodide by the reagent sodium thiosulfate, with the result that the partial pressure of I_2 at the drop surface is very small compared with that in the atmosphere. The experimental work of Taylor[4], performed in a column of controlled geometry, confirms this assumption for the aqueous thiosulfate absorption process. These tests show the overall mass transfer rate of I_2 to be independent of liquid flow rate, a parameter which would be expected to alter liquid film resistance. By contrast the transfer rate was found to be gas-velocity dependent, indicative of the fact that gas film resistance is controlling.

In a gas film controlled process, the dependence of the mass transfer rate on film conditions is expressed by

$$V_g = \frac{D_v}{d} (2 + 0.6 \, Re^{1/2} \, Sc^{1/3}) \qquad (1)$$

where

V_g = transfer coefficient expressed moles transferred per unit time, area, and concentration differential, cm/sec.

D_v = diffusion coefficient for iodine in air, cm^2/sec.

d = drop diameter, cm.

Re = Reynolds number, $\frac{d \, v \, \rho}{\mu}$, where ρ, v and μ are the density, velocity, and viscosity of the vapor, respectively (consistent units).

Sc = Schmidt number, $\frac{\mu}{\rho \, D_v}$ (consistent units).

Equation (1) is well substantiated by experiments in a variety of systems in which the gas film resistance is controlling, as reported by Ranz and Marshall[5].

The value of V_g thus derived was used in conjunction with the calculated droplet area for 700 μ mean diameter spray to predict the I_2 absorption rate used in the analyses reported in Chapter 12 of the PSAR.

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(4) R. F. Taylor, "Absoprption of Iodine Vapour by Aqueous Solutions," Chemical Engineering Science, 1959, Vol. 10, pp. 68-79.

(5) W. E. Ranz and W. R. Marshall, Jr., "Evaporation from Drops," Chemical Engineering Progress, Vol. 48, No. 3, p. 141 and No. 4, p. 173

It is interesting to note that recent experiments performed in the Nuclear Safety Pilot Plant in which fission product iodine was evolved by melting UO_2 under reducing conditions, the in-containment behavior of iodine was characteristic of HI and/or particulate iodides. That is, the iodine was almost totally collected with the condensate and by the particulate filters. The reducing conditions of the experiments were due to the presence of molten metal cladding and of hydrogen liberated by the cladding-steam reaction. These conditions would also prevail in the early stages of the PWR meltdown accident, suggesting that the prevalent vapor form of iodine would be the highly soluble HI, rather than I_2 as assumed. The spray removal effectiveness is therefore less dependent on the liquid phase reaction than assumed in the analysis, since a large fraction of the airborne iodine may be in the reduced state when absorbed or condensed.

9. The operation of some engineered safeguards systems will require that large quantities of radioactive liquid be pumped outside the containment under accident conditions thereby extending the effective containment boundary. Estimate the amount of leakage of radioactivity (liquid and gaseous) from lines, valves, pumps, etc. outside of containment under accident conditions and discuss how leakage will be controlled to limit potential off-site doses under accident conditions. State and justify the maximum leakage that can be tolerated from these systems before off-site doses exceed Part 100 values. In consideration of the importance of achieving low leakage of radioactive materials from your facility under accident conditions, discuss the advisability of installing some or all of this equipment inside the containment vessel.

Answer

The safety injection system, shown in Figure 6-1(R) has all of its pumping equipment and most of its piping and valving located outside the containment and within the auxiliary building. During the recirculation phase of operation following a loss of coolant, the system recirculates spilled reactor coolant and borated injection water from the containment sump through the residual heat exchangers, located within the auxiliary building, and returns the water to the containment for long-term cooling of the core or containment spray. The safety injection system is the only system which can circulate spilled reactor coolant and borated injection water outside the containment boundary. All components of the system are shown on the flow diagram in Figure 6-1(R), and with the exception of the relatively short spans of pipe in the trunk between the auxiliary building and the containment, all components which can contain the recirculated containment sump water are located within one region of the auxiliary building.

The system is designed, along with the auxiliary building ventilation and air filtering equipment, to assure that the total leakage release from the plant will be substantially below 10CFR100 limits and in fact do not exceed the limits of 10CFR20. This criterion is met, first, by minimizing leakage from the system and, second, by assuring that such leakage that does occur, will be confined initially within the auxiliary building and will be filtered through activated charcoal filters before being released to the environment.

A high degree of inherent leak tightness in the safety injection system results from strict quality control procedures; the use of all-welded joints, seal welded flanges and valve bonnets packless valves, where practical, and the specification of stringent backseat and seat leakage requirements on conventional valves; and the provision for injection-type double-seal pumps and valves with pressure in the pressurized zones is maintained above the recirculation loop working pressure. Operational leak tightness is assured by the provision of redundant isolation valves at system boundaries, and by minimization of connections which must be closed to assure isolation of the system from the environment. No immediate valve operation is required in order to effect isolation of the recirculation loop from the environment following the loss-of-coolant accident. The necessary valves are closed as part of initiating the recirculation phase of operation.

Ventilation of the Auxiliary Building

The ventilation system is arranged so that the two purge and dilution exhaust fans take suction from the end of the auxiliary building in which the safety injection equipment, piping and valves are located. The normal flow path for air from the auxiliary building discharges through a duct to the suction of these fans which then discharge to the plant discharge duct through a roughing and absolute filter. Prior to initiating recirculation flow in the safety injection system, the direct path to fan suction is blocked by closing two bubble-tight, remotely-operated butterfly valves. An alternate path is opened to fan suction through two independent and parallel charcoal filter assemblies each of which is isolated during normal plant operation by its individual remotely operated bubble-tight butterfly valve.

The flow rate through the auxiliary building with only one of the two purge and dilution fans operating and with only ony filter flow path open, is sufficient to assure a negative pressure in the region of the auxiliary building containing the safety injection system equipment. Either one of these fans can be powered from the emergency diesel.

The charcoal filters are the same type used in the containment ventilation filtration units and can be removed periodically for testing of filter effectiveness using the same test equipment provided for the containment filters. As in the case of the containment filters, provisions will be made for periodic checking of the installed filter assembly leak tightness using aerosol test procedures.

Provisions for Assuring System Leak-tightness

Leak-tightness is assured by attention to isolation of system boundaries, use of all-welded construction, and the sealing of individual points of potential leakage.

Control of the system boundary has been achieved first by limiting connections to the recirculation flow path to those times essential to the engineered safeguards. For those paths where connections must be made (the refueling water storage tank and the pump recirculation path used in system testing) isolation of the flow path is assured by two remotely operated valves in series. Leakage across the seats in these valves will be specified and tested to 3 cc/hr/inch of nominal pipe size in accordance to MSS-SP-61[1].

Welded construction is used throughout the system wherever it is practical and where other joints must be specified they are seal welded as follows:

a) Piping flanges, where used, will be seal welded.
b) Valve body-to-bonnet joints will be seal welded.
c) All instrumentation connections and joints will be seal welded.
d) System vent points will be capped and seal welded.

Valving will be specified for exceptional tightness and where possible such as in the instrument valves, packless diaphragm-type valves will be used. All manual valves will be provided with backseats, which are capable of limiting leakage to less than one cc per hour per inch of stem diameter, assuming no credit is taken for the valve packing. Those valves which are normally open, will be backseated. Normally closed valves will be installed with recirculation flow under the seat to prevent leakage of the recirculated

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1 Manufacturers Standardization Society - SP-61, "Hydrostatic Testing of Steel Valves"

water through the valve stem packing. Relief valves will be totally enclosed. Remotely operated valves and modulating valves will be supplied with double packing and seal water injection between the packings to prevent all outleakage of potentially radioactive water.

Double mechanical seals are used for the residual heat removal pumps, the containment spray pumps, and the safety injection pump shaft seals. Leakage of potentially radioactive water to the atmosphere in these pumps is prevented by injection of borated seal water into the zone between the mechanical seal.

Seal Water Injection System

A seal water injection system is provided to prevent any leakage of radioactive fluid through moving seals by providing a pressurized water seal between the primary and secondary seals of safety injection system pumps and through the double packing provided for all modulating and remotely operated valves. The system consists of:

a) Seal water supply tanks
b) Makeup pumps
c) compressed air supply
d) Pressure regulators
e) Headers and injection lines, and the valves and instrumentation needed for operation and control

The pressure at each seal water injection location is maintained at least 10 psi greater then the operating pressure at the seal. This is accomplished by dividing the recirculation loop seal pressure requirements into three classes: 1) Pressure at the safety injection and containment spray pumps suction, 2) Pressure at the residual heat removal pump suction, and 3) Pressure at the containment spray pump discharge. All valves and pumps that require seal water fall in one of these classes.

Three seal water supply tanks, one for each pressure class, are pressurized by a compressed air supply to maintain continuous seal pressure. This is accomplished by a differential pressure controller on each tank. A pressure signal is obtained from redundant pressure detectors located at the highest pressure point of the equipment in each pressure class. This signal is fed to a differential pressure controller which regulates the required pressure in the seal water supply tank by admitting compressed air to increase pressure and by venting to decrease pressure.

Redundant level indicators and alarms in the central control room are used to monitor the seal water supply in each of the three tanks.

Pressure in each seal point is indicated in the control room to show that the correct pressure is fed through to each pressurized zone between seals. Flow indicators and alarms in the control room are used to monitor the seal water flow for each piece of equipment. This instrumentation provides information to back up the supply tank level instrumentation and serves to identify the points of unusual water usage.

Each seal water supply tank has a capacity of 400 gallons, and is so sized to provide at least 15 minutes to isolate a pump which has experienced gross failure of an outside seal member. The flow capacity of the system will keep up with demand during this period and will assure leakage of only uncontaminated seal water into auxiliary building.

The capacity of each tank is far in excess of that needed to handle the expected demand. (When functioning properly, the leak rate from the pump mechanical seals is approximately one drop per minute.) Makeup to the system is provided at 5 gpm to handle long-term leakage well in excess of the expected quantities. This flow can be provided by one of the two makeup pumps, either of which can be driven from the emergency diesel.

The makeup water supply is obtianed from a 20,000 gallon supply in the refueling water storage tank. This supply will remain following safety injection since the suction connections for the safety injection equipment are located above the 20,000 gallon level.

Testing

The recirculation piping is initially hydrostatically tested at 150 per cent of design pressure of each loop. The entire loop is also pressurized during periodic testing of the safeguards components. The recirculation piping will also be leak tested at the time of the periodic re-tests of the containment (with water in the piping outside the isolation valves).

Since the recirculation system is operated at a pressure in excess of the containment pressure, it will be leak tested during periodic re-tests at the recirculation operating pressures. This will be accomplished by running each recirculation pump (safety injection, spray, and residual heat removal pumps) in turn at near shut off head conditions and checking the discharge, test and mini-flow lines. The suction lines will be tested by running the residual heat removal pumps and opening the flow path to containment spray and safety injection pumps in the same manner as the actual operation of the recirculation loop, thus pressurizing the entire suction header. The seal water injection system performance will be checked during these tests. During these tests, all system joints, valve packings, pump seals, leakoff connection, or other petential points of leakage will be visually examined.

Leakage Evaluation

Leakage to the atmosphere is limited to two potential sources: 1) Stem leakage of normally open backseated valves and 2) Isolation valves at the recirculation loop boundaries. All other leak sources are either seal welded or sealed by borated water injection.

There are 42 normally open valves in the recirculation flow path. In evaluating the nominal system leakage, it was assumed that the average valve stem diameter is one inch and that each valve is leaking at the specified rate of 1 cc/hr/in of stem diameter. The resulting leakage is 42 cc/hr.

Considering the isolation valves at the recirculation loop boundaries, there are two possible flow paths: a) the suction line at the refueling water storage tank and b) safety injection and spray pump minimum flow protection return lines. These lines are isolated by redundant remotely operated isolation valves and it is assumed that the leakage across the seat of each valve is at the specified value (3 cc/hr/inch of nominal pipe size). The resulting leakage is 45 cc/hr. from these sources.

The total leakage from all sources is, therefore, 87 cc/hr.

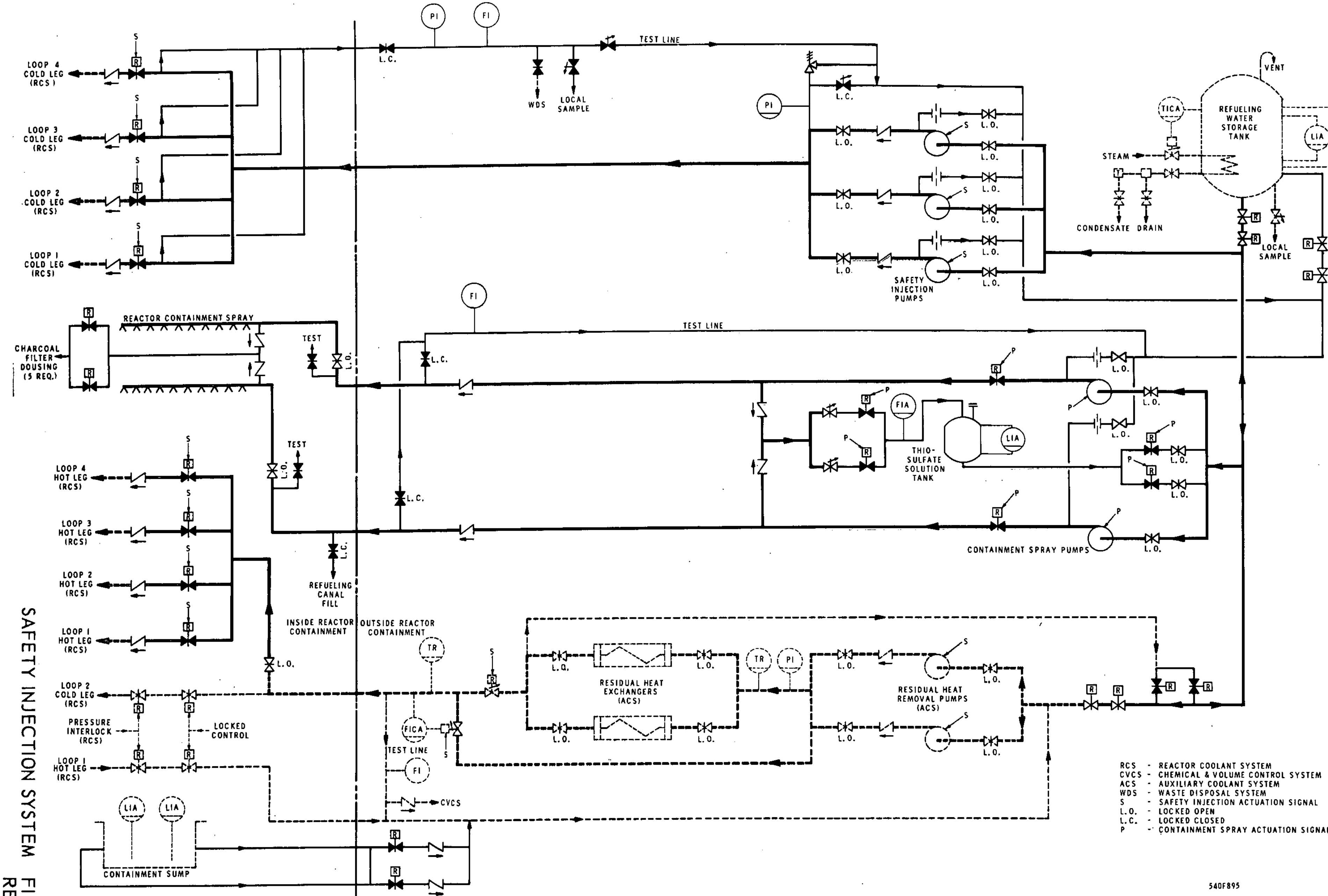
Off-site Exposures Due to Recirculation Loop Leakage

The off site exposures due to the leakage from the recirculation loop are determined on the following basis. The entire core inventory of iodine is assumed to be in the sump water with a concentration of approximately 9.5×10^{-2} equivalent curies of I-131 per cubic centimeter at the start of recirculation and that natural decay is the only mechanism for source strength reduction. Approximately 20% of the leaking liquid flashes to vapor in cooling to ambient conditions and carries that fraction (20%) of the iodine in the liquid to the auxiliary building atmosphere. This assumed a liquid temperature of 290°F although the expected temperature is progressively lowered below 200°F during recirculation. All of the vapor and iodine liberated to the auxiliary building atmosphere is sent through a 90% efficient charcoal filter unit (expected efficiency is greater than 99% for these low temperature-humidity conditions) prior to reaching the environment via the plant vent. The dispersion models for the release from the vent are those presented in Chapter 12 of the PSAR.

For a total leakage of 87 cc/hr, the 2 hour dose at the site boundary is 0.07 rem to the thyroid. This is a factor of 4.3×10^{3} below the limits of 10CFR100. The 24 hour dose at the low population zone for this same leakage is 0.24 rem to the thyroid which is a factor of 1.25×10^{3} below the limits of 10CFR100.

The total doses at the site boundary and the low population zone due to both the leakage from the containment prior to isolation and the leakage from this loop during the reciruclation phase are 1.0 rem to the thyroid and 0.75 rem to the thyroid, respectively, which is about the yearly total thyroid dose obtained due to a continuous exposure to the allowable concentrations for iodine given in 10CFR20.

Additional contributions to the dose after the 24 hour period will be very small as the temperature of the recirculated water will have been reduced sufficiently so that almost no flashing of any leaking coolant will occur.



SAFETY INJECTION SYSTEM FIGURE 6-1
REVISION-1

10. Discuss the operation of the emergency diesel power supply system under accident conditions with no normal power sources available. Indicate how the proper equipment is selected for operation (assume failure of one bus or diesel) and how unnecessary loads are dropped to prevent overloading and possible tripping of the remaining diesels.

Answer

Description of Emergency Power Supply

There will be three (3) 50% capacity emergency generators installed for Indian Point Station, Unit #2. Any two units, as a back-up to the normal standby AC power supply (Consolidated Edison 138 KV system) will be capable of sequentially starting and supplying the power requirements of one complete set of safeguards equipment. Figure 10, as attached, is a one-line diagram of the 480 volt bus arrangement of the emergency diesel-generator units and the engineered safeguards equipment. The equipment automatically started during the injection phase is:

One residual heat removal pump
One safety injection pump
One service water pump
One containment spray pump
Four of five containment ventilation fans

The loads will be changed manually from the above during the recirculation phase to provide cooling to the containment and core by either the fan coolers or the recirculation of coolant from the containment sump to the core and to the containment by way of the spray headers. For example, loads for the recirculation phase will be:

One residual heat removal pump
One safety injection pump
One service water pump
One component cooling water pump
One containment spray pump
One auxiliary building ventilation fan
Additional fan coolers as power availability permits

The diesel units will be started on loss of voltage on the 480 volt buses, which signal will also trip all motor feeder, main supply, and tie breakers on the 480 volt buses. After each unit comes up to speed and voltage, requirement for a safeguard system operation will initiate the closure of the emergency generator supply breakers to their respective 480 buses, as follows:

Diesel-generator unit 1 to 480 volt Bus 5
Diesel-generator unit 2 to 480 volt Bus 2 and 3
Diesel-generator unit 3 to 480 volt Bus 6

Upon energization of the 480 volt buses, the following two sequences will be started simultaneously:

<u>Sequence 1 (Equipment Connected to Bus 5 and 2)</u>

1. Start safety injection pump (Bus 5) and energize motor control center (Bus 5) to supply power to valves. If safety injection pump on Bus 5 did not start, start safety injection pump on Bus 2.
2. Start first containment ventilation fan (Bus 5)
3. Start second containment ventilation fan (Bus 5)
4. Containment spray pump (Bus 5) can now be started by a high containment pressure signal. If containment spray pump on Bus 5 did not start, start containment spray pump on Bus 2.

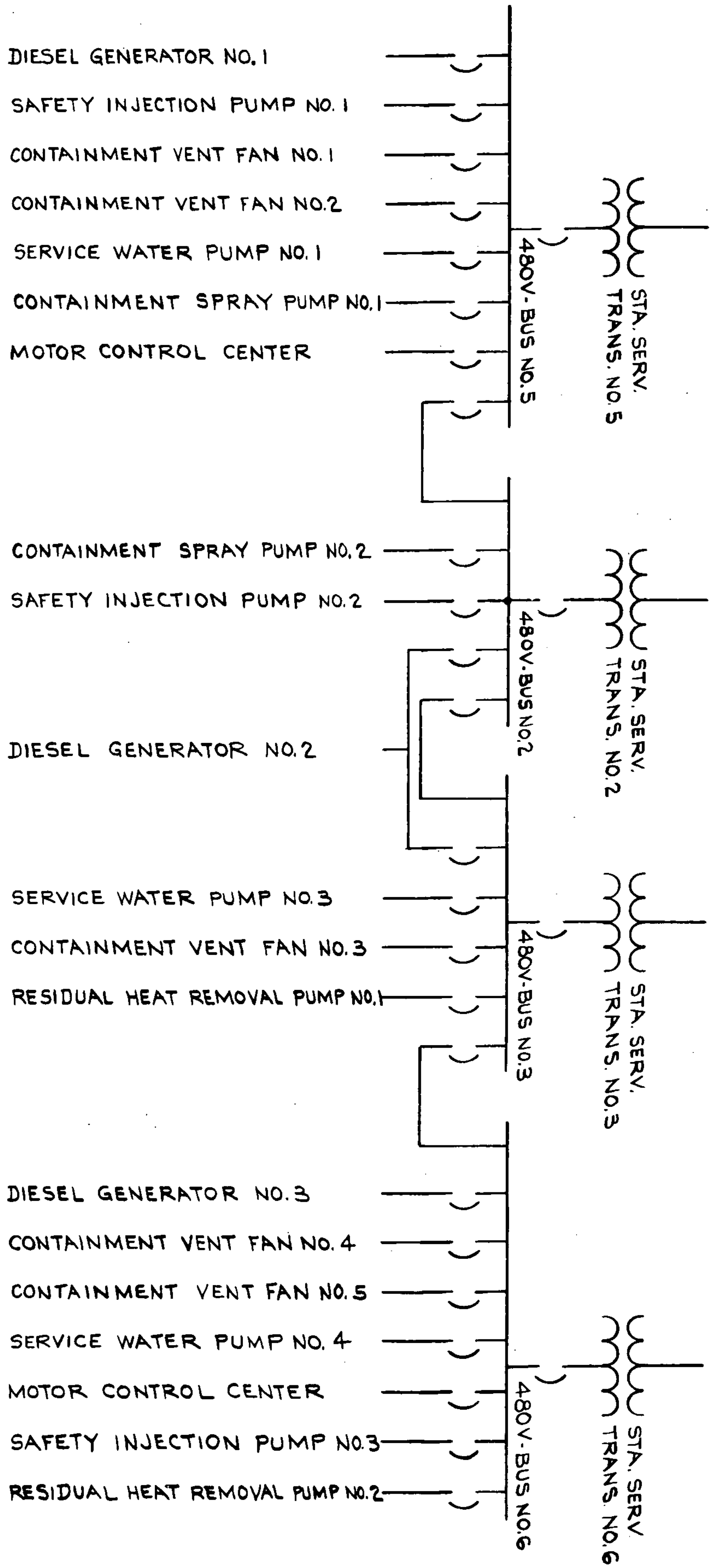
<u>Sequence 2 (Equipment Connected to Bus 6 and 3)</u>

1. Start residual heat removal pump (Bus 6) and energize motor control center (Bus 6) to supply power to valves. If residual heat removal pump on Bus 6 did not start, start residual heat removal pump on Bus 3.
2. Start service water pump (Bus 6). If service water pump on Bus 6 did not start, start service water pump on Bus 3.
3. Start fourth containment ventilation fan (Bus 6)
4. Start fifth containment ventilation fan (Bus 6)
5. If one of the containment ventilation fans in either sequence 1 or sequence 2 did not start, then the third containment ventilation fan on Bus 3 will be started.

In the event an emergency diesel-generator unit did not start, appropriate tie breakers will be closed in order to energize the four 480 volt buses. Depending on which unit did not start, the buses will be energized as follows:

1. Diesel-generator 1 did not start - diesel-generator 2 supplies Bus 5 and Bus 2, diesel-generator 3 supplies Bus 3 and Bus 6.
2. Diesel-generator 2 did not start - diesel-generator 1 supplies Bus 5 and Bus 2, diesel-generator 3 supplies Bus 3 and Bus 6.
3. Diesel-generator 3 did not start - diesel-generator 1 supplies Bus 5 and Bus 2, diesel-generator 2 supplies Bus 3 and Bus 6.

The sequence of starting the motors will be the same as previously listed and will not be affected by the buses being supplied either by two or three diesel generators.



ENGINEERED SAFEGUARDS
BUS ARRANGEMENT FIGURE 10

11. Provide preliminary accident evaluations to support the results reported in the Preliminary Safety Analysis Report for the startup accident, steam line rupture, refueling, and control rod cluster ejection accident. For example, show such parameters as core reactivity, core temperature, and system pressures plotted against time for the worst condition during core lifetime. Consider the possible generation of curves that relate minimum reactor period to (a) integrated excursion energy and (b) average fuel temperature. These curves should consider cases of hypothetical reactivity insertions considerably greater than that resulting from the ejection of a single control cluster. For each accident, state the potential off-site doses.

Answer

The startup accident is assumed to occur as a result of an uncontrolled rod cluster withdrawal from a zero power condition. If the reactivity rate is excessive, an abnormally high rate of increase in flux will initiate an alarm in the control room. If corrective action is not taken to reduce the startup rate, an automatic cluster-stop signal deactivates the cycling mechanism on the cluster drive control causing cluster withdrawal to cease. Should this signal fail to block the withdrawal, trip relays will open automatically on a high startup rate signal from the intermediate range channels. Trip relays will also open when the nuclear power level detected by any two of four power range channels exceeds a preset fractional power limit established for startup.

The maximum reactivity insertion rate with control rod cluster withdrawal, even assuming simultaneous withdrawal at maximum speed of control banks out of sequence is less than $3.5 \times 10^{-4}\,\delta k/\mathrm{sec}$. The nuclear power response to a continuous reactivity insertion at a very low power level is characterized by a very rapid rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient which limits the power to a tolerable level prior to external control action. A reactor trip is completed within 2 seconds of initiation to terminate the accident. The entire transient is completed before there is any significant heat transfer to the coolant to either increase the thermal heat flux significantly or heat the coolant and add any significant positive reactivity if a positive Moderator coefficient exists. The results presented in the Final Engineering Report for the San Onofre Nuclear Generating Station Unit #1 are very typical of the response to a startup accident and similar results are expected for the Indian Point Unit #2 when the detailed design is established. In this San Onofre analysis, the maximum heat flux reached was less than 50% of full power and the maximum increase in reactor coolant average temperature is less than 10°F and there was considerable margin to DNB.

A break in the steam piping between the steam generators and the turbine affects the safety of the plant in two ways:

a) A rupture in this system increases the rate of heat extraction by the steam generators and causes cooldown of the reactor coolant. If the core power required to meet this apparent load increase is excessive or if the cooldown and depressurization result in inadequate subcooling, the reactor is tripped to ensure the minimum desired margin to DNB in this phase of the transient.

b) Cooldown after trip, due to continued discharge of steam reduces the shutdown reactivity margin. In the more severe cases, injection of borated water in addition to control cluster insertions is required to terminate this reactivity transient.

The most severe steam break accident, with respect to reactivity insertion during cooldown, exists at the end of core life when the most negative moderate coefficient will exist. Other adverse conditions assumed in the analysis include the highest worth control rod stuck in the withdrawn position and minimum shutdown margin of 1% available with the remaining control rods.

During plant design, a hypothetical steam break is analyzed for the condition of a complete double ended rupture of the main steam header in coincidence with the above adverse conditions. In addition, minimum effectiveness is assumed for the safety injection system which serves to terminate the accident. Under this unlikely combination of circumstances, a short term return to criticality could occur during the cooldown before the safety injection system has added sufficient boron to shut down. If the flux distribution resulting from a single stuck rod is sufficiently distorted, DNB may occur, causing clad deformation and possible clad rupture in a limited core region. Public and plant personnel safety is assured by the fact that the transient is self-limiting (rising temperatures, coolant voids, boiloff of all available coolant in the secondary system and continued boron addition act to terminate the transient) and by the fact that any fission products released are confined to the reactor coolant system.

The result of this transient is very much a function of the detailed core design; for example, the minimum available shutdown and power distribution with a stuck rod as well as the characteristics of the safety injection system. From the standpoint of the steam break accident, the Indian Point Unit #2 design does not significantly depart from the San Onofre design. The initial increase in steam flow as a result of the hypothetical break should be of the order of 200 to 300 per cent at no load conditions where the maximum initial steam pressure exists. This flow falls rapidly to the order of 20 per cent of full load flow within a minute as the steam generator pressure decreases. The reactor coolant system will cooldown to between 350 and 400°F during that time. Should a momentary return to critical occur during the cooldown, automatic initiation of the safety injection system will effectively shut the core down within one minute after the initiation of the accident. Without safety injection the eventual blowdown of all the secondary water and the subsequent gradual heatup of the reactor coolant would shutdown the reactor within five minutes.

In order to assume a rapid ejection of a control rod cluster from the core, a hypothetical rupture of the control rod mechanism housing must be postulated. For this accident, it will be demonstrated that the effects are definitely localized and that there is no resulting pressure surge that could further damage the reactor coolant system. The resultant power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals. The amount of fuel damage that could result from such an accident is governed by the peak power attained in the transient which in turn depends on the worth of the ejected rod and the power distribution attained with the remaining control rod pattern.

The results of a rod ejection analysis are a function of the detailed physics design including the core enrichment pattern, the grouping of the control rods, their radial location and their axial location as a function of load. During the evaluation of this accident, all normal operating conditions are studied to determine the most severe condition including tolerances for instrumentation errors, reactivity coefficients etc. A key consideration in the detailed core physics design is the specification of the combination of parameters that will result in a calculation of ejected rod worth and power peaking due to the ejected rod that would limit the maximum fuel temperature to a value that would preclude any additional damage to the reactor coolant system.

The accident offers the potential for considerable core damage to the extent of possible gross fuel melting, dispersion of molten UO_2 in the coolant and a subsequent rapid pressure surge or shock. The reactor will be designed and operated so that this accident will not cause further rupture of the reactor coolant system. For purposes of this analysis fuel dispersion is assumed to occur when the average fuel temperature exceeds the UO_2 melting temperature. If there is any likelihood of this dispersal, a detailed analysis is mad e to demonstrate that the rapid dissipation of this energy into the water does not create a significant pressure surge to jeopardize either the integrity of the vessel or that of the adjacent fuel.

The core design for the Indian Point Unit #2 does not represent a significant departure from the San Onofre design for which detailed rod ejection Analyses have been made. The resultant ejected rod worth and peaking factors for the San Onofre design were such that in the worst case there was not even center rod melting. The maximum hot spot fuel center temperature only reached 4060°F. This corresponds to a maximum inserted reactivity of 0.8%δk and a heat flux hot channel factor of almost 10. On the basis of the San Onofre analysis it is expected that the criteria as stated for the Indian Point Unit #2 can be met with a high degree of confidence.

The effect of the additional heat transfer to the reactor coolant system during this transient is evaluated to determine the maximum system pressure including the effects of fuel dispersal in the coolant, if necessary. For the San Onofre Plant there was not even center melting and thus no danger of molten fuel being expelled. The maximum calculated pressure resulting from the rod ejection transient was 2316 psia.

During refueling, the primary coolant is borated sufficiently to yield k_{eff}=0.9 for cold conditions with all rods in. This concentration is also sufficient to prevent criticality even with all control rods removed. Continuous mixing will be maintained through the reactor vessel by utilizing a residual heat removal loop. During this period, the neutron sources installed in the core and three separate BF_3 detectors with audible count rates provide direct monitoring of the core. Any appreciable increase in the neutron source multiplication including that caused by the maximum physical boron dilution rate (approximately 580 ppm per hour) is slow enough to give ample time to take corrective action i.e. turn off the makeup water pumps which are the only potential sources of clean unborated water. These pumps are tagged and locked out of service during the refueling operation so that dilution cannot occur.

Of the four accidents discussed above, only the rod ejection and the steam line rupture accidents provide any potential off-site exposures. The startup and refueling accidents do not provide any potential for off-site exposure as they do not result in the release of radioactivity from the fuel. Criteria for the permissible activity in the secondary coolant and the potential off-site exposure from this accident are detailed in answer to Question 19d.

The rod ejection accident is a loss-of-coolant accident with a small equivalent break area. The fission product release to the containment is dependent upon the detailed results of the analysis which will determine the extent of fuel dousage due to the ejected rod and associated reactivity transient. Quantative results for this accident are not available at this time as the evaluation cannot be performed until the final plant and core design are completed.

It can be stated, however, that the potential off-site exposures will be lower than those calculated for the hypothetical loss-of-coolant accident and presented in Chapter 12 of the PSAR. The missile shielding provided above the reactor vessel will stop the ejected control rod pressure housing and thereby maintain containment integrity. The rise in containment pressure due to reactor coolant blowdown will cause containment isolation and operation of the isolation valve seal water system as well as switching of the containment air recirculation system to the post-accident condition. (Recirculation flow is passed through the charcoal filter units.) Safety injection and reactor trip also would be initiated because of the low pressurizer level and pressure conditions.

Isolation of the containment in conjunction with the penetration and weld channel pressurization system will prevent release of fission products to the environment. Reactor trip and safety injection will terminate the accident and prevent further release of fission products to the containment and the air recirculation cooling and filtration will reduce the containment pressure and the airborne fission product inventory in the containment.

12. The borated safety injection water may be diluted by the non-borated primary coolant or secondary system water following a major pipe failure. Analyze the consequences of adding diluted safety injection water to the reactor assuming several dilution factors, and provide the corresponding periods and energy releases if the control rods are and are not assumed to be inserted.

Answer

The borated safety injection water will be diluted to some extent by the spilled reactor coolant following a major pipe failure. The dilution of this water by spilled water from a simultaneous rupture of the secondary system is not considered credible, in view of the missile protection provided for the steam generators and the support of these units to accept reaction forces resulting from the complete circumferentral severance of a reactor coolant pipe or steam line.

The water injected during safety injection from the refueling water storage tank will contain the same concentration of boron (as boric acid) as is used in the refueling shutdown. The tentative concentration for an equilibrium core cycle, is 1.2 weight percent of natural boric acid. This concentration is soluble well below any credible post-accident temperature in the containment and concentration would not decrease from this cause. Boiling of water in the core would tend to increase boron concentration in the reactor vessel and improve the shutdown margin.

Mass transfer with containment surfaces is not envisioned to produce any significant change in boron concentration.

The only method of dilution considered credible is the postulated mixing of spilled reactor coolant with refueling water-injected into the containment at a higher boron concentration. Since recirculation will not be started until a substantial quantity of the refueling water has been added, it is reasonable to assume that a substantial mixing has taken place before the water is returned to the reactor core.

Assuming one control cluster stuck in the withdrawn position, a mixture of at least 30% by volume borated refueling water in unborated reactor coolant is required to assure the recirculation of borated sump water at about 800 ppm. to effect a minimum of one per cent shutdown margin.

Even if a flow of unborated, saturated water is established at the maximum recirculation rate into a core, without credit for the negative reactivity effects of either fission products or control rods, the steam void production rate, based on a conservatively low estimate of the residual heat generation, is sufficient to prevent criticality. As the cooling of the unborated recirculation water by any of the engineered safeguards lowers temperature significantly below saturation conditions in the containment, the core would slowly approach a power level at which the sum of the nuclear power and the residual heat will be equal to the heat removal capacity of the engineered safeguard (recirculation through the residual heat exchangers to the core). This would amount to approximately 4% of full power.

Although mixing in the containment sump is probable during the injection phase and this power generation is very unlikely, such power generation is terminated after about 10 minutes, when the recirculation of water through the break accomplishes the transfer of the limited volume of spilled coolant, and borated water at refueling concentrations is returned to the core.

13. The steam generators provide the primary mechanism for dissipation of primary system heat in the event of complete loss of power. Indicate the water sources and capacity available to the steam generators under these conditions. Discuss how this water can be delivered, and how long the reactor can be safely cooled by these sources.

Answer

The main water supply to the steam generators in the event of a complete loss of outside power is the condensate storage tank. This tank is sized to meet the normal operating and maintenance needs of the turbine cycle systems; however, a minimum water level will be maintained, equivalent to the steam generation due to 24 hours of residual heat generation at hot shutdown conditions. The tank is designed to seismic Class I standards.

A point for connection to the tank from at least one alternate supply of water will be provided for long-term cooling (from the primary plant makeup water supply, or from the 1.5 million gallon tank for plant storage of city water). The backup system selected will be designed to seismic Class I standards and can be powered by the diesels.

Water delivery to the steam generators is continued in the event of a total loss of outside power by a single, turbine-driven emergency feedwater pump which takes suction from the condensate storage tank. As in the case of the storage tank, the emergency feedwater pump and its piping system are designed to seismic Class I standards. The pump is sized to maintain a water level in the steam generators within the range of the steam generator level indication considering the effects of steam dump after reactor trip, pump starting delay, and the residual heat generation rate.

The loss of feedwater supply will signal for the start of the emergency steam driven feedwater pump. The emergency turbine utilizes steam flow from the steam generators and the turbine driver exhausts directly to the atmosphere.

If, at the time of turbine trip and complete loss of off-site power, the plant is operating at full load, there will be a rapid reduction of steam generator water level because of the reduction in steam bubble void fraction on the secondary side of the steam generator and partly because steam flow continues after normal feedwater flow stops. At the end of two minutes, flow will be established from the emergency feedwater pump and further reduction in water level will be slow. The capacity of the pump is adequate to assure that the water level will not fall below the lowest level within the indicator range during the transient. Thus the tubes will not be uncovered during the accident to the extent that the heat sink will not be adequate for removal of reactor residual heat.

The reactor operator in the control room will monitor the steam generator water level and will control the feedwater addition.

The steam driven feedwater pump can be tested at any time by admitting steam to the turbine driver. The pump will deliver water from the condensate storage tank through its feedwater control valves to the feedwater line to the steam generators. The remaining valves in the system can be operationally tested when the drive turbine and pump are tested.

14. Provide the following information regarding the proposed instrumentation system:

    a. Discuss the instrumentation provided to prevent low water levels in the steam generators. Is this instrumentation redundant?

<u>Answer</u>

The following redundant instrumentation is supplied to protect against low steam generator water level.

(a) Two level transmitters on each steam generator provide signals for low and low-low alarms in addition to indication.

(b) Feedwater flow mismatch (low feedwater flow as compared to steam flow) is alarmed. Each steam generator has two complete channels.

(c) The reactor is tripped if any one of the four steam generators indicates feedwater flow mismatch (one of two) in coincidence with low water level (one of two). The coincidence circuit is duplicated for each steam generator.

14. Provide the following information regarding the proposed instrumentation system:

    b. Discuss how the cluster control system has been designed so that rod insertion time is not delayed as a result of pressure gradients generated by potential blowdown forces.

Answer

The effect of hydraulic forces on rod cluster control insertion time during a loss of coolant accident is considered. Any delay in insertion time will be considered in establishing the time to achieve reactor shutdown, and the core fission energy generated prior to this shutdown will be considered in evaluating the energy releases to the containment under loss-of-coolant accident conditions.

14. Provide the following information regarding the proposed instrumentation system.

    c. Discuss how the position of critical isolation valves will be indicated in the control room.

Answer

All remote and automatically operated containment isolation valves will have control switches and position indicating lights in the main control room.

Manual isolation valves will be located to assure accessibility following the hypothetical accident and will be so designed to allow easy determination of valve position by local inspection. Manual valves which must be closed during power operation will be locked closed and tagged. Opening of these valves during power operation may be performed only under the administrative control of the licensed operator, and an operator must remain in the area during the period that the valve is open.

14. Provide the following information regarding the proposed instrumentation system:

    d. A rupture of the tap feeding two of the three pressurizer-low-level channels can remove the intended automatic protection provided by this circuit. Please justify this proposed design.

Answer

The three pressurizer level transmitters will have three individual taps (3 sets).

14. Provide the following information regarding the proposed instrumentation system:

    e. Provide a list of all monitors that will be provided to indicate the reactivity status of the reactor, and the pressure, temperature and humidity conditions inside the containment after the MCA. Discuss the design lifetime of the critical components associated with this equipment when operated in the post-MCA containment environment.

Answer

Pressure in the containment is the variable that is required for post MCA monitoring. Three transmitters are provided and they are installed outside the containment (auxiliary building) to negate potential missile and/or environment damage. The pressure is indicated (all three channels) on the main control board. This pressure reflects containment temperature and monitors the effectiveness of the containment cooling systems.

There are other monitors inside the containment such as reactor coolant temperature, reactor coolant pressure, radiation monitors, etc., that will be effective in transmitting information to the control board. We claim no benefit nor do we require information from these channels following MCA.

Such information that is available must be considered bonus readings and will be used as the situation dictates.

There is no specific indication that each safeguard system is delivering coolant to the proper area in the design manner; however, the following instrumentation insures broad coverage of the effective operation of the safeguard system:

(1) Containment Pressure

Three channels, monitoring containment pressure, reflect the effectiveness of containment spray, and reactor cooling in that high pressure indicates high temperatures and reduced pressure indicates reduced temperatures.

(2) Containment Sump Level

Redundant (two) containment sump B level indicators will show that water has been delivered to the containment in the early stages of post MCA and, in the later stages, will show that the residual heat removal pumps are effective in recirculation by maintaining the sump level. These transmitters will be designed to withstand MCA conditions.

(3) Refueling Water Storage Tank Level

These redundant (two) channels indicate that safety injection and spray have removed water from the storage tank. They are outside of the reactor containment.

(4) Safety Injection Pumps Discharge Pressure

This channel will clearly show that the safety injection pumps are operating. The transmitter is outside the containment.

(5) Pump Energiziation

All pumps will have indicator lights on the control board indicating colsure of the motor feeded breakers or starters.

(6) Valve Position

All remote operated engineered safeguards valves will have limit switches and assc indicator lights on the control board to show proper positioning of the valves.

(7) Residual Heat Exchangers

Combined exit flow is indicated and combined inlet temperature is recorded on the control board to monitor operation of the residual heat exchangers. In addition, the exit temperature of each heat exchanger is indicated. These transmitters are outside reactor containment.

(8) Air Coolers

The service water supply flow and exit temperature of each of the four coolers are alarmed in the control room if the flow is low or if the temperature is high. The transmitters are outside the reactor containment. In addition, the exit flow is monitored for radiation and alarmed in the control room if high radiation should occur. This is a common monitor and the faulty cooler can be located locally by manually valving each one out in turn.

(9) In addition to the above, the following local instrumentation is available.

a. Residual heat removal pumps discharge
b. Residual heat exchanger combined exit temperature
c. Containment spray test lines flow
d. Safety injection test line pressure and flow

14. Provide the following information regarding the proposed instrumentation system:

    f. Provide experimental evidence to indicate the sensitivity of the external ion chambers to changes in the axial and radial flux distribution. Relate this information to internal monitor readings, if possible.

Answer

Westinghouse had made a number of tests during the past four years to determine the extent to which out-of-core neutron detectors can depict conditions within the core. The conclusion from these studies was that a set of 4 long ionization chambers, each approximately equal in length to the core height, would provide the best means using external detectors both to measure average core power and to detect flux tilts. Each detector would be located opposite the "corner" of one quadrant and the internal construction of the chambers would be of two divided sections of equal length. The total current from the two sections would be used in the reactor protection system as giving the best measure of average core power, and the individual section currents would be used for detection of flux tilts.

The first of these studies was a tabulation of calibration corrections required during the operation of Core I at Yankee (Rowe). Where rod motions or other reasons caused the nuclear instrumentation detector readings to drift by more than 2 or 3%, recalibrations were made. Plotting the corrections resulted in graphs which showed changes in the detector readings versus control rod group positions for all rod groups used during the life of Core I. Measurements of the flux distribution changes in the detector wells for detectors located opposite the bottom half and top half of the core were also compared with flux wire measurements made within the core, particularly in fuel assemblies nearest to the external detectors.

Somewhat similar data were also made available to Westinghouse for the Indian Point Unit #1 Core A, where 18 external detectors are located in three sets of 6 each, one set 1/4 of the core height up from the bottom, one set at the axial center and one set 1/4 of the core height down from the top. The detailed results of the, Yankee core I studies of detector responses were made under LRD 2.05 (Development of Nuclear Instrumentation System) and have been reported in the LRD quarterly reports.

Further evaluations of the long chamber system are currently being carried out at the CVTR facility. Three long chambers (without divided sections) have been installed to replace the normal-size chambers used in the power range channels. Data on these chambers have been accumulated that indicate approximately 4 to 1 improvement in depicting average core power over the previously used chambers. The next step in the test program was to construct and install a long chamber with divided sections (one compensated and the other uncompensated). Early data on this latter chamber indicates that it is performing satisfactorily with respect to sensitivity and compensation.

15. Provide the distance and location of the Chelsea intake for New York City and discuss the possibility of transport of activity from the plant to this point.

Answer:

The City of New York's Chelsea Pumping Station is located about one mile north of Chelsea, New York, on the east bank of the Hudson River. Water will be pumped from intakes in the river at the rate of 100 million gallons per day into the city reservoir system as required to supplement the primary supply from watersheds. The effect of this withdrawal is taken into account in the analysis. The pumping station is 22 miles upriver from Indian Point measured along the centerline of the river.

All liquid wastes will be discharged from Indian Point Unit No. 2 at concentrations below the maximum permissible concentration for drinking water specified by 10 CFR 20 which is also the practice for Unit No. 1. Experience with Unit No. 1 and other operating reactors shows that the radioactive discharges averaged over a period of operation are far below drinking water MPC. During the month of highest discharges for Unit No. 1 in the three years of operation the average rate of discharge was less than 0.2 curies per day and the concentration was below drinking water MPC at all times.

The Indian Point Unit No. 2 liquid waste disposal system, described in Chapter 11 of the Preliminary Safety Analysis Report, will include gas stripping, evaporation and demineralizing systems to process the liquids as well as holdup tanks to allow radio active decay before release to the river. All wastes will be processed, stored and diluted as required to reduce radioactivity to acceptable levels before discharge from the plant. The holdup tanks are located in a leaktight pit, and thus there is no means for tank leakage to escape from the plant. The liquid wastes to be discharged will be pumped first to a monitor tank where they will be sampled before being released to the condenser discharge tunnel, and will be continuously monitored as they are released. If an unexpected increase in radioactivity is sensed by the monitor, the discharge valve will be closed automatically to stop the release.

Liquid wastes discharged from Indian Point at normal rates, and indeed at rates far in excess of normal, will not create concentrations at Chelsea near MPC. This can be shown by a simplified conservative dispersion study using the most unfavorable river conditions.

The dispersion of contaminants in the river is controlled by three factors: the runoff flow or net downstream flow, the tidal dynamics, and the circulating flow caused by differences in water density resulting from salinity gradients. Runoff flow is dependent on seasonal weather conditions and is most favorable during the wet season of the year when the high runoff would tend to dilute the liquid wastes from Indian Point and sweep them downstream away from Chelsea. The Hudson River salinity and, therefore, water density decreases with distance upstream. The salt water gradient from downstream to upstream creates an upstream movement of salt counteracted by the downstream flow.

The total effect of river contamination at any point in the river can be expressed in terms of these flows by the following one-dimensional dispersion equation:

$$E \frac{\delta^2 \ell}{\delta x^2} - U \frac{\delta \ell}{\delta x} - K\ell = \frac{\delta \ell}{\delta t}$$

Where: E = dispersion coefficient

ℓ = contamination level

x = distance along river

U = runoff flow

K = decay constant

t = time

The dispersion coefficient E expresses the longitudinal mixing which is caused primarily by the tidal turbulence and the density gradient. During one tidal cycle the discharge from Indian Point cannot reach Chelsea and, therefore, several tidal cycles are required to move the contaminants upstream. The dispersion coefficient has been determined empirically by measuring the concentration of salt at many locations in the river.

Two cases of liquid waste discharges from Indian Point have been studied: continuous discharges, and an instantaneous release postulating an accidental discharge. The most unfavorable condition is during a drought season when runoff is low. For this study a runoff flow of 4100 cfs was used which is the flow measured during the recent northeastern drought in 1964 and is one of the lowest river flows ever measured for more than a few days at a time.

The continuous discharge case assumed equilibrium conditions in the river with time and, therefore, $\delta \ell \delta t$ in the above equation is zero. The continuous release must be maintained for a very long time to reach the equilibrium condition. To attain equilibrium at Chelsea, the release must be maintained for over 1000 days. It is incredible that the unfavorable river conditions assumed in this analysis would persist for such a long period. We can, therefore, for the time being consider only Cesium-137 and neglect K. The composition of the wastes being discharged is such that all nuclides but Cesium-137 will have decayed to insignificant levels in this time. For these conditions it was found that in order to reach an MPC of 2×10^{-5} μc/ml at Chelsea, Cesium-137 would have to be discharged continuously from Indian Point at a rate exceeding 550 curies per day. However, radiochemical analysis of the wastes from Indian Point Unit No. 1 has shown that Cesium-137 actually accounts for about 2% of the total activity. The discharge rate at Indian Point that will result in a concentration of 2×10^{-5} μc/ml at Chelsea is thus 27,000 curies per day.

However, Iodine-131 is limiting for the continuous release because of its lower MPC. The decay of the nuclides discharged has been factored into the calculation and shows that the limiting MPC at Chelsea, taking into account the isotopes present at the time, occurs 5 days after the release has started. For this case 87 curies per day of the mixture could be released from Indian Point before the MPC at Chelsea would be reached. Taking into account the highest experienced fraction of Iodine-131 in the mixed isotopes release from Indian Point during Unit No. 1 operation, 1000 curies per day of total activity could be released continuously without exceeding drinking water MPC at Chelsea.

The calculated MPC at Chelsea would be an average value and the tidal cycle would cause variations in the order of $\pm$ 20%. However, since the 1000 curies per day release from Indian Point is many times the allowable release rate, it is insignificant that the calculated value is an average. The allowable release rate is defined as the maximum continuous release rate for Indian Point at MPC and is less than 1/100 of the release rate calculated above.

To solve the dispersion equation for an instantaneous release the expression $\delta \ell / \delta t$ denotes the time factor in moving the contamination upstream. In this case the radioactive decay of Iodine-131 was considered, and the contribution of shorter lived nuclides was neglected. The peak concentration at Chelsea would be reached about 9 days following the release from Indian Point. For this case, the MPC is based on a mixture of Iodine-131, Cesium-137, and Strontium-89, as is calculated to be 5.5×10^{-7} uc/ml based on least favorable ratios of nuclides discharged from Indian Point Unit No. 1. In order to reach MPC at Chelsea the time of peak concentration, 5750 curies would have to be instantaneously discharged from Indian Point. Considering that Iodine-131 is only about 5% of the total activity the wastes initially discharged, 120,000 curies of waste could be instantaneously released from Indian Point and not exceed drinking water MPC at Chelsea.

16. The inversion frequency assumed for the 30 day meteorology does not appear to be conservative since it is near the average value for two years. Please justify the selection of this value.

Answer

The degree of conservatism attached to the assumed meterology for the 30 day accident case must be evaluated from the joint assumptions concerning the postulated meteorological conditions. These are:

1) Inversion conditions prevail for 42.4% of the time
2) The wind direction is within a narrow 20° sector for 35% of the time (p. 12-42).

This is equivalent to assuming that in the model 20° sector, the inversion frequency is 14.8 percent for the 30 day period. The observed annual maximum inversion frequency for a 20° sector is 6.2% (p. 29, Table 3-3, NYU Tech. Report 372.3, Section 1.6, Exhibit B, Vol. 1). If we assume that the inversion frequency is spread uniformly throughout the year, almost three months worth of inversions in the model 20° sector are considered to occur in the first 31 day month after the accident. The assumption of uniform spread of inversion frequency over the year is examined critically in section 1.6.2 pp. 1-11 through 1-13, where an attempt is made to isolate those local meteorological conditions at Indian Point which might yield the highest 30 day dose. It is concluded that the "worst" meteorological conditions are associated with the nocturnal down-valley flow which is most frequent during September and October. A rather detailed study of the diurnal course of these winds and associated synoptic conditions, led to the conclusion that "the meteorology assumed in the Safety Analysis beyond the first 24 hours is about right for the worst months (September and October) and is undoubtedly conservative with varying degrees of conservatism for about ten months of the year." There is a further element of conservatism associated with the above conclusions for they are based on the assumption that the steadiness of the wind is identically "one" for the period 00-08 hrs. under weak large scale pressure gradient conditions (See Fig. 1.6-2 and 1.6-3). The observed steadiness of the wind averages about 0.88 from 00-0800 hrs. during weak pressure gradient conditions. The maximum steadiness value for any hour is slightly less than 0.95, but this value occurs only under virtually zero pressure gradient conditions. Although a unique distribution of wind directions cannot be associated with a given value of steadiness, consider the following distributions:

A. Wind distribution within $\pm$ 10° of some mean 90% of the time and at right angles to the mean 10% of the time. The resulting steadiness value is slightly less than 0.9.

B. Wind uniformly distributed around a mean direction from $\bar{\Theta}$ + 1/2 radian to $\bar{\Theta}$ - 1/2 radian with the wind speed uncorrelated with wind direction. The steadiness is then

$$\int_{-1/2}^{+1/2} \cos \Theta d\Theta \simeq 0.95$$

The diffusive consequences of distribution A is simply to reduce F i for the inversion case in χ/Q equation on p. 12-42 by about 10%. This would have no significant effect on the calculations. Distribution B, however, would change the calculated values significantly, because although F i is unchanged for the inversion case, β in the denominator of the χ/Q equation would be increased by a factor of 3 thus reducing the contribution of the inversion case to the total dose by a factor of "3".

The correct interpretation probably lies between hypothesis A and B. If one carefully draws the frequency density curve for I conditions from Table 3.3, it is apparent that the frequency density between 000-040° is quite flat.

This leads to the conclusion that the direction of the nocturnal valley wind oscillates over a range of 30-40° rather than the range of 20° assumed in the analysis on p. 1-13 and in the calculations on p. 12-42. There is an element of possible conservatism here that ranges from a factor of 1.5 to about 2.0. On the other hand, although highly improbable, it is possible that relatively light pressure gradient conditions might be found for 30 consecutive days after the accident. If this were to occur and if the range of the valley wind is assumed to be 35° the calculations presented in the exhibit would still be correct. In short, it is felt that the 30 day accident analysis represents a realistic estimate of what might happen if the accident coincided with a sequence of meteorological conditions considered to be most unfavorable.

17. Discuss the relation to nuclear safety of any system or equipment at the Indian Point site that will be shared by both reactors. Provide the infinite and maximum 8 hour thyroid and whole body doses for both control rooms following the potential MCA at either facility.

Answer

The systems and equipment which are common to Units #1 and #2 are limited to service utilities such as city water, house service compressed air and building heating steam. General facilities such as offices, laboratories, washrooms and locker rooms and showers will also be common to the two units. The only major funtional facility common to both plants is the condenser discharge canal shared by the separate condenser circulating water systems. The use of these common facilities and equipment will not have any effect on the nuclear safety of either plant. All equipment and systems associated with the nuclear safety of Unit #2 are completely separate from the nuclear safety systems of Unit #1.

The infinite and eight hour thyroid and whole body doses for both control areas following the potential maximum credible accidents in each unit are summarized in the following table:

	8 Hour Dose		Infinite Time Dose	
	Whole Body	Thyroid	Whole Body	Thyroid
Accident in Unit No.1 (See Note A)	0.20 rem	negligible	1.2 rem	ngeligible
Accident in Unit No. 2 (See Note B)	0.13 rem	2.5 rem	0.44 rem	2.5 rem

Note A

This accident involves core meltdown with containment integrity augmented by the fact that all potential leakage would be released from the top of the superheater stack of Unit No. 1. The location of the control room at less than 100 feet from the base of this stack results in negligible diffusion from the point of release to the control room as shown in Section 7 and in the Supplement, Appendix B, of the Final Harards Summary Report for the Consolidated Edison Indian Point Reactor Core B, Docket 50-3. Whole body exposure was calculated as described in Section 7.2 of the same report.

Note B

This case in the hypothetical meltdown accident described in Section 12.2.4 of the PSAR. Containment leakage is terminated one minute after the accident by the combined effectiveness of the Isolation Valve Seal Water System and the Penetration and Weld Channel Pressurization System. The dispersion model described in Section 12.2.4.8 of the PSAR was used to determine the fission product iodine concentration. It was assumed that control room occupants are exposed to air at the concentration existing outside the building throughout the stated period. The whole body dose was obtained without consideration for the shielding afforded by the control room structure, and is consistent with the results plotted in Figure 11-5 of the PSAR.

18. Provide the anticipated pressure-flow characteristics for the safety injection and the charging pumps.

Answer

Safety Injection Pumps

The safety injection pumps are the horizontal-centrifugal type. The following pressure-flow characteristics have been specified:

Design Flow	400 gpm
Design Head	2500 ft
Shut-off Head	3500 ft
Maximum Flow	650 gpm
Head at Maximum Flow	500 ft
Number of Pumps	3

A head versus flow characteristic curve will be prepared by the pump manufacturer after his selection. The effect of the actual design curve upon the loss-of-coolant transient analysis will be investigated.

Charging Pumps

The charging pumps are the positive displacement type with variable speed drive having the following preliminary characteristics:

Design Flow	125 gpm
Normal Head	2500 psi
Number of Pumps	2

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    a. For missile protection of the containment and other engineered safeguards during an assumed instantaneous rupture of the largest pipe of the primary system.

Answer

Missile protection for the Indian Point plant will be provided to comply with the following criteria:

a) The containment and liner shall be protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to an including the double-ended severance of a main coolant pipe.

b) The engineered safeguards systems and components required to maintain containment integrity and to meet the site criteria of 10 CFR 100 shall be protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria will be developed and implemented using the following considerations:

a) The reactor coolant system will be surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe and designed to stop the missiles.

b) The structural design of the missile shielding will take into account both static and impact loads and will be based upon the state of the art missile penetration data.

c) Missile velocities will be calculated considering both fluid and mechanical driving forces which can act during missile generation.

d) Components of the reactor coolant system will be examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.

The types of missiles for which missile protection will be provided are:

a) All valve stems up to and including the largest size to be used
b) All valves up to and including the largest size to be used
c) Massive chunks of metal up to 6 inches thick
d) All valve bonnets
e) All instrument thimbles
f) Various type and sizes of nuts and bolts
g) Pieces of pipe up to 10-inch diameter striking broadside or end on
h) Complete control rod drive mechanisms
i) Reactor vessel head bolts

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    b. For pipe motion resulting from an assumed instantaneous rupture of the largest pipe of the primary system.

Answer

The steam generators will be supported in a manner which will prevent rupture of the secondary side of a steam generator and/or main steam and feedwater piping as a result of the thrust forces created by the rupture of a reactor coolant pipe. As necessary, each steam generator will be provided with additional supports which will supplement the gravity supports and will limit the motion of the steam generators under the reaction forces due to the reactor coolant pipe break to a distance that is compatible with the flexibility of the main steam and feedwater piping. The support system design will be sufficient to resist the forces from an assumed break at any point in the reactor coolant system.

Ancillary piping, exclusive of safety injection piping, will not be provided with any special supports to prevent rupture or failure as a result of a reactor coolant pipe rupture. Supports for the reactor vessel and the other main reactor coolant piping will be sufficient to prevent rupture or failure of any safety injection line not connected to a pipe assumed to rupture. All piping which is connected to the reactor coolant system and which penetrates the containment will be provided with sufficient anchorage and load limit controls to prevent violation of the containment at the containment penetration as a result of a reactor coolant pipe break.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    c. For steam generators with respect to tube or tube sheet failure due to rupture of either the primary or secondary piping.

Answer

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2250 psi across the tubes and tube sheet from the primary side or a maximum pressure differential of 1100 psi across the tubes and tube sheet from the secondary side, respectively. Under these conditions there shall be no rupture of the primary to secondary boundary (tubes and tube sheet). This criteria prevents any violation of the containment.

To meet this criterion it has been established that under the postulated conditions, where a primary to secondary side differential pressure of 2250 psi exists, the stresses in the tube sheet ligaments (the highest stressed member) shall not exceed 90% of the yield stress at the operating temperature. An examination of the stresses under these conditions shows that the primary to secondary side differential could go as high as 2485 psi before the stress limitation of 90% of the yield strength is reached in the tube sheet ligaments. The membrane stress in the tubes under a 2250 psi differential is about 21,000 psi which is well under both the ASME code allowable stress (23,300 psi) and 90% of the 27,400 psi yield strength for the Inconel tubes at 650°F.

In the case where the secondary to primary differential is 1100 psi, the tube sheet stresses will obviously not exceed 90% of yield. Under this 1100 psi differential, the tube membrane stress of 10,250 psi (compression) which is 37% of the yield strength for Inconel at 650°F. Actual pressure tests on 3/4 in. O.D. - 0.058 in. wall Inconel tubing show collapse under external pressure in the order of 5700 to 5900 psi. Extrapolating this data to 7/8 in. O.D. - 0.050 in. wall tubes, as used in the Indian Point Unit No. 2, indicates tube collapse would occur at about 2630 psi (at 650°F). This gives a factor of safety of 2.4 against collapse under the postulated 1100 psi secondary to primary differential pressure.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    d. For the maximum permissible primary and secondary coolant activity during unrestricted power operation.

Answer

The design primary and secondary coolant activities are determined in such a way that normal or accidental releases of activity will not exceed the 10CFR20 yearly dose limits at the site boundary so that potential exposures will be well below the requirements of 10CFR100.

The resulting design fission products activities in the primary coolant are:

800 equivalent micro curies of Xe-133 per milliliter of primary coolant,
60 equivalent micro curies of I-131 per milliliter of primary coolant
30 υ c of Cs-137 per cubic centimeter of primary coolant

These concentrations are much higher than those expected during normal unrestricted power operation.

The resulting design fission products activities in the secondary coolant are:

6.15×10^{-3} equivalent micro curies of I-131 per milliliter of steam generator secondary coolant, and
2.76×10^{-2} micro curies of Cs-137 per milliliter of steam generator secondary coolant,
Fission gases are not considered because they will not accumulate.

The steam generator tubes and tube sheet form a barrier between the primary and secondary systems so that no appreciable leakage of primary coolant into the secondary coolant is expected during normal unrestrected power operation and the levels will be well below these values.

The postulated failures evaluated to establish the limits for the primary coolant activity levels are: (1) steam generator U-tube rupture, (2) uncontrolled discharge of the fission gases stored in one gas storage tank after an emergency shutdown, and (3) volume control tank rupture. The atmospheric model is as illustrated in the answer to Question 19-f. The maximum whole body dose to an individual standing at the site boundary is 0.5 rem and the thyroid dose is 1.0 rem.

When a complete severance of a steam generator U-tube is assumed, while the primary coolant is at its limit concentration, the activity in the blowdown header will increase, the affected steam generator will be identified and the remote-operated shutoff valve on the steam line of this steam generator will be closed. A high activity alarm on the air ejector of the secondary system will divert the air ejector flow to the containment. The design of the system will be such that the gaseous activity released to the environment prior to diverting the air ejector to the containment will not exceed the amount required to meet the 0.5 rem whole body dose criterion (about 13,500 curies of Xe-133).

The limit for the activity level in a gas decay tank is calculated in more detail in answer to Question 19-f. A volume control tank rupture is assumed to occur when a primary coolant feed has practically filled up the tank ($\sim$600 ft^3). As a consequence of rupture, less than 5% of the stored water is calculated to evaporate in cooling to ambient temperature. It is conservatively assumed that the evaporation of this amount of water causes all of its contained fission products, including the iodine and cesium, to become airborne. All of the gaseous fission products in the 600 ft^3 are assumed to escape. The fission product release is exhausted through absolute and charcoal filters prior to reaching the environment via the plant vent. A 90% removal efficiency by the filter is assumed for the iodine.

The worst postulated failure that can lead to the release of secondary coolant fission products is a steam line break. The consequence of this rupture is assumed to be a rapid flashing of most of the water in the shell side of the steam generators and subsequent evaporation of all the secondary water present in the feedwater system up to and including that in the hot well of the condenser. This water is assumed to be at the limit contamination and to carry over all the fission products in solution.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    e. For containment vessel penetrations. Provide a list of all penetrations and the type of isolation planned.

Answer

Table 19-e is a list of all containment penetrations with suitable references to criterion No. 22 on containment isolation, the corresponding process flow diagrams in the Preliminary Safety Analysis Report, and the seal water injection provisions.

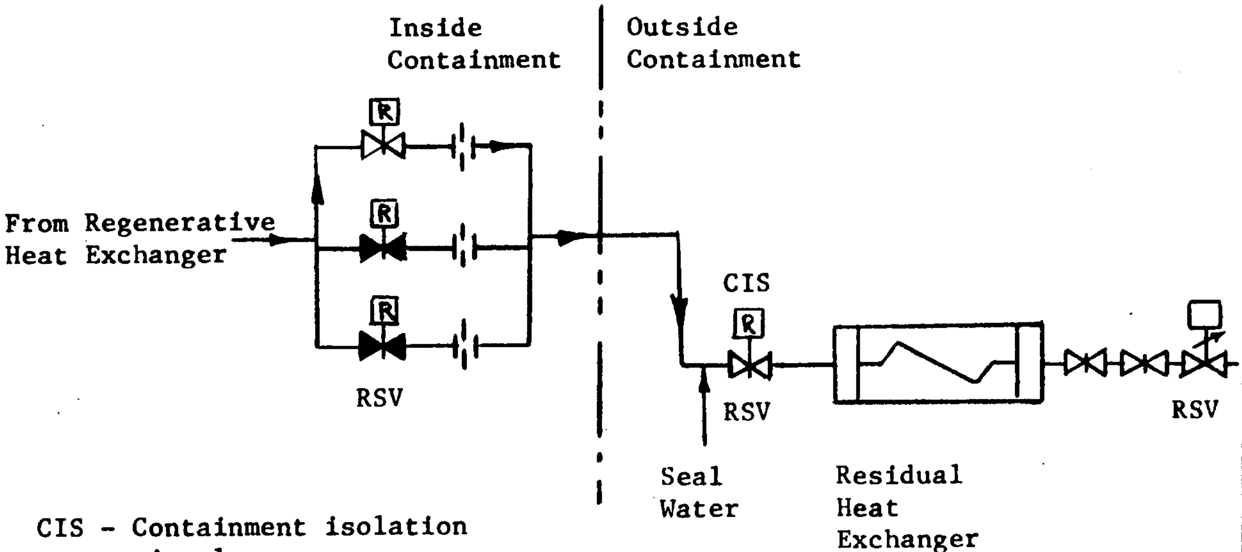
In many cases, additional isolation equipment is provided above that shown in the flow diagrams published with the Preliminary Safety Analysis Report. Where such changes have been made, they are indicated in the list.

The philosophy for isolation of the lines in the safety injection system is discussed in greater detail in the answer to Question 9.

REPRESENTATIVE PENETRATION ISOLATION POINTS

FIG. 19e-1 LETDOWN LINE - Penetration No. 30

Ref. Figure 9-1, Preliminary Safety Analysis Report



CIS - Containment isolation signal

FIG. 19e-2 SEAL WATER RETURN - Penetration No. 36

Ref. Figure 9-1, Preliminary Safety Analysis Report

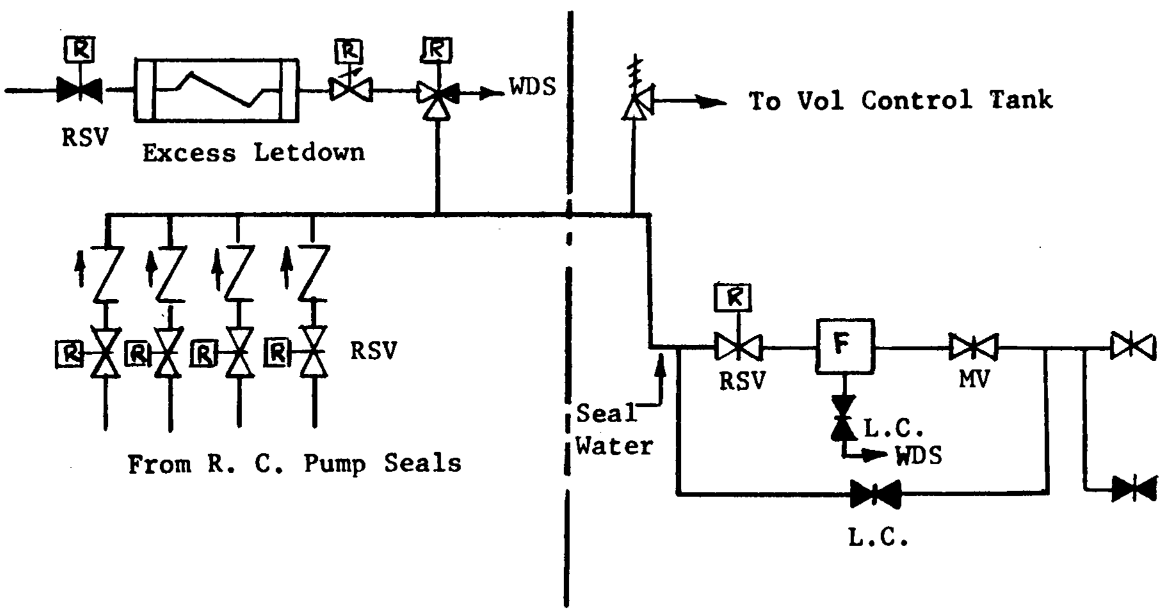


TABLE 19e

INDIAN POINT UNIT NO. 2

CONTAINMENT PENETRATION LIST

		SYSTEM	REFERENCE FLOW DIAGRAM	APPROX LINE SIZE inches	CONTAINMENT ISOLATION INSIDE TYPE	INSIDE POSITION	OUTSIDE TYPE	POSITION	SEAL WATER INJECTION	FLUID G-GAS W-WATER	TEMPERATURE >200F; HOT <200F; COLD	PENETRATION CLASS CRITERION 22
1	Pressurizer Relief Tank Vent	RCS	Fig.4-2 PSAR*	3/4	RSV	Closed	RSV+RSV	Open	A	G	Cold	2
2	Nitrogen Supply	RCS	Fig.4-2 PSAR*	3/4	Check	---	RSV+Press. Reg.	Closed	A A	G G	Cold	3
3	Pressurizer Relief Tank Makeup	RCS	Fig.4-2 PSAR*	2	Check	---	RSV+Check	Closed	M	W	Cold	3
4	Primary System Vent Header	RCS	Fig.4-2 PSAR*	2	---	---	RSV+RSV	Open	A	G	Cold	2
5	Drain Header	WDS	Fig.11-1 PSAR*	2	---	---	RSV+RSV	Open	M	W	Cold	2
6-9	Steam	Sec.	---	28	Missile Protected		non return + throttle stud	Open	No	G	Hot	4
10-13	Feedwater	Sec.	---	8	Missile Protected		Check+MV	Open	No	W	Hot	4
14-17	Steam Generator Blowdown	Sec.	---	2	---	---	RSV+RSV	Open	A	W	Hot	2
18	Residual Heat Removal Loop out	ACS/SIS	Fig.6-1(R) Ques.#9	10	RSV	Closed	Closed System	---	M	W	Hot	See Question #9
19	Residual Heat Removal Loop in	ACS/SIS	Fig.6-1(R) Ques.#9	8	RSV	Closed	Closed System	---	No	SIS Recirculation		See Question #9
20-23	R.C. Pump Cooling Water in	ACS	Fig.9-2 PSAR*	4	Check	---	MV+Closed System	Open	M	W	Cold	3
24-27	R.C. Pump Cooling Water out	ACS	Fig.9-2 PSAR*	4	MV	Open	RSV+Closed System	Open	M	W	Cold	2
28	Excess Letdown Heat Exchanger Cooling Water in	ACS	Fig.9-2 PSAR	2	Missile Protected		MV+Closed System	Open	No	W	Cold	4
29	**Excess Letdown Heat Exchanger** Cooling Water out	ACS	**Fig.9-2 PSAR**	2	**Missile Protected**		**MV+Closed System**	Open	No	W	Cold	4
30	Letdown Line	CVCS	Fig.19e-1 Ques.#19*	3	RSV	Open	RSV+RSV	Open	A	W	Hot	1
31	Charging Line	CVCS	Fig.9-1 PSAR	3	Check	---	MV+Closed System		M	W	Cold	3
32-35	R.C. Pump Seal Water Supply	CVCS	Fig.9-1 PSAR	2	Check	---	RSV+Closed System	Throttle	M	W	Cold	3
36	R.C. Pump Seal Water Return	CVCS	Fig.19e-2 Ques.#19*	4	RSV	Open	RSV+MV	Open	M	W	Cold	2
37	**Reactor Cool. System** Sample Line	**SS**	**Fig.9-3 PSAR***	**3/8**	**RSV**	**Closed**	**RSV+RSV**	**Closed**	A	W	Hot	1
38	Residual Heat Removal Loop	SS	Fig.9-3 PSAR*	3/8	RSV	Closed	RSV+RSV	Closed	A	W	Hot	1

CONTAINMENT ISOLATION

		SYSTEM	REFERENCE FLOW DIAGRAM	APPROX LINE SIZE inches	INSIDE TYPE	INSIDE POSITION	OUTSIDE TYPE	POSITION	SEAL WATER INJECTION	FLUID G-GAS W-WATER	TEMPERATURE >200F; HOT <200F; COLD	PENETRATION CLASS CRITERION 22
39	Fuel Transfer Tube	Fuel Trans	Fig.5-4 PSAR	~20	Double Gasket Blind Flange		MV	Closed	No	G	Cold	6
40	Instrument Air	Sec. Syst.	--- ---	---	---	---	Check+RSV	Open	A	G	Cold	3
41	Gas Analyzer	RCS	Fig.4-2 PSAR*	3/8	---	---	RSV+RSV	Closed	A	G	Cold	2
42	Containment Air Sample in	Rad. Mon.	--- ---	1	Check	---	MV+Closed System	Open	A	G	Cold	3
43	Containment Air Sample out	Rad. Mon.	--- ---	1	---	---	RSV+RSV	Open	A	G	Cold	2
44	Refueling Canal Fill	SIS	Fig.6-1(R) Ques.#9	8	MV	Closed	Closed System	---	M	W	Cold	See Question #9
45-46	Containment Purge Ducts	Vent.Syst.	--- ---	~36	RSV	Closed	RSV	Closed	No	G	Cold	6
ENGINEERED SAFEGUARDS PENETRATIONS												
47	Safety Injection Line	SIS	Fig.6-1(R) Ques.#9	6	RSV+Check	Closed	Closed System	---	No	W	Cold	See Question #9
48-49	Containment Spray	SIS	Fig.6-1(R) Ques.#9	3	MSV+Check	Open	Closed System	---	M	W	Cold	See Question #9
50-51	Containment Sump Recirc. Lines	SIS	Fig.6-1(R) Ques.#9	13	---	---	Closed System	---	No	W	Cold	See Question #9
52	Safety Injection Test Line	SIS	Fig.6-1(R) Ques.#9	2	---	---	Closed System	---	M	W	Cold	See Question #9
53	Ventilation Coolers-water in	Ventilation System	--- ---	6	Closed System		MV	Open	No	W	Cold	4
54	Ventilation Coolers-water out	Ventilation System	--- ---	6	Closed System		MV	Open	No	W	Cold	4
55-58	Penetration Pressurization	Pene.Press. System	Fig.5-6 PSAR	1	Missile Protected		MV+Closed	Open	No	G	Cold	4

Notes: *Indicates a change from the isolation shown on the reference drawing in the Preliminary Safety Analysis Report (PSAR)

Valves:
RSV - Remote Stop Valve
MV - Manual Valve
RSV+RSV - Remote Stop Valves in series

Seal Water:
A - Automatic Seal Water Injection
M - Manual Seal Water Injection

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    f. For the radioactive gas hold-up tank. What is the maximum radioactive fission product inventory that can be stored in the gas storage tanks? Describe the environmental consequences of a storage tank leak. Describe provisions to monitor gaseous releases for iodine.

Answer

The gas decay tanks contain the gases vented from the reactor coolant system, the volume control tank, and the liquid waste tanks in the waste disposal system. Sufficient volume is provided in the tanks to store the gases evolved during a reactor shutdown. The system is adequately sized to permit storage of these gases for at least 45 days prior to discharge. The tanks are designed and constructed in accordance with the applicable requirements of Section III of the ASME Code for Class C Nuclear Vessels, and as Class I components under the seismic design criteria.

The 45 day retention period is selected as the maximum foreseeable holdup time because in this period the shorter lived radioactive gaseous isotopes received by the waste system will have decayed to a level which is less significant than that of long-lived Kr-85. Further holdup would serve no useful purpose, as the yearly production of Kr-85 can be dispersed to the atmosphere well within the limits of 10CFR20. As indicated on page 11-5 of the Preliminary Safety Analysis Report, these 10CFR20 limits would allow a discharge of approximately 1.42×10^7 curies of Kr-85 per year with dispersion conditions represented by the worst monthly data which is orders of magnitude above the expected yearly releases of Kr-85.

Maximum Fission Product Inventory and Consequences of Release

The radioactive inventory of any one gas holdup tank less than that quantity whose inadvertant release could be detrimental to the health and safety of the public. Because of the high standards of integrity imposed in the design of the tanks, and close control over the venting operation, it is acceptable to store in one tank a quantity of gas whose release would not expose an individual at the site boundary to a potential dose exceeding 10CFR100 limits.

In the Indian Point Unit #2 waste disposal system, the maximum anticipated quantity of gaseous wastes in one tank is approximately equivalent to 13,500 curies of Xe-133. An inadvertant release of this inventory would result in a maximum exposure of 0.5 rem at the site boundary during the passage of the cloud. This dose is well below the limit of 25 rem stated in 10CFR100. The following assumptions were made in calculating the exposure:

a. Gas is released to the plant vent, either by uncontrolled discharge through the normal venting line or by leakage to the compartment ventilation system.

b. Dispersion in the atmosphere from the point of release occurs first by turbulent mixing (downwash) in the wake of the containment building and subsequently by diffusion under inversion conditions (see pages 12-41 of the Preliminary Safety Analysis Report).

c. The wind speed throughout the release is one meter per second.

d. The receptor remains at the point of maximum ground level concentration during the entire passage of the cloud.

Protection Against Release of Iodine and Particulates

Experience with the operation of waste gas processing facilities in other PWR plants (Saxton and Yankee) indicates that the gas holdup tanks will not contain iodine or particulate activity in quantities whose release would be of any consequence. Nevertheless, the gases to be discharged will be sampled to determine its isotopic content (including that of radioiodine). The plant vent gas and particulate monitor is also used to determine the discharge of activity to the environment.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    g. For the liquid waste collection tanks considering necessary capacity during accidental release.

Answer

Three liquid waste collection tanks, each sized to hold two-thirds (2/3) of the reactor coolant liquid volume, will be used to process the normal wastes produced. The contents of one tank will be passed through the liquid processing train while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity if needed.

The liquid waste collection tanks are located in the tank pit adjacent to the auxiliary building. Any liquid leakage from these tanks or piping will be collected in the tank pit sump to be pumped back into the liquid waste system. The tank pit sump and basement volume are sufficient to hold the full volume of a liquid waste collection tank without overflowing to areas outside the building. The tank pit will be vented to the ventilation system so that any gaseous leakage would be safely discharged via this route.

The system will be designed so that gaseous activity which could be released in the unlikely event of a tank rupture will not result in potential off-site doses in excess of the limits of 10 CFR 20. The potential doses will be checked during the course of detailed design based on conservative assumptions for gaseous activity present in the liquid, for release of gaseous activity from the spilled liquid and the site dispersion model for release from the plant vent.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    h. For the fuel hold-down fixtures considering the uplift forces during a major loss of coolant accident.

Answer

For all break sizes up to the largest pipe connected to the Reactor Coolant System (or equivalent), a design objective of the rod cluster control is to assure shutdown by rod insertion only. For all breaks, up to the double-ended severance of a reactor coolant pipe, the accidents will be investigated to show that the reactor will be shutdown, considering all available means. The effects of hydraulic forces on the rods and internals will be evaluated in assuring that these criteria are met.

On the basis of the above, times to establish shutdown of the reactor will be determined, and any energy from delay in shutting the reactor down will be appropriately handled in the containment cooling systems design heat loads.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

    i. To limit core drop if the upper support fails. Explain how the "in-core" instrumentation structure will be designed to limit the core drop. What are the consequences of the maximum potential reactivity insertion under these conditions?

Answer

Because of the care taken in the design and fabrication and the relatively low stresses imposed during normal operation, the failure of the core support flange and the subsequent drop of the core is considered to be incredible. If such an accident is postulated, the downward vertical displacement of the internals will be limited by the in-core instrumentation structure and energy absorbing devices attached to the instrumentation tie plate at locations concentric with the 4" diameter instrumentation guide tubes. In the event of this accident, the energy absorbers would contact the vessel bottom head. The load would transfer from the vessel to the energy device, through the instrumentation tie plate directly to the instrument guide tubes above the point of load application, and indirectly to other instrument guide tubes. The guide tubes, bolted to the instrumentation tie plate at their lower end, are supported from the underside of the core support forging at their upper end.

The energy absorbers, cylindrical in shape, will be contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design will be determined so as to limit the forces imposed to a safe criteria. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing device.

The free fall in the hot condition will be on the order of 1/2 inch, and there will be an additional strain displacement in the energy absorbing devices of approximately 3/4 inch. In the cold condition the free fall will be on the order of 1 inch and again an additional strain displacement of approximately 3/4 inch.

In summary, the design criteria are:

1) To limit the downward vertical displacement to a safe distance.

2) To limit the imposed forces so that the reactor vessel is within ASME Section III Code standards and that energy absorbing devices in the internals be used to absorb the dynamic energy of the internals drop.

In the event of this vertical displacement, core alignment will be maintained by the radial supports which are described in Chapter 3 of the Preliminary Facility Description and Safety Analysis Report. Control rods will insert under this condition.

The reactivity addition due to such a 1-1/4 inch drop with the rods in their most reactive position is of the order of 0.2 to 0.3% δk and is well within the capability of the reactor protection system to safely shut down the reactor.

19. State proposed design criteria, justification, how these criteria will be fulfilled by the design proposed, and (where applicable) test methods:

j. For the operational reliability of reactor safety, containment isolation, and engineered safeguards systems. Important equipment such as sensors, valves, solenoids, breakers, switches, pumps, cooling water, injection water, etc. should be considered. The "fail safe" and/or redundant features or lack of some should be discussed and, where important to an evaluation of system adequacy, test provisions and the criteria relative to frequency of tests should be stated and justified.

Answer

The details of the reactor safety, containment isolation and engineered safeguards systems are discussed in Question 2, Criteria 15, 22 and 23 and Questions 6, 7, 8, 9 and 10.

CONTAINMENT DESIGN INFORMATION

1. In load criteria on p. 5-8 (Exh. B, Vol. 2, Part A), what values of parameters correspond to the values of T, TLT', TL'? What are the temperatures in the interior and at the steel locations?

2. How is the lateral force (shear) in the structure carried? Is there shear reinforcement? For the liner, elastic stability provisions and load capacity based on yield are noted on p. 5-9. How does the steel liner participate in carrying the shear and other loads? What anchorage means is contemplated?

3. How will the splicing of the large 14S and 18S bars be handled? A general sketch of the contemplated reinforcing bar patterns is desirable.

4. What special provisions or special studies will be made to insure the adequacy of the penetrations (large and small) in terms of retaining strength and ductility while preventing leakage. Details of the concept of reinforcement around penetrations are desirable.

5. A tabulation of sources of stress, along with the appropriate allowable stress (or permissible resisting load and load factor) values, would help clarify the design approach. Also, a discussion of allowable ductility and provisions for obtaining same is desirable.

6. What magnitude of vertical acceleration in earthquakes will be considered?

7. In citation (ii) it is noted that "...any vertical acceleration would be counteracted by the weight of the building". This statement is not correct. Also, vertical seismic motion should be assumed to act simultaneously with horizontal excitation. A more scientifically valid criterion for the earthquake analysis is required.

8. In the table of damping values given on p. 5-16, the damping factor for the containment structure is shown as 7.0 per cent of critical and for the concrete support structure, item 2, 5 per cent. Similar values are shown in item 5. On what basis were these selected? Such large values correspond to rather heavily cracked concrete sections, stressed well into the yielding range. Lower values would be much more reasonable.

9. A description of the actual analysis techniques that will be employed in arriving at the design would be helpful. Only indirect statements about the procedures to be followed are given in the report on p. 5-16. What rigorous and acceptable procedures will be followed? How will the response spectra be employed in the procedures?

10. What criteria exist for adding the stresses arising from the different loadings, in contrast to combining loads? Since the loads act in different directions in many cases, a stress (or load resistance) combination approach would appear to be more rational. Discussion and comment is needed.

11. What wind loads will be assumed in the design?

Answer

The answers to the specific questions are included in the attached summary of Containment Design Information. The following list gives the section in which these questions are answered.

Question 1	Section II
Question 2	Sections I, IV
Question 3	Section V
Question 4	Section VIII
Question 5	Sections II, IV, V, VII
Question 6	Section III
Question 7	Section III
Question 8	Section III
Question 9	Section III
Question 10	Sections II, IV
Question 11	Section II

CONTAINMENT DESIGN INFORMATION

PREFACE

The following summary of containment design criteria is presented in response to the additional Containment Design Information requested. The answers to the eleven itemized questions are contained in this summary.

Numbers in parenthesis shown thus, (7), refer to reference material listed in the bibliography also included with this report.

The order of presentation by sections is as follows:

Section I	General Concept
Section II	Design Stress Criteria
Section III	Seismic Criteria
Section IV	Stress Analysis
Section V	Reinforcing Steel
Section VI	Concrete
Section VII	Liner
Section VIII	Wall Penetrations & Wall Openings
Section IX	Shop and Field Testing
Section X	Bibliography

I. GENERAL CONCEPT

The reactor containment is a reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a generally flat base supported on rock. The inside surface of the structural concrete is lined with steel plate anchored to the concrete shell. The liner is designed and fabricated to prevent leakage through it due to an accident resulting in the loss of reactor coolant and release of radioactive material. The entire containment is designed to contain all radioactive material which might be released from the reactor core during a loss-of-coolant accident.

The containment which is shown on Figure 1 has side walls which are 147 feet from the liner on the horizontal based to the spring line of the dome and has an inside diameter of 135 feet. The thickness of the reinforced concrete base is 9 feet, the side walls are 4 feet 6 inches and the dome thickness is 3 feet 6 inches. The change in wall thickness of the dome and the cylinder will be accomplished above the spring line such that the inside radius of the dome and cylinder will be equal. The bottom horizontal liner plate will be covered with 2 feet of concrete, the top of which will form the floor of the containment. The internal pressure within the containment is self-contained in that the vector is zero. Therefore, there is no need for mechanical anchorage between the bottom mat and the rock.

The basic structural consideration in the design of the containment structure considers the bottom mat, side walls and dome acting as one structure under all possible loading conditions. The loads considered in the design result from gravity, internal pressure and temperature due to a loss of coolant accident, external earth pressure, earthquake and wind as enumerated in further detail in the discussions which follow. The liner is anchored to the concrete shell by means of Nelson studs so that it forms part of the entire composite structure under all loading in such a manner as to insure vapor tightness from the internal and external loadings. The reinforcing in the structure will have an elastic response to all loads with limited maximum strains to insure the integrity of the steel liner.

II. DESIGN STRESS CRITERIA

The concept of this reinforced concrete containment has as the basic design philosophy the ultimate strength design as set forth in Part IV-B, "Structural Analysis and Proportioning of Members - Ultimate Strength Design," in the ACI standard building code requirements for reinforced concrete (ACI 318-63) published June 1963. The required load capacities of structural elements will be determined by computing stresses resulting from the loading combinations given below.

Loading Combination	Required Load Capacity of Section	
Operating plus loss-of coolant accident	0.95D + 1.5P + 1.0 (T + TL)	(1)
Operating plus loss-of coolant accident plus design earthquake	0.95D + 1.25P + 1.0 (T' + TL') + 1.25E	(2)

D: Dead load of structure including effect of any hydrostatic pressure.

1.5P: Pressure loads resulting from pressure 50% greater than design.

T: Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.5 times design pressures.

TL: Load exerted by the exposed liner based upon temperature associated with 1.5 times design pressures.

1.25P: Pressure loads resulting from pressures 25% greater than design.

T': Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times design pressures.

TL': Load exerted by the exposed liner based upon temperatures associated with 1.25 times design pressures.

E: Load due to acceleration from the design earthquake.

The maximum operating temperature is 120°F. The design 24-hour mean-low ambient temperature is -5°F. The maximum liner temperature under accident conditions is 247°F.

The temperature gradient through the wall is essentially linear and a function of normal operating temperature internally and ambient temperature externally.

Because of the insulating properties of the concrete, accident liner temperatures have a negligible effect on the gradient through the wall. For example, two thousand seconds after the accident, at which time the pressure and liner temperature have fallen off, the accident temperature affects only the inside six inches of the concrete wall.

No member will have a load capacity less than that required by formulae given above for the stresses caused by the most severe loading combination. The stresses in the reinforcement produced by these loading combinations must not exceed yield times the capacity reduction factor across the section being analyzed.

In any element in which the stresses resulting from wind load exceed those resulting from earthquake load, the design wind load of 30 psf will be used in the design of that element. However, calculations to date indicate that wind does not control in this situation for the required unit wind load and the geometry of the containment.

III. SEISMIC CRITERIA

The design of the containment which is a Class I structure will utilize the "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design will be based on the acceleration response spectrum curves developed by G. Housner for the ground acceleration at El Centro. Seismic determinations have been computed as outlined in the AEC TID-7024[6] and Portland Cement Publication[7].

Damping factors will be used as indicated in Table 5-4.

The damping factors shown in Table 5-4 have been reduced for components 1 and 5 (a) from those shown in the Preliminary Safety Analysis Report. The basis for this is the paper presented by J. A. Blume at the ASCE Conference on Earthquake Engineering in 1960 titled "Reserve Energy Technique." Blume writes in part "....5 per cent of critical damping is recommended because (a) it is considered a reasonable, nominal value of damping in the elastic range, (b) it is adequate to iron out many of the extreme peaks and valleys of spectral response for lesser damping,..."

TABLE 5-4

DAMPING FACTORS

	Component	Per Cent of Critical Damping
1.	Containment Structure	5.0
2.	Concrete Support Structure of Reactor Vessel	5.0
3.	Steel Assemblies:	
	(a) Bolted or Riveted	2.5
	(b) Welded	1.0
4.	Vital Piping Systems	0.5
5.	Concrete Structures above Ground:	
	(a) Shear Wall	5.0
	(b) Rigid Frame	5.0

The ground acceleration has been determined to be 0.1 g applied horizontally, and 0.05 g applied vertically. These values have been resolved as conservative numbers based upon the recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

The natural period of vibration is computed by the Rayleigh method; in this method the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to assure full and continued elastic response during seismic motions. Further, both bending and shear deformations are considered (see sketch A).

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined and the end deflection is given the value $\phi' = 1.0$ with corresponding values determined for other sections. The natural period of vibration (T) for the structure is then determined by the relation

$$T = 2\pi \left[\frac{Y_o \, \Sigma \, \phi'^2 \, dm}{g \, \Sigma \, \phi' \, dm} \right]^{1/2} \qquad (3)$$

This expression is derived by setting potential energy equal to kinetic energy and solving for T, wherein terms are defined as follows:

Y_o = Maximum Actual Deflection

$$\phi' = \frac{\text{Deflection of Section Under Consideration}}{\text{Maximum Actual Deflection}}$$

g = Acceleration Due to Gravity

dm = Weight of Section Under Consideration

Using the derived period, T, and entering the average acceleration spectral curves and applying 5% critical damping, a spectral acceleration for the containment is selected. Since this average curve is based upon a ground acceleration 0.33 g, the average spectral acceleration is multiplied by 0.1/0.33 for the containment structure acceleration with the seismic loading to be used for this plant. This value is derived to determine the base shear. The distribution of base shear will be upon a triangular loading assumption based upon the formula (see sketch A-1)

$$F_x = \frac{V\, w_x\, h_x}{\Sigma\, w\, h}$$

which is from the Structural Engineers Association of California (SEAOC) Code, with zero loading at the base to a maximum loading at the spring line of the dome. Above this line the loading will decrease due to change in section. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

IV. STRESS ANALYSIS

The analysis of the bottom mat, cylindrical wall, and dome of the containment will be in accordance with the principles stated in this section.

All membrane stresses will be carried by the reinforcement and steel liner and none by the concrete. This statement reiterates a basic ground rule; namely, that the concrete will not be counted upon to resist stresses other than compression, bond and shear. However there will be need for radial shear reinforcing in the lower portion of the wall in the form of stirrups or bent bars, and the need for diagonal shear reinforcing in the circumferential direction to resist earthquake shears for the full height of the wall and a distance above the spring line into the dome until a point is reached where the dome liner can resist the total shear. Some of the load carrying capacity of the diagonal shear bars will be used to resist membrane stresses for the design condition which includes 1.5P loading and during loading resulting from the pressure test.

The analysis of discontinuity stresses at changes in section or direction of the containment shell will be made on the assumption that the shell is an elastic homogeneous isotropic material(1)(2)(3). (See sketch B.) In determining discontinuity moments and shears, the mat is considered as offering complete fixity. The entire concrete section of the wall is used in the evaluation of the flexural rigidity. As cracking occurs and the reinforcement takes up the load, a redistribution of stress occurs, and the stiffness of the wall is greatly reduced thereby reducing the discontinuity moments and shears. As a conservatism in the design, however, the reinforced concrete wall is designed to accommodate the moments and shears determined on an uncracked section analysis. The differential equation

$$D \frac{d^4 w}{dx^4} + \frac{Eh}{a^2} w = Z \qquad (5) \qquad (5)$$

represents the basis of solution for all problems of symmetrical deformation of circular cylindrical shells of constant wall thickness.

The solution of this differential equation is

$$w = \frac{e^{-\beta x}}{2\beta^3 D} [\beta M_o (\sin \beta x - \cos \beta x) - (Q_o \cos \beta x)] \qquad (6)$$

For the fixed end condition,

$$(w)_{x=o} = -\frac{1}{2\beta^3 D} (\beta M_o + Q_o) = \delta \tag{7}$$

and

$$\left(\frac{dw}{dx}\right)_{x=o} = \frac{1}{2\beta^2 D} (2\beta M_o + Q_o) = 0 \tag{8}$$

where D is the flexural rigidity

W is the radial deflection of the wall

X is the distance from the intersection of the wall and base

E is the modulus of elasticity

a is the mean radius of the wall

h is the wall thickness

Z is the load intensity

$$\beta = \sqrt[4]{\frac{Eh}{4 a^2 D}}$$

δ is the unrestrained radial deflection of the wall

M_o is the moment at the base

Q_o is the shear at the base

Hence, the base moment and shear can be evaluated and the distribution of moments and shears above the base can be determined.

Temperature stresses of the reinforcing in the containment shell due to maximum thermal gradient will not influence the capacity of the structure to resist forces developed. This conclusion is based on the following rationale (see sketch D):

> Temperature effects will induce stresses in the structure which are internal in nature; i.e., tension outside and compression in the concrete inside (sketch D-1); and the resultant force is zero. Loading combinations (sketches D-2, D-3) concurrent with these

temperature effects may cause local stresses in the outside horizontal and vertical rows of bars to reach yield (sketch D-4); however, as local yielding is reached, any further load is transferred to the unyielded elements (sketch D-5). At the full yield condition, the magnitude of final load resisted across a horizontal and vertical section will be identical to that which would be carried if the temperature effects were not considered. This approach will not affect the overall carrying capacity of the structure and the factor of safety of the structural elements will be consistent. Ultimate strength design strengthens this argument because a basic assumption of ultimate strength design approach is that of transferring stresses through local redistribution and equalization.

Sketch C displays the condition at the corner where the vertical wall ties into the horizontal mat[5]. The deformation of the rock under load is very small compared to the deflection required for the reinforced concrete mat to develop significant flexure. The friction provided by the rough rock surface and the keying action of pit under the reactor vessel prevent relative horizontal motion between the mat and the rock. It is acceptable therefore to provide only an area of reinforcing in each direction equal to 0.25% of area of concrete in the mat, except for the area near the cylindrical wall where there are shears and moments caused by end restraints and by tension forces in the wall resulting from pressure uplift on the dome. All overburden and loose or broken rock will be removed to solid foundation material and backfilled with concrete.

Since the final grades of backfill material outside of the containment will be as shown in Figure 1, there will be a differential earth pressure from one exterior face of the containment to another. This differential has been considered as a static load on the structure.

It will be assumed that during the 115% pressure test of the containment at 54 psig, the liner will contribute to the net overall cross-sectional strength of the structure. The liner, since it will be anchored to the shell by Nelson studs at approrpriate intervals, will therefore not be loaded beyond a .95 yield. The liner will make only a small contribution at accident loading since it will tend to expand faster than the concrete at increased temperatures, and therefore will be stressed first in tension and then in compression. Insulation material will be applied to the inside of the liner cylinder to maintain stress less than .95 yield and to insure elastic stability where thermal stresses are not relieved by expansion of the wall at the junction of the cylinder and the bottom liner.

As previously stated, the liner will assume tangential shear due to seismic loading only in the dome of the containment in areas where the liner is adequate. Internal pressures cause circumferential and meridional tensile membrane stresses in the dome and cylinder as well as secondary radial moments and shears caused by discontinuities.

Dead load results in compressive meridional stresses and compressive and tensile circumferential membrane forces in the dome. The cylinder walls will be in compression vertically, with no circumferential forces except for small forces at the spring line.

Earthquake and wind will result in circumferential shear forces with the maximum force/foot parallel to the direction of motion. The overturning moment will cause vertical forces in the wall and the maximum tensile and compressive forces 180° apart in the direction of motion.

When the liner is in compression as was discussed previously, a tensile force is imposed on the reinforcing bars.

The preliminary magnitudes of forces, deflections, moments, and shears resulting from the loading combinations given in Section II are shown on Figures 2 and 3. Preliminary sketch F is a conceptual arrangement of reinforcing bars.

V. REINFORCING STEEL

Concrete reinforcement will be high-strength billet steel conforming to ASTM A-432 with a guaranteed minimum yield strength of 60,000 psi and ultimate minimum strength of 90,000 psi.

The design limit for tension members (i.e., the capacity of a section required for the loading combination listed in Section II) will be based upon the minimum guaranteed yield stress of the reinforcing steel. The load capacity so determined will be reduced by a capacity reduction factor "ϕ" which will provide for the possibility that small adverse variations in material strengths, workmanship, dimension and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under capacity. For tension members the coefficient "ϕ" will be conservatively established as 0.95. This compares with a coefficient of 0.90 utilized in ACI-318 for ultimate design of flexural members. In a tension member, unlike the case of a flexural member, only the variation of steel strength and not concrete strength is of concern. Also the effect of reinforcement misplacement is not critical as it is for a flexural member. Therefore, the capacity reduction factor of 0.95 is considered to be conservative. All other "ϕ" values conform to ACI-318 requirements.

All reinforcing bars will be spliced by the Cadweld process. This procedure does not involve a change in chemistry in the reinforcing and therefore will not result in a change in ductility. Spot sampling will be made at which time the appropriate length of reinforcing bar, including the weld, will be cut out and tested in tension in order to prove the integrity and reliability of the connection. The minimum elongation of 7% for ASTM-A-432 reinforcing allows more strain than the structure as a whole is capable of withstanding.

VI. CONCRETE

All concrete materials will be in accordance with ACI-318-63. Portland Cement will conform to "Specifications for Portland Cement", ASTM C-150-64, Type I (Normal), or Type II (moderate heat of hydration requirements), or Type III (high early strength). Shrinkage compensating cement will be used in areas where the elimination of secondary stresses due to shrinkage is of importance. Concrete aggregates will conform to "Specifications for Concrete Aggregates," ASTM C-33-64.

Water for mixing concrete will be clean and free of injurious quantities of substances harmful to the concrete or the reinforcing steel.

The strength of the concrete will be specified and shown on the drawings so as to meet the following requirements.

1. For flexural elements the extreme fiber stress in compression will conform to the limits established in ACI 318-63.

2. The shear as a measure of diagonal tension will conform to the limits established in ACI 318-63.

3. The ultimate compressive strength for a standard cylinder of reinforced concrete to be used in this design will be a minimum of 3000 psi in 28 days, or higher as required. In areas where lean fill will be required, the concrete will have a compressive strength of minimum 1500 psi in 28 days.

The concrete will be sampled and tested during construction in accordance with ACI 318-63 to insure compliance with the specifications. An independent testing laboratory will be retained to design the concrete mixes, take samples and perform all tests.

VII. LINER

With the exception of the equipment and personnel hatches, none of the liner components is subject to low ambient temperatures. Normal internal temperatures will be maintained between a minimum of 50°F and a maximum of 120°F. The liner material is ASTM-A442, Gr. 60, which has a minimum guaranteed yield point of 32,000 psi and is a low carbon/high manganese steel made with a grain structure that is specified primarily to insure ductility. Standard rolling mill practice results in nil ductility transition temperature for this material which will be at least 30°F below the operating temperatures of the plant.

Other reactor containment materials of construction are itemized in Table 5-2, page 5-12, of the Preliminary Safety Analysis Report.

VIII. WALL PENETRATIONS AND WALL OPENINGS

The adequacy of penetrations in retaining strength and ductility while preventing leakage will be insured by the following measures. The materials for all components as specified in Table 5-2 of the Safety Analysis Report are selected primarily because of their high ductility.

By design, all penetrations will withstand all stresses imposed on them as a result of normal plant operation and the hypothetical loss-of-coolant accident. Specifically, the joint between the penetration sleeve and the building liner plate will be reinforced with a steel plate ring. The sleeve will be anchored to the concrete by means of reinforcing bars welded to a steel ring which is in turn welded to the sleeve. The penetration end plates through which the pipes or electric cable pass will be designed to withstand the penetration's internal air pressure during normal operation and also the containment internal pressure during the hypothetical loss-of-coolant accident.

Load transfer around penetrations will be accomplished by maintaining continuity of main reinforcing bars by bending and the addition of diagonal reinforcing to insure the transfer of tensions, bending moments and shears. At the equipment access opening, a reinforced concrete frame is provided to carry stresses around the opening and to resist bending and torsional moments created by the load transfer. Again, main reinforcement will be bent to maintain continuity or stress and additional diagonals will be provided to insure load transfer (see sketches G and H).

IX. SHOP AND FIELD TESTING OF CONTAINMENT

The integrity of the containment liner plates, welds and penetration materials will be assured by sufficient non-destructive testing. The plates are 1/4" thick on the horizontal bottom mat, 3/8" in the cylindrical wall section and 3/8" in the dome.

Shop and field weld tests for welds and welders are outlined in the Preliminary Safety Analysis Report. Quality control will be obtained and assured by a quality control engineer and the craft supervisor assigned to the job.

X. BIBLIOGRAPHY OF SOURCE MATERIAL USED AS REFERENCE FOR CONTAINMENT ASSUMPTIONS AND DESIGN (EXCEPT CODES)

1. S. Timoshenko, Theory of Plates and Shells, First Edition - McGraw-Hill Co., Inc. - 1940.

2. S. Timoshenko and S. Woinowsky-Krieger - Theory of Plates and Shells, 2nd Edition - McGraw-Hill Co., Inc. - 1959.

3. M. Hetenyi, Beams on Elastic Foundations - University of Michigan Press - 1961.

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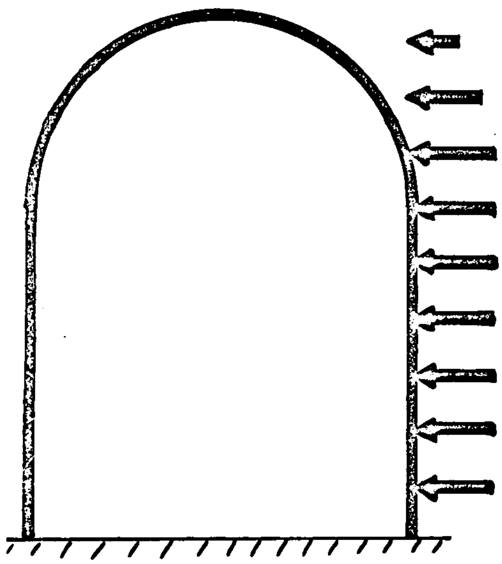
5. P. Ferguson - Reinforced Concrete Fundamentals - John Wiley and Sons, Inc. - 1958.

6. United States Atomic Energy Commission - 1963 - Nuclear Reactors and Earthquakes, TID-7024.

7. J. Blume, N. Newmark, L. Corning - Design of Multistory Reinforced Concrete Buildings for Earthquake Motions - Portland Cement Association - 1961.

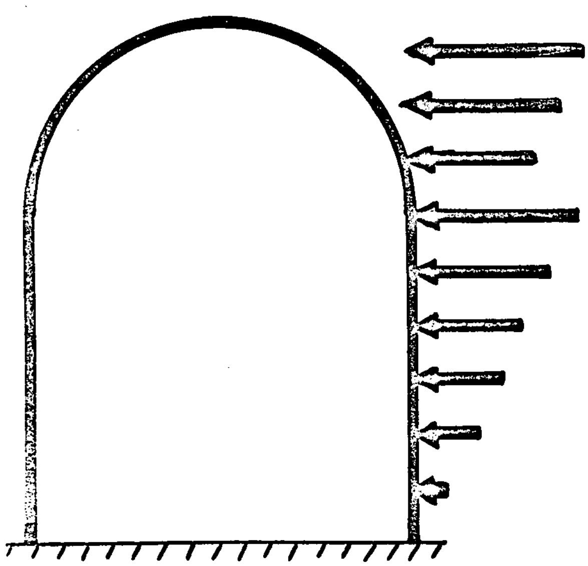
SEISMIC ANALYSIS



$$T = 2\pi \left[\frac{Y_0 \sum \phi'^2 dm}{g \sum \phi' dm}\right]^{1/2}$$

RAYLEIGH'S METHOD

SK-A

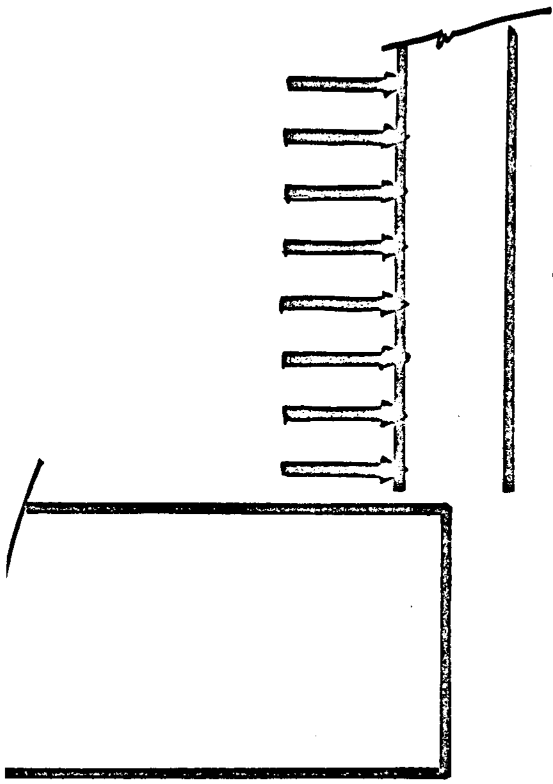


$$F_x = \frac{V w_x h_x}{\sum w h}$$

DISTRIBUTION of BASE SHEAR

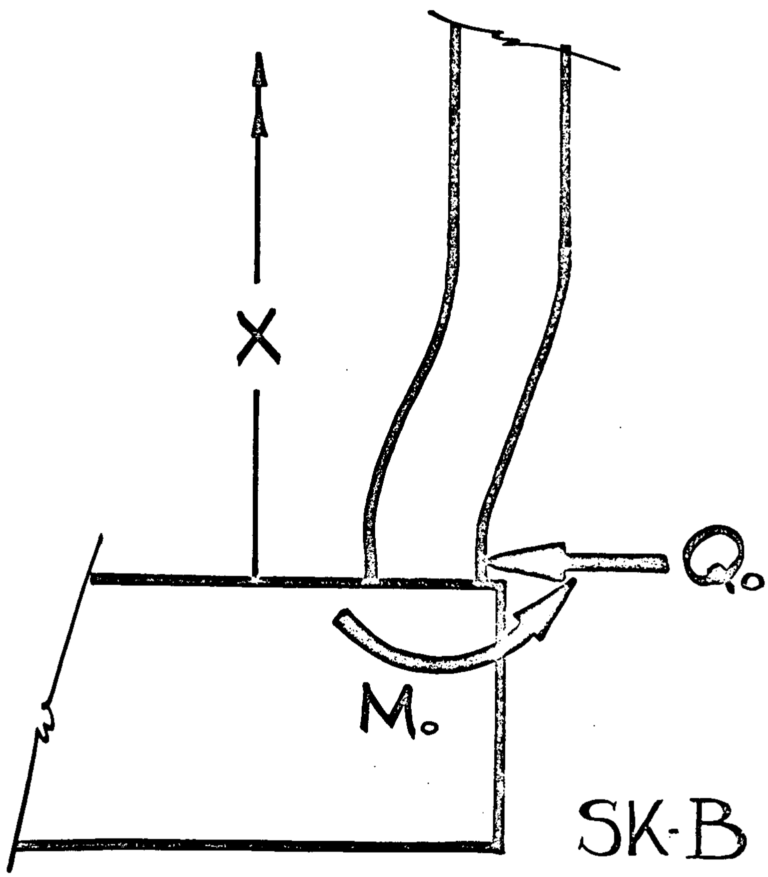
SK-A1

DISCONTINUITY EFFECTS



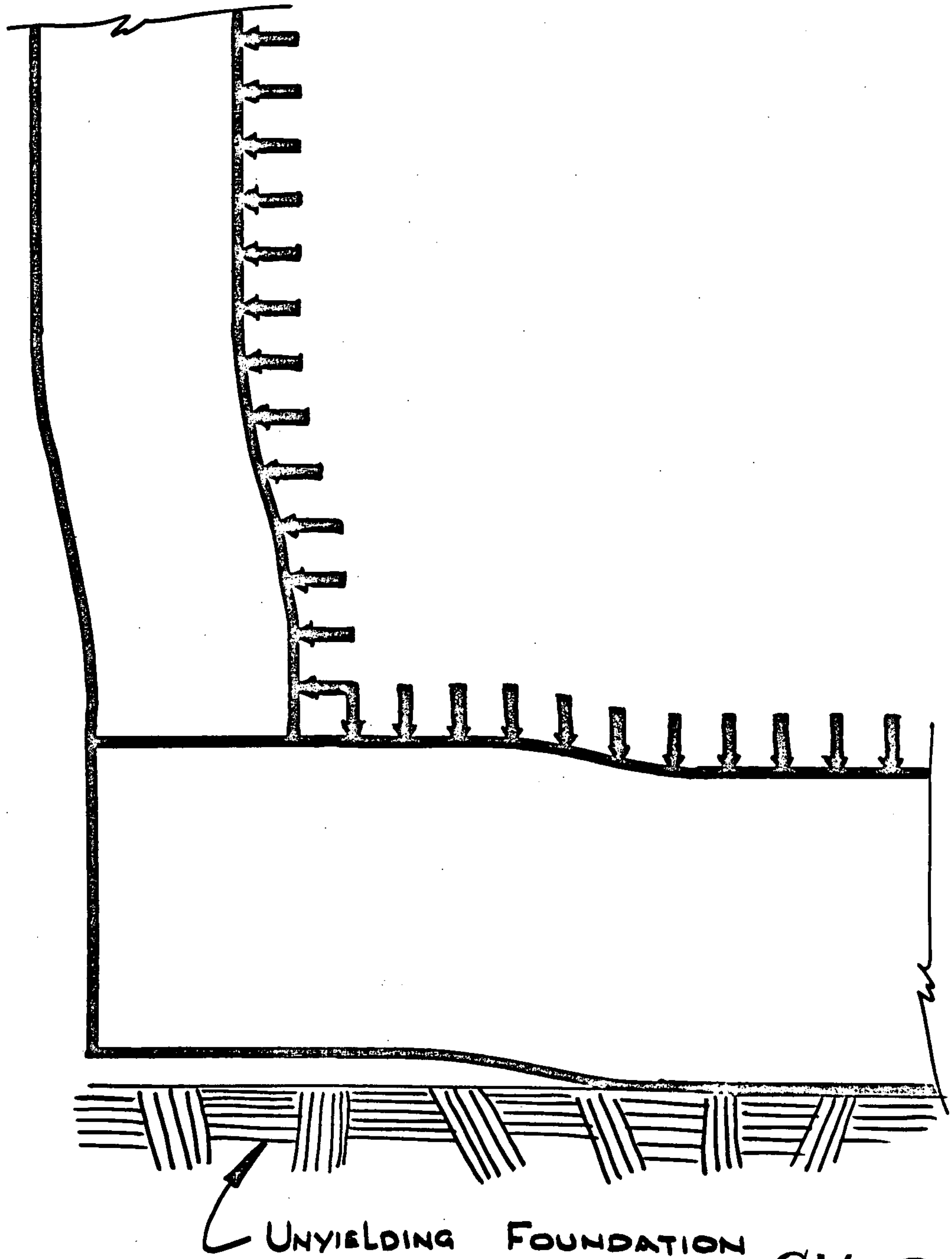
$$D\frac{d^4 w}{d x^4} + \frac{Eh}{a^2} w = Z$$

$$w = \frac{e^{-\beta x}}{2\beta^3 D}\left[\beta M_o(\sin\beta x - \cos\beta x) - (Q_o \cos\beta x)\right]$$



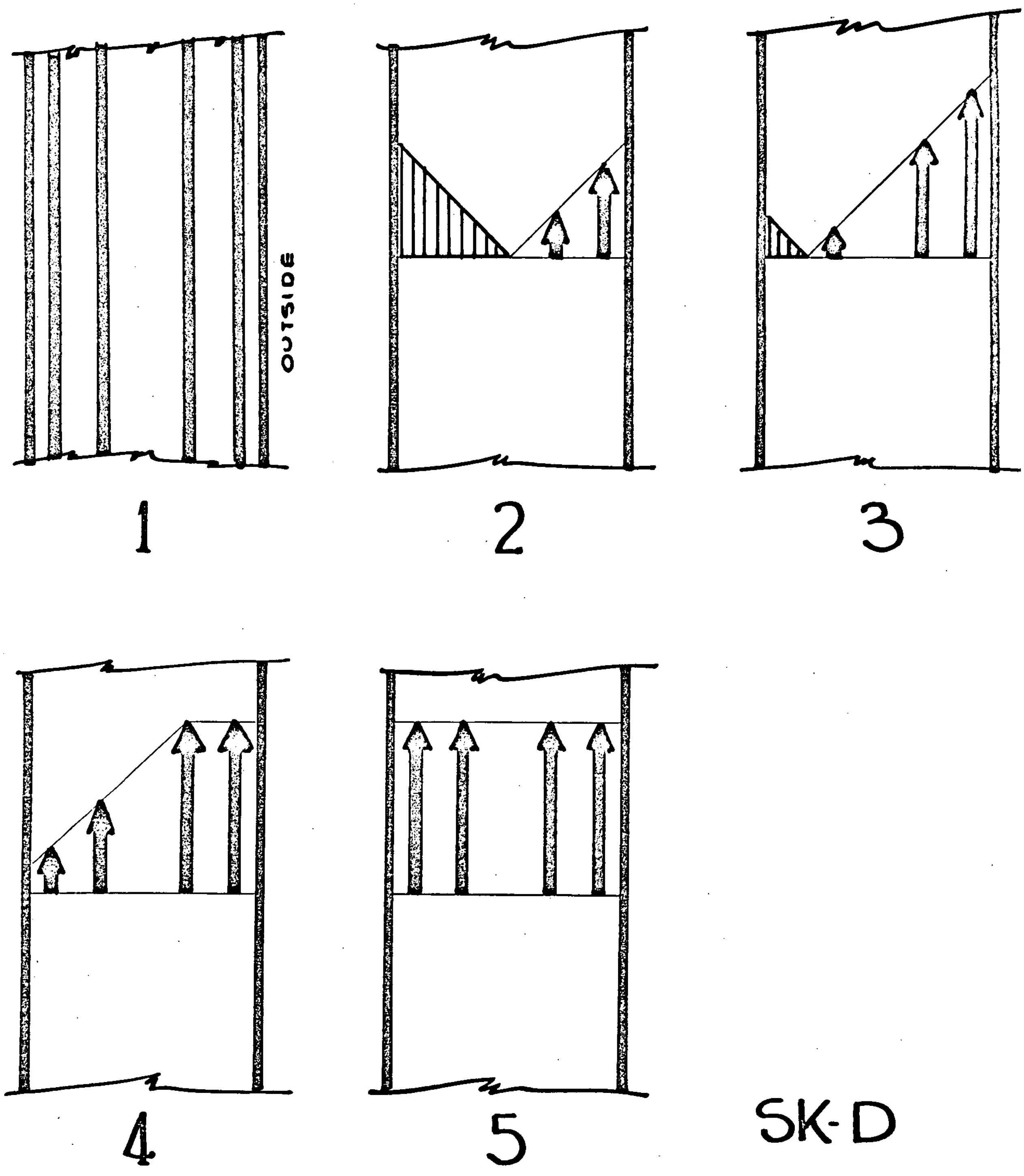
SK-B

PRESSURE DEFORMATION TENDENCY

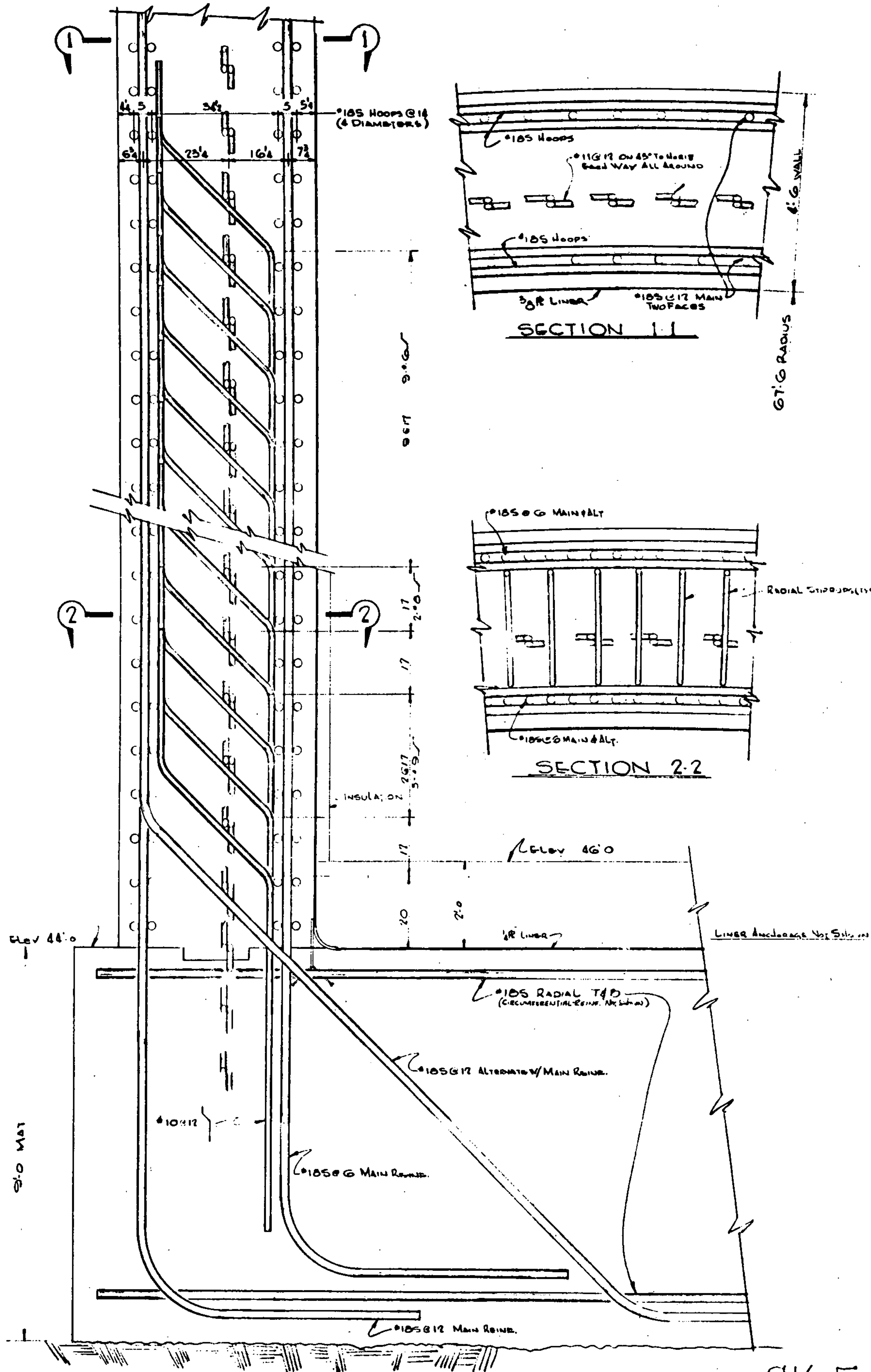


SK-C

STRESS REDISTRIBUTION



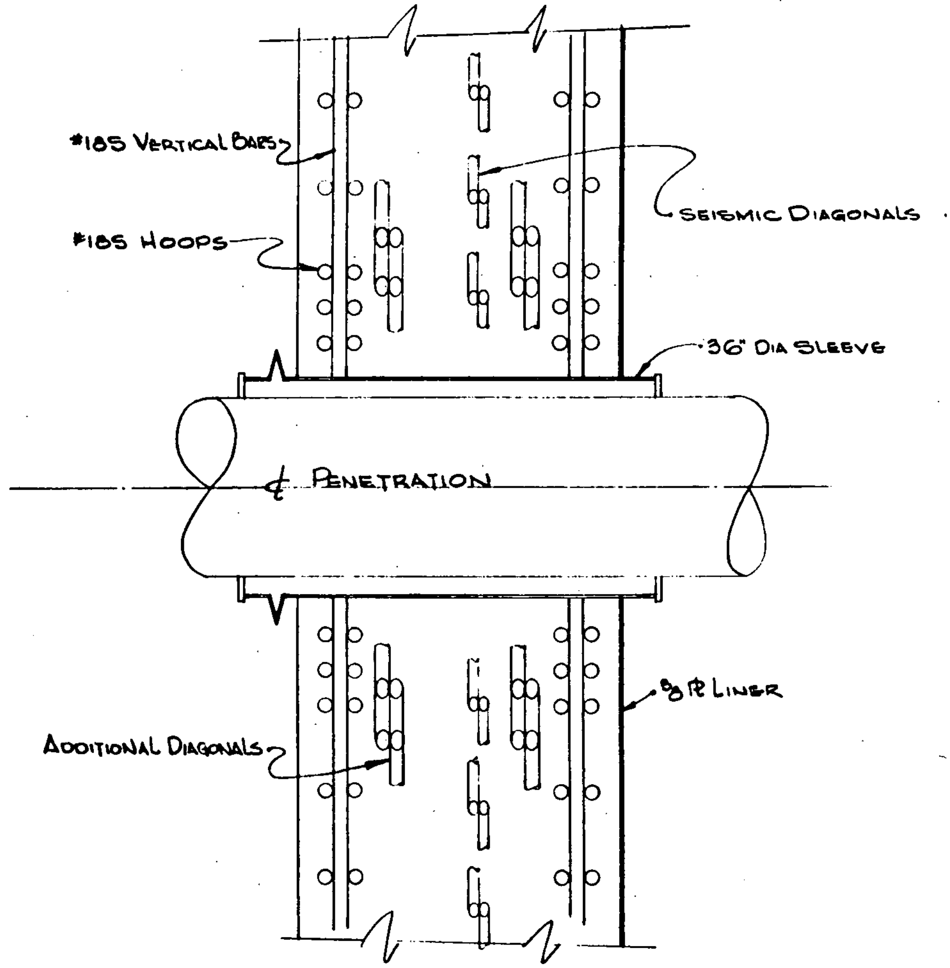
SK-D



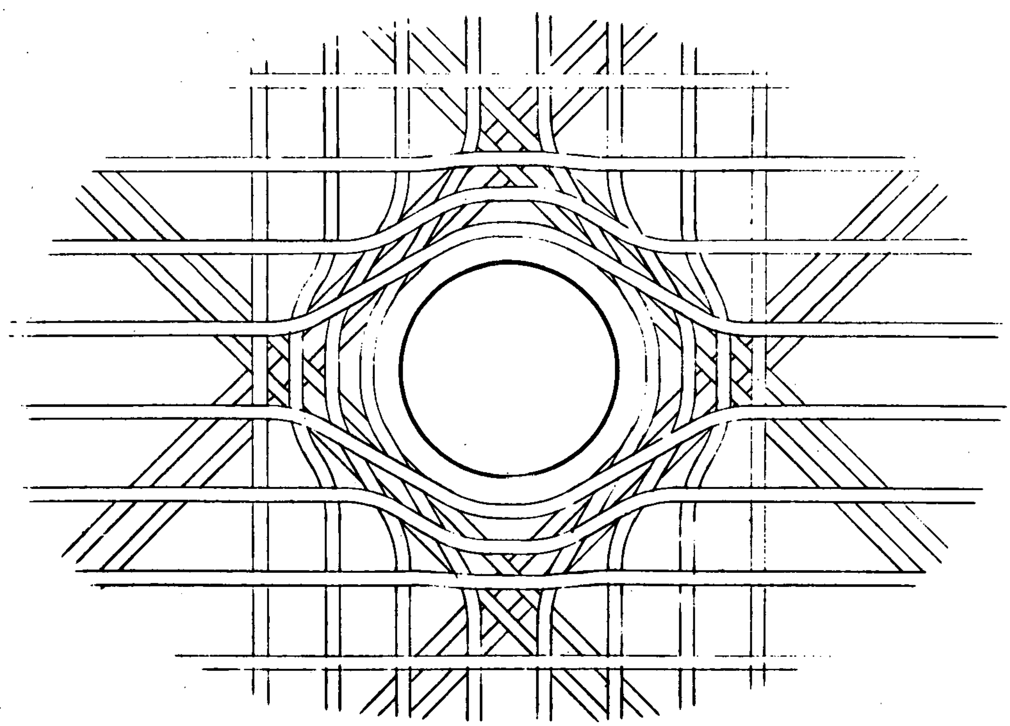
NOMINAL SHELL REINFORCING

SCALE 1"=1'-0

SK-F

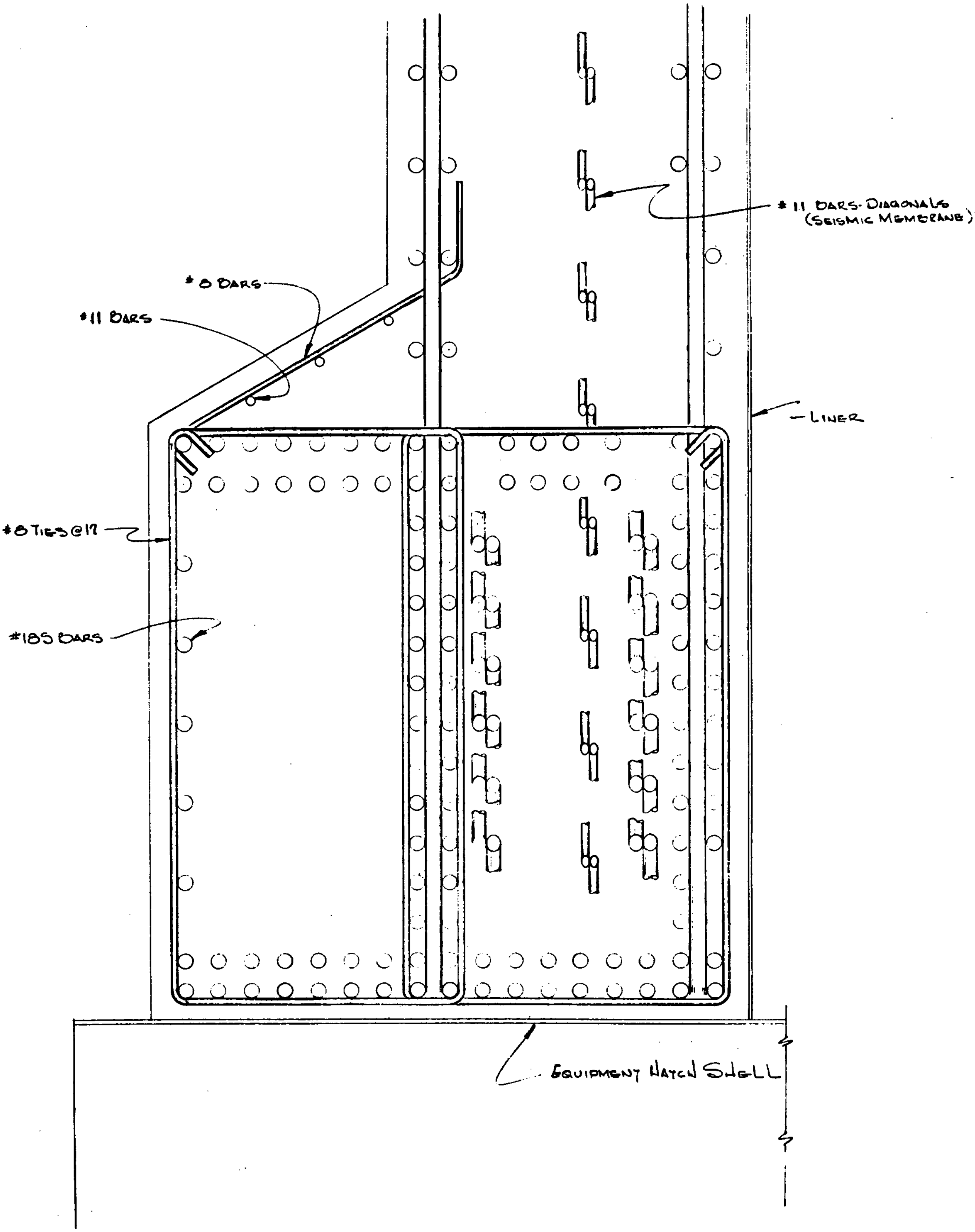


SECTION THROUGH WALL



PENETRATION ELEVATION

SK-G

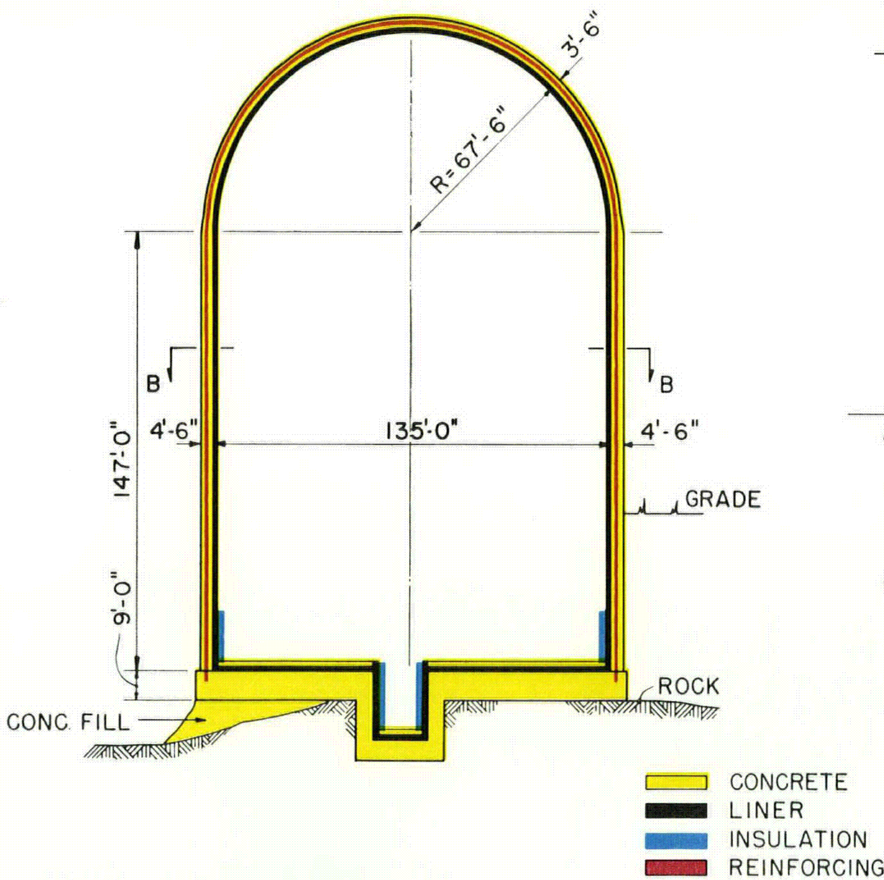


CONCEPTUAL REINFORCING DESIGN AROUND HATCH

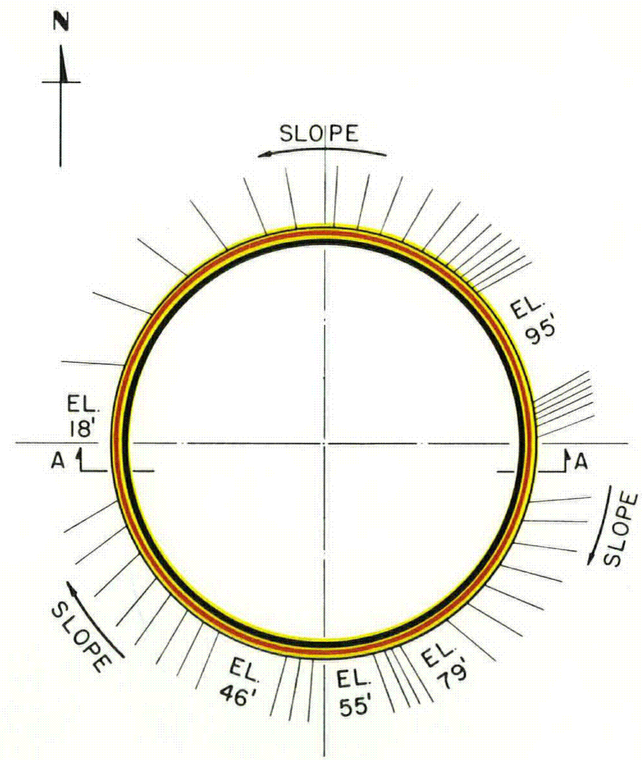
SCALE 1/2"=1'-0

SK-H

INDIAN POINT UNIT NO. 2, CONTAINMENT



SECTION A—A



SECTION B—B

FIGURE 1

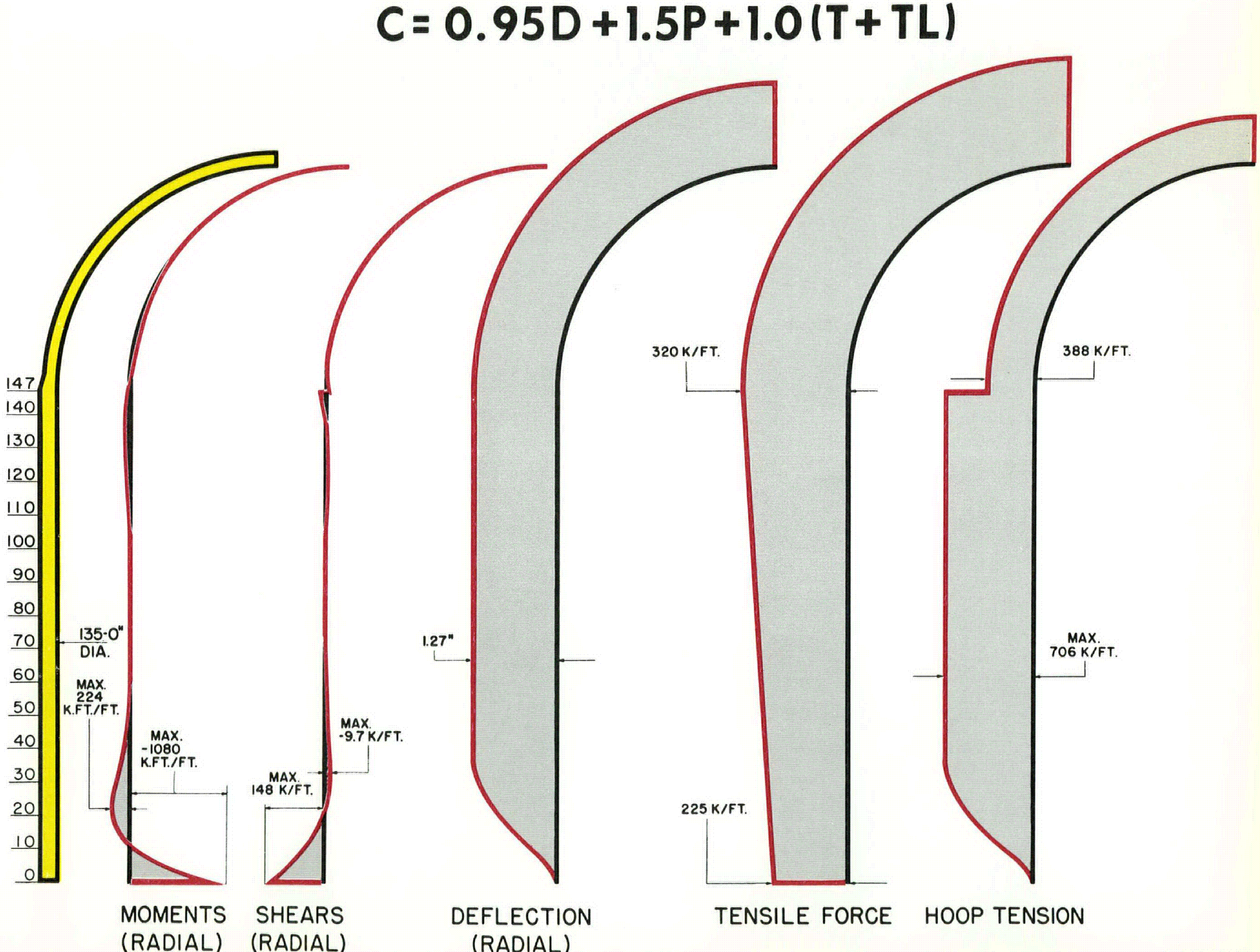

C= 0.95D +1.5P +1.0(T + TL)
147
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0
135-0" DIA.
MAX. 224 K.FT./FT.
MAX. -1080 K.FT./FT.
MAX. 148 K/FT.
MAX. -9.7 K/FT.
1.27"
320 K/FT.
225 K/FT.
388 K/FT.
MAX. 706 K/FT.
MOMENTS (RADIAL)
SHEARS (RADIAL)
DEFLECTION (RADIAL)
TENSILE FORCE
HOOP TENSION

FIGURE 2

$$C = 0.95D + 1.25P + 1.0(T' + TL') + 1.25E$$

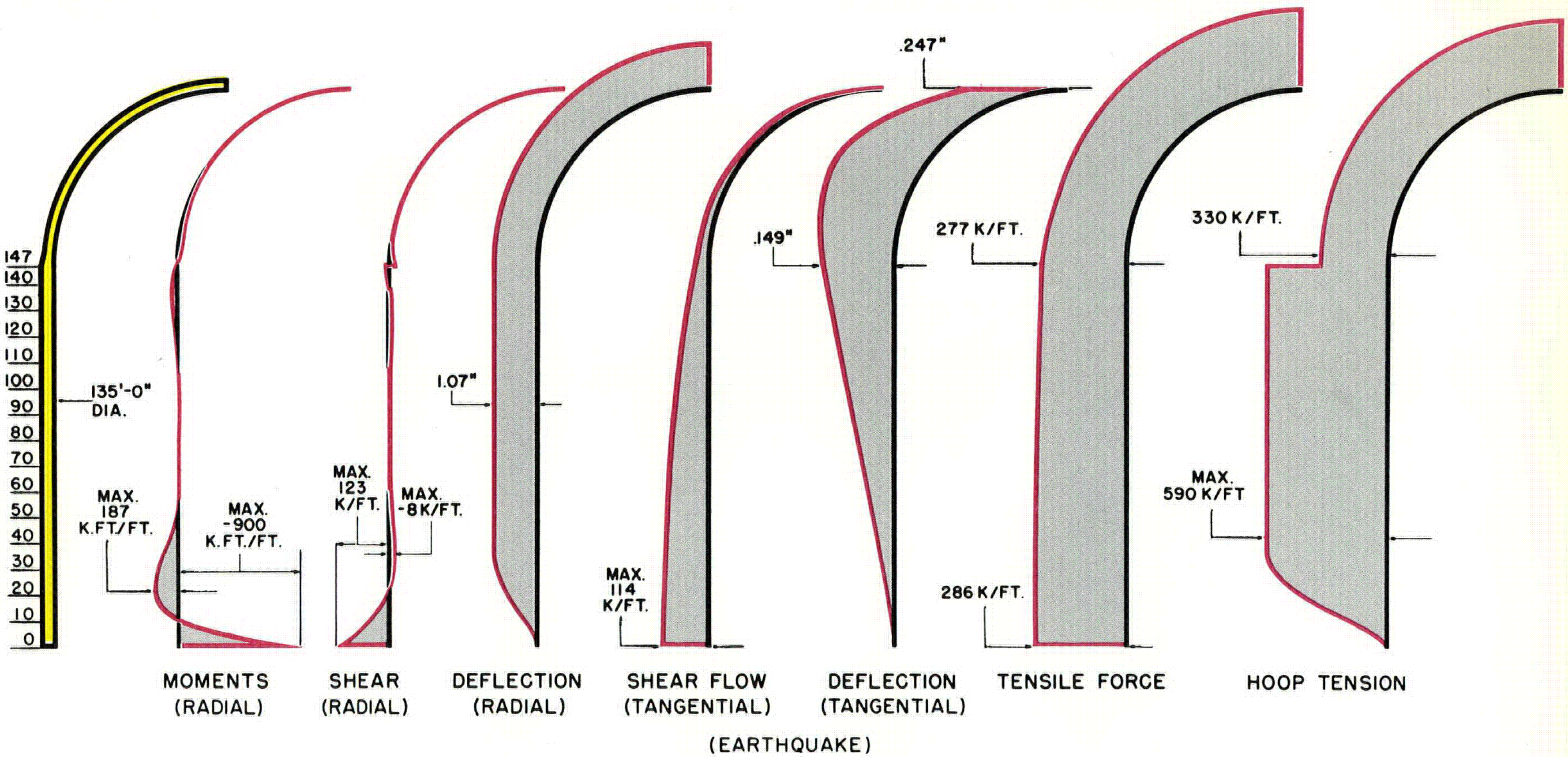


FIGURE 3

431 77 50T

1