POWER AUTHORITY OF THE STATE OF NEW YORK

10 COLUMBUS CIRCLE NEW YORK, N. Y. 10019

(212) 397-6200

August 28, 1981
IPN-81-66

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Seismic Qualification of Auxiliary
Feedwater System (Generic letter No. 81-14)

Dear Sir:

The Authority is submitting herewith Attachment I in response to the subject item.

The enclosed study concluded that the Auxiliary Feedwater System at Indian Point 3 is seismically qualified, with the exception of a few system components identified and analyzed.

Should you or your staff have further questions please contact us.

Very truly yours,

J. P. Bayne
Senior Vice President
Nuclear Generation

cc: attached

A001
S
1/1

cc: Mr. T. Rebelowski
Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Mr. Ron Barton
United Engineers & Constructors, Inc.
30 S. 17th Street
Philadelphia, Pa. 19101

DOCUMENT/ PAGE PULLED

ANO. 8109010366

NO. OF PAGES 4

REASON

☐ PAGE ILLEGIBLE

☐ HARD COPY FILED AT: PDR CF

OTHER ______

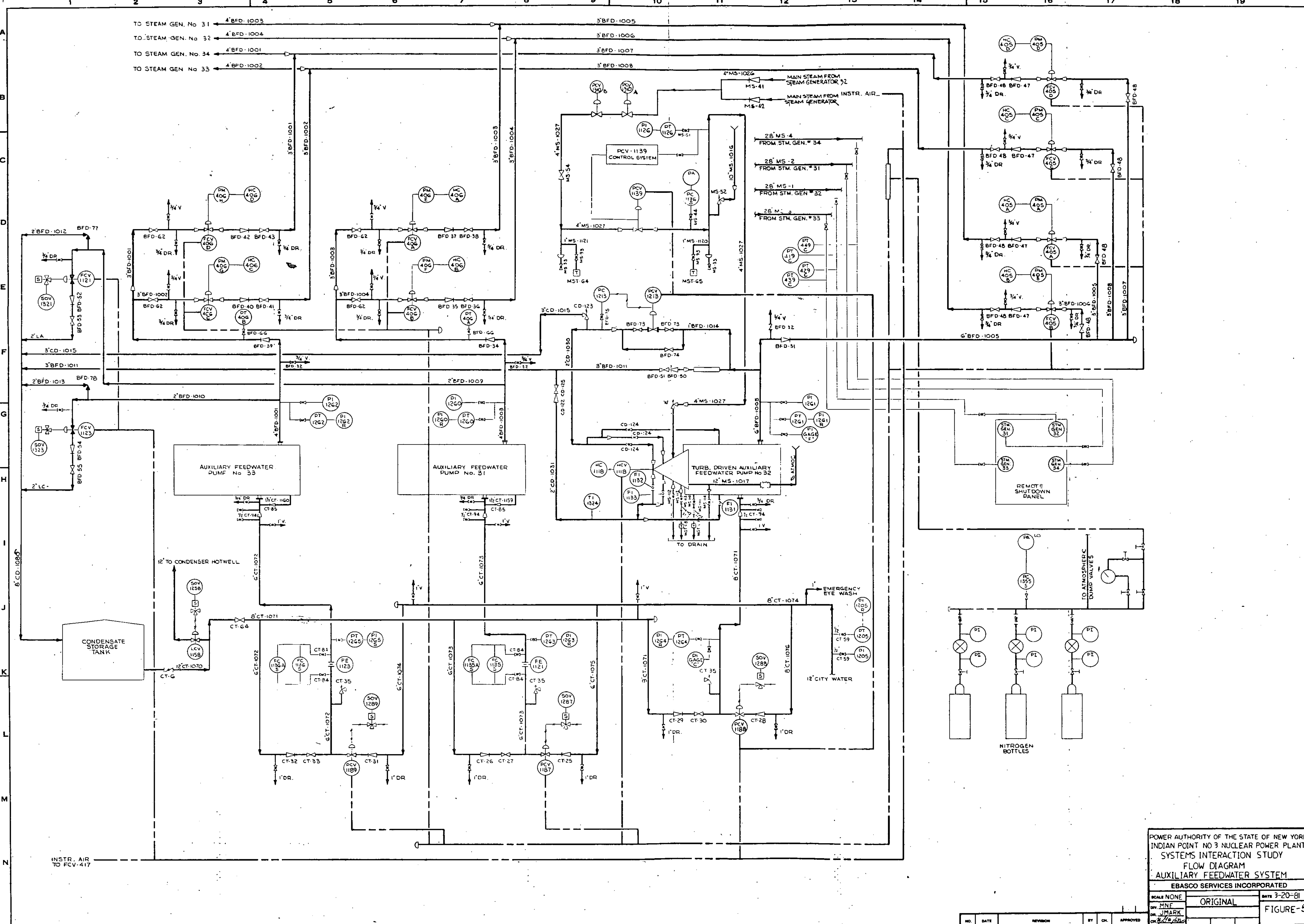
☐ BETTER COPY REQUESTED ON __/__/__

☒ PAGE TOO LARGE TO FILM.

☒ HARD COPY FILED AT: PDR CF

OTHER ______

☐ FILMED ON APERTURE CARD NO 01-04

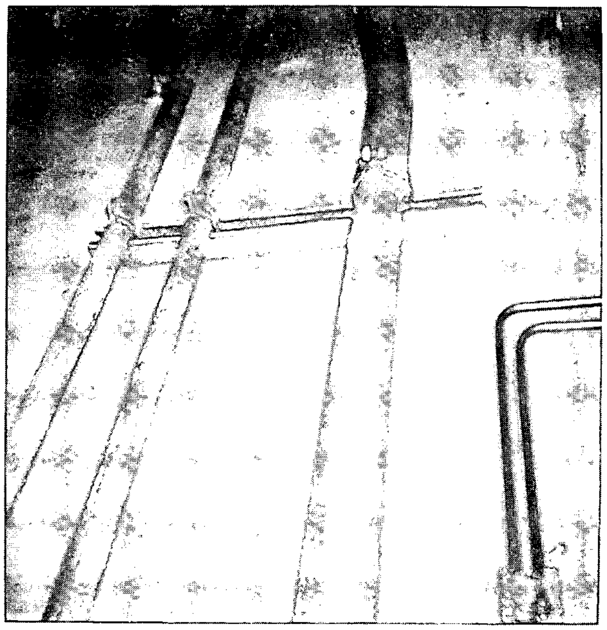


POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT NO 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
FLOW DIAGRAM
AUXILIARY FEEDWATER SYSTEM
EBASCO SERVICES INCORPORATED

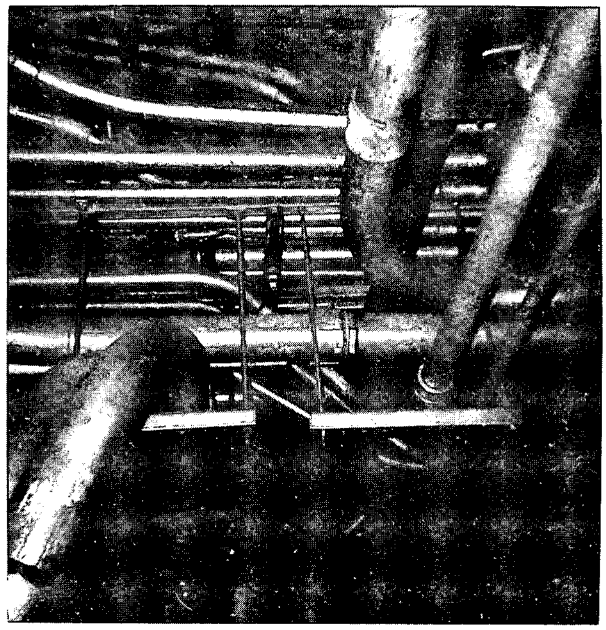
SCALE NONE | ORIGINAL | DATE 3-20-81
DFT. MNF
DR. JMARK
FIGURE-5

APPENDIX B

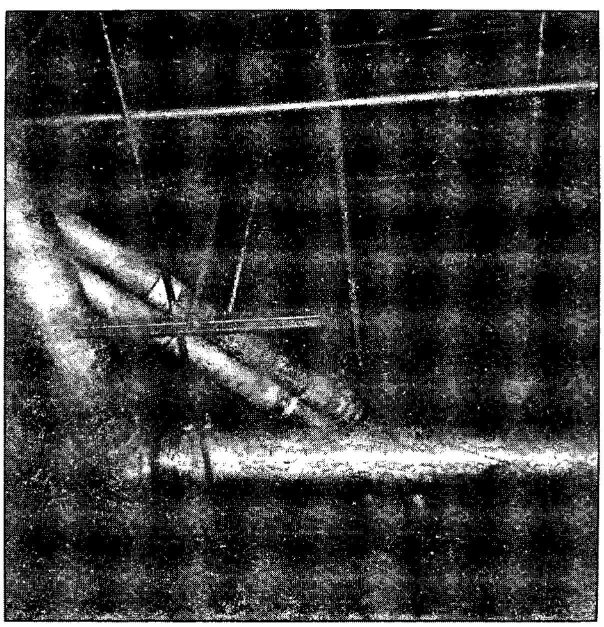
PHOTOGRAPHS



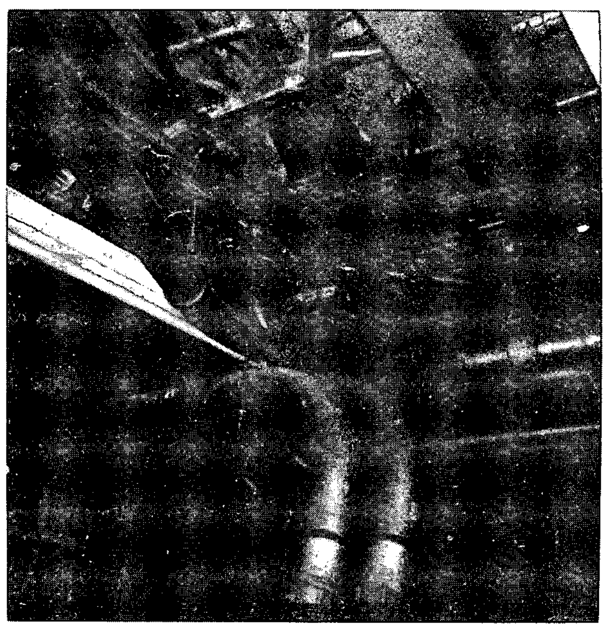
1



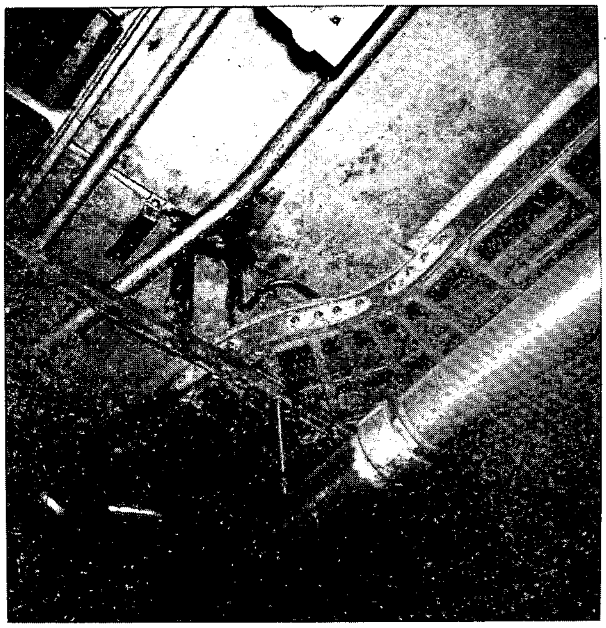
2



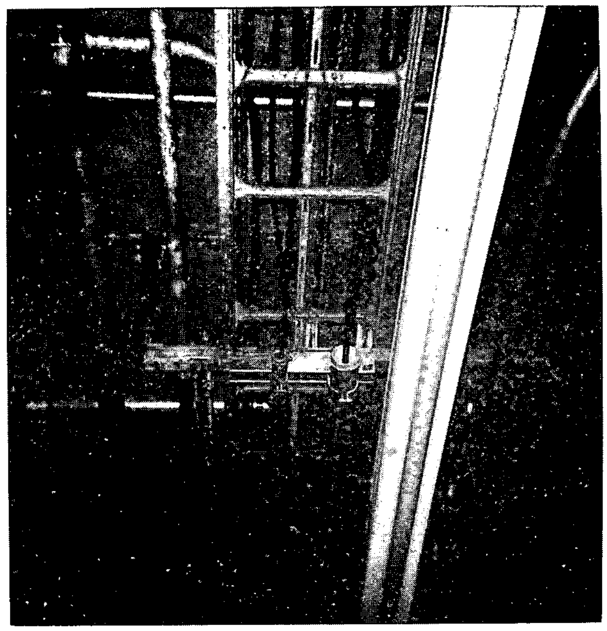
3



4

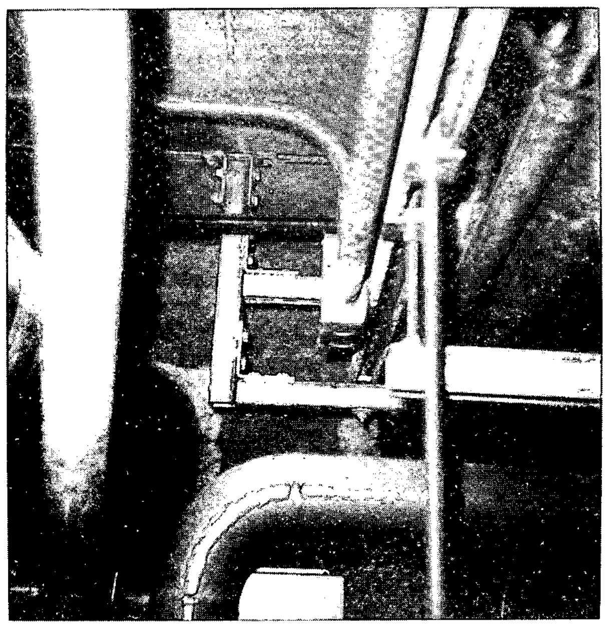


5

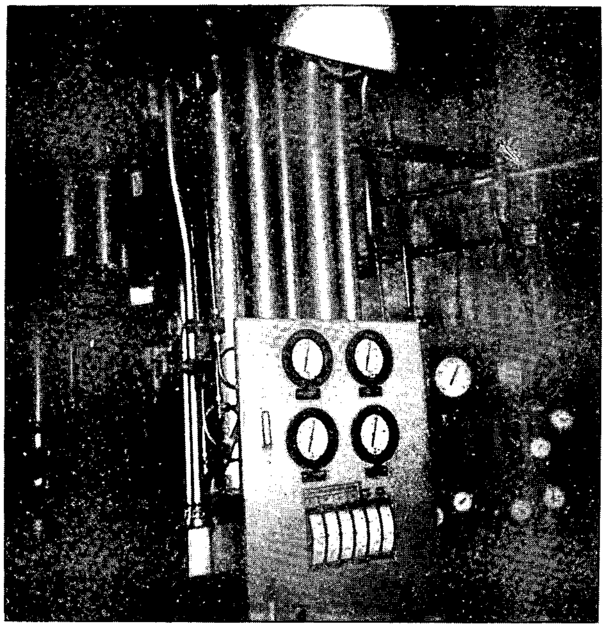


6

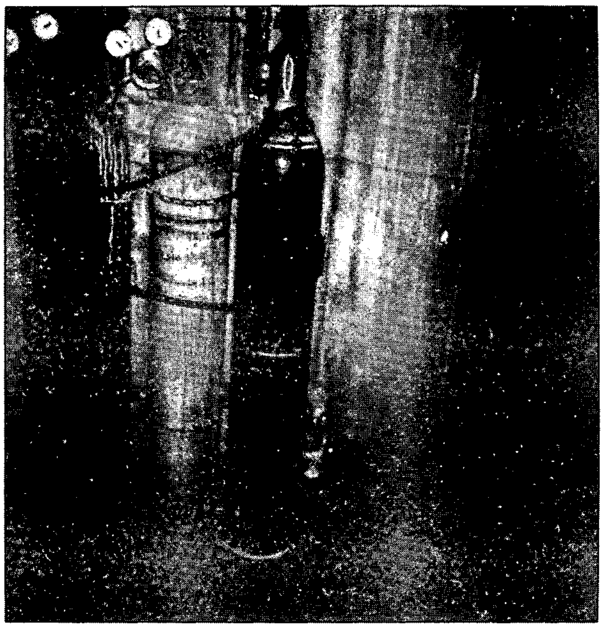
7



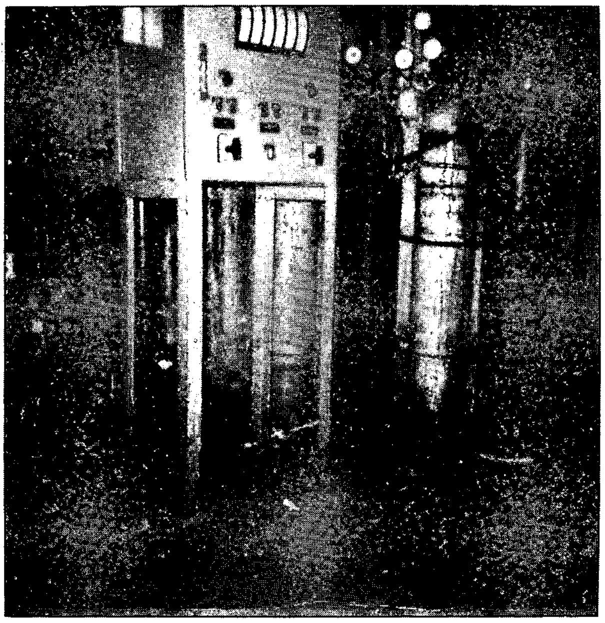
8



9



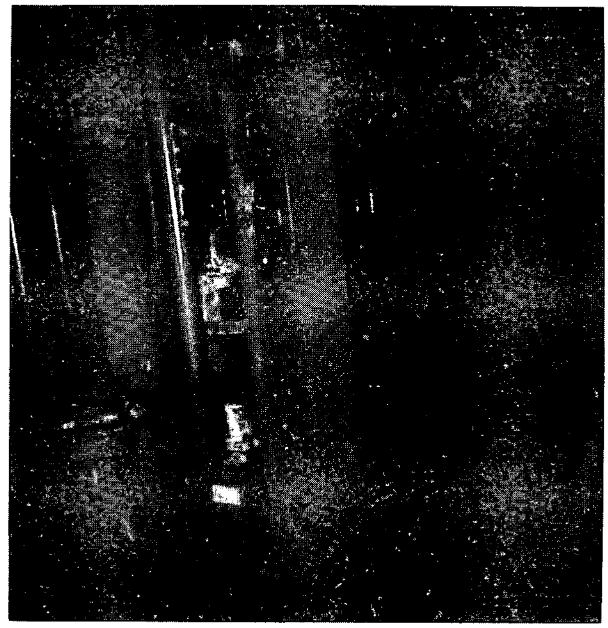
10



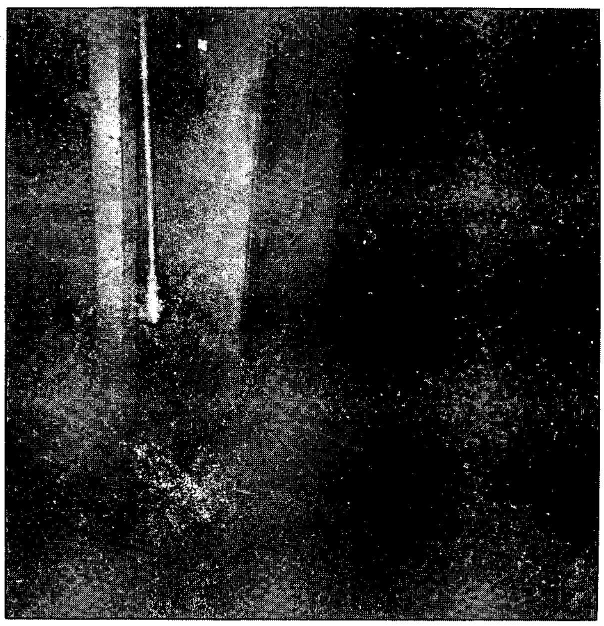
11

12

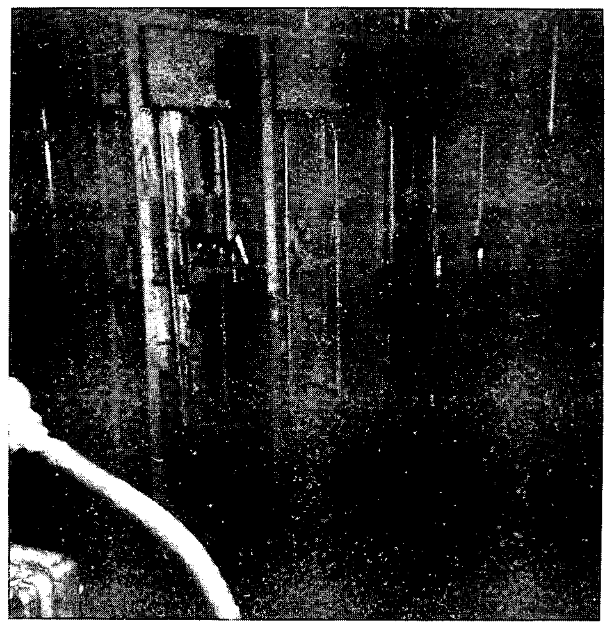
13



14



15



16

APPENDIX C

SEISMIC SPAN AND ACCELERATION TABLES

TABLE-1

SEISMIC SPAN TABLE

FOR PIPING

NOMINAL PIPE SIZE, INCH	WATER SERVICE, FT	STEAM, GAS, AIR SERVICE, FT
1/8	3.5	3.6
1/4	4.0	4.0
3/8	4.5	4.6
1/2	5.0	5.0
3/4	5.6	5.9
1	6.3	6.6
1-1/4	7.0	7.6
1-1/2	7.6	8.0
2	8.4	9.0
2-1/2	9.3	10.0
3	10.2	11.0
3-1/2	10.8	11.8
4	11.4	12.5
5	12.6	14.0
6	13.5	15.0
8	15.4	17.5
10	17.0	19.8
12	18.0	21.0

NOTE: SEISMIC SPAN WILL BE REDUCED TO 70% FOR STRAIGHT RUN WITH CHANGE OF DIRECTION AND 50% FOR CONCENTRATED WEIGHT.

TABLE-2

MAXIMUM SEISMIC ACCELERATIONS IN AFW PUMP BUILDING

ELEVATION	SAFE SHUTDOWN EARTHQUAKE (SSE) NORTH SOUTH	EAST WEST	VERTICAL
Upto 54'	0.5	0.6	0.4
66'	0.6	1.3	0.4
78'	0.7	2.3	0.4

NOTES:

(1) ABOVE VALUES ARE OBTAINED FROM SSE RESPONSE SPECTRA FOR SHIELD WALL FOR 2% DAMPING.

(2) THE OBE ACCELERATION VALUES ARE TWO-THIRDS OF SSE VALUES.

(3) AN AMPLIFICATION FACTOR 1.5 SHOULD BE USED FOR CALCULATING SEISMIC LOADS.

APPENDIX D

SEISMIC INTERACTIONS ANALYSIS REPORT

CHAPTER 6

6.0 EVALUATION CRITERIA

The evaluation of event induced systems interactions and their effects on plant safety rests heavily on experienced engineering judgement. Reliance is placed on assigned engineering and design personnel in various relevant disciplines applying their knowledge and experience in evaluating the problems.

6.1 INTERCONNECTED SYSTEMS

The evaluation of interconnected system interactions and their effects on plant safety will be based upon satisfying the failure criterion presented in Section 5.3.3 using the techniques of failure mode and effects analysis. As described in Section 5.3.3, postulated system interactions induced by random failures of safety related components will be considered acceptable if it does not compromise the functional capability of the system to perform it's intended safety function.

6.2 NONCONNECTED SYSTEMS

6.2.1 Evaluation of Sources

Potential sources are evaluated as part of the program to determine if events can credibly lead to detrimental interaction with targets.

a. Events will not lead to interaction because of defensible qualification of the sources by analysis, test, or experience with the same or similar items.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.1 Evaluation of Sources (Cont'd)

b. Events may lead to damage or failure of the sources, but the credible failure modes are no threat to the safety function of the target.

c. Events may lead to a credible failure mode of the source which has the potential to cause adverse interaction.

6.2.1.1 The following criteria provide minimum guidance for evaluation of sources for seismically induced events:

a) Structural Source Evaluation

All structural sources are evaluated by the single failure criterion:

Any non safety related structural element determined to be a potential source will be assumed to fail, unless seismic qualification by analysis, test or comparison to similar previously qualified elements has been performed to ensure integrity.

b) Mechanical Source Evaluation

The following is a set of failure modes for mechanical equipment which must be considered when evaluating potential sources in these categories.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1. (Cont'd)

b) Mechanical Source Evaluation (Cont'd)

In addition to the specific failures below, complete loss of power for all source equipment and control power has been postulated. Relative motion between the source and target are considered during the walkdown examination.

- Overturning of tanks, pumps, filters or other unsupported equipment where the center of gravity location as measured from the base is longer than one-half the base width in all directions. Each direction will be evaluated independently. A horizontal acceleration equivalent to at least that value associated with the plant SSE, would be required to overturn an unsupported component whose height is less than 1/2 base width from the base. Overturning is not considered where the distance from the base to the center of gravity is small. Further conservatism is obtained because mechanical equipment is held down by bolting, brackets, etc. However, if any component structure or system experiences a horizontal acceleration of greater than the SSE, it will be evaluated on a case by case basis.

- All non-seismically qualified valves, pumps, tanks and vessels are assumed to fail in the "worst credible mode" possible. (E.g., partial failure of valves and operation of pumps below design flow rate have to be considered).

The "worst credible mode" will be based on sound engineering judgement.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS(Cont'd)

6.2.1.1 (Cont'd)

c) Electrical Source Evaluation

Several categories of failure type must be considered with regard to seismic effects on electrical sources (equipment and cabling). They are discussed below:

c.1 Electrical Equipment

c.1.1 Overturning of cabinets, transformers, switchgear or other unsupported equipment where the center of gravity location as measured from the base is longer than one-half the base width in all directions. Each direction will be evaluated independently.

The same considerations discussed in regard to overturning of mechanical equipment apply to electrical equipment, i.e., overturning is assumed only for cases where the distance to the center of gravity is significant compared to the base width.

c.1.2 All nonseismically qualified electrical equipment (except cable trays will be assumed to fail in the worst credible mode possible. The "worst mode failure" will be based on sound engineering judgement.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1 Evaluation of Sources (Cont'd)

6.2.1.1 (Cont'd)

c) Electrical Source Evaluation (Cont'd)

c.1.3 All nonseismically supported electrical equipment (except raceways) will be assumed to be a source of the "worst possible" physical and electrical interaction.

c.2 Cable Trays

c.2.1 Seismically Supported Cable Trays

Cable trays that are determined to be seismically supported/restrained are assumed to remain physically intact in the event of an SSE (i.e., they do not become a source) and also that they will develop no electrical faults as built.

c.2.2 Non-Seismically Supported Cable Trays

A non-seismic cable tray in the vicinity of essential safety related equipment is to be a potential source and assumed to collapse. Also cables contained within the tray are assumed to develop electrical faults. The "vicinity" is defined by the criteria assumed and illustrated in Figure 6-1 & 6-2.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

c) Electrical Source Evaluation (Cont'd)

c.3 Conduits

Non-seismically supported/restrained conduits are assumed to be the source of mechanical and electrical interactions in an SSE.

d) HVAC Source Evaluation

d.1 Non-seismically supported ductwork that run directly over essential safety related targets will be considered a source of potential interaction. The interaction boundary envelope is illustrated in Figure 6-3.

d.2 While considering systems interaction of HVAC systems, the effects of ductwork crimping, adverse operation (or non-operation) of non-safety related fans that might spread combustible or toxic fumes through the ductwork has to be considered.

d.3 Failure of in-line HVAC equipment will follow the source evaluation criteria for Mechanical equipment. Support failure resulting in tipping, falling, sliding or overturning may occur. Overturning will be assumed possible when the distance as measured from the base to the center of gravity is more than one-half the width of the base. Each direction will be evaluated independently.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

e) Piping System Source Evaluation

High energy pipe rupture, jet impingement, flooding and internal missile analyses are not included in this seismically induced interaction assessment except in the cases where these effects are seismically induced.

All piping and asscoiated components identified as an essential safety related component fall under the category of targets. Also they are assumed to be seismically supported or restrained and hence will not become seismically induced souces.

Non-seismically designed piping will be considered as a source in the following context:

Physical Impact: All non-seismically designed/supported piping running in the vicinity of targets could fall or physically impact the target within the pipe's volume of influence. The volume of influence is defined as five (5) pipe diameters or five (5) feet whichever is greater, laterally from the pipe center line. The pipe is assumed to fail anywhere along the piping run, during or post SSE. This criteria is illustrated in Figure 6-4.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

e) Piping System Source Evaluation (Cont'd)

Flooding: A non-seismic piping run in the vicinity of target equipment will be assumed to have a circumferential or longitudinal rupture during or post SSE that could flood the room (attention must be paid to the instrumentation cabinets, motors, etc. in the room), flood any cable tray runs immediately above or below the piping run.

Environmental: Piping failures or a resulting chain interaction could cause unacceptable environmental conditions enveloping a target equipment, (e.g., auxiliary steam line failures could result in a steam environment with elevated temperatures and high humidity). Specific targets could either cease functioning or malfunction in this environment.

f) Instrumentation and Control, Source Evaluation

All instrumentation that is not seismically qualified will be assumed to malfunction in the "worst credible mode". Instrumentation that is not seismically mounted will be assumed to fail structurally and could becomes missile. The "worst credible mode" will be based on engineering judgement.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.2 The following criteria provide minimum guidance for evaluation of sources for pipe failure induced events

The criteria provided by the NRC Standard Review Plans 3.6.1 and 3.6.2 with companion Branch Technical Positions BTP APCSB 3-1 and MEB 3-1 were used to evaluate systems interactions associated with pipe failure induced events Table 6-1 summarizes the acceptance criteria for external and internal challenging events relative to the system, component or structure being evaluated.

6.2.1.3 The following criteria provide minimum guidance for evaluation of sources for missile (internally and externally) generated induced events.

The criteria provided by the NRC Standard Review Plans 3.5.1, 3.5.2 and 3.5.3 were used to evaluate systems interactions associated with the effects of internally and externally generated missile systems interactions. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system component or structure being evaluated.

6.2.1.4 The following criteria provide minimum guidance for evaluation of sources associated with flooding induced events.

The criteria provided by the NRC Standard Reveiw Plans 3.4.1 and 3.4.2 were used to evaluate adverse systems interactions associated with the effects of flooding. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system, component or structure being evaluated.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.5 The following criteria provided minimum guidance for evaluation of sources resulting from the effects of fire induced events.

The criteria provided by the NRC Standard Review Plan 9.5.1 with companion Branch Technical Position BTP APCSB 9.5-1 were used to evaluate adverse systems interactions associated with the effects of fire. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system component or structure being evlauated.

6.2.1.6 The following criteria provide minimum guidance for evaluation of sources resulting from the effects of severe environment.

The criteria provided by the NRC Standard Review Plans 3.3, 3.4, 3.5, 3.6 and 3.11 were used to evaluate systems interactions resulting from severe environmental conditions. In addition the guidance provided by IE Bulleting 79-OIB was used to the degree practicable for this evaluation. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system, components or structure being evaluated.

6.2.2 Modification Criteria

Modifications may be required to resolve identified event induced adverse systems interactions. These modifications may be any of the following:

a. Modification of the source to eliminate the adverse behavior by bracing, supporting, or reinforcing the source component.

b. Shielding or relocation of the target to preclude the physical interaction.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2 Modification Criteria (Cont'd)

c. Modification of the target to permit retention of the required safety function in spite of the interaction.

d. Alteration of system design to provide alternate means of accomplishing the safety function.

The criteria for structural or mechanical modifications are the same as documented for safety related structures and equipment.

For relocation or modification of non-safety related equipment, the criterion for acceptability is that the modified configuration, when re-evaluated for interactions using the evaluation criteria previously stated, is found to have resolved the original interaction and not created any new interactions.

6.2.2.1 Interaction Effects Evaluation Criteria

Once an interaction is identified as sufficiently credible to require more evaluation than can be done from inspection, it must be resolved in an acceptable manner and the resolution documented. Interactions considered are direct physical interactions such as target impact from a falling or moving source. Typical interactions are listed below.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Mechanical:

- impact from vibrating bodies
- impact from falling bodies
- pipe whip
- missiles

Electrical:

- unwanted open circuit (loss of control power)
- unwanted closed circuit
- unwanted energization

Pneumatic:

- loss of pressure (loss of control)
- unwanted pressurization
- jet impingement
- hostile gas

Hydraulic:

- loss of pressure
    - (a) loss of control
    - (b) loss of lubrication
- unwanted pressurization
- jet impingement
- flooding
- hostile fluids

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Environmental:

- elevated temperatures
- humidity
- radiation

Interactions are evaluated for their impact on the required safety functions and redundancy of identified targets. The results of the evaluation will then determine the method of resolution. In order of preference, the following are categories of acceptable methods of resolution of identified interactions.

a. Target Operability Evaluation:

The first approach to resolution is to show that the target's safety function is not impaired. This may be accomplished by studying the means by which impairment occurs and the possible extent of the impairment. For example, a pneumatically operated valve may be required to close during shutdown, but falling equipment could sever the air line so air supply to the operator is lost. If the valve is a "fail open" type, then shutdown capability is compromised, but if the valve is a "failed closed" type, then shutdown capability is not compromised even though the air supply is lost. In this example it is also necessary to consider consequences of crimping the air line, as well as the effect of a lost air line.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

a. Target Operability Evaluation: (Cont'd)

This example is typical of the reasoning process that is necessary in the evaluation of each interaction. A substantial degree of engineering judgement is, of necessity, expected to be used. Decisions based on judgement, along with the rationale, are documented.

b. Source Behavior Evaluation:

The second approach to resolution is to perform a more careful evaluation of the source behavior resulting from an event. If tests, analysis, or applicable experience can be developed to demonstrate that the item in question is qualified to withstand the postulated event, the interaction can be declared resolved on the basis that it will not credibly occur. Identification and resolution of indirect or chain-reaction source events shall use individual source failure criteria for each component source.

c. Modification:

If resolution is not possible by analysis or by test, the Interaction Team will recommend that physical modifications be made to prevent detrimental interaction. The range of possible modifications includes guard structures, protective covers, and restraining structures. The criterion is to prevent impairment of function.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

d. Change of Procedures:

The last method of resolution is by reordering the operating procedures or defining alternate means of providing the required safety functions. The Interaction Team will not specify procedural changes to resolve an adverser systems interaction, other than to present generic options.

The evaluation and resolution methods are discussed below in more detail.

Evaluation of Direct Interaction Effects

Where evaluation is directed to showing that the safety function of a target is not impaired by an identified direct interaction, the following guidance has been established. For cases not covered, criteria are developed and documented to provide an analagous level of rigor to the guidance herein provided.

a. Dynamic effects of breaks in piping are evaluated using the criteria in Section 6.2.1.2. For example one criterion to be used is that no damage will result if the target pipe size is at least equal to the size of the source pipe and the wall thickness of the target pipe is at least equal to that of the source pipe.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Evaluation of Direct Interaction Effects (Cont'd)

b. Direct impact of missiles or falling objects on structures and components are evaluated when necessary using the criteria of Sections 6.2.1.3. Care must be taken to consider such appurtenances as instruments, power connections, cooling and lubrication connections.

c. Direct impact of missiles or falling objects on HVAC ducts have to be evaluated on a case by case basis.

d. Flooding effects of broken or leaking pipes are evaluated using the criteria of Section 6.2.1.4.

e. The effects of fire are evaluated using the criteria of Section 6.2.1.5.

f. Environmental effects of broken or leaking piping, tanks, etc. are evaluated by comparison of the estimated environment with the target's qualification profile. Helpful criteria and data are contained in Section 6.2.1.6.

Evaluation of Secondary Effects or Cascading Influences

Two types of secondary effects on cascading influences are considered; chain-reaction failures and degraded operation.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Evaluation of Secondary Effects or Cascading Influences

For the chain-reaction events, the criteria for evaluation are the same as for the direct interactions and are successively applied to each member of the chain. It must be remembered that each step in chain scenarios has an associated probability less than one and that judgement must be applied to consider only credible scenarios.

In order for the plant to safely shut down, it is necessary for the required safe shutdown valves and drive elements to operate in the required manner, or fail in the required position. For this to occur. their control systems must remain intact after the interaction event, or else be damaged only in such a way to fail in the design failure mode. For example, if an air operated valve is required to fail in a certain mode, the design is such it will go to that failure mode on loss of air. If, however, the air line between the control device and the valve were to be impacted during a seismic event, the line might be pinched. This could prevent the venting of air and thereby prevent the valve from failing in its proper mode.

In electrically operated devices, a non-qualified component could impact the signal cable and cause damage which would adversely affect proper operation.

6.0 EVALUATION CRITERIA (Cont'd)

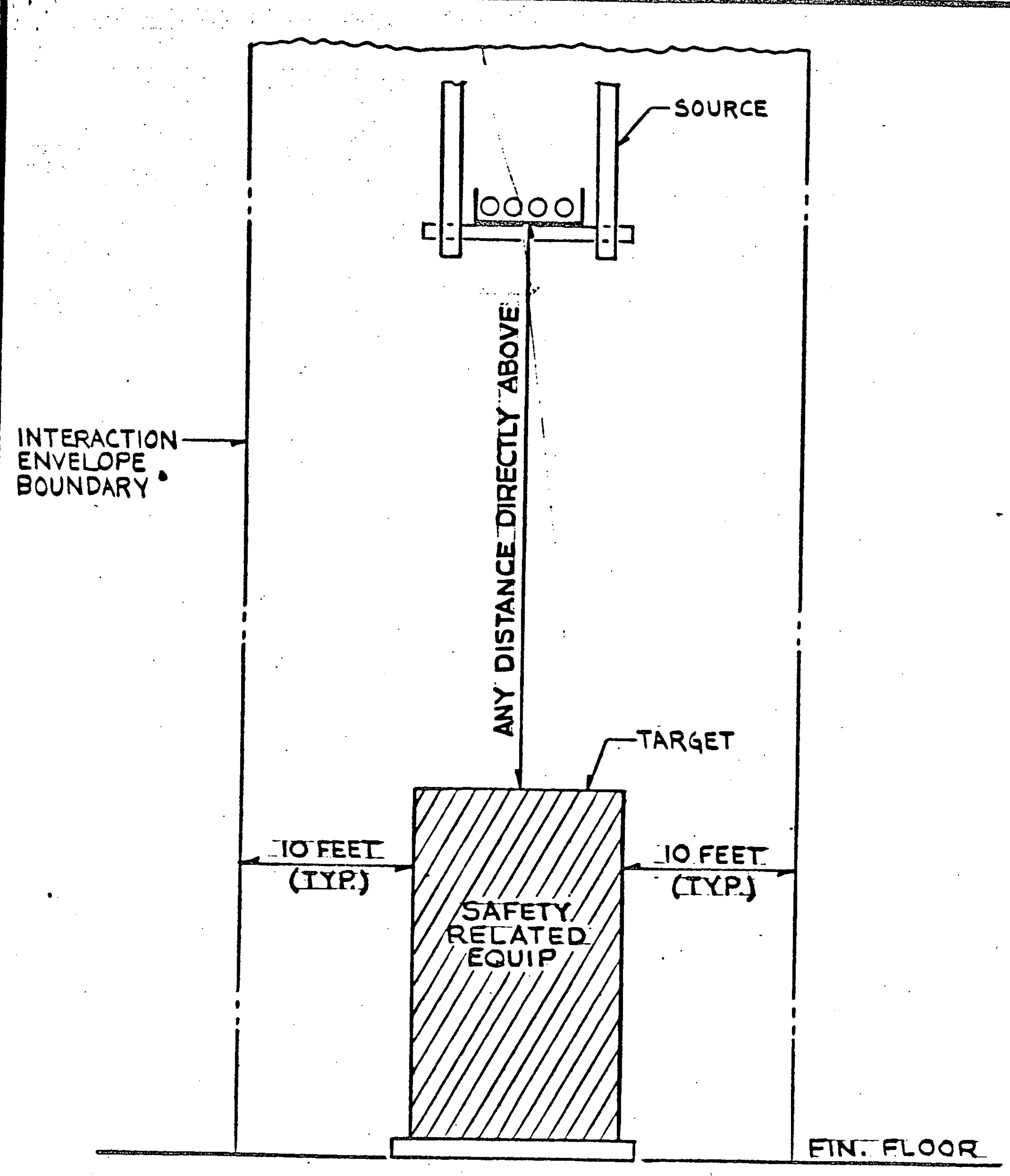
6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Evaluation of Secondary Effects or Cascading Influences (Cont'd)

The walkdown will identify process tubing, instrumentation, electrical cables and cable trays requiring protection from unacceptable interactions.

When questionable secondary interactions are identified which are not readily evaluated to be acceptable, the resolution then becomes one of modification such as redesign or replacement of the source equipment or the rerouting or upgrading of control and electrical wiring and/or process and air tubing.

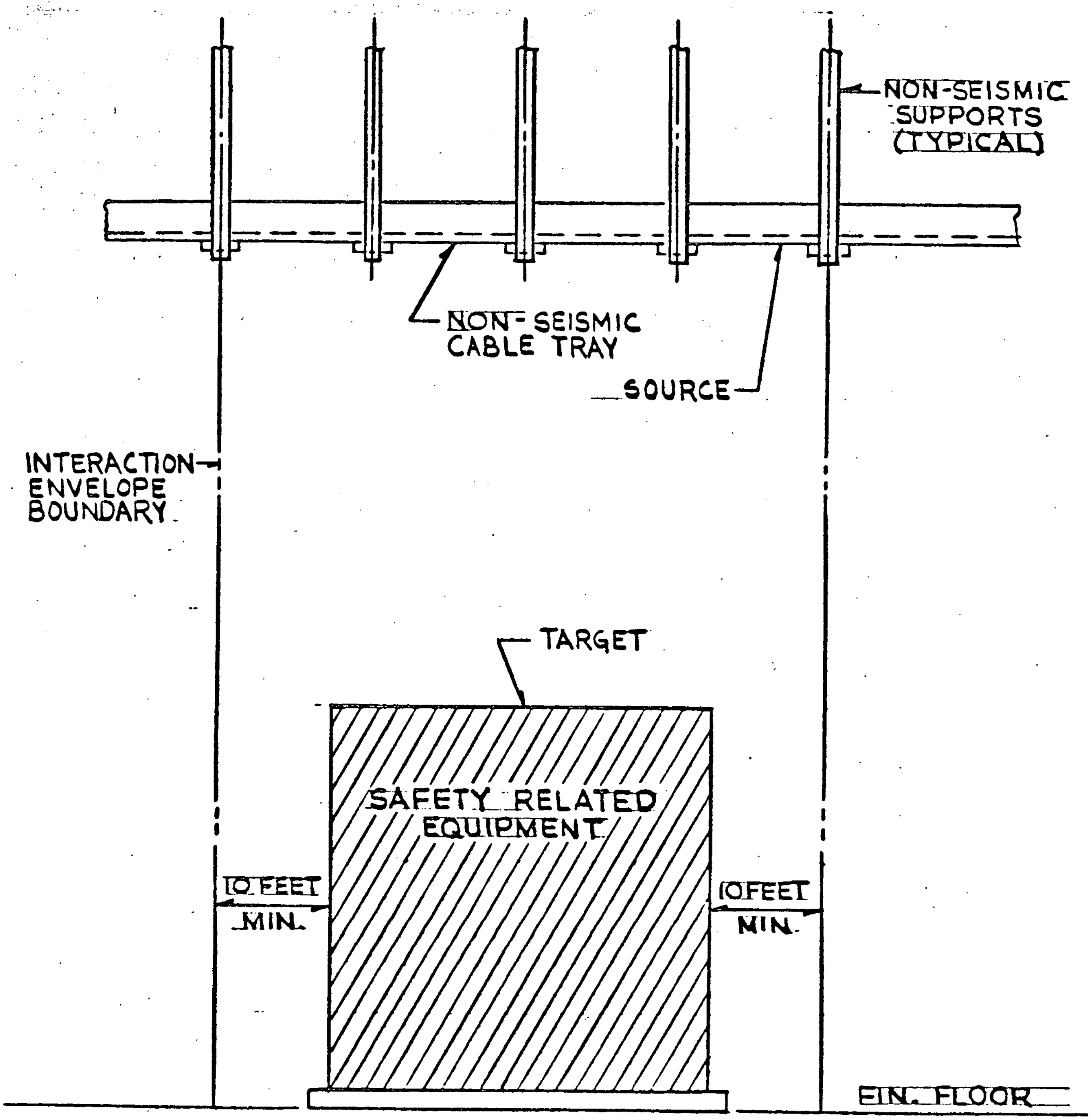


EBASCO SERVICES INCORPORATED

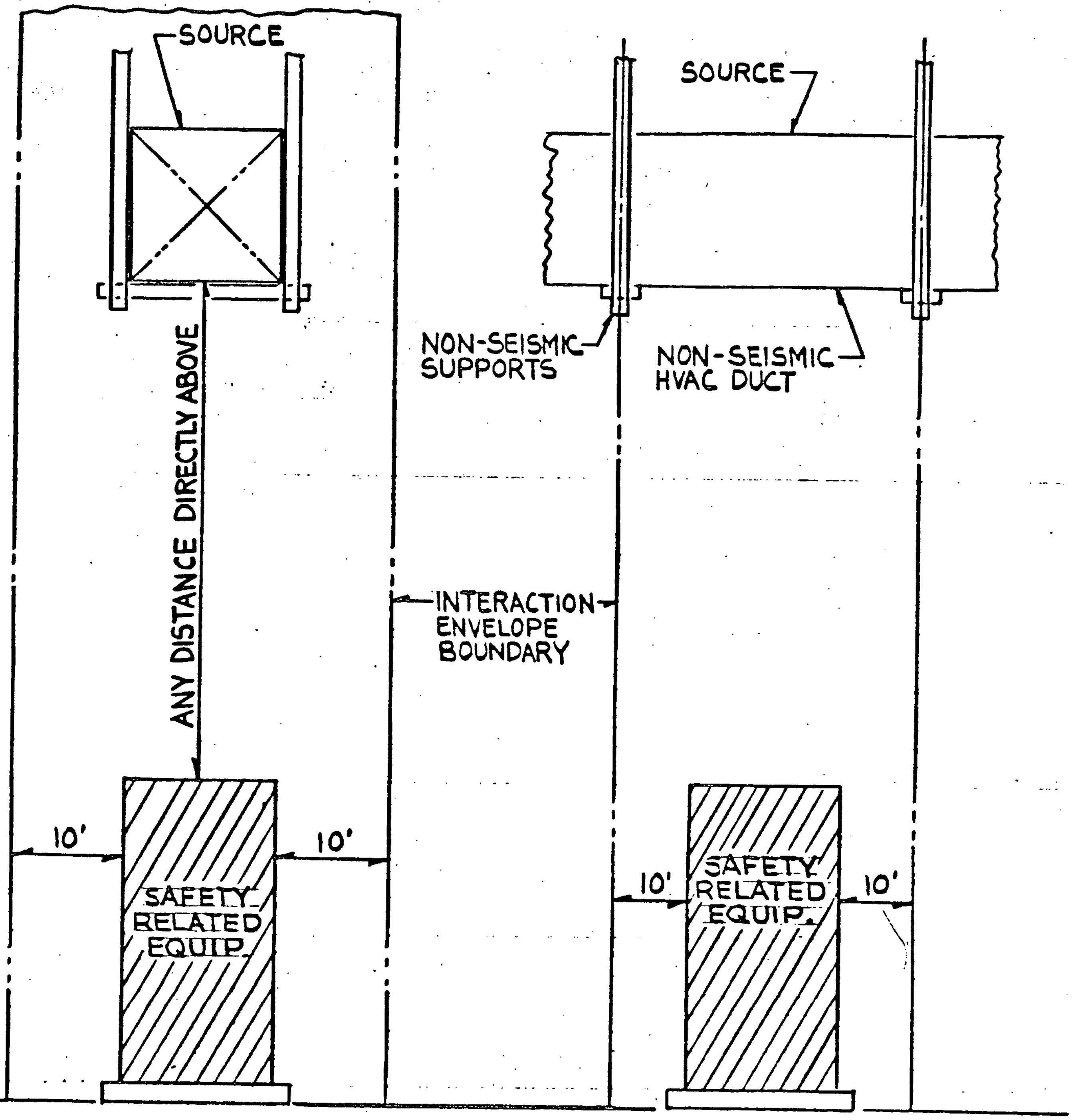
DIV. MNE DR. JM
DATE 3-20-81 CH.
SCALE None

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT No. 3
NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
INTERACTION CRITERIA
ELECTRICAL CABLE TRAYS

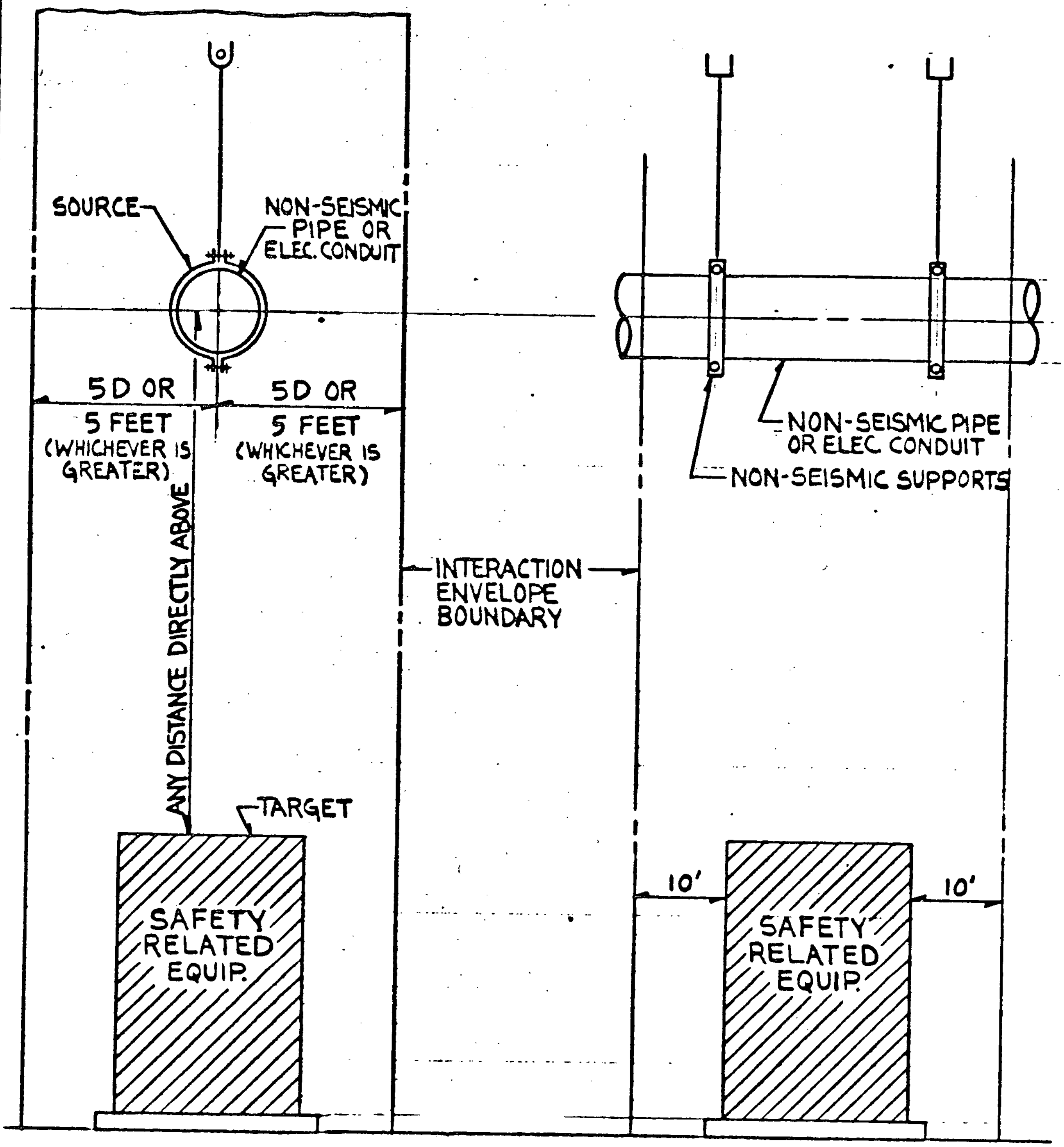
FIGURE 6-1



EBASCO SERVICES INCORPORATED DIV. MNE DR. JM DATE 3-20-81 CH. SCALE NONE	POWER AUTHORITY OF THE STATE OF NEW YORK INDIAN POINT No. 3 NUCLEAR POWER PLANT SYSTEMS INTERACTION STUDY INTERACTION CRITERIA ELECTRICAL CABLE TRAYS	FIGURE 6-2



EBASCO SERVICES INCORPORATED		POWER AUTHORITY OF THE STATE OF NEW YORK INDIAN POINT No. 3 NUCLEAR POWER PLANT SYSTEMS INTERACTION STUDY INTERACTION CRITERIA HVAC DUCTWORK	FIGURE 6-3
DIV. MNE DR. JM DATE 3-20-81 CH. [illegible] SCALE NONE	APPROVED		



EBASCO SERVICES INCORPORATED

DIV. MNE DR. JM
DATE 3-20-81 CH. [illegible]
SCALE NONE

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT No. 3
NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY
INTERACTION CRITERIA
MECH PIPES OR ELECT CONDUITS

FIGURE 6-4

APPENDIX A-2.2 Results on the AFS of Nonconnected Systems Interactions (Cont'd)

APPENDIX A-2.2.1 Systems Interactions Induced by the Effects of a Safe Shutdown Earthquake (SSE) (Cont'd)

GENERAL DISCUSSION (Cont'd)

The plant walkdown activities were consistent with the methodology guidelines and evaluatioon criteria described in Chapters 5 and 6.

SUMMARY AND CONCLUSION

In general due to the lack of documentation, it was impossible to complete a comprehensive review of the seismic design classification. In those instances where documentation existed an appropriate reference was included.

Structures, systems and components that were not substantiated by seismic documentation consistent with the quality assurance requirements of Appendix B to 10CFR Part 50 were assumed to be nonseismic and were evaluated with respect to their effects on other Seismic Category I items.

Acceptable and unacceptable system interactions resulting from the failure of nonseismic structures, systems or components are presented in Appendix A-4.

From a review of the results of the seismic system interactions, the following items are considered to be the major contributors to the identification of adverse systems interactions,

1 - Crane/monorail structure located directly above the two (2) motor driven and the turbine driven auxiliary feedwater pumps.

2 - 4" nonseismic floor drain pipe directly above the electrical cable trays containing essential safety related equipment.

APPENDIX A-2.2 Results on the AFS of Nonconnected Systems Interactions (Cont'd)

APPENDIX A-2.2.1 Systems Interactions Induced by the Effects of a Safe Shutdown Earthquake (SSE) (Cont'd)

SUMMARY AND CONCLUSION (Cont'd)

3 - Space heaters and electrical lighting fixtures located directly above essential safety related equipment and structures.

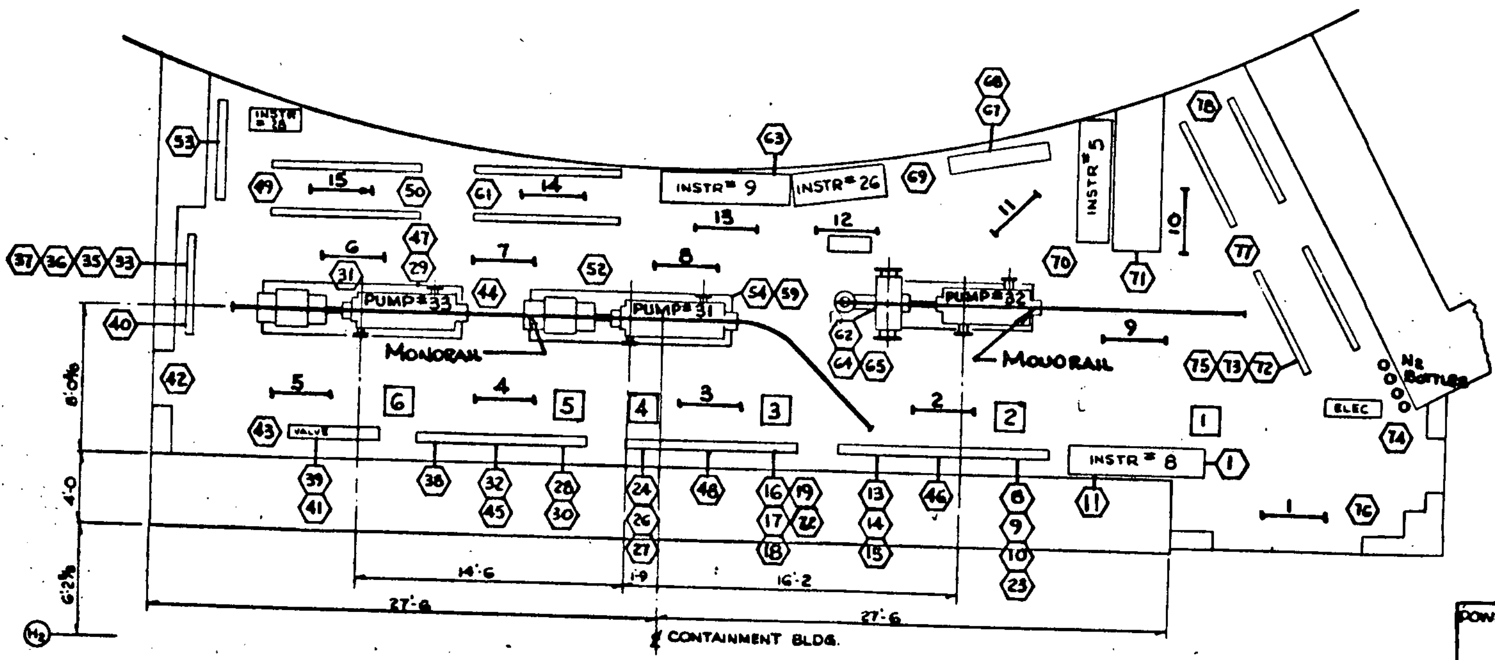
4 - Nonseismic electrical cable trays and conduit routed directly above essential safety related equipmenmt and structures.

5 - Large nonseismic instrument rocks located within close proximity to essential safety related equipment and structures.

6 - Large roll-up door located in the shieldwall whose structural failure could affect the flow control stations of the turbine driven auxiliary feedwater pumps.

Modifications including the possible use of guard structures, protective covers, and restraining devices are expected to prevent impairment of function due to the above concerns.

No PICTURE NUMBER
No LIGHT FIXTURE
No SPACE HEATER



PLAN EL 18'-6

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT No 3
NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY
AUXILIARY FEEDWATER BUILDING

EBASCO SERVICES INCORPORATED

SCALE NONE
APPROVED
DATE 3/ /81
DRAWN MAE
CHECKED MG

FIGURE A-TM.1

LINE No. / TARGETS		01 4" DR. LINE	02 ELEC. BOX FM-279	03 ELEC. BOX SX-2	04 SPACE HTR #2	05 LIGHT #2	06 WALL SW BOX	07 EXIDE LTG. & BOX	08 PSSW FOR HTR EUH 310	09 MONO RAIL	10 2" STA. AIR	11 EUH 311	12 ELEC. BOX SX-1	13 WALL BOX (THERMAL COUPLES) 321 & 322	14 SPACE HTR #3	15 LIGHT #3	16 SPACE HTR #4	17 2" CONDUIT	18 2" STA. AIR	19 ELEC. BOX 232 & CONDUIT	20 4" DR. LINE	21 SPACE HTR #5	22 2" CONDUIT	23 SWITCH BOXES & CONDUIT	24 SPACE HTR #6	25 SPEAKER	26 ELEC. BOX & Q27 CONDUIT	27 PUMP #33 POWER 4" CONDUIT	28 ELEC. BOX & SX-5 CONDUIT	29 LIGHT #7	30 3" CONDUIT	31 MONO RAIL	32 LIGHT #5	33 1½ & 4" CONDUIT
01	12" CT-1070																											41/43	41/43					
02	8" CT-1071 (HDR)							1			16				18	19	19	34	34			28			38	39								
03	8" CT-1071					46	13	13	13	46																								
04	6" CT-1072																																	46
05	1½" CT-1160																							32								45	45	45
06	6" CT-1073									48		17	17	17	18	19	19			24	27	48												
07	1½" CT-1159																																	
08	8" CT-1074 (HDR)	01			10					14/15	16				18	19	19	22	22			28												
09	6" CT-1074																				26				38									
10	6" CT-1075										16	16/17	16/17	17	18	19	19																	
11	8" CT-1076		08	09														23	23															
12	4" BFD-1001									47																				44/47	44	44		47
13	3" BFD-1001																																	
14	3" BFD-1002																																	
15	4" BFD-1003									59													29								59	54		

BUILDING: AUXILIARY FEEDWATER PUMP BUILDING

FIRE ZONE: 23 ELEVATION 18'-6

LOCATION WITHIN FIRE ZONE: NOT APPLICABLE - SEE KEY PLAN

EBASCO SERVICES INCORPORATED

DIV. MUE DR MG
DATE 3/24/81 CHK
SCALE NONE
APPROVED

POWER AUTHORITY OF STATE OF NEW YORK
INDIAN POINT No 3
NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
INTERACTION MATRIX

MATRIX No. L1
SHEET 1/39

SOURCE

LINE

TARGETS

No.	Line	01 2" Conduit	02 Mono Rail	03 Light #6	04 1½" Conduit	05 Elec. Box SX-7	06 Ventilation Louvers	07 Light #7	08 3" Conduit	09 Mono Rail	10 Light #8	11 4" Dr. Line	12 Elec. Box SX-6	13 Light #11	14 Light #12	15 3" Conduit	16 Conduits (Small)	17 Speaker	18 Conduit Box	19 Nitrogen Bottles	20 Space Htr #1	21 Roll-Up Door	22 Telephone Sta.	23 4" Dr. Line	24 Conduits From Rack 9 & 26	25
16	3" BFD-1003																									
17	3" BFD-1004																								63✓	
18	6" BFD-1005													✓70	✓70	70/71	✓71									
19	3" BFD-1005																	72✓	73✓	74✓	75✓	76✓	77✓			
20	3" BFD-1006																	72✓	73✓	74✓	75✓	76✓	77✓			
21	3" BFD-1007																							78✓		
22	3" BFD-1008																							78✓		
23	2" BFD-1009						40✓		52✓	52✓	52✓	53✓	53✓													
24	2" BFD-1010	✓29	31✓	31✓	33✓	36✓	40✓	✓44	44✓	44✓																
25	3" BFD-1011																									
26	2" BFD-1012				37✓																					
27	2" BFD-1013				35✓/37✓																					
28	1" BFD-1014																									
29	3" CD-1015																									
30	2" CD-1030																									

BUILDING: Auxiliary Feed Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: Not Applicable

EBASCO SERVICES INCORPORATED

DIV. MNE DR. MG

DATE 3/24/81 CH. M

SCALE NONE

APPROVED

POWER AUTHORITY OF STATE OF NEW YORK
INDIAN POINT No. 3
NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY
INTERACTION MATRIX

MATRIX No. L2

SHEET 2/39

INCHES

SOURCE

TARGETS	LINE	01 ELEC. BOX SX-7	02 VENTILATION LOUVERS	03 3" CONDUIT	04 SECURITY ALARM BOX & CONDUIT	05 MONO RAIL	06 LIGHT # 2	07 4" CONDUIT	08	09	10
31	2" CD-1031										
32	4" MS-1027			62		62 64 65	64/65	64/65			
33	1" MS-1120										
34	1" MS-1121										
35	10" MS-1016										
36	12" MS-1017										
37	2" LA (LINE)	36	40								
38	2" LC (LINE)		40	42	42						

BUILDING: Auxiliary Feed Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

EBASCO SERVICES INCORPORATED	POWER AUTHORITY OF STATE OF NEW YORK INDIAN POINT No 3 NUCLEAR POWER PLANT	MATRIX No. L3
DIV. MNE DR. MG DATE 3/4/82 CH. [illegible] SCALE NONE — APPROVED	SYSTEMS INTERACTION STUDY INTERACTION MATRIX	SHEET 3/39

SOURCE

LINE

TARGETS

		01 4" Conduit	02 Light #4	03 Light #15	04 Light #8	05 5" Conduit	06 3" Conduit	07 Light #14	08 Wall Lighting & Conduit	09 Conduits From Rack 9 & 26
06	6" CT-1073	48	48							
13	3" BFD-1001			49/50						
14	3" BFD-1002			49/50						
15	4" BFD-1003				54/59	54/59	54/59			
16	3" BFD-1003							61	61	63
17	3" BFD-1004							61	61	

BUILDING: Auxiliary Feed Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

EBASCO SERVICES INCORPORATED		POWER AUTHORITY OF STATE OF NEW YORK INDIAN POINT No 3 NUCLEAR POWER PLANT	MATRIX No. L4
DIV. MUE DR 36	APPROVED	SYSTEMS INTERACTION STUDY	SHEET 4/29
DATE 3/24/81 CHK		INTERACTION MATRIX	
SCALE NONE			

SOURCE

LINE		01 ELEC. BOX SE-G	02 LIGHT # 20	03 LIGHT # 21	04 ELEC. PANEL	05 SPACE HTR # 20	06 SPACE HTR # 21	07 LIGHT FIXTURES 22, 23 & 24	08	09	10
101	4" BFD-1001	80/82	81		84	83	85	85			
102	4" BFD-1002	80/82	81		84	83	85	85			
103	4" BFD-1003	80/82	81	82				85			
104	4" BFD-1004	80/82	81	82				85			
105	4" MS-1027	80/82		82							
106	10" MS-1016	80/82		82							
107	12" MS-1017	80/82	81								

TARGETS

BUILDING: AUXILIARY FEEDPUMP BUILDING

FIRE ZONE: 52A

LOCATION WITHIN FIRE ZONE: NA

EBASCO SERVICES INCORPORATED

DIV. MHE DR. MG

DATE 3/24/81 CH. [illegible]

SCALE NONE

APPROVED

POWER AUTHORITY OF STATE OF NEW YORK
INDIAN POINT No 3
NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY
INTERACTION MATRIX

MATRIX No. 65

SHEET 5/29

SOURCE

VALVES

TARGETS

		01 Wall SW Box	02 Exide Ltg. & Box	03 PSSW for HTR EUH 310	04 Speaker	05 Elec. Box & Q 27 Conduit	06 Mono Rail	07 Light # 2	08 4" Conduit	09
	~~12" CT-1070~~									
01	CT-6									
	~~8" CT-1071~~									
02	CT-64 (HOR.)				39✓	39✓				
03	CT-82 (1" DR)									
04	CT-29	13✓	13✓	13✓						
05	CT-30	13✓	13✓	13✓						
06	CT-35 (RELIEF)									
07	CT-84 (INSTR.)									
08	CT-84 (INSTR.)									
09	CT-83 (1" VENT)									
10	CT-94						46	46	46	
11	CT-94						46	46	46	
12	CT-98 (3/4" DR)									
13	CT-83 (1" VENT)									

BUILDING: AFW PUMP BUILDING

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

EBASCO SERVICES INCORPORATED

DIV. NUE CR. MG

DATE 3/4/81 CH. [illegible]

SCALE NONE

APPROVED

NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY

MATRIX

MATRIX No. VI

SHEET 6/39

SOURCE

VALVES

TARGETS

		01 ELEC. BOX FM-279	02 ELEC. BOX SX-2	03 EUH 311	04 ELEC. BOX SX-1	05 WALL BOX THERMAL COUPLE 321 & 322	06 SPACE HTR #3	07 LIGHT #3	08 SPACE HTR #4	09 4" DR. LINE	10 ELEC. BOX SX-3	11 SPACE HTR #6	12	13	14	15
	8" CT-1074															
38	CT-83 (1" VENT)											38				
	6" CT-1074															
39	CT-82 (1" DR.)															
40	CT-31									26	30					
41	PCV-1189									26	30					
42	SOV-1289									26	30					
	6" CT-1075															
43	CT-8 (1" DR.)															
44	CT-25 (CHECK)			17	17	17	18	19	19							
45	PCV-1187			17	17	17	18	19	19							
46	SOV-1287															
	8" CT-1076															
47	CT-82 (1" DR.)															
48	CT-28	08	09													

BUILDING: AFW Pump Bldg

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

EBASCO SERVICES INCORPORATED

DIV. MNE DR. MG CH [illegible]

DATE 3/4/81

SCALE None

APPROVED

POWER AUTHORITY OF STATE OF NEW YORK
INDIAN POINT No 3
NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY
INTERACTION MATRIX

MATRIX No. V4

SHEET 9/39

SOURCE

INSTR.

TARGETS		01 Conduit for Box 321 & 322	02 2" Conduit	03 2" Sta. Air	04 Elec. Box 232 & Conduit	05 Light # 3	06 Space Htr #4	07 4" Dr. Line	08 Switch Boxes & Conduit	09 Mono Rail	10 Light # 5	11 4 & 1½" Conduit	12 Light # 2	13	14	15
	8" CT-1071															
01	SENSING LINE OF PT-1264									8,9,46		8,9,46	8,9,46			
02	PI-1264									8,9,46		8,9,46	8,9,46			
	6" CT-1072															
03	FE-1123															
04	FC-1136-S		34	34					32							
05	FC-1136A-S		34	34					32							
06	PT-1265									45	45	45				
07	PI-1265									45	45	45				
	6" CT-1073															
08	FE-1121	20/21	22	22												
09	FC-1135-S				24	25	25	27								
10	FC-1135A-S							26								
11	SENSING LINE OF PT-1263															
12	PI-1263															

BUILDING: AFW

FIRE ZONE: 23

LOCATION: NA
WITHIN FIRE ZONE

EBASCO SERVICES INCORPORATED

DIV. I&C DR. DK
DATE 3-24-81 CH.
SCALE None

APPROVED

POWER AUTHORITY OF STATE OF NEW YORK
INDIAN POINT No 3
NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY
INTERACTION MATRIX

No. II

SHEET 21/39

ELECTRICAL

SOURCE

TARGETS

		01 Conduit 755	02 Conduit 7551	03 2" Conduit	04 Light #5	05 5" Conduit	06 Mono Rail	07 1" Conduit	08 Light #6	09 3" Conduit	10 Speaker	11 Instr. Air	[illegible] Space Htr #6	13 4" Dr. Line	14 2" Sta. Air	15 Space Htr #5	16 Light #4	17 Disc Switch EUH	18 Light #3	19 SX-3	20 Space Htr #4	21 Pull Box & other Small Conduit	22 4" Conduit	23 Pull Box EUH-	24 EUH- 3" Conduit	25 Elec Box Q-26	26 From Q-2G 1½" Conduit	27 From Q-2G 2½" Conduit	28 From Q-2G 3" Conduit	29 1½" Conduit from Rack 9 & 26	3[illegible] 1½" Conduit from SX-1	[illegible] 2½" Conduit from X-32	32 Power Supply to EUH #3 & 4	33 2" Conduit SX-2
01	TRAY- I			117✓										112										114	114					115	116	116	117	118✓
02	TRAY- II													113/202								113/202	113/202			113/201	113/201	113/201	113/201	115	116			118
03	TRAY- III													112										114	114					115	116	116	117	118✓
04	TRAY- IV																										113/201	113/201	113/201	115	116✓			
05	SX-5 LCV-5 & Conduit	100	100	100																														
06	Conduit 755			101	101	101	101	101✓	101✓	101																								
07	Conduit 7551			101	101	101	101	101	101✓	101																								
08	Motor #33			102		102	102	102	102	102																								
09	Conduit CT-C4			103	103			103			103✓	103✓	103✓																					
10	SX-3 PCV-1189 & Conduit			104				104✓						104✓	104	104	104																	
11	X232 Junction Box													105				105	105		105													
12	Conduit FC-1135													105				105	105✓		105													
13	Conduit FC-1135A													105✓				105	105✓		105													
14	Conduit FC-113C													106✓		106				106✓														
15	Conduit FC-113CA													106✓		106				106✓														
15A	Motor #31						X																											

BUILDING: Auxiliary Feedwater Pump Bldg

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

EBASCO SERVICES INCORPORATED

DIV. ELEC DRG. PJ
DATE 3/24/81 CH. [illegible]
SCALE NONE

APPROVED

POWER AUTHORITY OF STATE OF NEW YORK
INDIAN POINT 3
NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY

MATRIX No. E1

SHEET 33/39

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Bldg

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 01-09-03-46

INTERACTION SKETCH NO.:

FIGURE 5-1, PHOTO 46

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) AFW Pump Mono Rail
(Target) 8"CT-1071 AFW Pump Supply Line From Header

DESCRIPTION OF POSTULATED INTERACTION:

Mono Rail fails during Seismic Event and hits 8"CT-1071 Supply line from header

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - Mono rail is of sufficient mass to damage the supply line CT-1071. Assuming a concurrent single failure of PCV1188, No Suction flow to the turbine driven AFW pump is afforded.

M Gucciardi 2/9/81
Interaction Engineer/Date

[signature]
Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 01-09-06-48

INTERACTION SKETCH NO.:

FIGURE 5-1, PHOTO 48

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) MONO RAIL

(Target) 6" CT-1073 - SUPPLY LINE FROM COND STORAGE TANK HEADER TO AFW PUMP Nº 31

DESCRIPTION OF POSTULATED INTERACTION:

Mono Rail fails during seismic event and damages 6" CT-1073 Supply line to AFW Pump Nº 31

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - Loss of Suction flow to AFW Pump Nº 31 - Common mode failure - also damages 8" CT-1074 Header causing loss of suction flow to AFW Pumps Nº 32 & 33

M Gagliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 01-09-08-14/15

INTERACTION SKETCH NO.:

FIGURE 5-1, PHOTO 14 & 15

IDENTIFICATION OF INTERACTION COMPONENTS:

(SOURCE) AFW PUMP MONO RAIL

(TARGET) 8" CT-1074 SUPPLY HEADER FROM CITY WATER SUPPLY

DESCRIPTION OF POSTULATED INTERACTION:

Mono rail fails during seismic event and damages 8" CT-1074 Supply header from City Water Supply.

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - see sheet 11 for reasons

M Gogliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 01-09-12-47

INTERACTION SKETCH NO.:

FIG 5-1, PHOTO 47

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) AFW Pump Mono Rail

(Target) 4" BFD-1001 Feed supply from Pump No 33 to Steam Generators No 33 & 34

DESCRIPTION OF POSTULATED INTERACTION:

Mono Rail fails and damages 4" BFD-1001 feed supply from Pump No 33 to Steam Generators No 33 & 34.

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - Lose flow to SG's # 33 & 34 and with common mode failure of loss of suction to Turb Driven Pump & Motor Driven Pump 31, NO AFW to Stm generators

M Gagliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 01-09-15-59

INTERACTION SKETCH NO.:

FIGURE 5-1, PHOTO 59

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Mono Rail

(Target) 4" BFD-1003, Feed Supply from Pump 31 to Steam Generators No 31 & 32

DESCRIPTION OF POSTULATED INTERACTION:

Mono Rail fails and damages 4" BFD-1003 feed supply from pump No 31 to Steam generators No 31 & 32

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - Similar to Interaction 01-09-12-47, sheet 13. SEE FOR DETAILS

M Gogliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 01-31-04-45, 01-31-12-44, 01-31-15-54

INTERACTION SKETCH NO.:

5-1, PHOTO 45, 44 & 54

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Mono Rail

(Target) 6"CT-1072
4"BFD-1001
4"BFD-1003

DESCRIPTION OF POSTULATED INTERACTION:

Mono Rail fails from Supports and hits
6"CT-1072, 4"BFD 1001, 4"BFD 1003

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - lose Suction and discharge flow.

M Gogliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BUILD

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 02-02-24-31

INTERACTION SKETCH NO.:

5-1, Photo 31

IDENTIFICATION OF INTERACTION COMPONENTS:

Source - Monorail

Target - 2" BFD-1010

DESCRIPTION OF POSTULATED INTERACTION:

Monorail fails from support and hits 2" BFD-1010 causing rupture of the line,
Monorail also hits 2" BFD 1009 causing rupture of the line (see L-2 sheet 9)

EVALUATION & DISPOSITION OF INTERACTION: Unacceptable

Monorail failure and subsequent rupture of lines 2BFD-1010 and 1009 results in loss of function of motor driven pumps No 31 and 33. Coincident with a single failure of the turbine driven pump, results in complete loss of auxiliary feedwater supply to all steam generators.

N J Gogliardi 2/9/81
Interaction Engineer/Date

R J [signature] 2/10/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 02-06-23-40, 02-06-24-40

INTERACTION SKETCH NO.:

5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Ventilation Louvers
(Target) 2"BFD-1009
(Target 2"BFD-1010

DESCRIPTION OF POSTULATED INTERACTION:

Ventilation Louver fails from support and hits 2"BFD-1009 and 2"BFD-1010

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction.
See L-2 sheet 2 for evaluation.

N. Gagliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 02-09-23-52, 02-09-24-44

INTERACTION SKETCH NO.:

5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

Source) Monorail

Target) 2"BFD-1009
2"BFD-1010

DESCRIPTION OF POSTULATED INTERACTION:

Monorail fails from support and hits 2"BFD-1009 and 2"BFD-1010

EVALUATION & DISPOSITION OF INTERACTION: Unacceptable interaction
Interaction results in total loss of auxiliary feedwater flow to steam generators when assumed coincident single failure is Turbine driven pump failure to start.

N. Gogliardi 2/9/81
Interaction Engineer/Date

[signature]
Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Bldg

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 02-20-19-75, 02-20-20-75

INTERACTION SKETCH NO.:

A5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Space Heater #1

(Target) 3" BFD-1005
3" BFD-1006

DESCRIPTION OF POSTULATED INTERACTION:

Space Heater #1 fails from support and hits 3" BFD-1005 and 3" BFD-1006

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - Heater can cause loss of function of 3" BFD-1005 and 3" BFD-1006

N Gogliardi 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Bldg

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: UA

INTERACTION NO.: 02-22-19-77, 02-22-20-77

INTERACTION SKETCH NO.:

IDENTIFICATION OF INTERACTION COMPONENTS:
(Source) Communication Center
(Target) 3" BFD-1005
3" BFD-1006

DESCRIPTION OF POSTULATED INTERACTION:
Communication Center fails from support structure and hits 3" BFD-1005 and 3" BFD-1006

EVALUATION & DISPOSITION OF INTERACTION:
Unacceptable interaction - Communication center can cause loss of function of 3" BFD 1005 and 1006

[signature] 2/9/81
Interaction Engineer/Date

[signature] 2/10/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N/A

INTERACTION NO.: 02-23-21-78, 02-23-22-78

INTERACTION SKETCH NO.:

A5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) 4" plumbing drain line
(Target) 3" BFD-1007
3" BFD-1008

DESCRIPTION OF POSTULATED INTERACTION:

4" plumbing line fails from support and hits 3" BFD-1007 and 3" BFD-1008

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction, 4" line can damage 3" BFD-1007 and 3" BFD-1008 causing loss of function.

[signature] 2/9/81
Interaction Engineer/Date

[signature]
Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 03-05-32-62/64/65

INTERACTION SKETCH NO.:

A5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Monorail

(Target) 4" MS-1027

DESCRIPTION OF POSTULATED INTERACTION:

Mono rail fails from support and hits 4" MS-1027

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - loss of steam supply to turbine.

V. Jabardi 2/9/81
Interaction Engineer/Date

RJ Drago
Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 52A

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 05-05-101-83, 05-05-102-83

INTERACTION SKETCH NO.:

A5-2

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Space Heater #20
(Target) 4" BFD-1001
4" BFD-1002

DESCRIPTION OF POSTULATED INTERACTION:

Space heater #20 fails from support and hits 4" BFD-1001 and 4" BFD-1002

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - Space heater #20 can cause loss of function of lines.

[signature] 2/10/81
Interaction Engineer/Date

[signature]
Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 52A

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 05-06-101-85, 05-06-102-85

INTERACTION SKETCH NO.:

A5-2

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Space Heater #21
(Target) 4" BFD-1001
4" BFD-1002

DESCRIPTION OF POSTULATED INTERACTION:

Space heater #21 fails from support and hits 4" BFD-1001 & 1002

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - loss of function redesign

[signature] 2/9/81
Interaction Engineer/Date

[signature]
Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 06-04-02-39

INTERACTION SKETCH NO.:

A5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Paging Speaker
(Target) CT-64 w/ limit switches

DESCRIPTION OF POSTULATED INTERACTION:

Paging speaker fails from support and hits CT-64 w/limit switches

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - speaker can damage limit switches which are required for status indication.

V Tagliarck 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Bldg

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 09-06-44-18, 09-06-45-18

INTERACTION SKETCH NO.:

A5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) Space heater #3
(Target) CT-25 check valve
PCV-1187

DESCRIPTION OF POSTULATED INTERACTION:

Space heater #3 fails from support and hits C-25 and PCV-1187

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction - PCV-1187 can be damaged and lose functionability

V Goghari 2/9/81
Interaction Engineer/Date

Reviewer/Date
2/10/81

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW PUMP BLDG

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: ~~[illegible]~~, 09-07-45-19, ~~[illegible]~~, 09-08-45-19

INTERACTION SKETCH NO.: A5-1

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source) lighting fixture #3, Space heater #4
(Target) PCV-1187

DESCRIPTION OF POSTULATED INTERACTION:

lighting fixture and space heater #4 fail from supports and hit PCV-1187

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable interaction — loss of function is credible.

N. Gogliardi 2/9/81
Interaction Engineer/Date

[signature] 2/10/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: 01-09-11-12, 02-09-11-12, Matrix No I-1

INTERACTION SKETCH NO.: 5-3, Photo 46 for sources, Photo 8 & 9 for targets

IDENTIFICATION OF INTERACTION COMPONENTS:
Targets: PT-1264 sensing line & PI-1264
Sources: Mono Rail, 4" & 1/2" Conduits, Light #2

DESCRIPTION OF POSTULATED INTERACTION:
Seismically induced failures of the sources can cause damages to PT-1264 and its instrument tubing thereby resulting in water spillage and a loss of AFW Pump #2 suction pressure indication in the control room (for NPSH protection).

EVALUATION & DISPOSITION OF INTERACTION:
Interaction is unacceptable.
Review seismic documentation. If the sources are not seismically supported, provide seismic supports.

[signature] 2/11/81
Interaction Engineer/Date

[signature] 2/20/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: 06-09-010-011, 07-09-010-11; Matrix No. I-1

INTERACTION SKETCH NO.: 5-3, Photo 45

IDENTIFICATION OF INTERACTION COMPONENTS:

Targets: PT-1265 sensing line & PI-1265 (for measurement of Pump 33 suction pressure)

Sources: Monorail, Light #5, 4" and 1/2" conduits

DESCRIPTION OF POSTULATED INTERACTION:

Seismically induced failures of the sources can cause damages to PT-1265 and its instrument tubing thereby resulting in water spillage and a loss of AFW Pump 33 suction pressure indication in the control room.

EVALUATION & DISPOSITION OF INTERACTION:

Interaction is unacceptable

Review seismic documentation. If the sources are not seismically supported, provide seismic supports.

D Kavanaugh 2/11/81
Interaction Engineer/Date

N Gagliardi 2/20/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 34-07-19-108

INTERACTION SKETCH NO.: Fig 5-4, photo 108

IDENTIFICATION OF INTERACTION COMPONENTS:

Source : Electrical Unit Space Heater #2

Target : Box SX-2 and its associated conduits

DESCRIPTION OF POSTULATED INTERACTION:

Space Heater #2 fails from its support and hit the target

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction: Unit heater with a motor driven fan has sufficient mass to cause an unacceptable damage.

G C Pan 2-12-81
Interaction Engineer/Date

[signature] 2/23/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 34-02-18-107, 34-02-19-108, 34-03-18-107, 34-03-19-108
34-04-18-107, 34-05-18-107

INTERACTION SKETCH NO.: Fig 5-4, photo 107, 108

IDENTIFICATION OF INTERACTION COMPONENTS:

Sources: 1" Instrument Air, 2" Conduit, 1" Conduit, Space Heater DS

Targets: Box SX-1 and conduit to PCV-1187
Box SX-2 and Conduit to PCV-1188

DESCRIPTION OF POSTULATED INTERACTION:

Instrument Air piping, electrical conduits and Heater Disconnect Switch fail from their support and hit the targets identified.

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - a. Boxes SX-1 and SX-2 are unlikely to be damaged. Boxes are wall mounted.
b. Flexible conduits for PCV-1187 & PCV-1188 are very suseptible to any falling objects

G C Pan 2-12-81
Interaction Engineer/Date

N Gagliardi 2/23/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 33-13-01-112, 33-13-02-113/202, 33-13-03-112

INTERACTION SKETCH NO.: Fig 5-4, photo 112, 113, 202

IDENTIFICATION OF INTERACTION COMPONENTS:

Source: 4" Drain Line
Target: Tray I, II, III.

DESCRIPTION OF POSTULATED INTERACTION:

4" Drain line fails from supports and hit Cable Tray I, II, III directly below.

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction

G C Pall 2-12-81
Interaction Engineer/Date

[signature] 2/22/81
Reviewer/Date

Ebasco Services Incorporated

 E1

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 33-20-11-105, 33-20-12-105, 33-20-13-105

INTERACTION SKETCH NO.: Fig 5-4, Photo 105

IDENTIFICATION OF INTERACTION COMPONENTS:

Source: Space Heater #4

Target: Box #X232 and conduits to trays
Conduits to FC-1135 & FC-1135A

DESCRIPTION OF POSTULATED INTERACTION:

Space Heater #4 falls from its support and hits target

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - Damage to box #X232 and conduits will disable the operation of FC-1135 and FC-1135A

GC Pan 2-11-81
Interaction Engineer/Date

[signature] 2/23/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: NA

INTERACTION NO.: 33-06-06-101, 33-06-07-101, 33-06-08-101

INTERACTION SKETCH NO. Fig 5-4, photo 101, 102

IDENTIFICATION OF INTERACTION COMPONENTS:

(Source): Overhead Mono Rail
(Target): Motor #31
Motor #33
Feeder Conduits #755 & 7551 for motor #33

DESCRIPTION OF POSTULATED INTERACTION:

Mono rail falls from supports and hits motor #33, its feeder conduits 755 & 7551 and motor #31

Both motors are 400 HP, 440 Volt with open dripproof enclosure.

EVALUATION & DISPOSITION OF INTERACTION:

Unacceptable Interaction - Targets may be damaged by the failure of the mono rail support. Motor terminal boxes, which are made of fabricated steel plates, are deemed to be the most susceptible.

G C Pan 2-11-81
Interaction Engineer/Date

[signature] 2/12/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: generic

INTERACTION SKETCH NO.: Photo No. (later)

IDENTIFICATION OF INTERACTION COMPONENTS:

Volume tanks of the AFW turbine steam supply pressure control valve, PCV-1139.

DESCRIPTION OF POSTULATED INTERACTION:

These tanks must be seismically supported. The existing supports are questionable.

EVALUATION & DISPOSITION OF INTERACTION:

Review seismic documentation of the tanks and their supports.
If found unacceptable, provide new supports.

D [signature] 2/9/81
Interaction Engineer/Date

[signature] 2/20/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: 19-09-10-11-12, 20-09-10-11-12, 21-09-10-11-12, 22-09-10-11-12, Matrix No. I-2

INTERACTION SKETCH NO.: 5-3, Photo No. 44, 47

IDENTIFICATION OF INTERACTION COMPONENTS:

Targets: PT-406B, PI-1262, PT-1262, PT-1262R

Sources: Light # 7, 3' conduit, Monorail, instrument tubings between Rack #2E & Rack #5

DESCRIPTION OF POSTULATED INTERACTION:

Seismically induced failures of the sources can cause damages to the targets thereby resulting in water spillage and loss of functions the pressure transmitters which sense outlet pressure of AFW Pump 31.

Seismically induced failure of instrument lines between Rack #2E and Rack #5 is aggravated by the fact that Rack #2E is not anchored.

EVALUATION & DISPOSITION OF INTERACTION:

Interaction is unacceptable.

Review seismic documentation and provide seismic supports for the sources if they are not so supported.

Danai Kanchanaoy 2/4/81
Interaction Engineer/Date

M. [illegible] 2/20/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: 17-07-08-13, 18-07-08-13, Matrix No. I2

INTERACTION SKETCH NO.: 5-3, Photo No. 12 & 23

IDENTIFICATION OF INTERACTION COMPONENTS:

Targets: PCV-1188 Valve Station (including valve, solenoid, instrument air lines, electrical wires and terminal box)

Sources: 2" conduit, 2" station air pipe, Light #2.

DESCRIPTION OF POSTULATED INTERACTION:

Seismically induced failures of the sources can cause: 1) a loss of function of the city supply valve PCV-1188 of AFW Pump 32 and 2) instrument air piping rupture.

EVALUATION & DISPOSITION OF INTERACTION:

The rupture of instrument air piping is a common mode failure in the entire instrument air piping in the pump room. Such failure will cause pneumatically operated valves in the room to be inoperable. The loss of function of PCV-1188 will require manual operation to open the valve. However, PCV-1188 does not have a hand operator. Interaction is unacceptable. Review seismic documentation and provide seismic supports for the sources if they are not so supported.

D Kavchimaga 2/11/81
Interaction Engineer/Date

[signature] 2/20/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: 15-01-02-03-04-05-07-08, 16-01-02-03-04-05-06-07
Matrix No. I-2

INTERACTION SKETCH NO.: 5-3, Photo No. 17, 18, 19, 22

IDENTIFICATION OF INTERACTION COMPONENTS:

Targets: PCV-1187 Valve Station (including valve, solenoid, instrument air lines, electrical wires and terminal box)

Sources: Electrical box SX-1, space heater # 3, Light # 3, Space heater #4, 2" conduit, 2" station air pipe.

DESCRIPTION OF POSTULATED INTERACTION:

Seismically induced failures of the sources can cause: 1) a loss of function of city water supply valve PCV-1187 of AFW Pump 31 and 2) instrument air piping rupture.

EVALUATION & DISPOSITION OF INTERACTION:

The rupture of instrument air piping is a common mode failure in the entire instrument air piping in the pump room. Such failure will cause pneumatically operated valves in the room to be inoperable.
The loss of function of PCV-1187 will require manual operation to open the valve. However, PCV-1187 does not have a hand operator.
Interaction is unacceptable. Review seismic documentation and provide seismic supports for the sources if they are not seismically supported.

D Kanchanat 2/11/81
Interaction Engineer/Date

[signature] 2/20/81
Reviewer/Date

Ebasco Services Incorporated

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
EXTERNALLY INDUCED SYSTEMS INTERACTION
INTERACTION DOCUMENTATION SHEET

BUILDING: AFW Pump Building

FIRE ZONE: 23

LOCATION WITHIN FIRE ZONE: N.A.

INTERACTION NO.: 09-04-05-06-07, 10-07. Matrix No I-1

INTERACTION SKETCH NO.: 5-3, Photo No. 24, 25, 26, 27

IDENTIFICATION OF INTERACTION COMPONENTS:

Targets: FC-1135-S, FC-1135A-S (Flow switches to monitor AFW Pump 31 suction flow)

Sources: Elec. box 232 and conduit, Light #3, space heater #1, 4" drain line

DESCRIPTION OF POSTULATED INTERACTION:

Seismically induced failures of the sources can cause damages to the targets and their sensing lines thereby resulting in water spillage and a loss of Pump 31 suction flow alarm in the control room.

EVALUATION & DISPOSITION OF INTERACTION:

The loss of suction flow alarm is not critical. However, the water spillage can cause damages to other components in the area. Therefore, seismic documentation review should be conducted. If the sources are not seismically supported, provide seismic supports

D Kanchanas 2/11/81
Interaction Engineer/Date

M Gagliardi 2/20/81
Reviewer/Date

Ebasco Services Incorporated

ATTACHMENT A

CONTROL OF HEAVY LOADS

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
NOVEMBER 17, 1981

RESPONSES TO REQUESTS FOR INFORMATION IN SECTIONS 2.2, 2.3, AND 2.4 OF ENCLOSURE 3 TO NRC JULY 31, 1980 LETTER

2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE VICINITY OF FUEL STORAGE POOL

NUREG 0612, Section 5.1.2, provides guidelines concerning the design and operation of load-handling systems in the vicinity of stored, spent fuel. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that in this area, either the likelihood of a load drop which might damage spent fuel is extremely small or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I through III.

RESPONSE: For the reasons given in the response to Item 3 of the Authority's June 22, 1981 submittal, the Spent Fuel Storage Building crane has been excluded from consideration until such time as a decision is made regarding a spent fuel shipping cask. Currently, no heavy loads are carried within the vicinity of the spent fuel pool.

2.3 SPECIFIC REQUIREMENTS OF OVERHEAD HANDLING SYSTEMS OPERATING IN THE CONTAINMENT

NUREG 0612, Section 5.1.3, provides guidelines concerning the design and operation of load-handling systems in the vicinity of the reactor core. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that in this area, either the likelihood of a load drop which might damage spent fuel is extremely small or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I through III.

ITEM 2.3.1. Identify by name, type, capacity, and equipment designator any cranes physically capable (i.e., taking no credit for any interlocks or operating procedures) of carrying heavy loads over the reactor vessel.

RESPONSE: The only handling system within containment physically capable of carrying heavy loads over the reactor vessel is the Containment Polar Gantry Crane. The crane was designed by Whiting Corporation and possesses a main and auxiliary hoist with capacities of 175 tons and 35 tons, respectively.

ITEM 2.3.2. Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads, or are permanently prevented from the movement of any load either directly over the reactor vessel or to such a location where in the event of any load-handling-system failure, the load may land in or on the reactor vessel.

RESPONSE: The only other handling system inside the containment is the Manipulator Crane used for refueling operations. It is sized to handle single fuel assemblies, i.e., no heavy loads as defined in NUREG 0612 are handled by this handling system.

ITEM 2.3.3. Identify any cranes listed in 2.3.1 above which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: A probabilistic failure analysis of the Polar Crane has been performed applicable to removal and installation of the Reactor Head and the Upper Internals. Drops of these two components are controlling with respect to evaluating the potential for damaging the vessel nozzles or spent fuel in the core. The failure analysis utilized fault tree methodology and addressed all ways the polar crane system could fail, including failure of control circuitry, protective devices, brakes, structural failures of the crane or lifting rigs, and operator errors. The results of this analysis indicated that the probability of dropping the head or internals after initial lift off and leveling of the load is extremely small. Initial lift off and leveling of the load involves raising the load a height of approximately 1½ feet. The duration of this operation is approximately 15 minutes. Although still small, the probability of a drop during initial lift is somewhat larger than a drop from a greater height. Therefore, structural analyses have been performed to determine if the vessel nozzles or fuel in the core could be damaged if such a drop during initial lifting should occur. These are described in the response to Item 2.3.4.c.

One other load is carried over the open reactor vessel that could potentially damage spent fuel in the vessel. This is the Reactor Vessel Weld ISI tool. Its weight is approximately 5 tons. For this particular lift, which is performed by the Auxiliary Hoist, adequate load handling reliability will be assured on the same basis as for loads lifted by the Auxiliary Hoist in the Annulus Region. This basis is described in the response to Item 2.4.1.

ITEM 2.3.4. For cranes identified in 2.3.1 above not categorized according to 2.3.3, demonstrate that the evaluation criteria of NUREG 0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in your response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the containment and your determination of compliance. This response should include the following information for each crane:

ITEM 2.3.4.a. Where reliance is placed on the installation and use of electrical interlocks or mechanical stops, indicate the circumstances under which these protective devices can be removed or bypassed and the administrative procedures invoked to ensure proper authorization of such action. Discuss any related or proposed technical specifications concerning the bypassing of such interlocks.

RESPONSE: In no cases is reliance placed on mechanical stops or electrical interlocks.

ITEM 2.3.4.b. Where reliance is placed on other, site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications and discuss administrative or physical controls provided to ensure the continued validity of such considerations.

RESPONSE: Loads always lifted when the reactor vessel head is in place or the reactor is defueled were not considered as loads that could potentially drop into the core. These are: the CRDM missile shields, the CRDM missile shield support beams, the reactor vessel head stud tensioners, and the lower internals. No administrative controls are required to enforce this situation, because it is physically impossible to disassemble or reassemble the reactor such that these loads would be carried over an open vessel.

There are a number of other loads that could be moved within the containment when the reactor vessel head is removed. Procedures prohibit movement of any of these loads, including the crane load block, over the refueling cavity when the reactor vessel head is removed and there is irradiated fuel in the vessel. These procedures will be reviewed with operators as part of the qualification and training program and will be strictly enforced by individuals in charge of lifts by the Polar Crane. These administrative controls are judged to be adequate to preclude postulating that any of these loads drop into or onto an open reactor vessel.

ITEM 2.3.4.c. Analyses performed to demonstrate compliance with Criteria I through III should conform with the guidelines of NUREG 0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

RESPONSE: There are three potential consequences of interest when considering load drops onto the open reactor vessel. These are: (1) loss of reactor vessel integrity, (2) fuel cladding damage and the resultant radiological dose, and (3) fuel crushing and the possibility of a resulting criticality condition. Criteria I-III in Section 5.1 of NUREG 0612 addresses each of these potential consequences. The evaluations below have been performed to address these issues:

Reactor Vessel Integrity - Structural Evaluation

During normal refueling operations, the reactor pressure vessel (RPV) head assembly is initially lifted a small distance above the flange and checked for levelness. It is then raised a height of 29.0 feet above the flange. Once at the desired height, the RPV head is moved west towards its storage stand which rests on the operating deck. Reassembly is in reverse. The potential for fuel damage or a loss of safe shutdown capability affecting the ability to get water to the core resulting from a loss of integrity of RPV nozzles was evaluated.

Based on the failure analysis described in the response to Item 2.3.3, the RPV head was assumed to drop 1.5 feet through air impacting on the RPV flange. The general methods of analysis which are documented in WCAP-9198[1/] were incorporated using parameters which are applicable to the Indian Point plant.

1/ Alexander, D. W., Shakeley, R., and Dudek, D. F., Reactor Vessel Head Drop Analyses," WCAP-9198, Westinghouse Electric Corporation, January, 1978.

The RPV head was found to impact the RPV flange at a velocity of 9.83 ft./sec. During the postulated head drop, the RPV head loads the shell, but does not load the fuel. Since the head is postulated to be lifted to only 1.5 feet at the time of the drop, the head is still engaged on its guide studs and the control rod drive shafts are still within their respective head penetrations. For this reason, the drop is not expected to impart a significant impact load to the control rod drive shafts. Loading the control rod drive shafts is the only feasible mechanism for loading the fuel as a result of this drop. On this basis, damage to the fuel is not predicted.

The major portion of impact load of the RPV head is transmitted directly to the RPV flange. The load path is through the RPV shell to the two inlet nozzles and two outlet nozzles from which the RPV is supported. The dynamic model conservatively neglects energy absorption by the guide studs or the reactor internals. The stiffnesses of the RPV shell, the supported inlet and outlet nozzles, and the RPV support are modeled along with the associated masses of the actual system. The total impact load was calculated to be 47.2 million pounds. The load was assumed to be distributed to each nozzle in proportion to their stiffness resulting in a maximum principal stress in the outlet nozzle of 26,750 psi. This compares to an allowable stress of 84,000 psi. Based upon this calculation, a loss of nozzle integrity is not predicted, and the reactor coolant pressure boundary remains intact. Therefore, sources of cooling water which are provided through attached piping such as RHR or safety injection remain available.

In performing the RV head drop analysis, the following exceptions were taken to Appendix A of NUREG 0612:

(1) NUREG 0612 requires that the RPV head drop be evaluated for a fall from its maximum height. This evaluation was limited to a nominal height of 1.5 feet corresponding to a drop during initial liftoff. The basis for this exception is provided in the response to Item 2.3.3.

(2) The evaluated head drop is bounding in producing a maximum load to the RPV nozzles and the fuel. Off center drops over the RPV are not evaluated because the head is assumed to drop when still engaged on the guide studs. The orientation for probable drops is essentially flat and flange to flange based on the small drop height assumed and the fact that the head is still engaged on the guide studs.

Fuel Damage

As indicated above, no fuel damage was predicted as a result of a reactor vessel head drop. However, the limiting situation for fuel damage was judged to be the postulated drop of the upper internals package into the vessel. A conservative structural evaluation was performed of a drop of the upper internals during initial lifting as described below.

During normal refueling operations, the reactor vessel upper internals is initially lifted a small distance and checked for levelness. It is then lifted approximately 25 feet above the top of the core. Once at the desired height, the upper internals is moved west towards its storage stand which rests on the refueling cavity floor. For the reasons described in Section 2.3.3, it was postulated that the Polar Crane or the Upper Internals Lifting Rig fails during initial liftoff of the upper internals. The height of this drop was assumed to be 1.5 feet.

The impact velocity was calculated to be 9.83 feet per second. The effects of drag, bouyancy, and a "dashpot" effect due to the tight tolerance with the core barrel were conservatively neglected.

The total kinetic energy for the drop of the 143,000 pound (including load block and lifting rig) upper internals structure was calculated to be 2,145 thousand foot-pounds. This energy is assumed to be dissipated evenly by each of the 193 fuel assemblies in the core. The fuel cladding was considered to fail at a plastic strain of 1 percent. This criteria is based upon the irradiated properties of Zircalloy-4, the cladding material.

The impact load is transmitted from the upper core plate to the upper nozzle of the fuel assembly, through the 20 guide tubes, and to the lower nozzle of the fuel assembly. The fuel rods are not significantly loaded unless either the upper nozzle contacts the fuel due to elastic shortening and/or buckling of the guide tubes or the fuel assembly deflects laterally as a composite element.

It was found that the guide tubes reach their elastic limit prior to buckling elastically. The energy absorbed by axial deformation up to the elastic limit is 25,900 foot-pounds for the entire core. It is expected that the guide tubes would then buckle inelastically. The additional energy absorbed in this failure mode until the fuel assembly upper nozzle impacts the fuel rods is neglected.

Individual fuel rods are predicted to buckle elastically between spacer guides at a load of 120 pounds. This corresponds to 8,730 foot-pounds of energy due to axial deformation for the entire core. The additional energy of 180 thousand foot-pounds can be absorbed beyond the point of critical buckling through bending until the cladding strain reaches a value of 1 percent plastic. The fuel rod is assumed to take a sinusoidal shape based upon a pinned-pinned boundary condition. Accordingly, the deflection along the fuel rod is given by,

$$Y = A \sin \frac{\pi}{L} X \qquad (1)$$

where L = length of fuel rod between spacer grids
A = lateral deflection of fuel rod at mid span
X = distance along span
Y = lateral deflection of fuel rod at a distance X along the span

From beam theory,

$$1/R = \frac{d^2y}{dx^2} = -\frac{M(x)}{EI} \qquad (2)$$

where R = radius of curvature
M(x) = moment at a point x
E = youngs modulus
I = moment of inertia

The strain energy in bending is given by,

$$U_b = \frac{1}{2}\int_0^L \frac{M(x)^2}{EI}\,dx \tag{3}$$

From (2) and (3), it follows that

$$U_b = \frac{1}{2}\int_0^L EI\left(\frac{d^2y}{dx^2}\right)^2 dx \tag{4}$$

Differentiating the approximated deflection curve (1),

$$\frac{d^2y}{dx^2} = -\left(\frac{\pi}{L}\right)^2 A \sin\frac{\pi}{L}x \tag{5}$$

and substituting (5) into (4),

$$U_b = \frac{\pi^4 A^2 EI}{2L^4}\int_0^L \sin^2\left(\frac{\pi}{L}\right)x\,dx$$

$$U_b = \frac{\pi^4 A^2 EI}{4L^3} \tag{6}$$

From (6) it follows that

$$A = \sqrt{\frac{4U_b L^3}{\pi^4 EI}} \tag{7}$$

From (2) and evaluating (5) at x = L/2,

$$\frac{1}{\beta} = A\left(\frac{\pi}{L}\right)^2 \tag{8}$$

The bending strain at any fiber at a distance y from the neutral axis is given by,

$$\epsilon_b = y/R \tag{9}$$

Substituting (8) into (9)

$$\epsilon_b = Ay\left(\frac{\pi}{L}\right)^2 \tag{10}$$

Combining the bending strain from (10) with the axial strain, a total strain of 0.22 percent was calculated. This compares to a yield strain of 0.29 percent and the allowable plastic strain of 1 percent.

Based upon this analysis where the total kinetic energy is conservatively assumed to be taken by the fuel, a fission product release is not predicted from the fuel.

Criticality Considerations

The potential for a criticality condition as a result of a load drop into the core has been evaluated independent of the specific load being considered. Criterion II, Section 5.1 of NUREG 0612 requires that the resultant keff not be greater than 0.95. Additionally, Section 4.2 of Appendix A to NUREG 0612 provides guidelines for neutronics analyses of PWR cores. Since the Indian Point reactor utilized the same fuel geometry analyzed in Section 2.2 of NUREG 0612, we believe the analyses are applicable. In this case, the maximum increase in keff due to fuel crushing would be about 0.02. Since the Indian Point Technical Specifications require at least 10% Δ k/k during reactor vessel head removal and while loading and unloading fuel from the reactor, Criterion II of Section 5.1 is satisfied as the maximum achievable keff is less than 0.92.

2.4 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, CORE DECAY HEAT REMOVAL, OR SPENT FUEL POOL COOLING

NUREG 0612, Section 5.1.5, provides guidelines concerning the design and operation of load-handling systems in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that in these areas, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small or that damage to such equipment from load drops will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1 and their loads in your response to 2.1.3.3.

ITEM 2.4.1: Identify any cranes listed in 2.1.1 above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: The load handling reliability of two handling systems has been evaluated because of the potential impact of loads on equipment required to achieve and maintain safe shutdown. The evaluation of each is described below:

Auxiliary Hoist of the Polar Crane

The Polar Crane Auxiliary Hoist has a capacity of 35 tons and has a hook travel that can service the Annulus Region between the containment wall and the crane wall outside of the gantry legs. For the purpose of addressing the NUREG 0612 guidelines for this region of the containment, the load handling reliability of the Auxiliary Hoist has been evaluated against the criteria of Section 5.1.6. Based on the discussion below, adequate load handling reliability of the Auxiliary Hoist in the Annulus Region is demonstrated and, therefore, load drops into this region have not been postulated.

The auxiliary hoist is mounted on the trolley frame and fully satisfies the criteria in CMAA-70-1975 and ANSI B30.2-1976. For most load handling operations, the auxiliary hoist satisfies the intent of Section 5.1.6 of NUREG 0612 (i.e., dual load path or increased safety factors in lieu of normal 5:1).

The auxiliary hoist components are designed with a 5:1 design safety factor on ultimate strength . For loads of less than 17.5 tons, the design safety factor for the hoist will be better than 10:1. With the exception of the equipment hatch door/airlock, all loads typically carried in the Annulus Region are less than 17.5 tons. The equipment hatch door weighs approximately 25 tons, which still results in a minimum safety factor on crane load bearing components of 8:1.

In addition, the auxiliary hoist has eight parts of 7/8" wire rope. Based on published breaking strengths, the rope has a breaking strength of 245 tons. This gives a factor of safety for the wire rope of better than 14:1 for loads less than 17.5 tons and approximately 10:1 for the 25-ton equipment hatch door. Furthermore, redundant holding brakes are provided of greater than 150% capacity that are engaged when power to the hoist is lost or removed. To satisfy the intent of Section 5.1.6 of NUREG 0612, the following actions will be taken:

(1) Certified slings (ANSI B30.9) will be utilized with the auxiliary hoist for loads lifted in the Annulus Region.

(2) An extensive inspection program will be provided for ropes, brakes, and limit switches. This will include a thorough visual inspection prior to each refueling outage and checking for proper functioning of brakes and limit switches.

(3) More stringent criteria on rope replacement will be utilized (replace when six or more randomly distributed wires in one rope lay are found damaged, in lieu of the ANSI B30.2 criteria of 12 or more).

(4) A second upper limit switch will be installed on the auxiliary hoist.

(5) As indicated in the Authority's June 22, 1981 submittal, load handling and operator qualification procedures have been upgraded to meet the guidelines of NUREG 0612 and ANSI B30.2-1976. These procedures will be fully implemented by January, 1982.

Auxiliary Feedwater Pump Building Monorails

To assure that the likelihood of a load drop is sufficiently small that a load drop need not be postulated from the Auxiliary Feedwater Pump Building Monorails, the design of this handling system was compared to the criteria of Section 5.1.6 of NUREG 0612. Since NUREG 0612 pertains to overhead bridge cranes, it is not directly applicable to handling systems such as these monorail hoists. Accordingly, this comparison was performed to assure that the intent of the Section 5.1.6 criteria is satisfied. The following provides the results of this comparison to show that the intent of Section 5.1.6 is satisfied:

(1) The monorails and their attaching hardware were designed to AISC specifications for a rated load of 5 tons each. The AISC specifications call for a design safety factor of 5:1 against ultimate strength for the maximum stress. This gives an ultimate capacity of 25 tons or a safety factor of 13:1 for the maximum loads anticipated for these monorails.

(2) These monorails do not have a hoist permanently attached. To provide increased safety margins to meet the intent of Section 5.1.6 of NUREG 0612, procedures will require use of a hoist with ratings that are at least 2½ times greater than the weight of the load to be handled. Hoists that meet ANSI B30.16 or some other equivalent industry standard will be used. Such hoists are designed to manufacturer's specifications that require all components to meet a design safety factor of better than 5:1 on ultimate strength. This will result in the selection of a hoist that gives a design safety factor of better than 12:1 for the maximum loads that would be handled over the auxiliary feedwater pumps.

(3) Certified slings (ANSI B30.9) will be utilized when handling loads with the auxiliary feedwater pump monorails.

(4) Dynamic loads need not be considered for these hoists. The hoists are hand-driven type with a pawl-rachet holding device that is secured by a friction type disc brake. Lowering is accomplished by driving against the holding brake. The dynamic load would only occur on hoisting due to the pawl action; however, this load would be small. For these hoists, a load drop during hoisting is not of safety concern. The concern is only if a drop were to occur in transporting the load along the monorail over an auxiliary feedwater pump, motor, or piping.

(5) Hoists and the monorail system are inspected and maintained in accordance with ANSI B30.11, ANSI B30.16, and manufacturer's criteria.

ITEM 2.4.2. For any cranes identified in 2.1.1 not designed as single-failure-proof in 2.4.1, a comprehensive hazard evaluation should be provided which includes the following information:

ITEM 2.4.2.a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1.3.c. Impact areas should be identified by construction zones and elevations or by some other method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

RESPONSE: The requested information is provided in Attachment 1, Tables 3 through 10 and Figures 3 through 10. Layout drawings showing the location and surrounding equipment for the Auxiliary Feedwater Pump monorail system were included in the Authority's June 22, 1981 response.

ITEM 2.4.2.b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned considerations should be supplemented by the following specific information:

RESPONSE: This information is provided on Tables 3 through 10 in Attachment 1; see those items relying on hazard elimination Category c (right-hand column).

ITEM 2.4.2.b(1): For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

RESPONSE: CONTAINMENT POLAR CRANE

Systems evaluations were performed for a number of the regions inside containment. The approach and assumptions used to perform these evaluations are described below. The evaluation of each region for which systems evaluations were utilized is also provided.

Evaluations

Postulated load drops were evaluated using systems evaluations in Regions 3, 4, 5, 6, 7, 8, 9, and 10 (shown in Figures 3 through 10). These systems evaluations typically involved determining whether a load drop could cause loss of the primary core cooling mode at the time of the drop or, if the primary cooling mode could be lost, determining if backup cooling modes could be lost from the same drop.

Plant Conditions and Cooling Modes

The initial plant conditions for all systems evaluations was taken to be the "Cold Shutdown" or "Refueling Operation" condition as defined in the facility Technical Specifications. Heavy load handling operations typically don't begin until at least four days after shutdown. Cooling for both of these conditions is normally provided by the RHR loop of the Auxiliary Coolant System. Cases were considered for the situations of both RV head removed and RV head in place. Backup cooling modes, in the event of loss of RHR cooling, were identified from plant emergency procedures for loss of all RHR cooling. Several backup modes of cooling are possible. Not all backup modes were included in the evaluations, i.e., sufficient core cooling capability could be demonstrated without the need to include all possible modes identified in the procedures.

Event Trees

In order to identify which combinations of equipment failures could potentially result in a loss of core cooling capability, a set of event trees was developed. These event trees cover five cases that could be encountered for load drops inside containment.

They are:

Case 1 - Reactor Vessel Head Removed - Load Drop Does Not Result in an Unisolable Reactor Coolant System (RCS) Pipe Break

Case 2 - Reactor Vessel Head in Place - Load Drop Does Not Result in an Unisolable RCS Pipe Break

Case 3A - Reactor Vessel Head Removed - Load Drop Results in an Unisolable RCS Pipe Break

Case 3B - Reactor Vessel Head In Place - Load Drop Results in a Small Unisolable RCS Pipe Break

Case 3C - Reactor Vessel Head In Place - Load Drop Results in a Large Unisolable RCS Pipe Break

The event trees for these cases are displayed in Figures 11 through 15.

The event trees for the most part identify success and failure paths at the system level. For any particular load drop, the success or failure of a particular system was evaluated by determining whether any of the components required for operation of that system located inside containment could potentially be damaged by the load drop. If components could be damaged, then a determination as to whether loss of that system component could result in loss of the system function was made. Once the success or failure of the system of interest for each case was determined, the path on the event tree corresponding to the particular load drop event being postulated could be identified.

If the path for a particular drop scenario corresponded to successful maintenance of core cooling (indicated by the term "OK"), then no further evaluation of that drop scenario was required. If the path is one that culminates with an asterisk, then alternative core cooling modes were considered, i.e., cooling modes other than those included in the event trees.

Assumptions Regarding Loss of Equipment

The loss of equipment was evaluated in a conservative manner using the following assumptions:

(1) Except in cases where more localized damage could be justified, all equipment in a given region (at all elevations) was assumed to be lost. In the cases of Regions 6 and 7, the regions were subdivided for evaluation purposes into four subregions corresponding to each quadrant of the containment. This is justified for Region 7 because the effects of load drops below the operating deck, if there should be any, are expected to be localized, i.e., gross failure of large sections of the operating deck is not predicted. The deck was subdivided into four quadrants roughly corresponding to the NE, SE, SW, and NW regions of the floor. This was chosen because load handling and laydown areas are, for the most part, restricted to the four corner areas on either side of the two steam generator compartments.

Each of Regions 8 and 10 was subdivided into two subregions (North and South) for evaluation purposes.

(2) If RCS piping or connecting piping was in the region, an RCS pipe break was assumed to occur and its effect on core cooling evaluated assuming the simultaneous loss of other equipment in the region that could be impacted.

(3) In the case of Region 6, Reactor Coolant Pump Motor drops down through the corresponding openings in the operating deck were assumed to affect a significantly larger area below the deck than defined for the region at the 95' el.

(4) If instrumentation required to operate a component was within an impacted region, the component was assumed to be lost, e.g., if a steam generator level instrument was predicted to be lost, then the affected steam generator was assumed to be lost.

Steps in the Systems Approach

The steps used to perform systems evaluations of the potential effects of load drops inside the crane wall are outlined below:

(1) Select a region for consideration.

(2) Identify the equipment within the region that could be important to core cooling considerations.

(3) Identify the cases (event trees) that apply to that region.

(4) Assuming the equipment within the region is lost, determine whether the system function is lost.

(5) Using the results of (4), i.e., success or failure of the system, determine for each case which path on the event tree represents the load drop event being considered.

(6) If the path represents successful maintenance of core cooling for all cases, then no further analyses are required for the region.

(7) If the path represents a failure to demonstrate adequate core cooling with the core cooling modes included in the event tree, consider alternative cooling modes.

Systems Evaluation Results

Evaluation of Regions 3 and 4 - Areas Over RHR Heat Exchanger Compartment

There are two potential drop areas of interest that make up Regions 3 and 4. The first is the grating in the NE quadrant of containment. Although plant procedures prohibit movement of heavy loads over this region, there are no physical limitations that would prohibit Polar Crane travel over the region. In addition, the capacity of the grating is such that it can not be shown to withstand load drops of any significant weight or height of carry.

The second drop area is the head storage stand area. Structural analysis of a drop of the head on its storage stand predicts that scabbing from the underside of the 95' el. slab into the RHR Heat Exchanger compartment could occur.

In lieu of demonstrating that the intervening structures, i.e., the operating deck or the grating, can protect the equipment below from the effects of a load drop, a system evaluation was performed. The equipment identified in Table I was assumed lost as a result of a load drop on Regions 3 or 4. This equipment is located in the RHR Heat Exchanger Compartment. Also indicated in Table I is whether or not the equipment failures are predicted to result in loss of the system function. In some cases, remarks are included to explain the system failure conclusions.

The conclusions regarding the system failures were then used to enter the event trees applicable to the postulated load drop. The applicable event trees are those for Cases I and 2 (see Figures 11 and 12 in Attachment I). The Case 3 event trees are not applicable, because no unisolable RCS pipe breaks are predicted as a result of drops into this compartment.

For Case I, since RHR is predicted to fail, the primary cooling mode is assumed lost. However, the backup cooling mode, HPI and the Fan Cooler units, are not predicted to be lost as a result of the drop. Therefore, successful core cooling is maintained. The path on the event tree representing this success is Path 5.

For Case 2, again RHR is predicted to fail. However, none of the equipment in either of the two backup cooling modes displayed in the event tree is predicted to fail. Therefore, core cooling is maintained, as represented by Path 23.

TABLE I - SYSTEMS EVALUATION OF REGIONS 3 AND 4

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	None		No RCS Pipe Break	
RHR	Heat Exchangers (2) Inlet & Outlet Piping Inlet & Outlet Valves	Inlet-line 9, Outlet Lines 355 and 358	Yes	
CCW	Inlet & Outlet Piping to RHR Heat Exchangers and to Recirculation Pumps		Yes	
Recirculation portion of SI	Pumps (2) Discharge Piping to RHR Heat Exchanger Sump	Line 293	Yes	
PI Portion SI	Piping Valves	Line 351 from Acc. #1 to Loop 1 Cold Leg MOV 894A Chk 895A Line 355 (to 351) Chk 838A	No	Lines 754 and 753 can be isolated from remainder of SI by MOVS in Line 16 outside containment
Fan Coolers	None	Line 754 - HPI to Loops 3 CL	No	Located in the Annulus
Steam Generators	None	Line 753 - HPI to Loop 1 CL	No	
Feedwater	None	MOV 856E CHK 857L	No	
Atmosphere Steam Dump	None		No	

Note: Other systems unaffected by a drop in this region are Pressurizer/PZR pressure control, RC pumps, and CVCS.

Evaluation of Region 5 - Reactor Cavity

The load drops of interest for this region include the reactor head and the upper internals package. Load drops of either of these two components could potentially damage the reactor cavity to the extent that the inventory might be discharged to the containment floor below. The issue is whether or not equipment below required to maintain core cooling could be impacted or damaged from flooding.

There is no RHR or CCW equipment below the cavity floor. Therefore, the primary cooling mode is predicted to be unaffected by the postulated load impact. Further, as part of a previously performed ECCS performance analysis, a water level inside containment has been calculated based on a larger volume of water than could be discharged from the reactor cavity.

The water volume used for the ECCS analysis was over 420,000 gallons. During refueling, the reactor cavity is filled with approximately 342,000 gallons of borated water. The water level calculated for the ECCS analysis resulted in a water level up to about the 50' el. or approximately 4' above the floor level of 46'. The 50' el. water level was used as a bounding value for evaluating the possible flooding effects of a loss of reactor cavity integrity. The review indicated that there are no RHR or CCW valves affected by a 50' el. water level. Therefore, the primary mode of cooling in the cold condition is not predicted to be lost as a result of a postulated heavy load drop onto the reactor cavity floor.

Evaluation of Region 6 - Reactor Coolant Pumps

As indicated in Section 2.4.1.1, Region 6 was subdivided into four subregions: 6NE, 6SE, 6SW, and 6NW for evaluation purposes. The load drop of interest for these regions is a drop of a Reactor Coolant Pump Motor onto the pump. This could potentially occur from a height above the 95' elevation when raising or lowering a pump motor through the grating covered hatch at the operating deck. The Reactor Coolant Pump/Motor Mating surface is located at about the 70' el. Accordingly, a drop of a 32-ton pump motor onto the pump of over 25' could be postulated.

To evaluate the consequences of such pump motor drops, a systems evaluation was undertaken for each of the four regions. The equipment and associated systems identified in Table 2 were assumed lost as a result of a pump motor drop. The system failures identified were then used to enter the event trees. For the pump motor drop, all cases were considered. The results are presented in Table 3. As Table 3 indicates, core cooling can be maintained for all postulated drop scenarios.

Evaluation of Region 7 - Operating Deck - 95' el.

Region 7 was subdivided into four subregions: 7NE, 7SE, 7SW, and 7NW for evaluation purposes. The postulated load drop of principal interest is the RC Pump motor onto the operating deck.

Structural evaluations were performed to verify that drops onto the operating deck could not result in gross failure of large sections of the deck, i.e., localized failures only such as scabbing from the underside of the deck are anticipated.

TABLE 2 - SYSTEMS EVALUATION OF REGIONS 6NE, 6NW, 6SE, AND 6SW

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	RCS piping	Loop Cold Legs - one per region	RCS Pipe Break Possible	
RHR	Piping	Cold Leg Injection Lines - one per region. RHR Return Line in 6SE	Yes	
CCW	Piping	RCP Cooling Lines	No	The CCW to RCPs can be isolated from CCW loop to RHR Hx
Recirculation portion of SI	None		No	
[illegible]PI portion of SI	Piping	Injection Lines to Cold Legs - one per region. Injection Lines to Hot Legs in Regions 6NW, 6SW, 6SE	No	Broken injection lines can be isolated from remainder of SI system by MOVs located outside of Crane Wall or Missile Banier
Reactor Coolant Pump (RCP)	RCP and Associated Auxiliaries	One pump per region	Yes	Can affect one pump only
Pressurizer	Pressurizer Instrumentation, and Pressure Control	• PZR Heaters & Spray Lines • Level and Pressure Instruments	For 6NE drop only	Located in Loop 4. Therefore, assumed lost for 6NE

TABLE 2 - SYSTEMS EVALUATION OF REGIONS 6NE, 6NW, 6SE, AND 6SW

(continued)

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
Fan Coolers	None		No	Annulus
Steam Generators	None		No	
Feedwater	None		No	
Atmosphere Dump	None		No	
Chemical and Volume Control System - Charging and Letdown	Piping	Charging and Letdown Lines	For 6SW and 6SE only	Charging and letdown connections are in Loop 1. Piping travels in vicinity of 6SE RCP enroute to and from regenerative heat exchanger in SE quadrant.

TABLE 3
REGION 6 - EVENT TREE ASSESSMENT

SUBREGION	CASE	PATH	CONCLUSION
6NW	1	5	OK
	2	23	OK
	3A	1	OK
	3B	1	OK
	3C	1	OK
6SW	1	5	OK
	2	19	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
6SE	1	5	OK
	2	23	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
6NE	1	5	OK
	2	15	OK
	3A	1	OK
	3B	14	OK
	3C	1	OK

A systems evaluation very similar to that performed for Region 6 was performed for Region 7. The equipment and associated systems identified in Table 4 were assumed lost as a result of drops from above the operating deck. The system failures identified were then used to enter the event trees. All cases were considered. The results are presented in Table 5.

Table 5 indicates that if all equipment of interest in subregion 7NE is assumed lost, core cooling can not be accomplished by the cooling modes included in the event trees for Cases 2 and 3B. The principal difference in the analysis of Regions 6NE and 7NE is that the PORV piping is assumed lost for Region 7NE. The PORV Piping runs from the top of the Pressurizer out of the NW corner of the Pressurizer Compartment at about the 127' el.; down the Pressurizer Compartment wall to the 103' el. It then runs northwestward across the operating deck to a penetration in the floor just inside the crane wall. It proceeds downward at an angle through the crane wall to the Annulus region where it ultimately ties into the Pressurizer Relief Tank.

Prior to considering alternative cooling modes as the event trees suggest, the assumption that all equipment in Region 7NE is lost from a single load drop was evaluated. It was concluded that it was not reasonable to assume the loss of PORV piping in conjunction with loss of RHR. The RHR injection line that could be lost from a load drop is below the operating deck east and south of the RC Pump in Loop 4. The PORV discharge line is above the operating deck, north and west of this pump and the Pressurizer Compartment. It is extremely unlikely that a load drop that could damage one system could also damage the other. For this reason, it is concluded that core cooling can be maintained for Cases 2 and 3B.

<u>Evaluation of Region 8 - Steam Generators</u>

A load drop onto Region 8 could impact one or two steam generators. However, a breach of the steam generator shell at cold conditions would have no effect on the primary core cooling mode (RHR).

TABLE 4 - SYSTEMS EVALUATION OF REGIONS 7NE, 7NW, 7SE, AND 7SW

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	RCS Piping	RCS Cold and Hot Leg Piping	RCS Pipe Break Possible	
RHR	Piping	Cold leg injection - one per region. RHR return line in 7SE	Yes	
CCW	Piping	RCP cooling lines One loop of CCW to RHRAX in Crane Wall -75W	No	The CCW to RCPs can be isolated from the CCW to the RHR Hx One loop of CCW to RHR HX (outside Crane Wall) is available
HPI portion of SI	Piping	Injection lines to to Cold Legs - one per region - Injection lines to Hot Legs in 7NW, 7SW, 7SE	No	Injection lines can be isolated from remainder of SI system by MOV located outside Crane Wall
Steam Generator	Piping - steam/ blowdown		No	Affects one loop only
Recirculation portion of SI	None		No	
Reactor Coolant Pump	Auxiliaries	One pump per region	No	Can affect one pump only
Pressurizer	Instrumentation, Pressure Control and Pressure Relief	o PZR Heaters & Spray Lines o Level & Pressure Instruments o PORV Piping	For 7NE drops only	

TABLE 4 - SYSTEMS EVALUATION OF REGIONS 7NE, 7NW, 7SE, AND 7SW

(continued)

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
Fan Coolers	None		No	
Feedwater	Piping		No	Piping separated to steam generator
Atmosphere Dump	None		No	
Chemical & Volume Control System - Chg and Letdown	Piping		For 7SW and 7SE drops only	Charging and letdown are from Loop 1 in SW quadrant. Piping travels into SE quadrant to and from regenerative heat exchanger.

TABLE 5
REGION 7 - EVENT TREE ASSESSMENT

SUBREGION	CASE	PATH	CONCLUSION
7NW	1	5	OK
	2	23	OK
	3A	1	OK
	3B	1	OK
	3C	1	OK
7SW	1	5	OK
	2	19	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
7SE	1	5	OK
	2	19	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
7NE	1	5	OK
	2	17	Consider Alternative Cooling Modes
	3A	1	OK
	3B	16	Consider Alternative Cooling Modes
	3C	1	OK

Evaluation of Region 9 - Instrument Racks - NE Quadrant

A heavy load drop could potentially penetrate the grating over the instrument rack and valve access area in the NE quadrant of the operating deck.

The equipment and systems assumed to be lost are indicated in Table 6. This information was used to enter the event trees for Cases 1 and 2. Cases 3A, 3B, and 3C were not considered because no unisolable RCs pipe break is predicted.

For Case 1, successful core cooling is predicted. Path 5 represents the success path. For Case 2, successful core cooling is represented by Path 23.

Evaluation of Region 10 - Slabs Between Steam Generators

The load drop of interest for this region is the drop of a CRDM Missile Shield. The area between the steam generators is the laydown area for the four 23-ton missile shields (two on each side). Since the missile shields are only lifted when the reactor vessel head is in place, Cases 2 and 3A do not apply. Path 1 is applicable for Case 2, because no damage to RHR or CCW piping is predicted if the missile shield drop were to result in damage to equipment below the slabs. Further, the potential consequences of a postulated RCS pipe break from a drop are bounded by the RCS pipe break cases considered for Regions 6 and 7.

TABLE 6 - SYSTEMS EVALUATION OF REGION 9

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	None		No RCS Pipe Break	
RHR	Piping	Line 361 to Loop 4 cold leg injection line	Yes	
CCW	None		No	
HPI portion of SI	Piping & Valves	Line 16 - HPI to RCS MOV -856C; CK 8575 Line 846 - HPI to RCS MOV-8561C; CHK 857W	No	
Recirculation portion of SI	None		No	
Fan Coolers	None		No	
Instrumentation Racks	Steam Generator Level Indication		Yes	All level indication would be lost, however, steam generators would be available
	RC Pump Seal & Cool. Water Flow		No	Cooling flow information would be lost, but RC pumps still available
	RC Flow		No	Not useful
Instrument Sensing Lines	Pressurizer Press and Level Indicators		No	One channel of Pressure and Level is routed outside the Crane Wall.
Chemical and Volume Control System - Charging and Letdown	None		No	

Note: Other systems unaffected by this load drop include the pressurizer, RC pumps, Steam Generators and Feedwater, and Atmospheric Dump.

ITEM 2.4.2.b.(2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the bypassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

RESPONSE: Neither mechanical stops or electrical interlocks have been relied on.

ITEM 2.4.2.b.(3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance sequencing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the continued validity of such considerations.

RESPONSE: No load/target combinations have been eliminated on the basis of site specific considerations.

ITEM 2.4.2.c. For interactions not eliminated by the analysis of 2.4.2.b, above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: See response to 2.4.1.

ITEM 2.4.2.d. For interactions not eliminated in 2.4.2.b or 2.4.2.c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG 0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided:

RESPONSE: All handling systems and load impact regions have been evaluated.

ATTACHMENT I

TABLE 1

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 1 - REACTOR VESSEL (SEE FIGURE 1) 1A - REACTOR VESSEL HEAD REMOVED		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
RV HEAD (169 TONS)	Vessel flange is at 69' el. Fuel in core is at 56' el. Head and internals assumed dropped 1.5' after lift off.	Reactor Vessel - Vessel Integrity Considerations	e. Head drop analysis indicates stress is within code allowables
		Irradiated Fuel Assemblies in the core	e. Breach of fuel cladding not predicted for head drop.
REACTOR INTERNALS (67 TONS)		Irradiated Fuel Assemblies in the Core	e. Breach of fuel cladding not predicted for internals drop. Criticality not predicted assuming optimum uranium-water ratio from fuel crushing.
ISI TOOL (5 TONS)		Irradiated Fuel Assemblies in the Core	d. Likelihood of handling system failure for this load is extremely small.
CRANE LOAD BLOCK (4.5 TONS) REACTOR COOLANT PUMP MOTORS (32 TONS) CONCRETE HATCH COVER (7.5 TONS) PZR MISSILE SHIELD (7.6 TONS)		Reactor Vessel and Irradiated Fuel Assemblies in the Core	Procedures prohibit carrying any of these loads over the reactor cavity when the head is removed and irradiated fuel is in the vessel.

TABLE 1 (CONTINUED)

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 1B - REACTOR VESSEL REACTOR VESSEL HEAD IN PLACE (SEE FIGURE 1)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
CRDM MISSILE SHIELD BLOCKS (23 TONS) CRDM MISSILE SHIELD SUPPORT BEAMS RV HEAD STUD TENSIONERS CRANE LOAD BLOCK (4.5 TONS)	Impact area would be shield support beams at approximately the 95' el or the vessel head lifting rig, slightly below the 95' el.	Reactor Vessel - Vessel Integrity Considerations	e. Bounded by RV Head Drop Analysis in terms of load on vessel nozzles. The shield blocks (heaviest load) would impact the shield support beams and possibly head rig, if dropped. Expected that only a small amount of energy would be transferred to the nozzles, if any. Loss of RCS pressure boundary integrity at cold conditions from possible damage to CRDM housings as a result of a drop will have no effect on the capability to cool the core. Damage to housings would be expected to be very limited because of protection afforded by RV head lifting rig which is permanently in place.

TABLE 2

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 2 - ANNULUS REGION BETWEEN THE CRANE WALL AND THE CONTAINMENT LINER (SEE FIGURE 2)		
	ELEVATION	**SAFETY-RELATED EQUIPMENT**	**HAZARD ELIMINATION CATEGORY**
PZR MISSILE SHIELD (7.6 TONS) RV HEAD STUD TENSIONERS CONTAINMENT EQUIPMENT HATCH PLUG (25 tons)	Impact area would be grating or checkered plate at the 95' el. Equipment is at lower elevations.	Equipment required to maintain Long Term Cooling.	d. Likelihood of handling system failure for all heavy loads handled by the Auxiliary Hoist in the Annulus Region is extremely small

TABLE 3 CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA	REGION 3 - AREA OVER GRATING WITHIN CRANE WALL IN NORTHWEST QUADRANT OF CONTAINMENT (SEE FIGURE 3)		
LOADS	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Systems evaluation is independent of load considered.	Impact area is 95' el. Equipment is at lower elevations.	• RHR Heat Exchangers and associated piping • CCW piping to RHR Heat Exchangers and Recirculation Pumps • Recirculation pumps and sump • 1 of 4 HPI cold leg injection lines • 1 of 4 RHR cold leg injection lines	c. Evaluated loss of RHR cooling at cold conditions, both with head off and head in place. Core cooling maintained with equipment that is unaffected by a postulated load drop into the region. In addition, loads are prohibited from movement over this region when there is irradiated fuel in the reactor vessel.

TABLE 4

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 4 - RV HEAD STORAGE AREA (SEE FIGURE 4)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Systems evaluation is independent of load considered.	Impact area is 95' el. Equipment is at lower elevations.	• RHR Heat Exchangers and associated piping • CCW Piping to RHR Heat Exchangers and recirculation pumps • 1 of 4 HPI cold leg injection lines • 1 of 4 RHR cold leg injection lines	See Discussion for Region 3

TABLE 5

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 5 - REFUELING CANAL (SEE FIGURE 5)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Systems evaluation is independent of load considered.	Impact area is bottom of refueling canal at approximately 69' el. in west end and 60' el. in east end.	Possible Equipment Required to Maintain Long Term Cooling - Concern is leakage from pool to levels below and possible resultant flooding damage to equipment.	e. Volume of water in refueling canal is less than volume of water that could be dumped to the containment floor during a large LOCA. Flooding of safety-related components was previously evaluated for LOCA - modifications made to assure operability of all components. Checked for additional components associated with normal long term cooling mode, i.e. RHR. All are above LOCA water level. In addition, most of the loads listed are prohibited from movement over the refueling canal by procedure.

TABLE 6

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 6 - AREA AROUND REACTOR COOLANT PUMPS - FOUR SEPARATE REGIONS; ONE IN EACH QUADRANT OF CONTAINMENT (FIGURE 6)		
	ELEVATION	**SAFETY-RELATED EQUIPMENT**	**HAZARD ELIMINATION CATEGORY**
SYSTEMS-EVALUATION IS INDEPENDENT OF LOAD CONSIDERED	Impact area is RCP motor/pump connection at 69' el. Opening for motor removal is at 95' el.	For Each of Four Regions • 1 of 4 RHR Cold Leg Injection Lines • RHR Return Line From Loop 2 Hot Leg (SE Quardrant Only) • CCW Piping to RC Pump • Reactor Coolant Pump • Pressurizer Spray, Heaters and Instruments (NE Quadrant Only)	c. There is adequate separation of equipment to assure that core cooling can be maintained in the event of loss of the primary cooling mode (RHR) and/or a RCS pipe break at cold conditions.

TABLE 7

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 7 - OPERATING DECK INSIDE CRANE WALL - FOUR SEPARATE REGIONS: ONE IN EACH QUADRANT OF CONTAINMENT (SEE FIGURE 7)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is at 95' el. Equipment is at lower elevations.	Same equipment as identified for Region 6 except for following additions: • Damage to charging 2nd letdown piping can occur in both SW and SE Quadrants • Damage to a Steam Generator and its associated piping could occur • Damage to PORV piping to the Pressurizer Relief Tank could occur in the NE Quadrant (piping exposed above 95' el. only).	c. There is adequate separation of equipment to assure that core cooling can be maintained in the event of loss of the primary cooling mode (RHR) or a RCS pipe break at cold conditions.

TABLE 8

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 8 - STEAM GENERATORS (SEE FIGURE 8)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is top of steam generator shell at approximately 125' el.	STEAM GENERATORS	e. Consequences of Steam Generator shell rupture from a load drop have no effect on fuel in the core or core cooling capability at cold conditions.

TABLE 9 CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 9 - GRATING OVER INSTRUMENT RACKS AND VALVE ACCESS AREA - NE QUADRANT OF OPERATING DECK (SEE FIGURE 9)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is grating at the 95' el. Instrument Racks and valve access located at 68' el.	• 1 of 4 RHR Cold Leg Injection Lines • Steam Generator Level Indication • Pressurizer Pressure and Level Indiaation	c. There is adequate equipment separation to assure that core cooling capability is maintained in the event of loss of the primary cooling mode (RHR).

TABLE 10

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 10 - AREA BETWEEN STEAM GENERATORS - SLABS - BOTH NORTH AND SOUTH OF REFUELING CANAL WITHIN STEAM GENERATOR ENCLOSURES (SEE FIGURE 10)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is 95' el. Equipment located at lower elevations.	• RCS piping	c. & e. No damage to equipment for primary core cooling mode (RHR) predicted. RCS pipebreak consequences bounded by Region 6 and 7 evaluations.

LOCATION	AUXILIARY FEEDWATER PUMP BUILDING		
IMPACT AREA / LOADS	• MOTOR DRIVEN FEEDWATER PUMP HOUSING		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
FEEDWATER PUMP MOTOR (2860 lbs)		Motor driven pump that is not out for service and the steam driven pump. Potential effects are: (1) a drop of the motor on the pump housing for the pump that is out of service could potentially lead to flooding damage of the motor driven pump that is operable or the steam driven pump, or (2) a drop of the motor on the operable motor driven pump could take out the operable motor driven pump and lead to flooding damage of the steam driven feedwater pump.	d. Likelihood of load drop is extremely small.

TABLE 12

CRANE: AUXILIARY FEEDWATER PUMP BUILDING MONORAIL (PUMP 22)

LOCATION	AUXILIARY FEEDWATER PUMP BUILDING		
IMPACT AREA	• STEAM DRIVEN AUXILIARY FEEDWATER PUMP • STEAM DRIVEN PUMP SUCTION OR DISCHARGE PIPING		
LOADS	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
FEEDWATER PUMP TURBINE (1.5 TONS) TURBINE MISSILE SHIELD (1.9 TONS)		Motor driven feedwater pumps. The potential effect is that damage to the pump housing or suction piping could lead to flooding damage to motor driven feedwater pumps or electrical equipment if the suction line is not isolated. Plant procedures require isolation of the suction line prior to making heavy lifts over the pump.	d. Likelihood of handling system failure for these loads is extremely small.

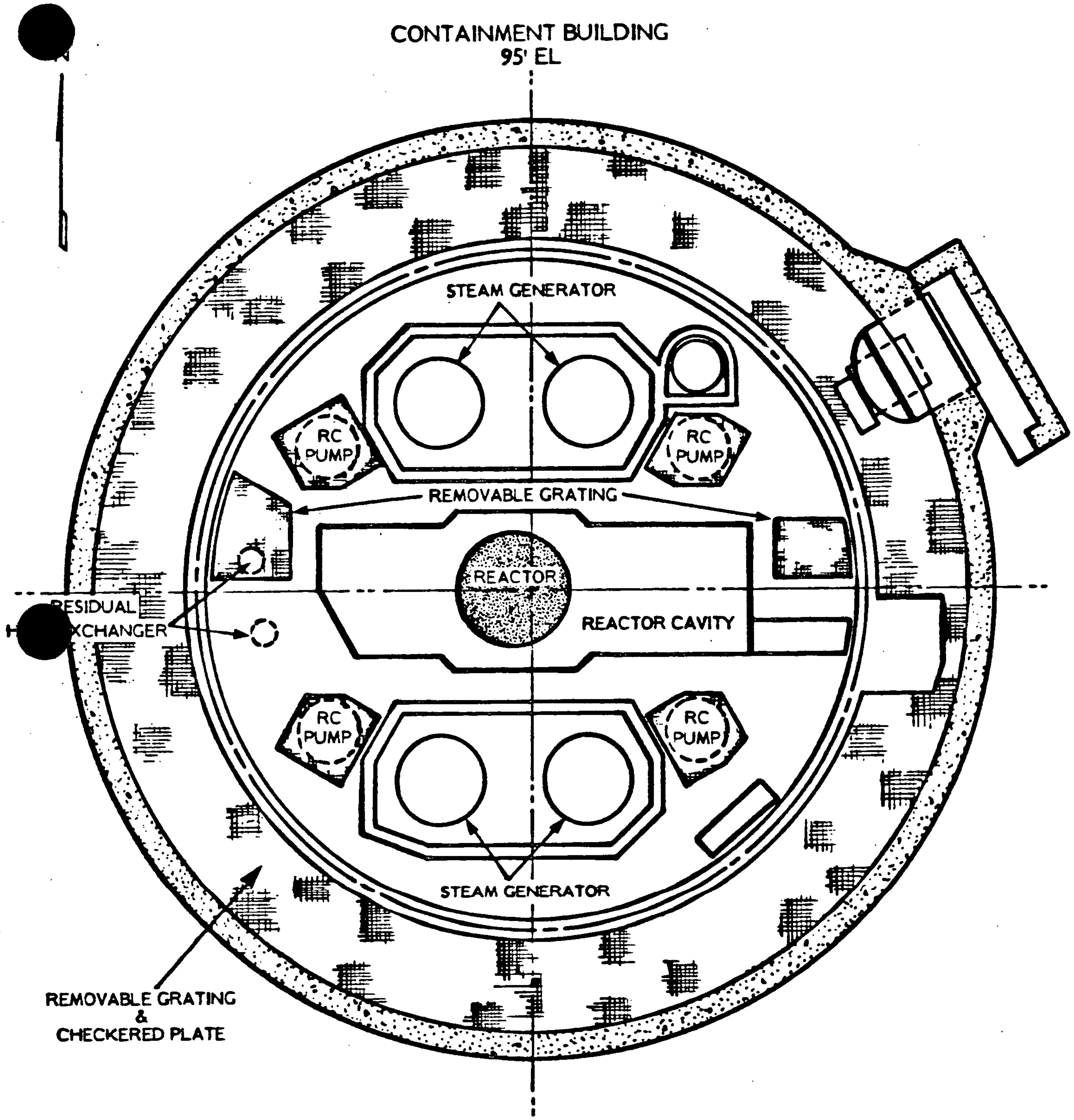


FIGURE I

REGION I - REACTOR VESSEL

CONTAINMENT BUILDING
95' EL

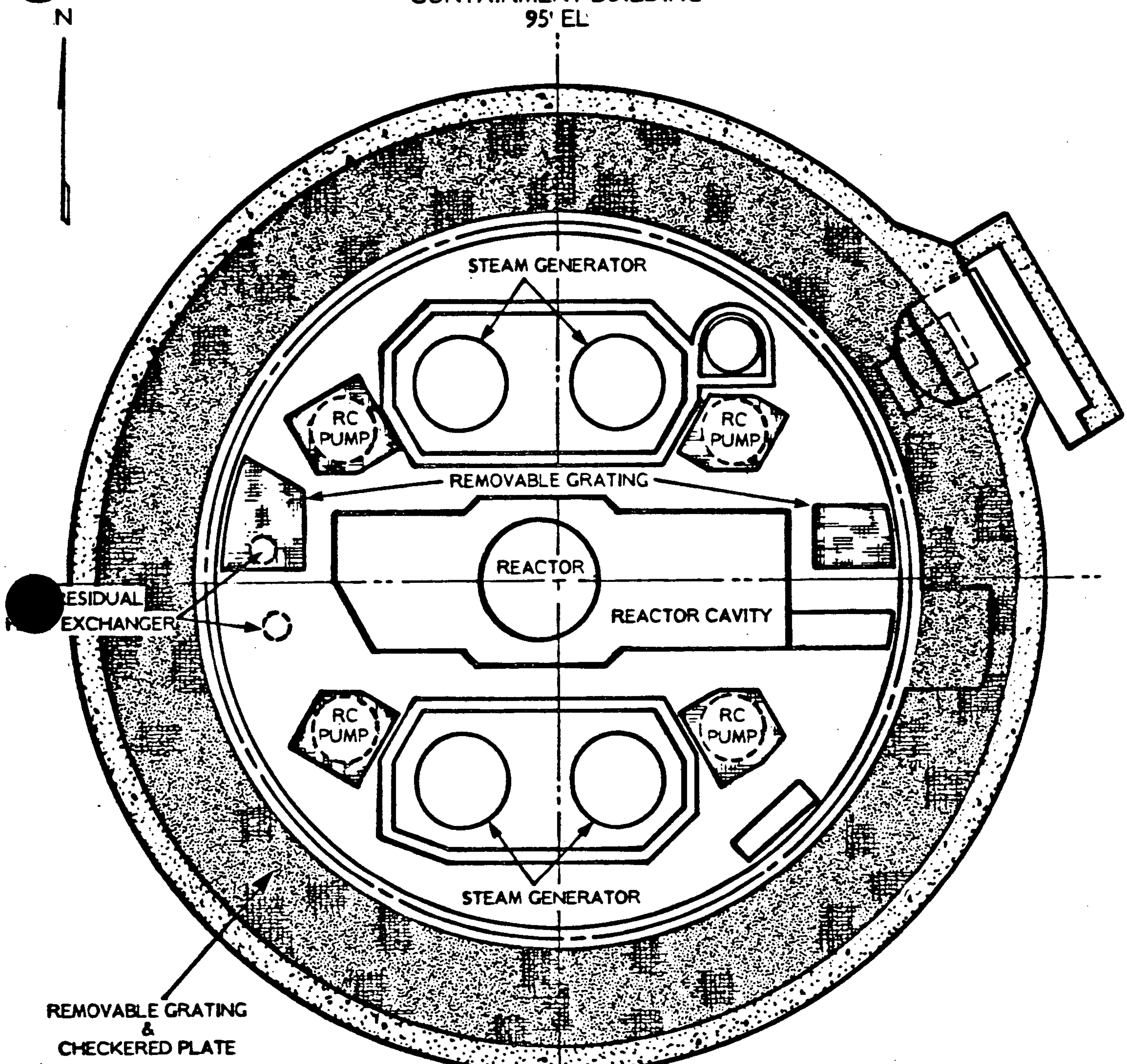


FIGURE 2
REGION 2 - ANNULUS

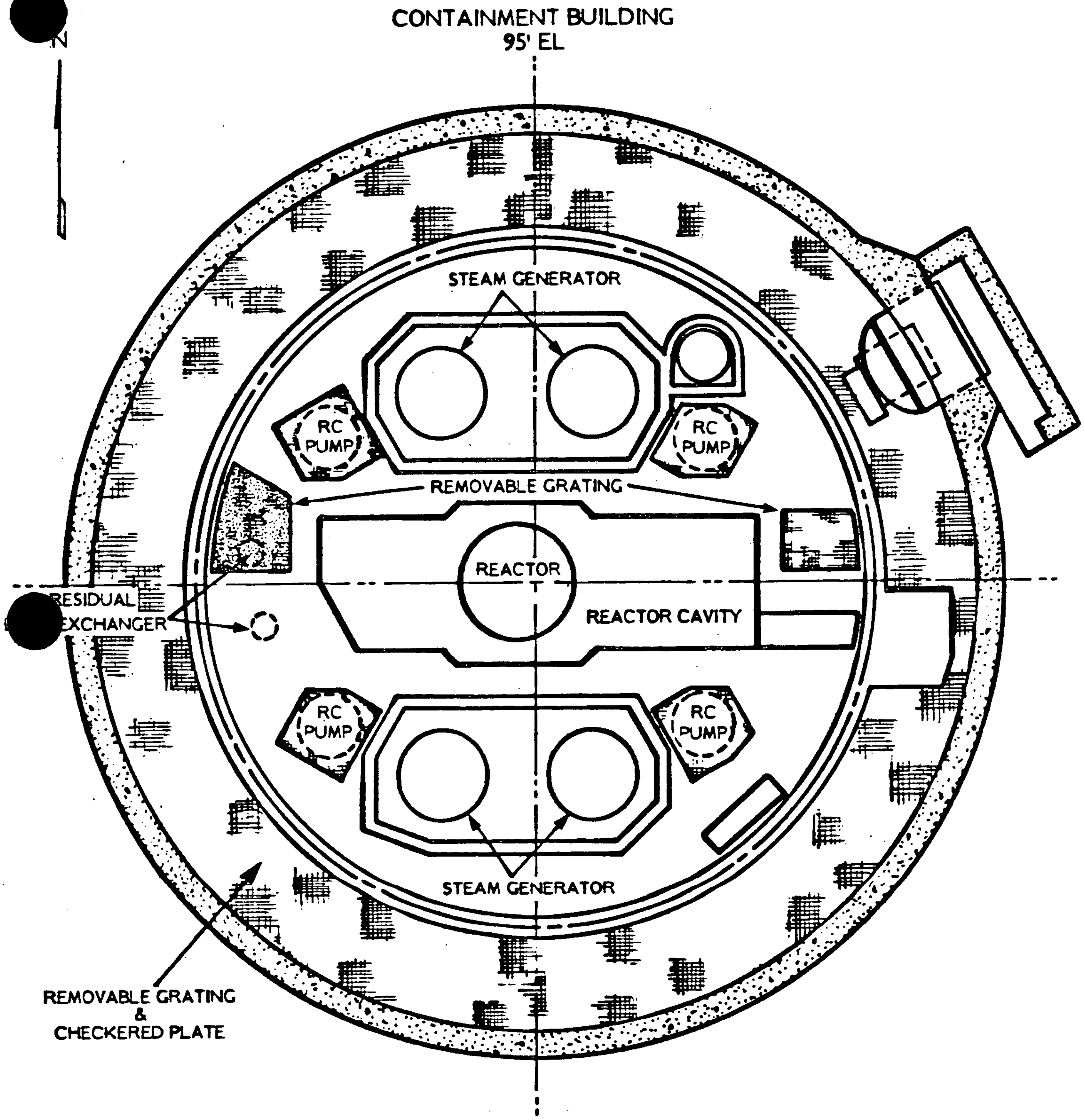


FIGURE 3

REGION 3 - GRATING OVER RHR HEAT EXCHANGERS

CONTAINMENT BUILDING
95' EL

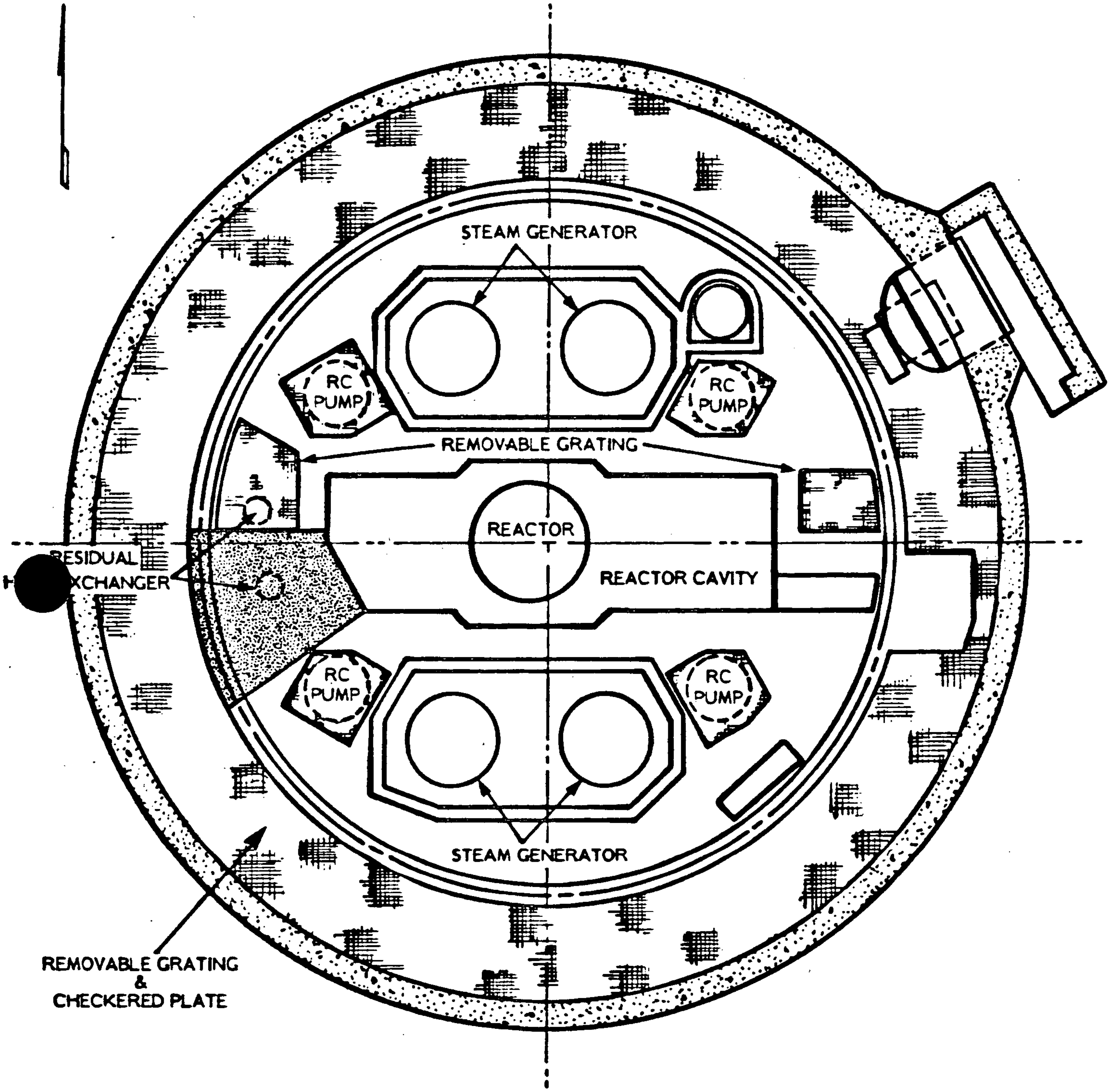


FIGURE 4

REGION 4 - OPERATING DECK HEAD STORAGE STAND OVER RHR HEAT EXCHANGERS

CONTAINMENT BUILDING
95' EL

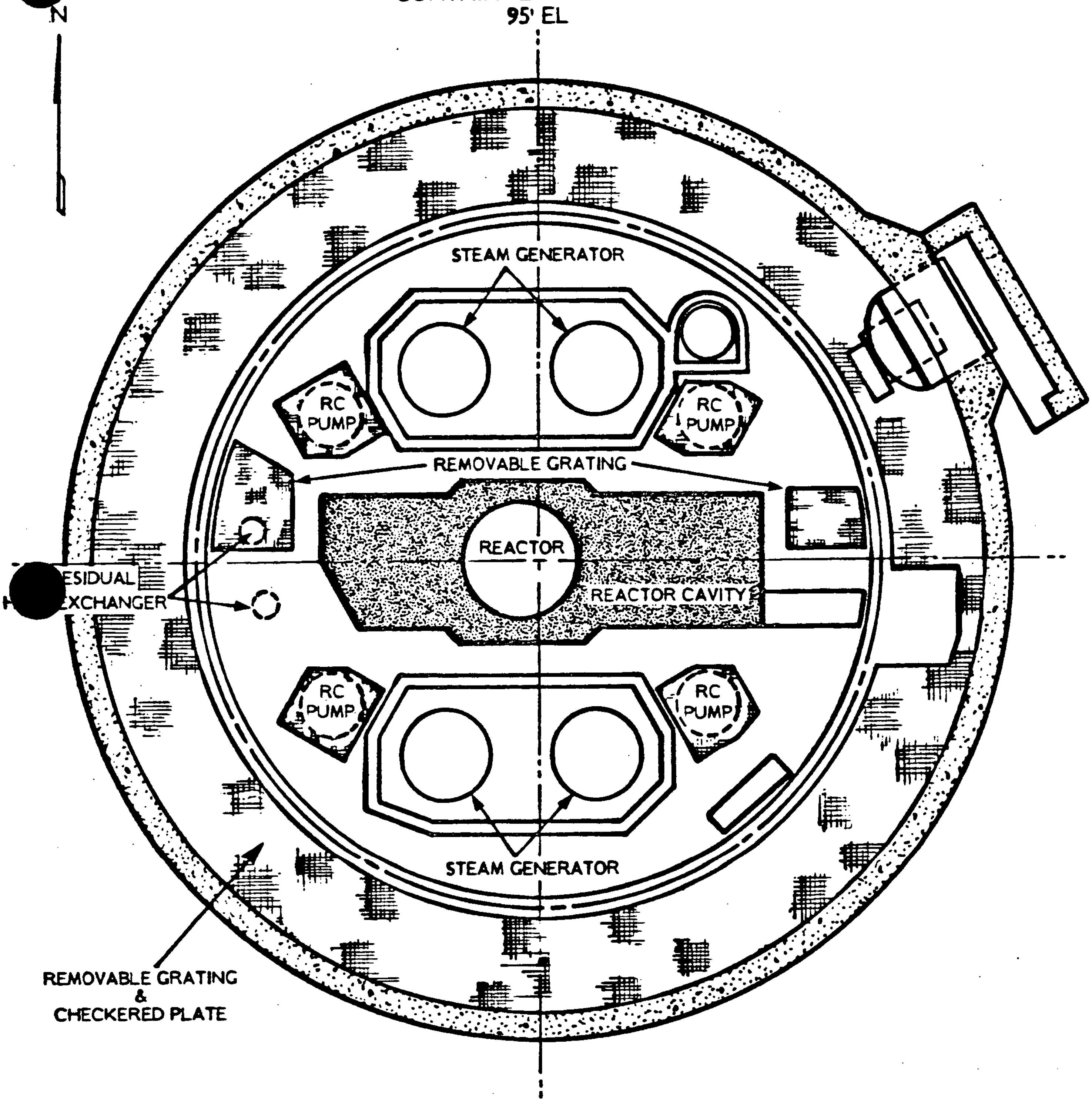


FIGURE 5

REGION 5 - REACTOR CAVITY

CONTAINMENT BUILDING
95' EL

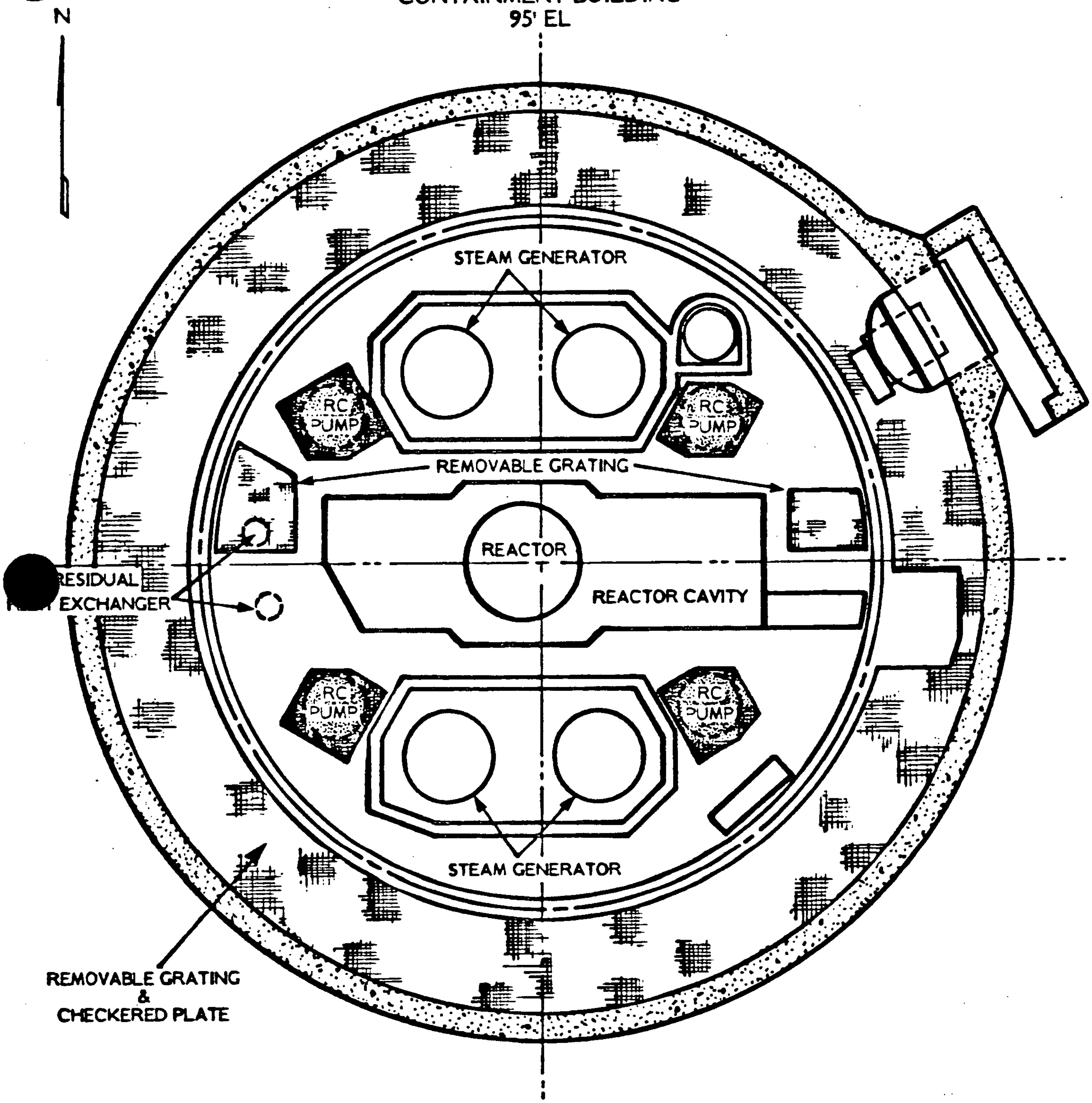


FIGURE 6

REGION 6 - REACTOR COOLANT PUMP AREA

CONTAINMENT BUILDING
95' EL

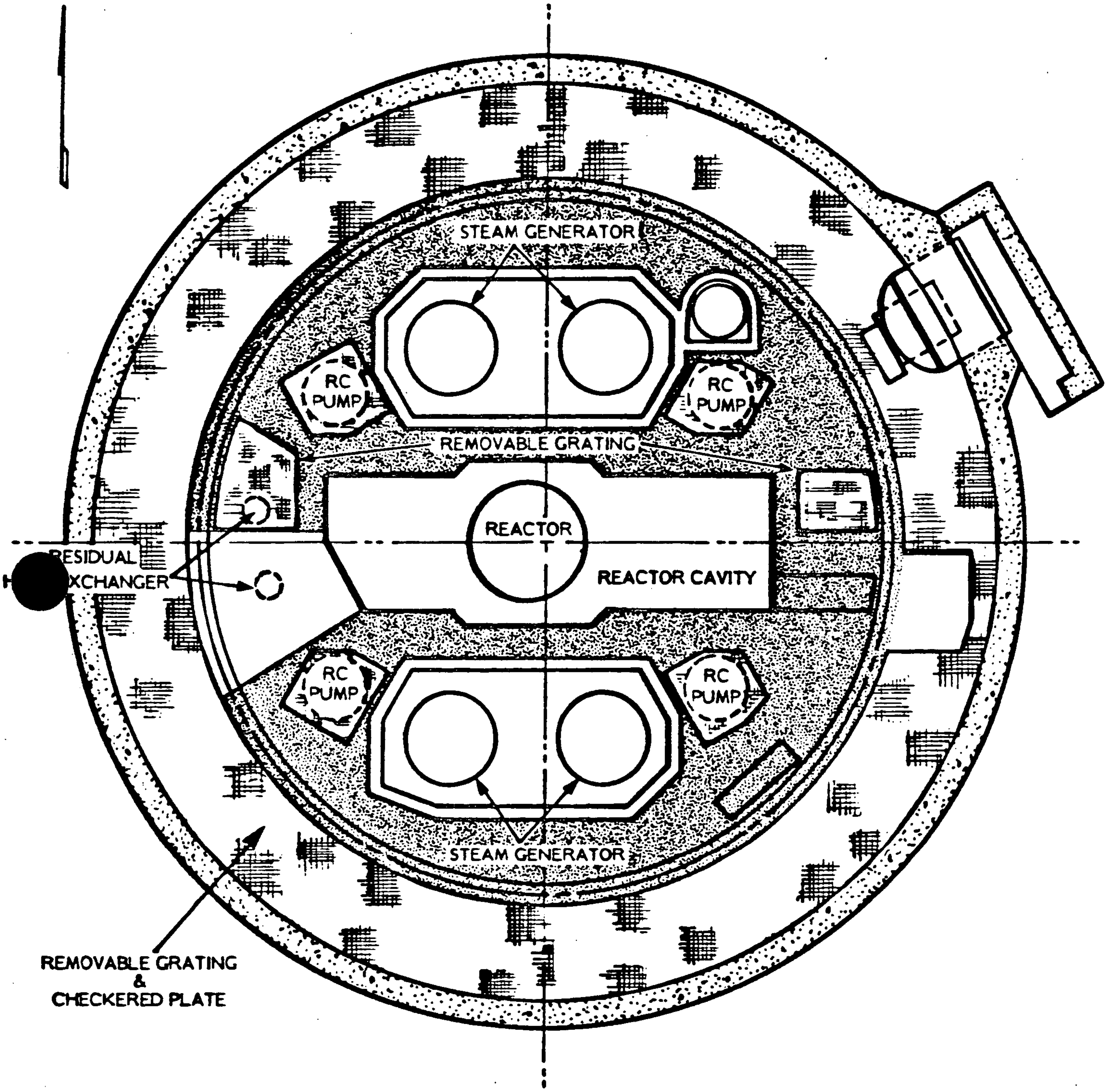


FIGURE 7
REGION 7 - OPERATING DECK INSIDE CRANE WALL

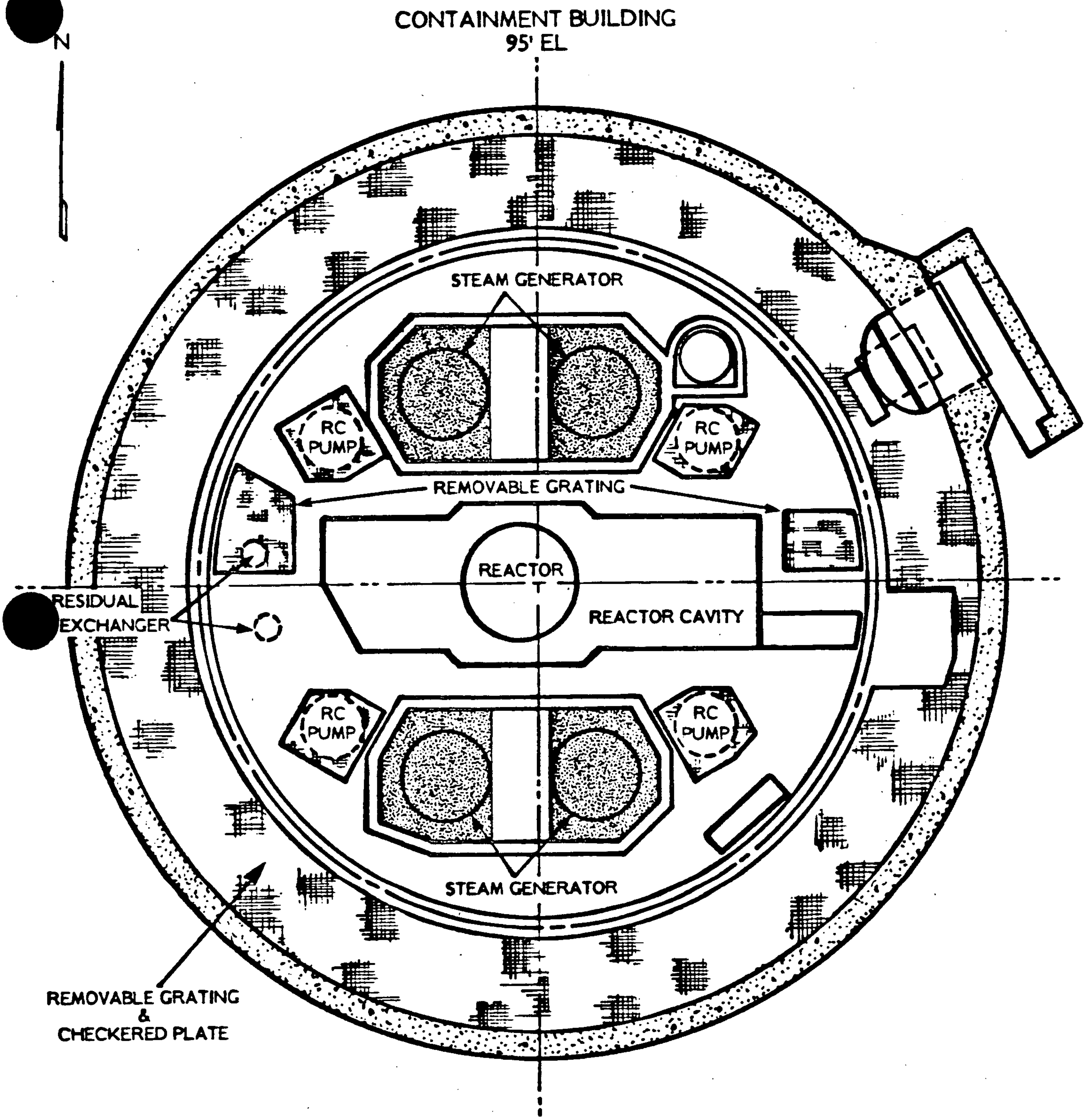


FIGURE 8

REGION 8 - STEAM GENERATORS

CONTAINMENT BUILDING
95' EL

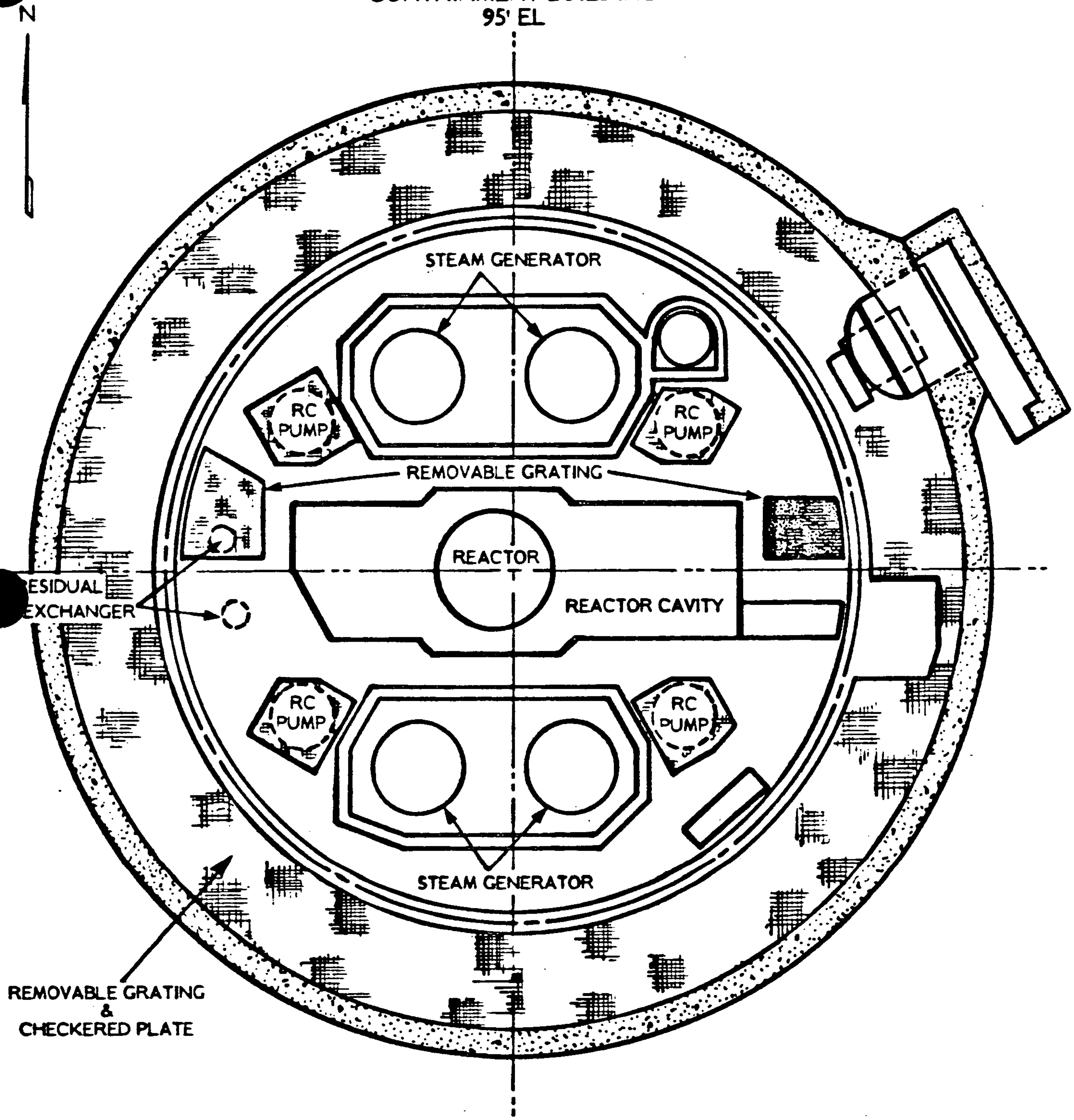


FIGURE 9

REGION 9 - GRATING OVER INSTRUMENT RACKS AND VALVE ACCESS AREA

CONTAINMENT BUILDING
95' EL

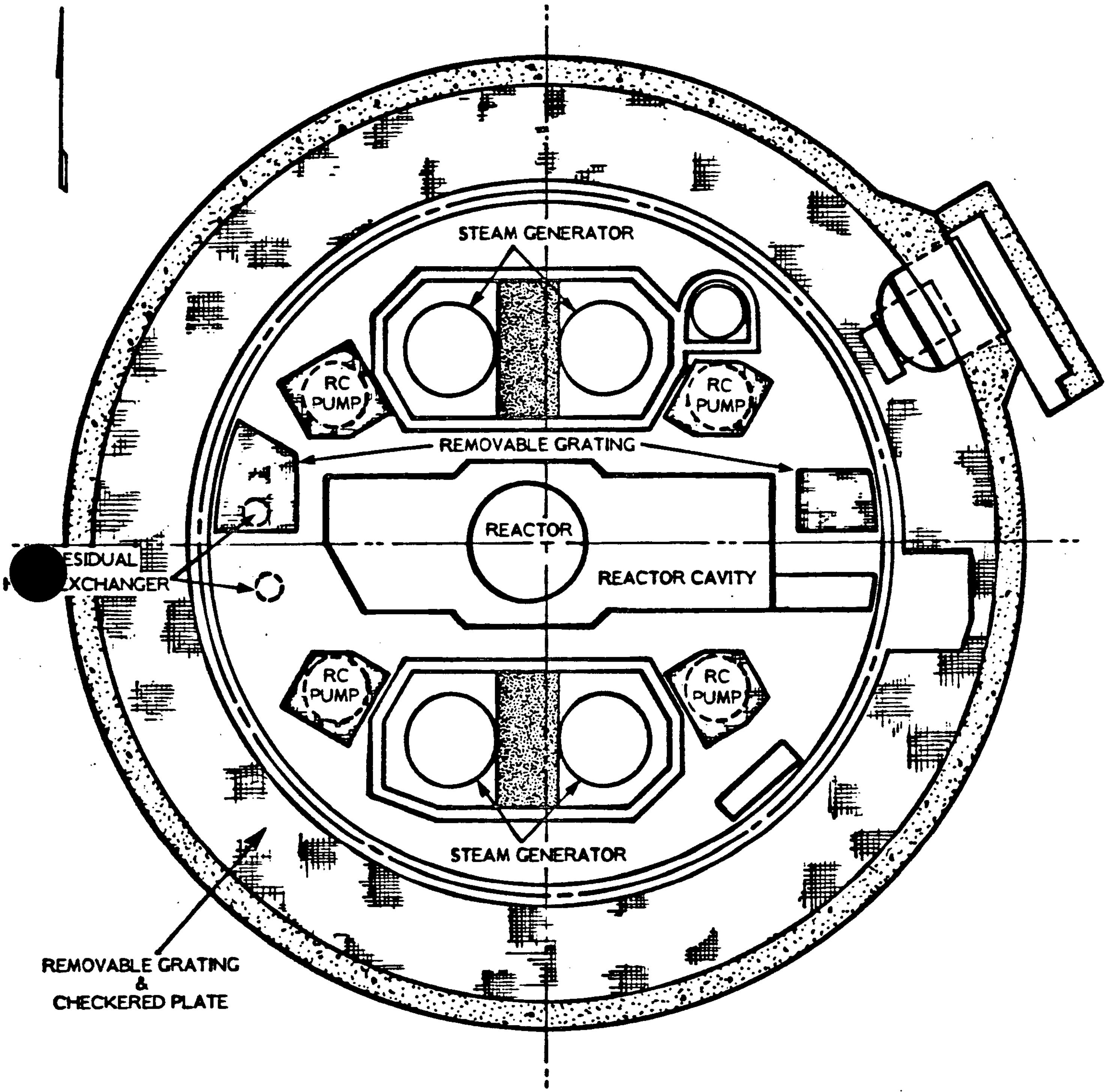


FIGURE 10

REGION 10 - SLABS BETWEEN STEAM GENERATORS (95' EL)

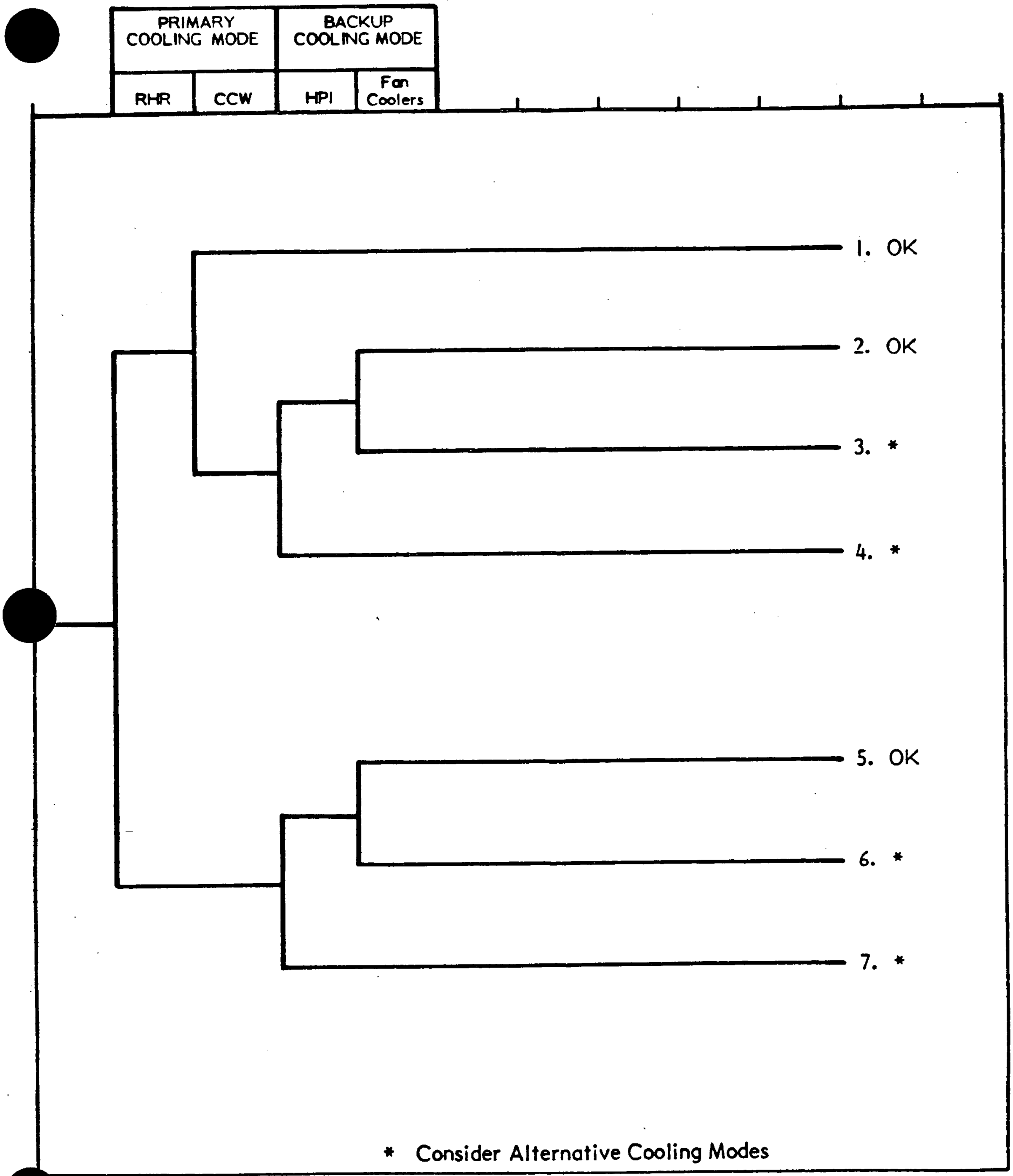


FIGURE 11
CASE I - RV HEAD REMOVED - NO RCS BREAK

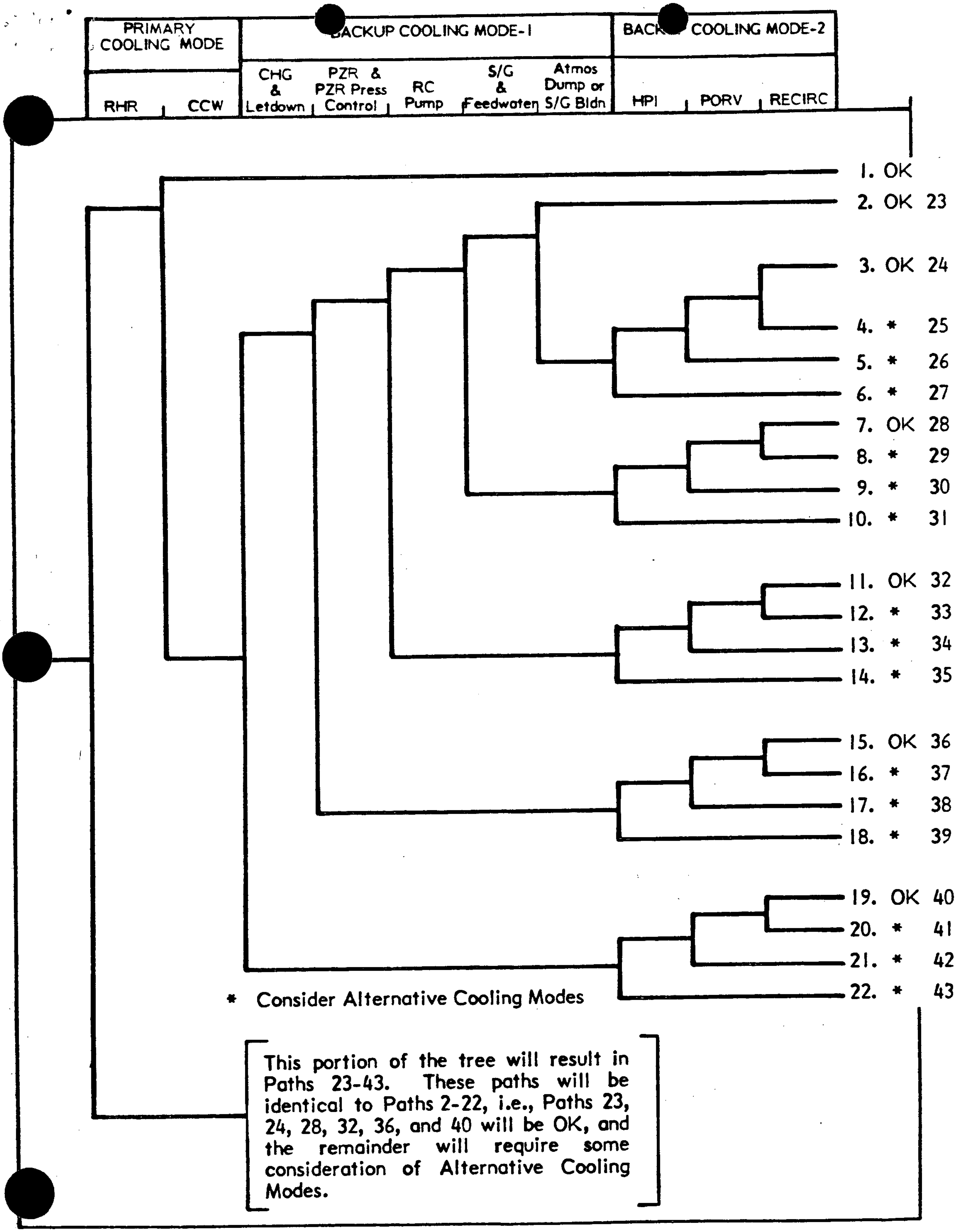


FIGURE 12
CASE 2 - RV HEAD IN PLACE - NO RCS BREAK

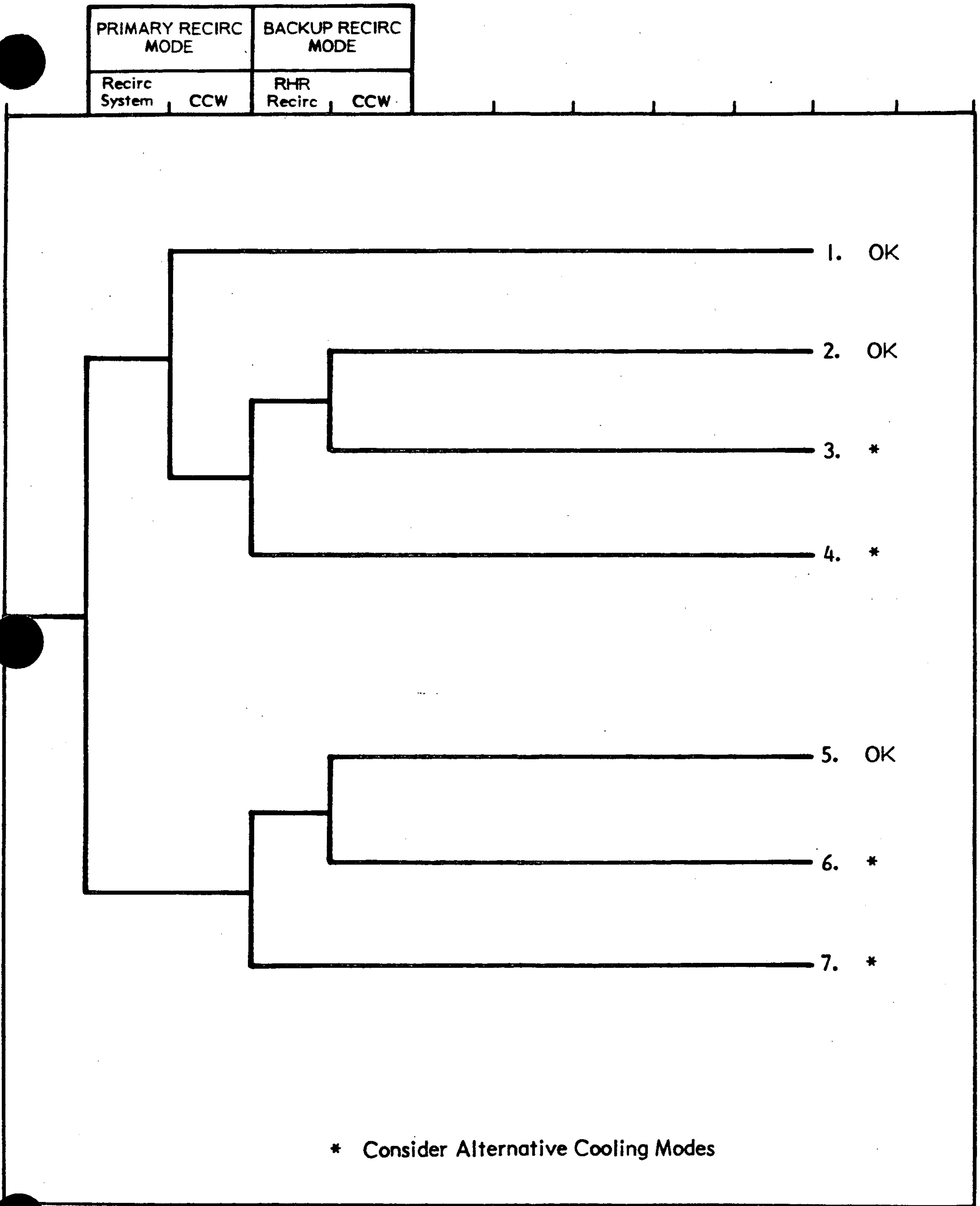


FIGURE 13

CASE 3A - RV HEAD REMOVED - RCS BREAK

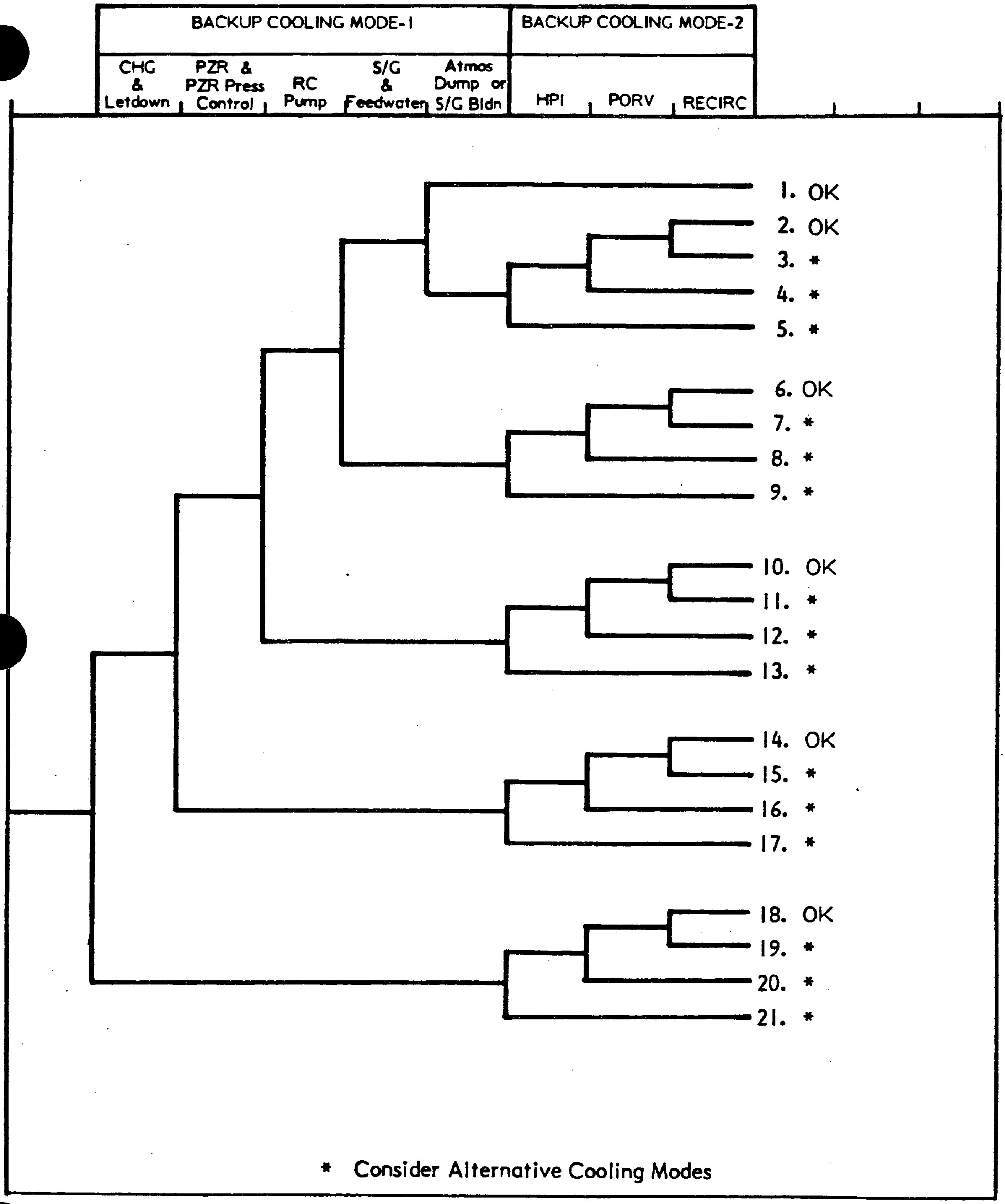
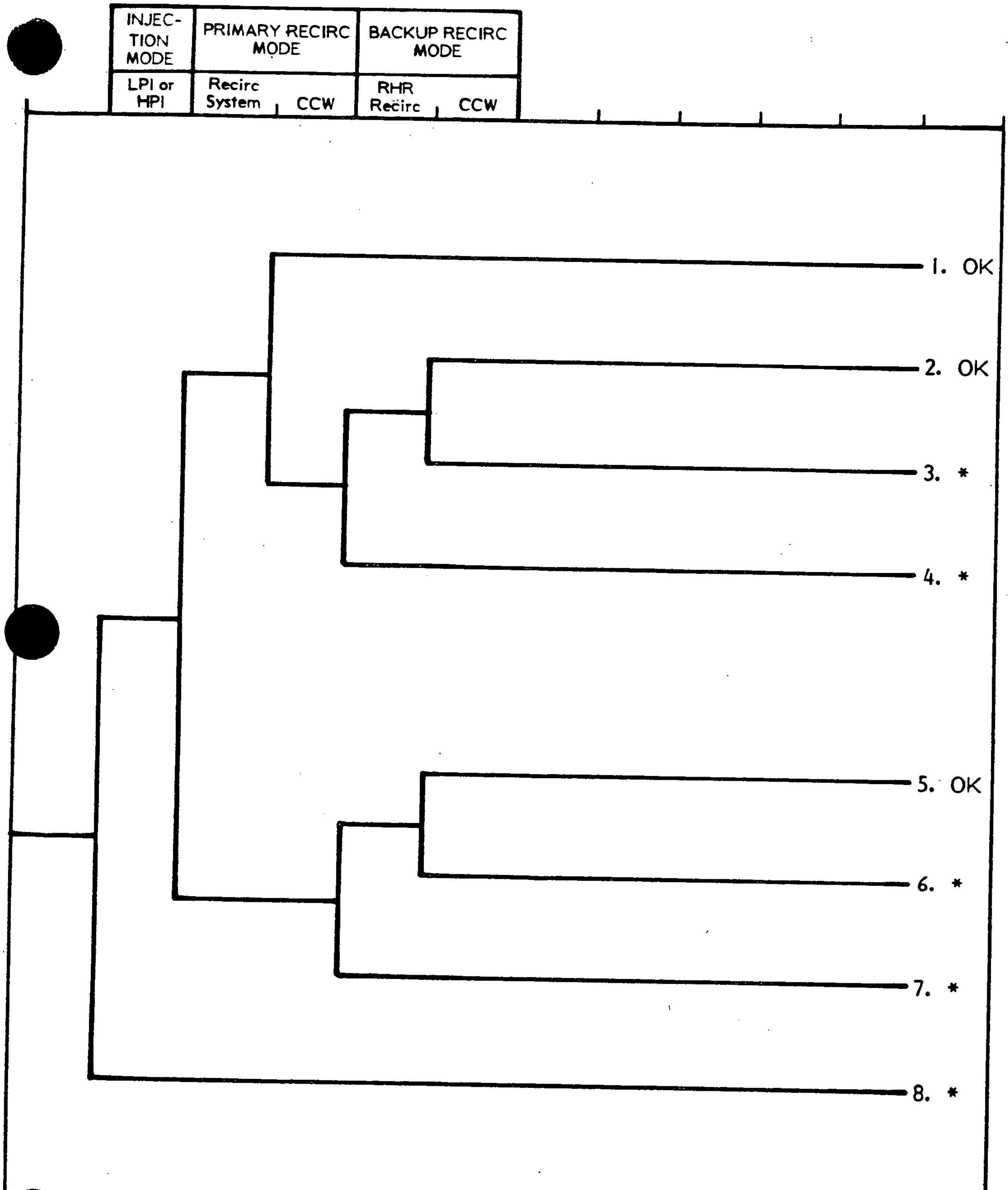


FIGURE 14

CASE 3B - RV HEAD IN PLACE - SMALL RCS BREAK



* Consider Alternative Cooling Modes

FIGURE 15
CASE 3C - RV HEAD IN PLACE - LARGE RCS BREAK