Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

Docket 50-508

October 22, 1982
G03-82-1085

Ms. Janis D. Kerrigan,
Chief Licensing Branch #3 (Acting)
Division of Licensing
US Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: WASHINGTON NUCLEAR PROJECT 3
RESPONSES TO NRC ACCEPTANCE REVIEW
QUESTIONS

References: a) Letter D. G. Eisenhut to R. L. Ferguson, dated 8/20/82

Reference a) transmitted a set of questions generated during the NRCs acceptance review of the WNP-3 Operating License Application. These questions addressed both the WNP-3 Final Safety Analysis Report (FSAR) and the Environmental Report (ER). Responses to the questions associated with the FSAR are enclosed in Attachment 1. Responses to the questions associated with the ER will be transmitted under separate cover.

Several of these questions address items which are still being evaluated by the Supply System. In those cases where our evaluation is not yet complete, we have provided a schedule detailing when we will be able to present the results for review.

In those cases where it is considered necessary or desirable to amend the FSAR due to our responses, we have provided marked up FSAR pages which show, in detail, what our intentions are. It is hoped that this will allow early review by the NRC prior to actual incorporation of the revised information into the FSAR.

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WASHINGTON NUCLEAR PROJECT 3 RESPONSES TO NRC ACCEPTANCE REVIEW QUESTIONS

If you require additional information or clarification, the Supply System point of contact for this action is Mr. K. W. Cook, Licensing Project Manager (206/482-4428 ext. 5436)

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G. D. Bouchey, Manager
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WNP-3 Files - Richland

Attachment 1-1
2266A

Question No.

100.1 (1.8.3 and other sections)

Table 1.8-3 addresses conformance and exceptions to the Standard Review Plant (SRP), NUREG-75/087. You have stated that WNP-3 will be reviewed and evaluated relative to NUREG-0800. This review and evaluation should conform to the following:

1. Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982, shall include an evaluation of the facility against the Standard Review Plant (SRP) in effect on May 17, 1982, or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

2. The evaluation shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.

3. The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.

In addition, a schedule should be provided for completion of the review and evaluation.

Response

The Supply System has initiated a program to provide the required Standard Review Plan (SRP) compliance review in accordance with recent modifications to the Commission regulations. This review and evaluation process is being conducted based upon the acceptance criteria contained in NUREG-0800. The initial portion of this program was culminated by changes in the WNP-3 FSAR provided in Amendment 1. The changes are summarized below:

1. Insertion of revised material in Subsection 1.8.3 and Table 1.8-3 (consistent with NUREG-0800 review rather than NUREG-75/087).

Question No.

100.1 (1.8.3 and other sections) (contd.)	The revised Table 1.8-3 provides a listing of SRP acceptance criteria and indicates those for which the WNP-3 design is in conformance and those for which differences exist. In addition, a schedule for completion of the required evaluations is included.

In several instances the compliance review was not completed or a schedule for completion was not available prior to submittal of Amendment 1. These areas are expected to be completed and submitted in the next FSAR Amendment.

The evaluation process to be utilized is described in Subsection 1.8.3 of the FSAR as modified in Amendment 1.

Attachment 1-3
2266A

Question No.

100.2 The WNP-3 FSAR contains numerous references to the CESSAR-FSAR but does not specifically address the Safety Evaluation Report (NUREG-0852) for the CESSAR-FSAR. This Safety Evaluation Report (SER) imposes requirements on applicants utilizing the CESSAR-FSAR and identifies open items. The applicant should provide a plan identifying and addressing the interface between NUREG-0852 and the WNP-3 FSAR to assure that the SER requirements are addressed in the WNP-3 FSAR and are, or will be, incorporated in the design and operation of WNP-3. Provide a schedule for implementation of this plan. Types of information to be addressed in this plan are as follows.

1. Open items identified in the SER. This should include both items identified for final resolution by the licensee as well as those for Combustion Engineering resolution. Although the latter items may not require specific licensee action at this time, licensee tracking is necessary to insure that any resolution is incorporated into the WNP-3 design.

2. Specific license conditions and technical specifications which are imposed by NRC on applicants referencing CESSAR.

3. Interface requirements identified by NRC which differ from, or are in addition to, those identified in CESSAR.

The following are specific examples of items from the CESSAR-SER which should be addressed.

1. The SER identifies, in Section 15.3.9, specific items which must be implemented by the licensee as an interim fix for anticipated transients without scram until rulemaking and formulation of final requirements are completed. These items are not discussed in Section 15.8 of the WNP-3 FSAR.

2. The SER requires, in Section 7.3.2, that control logic be configured such that an ESFAS signal will override MSIS. This is not consistent with the statement in Section 7.3.1 of the WNP-3 FSAR which states that "there are no overrides on any MSIS actuated devices with the exception of the atmospheric dump valves".

3. The SER requires specific plant technical specifications in Section 5.2.2 which should be addressed in the WNP-3 FSAR.

Response

The Supply System is in the process of developing a plan for identifying and addressing the interface between NUREG-0852 and the WNP-3 FSAR. This plan will be provided by December 1982.

Attachment 1-4
2266A

Question No.

100.3 (All) Describe your system for monitoring updates to the CESSAR-FSAR and SER and incorporating these updates into the WNP-3 systems, operations and documentation.

Response

Monitoring updates to the CESSAR-FSAR to assure the material is properly incorporated involves reviews by Combustion Engineering (CE), Ebasco Services (A/E) and the Supply System. For each amendment to CESSAR-FSAR, CE reviews the material for applicability to WNP-3 and provides their assessment to the Supply System and Ebasco for review and concurrence. For those areas where the update material is not applicable to WNP-3 a change notice is prepared providing WNP-3 specific input to replace the CESSAR-FSAR update. The change notice is reviewed in accordance with the FSAR amendment review procedures and incorporated into the WNP-3 FSAR in a subsequent amendment. Should the WNP-3 design be modified to incorporate CESSAR design changes the WNP-3 FSAR will be amended, at a later date, to reference the CESSAR FSAR material.

As discussed in NRC Question 100.2, a review of the CESSAR-FSAR Safety Evaluation Report (SER) has been initiated. Updates to the SER are expected to be in the form of supplements to NUREG-0852. These will be reviewed in the same manner as CESSAR-FSAR updates to determine applicability to WNP-3 and provide amendments to the WNP-3 FSAR where required.

Attachment 1-5
2266A

Question No:

100.4 Correct the following deficiencies in the General Information:

(General Information)

a. 10CFR50.33(d)(2)(-i) requires "names, addresses and citizenship" of the Directors. The tendered application gives the names and addresses but the citizenship is not included.

b. 10CFR50.33(d)(2)(iii) requires a statement as to whether the corporation is "owned, controlled or dominated by an alien, a foreign corporation, or foreign government and if so, give details." The tendered application is silent on this requirement.

Response

a) All Directors and Principal Officers of the Supply System are citizens of the United States, as are the Directors and Principal Officers of Pacific Power and Light, Portland General Electric, Puget Sound Power and Light Company, and Washington Water Power Company.

b) Neither the Supply System nor any of the private owners of WNP-3 are owned, controlled, or dominated by an alien, a foreign corporation or a foreign government. The private owners of WNP-3 are:

Pacific Power and Light
Portland General Electric
Puget Sound Power and Light Company
Washington Water Power Company

Attachment 1-6
2266A

Question No.

210.1 (3.8.2.4)	Provide a discussion of the extent of compliance with Subsection NE of the ASME Code, Section III, Division I for the procedures used in the design and analysis of the steel containment.

Response

The steel containment vessel is designed and constructed in full compliance with Subsection NE of the ASME Code, Section III, Division I and will be ASME Code stamped as stated in Subsection 3.8.2.2.3 of the FSAR.

Attachment 1-7
2266A

Question No.

210.2 (3.9.1.2) Regulatory Guide 1.70 states that the description of the computer programs used in dynamic and static analysis should include the extent of the programs application, and the design control measures employed to demonstrate the applicability and validity of each program. Reference or provide this information.

Response

The following tabulation provides a listing of those computer programs used in dynamic and static analysis of WNP-3 and reference to the appropriate subsection providing program application and the design control measures employed to demonstrate the applicability and validity of each program.

Computer Code	Application Reference	Validity Reference
PIPESTRESS 2010	3.9.1.2.2.1	3.9.1.2.2.1
PLAST 2267	3.9.1.2.2.2(a)	3.6B.2
CALPLOT F	3.9.1.2.2.2(a)	3.6B.1
CALPLOT FII	3.9.1.2.2.2(a)	3.6B.1
WHAMMOC II	3.9.1.2.2.2(a)	3.6B.3
RELAP 3	3.9.1.2.2.2(a)	3.6B.1
RELAP 4	3.9.1.2.2.2(a)	3.6B.1
NASTRAN	3.9.1.2.2.2(b)	3.7.2.1

The FSAR will be updated to reflect this response in Subsections 3.6.2.2.1 and 3.6B.1.

Q 210.2
1 of 2

c) Breaks are postulated to occur either instantaneously, with a break opening time of 0.001 seconds or, with a break opening based on analytical or experimental methods to determine break opening time for a particular case.

Blowdown forces are determined by either a simplified conservative method or a detailed computerized method as described in a, and b, below. The method to be used is determined by the forcing function input required by the methods of dynamic response analysis, which are discussed below in Subsection 3.6.2.2.2.

a) The predicted blowdown force (F_{BD}) on a pipe with flow area A fed by an infinite source volume can at pressure P_o be described by

$$F_{BD} = K P_o A$$

Where K is the thrust coefficient for the case of steam, saturated water choked steam flow is assumed and, based on this flow, the resulting steady state force thrust coefficient is K = 1.26. The source pressure P_o is taken as the maximum operating pressure experienced by the piping system and the resulting blowdown force is assumed to be a step function with zero rise time. For the case of subcooled liquids, such as feedwater, the force time history will consist of two values; an initial * magnitude equal to $(2.0)P_oA$ at time zero, decreasing linearly to $(1.26)\ P_{sat}A$ at a time, t = L/C, where L is the length of the line from the break to the reservoir or pump, C is the speed of sound in the fluid, and Psat the saturation pressure corresponding to the maximum operating temperature in the line. *

b) The following is an alternate less conservative computerized method for calculating blowdown force:

Computer codes, Relap-3 (Ref. 1) Relap-4 (Ref.2) or WHAMMOC II (see Appendix 3.6B) are used to determine thermodynamic properties for straight pipe lengths defining changes in direction. See Subsection 6.2.1.4 for typical examples of output for a main steam line break. From this data a ~~computer code CALPLOTF II has~~ CALPLOTF (in conjunction with RELAP 3) and its latest revision CAPLOTF II computer codes have been written to convert the transient flow conditions calculated in a piping system by the WHAMMOCII, or the RELAP series of computer codes into transient forces on the piping system. Specifically, CALPLOTFII calculates and plots the forces on straight lengths of pipe between changes in pipe direction (bends), or between a change in direction and a pipe break. For discussion of Both CALPLOTF and CALPLOTFII see Appendix 3.6B.

3.6.2.2.2 Pipe Whip Dynamic Response Analysis

Using the blowdown forcing functions, dynamic analysis is performed in order to determine: loads on restraints, loads on piping nozzles, loads on valves and maximum displacements of whipping pipe, unless studies show the whipping pipe is of no consequence. For this purpose these quantities can be determined by simplified conservative methods as described in a) below or by

CALPLOT F is a post processor code used to calculate transient piping forces using the magnetic Tape created by Relap-3 Computer Code.

3.6B.1 CalPLOTF and CALPLOTFII

'A' The CALPLOTFII computer code is a post processor code used to calculate transient piping forces using, as input, the magnetic tape created by the RELAP4 Mod6 or the WHAMMOCII computer code. This tape contains the calculated thermodynamic and flow conditions required by CALPLOTFII to calculate forces. A more complete description of the CALPLOT F CALPLOTFII computer code ~~is~~ are contained in Appendix 3.6C of the Waterford SES Unit No. 3 Final Safety Analysis Report (Docket No. 50-382).

3.6B.2 PLAST

The PLAST computer code uses the forces calculatei by CALPLOTFII to determine the pipe whip restraint reaction loads by performing a dynamic structural analysis on a lumped mass parameter of the piping system. A description of the PLAST computer code is contained in Appendix 3.6B of the Waterford SES Unit No. 3 Final Safety Analysis Report (Docket No. 50-382).

3.6B.3 WHAMMOCII

A Computer Code for Performing One of Two Phase Water Hammer Analysis.

3.6B.3.1 Summary

A computer code named WHAMMOCII has been developed to solve rapid fluid transients (acoustic hammer) problems for water, water with dissolved air, steam, or two-phase one or two component fluid networks. The transient fluid conditions, thermodynamic conditions, and the forces on a piping network can be calculated, following any initiated rapid transient.

The WHAMMOCII computer code uses the method of characteristics to solve the continuity and momentum equations assuming compressible, adiabatic, homogeneous one or two-phase no slip flow, and includes the effects of piping deformation and cavitation on pulse speed propagation. The pressure predictions of the WHAMMOCII computer code are compared with various experiments including two subcooled water blowdown test, and a rapid valve closure transient in a pipe containing water with dissolved air. The code's prediction of the forces generated by a subcooled blowdown are also compared. The nomenclature used in WHAMMOCII equations is given in Table 3.6B-1.

3.6B.3.2 Introduction

Fluid hammer phenomena have become a major concern of the utility industry for the design of piping networks. At present, no publicly available computer code can analyze the water hammer phenomena which results in a transition from one to two-phase flow. Consequently, the computer code WHAMMOCII(1) has been developed to analyze acoustic wave (hammer) problems for one-phase water or steam conditions, for one-phase water with dissolved air, for two-phase multicomponent (air-steam-water) flows, and for the determination of phase transitions. The code is not intended to be used for the calculation of nonacoustic phenomenon effects such as shock waves. WHAMMOCII can calculate the fluid thermodynamic properties, and the piping fluid forces following any initiating rapid transient such as an opening or closing valve, a starting or tripping pump, a pipe break or any known time dependent pressure boundary condition. A one-phase condition, a homogeneous two-phase condition, or a column separation condition for one or two-phase flow can be predicted by the code.

Attachment 1-8
2266A

Question No.

210.3 (3.9.3B)	Either supply the information identified as later in Appendix 3.9.3B or provide a schedule for submittal of this information.

Response

The information identified as "later" within Appendix 3.9.3B will be supplied by September 1983.

The FSAR will be amended to reflect this.

Q 210.3
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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2AF-VD035SA	Target Rock Corp	TRP-2341 Rev B
2AF-VD036SB	Target Rock Corp	TRP-2341 Rev B
2AF-VD039SA	Target Rock Corp	TRP-2341 Rev B
2AF-VD040SB	Target Rock Corp	TRP-2341 Rev B
3AF-VD098SB	Target Rock Corp	TRP-2341 Rev B
3AF-VD099SA	Target Rock Corp	TRP-2341 Rev B
3AF-VD100SB	Target Rock Corp	TRP-2341 Rev B
3AF-VD101SA	Target Rock Corp	TRP-2341 Rev B
2BD-VE037SA	Borg-Warner Corp	*
2BD-VE038SA	Borg-Warner Corp	*
2BD-VE039SB	Borg-Warner Corp	*
2BD-VE040SB	Borg-Warner Corp	*
2BD-VR033SA	Borg-Warner Corp	*
2BD-VR034SA	Borg-Warner Corp	*
2BD-VR035SB	Borg-Warner Corp	*
2BD-VR036SB	Borg-Warner Corp	*
2BD-VR077SA	Borg-Warner Corp	*
2BD-VR078SA	Borg-Warner Corp	*
2BD-VR079SB	Borg-Warner Corp	*
2BD-VR080SB	Borg-Warner Corp	*
2BD-VR089SA	Borg-Warner Corp	*
2BD-VR090SA	Borg-Warner Corp	*

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2BD-VR091SB	Borg-Warner Corp	*
2BD-VR092SB	Borg-Warner Corp	*
3CC-B004SA	Litton Contromatics	No. 14225-5A Rev 1
3CC-B005SB	Litton Contromatics	No. 14225-5A Rev 1
3CC-B066SN	Litton Contromatics	No. 14225-5A Rev 1
3CC-B067SN	Litton Contromatics	No. 14225-5A Rev 1
3CC-B068SN	Litton Contromatics	No. 14225-5A Rev 1
3CC-B069SN	Litton Contromatics	No. 14225-5A Rev 1
3CC-B507SA	Litton Contromatics	No. 14225-2A Rev 1
3CC-B508SA	Litton Contromatics	No. 14225-2A Rev 1
3CC-B509SB	Litton Contromatics	No. 14225-2A Rev 1
3CC-B510SB	Litton Contromatics	No. 14225-2A Rev 1
3CC-B511SA	Litton Contromatics	*
3CC-B512SB	Litton Contromatics	*
3CC-B513SA	Litton Contromatics	No. 14225-2A Rev 1
3CC-B514SA	Litton Contromatics	No. 14225-2A Rev 1
3CC-B515SB	Litton Contromatics	No. 14225-2A Rev 1
3CC-B516SB	Litton Contromatics	No. 14225-2A Rev 1
2CC-B521SB	Litton Contromatics	No. 14225-3A Rev 1
2CC-B522SB	Litton Contromatics	No. 14225-3A Rev 1
2CC-B523SA	Litton Contromatics	No. 14225-2A Rev 1
2CC-B524SA	Litton Contromatics	No. 14225-2A Rev 1

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2CC-B525SB	Litton Contromatics	No. 14225-2A Rev 1
2CC-B526SB	Litton Contromatics	No. 14225-2A Rev 1
3CC-B531SA	Litton Contromatics	No. 14225-2A Rev 1
3CC-B532SB	Litton Contromatics	No. 14225-2A Rev 1
3CC-B539SA	Litton Contromatics	*
3CC-B540SB	Litton Contromatics	*
3CC-VE255SB	Target Rock Corp	TR 2684 Rev B
3CC-VE535SA	Target Rock Corp	TR 2684 Rev B
3CC-VE536SB	Target Rock Corp	TR 2684 Rev B
3CC-VE638SA	Target Rock Corp	TR 2684 Rev B
2CH-VP011SBR	Borg-Warner Corp	*
2CH-VP029SAR	Borg-Warner Corp	*
2CH-VP030SBR	Borg-Warner Corp	*
2CH-VS041SA	Borg-Warner Corp	*
2CH-VS042SBR	Borg-Warner Corp	*
2CH-VS044SA	Borg-Warner Corp	*
2CH-VS047SBR	Borg-Warner Corp	*
2CH-VW037SAR	Borg-Warner Corp	*
2CH-VW040SBR	Nissho-Iwai American Corp	*
2CH-VW401SBR	Atwood & Morrill	No. 62 Rev 0
2CH-VW404SAR	Atwood & Morrill	No. 62 Rev 0

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS

FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2CS-B001SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2CS-B002SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2CS-VQ005SAR	Target Rock Corp	TRP-2341 Rev B
2CS-VQ006SBR	Target Rock Corp	TRP-2341 Rev B
2CS-VS015SAR	Atwood & Morrill	No. 34 Rev B
2CS-VS016SBR	Atwood & Morrill	No. 34 Rev B
2CS-VS017SAR	Atwood & Morrill	No. 31 Rev C
2CS-VS018SBR	Atwood & Morrill	No. 31 Rev C
2CS-VS021SAR	Atwood & Morrill	No. 32 Rev A
2CS-VS022SBR	Atwood & Morrill	No. 32 Rev A
2CS-VS023SAR	Atwood & Morrill	No. 34 Rev B
2CS-VS024SBR	Atwood & Morrill	No. 34 Rev B
2CS-VS079SAR	Atwood & Morrill	No. 31 Rev C
2CS-VS080SBR	Atwood & Morrill	No. 31 Rev C
2CS-VS091SA	Atwood & Morrill	No. 33 Rev B
2CS-VS092SB	Atwood & Morrill	No. 33 Rev B
2CS-VU055SAR	Borg-Warner Corp	*
2CS-VU056SBR	Borg-Warner Corp	*
3EC-B001SA	Litton Contromatics	No. 14225-3A Rev 1
3EC-B003SA	Litton Contromatics	No. 14225-3A Rev 1
3EC-B004SA	Litton Contromatics	No. 14225-3A Rev 1
3EC-B007SB	Litton Contromatics	No. 14225-3A Rev 1

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2EC-B009SB	Litton Contromatics	No. 14225-3A Rev 1
2EC-B010SA	Litton Contromatics	No. 14225-3A Rev 1
2EC-B011SB	Litton Contromatics	No. 14225-3A Rev 1
2EC-B012SA	Litton Contromatics	No. 14225-3A Rev 1
3EC-B013SB	Litton Contromatics	No. 14225-3A Rev 1
3EC-B014SB	Litton Contromatics	No. 14225-3A Rev 1
2FP-VH005SB	Borg-Warner Corp.	*
2FP-VH006SB	Borg-Warner Corp	*
2FP-VH007SB	Borg-Warner Corp	*
2FW-VD021SB	Atwood & Morrill	201-14225-00 Rev 1
2FW-VD022SA	Atwood & Morrill	No. 36 Rev A
2FW-VD023SB	Borg-Warner Corp	*
2FW-VD024SA	Borg-Warner Corp	*
2FW-VD027SB	Atwood & Morrill	201-14225-00 Rev 1
2FW-VD028SA	Atwood & Morrill	No. 36 Rev A
2FW-VD036SB	Atwood & Morrill	201-14225-00 Rev 1
2FW-VD037SA	Atwood & Morrill	No. 36 Rev A
2FW-VD038SB	Borg-Warner Corp	*
2FW-VD039SA	Borg-Warner Corp	*
2FW-VD042SB	Atwood & Morrill	201-14225-00 Rev 1
2FW-VD043SA	Atwood & Morrill	No. 36 Rev A

* Later

DESIGN/SEISMIC QUALIFICATION REPORTS FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2HA-VS008SRA	Target Rock Corp	TR 2684 Rev B
2HA-VS009SRA	Target Rock Corp	TR 2684 Rev B
2HA-VS010SRA	Target Rock Corp	TR 2684 Rev B
2HA-VS027SRB	Target Rock Corp	TR 2684 Rev B
2HA-VS028SRB	Target Rock Corp	TR 2684 Rev B
2HA-VS029SRB	Target Rock Corp	TR 2684 Rev B
2MD-VE001SA	Borg-Warner Corp	*
2MD-VE002SA	Borg-Warner Corp	*
2MD-VE003SB	Borg-Warner Corp	*
2MD-VE004SB	Borg-Warner Corp	*
2MD-VE087SB	Borg-Warner Corp	*
2MD-VE088SB	Borg-Warner Corp	*
2MD-VE089SA	Borg-Warner Corp	*
2MD-VE090SA	Borg-Warner Corp	*
2MS-P012SA	Control Components Inc c/o Babcock & Wilcox	*
2MS-P016SB	Control Components Inc c/o Babcock & Wilcox	*
2MS-P021SB	Control Components Inc c/o Babcock & Wilcox	*
2MS-P023SA	Control Components Inc c/o Babcock & Wilcox	*
2MS-R006SA	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS

FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2MS-R007SA	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R008SA	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R009SA	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R010SA	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R011SA	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R013SB	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R014SB	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R015SB	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R018SB	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R019SB	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-R020SB	Dresser Industrial Valve & Instrument Division	SQ-37-7 Rev 1
2MS-VD001SA	Anchor Darling Valve Comp	E-6232-6/E-6210-1C
2MS-VD002SA	Anchor Darling Valve Comp	E-6232-6/E-6210-1C
2MS-VD003SB	Anchor Darling Valve Comp	E-6232-6/E-6210-1C
2MS-VD004SB	Anchor Darling Valve Comp	E-6232-6/E-6210-1C
2MS-VE055SA	Borg-Warner Corp	*

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2MS-VE082SB	Borg-Warner Corp	*
2MS-VE091SA	Borg-Warner Corp	*
2MS-VE092SB	Borg-Warner Corp	*
2MS-VE138SB	Borg-Warner Corp	*
2MS-VE140SA	Borg-Warner Corp	*
2NG-VE057SB	Borg-Warner Corp	*
2NG-VE058SA	Borg-Warner Corp	*
2PV-B001SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B003SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B004SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B005SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B007SA	Litton Contromatics	No. 14225-2A Rev 1
2PV-B009SB	Litton Contromatics	No. 14225-3A Rev 1
2PV-B010SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B014SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B016SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B017SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B018SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B019SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B023SB	Litton Contromatics	No. 14225-4A Rev 1
2PV-B024SA	Litton Contromatics	No. 14225-4A Rev 1

* Later

9

DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2PV-B025SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B029SA	Litton Contromatics	No. 14225-1A Rev 1
3PV-B033SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
3PV-B034SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
3PV-B035SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
3PV-B036SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B037SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B038SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B039SA	Litton Contromatics	No. 14225-2A Rev 1
2PV-B041SA	Litton Contromatics	*
2PV-B042SA	Litton Contromatics	*
2PV-B043SA	Litton Contromatics	*
2PV-B044SA	Litton Contromatics	*
2PV-B045SA	Litton Contromatics	*
2PV-B046SA	Litton Contromatics	*
2PV-B047SA	Litton Contromatics	*
2PV-B048SA	Litton Contromatics	*
2PV-B052SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B054SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B055SA	Litton Contromatics	No. 14225-3A Rev 1
3PV-B060SA	Litton Contromatics	No. 14225-1A Rev 1
3PV-B061SA	Litton Contromatics	No. 14225-1A Rev 1

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
3PV-B062SA	Litton Contromatics	No. 14225-1A Rev 1
3PV-B063SA	Litton Contromatics	No. 14225-1A Rev 1
2PV-B064SA	Litton Contromatics	No. 14225-4A Rev 1
2PV-B101SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B103SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B104SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B105SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B107SB	Litton Contromatics	No. 14225-2A Rev 1
2PV-B109SA	Litton Contromatics	No. 14225-3A Rev 1
2PV-B110SB	Litton Contromatics	No. 14225-3A Rev 1
2PV-B111SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B112SA	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B113SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B114SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B123SB	Litton Contromatics	No. 14225-4A Rev 1
2PV-B124SA	Litton Contromatics	No. 14225-4A Rev 1
2PV-B125SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
2PV-B129SB	Litton Contromatics	No. 14225-1A Rev 1
3PV-B133SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
3PV-B134SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
3PV-B135SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1
3PV-B136SB	McNally Pittsburg Mfg Corp	SRV-1502 Rev 1

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS

FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2PV-B137SB	Litton Contromatics	No. 14225-3A Rev 1
2PV-B138SB	Litton Contromatics	No. 14225-3A Rev 1
2PV-B140SB	Litton Contromatics	No. 14225-2A Rev 1
2PV-B141SB	Litton Contromatics	*
2PV-B142SB	Litton Contromatics	*
2PV-B143SB	Litton Contromatics	*
2PV-B144SB	Litton Contromatics	*
2PV-B145SB	Litton Contromatics	*
2PV-B146SB	Litton Contromatics	*
2PV-B147SB	Litton Contromatics	*
2PV-B148SB	Litton Contromatics	*
2PV-B152SB	Litton Contromatics	No. 14225-3A Rev 1
2PV-B154SB	Litton Contromatics	No. 14225-3A Rev 1
2PV-B155SB	Litton Contromatics	No. 14225-3A Rev 1
3PV-B160SB	Litton Contromatics	No. 14225-1A Rev 1
3PV-B161SB	Litton Contromatics	No. 14225-1A Rev 1
3PV-B162SB	Litton Contromatics	No. 14225-1A Rev 1
3PV-B163SB	Litton Contromatics	No. 14225-1A Rev 1
2PV-B164SB	Litton Contromatics	No. 14225-4A Rev 1
1SI-VP091SAR	Atwood & Morrill	*
1SI-VP092SAR	Atwood & Morrill	*
1SI-VP097SBR	Atwood & Morrill	*

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS

FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
1SI-VP098SBR	Atwood & Morrill	*
1SI-VP104SAR	Atwood & Morrill	*
1SI-VP107SAR	Atwood & Morrill	*
1SI-VP110SBR	Atwood & Morrill	*
1SI-VP113SBR	Atwood & Morrill	*
1SI-VP185SBR	Borg-Warner Corp	*
1SI-VP186SBR	Borg-Warner Corp	*
2SI-VQ011SAR	Borg-Warner Corp	*
2SI-VQ027SBR	Borg-Warner Corp	*
2SI-VQ033SA	Borg-Warner Corp	*
2SI-VQ035SB	Borg-Warner Corp	*
2SI-VQ046SA	Borg-Warner Corp	*
2SI-VQ049SB	Borg-Warner Corp	*
2SI-VQ087SA	Target Rock Corp	*
2SI-VQ088SB	Target Rock Corp	TRP-2341 Rev B
2SI-VQ115SA	Borg-Warner Corp	*
2SI-VQ170SA	Target Rock Corp	*
2SI-VQ171SB	Target Rock Corp	*
2SI-VQ172SAR	Target Rock Corp	TRP-2341 Rev B
2SI-VQ173SBR	Target Rock Corp	TRP-2341 Rev B
2SI-VQ201SAR	Target Rock Corp	TR 2684 Rev B
2SI-VQ202SBR	Target Rock Corp	TR 2684 Rev B
2SI-VQ203SA	Target Rock Corp	TR 2684 Rev B

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2SI-VQ204SB	Target Rock Corp	TR 2684 Rev B
2SI-VQ207SBR	Target Rock Corp	TR 2684 Rev B
2SI-VQ208SAR	Target Rock Corp	TR 2684 Rev B
2SI-VQ209SB	Target Rock Corp	TR 2684 Rev B
2SI-VQ210SA	Target Rock Corp	TR 2684 Rev B
2SI-VS054SAR	Nissho-Iwai American Corp	*
2SI-VS060SBR	Nissho-Iwai American Corp	*
2SI-VS131SA	Target Rock Corp	TR 2684 Rev B
2SI-VS134SA	Target Rock Corp	TR 2684 Rev B
2SI-VS137SA	Target Rock Corp	TR 2684 Rev B
2SI-VS140SA	Target Rock Corp	TR 2684 Rev B
2SI-VS193SB	Target Rock Corp	TR 2684 Rev B
2SI-VS194SB	Target Rock Corp	TR 2684 Rev B
2SI-VS195SB	Target Rock Corp	TR 2684 Rev B
2SI-VS196SB	Target Rock Corp	TR 2684 Rev B
2SI-VU001SAR	Atwood & Morrill	No. 39 Rev A
2SI-VU021SBR	Atwood & Morrill	No. 39 Rev A
2SI-VU055SAR	Atwood & Morrill	No. 38 Rev B
2SI-VU061SBR	Atwood & Morrill	No. 38 Rev B
2SL-VP100SABR	Borg-Warner Corp	*
2SL-VP101SABR	Borg-Warner Corp	*
2SL-VP102SABR	Borg-Warner Corp	*

* Later

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DESIGN/SEISMIC QUALIFICATION REPORTS
FOR A/E SUPPLIED ACTIVE VALVES

Valve Tag No.	Manufacturer	Manufacturer's Report No.
2SL-VP103SABR	Borg-Warner Corp	*
2SL-VP104SABR	Borg-Warner Corp	*
2SL-VP105SABR	Borg-Warner Corp	*

* Later

Question No.

210.4 (3.9.3.4)

The SRP (NUREG-0800) contains the following requirements:

All safety-related components which utilize snubbers in their support systems should be identified and tabulated in the FSAR. The tabulation should include the following information: (i) identification of the systems and components in those systems which utilize snubbers, (ii) the number of snubbers utilized in each system and on components in that system, (iii) the type(s) of snubber (hydraulic or mechanical) and the corresponding supplier identified, (iv) specify whether the snubber was constructed to the rules of ASME Code Section III, Subsection NF, (v) state whether the snubber is used as a shock, vibration, or dual purpose snubber, and (vi) for snubbers identified as either dual purpose or vibration arrestor type, indicate if both snubber and component were evaluated.

Provide or reference this material for snubbers utilized on all safety-related components.

Response

Information regarding snubbers on safety-related components requires input from both the A/E and NSSS vendor, as well as the Supply System. This information is being assembled now and will be available December 1982.

Attachment 1-10
2266A

Question No.

210.5 (3.9.6) Provide a schedule for submittal of the in-service testing program.

Response

The In-service testing program will be submitted one year prior to fuel load. This committment will be included in a subsequent amendment to the WNP-3 FSAR as indicated on the attached markup of FSAR page 3.9-154.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

This section addresses the program for inservice testing for operational readiness of ASME Code, Section III, Class 1, 2, and 3 pumps and valves. The detailed program, indicating valves and pumps to be tested and the test specifics, will be submitted as a separate document one year prior to Fuel Load. ~~The program will incorporate the requirements of the latest approved edition and addenda of ASME Section XI, Subsections IWP and IWV at the time of its preparation. As required by 50.55a(g), the program will be revised prior to commercial operations to incorporate the operational readiness requirements of the ASME Code Edition and Addenda in effect one year prior to date of issuance of the operating license.~~

Attachment 1-11
2266A

Question No.

210.6 (5.2.1.1) Provide in Table 5.2-1 or reference the code requirements (class, edition and addenda) for the RCS pumps and RCPB valves (A/E).

Response

The Code requirements for the RC Pumps and valves (A/E supplied) are as follows:

	Class	Edition	Addenda Unit 3	Addenda Unit 5
RC Pump	1	1974	None	None
RCPB Valves (A/E) - size 2-1/2 in. and larger	1	1977	None	None
- size 2 in. and smaller	1	1977	Summer 77	Summer 77

FSAR Table 5.2-1 will be amended to reflect this information.

Q 210.6
1 of 1

TABLE 5.2-1

REACTOR COOLANT SYSTEM PRESSURE BOUNDARY CODE REQUIREMENTS

ASME BOILER & PRESSURE VESSEL CODE
SECTION III - NUCLEAR POWER PLANT COMPONENTS

COMPONENT	CLASS	EDITION	ADDENDA UNIT 3	ADDENDA UNIT 5
REACTOR VESSEL	1	1971	SUMMER 73	WINTER 73
STEAM GENERATOR				
- PRIMARY SIDE	1	1971	SUMMER 73	WINTER 73
- SECONDARY SIDE	2	1971	SUMMER 73	WINTER 73
~~PRESSURIZER~~	1	1971	SUMMER 73	WINTER 73
RC PUMP	1	1974	NONE	NONE
REACTOR COOLANT PIPE	1	1971	SUMMER 73	WINTER 73
VALVES				
PRESSURIZER SAFETY	1	1974	SUMMER 75	SUMMER 75
PRESSURIZER SPRAY	1	1974	WINTER 75	WINTER 75
MOTOR OPERATED	1	1974	WINTER 75	WINTER 75
PNEUMATIC	1	1974	WINTER 75	WINTER 75
CONTROL ELEMENT DRIVE MECHANISM	1	1974	WINTER 75	WINTER 75
RCPB PIPE (A/E)	1	1974	SUMMER 76	SUMMER 76
RCPB VALVES (A/E)	1	(IN ACCORDANCE WITH THE CODE EDITION AND ADDENDA IN EFFECT AT TIME OF PURCHASE ORDER)		
RCPB VALVES (A/E)				
- size 2½ in and larger	1	1977	NONE	NONE
- size 2 in and smaller	1	1977	SUMMER 77	SUMMER 77

Attachment 1-12
2266A

Question No.

210.7 (5.4.1.1.2)	Subsection 5.4.1.1.2 references Table 5.4-1. Either supply the information identified as "later" in Table 5.4-1 or provide a schedule for its submittal.

Response

Attached are the Certified Material Test Reports provided by the material manufacturer. The CMTRs are indexed by the Pump Serial Number.

FSAR Table 5.4-1 will be amended to reflect this information.

6. ALLIS-CHALMERS CORP.
MR. WILLIAM HACKBARTH
DEPT. 8052 (AA)
P.O. BOX 512
MILWAUKEE, WISC. 53201

S/N 47423

TEST CERTIFICATE

MILL ORDER NO	CUSTOMER P.O.		CONSIGNED
73524-1	WAH-457571-EA	BL 51677 8T 4/4	ALLIS-CHALMERS CORP. 66TH AND GREENFIELD MAIN RECEIVING WEST ALLIS, WISC.

A.C. 05-685-134-412 REV.2 DTD 12/7/76 A-543-72A CL.1 TYPE-B ASME CODE SEC.3 SUB NB 1974 EDIT.
THRU SUMMER 1974 ADDENDA & CODE CASE 1358 N-1160 6/4/78

SHEET 1 OF 2

BEND TEST O.K. HOMOGENEITY TEST

CHEMICAL ANALYSIS

MELT NO.	C	MN	P	S	CU	SI	NI	CR	MO	V	TI	XXX	XXX		BASIC PROCE
C6758	17	30	014	015		32	3.59	1.81	47	00		VIP	STEEL		ELEC.
B0923	17	30	008	013		32	3.23	1.85	48	00		VIP	STEEL		ELEC.

Q210.7
1 of 5

PHYSICAL PROPERTIES

MELT NO.	SLAB NO.	YIELD	TENSILE	% ELONG. in 2"	% R.A.	BHN	IMPACTS TV +212°F.			FRACTURE APPEARANCE % SHEAR	DESCRIPTION
C6758	3*	957	1136	17			92	84	93	80-80-80	1- 10" X 30-1/2 ID X 66-1/2 OD
		960	1141	21		LATERAL EXPANSION IN INCHES					
							.078	.083	.085		
B0923	3**	948	1155	20			80	82	84	70-70-70	1- "
		919	1100	21		LATERAL EXPANSION IN INCHES					
							.073	.069	.071		
B0923	5**	963	1196	19			82	80	79	70-70-70	1- "
		938	1200	16		LATERAL EXPANSION IN INCHES					
							.078	.072	.073		

LONG. DROP WEIGHT TESTS FOR EACH SLAB PER E-208 (SIZE P-3) @ -40°F.
EXHIBIT NO BREAK. --- N.D.T. IS -50°F OR BELOW.

We hereby certify the above information is correct.

R. H. Klis

6. ALLIS-CHALMERS CORP.
MR. WILLIAM HACKBARTH S/N 47423
DEPT. 8052 (AA)
P.O. BOX 512
MILWAUKEE, WISC. 53201

TEST CERTIFICATE

MILL ORDER NO	CUSTOMER P.O.		CONSIGNEE
73524-1	WAM-457571-EA	BL 51677 BT 4/4	ALLIS-CHALMERS CORP. 66TH AND GREENFIELD MAIN RECEIVING WEST ALLIS, WISC.

A.C. 05-685-134-412 REV.2 DTD 12/7/76 A-543-72A CL.1 TYPE-B ASME CODE SEC.3 SUB NB 1974 EDIT.
THRU SUMMER 1974 ADDENDA & CODE CASE 1358 N-1160 6/4/78
SHEET 1 OF 2

BEND TEST O.K. HOMOGENEITY TEST

CHEMICAL ANALYSIS

MELT NO.	C	MN	P	S	CU	SI	NI	CR	MO	V	TI	XXX	XXX		BASIC PROCE
C6758	17	30	014	015		32	3.59	1.81	47	00		VIP	STEEL		ELEC.
B0923	17	30	008	013		32	3.23	1.85	48	00		VIP	STEEL		ELEC.

Q210.7
1 of 5

PHYSICAL PROPERTIES

MELT NO.	SLAB NO.	YIELD PS. KSI	TENSILE PS. KSI	% ELONG in 2"	% R.A.	BHN	IMPACTS TV +212°F.			FRACTURE APPEARANCE % SHEAR	DESCRIPTION
C6758	3*	957 960	1136 1141	17 21			92	84	93	80-80-80	1- 10" X 30-1/2 ID X 66-1/2 OD
						LATERAL EXPANSION IN INCHES	.078	.083	.085		
B0923	3**	948 919	1155 1100	20 21			80	82	84	70-70-70	1- "
						LATERAL EXPANSION IN INCHES	.073	.069	.071		
B0923	5**	963 938	1196 1200	19 16			82	80	79	70-70-70	1- "
						LATERAL EXPANSION IN INCHES	.078	.072	.073		

LONG. DROP WEIGHT TESTS FOR EACH SLAB PER E-208 (SIZE P-3) @ -40°F.
EXHIB. T NO BREAK. --- N.D.T. IS -50°F. OR BELOW.

We hereby certify the above information is correct.

AUTHORIZED SIGNATURE F. H. Klis

6. ALLIS-CHALMERS CORP.
MR. WILLIAM HACKBARTH
MILWAUKEE, WISC. 53201

S/N 47423

TEST CERTIFICATE

MILL ORDER NO.	CUSTOMER P.O.	
73524-1	WAH-457571-EA	BL 51677 BT 4/5

ALLIS-CHALMERS CORP.
WEST ALLIS, WISC.

MATERIAL HAS BEEN MANUFACTURED AND TESTED IN ACCORDANCE WITH PURCHASE ORDER REQUIREMENTS AND SPECIFICATIONS

SAME

SHEET 2 OF 2

BEND TEST   HOMOGENEITY TEST

CHEMICAL ANALYSIS

MELT NO.	C	MN	P	S	CU	SI	NI	CR	MO	V	TI	AL	B		
2															

PHYSICAL PROPERTIES

MELT NO.	SLAB NO.	YIELD PSI	TENSILE PSI	% ELONG	% R.A.	BHN	IMPACTS			DESCRIPTION

*RING AND TESTS HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, RE-HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, THEN TEMPERED 1140°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED.

**RINGS AND TESTS HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, RE-HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, THEN TEMPERED 1160°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED.

MAY 1 9 1977

Notary Public

We hereby certify the above information is correct.

SUPERVISOR TESTING

LUKENS STEEL COMPANY
COATESVILLE, PA 19320
TEST CERTIFICATE 47424

DATE 3-3-78

PURCHASER: ROTATING APPARATUS
MILWAUKEE, WISCONSIN 53201

CONSIGNEE: SIEMENS-ALLIS, INC.
WEST ALLIS, WISC. 53214

S/N 47424

MILL ORDER NO	CUSTOMER P.O.	
58479-1	WAN495421EA PT. 2	BL 8378 DD 4/11

A-543-72A CL. 1 TYPE-B MOD. BY A.C. 05-684-134-412 REV. 2 12/7/76 ASME CODE SECT. III SUB NB 1974 EDITION THRU SUMMER 1974 ADDENDA AND CODE CASE 1358 N-1160 6/4/78

BEND TEST O.K. HOMOGENEITY TEST

CHEMICAL ANALYSIS

MELT NO	C	Mn	P	S	Cu	Si	Ni	Cr	Mo	V	Ti	Al	XXXXX	BASIC PROC
52673	.15	.35	.009	.016		.30	3.16	1.94	.45	.00			VIP STEEL	ELEC

M

PHYSICAL PROPERTIES

MELT NO.	SLAB NO.	YIELD	TENSILE	% ELONG in 2"	% R.A.	BHN	IMPACTS TV +212°F.			FRACTURE APPEARANCE & SHEAR	DESCRIPTION
82673	5	967	1146	19			86	88	85	70-70-70	1- 10" x 30-1/2 ID x 66-1/
		927	1106	21	LATERAL EXPANSION IN INCHES		.076	.072	.075		

LONG. DROP WEIGHT TESTS PER E208 (SIZE P3) @ -40°F., EXHIBIT NO BREAK
NDT IS -50°F. OR BELOW

PLATE AND TESTS HEATED 1650-1700°F., HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, THEN RE-HEATED 1650-1700°F., HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, THEN TEMPERED 1160°F., HELD 1 HR. PER INCH MIN. AND WATER QUENCHED.

We hereby certify the above information is correct.

H. H. Kles

6. LARGE[illegible] [illegible]ATING APPARATUS
SIEMENS-ALLIS, INC.
ATTN: NANCY JENKINS, BUYER
BOX 2166
MILWAUKEE, WISCONSIN 53201

S/N47425

TEST CERTIFICATE

[illegible]7425

CONSIGNEE

MILL ORDER NO.	CUSTOMER P.O.	
58481 1	WAM495421EA PT.3	MP 81078 BT L

A-543-72A CL.1 TYPE-B ASME CODE SEC.III SUB NB 1974 EDIT. THRU SUMMER 1974 ADDENDA AND CODE CASE 13589 MODIFIED BY A.C. 05-684-134-412 REV.2 12/7/76 N1160 8/4/78 EXTENDED TO 9/30/78 BY ASME 8/4/78

BEND TEST O.K. HOMOGENEITY TEST

CHEMICAL ANALYSIS

MELT NO.	C	MN	P	S	CU	SI	NI	CR	MO	V	TI	XXA	XXB		BASIC PROC
B2551	.17	.32	.007	.013		.32	3.15	1.77	.49	.00		VIP	STEEL		ELEC.

4

PHYSICAL PROPERTIES

MELT NO.	SLAB NO.	YIELD PSI x1000	TENSILE PSI x1000	% ELONG. in 2"	% R.A.	BHN	IMPACTS TV +212°F.			FRACTURE APPEARANCE % SHEAR	DESCRIPTION
B2551	1	972 905	1130 1090	19 16			90	96	88	70-70-70	2- 10" X 30-1/2 ID X 86-1/2 OD
						LATERAL EXPANSION IN INCHES	.068	.066	.073		

LONG. DROP WEIGHT TESTS FOR ALL SLABS PER E208 (SIZE P3) @ -40°F. EXHIBIT NO BREAK. N.D.T. IS -50°F. OR BELOW.

RBH
9/5/78

PLATES AND TESTS HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, RE-HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, THEN TEMPERED 1160°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED.

We hereby certify the above information is correct.

INSPECTOR TESTING R. H. Ellis

B. LARGE ROTATING APPARATUS S/N47426
STIEMENS-ALLIS, INC.
ATTN: NANCY JENKINS, BUYER
BOX 2166
MILWAUKEE, WISCONSIN 53201

TEST CERTIFICATE 47426

MILL ORDER NO.	CUSTOMER P.O.		CONSIGNEE
58481 1	WAH495421EA PT.3	MP 81078 BT L	

THE MATERIAL HAS BEEN ... AND TESTED IN ACCORDANCE WITH PURCHASE ORDER REQUIREMENTS AND SPECIFICATIONS: SEC.III SUB NB 1974 EDIT. THRU SUMMER 1974 ADDENDA
A-543-72A CL.1 TYPE-B ASME CODE SEC.III SUB NB 1974 EDIT. THRU SUMMER 1974 ADDENDA
AND CODE CASE 13589 MODIFIED BY A.C. 05-684-134-412 REV.2 12/7/76
N1160 6/4/78 EXTENDED TO 9/30/78 BY ASME 8/4/78

BEND TEST O.K. HOMOGENEITY TEST

CHEMICAL ANALYSIS

MELT NO.	C	MN	P	S	CU	SI	NI	CR	MO	V	TI	XXM	XXB		BASIC PR
B2551	.17	.32	.007	.013		.32	3.15	1.77	.49	.00		VIP STEEL			ELEC.

5 of 5

PHYSICAL PROPERTIES

MELT NO.	SLAB NO.	YIELD PS. X 00	TENSILE PS. X 00	% ELONG. in 2"	% R.A.	BHN	IMPACTS TV +212°F.			FRACTURE APPEARANCE % SHEAR	DESCRIPTION
B2551	1	972 905	1130 1090	19 16			90	96	88	70-70-70	2- 10" X 30-1/2 ID X 86-1/2 OD
						LATERAL EXPANSION IN INCHES	.068	.066	.073		

LONG. DROP WEIGHT TESTS FOR ALL SLABS PER E208 (SIZE P3) @ -40°F. EXHIBIT NO BREAK
N.D.T. IS -50°F. OR BELOW.

REH
9/5/78

PLATES AND TESTS HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, RE-HEATED 1650°F./1700°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED, THEN TEMPERED 1160°F. HELD 1 HR. PER INCH MIN. AND WATER QUENCHED.

We hereby certify the above information is correct.

SUPERVISOR TESTING H. H. Elli

Attachment 1-13
2266A

Question No.

220.1 (3.4.2) As per Regulatory Guide 1.70, summarize for each safety-related structure, system, and component that may be so affected, the design basis static and dynamic loadings, including consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loadings, and the static and dynamic effects on foundation properties. Provide or reference this material.

Response

Seismic Category I structures do not require flood protection as described in Subsection 3.4.1. As a result design basis static and dynamic loadings, including consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings and coincident wind loadings resulting from flooding conditions are not applicable.

The treatment of groundwater induced hydrostatic loading is described in Subsection 2.4.13.5 which is referenced in Subsection 3.4.1.2.3.

The FSAR will be amended to reflect the aforementioned response.

Question 220.1
Page 1 of 1

For the portions of the vertical collection system on the wall adjacent to the Turbine Building, the vertical stand pipes will be four feet above the ground floor EL. 390 feet (see Figure 3.4.1-2). This elevation is above the highest level (3 ft) that circulating water has been estimated to rise in the event of a CWS break inside the building (see Subsection 10.4.5). Thus any water released into the Turbine Building will be prevented from directly entering the drainage collection system.

A portion of the standpipe is designed to be seismic Category I (see Figure 3.4.1-2) and as such will resist the passive pressure of the weathered sandstone in the embedded portion. Pipe straps are used to restrain the standpipe above the Turbine Building ground floor slab to resist the peak seismic acceleration of the RAB at grade elevation. The RAB response spectra and ground acceleration are described in Subsection 3.7.1.

A one inch gap is provided between the Turbine Building ground floor slab and the pipe, and is filled with a flexible compound.

Watertight seals are provided on all below-grade penetrations of the RAB to prevent groundwater seepage into the RAB and to stop water from a pipe break in the RAB from entering the groundwater drainage system. The GWDS is not classified as seismic Category I except for the manholes at the corners of the RAB and the portions of the vertical collection system on the wall adjacent to the Turbine Building. Subsection 3.8.4 provides additional discussion of the design of the Category I manholes. This classification is adequate since a failure of the GWDS (clogging of the drain pipes) during a seismic event would not cause an appreciable rise in the groundwater level for a minimum of 115 days (see Subsection 3.4.2.1). In addition, the GWDS will be inspectable to assure proper functioning at any time, including after an earthquake. Partial plugging of the GWDS will not reduce its effectiveness.

Each horizontal pipe has a full flow capacity of at least 385 gpm and can accept flow in either direction so that the drains will function satisfactorily if one outlet should become temporarily plugged. Furthermore, the vertical drains are six inches in diameter to minimize the chances of plugging.

In the unlikely case of a complete blockage of the system during the interval between inspections, the groundwater has been calculated to rise to an elevation of 365 feet above MSL in a minimum of 115 days during maximum recharge conditions. (See Subsection 3.4.2.1). The walls and mat are designed accordingly to withstand the resulting hydrostatic load (see Subsection ~~2.5.4.10).~~
2.4.13.5

Any blockage in the horizontal or vertical drains can be removed by using power augers or flushing with a high-pressure stream of water. Blockage in the drainage tunnel can be removed by either of the above two methods or by digging.

The vertical manholes at each corner of the building are designed to seismic Category I classification to provide access to the horizontal headers and the drainage tunnel in the event of an earthquake (for seismic criteria see

Attachment 1-14
2266A

Question No.

220.2 (3.7.1.1) Regulatory Guide 1.70 states that the basis for any response spectra that differ from the spectra given in Regulatory Guide 1.60 should be included in the FSAR section. You state that the vertical design response spectra does not comply with the recommendations of Regulatory Guide 1.60 but do not provide a basis for this divergence. Reference or provide this basis.

Response

The procedure for development of the vertical design response spectra was based entirely on the recommendations of Newmark, Blume, and Kapur (Reference 3.7.1-1) and was previously accepted by the NRC as documented in the Construction Permit SER.

As indicated in Subsection 3.7.1.1.2, the vertical design response spectra differ from those given in Regulatory Guide 1.60 only in the frequency range higher than 33 cps representing the maximum (peak) vertical ground acceleration. The peak vertical ground acceleration was chosen to be two-thirds of the peak horizontal ground acceleration based on the historical evidence. The available earthquake data (Reference 3.7.1-1) indicate that the ratio of the former to the latter ranges from 1/2 to 2/3. Subsection 2.5.2.6.2 contains further discussions of earthquake ground motions.

FSAR Subsection 3.7.1.1.2 will be changed to reflect the basis for divergence.

Q220.2
1 of 1

The Category I structures are founded on sandstone; hence, the peak rock acceleration determined in Section 2.5 represents the peak acceleration associated with SSE. The peak horizontal acceleration associated with SSE for the Category I structure is 0.32 g; that associated with OBE (1/2 SSE) is 0.16 g.

Horizontal design response spectra for spectral damping ratios of 0.02, 0.04, 0.07, and 0.10 for SSE are presented on Figure 3.7.1-3. Horizontal design response spectra for spectral damping ratios of 0.02 and 0.04 for OBE (1/2 SSE) are presented on Figure 3.7.1-4. These design response spectra are defined for the free field and are applicable at the plant grade level (EL. 390.0 ft).

3.7.1.1.2 Vertical Design Response Spectra

The smooth vertical design response spectra were developed following the procedure outlined by Newmark, Blume, and Kapur (Reference 3.7.1-1), which is illustrated on Figure 3.7.1-5. This procedure is based on modifying the horizontal design spectra as follows:

a) The vertical response spectrum is drawn as indicated by the dashed lines on Figure 3.7.1-5, by using two-thirds of the horizontal design spectrum from very low frequencies (i.e., long periods) through points D' and C', both of which pass through the same frequency as points D (frequency = 0.25 hertz) and C (frequency = 2.5 hertz).

b) Line D'C' is extended to point C" at which the vertical design spectrum becomes equal to the horizontal design spectrum.

c) The complete vertical design spectrum is given by the line D'C" BA and then merges into line G, which represents the peak vertical ground acceleration. The point of intersection on line G is approximately at a frequency of 50 hertz for all spectra with different spectral damping ratios.

The vertical design response spectra does not comply with the recommendations of Regulatory Guide 1.60 i.e., for frequencies higher than 33 cps the vertical design response spectra values do not follow the maximum ground acceleration line but decline to 2/3 of the maximum ground acceleration value between 33 and 50 cps and then remains at 2/3 of that value for frequencies higher than 50 cps. The peak vertical ground acceleration ~~based~~ was chosen to be 2/3 of the peak horizontal ground acceleration based on the historical evidence. The available earthquake data (Reference 3.7.1-1) indicate that the ratios of the former to the latter ranges from 1/2 to 2/3. Smooth vertical design response spectra constructed in accordance with the above procedure for an SSE peak vertical ground acceleration of 0.22 g and spectra damping ratios of 0.02, 0.04, 0.07, and 0.10 are presented on Figure 3.7.1-6. These design response spectra are defined for the free field and are applicable at the plant grade level (EL. 390.0 ft).

3.7.1.2 Design Time History

Attachment 1-15
2266A

Question No.

220.3 (3.7.2.6) Regulatory Guide 1.70 states the applicant should indicate the extent to which the procedures for considering the three components of earthquake motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Guide 1.92. Reference or provide this information.

Response

As discussed in Subsection 3.7.2.6 the procedures used for combining the three components of earthquake motion in the seismic design of structures are in conformance with Regulatory Guide 1.92 except for the mat design. The reason for the exception and the alternative method used in the mat design are also presented in Subsection 3.7.2.6.

Attachment 1-16
2266A

Question No.

220.4 (3.7.4) Provide an FSAR Subsection 3.7.4 on seismic instrumentation as outlined in the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.70.

Response

The Seismic Monitoring System is discussed in FSAR Subsection 7.6.1.1.7. FSAR Subsection 3.7.4, Seismic Instrumentation, as required by Regulatory Guide 1.70 will be added to the FSAR and will reference the Seismic Monitoring System.

In order to meet the requirements of NUREG-0800, Inservice Surveillance capability has been provided in the Seismic Monitoring System, such that the system instruments can be demonstrated to be operable through the performance of channel check, calibration and functional testing operation. The frequencies at which these operations will be performed will be provided in Chapter 16.

The FSAR will be revised to reflect the above response.

148.

Q220.4
1 of 3

d) the containment foundation. (H/V-12A/PSR-1200)

ADD Insert "I"

Support Systems

Auxiliary systems supporting the operation of the seismic monitoring system are:

Noninterruptible 120V vital ac system.

7.6.1.1.8 Loose Parts Monitoring System

The Loose Parts Monitoring System has the function of indicating to the operator the presence of loose metallic parts in the primary coolant loop as described in Subsection 4.4.6.1. The system aids the operator in detecting these loose parts before they cause damage to the primary system.

The display instruments utilized by the operator for monitoring loose parts are found on the loose parts detection panel, CP-14, and are listed in Table 7.6-5.

Figure 7.6-20 provides the general system arrangement of the local sensors and preamplifiers and of the equipment and indication available on CP-14 in the Main Control Room. Figure 7.6-21 depicts general sensor location. Located

7.6.1.1.9 Tornado Protection Valves

The Tornado Protection Valves (TPV) are provided on various safety-related HVAC Systems and are described in Section 9.4. The TPV'S are self-contained and close automatically as described in the HVAC System descriptions.

The Tornado Protection Valves (TPV) are capable of automatic return to normal operating position following the restoration of normal outside ambient atmospheric conditions. Operation of the valves during a tornado do not inhibit the valves from subsequently performing normal functions following the tornado. The valves remain open during normal operation allowing the passage of the design quantity of normal airflow. The description of the display instrumentation is found in Subsection 7.5.1.4.8 and it is listed in Table 7.5-17.

7.6.1.1.10 Safety Injection Tank Isolation Valve and Shutdown Cooling System Suction Line Valve Interlocks

Safety Injection Tank (SIT) isolation valve interlocks and Shutdown Cooling System (SDCS) suction line valve interlocks are described and analyzed in CESSAR-F Subsections 7.6.1 and 7.6.2. Final control wiring diagrams, for the above valves and their interlocks are provided on Figures 7.6-22 and 7.6-23.

d) IEEE-279 Section 4.11, 4.12, 4.13 and 4.14 - There are no operating bypasses on this system.

e) IEEE-279 Section 4.15, 4.16 and 4.17 - Not applicable to these systems.

f) IEEE-279 Section 4.20 - Displays provided in the Control Room for these systems are described in Tables 7.6-1,2 and 4.

7.6.2.1.4 RC PB Leak Detection, Seismic and Loose Parts Monitoring Systems

The RC Pressure Boundary Leak Detection System, Seismic Monitoring System, and Loose Parts Monitoring System are designed in accordance with Regulatory Guides 1.45, 1.12 and 1.133 respectively and since they are not protective systems the criteria of IEEE-279 is not totally applicable. Applicable sections are discussed below:

a) 4.1 "Automatic Initiation"

These systems are continuously monitoring plant conditions for conditions which could have an adverse effect on continued plant operation. The parameters monitored by each are discussed in Subsection 7.6.1.

b) 4.3 "Quality Control of Components and Modules"

Refer to Subsection 7.3.2.3.1c.

c) 4.4 "Equipment Qualification"

Refer to Subsection 7.3.2.3.1d.

d) 4.8 "Derivation of Inputs"

Refer to Subsection 7.3.2.3.1h

e) 4.9, 4.10 "Capability of Sensor Checks and Capability for Test and Calibration"

The instrumentation required by these systems is capable of being tested and calibrated in accordance with the methods described in Subsections 7.2.1.1 and 4.4.6.1.

e

ADD I-insert "II"

f) 4.18 "Access to Setpoint Adjustments and Calibration"

Refer to Subsection 7.3.2.3.1r.

g) 4.19 "Identification of Operation"

4.20 "Information Readout"

INSERT "I"

Inservice Serveillance Capability

Capability has been provided in the seismic monitoring system, such that the system instruments can be demonstrated to be operab.?, through the performance of channel check, calibration and functional testing operations.

For the frequencies at which these operations are to be performed refer to Chapter 16.

INSERT "II"

7.6.2.1.4 e) 4.9, 4.10 "Capability of Sensor Checks and Capability for Test and Calibration"

The instrumentation for RC Leak Detection, Loose Parts, and Seismic Monitoring Systems is capable of being tested and calibrated as described in Subsections 7.2.1.1, 4.4.6.1, and 7.6.1.1.7 respectively.

Attachment 1-17
2266A

Question No.

220.5 (3.8.2.5)	In Subsection 3.8.2.5.1 you refer to Tables 3.8.2-3 and 3.8.2-3a. These tables have not been provided. Provide these tables or a schedule for when they will be provided.

Response

The reference to Tables 3.8.2-3 and 3.8.2-3a is incorrect. The correct reference is to Table 3.8.2-2.

The FSAR will be amended to reflect the response to this question.

Q220.5

1of1

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 Hemispherical Head and Cylindrical Shell

The load combinations for the containment are specified in Subsection 3.8.2.3 and summarized in Table 3.8.2-2 ~~Tables 3.8.2-2 and 3.8.2-3a~~. The allowable stresses for each of these load cases are summarized in Table 3.8.2-11. Tables 3.8.2-12 through 3.8.2-17b compare the calculated stresses with the allowable stresses for those critical sections of the containment shown on Figure 3.8.2-18.

Primary membrane allowable stress levels conform to the general design rules as specified in ASME Subsection NE 3131.

Load combination cases eight and nine include an accident condition with seismic loads (OBE or SSE) and pipe rupture loads. As specified in Subsection 3.8.2.4, pipe rupture loads are investigated as local effects. The various load combinations and stress limits for each type of penetration are given in Tables 3.8.2-4, 3.8.2-5, and 3.8.2-6. Load combination cases 10 through 13 include pipe rupture, accident condition with seismic loads (OBE and SSE) and jet impingement loads. General design rules and stress allowables are defined in ASME Subsection NE 3131.2. 12

The load combinations are given in Table 3.8.2-2.

The allowable buckling stresses were determined by the following methods:

a) Allowable Buckling Stresses for Unstiffened Hemispherical and Ellipsoidal Heads

Compressive stress resultants in the top head are compared to the allowable stresses obtained from the paragraphs entitled "Biaxial Compression-Equal Unit Forces" and "Biaxial Compression-Unequal Unit Forces" of the Welding Research Council Bulletin No. 69(6). Using these allowables for the spherical dome is based on the assumption that the dome acts as a cylinder with the radius equal to the radius of the dome.

Three cases are considered (refer to Figure 3.8.2-16):

1) For a uniaxial compressive stress resultant and for biaxial unequal tensile and compressive stress resultants:

$$S_\phi \text{ allowable} = 1.8 \times 10^6 \frac{t}{R}$$

where: t = shell thickness = 1.181 in.

R = dome radius = 900 in.

Therefore S_ϕ allowable = 2362 psi

Attachment 1-18
2266A

Question No.

220.6 (3.8.3.3, 3.8.3.4 3.8.3.5)

As per Regulatory Guide 1.70, Revision 3, provide a discussion of the extent of compliance in the indicated sections with the following:

1. ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures".

2. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings".

3. Subsection NF of the ASME Code, Section III, Division 1.

Response

1. ACI 349 was not the design code for WNP-3. The concrete internal structures were designed in accordance with the provisions of ACI 318-71 as indicated on FSAR pages 3.8-84 and 3.8-95.

2. The design of steel internal structures is in compliance with the requirements of AISC. As indicated on page 3.8-90 the elastic working stress design methods of AISC Part 1 were used.

3. As described on page 3.8-94.

The design and analysis for the NSSS component supports is within NSSS supplier's scope of responsibility. Attaching structures designed to mate with the component supports and allow transfer of loads to the building structure are within AE's scope of responsibility. The ASME Code jurisdictional boundaries have been established such that these attaching structures are outside ASME Code jurisdiction and hence are designed in accordance with the provisions of AISC. Connection bolts between NSSS supports and attaching structures are in accordance with ASME Code, Section II and the additional requirements of Subsection 3.8.3.6.3(e).

Question No.

220.7 (3.8.4.1.3) Provide your schedule for furnishing design information for the Ultimate Heat Sink - Dry Cooling Towers.

Response

The following is the design information for the Ultimate Heat Sink (Dry Cooling Tower) Structure:

Dry Cooling Tower Structure

The Dry Cooling Tower Structure houses the Ultimate Heat Sink (UHS) components (Trains A and B) which provide heat rejection from the Component Cooling Water System (CCWS). The descriptions of the UHS and CCWS are given in Subsections 9.2.5 and 9.2.2, respectively.

The Category I Structure is located to the east of the Reactor Auxiliary Building and occupies an area of 520 ft by 85 ft, having its longitudinal axis oriented in the north-south direction as shown on Figure 1.2-2. It is predominantly a reinforced concrete structure supported on a 4-ft thick foundation mat. It extends from the top of the mat at EL. 390.00 ft to the roof grating level at EL. 437.33 ft. A wind wall that extends 17 ft above the roof level is provided for each longitudinal exterior wall. Within the structure, the heat exchangers are located at EL. 417.5 ft and supported by cross walls which are spaced approximately 49 ft apart. A fan deck is located at EL. 428.50 ft.

Air flow to the tower train is from each side - the east and west sides, and the concrete labyrinth or maze type design that provides exterior missile protection by the line-of-sight approach is provided for the air inlet. Air exhaust from the tower train is through the missile protection roof grating. Wind walls are provided to minimize exhaust air recirculation.

An electrical control building is located between two redundant tower trains and is an integral part of the tower structure housing Train A. The tower structures including the foundation mat for Trains A and B are separated by a 3-in gap to permit an unobstructed seismic displacement during an SSE. The control building is a two-story Category I reinforced concrete structure 85 ft by 24 ft in plan with an overall height of 54 ft. It houses station service transformers and HVAC equipment on the ground floor, and motor control centers on the second floor at EL. 416.5 ft.

Attachment 1-20
2266A

Question No.

220.7 (3.8.4.1.3) (contd.)

There are two Diesel Oil Storage Tank Buildings located adjacent to the Dry Cooling Tower structure, one at the north end of Train A and the other at the south end of Train B. Each tank building is a Category I reinforced concrete structure housing a 92,500-gal capacity diesel oil storage tank and transfer pump located at the ground level. It is supported on a separate foundation mat 41 ft by 36 ft in plan and 4 ft thick. The roof is at EL. 437.33 ft.

Loads for Dry Cooling Tower Structure

Snow or Ice Load: 30 psf per Horizontal Plane Projection.

Maintenance Load: 175 psf on Mat

250 psf on the Fan Deck

Static Analysis for Dry Cooling Tower Structure

The Dry Cooling Tower structure static analysis is performed by the finite element method using the MSC/NASTRAN program. The analyses provide stress resultants (shears, moments, and axial forces per unit length) and deformations for each element under each individual load specified in Subsection 3.8.4.3.1 as well as the load combinations required in Subsection 3.8.4.3.2. A static analysis for earthquake is performed for each of the three orthogonal directions, two horizontal and one vertical, for which the dynamic seismic loads have been separately established. For the superstructure, the SRSS method is used to obtain the combined effects of the three directions of seismic motion occurring simultaneously. For the mat analysis, the same approach as described in the analysis of the Category I Tank foundation mat is used.

FSAR Subsections 3.8.4.1.3, 3.8.4.3.1 and 3.8.4.4.1.2 will be amended to reflect the information of the Dry Cooling Tower Structure.

crack. The portions of high energy piping systems within the steam tunnel satisfy the requirements of Subsection 3.6.2.1.4 and therefore are not postulated to rupture. However, the tunnel is designed for a compartment pressure associated with a hypothetical crack in the main steam line. For additional discussion see Subsection 3.6A.2.3.2.

See Insert 1

3.8.4.1.3 ~~Ultimate Heat Sink Cooling~~ Dry Cooling Towers Structure

3.8.4.1.4 Category I Tank Enclosure Structure

The Tank Enclosure Structure is located as shown on Figure 1.2-2, and serves to support and enclose two 485,000 gal. Refueling Water Storage Tanks and one 380,000 gal. Condensate Storage Tank. The structure consists of a reinforced concrete foundation mat (175 ft long, 66 ft wide and five ft thick) and reinforced concrete enclosure walls ranging in height from 22.5 feet to 53 feet. The top of the mat is at EL. 390. Enclosure walls for the Refueling Water Storage Tanks form an open roof, box type structure in order to retain liquid in the event of tank rupture. Enclosure walls for the Condensate Storage Tank extend to EL. 443.0, and together with the steel missile shield roof, constitute missile protection for the tank. See Figures 3.8.4-2 through 3.8.4-5 for the masonry drawings of the Tank Enclosure Structure.

Insert 1

3.8.4.1.3 Dry Cooling Tower Structure

The Dry Cooling Tower Structure houses the Ultimate Heat Sink (UHS) components (Trains A and B) which provide heat rejection from the Component Cooling Water System (CCWS). The descriptions of the UHS and CCWS are given in Sections 9.2.5 and 9.2.2 respectively.

The Category I Structure is located to the east of the Reactor Auxiliary Building and occupies an area of 520 ft by 85 ft, having its longitudinal axis oriented in the north-south direction as shown in Figure 1.2-2. It is predominantly a reinforced concrete structure supported on a 4-ft thick foundation mat. It extends from the top of the mat at Elevation 390.00 ft to the roof grating level at Elevation 437.33 ft. A wind wall that extends 17 ft above the roof level is provided for each longitudinal exterior wall. Within the structure, the heat exchangers are located at Elevation 417.5 ft and supported by cross walls which are spaced approximately 49 ft apart. A fan deck is located at Elevation 428.50 ft.

Air flow to the tower train is from each side - the east and west sides, and the concrete labyrinth or maze type design that provides exterior missile protection by the line-of-sight approach is provided for the air inlet. Air exhaust from the tower train is through the missile protection roof grating. Wind walls are provided to minimize exhaust air recirculation.

An electrical control building is located between two redundant tower trains and is an integral part of the tower structure housing Train A. The tower structures including the foundation mat for Trains A and B are separated by a 3-in gap to permit an unobstructed seismic displacement during an SSE. The control building is a two-story Category I reinforced concrete structure 85 ft by 24 ft in plan with an overall height of 54 ft. It houses station service transformers and HVAC equipment on the ground floor, and motor control centers on the second floor at Elevation 416.5 ft.

There are two Diesel Oil Storage Tank Buildings located adjacent to the Dry Cooling Tower structure, one at the north end of Train A and the other at the south end of Train B. Each tank building is a Category I reinforced concrete structure housing a 92,500-gal capacity diesel oil storage tank and transfer pump located at the ground level. It is supported on a separate foundation mat 41 ft by 36 ft in plan and 4 ft thick. The roof is at Elevation 437.33 ft.

The following live loads are considered:

Shield Building

Snow, ice or construction loads on dome: 30 psf per horizontal plane projection.

Annulus negative pressure: 0.5 psi during normal operation

Reactor Auxiliary Building

Roofs: 30 psf except roof at EL. 495 for which the design load is 50 psf.

Floors: 100 psf or the equipment load in a designated area, whichever is greater.

Tank Enclosure Structure

Snow or ice load: 30 psf per horizontal plane projection.

See Insert 2

Maintenance load: 100 psf or 4000 lbs, whichever is greater.

T_o = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition. Thermal loads are induced by the thermal gradient existing across walls between the building interior and the ambient external environment during normal operating or shutdown conditions. Both winter and summer conditions are considered as follows:

1) Shield Building

	Summer Temp, F	Winter Temp, F	Shutdown F
Interior face	120	70	50
Exterior face, minimum 7 day average	-	19	19
Exterior face, maximum 7 day average	72	-	-
Exterior face sheltered in RAB	104	70	50

2) Reactor Auxiliary Building

	Summer Temp, F	Winter Temp, F	Shutdown F
Outdoor, air 7 day average	72	19	-
Outdoor, rock below EL. 361.5	50	50	-
Outdoor, rock from EL. 361.5 to grade	Varies linearly from 50 at 361.5 to 72 at grade		

Insert 2

Dry Cooling Tower Structure

Snow or Ice Load:	30 psf per Horizontal Plane Projection.
Maintenance Load:	175 psf on Mat
	250 psf on the Fan Deck

5

SeeInsert 3

3) ~~UHS Cooling~~ Dry Cooling Tower Structure ~~(Later)~~ a

4) Category I Tank Enclosure Structure

The Category I Tank Enclosure Structure is analyzed by the finite element method using the MSC/NASTRAN computer program. The analysis provides in plane stresses (shears, forces per unit length) and deformations in each wall and mat element under the influence of each individual load specified in Subsection 3.8.4.3.1c, as well as the specific combinations of loads as required in Subsection 3.8.4.3.2. A static analysis for earthquake is performed for each of the three orthogonal directions considered in the dynamic analysis. For the enclosure walls, the SRSS Method is used to account for all three directions of seismic motion occurring simultaneously. For the foundation mat analysis, the SRSS Method of combining three component seismic motions is not used because of the non-linear nature of the foundation resistance. Here a simultaneous application of two-component seismic motions, one horizontal and one vertical, is performed with a load multiplier of 1.2 for the horizontal motion. See Subsection 3.7.2.6.

The enclosure walls are analyzed for flexure and shear either as one-way or two-way slabs according to span ratios, and for in-plane shear occurring primarily from the earthquake loading. The base of the enclosure walls is assumed fixed.

The foundation mat is treated as a slab on elastic foundation having no tensile resistance.

The Category I Tank Enclosure Structure is investigated for stability with respect to sliding and overturning in the event of SSE.

b) Dynamic analysis

Analytical techniques for the seismic dynamic analysis are described in Subsection 3.7.2.

c) Missile analysis

The method for missile analysis is described in Section 3.5.

3.8.4.4.2 Structural Steel Framing

The portions of other Category I structures consisting of structural steel framing are analyzed for the effects of load combinations given in Subsection 3.8.4.3, using conventional procedures of structural analysis which are well documented (References 3.8-6 and 3.8-7). The Reactor Auxiliary Building has

Insert 3

(3) Dry Cooling Tower Structure

The Dry Cooling Tower structure static analysis is performed by the finite element method using the MSC/NASTRAN program. The analyses provide stress resultants (shears, moments, and axial forces per unit length) and deformations for each element under each individual load specified in subsection 3.8.4.3.1 as well as the load combinations required in subsection 3.8.4.3.2. A static analysis for earthquake is performed for each of the three orthogonal directions, two horizontal and one vertical, for which the dynamic

have been

seismic loads ~~bee~~ separately established. For the superstructure, the SRSS method is used to obtain the combined effects of the three directions of seismic motion occuring simultaneously. For the mat analysis, the same approach as described in the analysis of the Category I Tank foundation mat is used.

Attachment 1-21
2266A

Question No.

220.8 (3.8.4.8)	Provide a Subsection 3.8.4.8 that discusses the effects of masonry walls on other structures in accordance with SRP 3.8.4 (NUREG-0800).

Response

The response to this Question is being prepared in conjunction with the response to the request for additional information contained in Enclosure 4, item #6. The complete response will be available by December 1982.

Attachment 1-22
2266A

Question No.

220.9 (3.8.5.1) The SRP (NUREG-0800) and Regulatory Guide 1.70 state that a description should be provided of the relationship between adjacent foundations, including any separation provided and the reasons for such separation. Reference or provide this information. Also provide the dry cooling tower information identified as later or provide a schedule for its submittal.

Response

The location of the foundations for Category I structures is shown on Figure 1.2-2. The Category I foundations are: the common mat, the UHS Cooling Tower Foundations, the Condensate and Refueling Water Storage Tank Enclosure Foundation and the Diesel Oil Storage Tank Foundation.

A separate reinforced concrete mat is provided for each Dry Cooling Tower train structure. A foundation mat which is 272 ft long; 85 ft wide, and 4 ft thick supports both the Train A structure and Control Building. The foundation mat for Train B structure measures 248 ft long, 85 ft wide, and 4 ft thick. These two mat foundations are physically separated by a 3-in isolation joint and are both founded entirely on weathered sandstone. The Dry Cooling Tower foundations are located 50 ft from the common mat structures.

Two separate foundations are provided for Category I Diesel Oil Storage Tank enclosures. Each foundation measures 41 ft long by 36 ft wide and 4 ft thick. These foundations are adjacent to the Dry Cooling Tower Train A and Train B foundations. They are physically separated by a 2 in. isolation joint and are founded entirely on weathered standstone.

The Condensate and Refueling Water Storage Tank foundation is located 41.5 feet south of the common mat. The drumming station foundation is located between the two. A 1/2 thick isolation gap is provided between the tank foundation and the drumming station foundation and the common mat.

The separations between adjacent foundations have been provided to insure that there is no interaction under seismic conditions.

FSAR Subsection 3.8.5.1.1 will be amended to reflect this response.

Q220.9
1 of 2

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

insert 1

3.8.5.1.1 Foundations for Category I Structures

A common foundation mat is constructed to support the Reactor Building and the Reactor Auxiliary Building/Fuel Handling Building Category I Structures. This foundation mat is a continuous reinforced concrete structure which is 298 feet wide, 310 feet long and nine feet in thickness. It rests entirely on the fresh sandstone of the Astoria Formation at EL. 326 feet MSL. Horizontal shear loads are transfered to the sandstone by friction at the bottom of the mat and by bearing against the sandstone vertical walls.

~~In addition to the above mentioned common mat, there are individual mat type foundations supporting the UHS Cooling Tower Structure and the Category I Tank Enclosure Structure. (The Category I Tanks are the two Refueling Water Storage Tanks and the Condensate Storage Tank).~~

insert 2

The reinforced concrete mat for the Tank Enclosure Structure is 175 feet long, 66 feet wide and five feet thick. The top of the foundation mat is at elevation 390 feet and the mat rests entirely on weathered sandstone.

insert 3&4

The location of the foundations for Category I structures and their relationship with foundations for non Category I structures are shown on Figure 1.2-2. See Figures 3.8.5-1 through 3.8.5-6 for the typical masonry plans and sections of the Category I foundations.

insert 5

The cylindrical wall of the Shield structure is supported directly on the common foundation mat. A typical reinforcing patern for the wall-mat junction is shown on Figure 3.8.5-7.

The primary shield wall, secondary shield wall and refueling cavity wall of the Reactor Building internal structure, and the Steel Containment Vessel (descriptions of which are given in Subsections 3.8.3 and 3.8.2 respectively) are supported on concrete fill which transfers the loads, by bearing, to the common foundation mat below. For the Reactor Auxiliary Building (and Fuel Handling Building), load transfer to the mat is achieved through the walls and columns. The typical reinforcing pattern for the wall-mat junction is shown on Figure 3.8.5-8.

A drainage system, described in Subsection 3.4.5, is provided and no water proofing membranes are utilized.

Piling or similar foundation concepts are not used.

<u>The UHS Dry Cooling Tower</u>

Later

Q220.9
2 of 2

Insert 1

The location of the foundations for Category I structures is shown on Figure 1.2-2. The Category I foundations are: the common mat, the UHS Cooling Tower Foundations, the Condensate and Refueling Water Storage Tank Enclosure Foundation and the Diesel Oil Storage Tank Foundation.

Insert 2

A separate reinforced concrete mat is provided for each Dry Cooling Tower train structure. A foundation mat which is 272 ft long, 85 ft wide, and 4 ft thick supports both the Train A structure and Control Building. The foundation mat for Train B structure measures 248 ft long, 85 ft wide, and 4 ft thick. These two mat foundations are physically separated by a 3-in isolation joint and are both founded entirely on weathered sandstone. The Dry Cooling Tower foundations are located 50 ft from the common mat structures.

Insert 3

Two separate foundations are provided for Category I Diesel Oil Storage Tank enclosures. Each foundation measures 41 ft long by 36 ft wide and 4 ft thick. These foundations are adjacent to the Dry Cooling Tower train A and train B foundations. They are physically separated by a 2 in. isolation joint and are founded entirely on weathered sandstone.

Insert 4

The Condensate and Refueling Water Storage Tank foundation is located 41.5 feet south of the common mat. The drumming station foundation is located between the two. A ½ thick isolation gap is provided between the tank foundation and the drumming station foundation and the common mat.

Insert 5

The separations between adjacent foundations have been provided to insure that there is no interaction under seismic conditions.

Attachment 1-23
2266A

Question No.

241.1 (2.5.4 and 2.5.5)	In your write-up of FSAR Subsections 2.5.4 and 2.5.5, there are a number of omissions of symbols and functions used in your mathematical expressions. Please thoroughly proofread your text and correct it so that it can be reviewed.

Response

The errors and omissions in Subsections 2.5.4 and 2.5.5 have been corrected. The attached pages identify the corrections.

The FSAR will be amended to reflect the response to this question.

A-series, B-series, D-series, L-series and CT-series borings; and results are presented in Appendix 2.5A.

Typical values for the index properties of the fresh sandstone are summarized in Table 2.5-16.

2.5.4.2.2.1.2 Compressive Strength

The compressive strength of fresh sandstone specimens was determined in the laboratory by uniaxial compression tests. Tests were performed in general accordance with ASTM Standard D2938-71a. The test samples had a length to diameter ratio ranging from 2.0 to 2.5 to assure a fairly uniform stress distribution through the sample. The rate of loading was controlled to minimize the effects of rate of loading on the strength of the sample. The compressive strength of fresh sandstone was calculated by using the empirical relationship presented in ASTM Standard D2938-71a for specimens with L/D ratio equal to 2.

$$\sigma_c = \frac{0.889\ \sigma_m}{0.778 + \frac{0.222D}{L}}$$

where σ_m = compressive strength for L/D at failure

σ_c = compressive strength for L/D = 2
L = length of the specimen
D = diameter of the specimen.

Results are summarized in Table 2.5-16 and details of test procedures are discussed in Appendix 2.5B.

2.5.4.2.2.1.3 Static Deformation Properties

Deformation properties determined include tangent modulus E_T and Poisson's ratio μ_T. *

Tangent Modulus E_T - The tangent modulus E_T was determined by measuring incremental axial strains ($\Delta\epsilon_a$) and corresponding incremental axial stress ($\Delta\sigma_a$) in uniaxial compression tests. The tangent modulus E_T was computed at 50 percent of ultimate uniaxial strength as:

$$E_T = \frac{\Delta\sigma_a}{\Delta\epsilon_a}$$

Results are summarized in Table 2.5-16 and details of test procedures are discussed in Appendix 2.5B.

Poisson's Ratio μ_T - The Poisson's ratio μ_T was determined by measuring incremental axial strain ($\Delta\epsilon_a$) and incremental radial strain ($\Delta\epsilon_r$) in uniaxial compression tests. The Poisson's ratio μ_T was computed at 50 percent ultimate uniaxial strength as:

$$\mu_T = - \frac{\Delta \epsilon r}{\Delta \epsilon a}$$ *

Results are summarized in Table 2.5-16 and details of test procedures are discussed in Appendix 2.5B.

2.5.4.2.2.2 Dynamic Properties

Dynamic properties were investigated through field measurements and laboratory testing. The field investigation consisted of cross-hole shear-wave velocity measurements, and seismic-wave refraction measurements to determine compressional-wave velocities. The laboratory test program consisted of sonic velocity measurements to determine compressional-wave and shear-wave velocities on rock specimens.

2.5.4.2.2.2.1 Field Geophysical Measurements

Compressional-Wave Velocity $(V_p)f$ - Results of compressional-wave velocity measurements for the fresh sandstone are summarized in Table 2.5-16 and details of test procedures are presented in Subsection 2.5.1.2.11 and Appendix 2.5D.

Shear-Wave Velocity $(V_S)f$ - Results of shear-wave velocity measurements for the fresh sandstone are summarized in Table 2.5-16 and details of test procedures are presented in Subsection 2.5.1.2.11 and Appendix 2.5D.

Dynamic Deformation Properties - Based on elastic theory, in situ dynamic tangent modulus E_f and dynamic Poisson's ratio μ_f can be determined from the typical values for the compressional - and shear-wave velocities and for the rock mass ρ as (ASTM D2845-61): *

*

$$\mu_f = (V_P^2 - 2V_S^2)/[2\ (V_P^2 - V_S^2)]$$ *

$$E_f = \frac{[\rho V_{S+}^2 (3V_P^2 - 4V_S^2)]}{V_P^2 - V_S^2}$$ *

Results of calculations are presented in Table 2.5-16.

2.5.4.2.2.2.2 Laboratory Sonic Velocity Measurements

Sonic velocity measurements were conducted on saturated specimens of fresh sandstone axially loaded at 200 lb/in.^{2}. The axial stress was so selected that the mean stress $[(\sigma_1+\sigma_2+\sigma_3)/3]$ in the specimen was similar to the * calculated stress in the field.

Compressional-Wave Velocity $(V_p)_1$ - Results of compressional-wave velocity measurements on fresh sandstone specimens are summarized in Table 2.5-16 and details of test procedures are presented in Appendix 2.5C.

Shear-Wave Velocity $(V_s)_1$ - Results of shear-wave velocity measurements on fresh sandstone specimens are summarized in Table 2.5-16 and details of test procedures are presented in Appendix 2.5C.

Dynamic Deformation Properties - Based on elastic theory, dynamic tangent modulus E_1 and dynamic Poisson's ratio μ_1 can be determined from the representative values for the compressional- and shear-wave velocities and for the selected rock ρ mass using the equation presented in Subsection 2.5.4.2.2.2.1. *

2.5.4.2.3 Properties of Weathered Sandstone

The weathered sandstone below the plant grade is of low to moderate hardness, moderately weathered, yellow brown, coarse to fine grained; see Appendix 2.5A for classification criteria. The mineralogy of the weathered sandstone is described in Appendix 2.5I.

2.5.4.2.3.1 Static Properties

Static properties determined include index properties, compressive strength, and deformation properties.

2.5.4.2.3.1.1 Index Properties

Unit weight, saturation water content, specific gravity, core recovery, and RQD were derived by employing the procedures described in Subsection 2.5.4.2.2.1.1 and are presented in Table 2.5-15.

2.5.4.2.3.1.2 Compressive Strength

The compressive strength was derived by employing the procedures described in Subsection 2.5.4.2.2.1.2 and is presented in Table 2.5-15.

2.5.4.2.3.1.3 Static Deformation Properties

The tangent modulus E_T and Poisson's ratio μ_γ were derived by employing the procedures described in Subsection 2.5.4.2.2.1.3 and are presented in Table 2.5-15. *

2.5.4.2.4.2.1 Field Geophysical Measurements

Tuff Bed 1 could not be identified during seismic-wave refraction measurements. Shear-wave velocity of Tuff Bed 1 was measured at two locations. Results are summarized in Table 2.5-17, and details of test procedures are described in Subsection 2.5.1.2.11 and App. 2.5D.

2.5.4.2.4.2.2 Laboratory Sonic Velocity Measurements

Compressional and shear-wave velocities $(V_p)_1$ and $(V_S)_1$, dynamic tangent modulus E_1, and dynamic Poisson's ratio μ_1 were determined using the procedures described in Subsection ~~2.5.4.2.2.2~~ 2.5.4.2.2.2.2 and are presented in Table 2.5-17.

2.5.4.2.5 Properties of Class I Soil-Cement Fill Material

2.5.4.2.5.1 General

Soil-cement was used for the Class I fill to backfill the access ramps leading to both Reactor Auxiliary Building excavations. In addition, the soil-cement was used to stabilize the north face of the excavation for Reactor Auxiliary Building No. 5. Figure 2.5-117 shows the locations of soil-cement.

Based on a review of the unconfined strength and shear modulus of fresh sandstone previously measured (refer to Appendices 2.5B and 2.5C respectively), values of unconfined compressive strength of 600 lb/in^2 and shear modulus of 400,000 lb/in^2 were selected as representative of the fresh sandstone. A laboratory testing program was performed to determine a blend of processed concrete sand, Portland cement and water which would closely resemble the static and dynamic properties of the fresh sandstone.

2.5.4.2.5.2 Specimen Preparation and Testing

Representative samples were obtained of the Class I concrete sand from the site batch plant. Standard Proctor Compaction Tests (ASTM D-698-78 or ASTM D-558-57) were performed on the Class I concrete sand with cement contents of 0 percent, six percent, eight percent, 10 percent and 12 percent to obtain maximum densities and corresponding water contents (optimum water content). Subsequent to determining the optimum water content, additional compaction samples were prepared by varying the percent compaction for each cement content. These samples were then cured for seven, 28 and 101 days prior to further testing.

Sonic Velocity and Unconfined Compression Tests were performed on the cured samples. In the Sonic Velocity Tests compression and shear wave velocity measurements were made with a load of 16 kg applied to the sonic velocity heads. Unconfined Compression Tests were performed on specimens typically loaded using two different strain rates. Initially, a strain rate of about three percent/hour was applied until the strain reached between 0.05 percent

5

C = cohesion

q = surcharge pressure

N_γ = $\tan^6(45 + \psi/2) + 1$

N_C = $7 \tan^4(45 + \psi/2)$

N_q = $\tan^6(45 + \psi/2)$ *

ψ = rupture angle

2.5.4.10.2 Average Displacement

The average displacement of the mat under combined static and dynamic loading was found by using the equation derived from Boussinesq for the normal displacement of a semi-infinite elastic solid under the action of a normal load (Staff and Zienkiewicz 1969) and can be expressed as:

$$S = \frac{\bar{m} P (1 - \mu^2)}{\sqrt{A}\, E_E \left(1 + \frac{2D}{B}\right)}$$

where: S = average displacement

$\bar{m}$ = displacement coefficient dependent on the shape loaded surface and the distribution of load (Stagg and Zienkiewicz 1969)

μ = Poisson's ratio

E_E = design tangent modulus

P = total load applied

D = depth of embedment

B = width of foundation

A = area of foundation

2.5.4.10.3 Subgrade Modulus

The subgrade modulus K is calculated as:

$$K = \frac{P}{AS}$$

where: P = total load applied

A = total area on which load is applied

2.5.4.11.2 Design Parameters for Fresh Sandstone

2.5.4.11.2.1 Static Parameters

The required design parameters are grouped into four categories: in situ index properties, in situ compressive strength, in situ static deformation properties, and in situ permeability.

2.5.4.11.2.1.1 In Situ Index Properties

The design water content, total and dry unit weight, and laboratory compressional and shear-wave velocities of fresh sandstone are obtained based on the statistical method using values determined on rock specimens in the laboratory; see Subsection 2.5.4.2.2.1.1 The design RQD is obtained based on the statistical method using the RQD recorded on the summary logs. The design values for the field compressional and shear-wave velocities are selected in Subsection 2.5.1.2.11.

2.5.4.11.2.1.2 In Situ Compressive Strength

The representative value of compressive strength of the fresh sandstone cores is obtained based on the statistical method using results of the laboratory compression tests; see Subsection 2.5.4.2.2.1.2. Using a reduction factor RF, the representative compressive strength is reduced to incorporate the effects of in situ geologic discontinuities (Deere et al 1968). The design compressive strength is, thus, stated as:

$$\sigma_E = \sigma_{cs} \text{ (RF)}$$

where: σ_E = design compressive strength

σ_{cs} = statistically selected value of laboratory compressive strength

RF = $\left[(V_P)_f/(V_P)_l\right]^2$

The reduction factor is obtained using the design values for the field and laboratory compressional-wave velocities.

2.5.4.11.2.1.3 In Situ Static Deformation Properties

Two parameters are required to define the in situ static deformation properties, the elastic modulus E_E and the Poisson's ratio μ_E. The frequency and nature of geologic discontinuities are significant factors in determining the properties of fresh sandstone, and their effect can be incorporated in the design equations by use of the reduction factor applied to Young's modulus. *

selection of soil strength parameter for use in the analysis of the man made slopes is discussed in detail in Subsection 2.5.5.2.3.3. The values of elastic modulus E_i and Poisson's ratio μ are selected based on laboratory test results presented in Appendix 2.5B and summarized in Table 2.5-14. *

The cyclic strength is graphically expressed in terms of the variation of cyclic strength versus the number of cycles to cause 5 x 10^{-2} strain. This curve is based on laboratory results discussed in Appendix 2.5C and presented on Figures 2.5-101, 2.5-102, and 2.5-103. Values of dynamic shear modulus are graphically expressed in terms of the variation of shear modulus versus shear strain. This curve is based on field measurements and laboratory test results discussed in Appendix 2.5C and presented on Figure 2.5-104. The damping ratio λ versus shear strain curve based on laboratory test results described in Appendix 2.5C is presented on Figure 2.5C-13.

2.5.5.2.1.2 Weathered Sandstone

The properties of weathered sandstone were selected based on field and laboratory tests. A total of 40 index property tests were made on specimens from the plant area. The results of these tests are discussed in Appendix 2.5B and presented in Tables 2.5B-1 and 2.5B-2. Based on these results, a saturated unit weight of 130 lb/ft^3 and a submerged unit weight of 68 lb/ft^3 were selected.

Static strength parameters were selected based on results of 13 uniaxial compression tests on saturated specimens taken from borings in the plant area. These results are discussed in Appendix 2.5B and presented in Table 2.5B.2. Based on these results, the ultimate strength of the weathered sandstone varies between 325 $lb/in.^2$ and 800 $lb/in.^2$ (i.e., between 46.8 k/ft^2 and 115 k/ft^2). The grain size, nature of cementation, and the fracture pattern in uniaxial compression tests indicate that the weathered sandstone is a C-Ø material. If Ø = 0, the above range of ultimate strengths could be interpreted to correspond to cohesion ranging between C = 23.4 k/ft^2 to 57.6 k/ft^2. Fracture planes in the uniaxial compression tests indicate inclinations between 55° and 70° to the horizontal. Based on these inclinations, the sandstone could be interpreted to have an angle of internal friction Ø ranging between approx. 20° and 50°. This range of angle of internal friction Ø is in agreement with published data for sandstone (e.g., Stagg and Zienkiewicz 1969). Based on these values, a cohesion of 23 k/ft^2 and an angle of internal friction Ø = 16° were selected for analysis of the natural slopes. Analysis of man-made slopes that could have influence on the Category I structures were performed using the more conservative parameters of cohesion equal to 23 k/ft^2 and internal friction Ø = 0°. All of the above values are conservative since both the C and Ø selected represent the minimum values interpreted from the test data. In addition, the range of the tested strength values is very similar to the strength range for fresh sandstone.

Dynamic properties of the weathered sandstone were selected based on field shear-wave velocity measurements. Below EL. 390 (i.e., below the plant grade) the weathered sandstone has a shear-wave velocity greater than 3000 ft/sec; see Appendix 2.5D. Above EL. 390 the shear-wave velocity varies between 3000 ft/sec and 2300 ft/sec. Hence, below EL. 390 the shear modulus, damping ratio, and variation of shear modulus and damping ratio with shear strain are considered to be such as to produce no site-dependent effects. Above EL. 390 the shear moduli

2.5.5.2.2.2.2 Lowe's and Taylor's Methods

The modified Swedish Method suggested by the Corps of Engineers (1970) considers that the direction of forces between the slices remains constant throughout the force polygon. To ascertain the effect of change in the direction of these forces, the slip surface with the smallest factor of safety in Profile 1 was analyzed using a method proposed by Lowe (1967) in which the forces between the slices are considered to be in the same direction as the average chord. For the surface analyzed, the factor of safety showed only an insignificant change (9.9 instead of 10). The same slip surface was also analyzed using the Friction Circle method (Taylor 1948). The analysis gave a somewhat lower factor of safety (9.2 instead of 10) but does not produce a significant change in the results. The procedure utilized in Lowe's method and Taylor's method is shown on Figure 2.5-108.

2.5.5.2.2.2.3 Wedge Method

The stability of Profile 1 was analyzed. For this purpose wedges bounded by the surface of the weathered sandstone, the excavation surface and a planar slip surface were considered. Three planar slip surfaces were selected based on proximity to the Category I structures on a common mat with inclinations of 0°, 6° and 12° to the horizontal; see Figure 2.5-109. All planar surfaces intersect the interpreted weathered rock surface; thus the wedges defined by them do not encounter passive resistance on the downslope side. This is conservative because inclusion of a passive resistance in analysis would increase the factor of safety. Results of the analysis shows that there is an ample factor of safety (FS) with respect to wedge failure (FS = 16.7 to 34).

2.5.5.2.2.2.4 Sensitivity Analysis

To determine the effect of variation in slope geometry and material properties on static slope stability, a sensitivity analysis was made on Profile 1. Calculations based on the contours given on Figure 2.5-100 indicate that the slope of the sandstone surface varies from 19° to 43°. The interpreted slope of Profile 1 is 26°. For sensitivity analysis, Profile 1 was readjusted to a surface slope of 45°.

In the above analyses, the shearing strength of Ø weathered sandstone and fresh sandstone was defined by a cohesion C = 23 k/ft^2 (minimum laboratory value for Ø = 0) and an angle of internal friction ~~Ø~~ = 16° (minimum value based on fracture pattern). For sensitivity analysis, a cohesion C = 23 k/ft and an angle internal friction Ø = 0° were selected. *

The profile was analyzed using both modified Swedish Method and the wedge method. Results of the analyses are shown on Figure 2.5-110. Based on the modified Swedish Method, the smallest factor of safety is 6. Based on the wedge analysis, the smallest factor of safety is 8.5; i.e., in both cases there is ample factor of safety.

2.5.5.2.2.3 Dynamic Slope Stability Analysis

2.5.5.2.2.3.1 General Considerations

Profiles 1, 2, 3, and 3A were analyzed. The dynamic analyses were based on the same general considerations as the static analyses (given in Subsection 2.5.5.2.2.1) except for the magnitude of the lateral force applied to the wall of the excavation. Where applicable, the lateral force is a dynamic force generated by the relative motion of the Category I structures on a common mat. Its magnitude is obtained by a soil-structure interaction analysis described in Subsection 3.7.2.4. In the analysis the force is considered to be distributed uniformly over the vertical face of the excavation.

2.5.5.2.2.3.2 Modified Newark Method

The method proposed by Newark (1965) was applied to Profiles 1, 2, 3, and 3A. On each profile slip surfaces with the minimum static factors of safety were analyzed. As an example, Figure 2.5-111 shows the forces considered and the procedure used for slip surface on Profile 3A. The force NW is the force applied through the center of gravity of the slip surface. For a minimum value of N, the force NW is oriented in a direction perpendicular to the line joining the center of the slip surface with the center of gravity. If the static and dynamic shearing resistance are selected approximately equal, the approximate minimum value of N for a circular slip surface is given by

$$N = (FS - 1) \sin\beta$$ *

where:

FS = static factor of safety
β = angle between vertical and the radius to the center of gravity *
N = a coefficient when multiplied by the weight of the sliding mass gives the total dynamic force required to cause movement.

The minimum values of N were calculated and compared with the resultant value of N for SSE (N = 0.39). Minimum values of N were determined for Profiles 2, 3, and 4. The Newmark method cannot be directly applied to Profile 1 because of the lateral dynamic force that is a function of N.

Profile	Minimum Static Factor of Safety	Value of Coefficient N
1	10	NA
2	16	1.8
3	10	3.3
3A	9	2.5

As can be seen from the above table, the value of N required to cause movement in Profiles 2, 3 and 3A is greater than the value of N for SSE. Therefore, there is an ample factor of safety.

2.5.5.2.2.3.3 Pseudo-Static Analysis

The dynamic stability of Profiles 1, 2, 3 and 3A was also calculated by making pseudo-static analyses on the same slip surfaces. The procedure utilized is illustrated on Figure 2.5-111 for a slip surface for Profile 3A. The procedure is suggested by the U.S. Corps of Engineers (EM-1110-2-1902, April 1970). To each slice the horizontal seismic force was applied in a direction out of the slope, and the vertical seismic force was applied in a direction up

Profile 7 shows the thickest extent of residual soil underlying a cut slope and the lowest safety factors against a pseudo-static slip circle type failure. For these reasons, this profile was subjected to a dynamic slip circle type analysis as described in Subsection 2.5.5.2.3.8, using the average soil strength properties described in Subsection 2.5.5.2.3.3. The location of the worst circle along with the minimum factor of safety is shown on profile 7 (Figure 2.5-114).

Further evidence that the residual soil slopes will remain stable during and after the SSE can be seen by reviewing the residual strength of the residual soil materials. Figure 2.5-119 is a composite plot showing material shear strength under undrained conditions $[(\sigma_1 - \sigma_3)/2]$ versus axial strain for the various unconsolidated-undrained triaxial compression tests that were performed on the material (see Figures 2.5B-15 through 2.5B-18). These curves generally show a peak strength at small strain levels followed by a subsequent leveling off or decrease in resistance until a more or less constant undrained residual strength is reached. Also shown on this figure is the average static shear stress induced along the slip circle sliding surface for profile 7 having the lowest factor of safety under static conditions. (Figure 2.5-114 circle "c" $\phi = 26^{o}$ c = 750 psf τX = 1.4 ksf.). Comparing the induced shear stress against the material residual shear strength indicates that the residual shear strength is higher than the induced stress with a factor of safety of 1.4. *

Therefore, the material has adequate undrained strength to resist sliding during the SSE as shown in previous paragraphs of this section, as well as adequate residual strength to resist sliding following the seismic event.

2.5.5.2.3.5 Static Slip Circle Analysis

The static stability of the permanent man-made slopes against a slip circle type failure was determined using the ICES-LEASE-1 computer program which employs the Fellenius slip circle method.

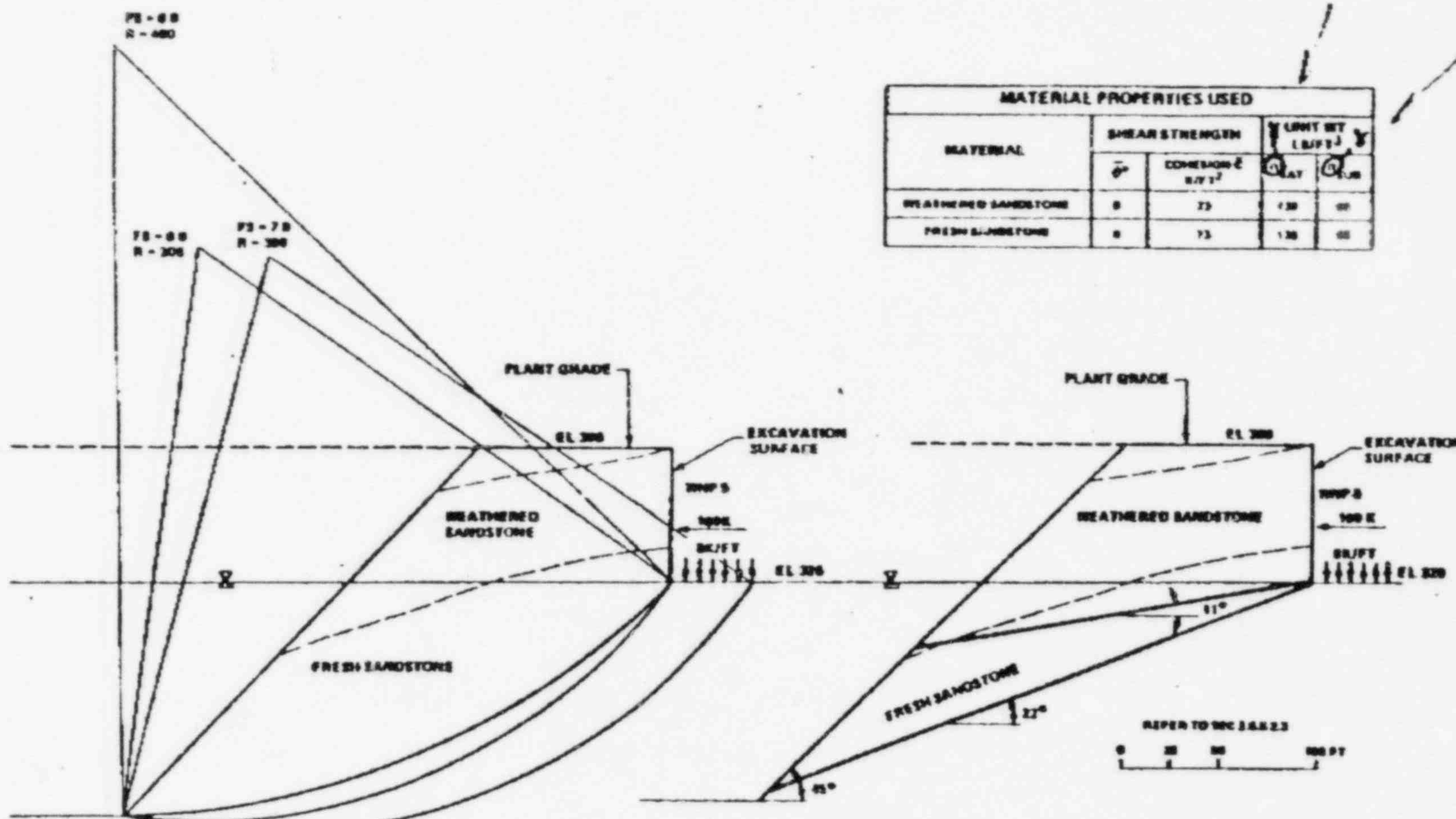
LEASE-1 is a subsystem of ICES designed to perform stability analyses of arbitrary slopes by the method of slices. The failure surfaces are assumed to be arcs of circles. This computer program locates the radius and center having the minimum factor of safety by starting at a specified trial center and using a random search technique.

The static stability analysis was performed on each of the five profiles discussed in Subsection 2.5.5.2.3.4 using each of the three sets of residual soil strength parameters discussed in Subsection 2.5.5.2.3.3.

Input data to the ICES-LEASE program included the following: (a) properties of the soil, including submerged weights, pore pressure, cohesion, angle of internal friction, etc., (b) water level data; (c) total weight of the overlying mass of soil and/or water; (d) the inclination of the failure surfaces at the bottom of the slice.

The factor of safety for the static analyses is defined as the ratio of the moment of the available shearing forces on the trial failure surface to the net moment of the driving force. The minimum allowable safety factor is 1.5 static.

Q241.1



MATERIAL PROPERTIES USED				
MATERIAL	SHEAR STRENGTH		UNIT WT (LB/FT³)	
	$\bar{\phi}$	COHESION $\bar{c}$ (LB/FT²)	γ_{SAT}	γ_{SUB}
WEATHERED SANDSTONE	0	23	130	[illegible]
FRESH SANDSTONE	0	73	130	[illegible]

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NATURAL SLOPES STATIC STABILITY ANALYSIS, PROFILE I, SENSITIVITY ANALYSIS

FIGURE 2.5-110

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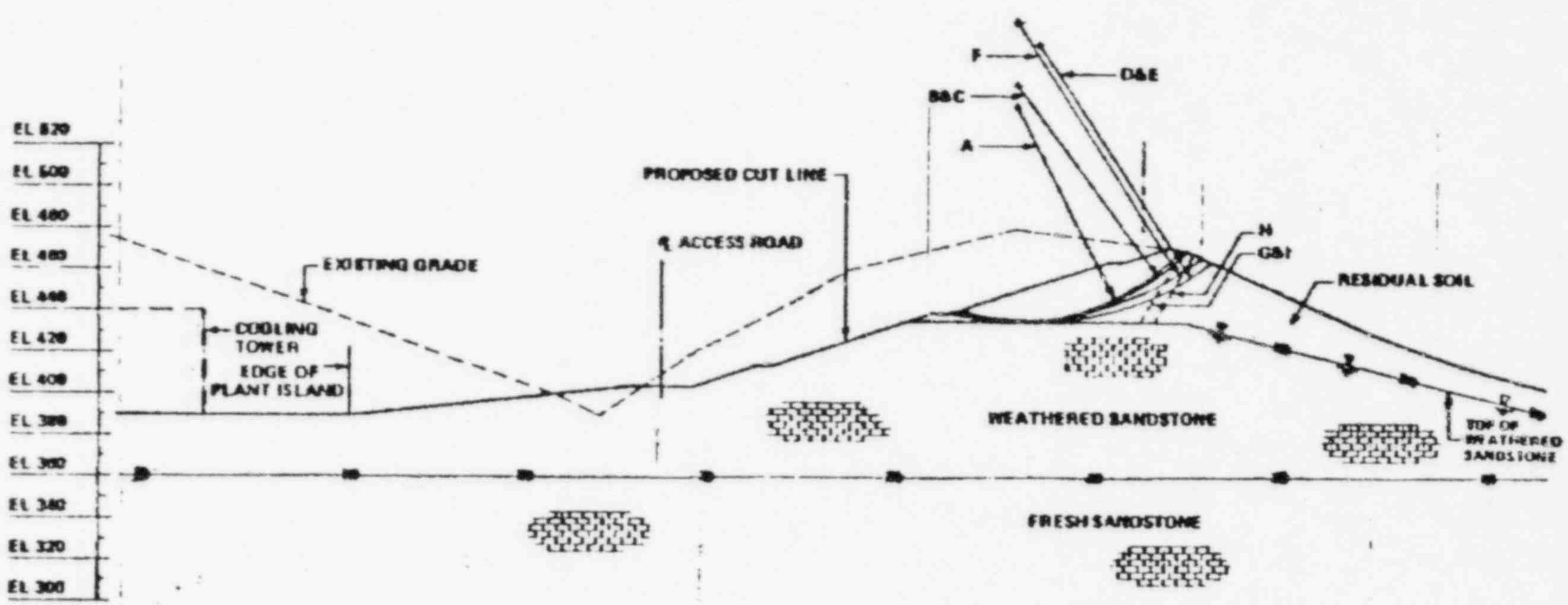

MATERIAL PROPERTIES					FACTOR OF SAFETY					
STRATA		UNIT WEIGHT (#/FT³)	COHESION (#/FT²)	FRICTION ANGLE	STATIC CIRCLE		PSEUDO STATIC CIRCLE		PSEUDO STATIC SLIDING WEDGE	
WEATHERED SANDSTONE		120	22,000	0	CIRCLE	S.F.	CIRCLE	S.F.	CIRCLE	S.F.
RESIDUAL SOIL	MIN	130	1,000	40	A	5.70	D	2.30	G	2.74
	AVG	130	1,000	30	B	4.38	F	1.91	H	1.76
	MAX	130	750	28	C	[illegible]	F	1.92	I	1.41

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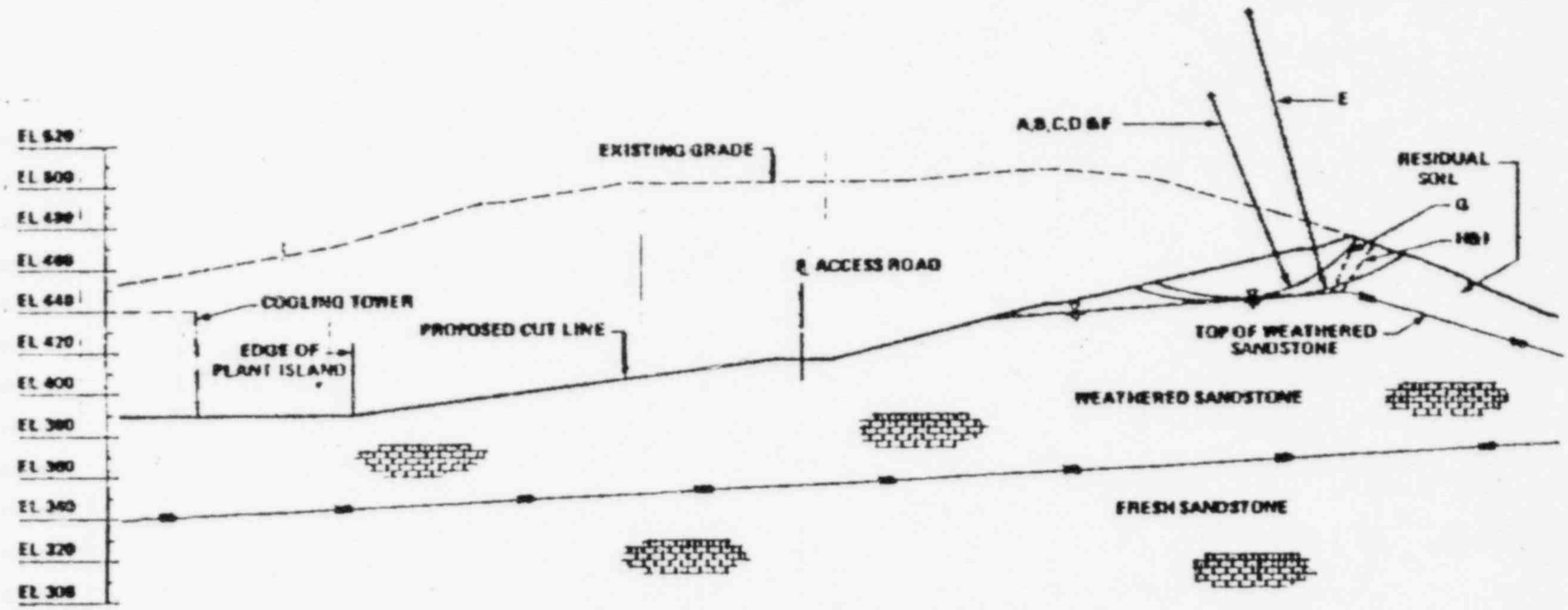
STABILITY ANALYSIS
MAN MADE SLOPES PROFILE NO. 8
FIGURE 2.5-112



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MATERIAL PROPERTIES					FACTOR OF SAFETY					
STRATA		UNIT WEIGHT (#/FT³)	COHESION (#/FT²)	FRICTION ANGLE	STATIC CIRCLE		PSEUDO STATIC CIRCLE		PSEUDO STATIC SLIDING WEDGE	
WEATHERED SANDSTONE		130	23,000	0	CIRCLE	S.F.	CIRCLE	S.F.	CIRCLE	S.F.
RESIDUAL SOIL	MIN.	120	[illegible]	40	A	8.23	D	3.95	G	3.70
	AVG.	120	1,000	30	B	6.70	E	2.95	H	2.30
	MAX.	120	700	20	C	5.35	F	2.04	I	1.80

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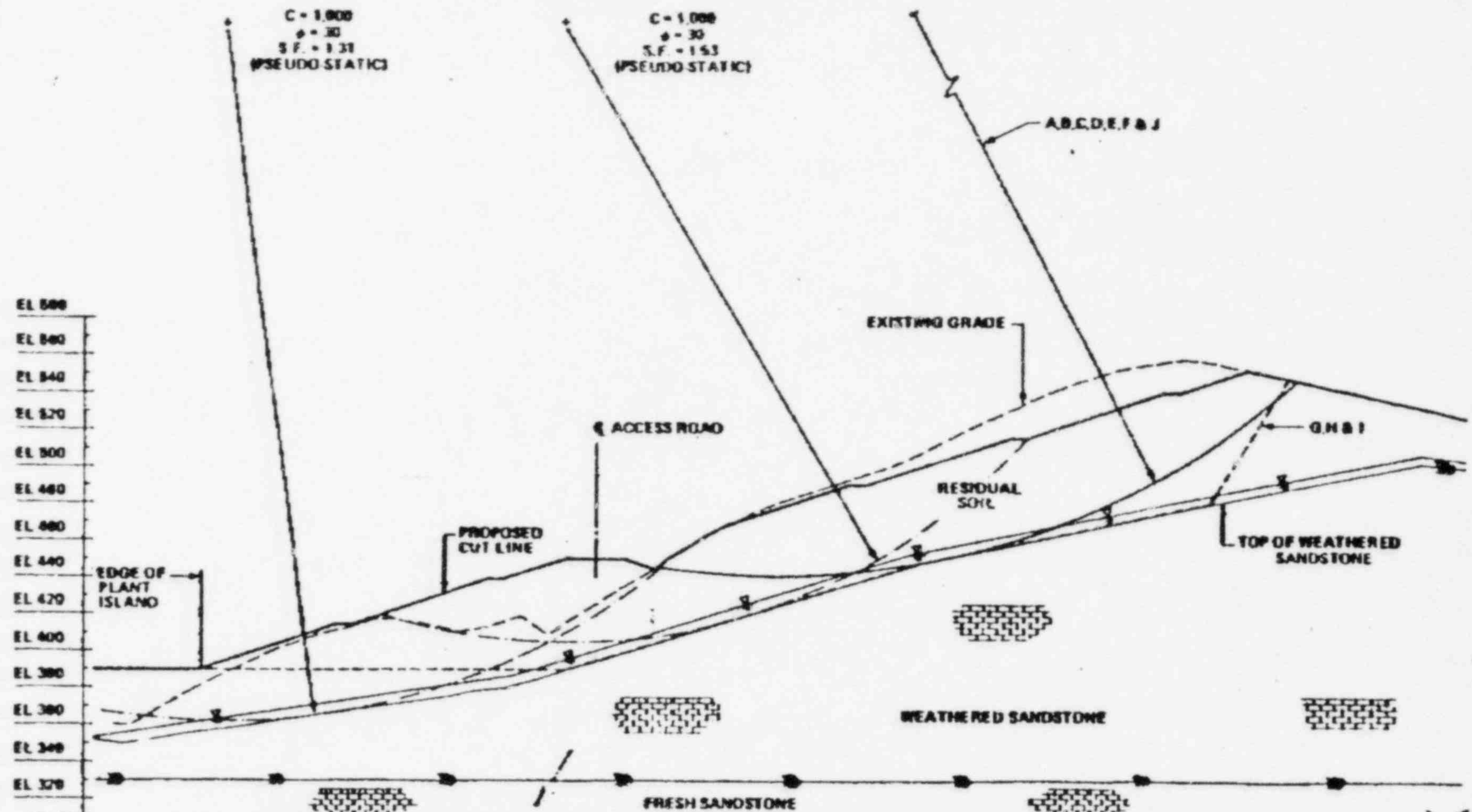
STABILITY ANALYSIS
MAN MADE SLOPES PROFILE NO. 6A
FIGURE 2.5-113



MATERIAL PROPERTIES					FACTOR OF SAFETY							
STRATA		UNIT WEIGHT (lb/ft³)	COHESION (lb/ft²)	FRICTION ANGLE	STATIC CIRCLE		PSEUDO STATIC CIRCLE		PSEUDO STATIC SLIDING WEDGE		DYNAMIC CIRCLE	
WEATHERED SANDSTONE		[illegible]	[illegible]	0	CIRCLE	S.F.	CIRCLE	S.F.	CIRCLE	S.F.	CIRCLE	S.F.
RESIDUAL SOIL	MAX	130	[illegible]	40	A	3.05	D	1.86	G	1.99		
	AVG	130	1,000	30	B	2.90	E	1.30	H	0.98	J	0.64
	MIN	130	750	26	C	2.31	F	1.13	I	1.17		

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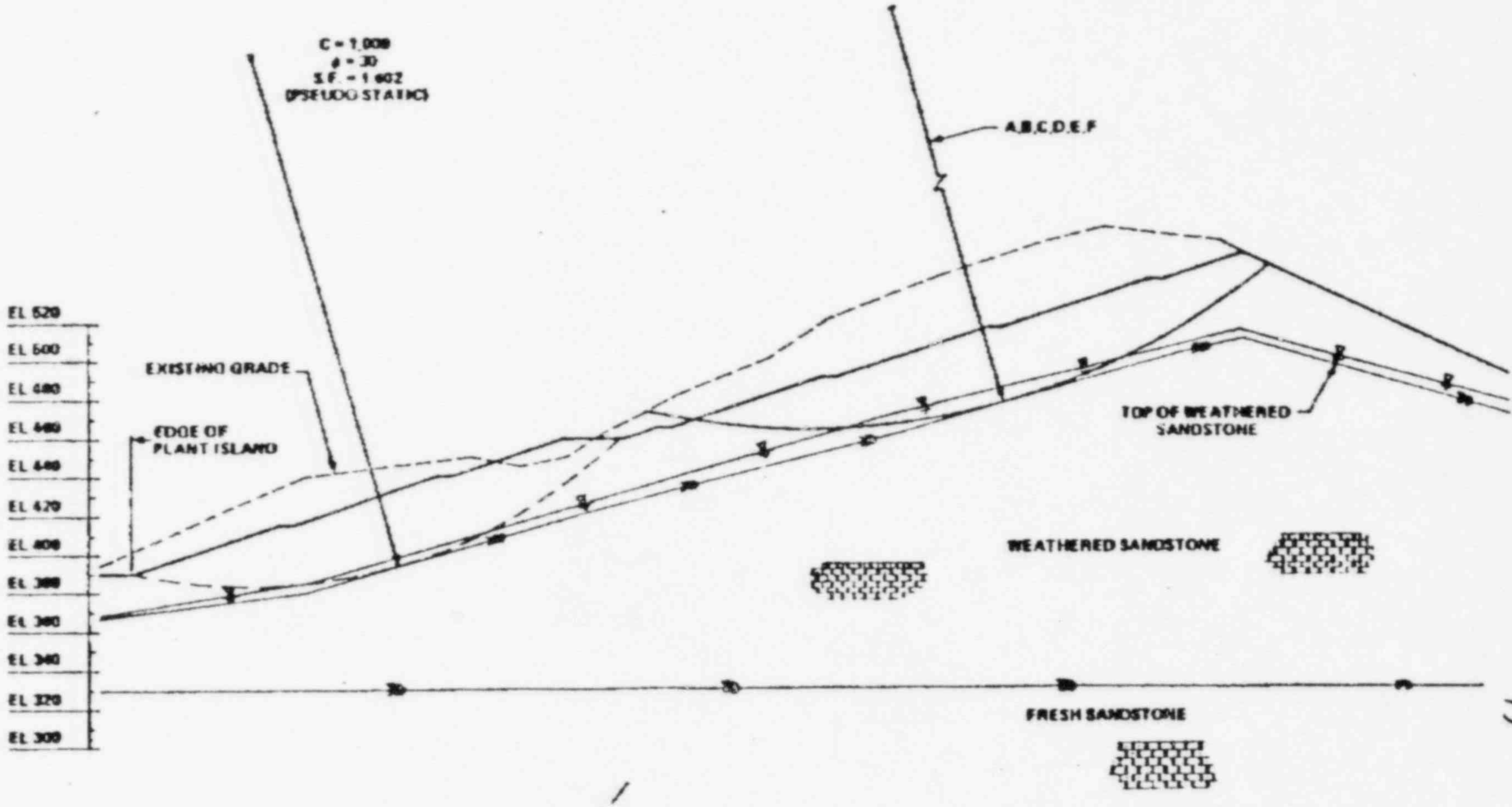
STABILITY ANALYSIS
MAN-MADE SLOPES PROFILE NO 7

FIGURE 2.5-114



MATERIAL PROPERTIES					FACTOR OF SAFETY			
STRATA		UNIT WEIGHT (#/FT.³)	COHESION (#/FT.²)	FRICTION ANGLE	STATIC CIRCLE		PSUEDO STATIC CIRCLE	
WEATHERED SANDSTONE		[illegible]	[illegible]	0	CIRCLE	S.F.	CIRCLE	S.F.
RESIDUAL SOIL	MAX	120	1,000	[illegible]	A	[illegible]	D	[illegible]
	AVE	120	1,000	[illegible]	B	[illegible]	E	[illegible]
	MIN	120	750	[illegible]	C	[illegible]	F	[illegible]

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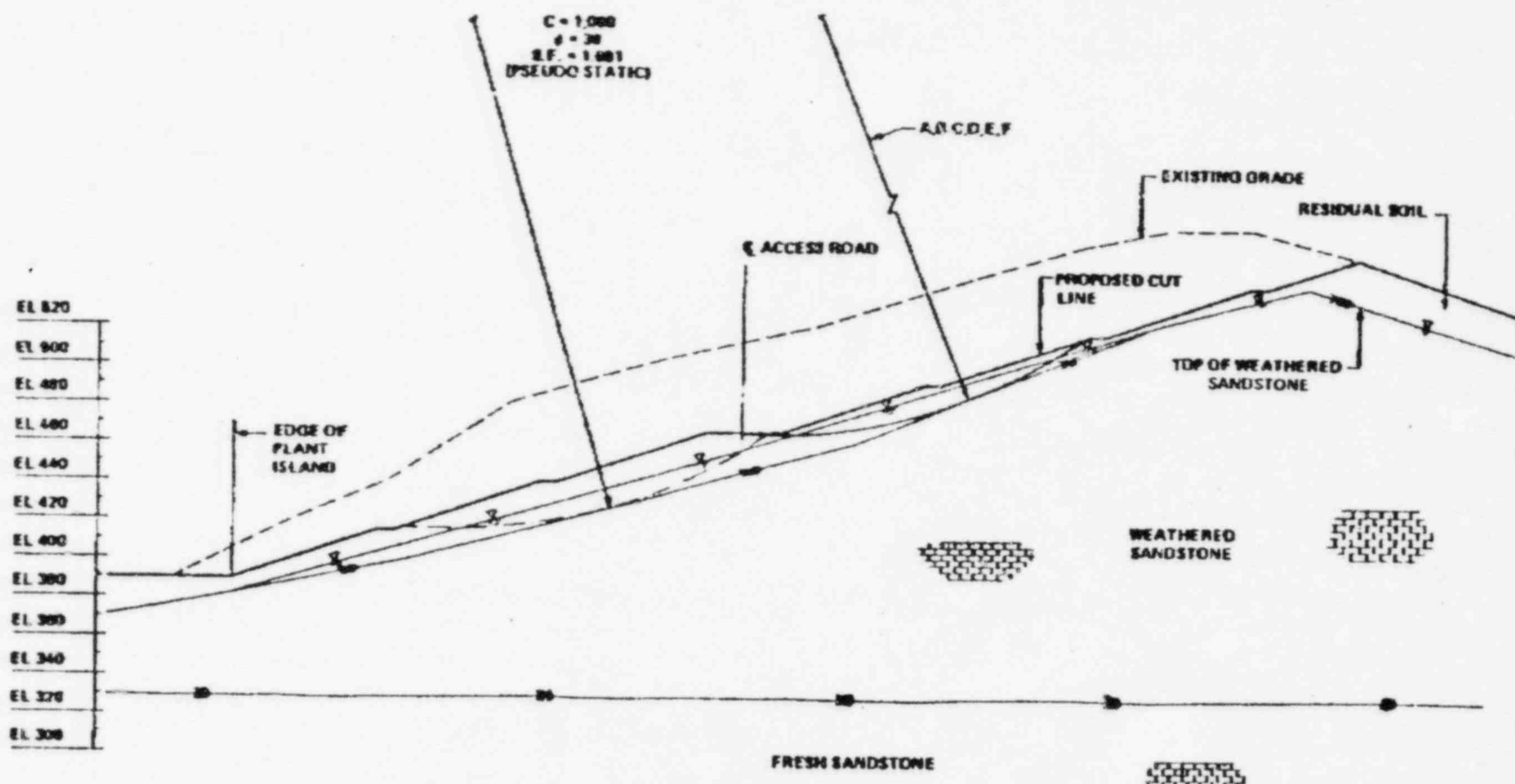
STABILITY ANALYSIS
MAN MADE SLOPES PROFILE NO. 7A
FIGURE 2.5-115



MATERIAL PROPERTIES					FACTOR SAFETY			
STRATA		UNIT WEIGHT (#/FT³)	COHESION (#/FT²)	FRICTION ANGLE	STATIC CIRCLE		PSEUDO STATIC CIRCLE	
WEATHERED SANDSTONE		120	[illegible]	0	CIRCLE	S.F.	CIRCLE	S.F.
RESIDUAL SOIL	MAX	120	1,000	40	A	[illegible]	D	[illegible]
	AVE.	120	1,000	30	B	[illegible]	E	[illegible]
	MIN	120	750	20	C	[illegible]	F	[illegible]

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STABILITY ANALYSIS
MAN MADE SLOPES PROFILE NO. 78
FIGURE 2.5-118

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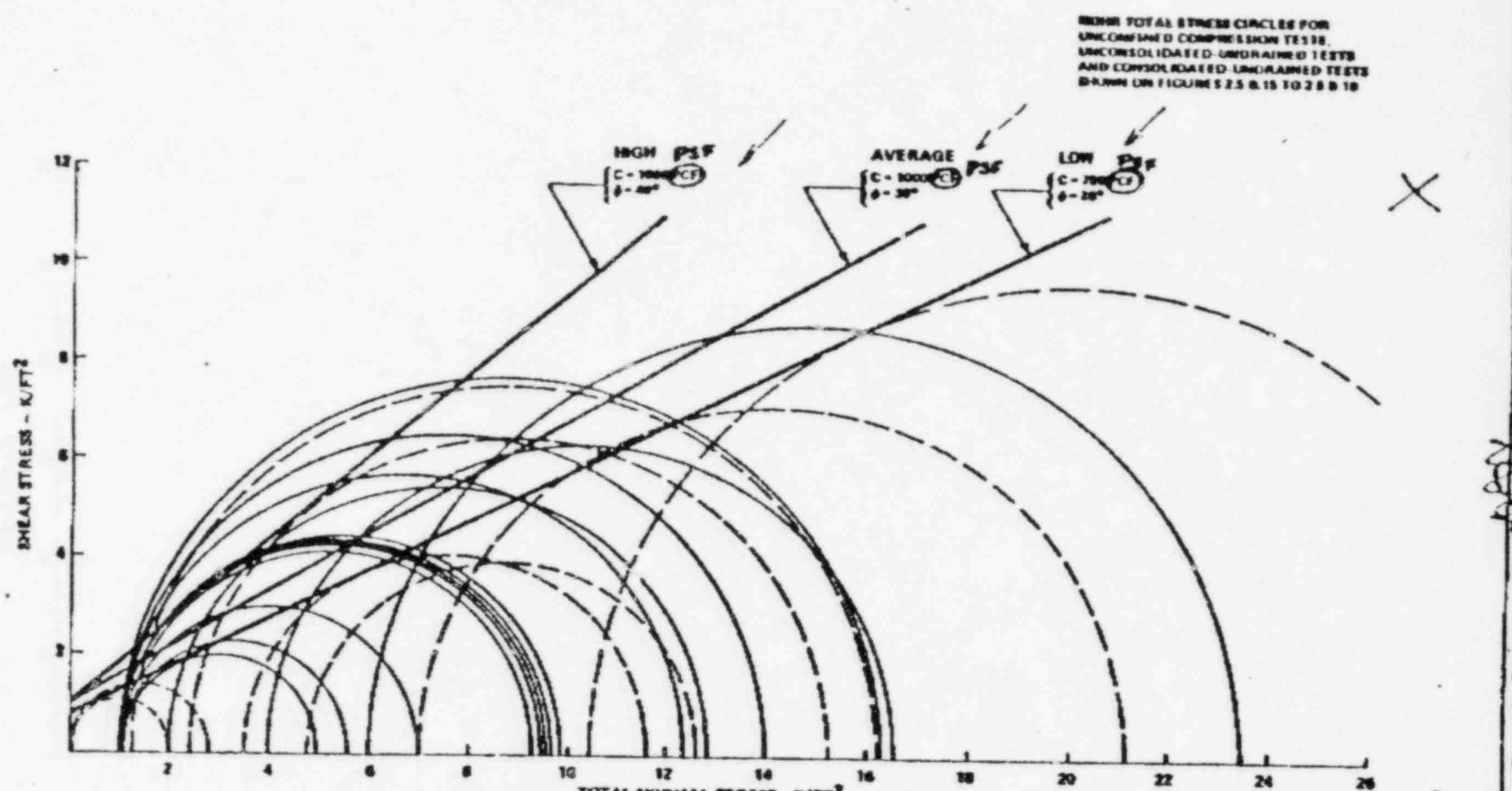


WASHINGTON PUBLIC POWER SUPPLY SYSTEM
Nuclear Projects 3 & 5
FINAL SAFETY ANALYSIS REPORT

RESIDUAL SOIL STRENGTH
FIGURE 2.5-118

Attachment 1-24
2266A

Question No.

241.2 (2.5.4.5.4) You have indicated that, during excavation, you discovered ground cracks adjacent to the excavation of Reactor Auxiliary Building (RAB-3). Provide plans and cross-sectional details clearly identifying the location of these cracks and the corresponding location of sets of pins with strain gauges that you installed to monitor the cracks. Provide the time-displacement monitoring records of the installed strain gauges. Also, discuss in detail, using appropriate drawings, the rock joint and dip pattern in this area. Also discuss if the cracks observed during excavation could pose any landslide or slope stability problems in the future during the plant operation.

Response

The locations of ground cracks and associated sets of pins with strain gauges are shown on Attachment I, Information Drawing - Instrumentation Locations. The results of all the time-displacement monitoring records of the installed strain gauges are shown on Attachment II, Crack Monitoring - Unit No. 3.

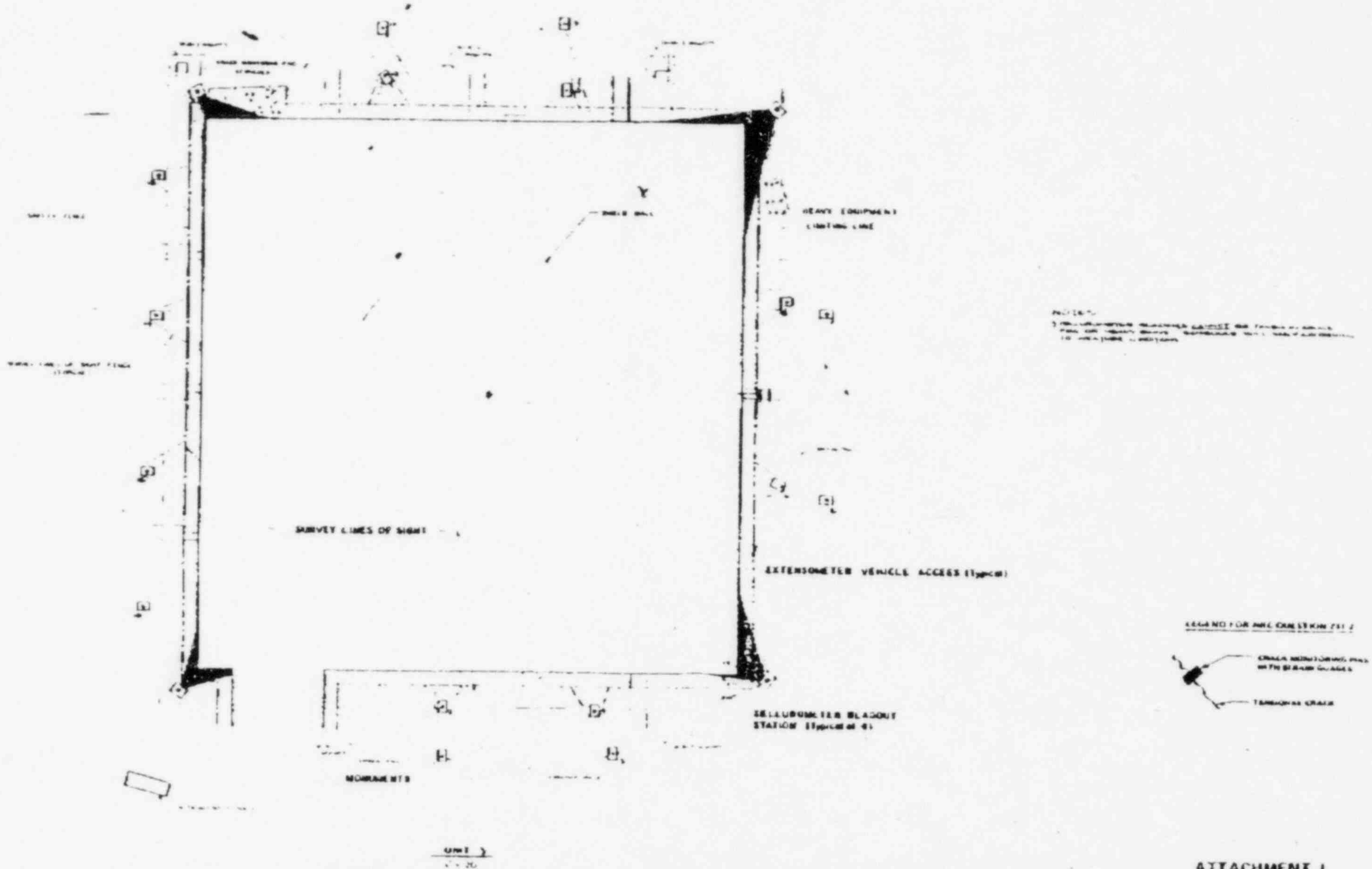
The ground crack monitoring program was initiated in order to provide early warning of rock instability or hazardous conditions as part of the overall Workman Safety - Related Instrumentation Program. Their development was caused by tensional stresses resulting from the excavation of the adjacent sandstone for the RAB-3 foundation. The excavation mapping program summarized in FSAR Appendix 2.5F indicated that extensive cracking did not occur on either the sandstone floor or walls of RAB-3.

The excavation for RAB-3 lies roughly 2,000 feet stratigraphically above the base of the Astoria Formation. The general bedding attitude of the Astoria Formation at the site in N 75 E, 12 N as indicated on FSAR Figure 2.5-53 Site Locality, Geology and Structure map. The Astoria Formation which is predominantly sandstone is typically thick-bedded and massive to faintly cross bedded. The distinguishing of true bedding from cross-bedding in many sandstone outcrops is difficult and results in some locally erratic bedding attitudes. As a result, the stability of the slopes adjacent to the seismic Category I structures were analyzed using conservative material properties under both static and dynamic loadings as described in FSAR Subsection 2.5.5. The results of the analyses indicate that the slopes adjacent to the seismic Category I structures are stable.

Question No.

241.2 (2.5.4.5.4) (contd.)

As a result of the time-displacement monitoring records, the extensive geologic mapping program and the stability analyses, it was concluded that the "hairline" cracks adjacent to the excavation of RAB-3 could not pose any landslide or slope stability problems in the future during plant operation.



Q241.2
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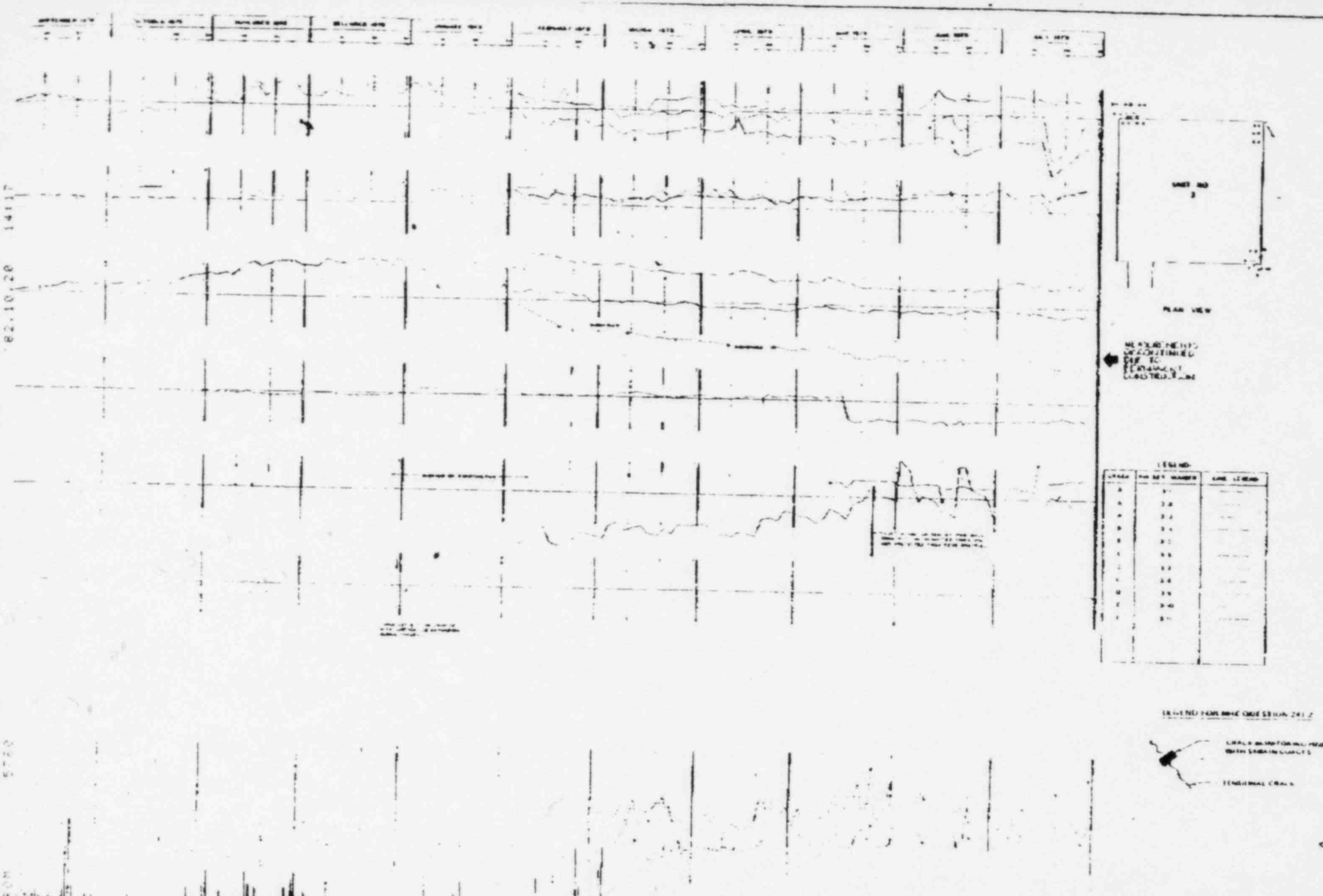
ATTACHMENT I
(NRC QUESTION 241.2)
INFORMATION DRAWING
INSTRUMENTATION LOCATIONS
(WORKMAN SAFETY RELATED)

82.10.20 14:17 5720 FROM



Q241.2
2 of 2

ATTACHMENT II
(NRC QUESTION 241.2)

CRACK MONITORING
UNIT NO. 3

Attachment 1-26
2266A

Question No.

241.3 (2.5.4.10) Provide in tabular form, the as-built dimensions of various seismic Category I structural foundations (length and width), foundation elevations, description (and thickness) of soil/rock characteristics between the foundation base and fresh sandstone, area foundation loads for static and dynamic conditions, corresponding bearing capacities, and resultant factors of safety.

Response

Attached in tabular form is the response to your question.

Q 241.3

1 of 1

CATEGORY I STRUCTURE	FOUNDATION DIMENSIONS (AS-BUILT)	FOUNDATION ELEVATIONS (MSL)	DESCRIPTION OF FOUNDATION BASE AND DEPTH TO FRESH SANDSTONE	FOUNDATION LOADING Static K/Ft^2	FOUNDATION LOADING Dynamic K/Ft^2	ALLOWABLE BEARING CAPACITY Static K/Ft^2	ALLOWABLE BEARING CAPACITY Dynamic K/Ft^2	FACTOR OF SAFETY Static	FACTOR OF SAFETY Dynamic
Condensate/Refueling Water Tank Foundation	178'-0 x 66'-0	Top El 390.00'	5'-0 Thick Mat on Weathered Sandstone 40' to Fresh Sandstone	2.5	6.0	50.0	50.0	20.0	8.3
Dry Cooling Tower & Control Bldg.	85'-0 x 272'-0	Top El 390.00'	4'-0 Thick Mat on Weathered Sandstone 35' to Fresh Sandstone	2.1	4.6	50.0	50.0	23.8	10.9
Shield Building, Containment, Int. Structure, Fuel Handling Bldg. Reactor Aux Bldg on common mat	298'-0 x 310'-0	Top El 335.00'	9'-0 Thick Mat Located in Fresh Sandstone	13.6	17.7	85.0	85.0	6.3	4.8

Attachment 1-27
2266A

Question No.

241.4 (2.5.4.10) From your FSAR write-up, it is not clear to the staff whether you are currently monitoring settlement of various Category I structures or not. Modify your FSAR to identify the settlement monitoring program and give reasons if you are not currently monitoring Category I foundation settlements. Provide location drawings of settlement monuments along with the time vs. settlement plots that include up-to-date rebound and settlement data obtained for all Category I structures where settlements have been monitored.

Response

Since all seismic Category I structures were founded directly on either the fresh or weathered sandstone settlement and rebound was not a factor in the design of the plant. The average displacement of the Category I structures on a common mat was due to the recompression of the fresh sandstone underneath the foundation mat placed at EL. 326'. This displacement was caused by the gradual reloading of the fresh sandstone during construction to the bearing stresses of approximately 81 lb/ft^2. This displacement was elastic in nature and most of it took place during construction. The post-construction displacement is computed using the equation presented in Subsection 2.5.4.10.2 to be less than one-quarter inch and the differential displacement will be even smaller. Therefore, displacement is not a factor in the design of the plant and no instrumentation is required for surveillance of foundations for safety-related structures.

Attachment 1-28
2266A

Question No.

241.5 (2.5.5.1) You mention that the slopes selected for stability analysis were those bounded by the interpreted surface of the weathered sandstone, shown on Figure 2.5-124. You have not included this figure in the FSAR. Provide the figure or a correct reference to its location in the FSAR.

Response

The correct reference is Figure 2.5-100.

FSAR Subsection 2.5.5.1 will be amended to reflect the response to this question.

2.5.5 STABILITY OF SLOPES

The plant grade is established at EL. 390 through a cut and fill operation; see Subsection 2.5.4.5.1. As shown in the excavation profiles presented on Figure 2.5-97, there are natural slopes in the north-south direction and man-made excavation slopes in the east-west direction.

To analyze the static and dynamic stability of natural and man-made slopes, the following procedure was utilized:

a) Select representative slope at the plant location.

b) Select applicable material properties.

c) Examine available analytical methods for static stability and select one method.

d) Analyze the static stability of all slopes by the selected method and identify the slope with smallest factor of safety.

e) Analyze the static stability of the slope with smallest factor of safety by other analytical methods. If required, also analyze other slopes.

f) As required, make a sensitivity analysis on the slopes analyzed in (e) to study the effect of variation in slope geometry and material properties.

g) Examine available methods for dynamic stability and select one method.

h) Analyze the dynamic stability of all slopes by the selected method and identify slope with smallest factor of safety.

i) Analyze the dynamic stability of the slope with smallest factor of safety and, if required of other slopes, by other analytical methods.

j) As required, make sensitivity analysis on the slopes analyzed in (i) to study the effect of variation in slope geometry and material properties.

2.5.5.1 Slope Characteristics

2.5.5.1.1 Selection of Slopes

Natural Slopes - To select profiles for natural slope stability analysis, the areas north and south of plant location below the plant grade (EL. 390) were considered. Below this elevation, materials generally underlying the slopes to the north are Helm Creek Deposit, residual soil, weathered sandstone and fresh sandstone. Generally, the materials underlying the slopes to the south are residual soil, weathered sandstone and fresh sandstone. On both slopes the thickness of residual soil below EL. 390 is relatively small ($<$10 ft) and would not affect the Category I structures; therefore, it is not considered in stability analysis. The Helm Creek Deposit was more than 100 ft. away from the edge of the excavation, and its dynamic properties were not ascertained. Therefore, its properties were not considered in analysis. Thus, the slopes selected for stability analysis were those bounded by the interpreted surface of the weathered sandstone. This surface, interpreted from the borings at the plant location, is shown on Figure 2.5-~~33~~ -100 Contours on Figure 2.5-100 refer to *

Attachment 1-29
2266A

Question No.

241.6 (2.5.5.1) Your bases for selecting the critical cross-sections for slope stability analyses of natural as well as man-made slopes are not adequately justified. Provide sufficient details of your reasons for the selection of critical slopes for the staff's independent evaluation.

Response

The bases for selecting critical cross-sections for slope stability analyses are explained below.

Attachment I shows the locations of critical cross sections selected for detailed slope stability analysis, superimposed on a contour map showing the top of the weathered sandstone.

Since the shear parameters of the weathered sandstone, fresh sandstone and tuff are the same, the stability of the natural slopes is a function of the steepness of the surface of the weathered sandstone. As a result the locations of Profiles 1, 2, 3 and 3A, shown on Attachment I, were selected for stability analysis.

Man-made cut and fill slopes lie primarily in sandstone and only partially in residual soil. All cut and fill slopes are three horizontal to one vertical. The stability of the slopes in residual soil were judged to be more critical than the cuts in rock and were analyzed under both static and dynamic conditions. The location of the residual soil and rock cuts are shown on Figure 2.5-117. Attachment I shows the location of the Profiles 5, 5A, 7, 7A and 7B, judged to be the most critical slopes, and were selected for detailed stability analysis. A detailed discussion of the stratigraphy of the profiles is presented in Subsection 2.5.5.2.3.4.

All factors of safety exceeded the minimum allowable values recommended by the Corps of Engineers Manual EM 1110-2-1902, April 1970.

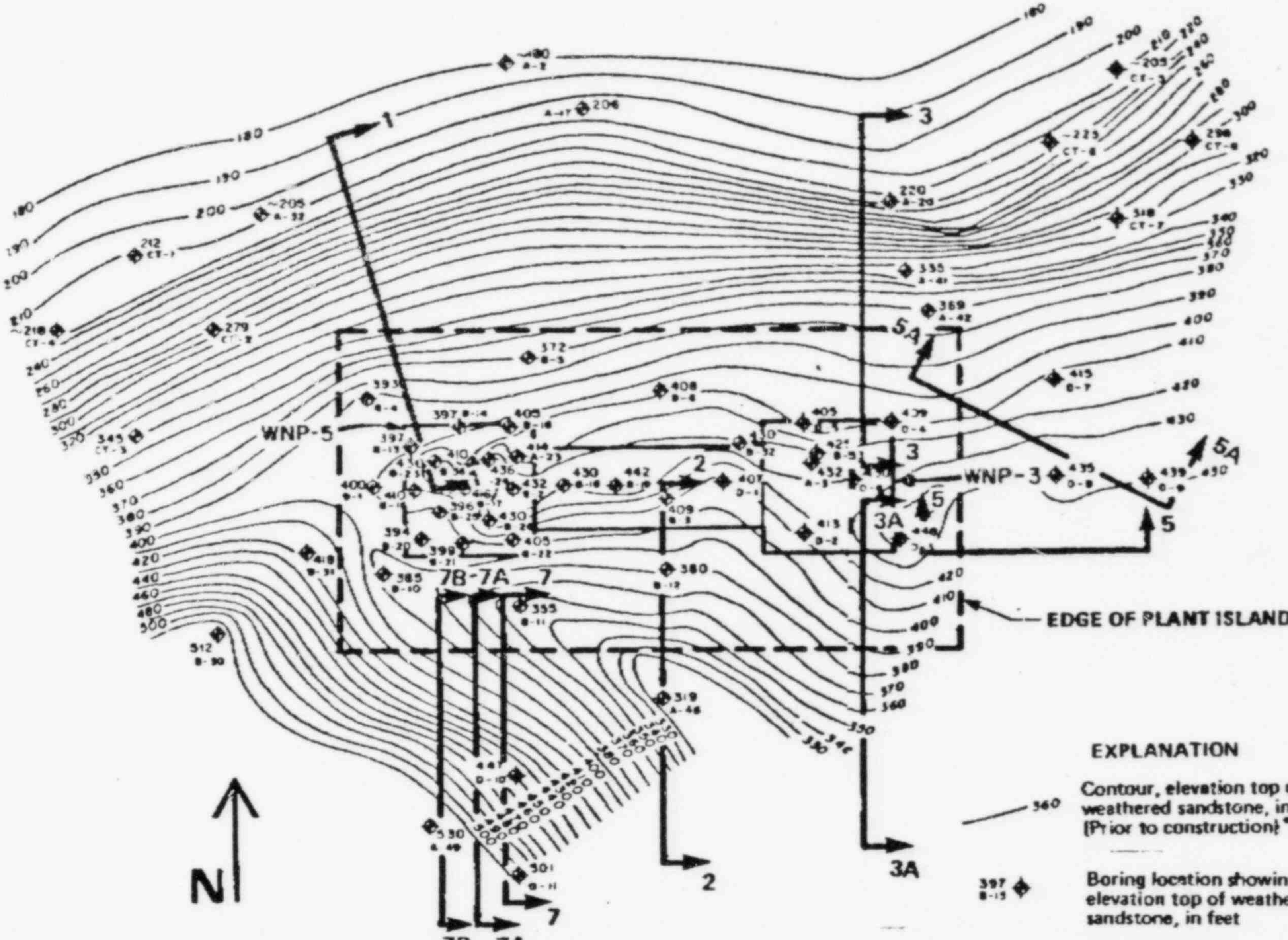
The slope stability analyses were performed using an assumed Reactor Building and Reactor Auxiliary Building common mat loading selected during the PSAR stage of 8.0 ksf. However, the loading has since been increased to slightly less than 14.0 ksf.

In order to verify the results of the stability analyses previously performed additional analyses using the 14.0 ksf common mat loading are deemed necessary.

Attachment 1-30
2266A

Question No.

241.6 (2.5.5.1) (contd.)	A preliminary review of the sensitivity of a 6.0 ksf loading increase on the Reactor Building and Reactor Auxiliary Building common mat indicates that the resulting factors of safety against a stability failure will not be altered significantly and will still exceed the minimum values recommended by the Corps of Engineers. A verification of critical cross-sections for slope stability analyses of natural as well as man-made slopes will be provided subsequent to performing the above analysis. The amended response to Question 241.6 will be provided on January 15, 1983.



EXPLANATION

360 — Contour, elevation top of weathered sandstone, in feet (Prior to construction)*

397 B-15 ◆ Boring location showing elevation top of weathered sandstone, in feet

* At completion of construction, elevation of plant island is +390

NOTE: This contour map is interpreted from presently available geologic boring data and topographic maps.

ATTACHMENT I
(NRC QUESTION 241.6)
PLANT LOCATION CONTOUR MAP
TOP OF WEATHERED SANDSTONE

241.6

Attachment 1-31
2266A

Question No.

241.7 (3.7.1.3) You mention that strain dependent damping coefficients used in the deconvolution analysis are shown on Figures 2.5-87 and 2.5-88. The figure numbers are incorrect.

Response

The strain dependent damping coefficients used in the deconvolution analysis are shown on Figure 2.5-122. The incorrect reference has been corrected.

FSAR Subsection 3.7.1.3 will be amended to reflect the correct reference.

Q241.7
1of1

Comparison of the spectral values of the vertical design spectra with the corresponding response spectra of the artificial accelerogram is made at the same period intervals used for the horizontal component presented in Subsection 3.7.1.2.1.

b) At Foundation Level

To obtain a vertical acceleration time history to be used at the base of the rock-structure interaction system (Subsection 3.7.2.4), a deconvolution analysis of the vertical design time history at the grade level was performed as described in Appendix 3.7.A.

3.7.1.3 Critical Damping Values

The damping ratios, expressed as percent of critical damping, which are used in the analysis of Category I systems and components, are presented in Table 3.7.1-1. These damping ratios are from the recommendations by N.M. Newmark, J.A. Blume, and K.K. Kapur (Reference 3.7.1-1) and are in accordance with Regulatory Guide 1.61. As indicated in Table 3.7.1-1, the two sets of damping ratios are specified to be used respectively for the OBE and SSE seismic analysis.

122

The damping coefficients of foundation rock for the site are taken to be strain level dependent as shown on Figures 2.5-37 ~~and 2.5-38~~, and are used to define damping for the rock elements in the deconvolution analysis (Appendix 3.7.A) and rock-structure interaction analysis (Subsection 3.7.2.4).

3.7.1.4 Supporting Media for Seismic Category I Structures

All seismic Category I structures are supported on either fresh or weathered sandstone.

The following seismic Category I structures are supported on an embedded common mat founded on fresh sandstone:

a) Reactor Auxiliary Building

b) Fuel Handling Building

c) Shield Building

d) Internal Structure

e) Steel Containment Vessel

Attachment 1-32
2266A

Question No.

241.8 (3.7A.2) You have stated that the engineering properties of fresh and weathered sandstone used in the deconvolution analysis are presented in Tables 2.5-7 and 2.5-8. These table numbers are incorrect. Provide the correct reference.

Response

The engineering properties of weathered and fresh sandstone used in the deconvolution analysis are presented in Tables 2.5-15 and 2.5-16, respectively.

FSAR Subsection 3.7A.2 will be amended to reflect the correct reference.

Q241.8

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APPENDIX 3.7A

DECONVOLUTION OF FREE-FIELD ACCELERATION TIME HISTORIES FOR ROCK-STRUCTURE INTERACTION ANALYSIS

3.7A.1 INTRODUCTION

This section describes the deconvolution analyses performed for the finite element rock-structure interaction analysis (Subsection 3.7.2.4) of the Satsop Site. The purpose of these analyses is to determine the base level motion which, when input to the base of the rock column representing the rock characteristics of the site, will produce a time history at the plant grade elevation whose response spectra are virtually identical to the design response spectra defined for the free field at the grade level.

The procedure used in determining the unknown base level motion is based on the continuous solution to the wave equation adapted for use with transient motion through the Fast Fourier Transform techniques. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent linear rock properties using an iterative procedure to obtain values for shear modulus and damping compatible with effective shear strain at the middle of each layer.

The design time histories at the grade level as discussed in Subsection 3.7.1.2 are used. Separate deconvolution analyses are performed for both the horizontal and vertical components of the free-field SSE and 1/2 SSE.

The rock characteristics, mathematical model and description of the analyses are discussed below.

3.7A.2 ROCK CHARACTERISTICS

The rock on which all Category I structures will be founded is fresh sandstone which has a shear wave velocity ranging from 3000 ft/sec. to 4300 ft/sec. Weathered sandstone overlies the fresh sandstone up to the plant grade elevation. The strain-dependent rock properties of the fresh and weathered sandstone were obtained from laboratory tests described in Section 2.5 Figures 2.5-121 and 2.5-122 present the dynamic shear modulus and damping ratio, respectively, as function of shear strain. The engineering properties of fresh and weathered sandstone used in deconvolution analyses are presented in Tables 2.5-~~7~~ 15 and 2.5-~~8~~ 16 respectively. The coefficient of earth pressure at rest is taken to be 0.6.

3.7A.3 MATHEMATICAL MODEL

The mathematical model used in performing the deconvolution analyses of both horizontal and vertical design time histories consists of one-dimensional system of homogenous, viscoelastic layers of infinite horizontal extent with rigid base defined at 570 ft below the plant grade elevation. Each layer is characterized by the thickness, h, mass density, p, ρ elastic modulus, and critical damping ratio, β.

Attachment 1-33
2266A

Question No.

241.9 (3.7A.3) Provide the thickness of the various soil and rock layers, their assumed or measured mass densities, shear wave velocities, moduli, and damping values for the model used in your deconvolution analysis. What is your basis for selecting a 570 ft depth of rock column for this model?

Response

As discussed in Subsection 2.5.4, the subsurface materials below the plant grade at EL. 390 ft involve essentially the weathered and fresh sandstone. Therefore, the deconvolution analysis model consists only of rock layers. As shown in Table 3.7A-1, the rock column, that measures 570 ft from the grade to the base, consists of five 14 ft thick layers and twenty 25 ft thick layers. An average unit weight of 128 lb/ft^3 is used for all rock layers since the range of variation is small as shown in Tables 2.5-15 and 2.5-16. The final iterated values of shear modulus and damping ratio for each layer are presented in Table 3.7A-1. The shear wave velocity associated with each rock layer can be determined from the rock density and the shear modulus of each layer. The average shear wave velocities for the SSE and OBE conditions are 3,848 fps and 3,883 fps respectively. The basis for selecting a 570 ft depth of rock column for the deconvolution analysis is given in Subsection 3.7.2.4.1.

Attachment 1-34
2266A

Question No.

241.10 (3.8.4.4) Describe in detail the procedure you used for calculating the subgrade stiffness that was used in MSC/NASTRAN for the analysis of the Category I Tank Enclosure Structure, and provide the values of the geotechnical parameters for staff review. Also provide the foundation loading results and factors of safety with respect to sliding and overturning of this structure for SSE conditions and reference results to Subsection 2.5.4.

Response

The subgrade stiffness for the analysis of the Category I Tank Enclosure Structure was calculated using

(1) $$Ks = \frac{2.16G}{Rd\ (1- \quad)\ A}$$

where: Ks = Modulus of subgrade reaction

G = 330 Ksi from Figure 2.5-121

= 0.40 from Table 2.5-16

A = Area but to a maximum of $10m^2$

Rd = reduction factor which varies from 5 to 20 to convert from laboratory data to field data.

Equation (1) was obtained from "Dynamics of Bases and Foundations" by D. D. Barker, McGraw Hill, 1962.

Substituting the indicated values into Equation (1) and using the maximum reduction factor for conservatism, a value for Ks of 500 $\#/in^3$ was determined and used in the static analysis for the Tank Enclosure Structure.

The factors of safety against sliding and overturning for the Tank Enclosure Structure and respectively, 1.6 and 1.5 (see Subsection 3.7.2.14) for SSE condition. The maximum bearing pressure (also due to SSE) is $6.0^K/ft^2$. The bearing capacity of weathered sandstone, on which the tank enclosure structure is founded, is $50^K/ft^2$ (factor of safety is 8.3).

Attachment 1-35
2266A

Question No.

241.11 (2.5.4.13) The type, location, and purpose of each instrument used for surveillance of foundations for safety-related structures should be presented.

Response

Since all seismic Category I structures were founded directly on either the fresh or weathered sandstone, settlement was not a factor in the design of the plant. Excessive hydrostatic pressure development will be guarded against through an in-service surveillance program providing for periodic inspections of the vertical drains, horizontal headers and drain tunnels as described in Subsection 3.4.2.2. Therefore, no instrumentation is required for surveillance of foundations for safety-related structures.

Attachment 1-36
2266A

Question No.

250.1 (3.5.1.3) If applicable for the case of turbine destructive overspeed, an analysis should be presented justifying the assumption of only one disc failure. Turbine overspeed acceleration characteristics, statistical distribution of destructive overspeed failure speeds, and related information should be considered in the evaluation of the probability of second wheel failure during the interval of physical disassembly caused by the first failure. Provide or reference this information.

Response

The analysis of the WNP-3 turbine for the case of destructive overspeed is presented in Westinghouse Electric Corp., Turbine Missile Report CT-24903. Subsection 3.3.2 of that report contains the rationale for considering only one disc failure.

Attachment 1-37
2266A

Question No.

250.3 (5.4.2.2)	Is Subsection 5.4.2.2, Steam Generator In-Service Inspection, intended to replace Subsection 5.4.2.2, Description, in CESSAR-F? If this is the case, provide a description of the WNP-3 steam generator. If Subsection 5.4.5 of CESSAR-F is applicable to WNP-3 this should be indicated or the appropriate information should be provided for this section.

Response

Subsection 5.4.2.2 "Steam Generator In-Service Inspection" presented within the WNP-3 FSAR is not intended to replace Subsection 5.4.2.2 "Description" presented within CESSAR-F. In addition Subsection 5.4.5 of CESSAR-F is applicable to WNP-3.

As a result of the aforementioned NRC Question (250.3), WNP-3 FSAR Subsection 5.4.2.2 will be amended.

Q250.3
1 of 2

5.4.2 STEAM GENERATORS

Refer to CESSAR-F Subsection 5.4.2. In addition, the following components and features are used in conjunction with the condensate and feedwater system (Subsection 10.4.7) to maintain proper steam generator water quality.

a) Steam and feedwater materials; Subsection 10.3.6

b) Gland steam condenser Subsection 10.4

c) Condensate demineralizer 10.4.6

d) Steam Generator Blowdown System Subsection 10.4.8

5.4.2.1 Design Basis

Refer to CESSAR-F Subsection 5.4.2.1.

5.4.2.2 Steam Generator In-service Inspection

~~5.4.2.2a Description~~
~~Refer to CESSAR-F Subsections 5.4.2.2 through 5.4.6.4~~
~~Refer to CESSAR-F Subsection 5.4.2.2a~~

5.4.2.2.1 In-service Inspection of ASME III, Class 1 and 2 Parts of the Steam Generators

All Class 1 parts of the steam generators subject to examination are listed in Table 5.2-11 Section C and consist of all pressure-retaining welds, pressure-retaining bolting, integrally welded supports, and components cladding which forms part of the primary pressure boundary.

All Class 2 parts of the steam generators subject to examination are listed in Table 6.6 under Class 2 Pressure Vessels, and consist of all pressure-retaining welds which are gross structural discontinuities, nozzle-to-vessel welds, integrally welded supports, and pressure retaining bolting which form part of the secondary pressure boundary.

5.4.2.2.2 In-service Inspection of Steam Generator Tubing

This subsection addresses the preservice and in-service inspections of steam generator tubing as required by ASME Section XI 1977 Edition with Addenda through Summer 1978. The preservice inspection of steam generator tubing will be done in accordance with Regulatory Guide 1.83 and Appendix IV of ASME Section XI 1977 Edition with Addenda through Summer 1978. The extent of in-service examinations of steam generator tubing will be determined by the code in effect at the time of program submittal.

a) Areas Subject to Examination

The areas subject to examination consist of the entire length of all the U-bend tubes in the steam generators.

2 of 2

indications such as dents which do not preclude probe passage, scratches, and variations in permeability will be recorded but will not be reported. All relevant indications which equal or exceed 20 percent of the tube wall thickness will be reported. All tubes found to contain defects, or tubes containing imperfections that equal or exceed the plugging limit, will be plugged. In accordance with the recommendations of Regulatory Guide 1.83, Rev. 1, the NRC will be informed if, during any inspection, it is determined that more than 10 percent of the total tubes inspected have detectable wall penetration greater than 20 percent, or more than three of the tubes inspected exceed the plugging limit.

f) System Leakage and Hydrostatic Pressure Tests

The system leakage and hydrostatic pressure tests conducted during fabrication and preservice testing are important in determining the initial quality of the steam generator tubing. The steam generator tubing will, throughout the service life of the steam generator see periodic system leakage and hydrostatic tests as defined in Subsection 5.2.4.7. During plant operation, the secondary system will be periodically monitored for radioactivity and the presence of boron using off-line analysis techniques to detect steam generator tubing leakage.

g) In-service Inspection Commitment

The preservice inspection of the steam generator tubing is conducted in accordance with ASME Section XI 1977 Edition with Addenda through Summer 1978. Subsequent in-service inspections will be conducted as indicated in Table 5.2. This program of in-service inspection is based on the requirements of ASME Section XI 1977 Edition with Summer 1978 Addenda. The in-service inspection program for the steam generator tubing will be updated as required.

h) Steam Generator Tube Plugging Limit

The Steam Generator Tube Plugging Limit will be determined by the code referenced in the ISI program.

In addition, the main steam system valves and arrangement are described in Subsection 10.3.2. Main steam line isolation system operability is discussed in Subsection 3.9.3.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

Refer to CESSAR-F Subsection 5.4.6

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

Refer to CESSAR-F Subsections 5.4.7 through 5.4.7.1.1 and Subsection 9.2.10 of this FSAR.

5.4.2.3 Economizer Integrity
Refer to CESSAR-F ~~5.4.6~~ Subsection 5.4.2.3

5.4.2.4 Steam Generator Materials
Refer to CESSAR-F Subsection 5.4.2.4

5.4.2.5 Tests and Inspections
Refer to CESSAR-F Subsection 5.4.2.5

5.4.3 Reactor Coolant Piping
Refer to CESSAR-F Section 5.4.3

5.4.4 Main Steam Line Restrictions
Refer to CESSAR-F Section 5.4.4

5.4.5 Main Steam Line Isolation System
[illegible]

Attachment 1-38
2266A

Question No.

270.1 (3.11.1)	Tables 3.11-1 and 3.11-2 are not complete. Provide the missing information or a schedule for providing it.

Response

Tables 3.11-1 and 3.11-2 will be revised quarterly to incorporate information as it becomes available. The first submittal shall be issued on December 1982 with an overall completion date scheduled for June 1984.

Attachment 1-39
2266A

<u>Question No.</u>

270.2 (3.11.1) Figures 3.11-14 and 3.11-15 have not been submitted. Provide a schedule for submitting these figures.

<u>Response</u>

Figures 3.11-14 and 3.11-15 will be submitted by January 1983.

The FSAR will be amended to reflect the inclusion of the two figures.

Q270.2
1of2

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	CCWS DRY COOLING TOWER ELECTRICAL EQUIPMENT ROOMS	FIGURE 3.11-14

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

Nuclear Projects 3 & 5

FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR FUEL OIL STORAGE TANK ROOMS

FIGURE 3.11-15

Attachment 1-40
2266A

Question No.

271.1 (3.10)	Table 3.10-1 is not complete. Provide the missing information or a schedule for providing it.

Response

Table 3.10-1 will be revised quarterly to incorporate information as it becomes available. First submittal shall be issued December 1982 with an overall completion date scheduled for February 1984.

Attachment 1-41
2266A

Question No.

280.1 (9.5.1.1.6) Table 9.5.1-2 provides a listing of unusually hazardous material. As per Regulatory Guide 1.70, Revision 3, discuss the conditions under which these materials are to be used.

Response

As per Regulatory Guide 1.70, Revision 3, the conditions under which unusually hazardous materials are used is listed in Table 9.5.1-2. The FSAR will be revised for Table 9.5.1-2 to change the column headed "Storage Conditions For Use" to read "Conditions of Use".

Q280.1
1 of 5

TABLE 9.5.1-2

UNUSUALLY HAZARDOUS MATERIAL

Hazardous Material	Approximate Amount	Plant Location	Conditions of Use ~~Storage Condition for Use~~	Expected Time Duration of Use

Note: No flammable liquids are used in the plant systems.

1. COMBUSTIBLE LIQUIDS

Diesel Oil (Storage)	92,500 gal.	Yard	Ambient	Constant
Diesel Oil (Day Tank)	1,100	DG-13	Ambient	Constant
Diesel Oil (Day tank)	1,100	DG-23	Ambient	Constant
Diesel Oil (Engine Sumps)	1,700	DG-11	Ambient	Constant
Diesel Oil (Engine Sumps)	1,700	DG-21	Ambient	Constant
Turbine Lube Oil (Dirty)	25,000	T-20	Ambient	Constant
Lube Oil (Clean)	25,000	T-20	Ambient	Constant
Lube Oil (Main Reservoir)	15,000	T-21	Ambient	Constant
Lube Oil (TDSGFW Pump Res)	835	T-9.2	Ambient	Constant
Lube Oil (TDSGFW Pump Res)	835	T-9.3	Ambient	Constant

2

TABLE 9.5.1-2 (Cont'd)

Hazardous Material	Approximate Amount	Plant Location	~~Storage Condition for Use~~ Conditions of Use	Expected Time Duration of Use
2. STRONG OXIDIZING AGENTS				
Sodium Hypochloride Tanks 2	7500 gal.	Yard	Ambient	Twice a day
3a. COMPRESSED GASES-FLAMMABLE				
Hydrogen Cryogenic	170,000 SCF	Yard	100 psig Gas	Constant
Hydrogen Cylinders - Low Pressure	76,200 SCF	Yard	100 psig Gas	Constant
Hydrogen Cylinders - High Pressure	7,245 SCF	Yard	2,400 psig Gas	Constant
Hydrogen (in bottles)	Supply System	Hot Lab (RAB EL. 402.00 and Cold Lab (Admin. Bldg. EL. 405.00)	Ambient	Continuous
3b. COMPRESSED GASES - NON-FLAMMABLE				
Carbon Dioxide Cryogenic	3 Tons	Yard	100 psig/ 150F	Intermittent
Nitrogen Cryogenic	300,000 SCF	Yard	700 psig/ 120F	Constant
Sulfur Dioxide Cylinders	4 Tons	Yard	35 psig/ 70F	Intermittent

TABLE 9.5.1-2 (Cont'd)

3

Hazardous Material	Approximate Amount	Plant Location	Conditions of Use ~~Storage Condition for Use~~	Expected Time Duration of Use
Oxygen	75,000 SCF	Yard	150 psig maximum Ambient	Continuous in processing one decay tank
Nitrogen (bottles)	1 bottle per post-accident Hydrogen Analyzer	RAB EL. 362.50	Ambient	Monthly for calibration and continuous post-accident
4a. CORROSIVE MATERIALS - ACIDS				
Sulfuric Acid (66 Be)	20,000 gal.	Water Treatment Building	Ambient	Intermittent
Sulfuric Acid (66 Be)	20,000 gal.	Chlorination Facility	Ambient	Intermittent
4b. CORROSIVE MATERIALS - CAUSTIC				
Note: All water treatement - no threat to safety-related areas.				
Sodium Hydroxide (50% Solution)	20,000 gal.	Water Treatment Building	Ambient	Intermittent
Ammonium Hydroxide (28% Solution)		Yard	Ambient	Intermittent
Hydrazine (35% Solution 55 gal. Bottles)	550 gal. Estimate	Turbine Building Zone T-1	Ambient	Intermittent

TABLE 9.5.1-2 (Cont'd)

Hazardous Material	Approximate Amount	Plant Location	Conditions of Use ~~Storage Condition For Use~~	Expected Time Duration of Use
Hydrazine (12 1/2% Solution) Day Tank	100 gal.	Turbine Building Zone T-1	Ambient	Continuous
Hydrazine (5% Solution) Day Tank	100 gal.	Turbine Building Zone T-1	Ambient	Continuous
Ammonium Hydroxide (12-15% Solution) Day Tank	250 gal.	Turbine Building Zone T-1	Ambient	Continuous
Ammonium Hydroxide (12-15% Solution) Day Tank	200 gal.	Turbine Building Zone T-2	Ambient	Continuous
Sodium Hydroxide (50% Solution) Day Tank	588 gal.	Turbine Building Zone T-2	Ambient	Intermittent
Sodium Hydroxide (50% Solution) Day Tank	250 gal.	Reactor Auxiliary Building Zone RW-51	Ambient	Intermittent
Sulfuric Acid (66 Be) Day Tank	288	Turbine Building Zone T-2	Ambient	Intermittent
Sulfuric Acid (66 Be) 2 Day Tanks	100	Reactor Auxiliary Building Zone RW-51	Ambient	Intermittent

TABLE 9.5.1-2 (Cont'd)

Hazardous Material	Approximate Amount	Plant Location	Conditions of Use ~~Storage Condition for Use~~	Expected Time Duration of Use
Ammonium Hydroxide (12-15%) Day Tank	100 gal.	Turbine Building Auxiliary Boiler Zone AB-1	Ambient	Intermittent
Sulfuric Acid (66 Be)	100 gal.	HVAC Chiller Room, Zone T-1.2, Turbine Building	Ambient	Intermittent

5. EXPLOSIVES OR HIGHLY FLAMMABLE MATERIALS

None

Attachment 1-42
2266A

Question No.

280.2 (9.5.1.3) As per Regulatory Guide 1.70, Revision 3, include in the evaluation of fire hazards in each zone, a discussion of the expected rate of fire development and maximum intensity, as these relate to fire detection response sensitivity and automatic and manual firefighting activities. Also, discuss the generation of smoke and other combustion products considering both the toxic and corrosive characteristics.

Response

The type of fire detection system provided for any given fire area is determined by the expected rate of development of the postulated fire in that area. Where a fast developing fire is postulated due to the nature of the combustibles present, a thermal detection system is provided for prompt actuation of the alarms (at local control panel and Main Fire Control Panel) and, when provided, either the pre-action valve, multi-cycle valve, or deluge system. Where a slow developing, smoldering fire is postulated due to the nature of the combustibles present, an ionization detection system is provided for early actuation of the alarms (at local control panel and Main Fire Control Panel) and, when provided, either the pre-action valve, multi-cycle valve, or deluge system.

Provision of an automatic fire suppression system in any given fire area is determined based on fire hazard present in the area as well as the presence of safety-related equipment or cable. Fire areas considered high hazard are those in which the postulated fire is expected to have a rapid rate of development and/or a high maximum intensity.

In area of low fire hazard, a detection system suited to the nature of the postulated fire and manual fire response equipment are provided on the basis of early detection system alarm and prompt fire brigade response, as detailed in Item 9 in the Fire Hazards Analysis. In areas of high fire hazard, an automatic suppression system, a detection system suited to the nature of the postulated fire, and manual fire response equipment are provided on the basis of quick actuation of the suppression system, early detection system alarm and prompt fire brigade response, as detailed in Items 8 and 9 in the Fire Hazards Analysis. In consideration of maintaining manual firefighting capability, the use of materials which generate smoke and other combustion products having toxic and/or corrosive characteristics, especially halogenated plastics, is minimized in the plant design to the extent possible. Such materials are used in restricted quantities and only where non-combustible materials are not available.

FSAR Subsection 9.5.1.3.1 will be amended to reflect the response above.

Q280.2
1of2

c) Containment and consequences of the fire within the considered fire area, and/or its effect on other fire areas.

d) Provision of properly located detectors to sense area fire or smoke conditions so that prompt fire control response can be made.

e) Effective use of manual fire control equipment and backup systems.

f) Adequate smoke removal to permit personnel to enter the fire area, assess the fire condition, and use manual equipment.

g) Effects from the postulated fire on required operation of essential equipment in the area.

h) Protection of redundant systems, equipment or trains, if located in the same fire area, to maintain operability. Separation or isolation of redundant equipment.

The fire hazard analysis was initiated by establishing the fire areas listed in Table 9.5.1-1. These are delineated in Figures 9.5A-1 through 9.5A-29. Boundaries for these areas are based on the nature of occupancy of the plant space, the amount and distribution of combustible materials within the area, and the location of safety-related systems and equipment.

Plant areas important to the plant's capability for safe shutdown, such as electrical penetration area, cable spreading rooms, diesel generator areas, switchgear and battery rooms, are designated as fire areas. Other areas of the plant are considered as fire zones within fire areas.

Fire areas are bounded by walls, floors, ceilings and penetration seals that provide a minimum three-hour fire resistive rating except where they are shown to be unnecessary. Fire zones within fire areas may be bounded entirely or partially with barriers having a three-hour fire resistive rating or less or may be defined by the area limits of fire suppression systems or of occupancies of different nature based upon the results of the fire hazard analysis.

See Insert 1

For each of the designated fire areas listed in Table 9.5.1-1, the fire hazard analysis, as detailed in Appendix 9.5A covers: identification, occupancy, boundaries, systems present in fire area, combustible loading, supplemental building features, fire detection, fire suppression, access and initial response, analysis of effects of postulated fire, equipment lists, assessment of postulated fire consequences, safety evaluation.

The Fire Hazard Analysis has not yet been completed. The safety evaluation is scheduled to be complete in the fourth quarter of 1982 and Appendix 9.5A will then be modified to describe the plants ability to achieve safe shutdown as well as any other changes to design which may occur as a result of this review. In response to the NRC staff request for information, the safety evaluation will consider the criteria of 10CFR50 Appendix R as well as plant design criteria. The amendment to Appendix 9.5A will provide a discussion of compliance with plant design criteria (BTP APCSB 9.5-1 and its Appendix A) as well as a comparison of plant design versus 10CFR50 Appendix R in response to the information request.

Insert 1

The type of fire detection system provided for any given fire area is determined by the expected rate of development of the postulated fire in that area. Where a fast developing fire is postulated due to the nature of the combustibles present, a thermal detection system is provided for prompt actuation of the alarms (at local control panel and Main Fire Control Panel) and, when provided, either the pre-action valve, multi-cycle valve, or deluge system. Where a slow developing, smoldering fire is postulated due to the nature of the combustibles present, an ionization detection system is provided for early actuation of the alarms (at local control panel and Main Fire Control Panel) and, when provided, either the pre-action valve, multi-cycle valve, or deluge system.

Provision of an automatic fire suppression system in any given fire area is determined based on fire hazard present in the area as well as the presence of safety related equipment or cable. Fire areas considered high hazard are those in which the postulated fire is expected to have a rapid rate of development and/or a high maximum intensity.

In areas of low fire hazard, a detection system suited to the nature of the postulated fire and manual fire response equipment are provided on the basis of early detection system alarm and prompt fire brigade response, as detailed in Item 9 in the Fire Hazards Analysis. In areas of high fire hazard, an automatic suppression system, a detection system suited to the nature of the postulated fire, and manual fire response equipment are provided on the basis of quick actuation of the suppression system, early detection system alarm and prompt fire brigade response, as detailed in Items 8 and 9 in the Fire Hazards Analysis. In consideration of maintaining manual firefighting capability, the use of materials which generate smoke and other combustion products having toxic and/or corrosive characteristics, especially halogenated plastics, is minimized in the plant design to the extent possible. Such materials are used in restricted quantities and only where non-combustible materials are not available.

Attachment 1-43
2266A

Question No.

280.3 (9.5.1.3) As per Regulatory Guide 1.70, Revision 3, where automatic fire suppression systems are installed, include an evaluation of the effects of the postulated fire both with and without actuation of the systems.

Response

The effects of a postulated fire in an area protected by an automatic fire suppression system will be limited to fire damage in the immediate area of inception with only limited propagation of the fire through the area. If the automatic fire suppression system has not actuated automatically, the postulated fire might involve cabling and equipment adjacent to the point of inception and the fire could propagate throughout the fire area. The extent of damage beyond the fire area will be limited by the three hour fire rated barriers enclosing the fire area. However, even without actuation of the automatic suppression system in the area, the fire will be sensed by the fire detection system which will alarm fire condition in the Control Room. The Control Room operator will dispatch the Fire Brigade for prompt assessment of the situation and initiation of effective manual firefighting through the use of portable fire extinguishers, hose lines and/or manual actuation of the automatic fire suppression system thus reducing the potential for the fire spread.

FSAR Subsection 9.5.1.3.1 will be amended to reflect the response provided above.

c) Containment and consequences of the fire within the considered fire area, and/or its effect on other fire areas.

d) Provision of properly located detectors to sense area fire or smoke conditions so that prompt fire control response can be made.

e) Effective use of manual fire control equipment and backup systems.

f) Adequate smoke removal to permit personnel to enter the fire area, assess the fire condition, and use manual equipment.

g) Effects from the postulated fire on required operation of essential equipment in the area.

h) Protection of redundant systems, equipment or trains, if located in the same fire area, to maintain operability. Separation or isolation of redundant equipment.

The fire hazard analysis was initiated by establishing the fire areas listed in Table 9.5.1-1. These are delineated in Figures 9.5A-1 through 9.5A-29. Boundaries for these areas are based on the nature of occupancy of the plant space, the amount and distribution of combustible materials within the area, and the location of safety-related systems and equipment.

Plant areas important to the plant's capability for safe shutdown, such as electrical penetration area, cable spreading rooms, diesel generator areas, switchgear and battery rooms, are designated as fire areas. Other areas of the plant are considered as fire zones within fire areas.

Fire areas are bounded by walls, floors, ceilings and penetration seals that provide a minimum three-hour fire resistive rating except where they are shown to be unnecessary. Fire zones within fire areas may be bounded entirely or partially with barriers having a three-hour fire resistive rating or less or may be defined by the area limits of fire suppression systems or of occupancies of different nature based upon the results of the fire hazard analysis.

See Insert 1

For each of the designated fire areas listed in Table 9.5.1-1, the fire hazard analysis, as detailed in Appendix 9.5A covers: identification, occupancy, boundaries, systems present in fire area, combustible loading, supplemental building features, fire detection, fire suppression, access and initial response, analysis of effects of postulated fire, equipment lists, assessment of postulated fire consequences, safety evaluation.

The Fire Hazard Analysis has not yet been completed. The safety evaluation is scheduled to be complete in the fourth quarter of 1982 and Appendix 9.5A will then be modified to describe the plants ability to achieve safe shutdown as well as any other changes to design which may occur as a result of this review. In response to the NRC staff request for information, the safety evaluation will consider the criteria of 10CFR50 Appendix R as well as plant design criteria. The amendment to Appendix 9.5A will provide a discussion of compliance with plant design criteria (BTP APCSB 9.5-1 and its Appendix A) as well as a comparison of plant design versus 10CFR50 Appendix R in response to the information request.

Q280.3
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Insert 1

The effects of a postulated fire in an area protected by an automatic fire suppression system will be limited to fire damage in the immediate area of inception with only limited propagation of the fire through the area. If the automatic fire suppression system has not actuated automatically, the postulated fire might involve cabling and equipment adjacent to the point of inception and the fire could propagate throughout the fire area. The extent of damage beyond the fire area will be limited by the three hour fire rated barriers enclosing the fire area. However, even without actuation of the automatic suppression system in the area, the fire will be sensed by the fire detection system which will alarm fire condition in the Control Room. The control room operator will dispatch the Fire Brigade for prompt assessment of the situation and initiation of effective manual firefighting through the use of portable fire extinguishers, hose lines and/or manual actuation of the automatic fire suppression system thus reducing the potential for the fire spread.

Attachment 1-44
2266A

Question No.

280.4 (9.5.1.3)	Many items in Appendices 9.5-1 through 9.5-21, "Fire Hazard Analyses by Fire Areas" are marked "Later". In the proposed completion of these items, they should be evaluated against the Standard Review Plan, (NUREG-0800).

Response

The items marked "Later" in Appendices 9.5A-1 through 9.5A-21 will be completed according to the following schedule:

Clarification of "Later" for cable trays: The function of cables in cable trays will not be given. Cable trains SA and SB will be protected. Accordingly, all "Later" designations for cable tray functions will be deleted. The function of cable trays will be marked "not applicable".

Function of equipment now designated "Later" will be completed for inclusion into the "Fire Hazard Analyses by Fire Areas" by April 1983.

These items will be reviewed and evaluated against NUREG-0800.

FSAR Appendices 9.5A-1 through 9.5A-21 will be amended to reflect this.

Q280.4

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9. Access and Initial Response (Cont'd)

Fire Zone	Adjacent Areas		Nearest		
	Zone	Corridors	Stair Tower	Elevator	Other Access
C-3	C-5 Via Ladder	None	2	4	None
C-4	C-5	None	1	4	None
C-5	C-4	None	2	4	None
C-6	C-4	None	1	4	None
C-7	Later	None	1	4	None
C-8	Later	None	2	4	None
C-9	Later	None	1, 2	4	None

10. Analysis of Effects of Postulated Fire

a. Combustibles - In Fire Area C, fire hazard combustibles include cable insulation, charcoal, lube oil and grease internal to the equipment.

b. Control of Hazards - The quantity of combustible material which may be involved in the postulated cable fire, and consequently, the magnitude of both the fire and the resultant damage to plant facilities, is reduced by the use of IEEE-383 cables, by the confinement of released combustible liquids through provision of drainage of released oil to area sumps and oil disposal systems.

The introduction of transient combustibles is not considered likely because the Reactor Building is not occupied during normal operation. However, transient combustible materials may be brought into the fire area for maintenance and repair or during plant shutdown. The introduction of transient combustibles resulting from normal maintenance and operation activities is controlled through administrative procedures.

c. Damage Limitation

The extent of damage within and beyond the fire zone on fire is limited by removal of heat, smoke and other products of combustion through use of the Containment Purge System, by partial structural barriers and separation within the area and by three-hour fire rated barriers enclosing the fire area.

2

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Cooling Units	2CC-B524SA	Containment Isolation
	2CH-VP028SNR	Containment Isolation
	2CH-VP029SAR	Containment Isolation
	2EC-B012-SA	Hydrogen$_2$ Purge Supply
	2EC-B010-SA	Hydrogen$_2$ Purge Supply
Transmitter	PT-268	Reactor Drain Tank Pressure
Thermocouple	TE-221	Reactor Heat Exchanger Letdown Outlet Pressure
Cable Trays	P27-NA	(Later) N/A
	P29-NA	(Later)
	C69-NA	(Later)
	C71-NA	(Later)
	L35-NA	(Later)
	P28-NB	(Later)
	C70-NB	(Later)
	L36-NB	(Later)
	L38-NB	(Later)

Fire Zone C-2, Steam Generator 1-SA Area

Equipment or Component	ID No. and Division	Function
Pumps	1A-SN	Reactor Coolant Pump
	1B-SN	Reactor Coolant Pump
Steam Generator	1-SA	Steam Generation
Heat Exchangers	1-SA	Steam Generator Blowdown
	3-SA	Steam Generator Blowdown
Valves	1SI-VP092SAR	Shutdown Cooling System Isolation
	2PV-B010-SA	Hydrogen$_2$ Purge Supply
	2PV-B016-SA	Hydrogen$_2$ Purge Supply

Fire Zone C-3, Steam Generator 2-SB Area

Equipment or Component	ID No. and Division	Function
Pumps	2A-SN	Reactor Coolant Pump
	2B-SN	Reactor Coolant Pump
Steam Generator	2-SB	Steam Generation
Heat Exchangers	2-SB	Steam Generator Blowdown
	4-SB	Steam Generator Blowdown
Valve	1SI-VP098SBR	Shutdown Cooling System Isolation

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone C-4, Containment Fan Cooler Area		
Unit Cooler	UC-5	CEDM Unit Cooler
Fan Cooler	FC-1A FC-1C	(Later) (Later)
Fans	UC-5FB UC-5FB	(Later) (Later)
Tanks	3-SA/B 4-SA/B	Safety Injection Safety Injection
Valves	1SI-VP104SAR 1SI-VP107SAR 1SI-VP123SBR 1SI-VP124SBR RLH-HV-6702S	Safety Injection Tank Isolation Safety Injection Tank Isolation Leakage Relief of Valve VP106SA/BR Leakage Relief of Valve VP103SA/BR Fuel Pool (C)
Transmitters	PT-100X	Pressurizer Pressure Control
	PT-100Y	Pressurizer Pressure Control
	PT-101A	Pressurizer Pressure Protective
	PT-101B	Pressurizer Pressure Protective
	PT-101D	Pressurizer Pressure Protective
	PT-102A	Pressurizer Pressure Protective
	PT-102B	Pressurizer Pressure Protective
	PT-102D	Pressurizer Pressure Protective
	PT-103	Restricted Range Pressure
	PT-104	Restricted Range Pressure
	PT-106	Restricted Range Pressure
	PT-199A	Pressurizer Pressure (Supplementary System)

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
	PT-199B	Pressurizer Pressure (Supplementary Protection System)
	PT-199D	Pressurizer Pressure (Supplementary Protection System)
	LT-110X	Pressurizer Level Control
	LT-110Y	Pressurizer Level Control
	PT-1013A	Steam Generator Pressure Protective
	PT-1013B	Steam Generator Pressure Protective
	PT-1013C	Steam Generator Pressure Protective
	PT-1013D	Steam Generator Pressure Protective
	LT-1113A	Steam Generator Level Protective (Wide Range)
	LT-1113B	Steam Generator Level Protective (Wide Range)
	LT-1113C	Steam Generator Level Protective (Wide Range)
	LT-1113D	Steam Generator Level Protective (Wide Range)
	LT-1114A	Steam Generator Level Protective
	LT-1114B	Steam Generator Level Protective
	LT-1114C	Steam Generator Level Protective
	LT-1114D	Steam Generator Level Protective
Cable Trays	P31-SA	(Later) N/A
	C73-SA	(Later)
	C75-SA	(Later)
	L39-SA	(Later)
	L41-SA	(Later)
	P30-SB	(Later)
	C72-SB	(Later)
	C74-SB	(Later)
	L42-SB	(Later)
	L44-SB	(Later)

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone C-5, Containment Fan Cooler Area		
Cont. Fan Cooler	FC-1B	(Later)
Tanks	1-SA/B	Safety Injection
	2-SA/B	Safety Injection
Valves	1SI-VP110SBR	Safety Injection Tank Isolation
	1SI-VP113SBR	Safety Injection Tank Isolation
	1SI-VP121SBR	Leakage Relief of Valve VP112SA/BR
	1SI-VP122CBR	Leakage Relief of Valve VP109SA/SB
	1CH-VP010SAR	Containment Isolation Valve
	1CH-VP001SBR	Pressure Control Valve
	1CH-VP039SAR	Auxiliary Spray to Pressurizer Isolation Valve
	1CH-VP009SBR	Safety Injection Isolation Valve
Transmitters	PT-1023A	Steam Generator Pressure Protective
	PT-1023B	Steam Generator Pressure Protective
	PT-1023C	Steam Generator Pressure Protective
	PT-1023D	Steam Generator Pressure Protective
	LT-1124A	Steam Generator Level Protective
	LT-1124B	Steam Generator Level Protective
	LT-1124C	Steam Generator Level Protective
	LT-1124D	Steam Generator Level Protective
Differential Pressure	PDT-240	Charging Back Pressure Valve
Cable Trays	P31-SA	(Later) N/A
	C73-SA	(Later)
	C75-SA	(Later)
	L39-SA	(Later)
	L41-SA	(Later)
	P30-SB	(Later)
	C72-SB	(Later)
	C74-SB	(Later)
	L42-SB	(Later)
	L44-SB	(Later)
Fire Zone C-6, Pressurizer Room		
Pressurizer	SA/SB	Primary Loop Pressurization
Valves Exchanger 4	3CC-B006SN	Steam Generator Blowdown Heat Isolation Valve
Exchanger 4	3CC-B008SN	Steam Generator Blowdown Heat Isolation Valve

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone C-6, Pressurizer Room (Cont'd)		
	3CC-B009SN	Steam Generator Blowdown Heat Exchanger 2 Isolation Valve
	3CC-B010SN	Steam Generator Blowdown Heat Exchanger 2 Isolation Valve
	3CC-B012SN	Steam Generator Blowdown Heat Exchanger 1 Isolation Valves
	3CC-B015SN	Steam Generator Blowdown Heat Exchanger 1 Isolation Valves
	3CC-B018SN	Steam Generator Blowdown Heat Exchanger 3 Isolation Valves
	3CC-B019SN	Steam Generator Blowdown Heat Exchanger 3 Isolation Valves
Fire Zone C-7, Head Assembly Laydown Area and Upper Level		
Valves	1RC-P501SA/BR	Pressurizer Spray Line Isolation
	1RC-P502SA/BR	Pressurizer Spray Line Isolation
	2PV-B010SA	$Hydrogen_2$ Purge Exhaust
Transmitters	PT-101C	Pressurizer Pressure Protective
	PT-102C	Pressurizer Pressure Protective
	PT-103	Restricted Range Pressure
	PT-199C	Pressurizer Pressure (Supplementary Protection System)
Hoisting Winch		(Later)
Hydraulic Power Pack		(Later)
Fire Zone C-8, Upper Levels		
Tanks	1-SA/B	Safety Injection
	2-SA/B	Safety Injection
Fan Cooler	FC-1D	(Later)
Hydrogen Recombiner	SB	(Later)
Valves	2PV-B109SA	Hydrogen Purge Exhaust
	2PV-B064SA	Containment Purge Exhaust
	2PV-B112SA	Containment Purge Exhaust

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone C-9, Reactor Pressure Vessel Area		
Reactor Pressure Vessel	SA/SB	Containment
Fans	E-1A	Reactor Cavity Cooling Supply
Fans	E-1B	Reactor Cavity Cooling Supply
Lift Rig		(Later)
Reactor Drain Tank		(Later)
Core Support Barrel Storage Stand		(Later)
Valve	KIH-HV-6701S	Fuel Pool
Transmitters	RE-HV-6701AS RE-HV-6702AS RE-HV-6701BS RE-HV-6702BS	Fuel Pool Fuel Pool Fuel Pool Fuel Pool

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment in Fire Area C may be damaged or lose function due to a fire.

The fire in Zone C-5 is limited to that zone because of the slow burning nature of the cable and the non-self propagating nature of IEEE-383 cable insulation. Redundant equipment in Fire Zone C-4 is beyond the influence of the postulated fire and continues to remain available.

The postulated fire and its associated effects in Fire Zone C-2 is limited to the immediate vicinity of the exposed reactor coolant pump by the limited amount of fuel, the reinforced concrete cubicle walls and pedestal design. The other pump 1B(SN) approximately 30 ft away and Steam Generator 1SA, in Fire Zone C-2, and the redundant equipment in Fire Zone C-3 are beyond the influence of the postulated fire and continue to remain available.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

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10. Analysis of Effects of Postulated Fire (Cont'd)

e. Fire Effects - The fire is contained within the three-hour fire barriers of Fire Area CSA. The redundant Fire Area CSB is enclosed and separated from CSA by three-hour fire resistive boundaries and is therefore not exposed by the fire.

f. Equipment in Area - Safety-related and non-safety-related equipment are listed below.

Equipment or Component	ID No. and Division	Function	
Fire Area CSA, Cable Vault A			
Damper	D7A-SA	Battery Room & Switchgear Ventilation	
Cable Trays	C39-SA	(Later)	N/A
	C41-SA	(Later)	"
	C43-SA	(Later)	
	C45-SA	(Later)	
	C37-SA	(Later)	
	C47-SA	(Later)	
	C49-SA	(Later)	
	C51-SA	(Later)	
	C53-SA	(Later)	
	L21-SA	(Later)	
	L23-SA	(Later)	
	L29-SA	(Later)	
	C25-SC	(Later)	
	C27-SC	(Later)	
	C77-SC	(Later)	
	C59-NA	(Later)	
	C61-NA	(Later)	
	C15-NA	(Later)	
	C25-NA	(Later)	
	C6-NA	(Later)	
	C17-NA	(Later)	
	C57-NA	(Later)	
	C5-NA	(Later)	
	C19-NA	(Later)	
	C21-NA	(Later)	
	C63-NA	(Later)	
	L3-NA	(Later)	
	L11-NA	(Later)	
	L13-NA	(Later)	
	L15-NA	(Later)	
	L17-NA	(Later)	

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function	
Fire Area CSA, Cable Vault A (Cont'd)			
	L33-NA	(Later)	N/A
	C16-NB	(Later)	
	C18-NB	(Later)	
	C20-NB	(Later)	
	C22-NB	(Later)	
	C6-NB	(Later)	
	C28-NB	(Later)	
	C30-NB	(Later)	
	C32-NB	(Later)	
	C34-NB	(Later)	
	L12-NB	(Later)	
	L14-NB	(Later)	
	L16-NB	(Later)	
	L20-NB	(Later)	
Fire Area CSB, Cable Vault B			
Damper	D-7B-8B	Battery Room & Switchgear Ventilation	
Cable Trays	C36-SB	(Later)	N/A
	C38-SB	(Later)	
	C40-SB	(Later)	
	C42-SB	(Later)	
	C44-SB	(Later)	
	C46-SB	(Later)	
	C48-SB	(Later)	
	C50-SB	(Later)	
	L26-SB	(Later)	
	L28-SB	(Later)	
	C52-SD	(Later)	
	C78-SD	(Later)	
	L22-SD	(Later)	
	C25-NA	(Later)	
	L33-NA	(Later)	
	L3-NA	(Later)	
	L13-NA	(Later)	
	L17-NA	(Later)	
	L15-NA	(Later)	
	L1-NA	(Later)	
	C18-NB	(Later)	
	C20-NB	(Later)	
	C28-NB	(Later)	

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Area CSB, Cable Vault B (Cont'd)		
	C30-NB	(Later)
	C32-NB	(Later)
	C34-NB	(Later)
	C25-NB	(Later)
	C14-NB	(Later)
	L14-NB	(Later)
	L16-NB	(Later)
	L2-NB	(Later)
	L4-NB	(Later)
	L20-NB	(Later)
	L12-NB	(Later)
	L33-NB	(Later)
	L3-NB	(Later)
	L13-NB	(Later)
	L17-NB	(Later)
	L15-NB	(Later)
	L30-NB	(Later)
	L32-NB	(Later)

N/A

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment in the fire areas may be damaged or loose function due to a fire.

The fire in Fire Area CSA is limited to a small length of SA cable trays which are damaged or lose function due to fire. The redundant Fire Area CSB contains the SB and SD cable trays and is beyond the influence of the fire so that it continues to remain available as required.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment or Component	ID No. and Division	Function
DG Intercooler Engine Driven Pump	A-SA	Supplies Cooling Water
DG Air Intake Silencer	A-SA	DG Air Intake
DG Crankcase Vacuum Fan	A-SA	DG Operation
DG Crankcase Vacuum Oil Separator	A-SA	DG Operation
DG Jacket Water Heat Exchanger	A-SA	DG Cooling System
DG Intercooler Heat Exchanger	A-SA	DG Cooling System
DG Lube Oil Heat Exchanger	A-SA	DG Lube Oil Cooling
DG Air Intake Filters	A-SA	DG Operation
Exciter Enclosure	A-SA	Protective
Neutral Ground Transformer	A-SA	Electrical Balance
Fire Zone DG-12, Electrical Equipment Room		
DG Control Panel	A-SA	(later)
480V MCC	A323-SA	(later)
Cable Tray	P9-SA	(later) N/A
Cable Tray	C53-SA	(later) N/A
Cable Tray	L21-SA	(later) N/A
Fire Zone DG-13, Diesel Day Tank Room A-SA		
Diesel Day Tank	A-SA	DG Set Fuel
Fire Area DGB, Diesel Generator B-SB		
Fire Zone DG-21, Diesel Generator Room B-SB		
Diesel Generator	B-SB	Required for emergency power during accident
DG Jacket Water Expansion Tank	B-SB	DG Cooling
DG Lube Oil Make-up Tank A/B	B-SB	DG Set Lubrication
Starting and Control Air Tank	B-SB	DG Startup
DG Engine Driven Fuel Oil Pump	B-SB	DG Oil Supply

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment or Component	ID No. and Division	Function
DG Jacket Water Engine Driven Pump	B-SB	Water to Coolers
DG Intercooler Silencer	B-SB	DG Air Intake
DG Crankcase Vacuum Fan	B-SB	DG Operation
DG Crankcase Vacuum Oil Separator	B-SB	DG Operation
DG Jacket Water Heat Exchanger	B-SB	DG Cooling System
DG Intercooler Heat Exchanger	B-SB	DG Cooling System
DG Lube Oil Heat Exchanger	B-SB	DG Lube Oil Cooling
DG Air Intake Filters	B-SB	DG Operation
Thermocouple	TE-HV-4921B1S	DG Room Temperature
Thermocouple	TE-HV-4921B2S	DG Room Temperature
Thermocouple	TE-HV-4921B3S	DG Room Temperature
Thermocouple	TE-HV-4921B4S	DG Room Temperature
Valve	3CC-B540SB	DG Room Isolation
Vent Line	3WG-VE006SA	DG A Intercooler Vent Line to Expansion Tank A
Cable Tray	P10-SB	(later) N/A
Cable Tray	C42-SB	(later)
Cable Tray	C42-SB	(later)
Cable Tray	L26-SB	(later)
Fire Zone DG-22, Electrical Equipment Room		
DG Control Panel	B-SB	(later)
480V MCC	B-323-SB	(later)
Cable Tray	P10-SB	(later) N/A
Cable Tray	C42-SB	(later)
Cable Tray	L26-SB	(later)

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Unit Coolers	UC-1A-SA	HPSI Pump Room A
	UC-2A-SA	Containment Spray Pump Room A
	UC-3A-SA	LPSI Pump Room A
	UC-6A-SA	Auxiliary Feedwater Pump 1A . 2A Room
Valves	3AF-FO23SA	Motor Driven Auxiliary Feed Pump A Discharge
	3AF-VD099SA	Auxiliary Feedwater System Isolation Valve
	3AF-VD101SA	Auxiliary Feedwater System Isolation Valve
	3CC-VE535SA	CS Pump A Cooler Isometric Valve
Control Valve	FCV-AF-8321AS	Motor Driven Auxiliary Feed Pump A

Fire Zone EC-3 Mechanical Penetration Area A

Equipment or Component	ID No. and Division	Function
Unit Cooler	UC-7A-SA	Mechanical Penetration Area Emergency Core Cooling System Area
Tank	X-SA/B	Spray Chemical Storage Tank
Pump	X	Spray Chemical Pump
Valves	2CC-B525SB	Containment Isolation
	2CC-B521SB	Containment Isolation
	3CC-B507SA	Safety Injection Isolation
	3CC-B508SA	Safety Injection Isolation
Control Valve	TCV-HV-4957AS	Mechanical Penetration Area 5A Temperature
Transmitter	TE-HV-4957AS	Mechanical Penetration Area 5A Temperature
Cable Trays	C43-SA	(Later)
	L21-SA	(Later)
	P13-SA	(Later)

N/A

Fire Area ECB Essential Cooling Equipment B

Fire Zone EC-4 Shutdown Cooling Heat Exchanger B

Equipment or Component	ID No. and Division	Function
Heat Exchanger	B-SB	Shutdown Cooling
Unit Cooler	UC-4B-SB	Shutdown Cooling Heat Exchanger B
Valves	3CC-B512SB	Shutdown Heat Exchanger B Isolation Valve
	3CC-VE536SB	Containment Spray Pump B Cooler Isometric Valve

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone EC-6 Mechanical Penetration Area B		
Valves	3CC-B067SN	Non Nuclear Safety Header Return Isolation
	3CC-B509SB	Safety Injection Isolation
	3CC-B510SB	Safety Injection Isolation
	2CH-VQ005SNR	Seal Injection Line Isolation Outside Containment
Unit Cooler	UC-7B-SB	Mechanical Penetration Area Emergency Core Cooling System Area
	2CH-VQ040SNR	Charging Line Isolation Outside Containment
	2CH-VP011SBR	Containment Isolation
	2CH-VP030SBR	Containment Isolation
	2EC-B009SB	Containment Ventilation Exchange
	2EC-B911SB	Hydrogen Purge Supply
Thermocouples	TE-HV-223	Reactor Heat Exchanger Letdown Outlet Temperature
	TE-HV-224	Ion Exchanger and Boronometer Inlet Temperature
Control Valve	TCV-HV-4957BS	Mechanical Penetration Area 5B Temperature
Transmitters	TE-HV-4957BS	Mechanical Penetration Area 5B Temperature
	FIS-CC-7241	Letdown Heat Exchanger Component Cooling Water Flow
Cable Trays	P13-SA	(Later) N/A
	B26-SB	(Later) "
	B27-SB	(Later) \
	C42-SB	(Later) \
	L26-SB	(Later) \
	P14-SB	(Later)
	C82-NB	(Later) \

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety related equipment may be damaged or lose function due to a fire and smoke. The postulated fire in Zone EC-3 will be evaluated later during the safety evaluation to determine the effect on redundant equipment and safe shutdown capability.

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10. Analysis of Effects of Postulated Fire (Cont'd)

components within a single train except as indicated in Section 3, and by three-hour rated fire resistive barriers enclosing the fire area except the boundary between EP-11 and EP-21.

d. Fire Postulation - A fire is postulated at the boundary of Fire Zone EP-11 from an exposure fire on the floor adjacent to the CEDM panels.

The fire could achieve a maximum development that directly exposes cable insulation inside the panels and non-safety-related cable trays directly over the panels. The cable fire could spread several feet in both directions along the trays and reach SA and SB cable trays which enter Fire Area RS from Zones EP-11 and EP-21.

e. Fire Effects - All fire zones in Fire Areas EPA and EPB are enclosed within three-hour fire resistive boundaries with the exception of Fire Zones EP-11 and EP-21, and are not exposed by the fire. Only a limited portion of the low permanent combustibles in EP-11 and EP-21 is involved in the fire.

Damage by fire to other safety-related equipment such as switchgear would be limited to one train because of the more than 50 foot separation between A and B trains in EP-11 and EP-21 and by enclosure of redundant trains in three-hour rated fire resistive walls in the other EP fire zones.

f. Equipment in Area - Safety-related equipment and non-nuclear safety-related equipment are listed below:

Equipment or Component	ID No. and Division	Function
Fire Area EPA, Electrical Penetration Area A		
Fire Zone EP-12, Battery Room SA		
Battery Racks	SA	Later
Fire Damper	D13A-SA	Battery Room & Switchgear Ventilation
Fire Zone EP-13, Battery Room SC		
Battery Racks	SC	Later

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone EP-14		
Motor Generator Set	A	(Later)
Fire Zone EP-15, Communications Room		
Communications Panels		(Later)
480V Motor Control Center	A111-SA A313-SA	(Later)
125V DC Panels	A-SA 5A-SA A-SA	Emergency Diesel Generator A Steam Turbine Auxiliary Feed Pump A H_2 Recombiner
13.8 kV Switchgear	RCP-SA	Reactor Trip
480V Power Cabinet	A-11	(Later)
Fire Zone EP-16, Electrical Penetration Area A		
Cable Trays	L21-SA L23-SA C23-SC L25-SC P55-NA P57-NA P59-NA	(Later) (Later) (Later) (Later) (Later) (Later) (Later)
Fire Zone EP-21, Electrical Penetration Area A		
Panel	D D	Control Element Drive Mechanism Control Element Drive Mechanism Auxiliary
Battery Charger	A1-SA A2-SA	(Later) (Later)
MUX	2	(Later)

N/A

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Radiation Monitor	RT-HV-6701AS	Airborne Effluent, Fuel Pool (C)
	RT-HV-6702AS	Airborne Effluent, Fuel Pool (C)
HVAC Dampers	D-18B-6B	Battery Room & Switchgear Ventilation
Cable Trays	P13-SA	(Later) N/A
	P15-SA	(Later)
	P17-SA	(Later)
	C51-SA	(Later)
	C53-SA	(Later)
	C43-SA	(Later)
	L21-SA	(Later)
	P9-SA	(Later)
	C41-SA	(Later)
	C45-SA	(Later)
	L23-SA	(Later)
	C47-SA	(Later)
	C49-SA	(Later)
	C37-NA	(Later)
	C59-NA	(Later)
	C61-NA	(Later)
	C63-NA	(Later)
	L13-NA	(Later)
	L15-NA	(Later)
	L17-NA	(Later)
	P19-NA	(Later)
	P21-NA	(Later)
	P23-NA	(Later)
	P55-NA	(Later)
	P57-NA	(Later)
	P59-NA	(Later)
	P56-NB	(Later)
	P58-NB	(Later)
	P60-NB	(Later)
	C34-NB	(Later)
	L30-NB	(Later)

Fire Zone EP-26, Access Area

Radiation Monitor	RE-HV5040A	(Later)
	RE-HV5041A	(Later)

10. Analysis of Effects of Postulated (Cont'd)

Equipment or Component	ID No. and Division	Function
Cable Trays	P9-SA	(Later) N/A
	C53-SA	(Later) "
	L21-SA	(Later)
	P21-NA	(Later)
	C63-NA	(Later)
	L15-NA	(Later)
	P23-NA	(Later)
	C61-NA	(Later)

Fire Zone EP-27, Electrical Shops

Sample Recycle Tank		(Later)
Sample Recycle Pumps		(Later)

Fire Zone EP-28, Hot Laboratory and Sample Rooms

Primary Sample Panels	A, B	(Later)
Secondary Sample Panels	A, B	(Later)
Radiation Monitor		(Later)
Chiller Unit		(Later)

Fire Area EPB, Electrical Area Penetration B

Fire Zone EP-11

480V Motor Control Center	B111-6B	(Later)
	B313-5B	(Later)
125V DC Panels	B-6B	Emergency Diesel Generator B
	5B-5B	Steam Turbine Auxiliary Feed Pump B
	B-6B	H_2 Recombiner
13.8 kV Switchgear	RCP-6B	Reactor Trip
480V Power Cabinet	B-11	(Later)
Panels	D	Control Element Drive Mechanism
		Control Element Drive Mechanism

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10. Analysis of Effects of Postulated Fire (Cont'.)

Equipment or Component	ID No. and Division	Function
Battery Charger	B1-SB	(Later)
	B2-SB	(Later)
MUX	3	(Later)
PPB	1211	Power Source for forward, reverse speed
Radiation	RT-HV-6701BS	Airborne Effluent, Fuel Pool (c)
Monitor	RT-HV-6702BS	Airborne Effluent, Fuel Pool (c)
HVAC Dampers	D-18A-SA	Battery Room Switchgear Ventilation
Cable Trays	L28-SB	(Later) N/A
	C36-SB	(Later)
	C38-SB	(Later)
	C44-SB	(Later)
	C40-SB	(Later)
	L26-SB	(Later)
	C48-SB	(Later)
	C50-SB	(Later)
	P18-SB	(Later)
	P16-SB	(Later)
	C42-SB	(Later)
	P14-SB	(Later)
	P10-SB	(Later)
	V4-SB	(Later)
	P55-NA	(Later)
	P57-NA	(Later)
	P59-NA	(Later)
	C63-NA	(Later)
	L13-NA	(Later)
	P61-NA	(Later)
	P63-NA	(Later)
	P56-NB	(Later)
	P58-NB	(Later)
	P60-NB	(Later)
	C34-NB	(Later)
	C30-NB	(Later)
	P62-NB	(Later)
	P64-NB	(Later)
	P24-NB	(Later)
	L32-NB	(Later)
	P22-NB	(Later)

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function	
	F20-NB	(Later)	N/A
	C32-NB	(Later)	
	L20-NB	(Later)	
	C28-NB	(Later)	
Fire Zone EP-22, Battery Room SB			
Battery Racks	SB	(Later)	
Fire Zone EP-23, Battery Room SD			
Battery Racks	SD	(Later)	
Fire Zone EP-24, Motor Generator Set & Control Panel Room			
Motor Generator Set	B	(Later)	
Fire Zone EP-25, Access Area			
480V Power C	B23	(Later)	
Radiation Monitor	RE-HV5040B	(Later)	
	RE-HV5041B	(Later)	
Cable Trays	V4-SB	(Later)	N/A
	F10-SB	(Later)	
	C48-SB	(Later)	
	C42-SB	(Later)	
	L26-SB	(Later)	
	P22-NB	(Later)	
	P24-NB	(Later)	
	C32-NB	(Later)	
	C34-NB	(Later)	
	L32-NB	(Later)	
Fire Zone EP-29, Electrical Penetration Area B			
Cable Trays	L26-SB	(Later)	N/A
	L28-SB	(Later)	
	L22-SD	(Later)	
	C24-SD	(Later)	
	P56-NB	(Later)	
	P58-NB	(Later)	

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Thermocouples	TE-HV-4986A1S	Control A Outdoor Temperature
	TE-HV-4986B1S	Control A Outdoor Temperature
	TE-HV-4987A1S	Control Room Supply Air A High Temperature
	TE-HV-4987AS	Control Room Set Air A Temperature
Temp Control Valve	TCV-HV-4987AS	Control Room Air A Temperature Control
Cable Trays	P52-SA	(Later) N/A
	C115-SA	(Later)
	144-SA	(Later)
Fire Zone HVA-2, Control Room Fans A-SA		
Return Air Fan	R-37A	Control Room Ventilation
Exhaust Air Fan	E-6A	Control Room Ventilation
Valves	3FV-B060-SA	Control Room Isolation
	3FV-B160-SA	Control Room Isolation
	3FV-B034-SA	Control Room Isolation
	3FV-B134-SA	Control Room Isolation
Radiation Monitors	RE-HV-4913A	Plant Vent 3 Particulate
	RE-HV-4914A	Plant Vent 3 Noble
Dampers	D-8A-SA	Control Room Vent. Filt. and Cooling
	D-86A-SA	Control Room Vent. Filt. and Cooling
	D-9A-SA	Control Room Vent. Filt. and Cooling
Cable Trays	P52-SA	(Later) N/A
	C115-SA	(Later)
	L44-SA	(Later)
Fire Zone HVA-3, Chillers and Pump Room A-SA		
Chilled Water Pump	P1A-SA	(Later)
Water Chiller	WC-1A(SA)	(Later)
Transmitter	PDT-EC-4905AS	Essential Services Chilled Water Chiller A Differential Pressure
Pressure Control Valve	PCV-EC-4905AS	

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone HVA-4, Switchgear AC Room A-SA		
Air Conditioning Unit	AC-1A-SA	Battery Room and Switchgear Ventilation
Air Cleanup Unit	CU-5A	Emergency Core Cooling System Area Exhaust
Expansion Tank	Tank A	Chilled Water Expansion
Exhaust Fans	CUR-5B-SB	Emergency Core Cooling System Area Exhaust
	E-9A-SA	Battery Room and Switchgear Ventilation
Return Fan	R-5A-SA	Battery Room and Switchgear Ventilation
Valves	2PV-B029-SA	ECCS Area Exhaust
	2PV-B154-SB	ECCS Area Exhaust
	2PV-B155-SB	ECCS Area Exhaust
	3PV-B133-SB	Control Room Isolation
	3PV-B161-SB	Control Room Isolation
Radiation Montiors	RE-HV-4986-A1S	Control Room Air Intake 1A
	RT-HV-4986-A1S	Control Room Air Intake 1A
	RE-HV-4986-B1S	Control Room Air Intake 1B
	RT-HV-4986-B1S	Control Room Air Intake 1B
Temp. Control Valve	TCV-HV-4911AS	Electrical and Battery Room A Recirculation Valve
Level Transmitter	LT-EC-4907A1S	Essential Services Chilled Water Chiller A Level
Dampers	D-1A-SA	Battery Room and Switchgear Ventilation
	D-2A-SA	Battery Room and Switchgear Ventilation
	D-3A-SA	Battery Room and Switchgear Ventilation
	D-79A-SA	Battery Room and Switchgear Ventilation
	D-80A-SA	Battery Room and Switchgear Ventilation
	D-81A-SA	Battery Room and Switchgear Ventilation
Cable Trays	P33-SA	(Later)
	C116-SA	(Later)
	L45-SA	(Later)

N/A

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone HVA-5, Relay Room A		
Control Room Relay Equipment	A	(Later)
Fire Area HVB, Control Room HVAC Area B		
Fire Zone HVB-1, Control Room AC and Cleanup Units B-SB		
Air Conditioning Unit	AC-2B-SB	Control Room Ventilation
Air Cleanup Unit	CU-1B	Filtration and Cooling
Thermocouples	TE-HV-4986A2S	Control Room B Outdoor Temperature
	TE-HV-4986B2S	Control Room B Outdoor Temperature
	TE-HV-4987B1S	Control Room Supply Air B High Temperature
	TE-HV-4987BS	Control Room Set Air B Temperature
Temperature Control Valve	TCV-HV-4987BS	Control Room Air B Temperature Control
Valves	3FV-B163-SB	Control Room Isolation.
	3FV-B136-SB	Control Room Isolation
	3CC-F0055B	Essential Cooling Water Chiller B Isolation Valve
Dampers	D-82B-SB	Control Room Vent Filtration
	D-84B-SB	Control Room Vent Filtration
	D-85B-SB	Control Room Vent Filtration
	D-87B-SB	Control Room Vent Filtration
	D-88B-SB	Control Room Vent Filtration
	D-83B-SB	Control Room Vent Filtration
	D-188B-SB	Control Room Vent Filtration
Cable Trays	P51-SB	(Later) N/P
	C114-SB	(Later)
	L43-SB	(Later)
Fire Zone HVB-2, Control Room Fans B-SB		
Return Air Fan	R-37B	Control Room Ventilation
Exhaust Air Fan	E-6B	Filtration and Cooling
Valves	3-PVB035-SA	Control Room Isolation Valve
	3-PVB135-SB	Control Room Isolation Valve
	3-PVB062-SA	Control Room Isolation Valve
	3-PVB162-SB	Control Room Isolation Valve

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Radiation Monitors	RE-HV-4913B	Plant Ventilation 2 Particulate
	RE-HV-4914B	Plant Ventilation 2 Noble Gas
Dampers	D-8B-SB	Control Room Ventilation
	D-86B-SB	Control Room Ventilation
	D-9B-SB	Control Room Ventilation
Cable	P51-SB	(Later) N/A
	L43-SB	(Later) N/A

Fire Zone HVB-3, Chiller and Pump Room B-SB

Equipment or Component	ID No. and Division	Function
Chilled Water Pump	P-1B-SB	(Later)
Water Chiller	WC-1B-(SB)	(Later)
Transmitter	HDT-EC-4905BS	Essential Service Chilled Water Chiller B Differential Pressure
Press. Control Valve	PCV-EC-4905BS	
Damper	D-17B-SB	Battery Room and Switchgear Ventilation

Fire Zone HVB-4, Switchgear AC Room B-SB

Equipment or Component	ID No. and Division	Function
Air Conditioning Unit	AC-1B-SB	Battery Room and Switchgear Ventilation
Air Cleanup Unit	CU-5B	Emergency Core Cooling System Area Exhaust
Expansion Tank	Tank B	Chilled Water Expansion
Return Fan	CUR-5A-SA	Emergency Core Cooling System Area Exhaust
	E-9B-SB	Battery Room and Switchgear Ventilation
Valves	2PV-B054-SA	Emergency Core Cooling System Area Exhaust
	2PV-B055-SA	Emergency Core Cooling System Area Exhaust
	2PV-B129-SB	Emergency Core Cooling System Area Exhaust
	3PV-B063-SA	Control Room Isolation
	3PV-B036-SA	Control Room Isolation
Radiation Monitors	RE-HV-4986A2S	Control Room Air Intake 2A
	RT-HV-4986A2S	Control Room Air Intake 2A
	RE-HV-4986B2S	Control Room Air Intake 2B
	RT-HV-4986B2S	Control Room Air Intake 2B

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Temp Control Valve	TCV-HV-4911BS	Electrical and Battery Room B Recirculation Valve
Level Transmitter	LT-EC-4907B1S	Essential Services Chilled Water Chiller B Level
Dampers	D-1B-SB	Battery Room and Switchgear Vent
	D-2B-SB	Battery Room and Switchgear Vent
	D-3B-SB	Battery Room and Switchgear Vent
	D-79B-SB	Battery Room and Switchgear Vent
	D-80B-SB	Battery Room and Switchgear Vent
	D-81B-SB	Battery Room and Switchgear Vent
Cable Trays	P52-SB	(Later) N/A
	C115-SB	(Later)
	L44-SB	(Later)
Fire Zone HVB-5, Relay Room B		
Control Room Relay Equipment	B	(Later)

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment could be damaged or lose function due to a fire.

The fire and its associated effects are contained by the three-hour fire rated fire resistive barriers enclosing Fire Area HVA. Fire Area HVB is beyond the influence of the same fire and continues to remain available as required.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated in the FHA at that time.

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10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Fire Area MA, Miscellaneous Areas A		
Fire Area MA-1, Fuel Handling Building H&V Room		
Fuel Pool Exhaust Unit	E-8	FHB Ventilation
FHB HV Unit	HV-1	(Later)
FHB Units	R-7A, R-7B	(Later)
480V MCC	231	(Later)
Dampers	D-10A-SA D-10B-SB	HVAC Dampers
Cable Trays	P21-NA C63-NA P11-NA L15-NA	(Later) (Later) (Later) (Later) N/A
Fire Zone MA-2, Electrical Area		
Valves	2PV-B018SB 2PV-B017SB 2PV-B019SB 2PV-B023SB 2PV-B024SB	Cont. Isol. Cont. Isol. Cont. Isol. Cont. Isol. Cont. Isol.
Transmitters Fuel Pool	RT-HV-5071AS	Airborne Effl. Mon.
Quad B	PT-SI-0352AS	Cont. Abs. Press.
Quad C	PT-SI-0352CS	Cont. Abs. Press.
Fuel Pool	RT-HV-5072AS	Airborne Effl. Mon.
Quad. "A"	PT-SI-0351AS	Cont. Abs. Press.
Monitors	RE-HV-5044A RE-HV-5043A	Plant Vent 4* Plant Vent 4*
4180V MCC	A-121	(Later)
Cable Trays	P11-NA C63-NA P21-NA L15-NA	(Later) (Later) (Later) (Later) N/A

10. Analysis of Effects of Postulated Fires (Cont'd)

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Equipment of Component	ID No. and Division	Function
Fire Zone MA-3, Emergency Diesel Generator H&V Room A		
Supply Air Unit	HV-2ASA	(Later)
Dampers	D-90A-SA	(Later)
	D-91A-SA	(Later)
	D-92A-SA	(Later)
	D-93A-SA	(Later)
	D-11B-SB	(Later)
	LO-2A-SA	(Later)
Return Fans	R-41A-SA	(Later)
Exhaust Fans	E-10A-SA	(Later)
	E-38A-SA	(Later)
	E-39A-SA	(Later)
	E-40A-SA	(Later)
480V Pwr C	A-23	(Later)
Cable Trays	P11-NA	(Later) N/A
	P21-NA	(Later)
	C63-NA	(Later)
	L15-NA	(Later)
Fire Area MA-4, Shield Building Cleanup Unit		
Shield Bldg under Clean Up Unit	CU-3A-SA	Maintain Shield Bldg negative pressure
Exahust Fan	E-4A	Hydrogen purge exhaust
Valves	2PV-B001-SA	(Later)
	2PV-B003-SA	(Later)
	2PV-B004-SA	(Later)
	2PV-B005-SA	(Later)
	2PV-B025-SA	(Later)
	2PV-B152-SB	(Later)
	2PV-B009-SB	Containment Isolation
Cable Trays	P9-SA	(Later) N/A
	C53-SA	(Later) N/A
Fire Area MB, Miscellaneous Areas B		
Fire Zone MB-1, Solid Radwaste Pump Room B		
Pump	B-SB	VRS Cond. Sample
	A,B	Feed Tank Metering Resin Metering
	A,B	Condensate Sampling

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10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Tank	A,B	Dewatering Feed
Cable Trays	P26-NB C34-NB L32-NB P10-NB C42-NB	(Later) (Later) (Later) (Later) (Later) e N/A
Fire Zone MB-2, Electrical Area B		
Valves	2PV-B124SA 2PV-B123SB 2PV-B164SB	(Later) (Later) (Later)
Monitors	RE-HV-5044B RE-HV-5043B	Plant Vent. 1 Noble Gas* Plant Vent. 1 Particulate*
480V MCC	B121	(Later)
480V Pwr C	B12	(Later)
Inst. Racks	RA-62SB RA-44	(Later) (Later)
Cable Trays	P12-NB P26-NB C34-NB L32-NB	(Later) (Later) (Later) (Later) e N/A
Fire Zone MB-3, Emergency Diesel Generator H&V Room		
Air Unit	HV-2B-SB	(Later)
Return Fans	R-41B-SB	(Later)
Exhaust Fans	E-10B-SB E-39B-SB E-40B-SB	(Later) (Later) (Later)
Dampers	D-90B-SB D-91B-SB D-92B-SB D-93B-SB LD-2B-SB	(Later) (Later) (Later) (Later) (Later)
Cable Trays	P12-NB P26-NB C34-NB L32-NB	(Later) (Later) (Later) (Later) e N/A

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10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Monitors	RT-HV-5071BS RT-HV-5072BS	Airborne Effl. Fuel Pool (FHB) Airborne Effl. Fuel Pool (FHB)
Fire Zone MB-4, Shield Building Cleanup Unit		
Shield Bldg Clean Up Unit	CU-3B-SB	Maintain Shield Blding under negative pressure
Exhaust Fan	B	Hydrogen purge exhaust
Isolation Valves	2PV-B052-SA 2PV-B101-SB 2PV-B103-SB 2PV-B104-SB 2PV-B105-SB 2PV-B125-SB 2PV-B110-SB 2PV-B111-SB 2PV-B113-SB	Shield Building Isolation (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later)
Transmitters	PT-SI-0351BS	Containment Absolute Pressure Quad B
	PT-SI-0351DS	Containment Absolute Pressure Quad B
	PT-SI-0352BS	Containment Absolute Pressure Quad B
	PT-SI-0352DS	Containment Absolute Pressure Quad B
Cable Trays	C42-SB P10-SB	(Later) N/A (Later) N/A
Fire Zone MB-5		
VRS Distillate Rcv'r	(Later)	(Later)
VRS Concentrator	(Later)	(Later)
VRS Distillate Condenser	(Later)	(Later)
VRS Distillate Pump	(Later)	(Later)
Secondary Particulate Filter	(Later)	(Later)
VRS CNDS Sample Tk A	(Later)	(Later)
VRS CNDS Sample Tk B	(Later)	(Later)
Heater ERH-16 Controlled (Later)		(Later)

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10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
VRS Rack #1	(Later)	(Later)
VRS Rack #2	(Later)	(Later)
SPF Inst Rack	(Later)	(Later)
Inst Rack #2	(Later)	(Later)
Inst Rack #4	(Later)	(Later)

* Loss of information/data on radiological measurements results from equipment damage.

11. Assessment of Postulated Fire Consequences

Safety-related and non-safety-related equipment located in Fire Areas MA may be damaged or lose function due to a fire. The fire and associated effects are contained by the three-hour rated fire resistive barriers enclosing Fire Area MA. Fire Area MB equipment required for shutdown is redundant to Fire Area MA equipment and is beyond the influence of the postulated fire so that it continues to remain available as required.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fire (Cont'd)

Introduction of transient combustibles resulting from normal maintenance and operation activities is controlled through administrative procedures.

c. Damage Limitation - The extent of damage within and beyond the fire area is limited by controlled removal of heat, smoke and other products of combustion through continued operation of the area ventilation systems, and by three-hour fire resistive barriers enclosing the fire area.

d. Fire Postulation - A fire is postulated in Fire Area RS resulting from an exposure fire.

The fire could achieve a maximum development that directly exposes the Remote Shutdown Panel, the cables within it, and the cable trays over the burn area. The Safety Train A and B cable trays are approximately four feet apart with no barrier between them.

e. Fire Effects - The three hour rated fire resistive barriers enclosing Fire Area RS are fully capable of containing the postulated fire and its effects within the fire area.

f. Equipment in Area - Safety-related and non-safety-related equipment in the fire area are listed below:

Equipment or Component	ID No. and Division	Function
Fire Area RS - Remote Shutdown Panel		
Remote Shutdown Panel	—	Alternate shutdown of plant
Cable Trays	C47-SA	(Later) N/A
	C41-SA	(Later)
	L23-SA	(Later)
	C44-SB	(Later)
	C36-SB	(Later)
	L28-SB	(Later)

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety-related equipment could be damaged or lose function due to a fire. The Control Room, which is beyond the influence of the fire, continues to remain available so that the capability to shutdown the plant remains.

12. Safety Evaluation

Safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Valve	FCV-CS-7111BS	Containment Spray Pump B Flow Control
	A	(Later)
FD Tanks	(B)	(Later)
FD Pump	(Later)	Floor Drain Sump 6 Floor Drain Sump 7
Tank	(Later)	Concentrate Storage
Pumps	(Later)	Concentrate Storage Transfer
	(Later)	Spent Resin Transfer
	(Later)	Spent Resin Dewatering
	A, B	Secondary High Purity Waste
	A, B	Inorganic Chemical Waste
	A, B	Secondary Particulate Waste
Tanks	Later	Spent Resin Storage
	A, B	Secondary High Purity Waste
	A, B	Inorganic Chemical Waste
	A, B	Secondary Particulate Waste
Cable Trays	P32-NB	(Later) N/A
	C84-NB	(Later)
	C66-NB	(Later)
	C68-NB	(Later)
	C86-NB	(Later)
	L48-NB	(Later)

Fire Zone RW-3, Access and Boric Acid Sample

Equipment of Component	ID No. and Division	Function
Transmitter	LT-CH0208	Holdup Tank Level
Tanks	A, B	Boric Acid Condensate Sampling
Pumps	A, B	Boric Acid Condensate Sampling
Instr Racks	RA-17	Boric Acid Sampling Syst "A" Instruments
	RA-18	Boric Acid Sampling Syst "B" Instruments
Cable Trays	P31-NA	(Later) N/A
	C87-NA	(Later)
	L13-NA	(Later)

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function	
Fire Zone RW-4, Radwaste Equipment			
Valve	2PV-8107SB	Isolation Fuel Handling Building	
Tanks	(Later)	Decontamination Sample	
	A, B	Caustic Addition	
	(Later)	Flush Tank	
Pumps	(Later)	Decontamination Sample	
	A-6N, B-6N	Condensate Makeup	
	(Later)	Caustic Addition Tank Metering	
	(Later)	Acid Addition Tank Metering	
	(Later)	Flush Pump	
Condenser	(Later)	Floor Drain Distillate	
Filters	(Later)	Detergent Waste	
	A, B	Fuel Pool	
	(Later)	Floor Drain	
	(Later)	Inorganic Chemical Waste	
	(Later)	Secondary High Purity	
	(Later)	Boric Acid	
	SN	Reactor Drain	
	1-6N, 2-6N	Purification	
	1-SA, 1-SB	Seal Injection	
Cable Trays	P32-NB	(Later)	N/A
	C64-NB	(Later)	
	C82-NB	(Later)	
	L12-NB	(Later)	
	P32-NB	(Later)	
	C68-NB	(Later)	
	C86-NB	(Later)	
	V4-SB	(Later)	
	C48-6B	(Later)	
	L26-6B		
Fire Zone RW-7, Radwaste Valve Gallery			
Seal Injection Filter Hatches		(Later)	
Automated Filter Cartridge Transfer Cart		(Later)	
Cable Trays	P32-NB	(Later)	N/A
	C64-NB	(Later)	N/A

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Pumps	1,2 SN	Boric Acid Makeup
		Floor Drain Sump 3
	A,B	Inorganic Chemical Waste Condenser
		Inorganic Chemical Waste Seal Water
	A,B	Inorganic Chemical Waste Recirculating
	A,B	Inorganic Chemical Waste Condenser
	B	Secondary Particulate Waste
		Floor Drain Sump 4
Tanks	B	Secondary Particulate Waste
	(Later)	Inorganic Chemical Waste Seal Water
	(Later)	Inorganic Chemical Waste Bottoms
	A,B	Inorganic Chemical Waste Condensate
Cable Trays	C66-NB	(Later) N/A
	C66-NB	(Later)
	C68-NB	(Later)
	C86-NB	(Later)
	P32-NB	(Later)
	CB4-NB	(Later)
	L48-NB	(Later)

Fire Zone RW-26, Access Areas, HVAC

Tank	(Later)	Auxiliary Condensate Flash
Pumps	A,B	Auxiliary Condensate Flash

Fire Zone RW-27, Access Areas, HVAC

Elevator Machinery	(Later)	(Later)

Fire Zone RW-28, Radiation Monitors, Hydrogen Analyzer

Valve	3CC-VE638SA	(Later)
Monitors	RE-HV5020A	(Later)
	RE-HV5021A	(Later)

Fire Zone RW-29, Radiation Monitors, Hydrogen Analyzer

Valve	3CC-VE255SB	(Later)
Monitors	RE-HV5020B	Radiation
	RE-HV5021B	Radiation

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Boric Acid Concentrator	(Later)	(Later)
Cable Trays	V4-6B C48-6B L26-6B	(Later)

Fire Zone RW-51, Radwaste Equipment

Valves	2CH-VQ001SNR 2CH-VW401SBR 2CH-VW404SAR 2CH-VW417SNR	Seal Injection Heat Exchanger Isolation RW Storage Tank Isolation RW Storage Tank Isolation RW Storage Tank Isolation
Transmitters	RE-SD-001 RT-SD-001 RE-WM-6106 RT-WM-6106 PIT-201 TIC-213 TE-231	Sump, Secondary High Purity Discharge* Sump, Secondary High Purity Discharge* Discharge to Neutralization Pond* Discharge to Neutralization Pond* Letdown Pressure Control Boric Acid Batching Tank Temperature Seal Injection Temperature

Fire Zone RW-61, Access and Radwaste Equipment

Transmitters	LT-CH0276 LT-CH0277 LT-CH0278 LT-CH0279	Holdup Tank Level Holdup Tank Level Holdup Tank Level Holdup Tank Level
Valve	FCV-CS-7111AS	Containment Spray Pump A Flow Control
Tanks	A,B,C,D,E A,B	Holdup Secondary High Purity Sample
Pumps	A,B	Secondary High Purity Sample

Fire Zone RW-62, Radwaste Equipment

Valve	2PV-B007SA	(Later)
Instrument Racks	RA-9 RA-10 RA-7 RA-8 RA-15 RA-16	Discharge to Neutralization Pond* (Later) (Later) (Later) (Later) (Later)

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10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Tanks		Detergent Waste (2)
		Floor Drain Condensate (4)
	1,2,3,4,5,6,7,8,9(SN)	Gas Decay Heat
	SN	Nitrogen Recycle
Compressors	1-SN	Waste Gas
	SN	Nitrogen Recycle
Gas Analyzer Package	SN	(Later)
Gas Stripper	SN	(Later)
Control Panel	SN	Waste H_2 Recombiner Control
Waste Hydrogen Recombiner Valve Rack	SN	(Later)
Waste Hydrogen Gas Analyzer	SN	(Later)
Cable Trays	V3-SA	(Later)
	GA3-SA	(Later)
	L21-SA	(Later)
	C85-NA	(Later)
	C87-NA	(Later)
	L13-NA	(Later)
	C81-NA	(Later)
	C83-NA	(Later)
	L13-NA	(Later)
Transmitters	FIT-204	Process Monitor Flow
	FT-210X	Reactor Makeup Water Flow Control
	FT-210Y	Boric Acid Makeup Flow Control
	LT-226	Volume Control Tank Level
	LIT-227	Volume Control Tank Level
Heat Exchanger	(Later)	Seal Injection
Tanks	(Later)	Chemical Addition
		Boric Acid Batching
	B	Caustic Addition
	B	Acid Addition
	SN	Volume Control
Pumps	(Later)	Chemical Addition
	B	Caustic Addition
	B	Acid Addition
	(Later)	Inorganic Chemical Waste Distillate

N/A

10. Analysis of Effects of Postulated Fires (Cont'd)

Equipment of Component	ID No. and Division	Function
Condensers	A, B	Inorganic Chemical Waste
Cable Trays	C82-NB	(Later)
	L12-NB	(Later)
	P32-NB	(Later)
	C64-NB	(Later)
	C80-NB	(Later)
	C86-NB	(Later)
	C84-NB	(Later)
	L48-NB	(Later)
	C66-NB	(Later)
	C68-NB	(Later)

N/A

Fire Zone RW-52, Radwaste Control Room

Equipment of Component	ID No. and Division	Function
Control Cabinet	HCP-3	HVAC Control
Multiplex Panel	6	(Later)
HVAC Unit	AC-3	(Later)
Cable Trays	C80-NB	(Later)
	C82-NB	(Later)
	C86-NB	(Later)
	C84-NB	(Later)
	L12-NB	(Later)
	C64-NB	(Later)
	C68-NB	(Later)
	C66-NB	(Later)
	L48-NB	(Later)
	C81-NA	(Later)
	C83-NA	(Later)
	C85-NA	(Later)
	C87-NA	(Later)
	L13-NA	(Later)

N/A

Fire Zone RW-91, Radwaste Storage and Decontamination

Equipment of Component	ID No. and Division	Function
Bridge Crane	(Later)	(Later)
10 Ton Trolley	(Later)	(Later)
Pumps	A, B	Sump #15

Fire Zone RW-92, Radwaste Storage and Decontamination

Equipment of Component	ID No. and Division	Function
Bridge Crane	(Later)	(Later)
15 Ton Monorail	(Later)	(Later)

*Loss of information/data on radiological measurements results from equipment damage.

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Area SWA, Essential Switchgear Area A-SA		
Fire Zone SW-1, Essential Switchgear Room A-SA		
480V MCC	311-SA	Control Room Isolation Valve
	312-SA	Shutdown Cooling Heat Exchanger A
	321-SA	Auxiliary Feedwater Pump 1A & 2A
	322-SA	Emergency Core Cooling System Heat Exchanger Isolation Valves
480V PC	A31-SA	Power to Shield Building
	A32-SA	Cleanup Unit
4.16 kV Swgr	A3-SA	Power to Chiller WC-1A-SA
UPS Panel and Transformer	A-SA	Uninterruptible Power Supply
Switchgear Test Cabinet	A-SA	(Later)
Damper	D-5A-SA	(Later)
Cable Trays	C41-SA	(Later)
	C37-SA	(Later)
	C43-SA	(Later)
	C49-SA	(Later)
	C51-SA	(Later)
	C53-SA	(Later)
	L21-SA	(Later)
	P9-SA	(Later)
	P13-SA	(Later)
	P15-SA	(Later)
	P17-SA	(Later)
	V3-SA	(Later)
	V5-SA	(Later)
	P47-SA	(Later)
	P49-SA	(Later)
	C105-SA	(Later)
	C103-SA	(Later)
	P43-SA	(Later)
	P45-SA	(Later)
	C99-SA	(Later)
	C101-SA	(Later)
	P35-SA	(Later)
	P37-SA	(Later)
	C91-SA	(Later)
	C93-SA	(Later)
	P39-SA	(Later)
	P41-SA	(Later)
	C95-SA	(Later)
	C97-SA	(Later)

N/A

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone SW-2, Emergency Electrical Room C-SC		
Battery Charger	SC-1 (SC) SC-2 (SC)	Charging Batteries
Panel	SC	25 Volt dc Panel
UPS Panel and Transformer	SC	Uninterruptible Power Supply
Cable Tray	C55-SC	(Later) N/A
Fire Area SWB, Essential Switchgear Area B-6B		
Fire Zone SW-3, Essential Switchgear Room B		
480V MCC.	311-6B 312-6B 321-6B A322-6B	Control Room Isolation Valve Shutdown Cooling Heat Exchanger B Auxiliary Feedwater Pump 1B & 2B Emergency Core Cooling System Heat Exchanger Isolation Valves
480V PC	B31-6B B32-6B	Power to Shield Bldg Cleanup Unit
4.16 kV Swgr	23-6B	Power to Chiller WC-1B-6B
UPS Panel and Transformer	B-6B	(Later)
Switchgear Test Cabinet	B-6B	Battery Room and Switchgear Ventilation
Damper	B-5B-6B	(Later)
Cable Trays	P10-6B P14-6B P16-6B P18-6B C50-6B C48-6B C46-6B C40-6B L26-6B V4-6B P10-6B C42-6B C36-6B V6-6B	(Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) (Later) N/A

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Cable Trays (Cont'd)	P46-SB	(Later)
	P48-SB	(Later)
	C100-SB	(Later)
	C102-SB	(Later)
	P42-SB	(Later)
	P44-SB	(Later)
	C96-SB	(Later)
	P38-SB	(Later)
	P40-SB	(Later)
	C92-SB	(Later)
	C94-SB	(Later)
	P34-SB	(Later)
	P36-SB	(Later)
	C88-SB	(Later)
	C90-SB	(Later)

N/A

Fire Zone SW-4, Emergency Electrical Room D-SD

Battery Charger	SD-1(SD) SD-2 (SD)	Charging Batteries
Panel	SC	25 Volt dc Panel
UPS Panel and Transformer	SC	(Later)
Cable Trays	C52-SD	(Later) N/A

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment in Fire Area SWA may be damaged or lose function due to a fire. The fire and its associated effects are contained by the three-hour rated fire resistive barriers enclosing Fire Area SWA. Fire Area SWB equipment is redundant to that contained in Fire Area SWA and is beyond the influence of the postulated fire so that it continues to remain available as required for safe reactor shutdown.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
125V dc Panels	NA, NB	(Later)
125V dc Distribution Panel	A	(Later)
Electro Magnetic Filter Control Panel	(Later)	(Later)
Electro Magnetic Filter Instrument Panel	(Later)	(Later)
Electro Magnetic Filter Coil Cooling Units	A, B	(Later)
Amertap Ball Collectors		(4) collectors
Fans	E-44A E-44B	(Later) (Later)
Multiplexer	MUX-8	(Later)
Instrument Air Receiver	(Later)	(Later)
Instrument Air Dryer	(Later)	(Later)
Cable Trays	C5-NA	Non-Nuclear Safety Systems Cables
	C7-NA	Non-Nuclear Safety Systems Cables
	C9-NA	Non-Nuclear Safety Systems Cables
	C1-NA	Non-Nuclear Safety Systems Cables
	C3-NA	Non-Nuclear Safety Systems Cables
	L1-NA	Non-Nuclear Safety Systems Cables
	L3-NA	Non-Nuclear Safety Systems Cables
	P3-NA	Non-Nuclear Safety Systems Cables
	V1-NA	Non-Nuclear Safety Systems Cables
	P1-NA	Non-Nuclear Safety Systems Cables
	C5-NA	Non-Nuclear Safety Systems Cables
	C25-NA	Non-Nuclear Safety Systems Cables
	C16-NA	Non-Nuclear Safety Systems Cables
	C18-NA	Non-Nuclear Safety Systems Cables
	C20-NA	Non-Nuclear Safety Systems Cables
	C22-NA	Non-Nuclear Safety Systems Cables
	C6-NA	Non-Nuclear Safety Systems Cables
	L12-NA	Non-Nuclear Safety Systems Cables
	C15-NA	Non-Nuclear Safety Systems Cables
	C17-NA	Non-Nuclear Safety Systems Cables
	C19-NA	Non-Nuclear Safety Systems Cables
	C21-NA	Non-Nuclear Safety Systems Cables
	L11-NA	Non-Nuclear Safety Systems Cables
	C8-NB	Non-Nuclear Safety Systems Cables
	C10-NB	Non-Nuclear Safety Systems Cables
	C2-NB	Non-Nuclear Safety Systems Cables

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Cable Trays	C4-NB	Non-Nuclear Safety Systems Cables
	C6-NB	Non-Nuclear Safety Systems Cables
	L2-NB	Non-Nuclear Safety Systems Cables
	L4-NB	Non-Nuclear Safety Systems Cables
	V2-NB	Non-Nuclear Safety Systems Cables
	P2-NB	Non-Nuclear Safety Systems Cables
	P4-NB	Non-Nuclear Safety Systems Cables
	C28-NB	Non-Nuclear Safety Systems Cables

Fire Zone T-1.1, Battery Rooms

125V dc Batteries	(Later)	(Later)
250V dc Batteries	(Later)	(Later)

Fire Zone T-2, Condensate Demineralizer Area

Resin Strainer Condensate	(Later)	(Later)
Demineralizer Vessels	1 thru 12	
Tanks	(Later)	Cation Regeneration Tank
	(Later)	Anion Regeneration Tank
	(Later)	Resin Storage Tank
	(Later)	Acid Tank
	(Later)	Caustic Tank
	(Later)	Ammonia Tank
	(Later)	Caustic Recovery Tank
	(Later)	Ultrasonic Resin Cleaner Drain
	(Later)	Tank
Pumps	A, B	Sluice Pumps
	A, B	Recycle Pumps
	A, B	Service Water Pumps
	A, B	Floor Drain Sump #12
MCC's	A215	(Later)
	A223	(Later)
Instrument Rack	RT-4	(Later)
Multiplexer	MUX 9	(Later)
Demineralizer Air Blower	(Later)	(Later)
Demineralizer Control Panel	(Later)	(Later)

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Cable Trays	V1-NA	Non-Nuclear Safety Systems Cables
	P1-NA	Non-Nuclear Safety Systems Cables
	P3-NA	Non-Nuclear Safety Systems Cables
	C1-NA	Non-Nuclear Safety Systems Cables
	C3-NA	Non-Nuclear Safety Systems Cables
	L1-NA	Non-Nuclear Safety Systems Cables
	C5-NA	Non-Nuclear Safety Systems Cables
	V2-NB	Non-Nuclear Safety Systems Cables
	P2-NB	Non-Nuclear Safety Systems Cables
	P4-NB	Non-Nuclear Safety Systems Cables
	C2-NB	Non-Nuclear Safety Systems Cables
	C4-NB	Non-Nuclear Safety Systems Cables
	L2-NB	Non-Nuclear Safety Systems Cables
	C6-NB	Non-Nuclear Safety Systems Cables

Fire Zone T-2.1, Unloading Bay

None

Fire Zone T-3, Condenser Tube Pull Iron

Pumps	A, B, C, D	Mechanical Vacuum Pumps
	A, B	Heater Drain Pumps
	A, B	Floor Drain Sump #13
	A, B, C, D	Amertap Reinjection Pumps
Instrument Air Aftercooler	(Later)	(Later)
Separators	A, B	(Later)
Instrument Rack	RT-5	(Later)
Radiation Monitors	RE-WM6213*	Liquid Effluent, Waste Management System Discharge
	RT-WM6213*	Liquid Effluent, Waste Management System Discharge
	RE-HV1400*	Condenser Mechanical Vacuum Pump
10 Ton Monorail Cranes	(2)	(Later)
Cable Trays	P1-NA	Non-Nuclear Safety Systems Cables
	P3-NA	Non-Nuclear Safety Systems Cables
	C1-NA	Non-Nuclear Safety Systems Cables
	C3-NA	Non-Nuclear Safety Systems Cables
	L1-NA	Non-Nuclear Safety Systems Cables
	V1-NA	Non-Nuclear Safety Systems Cables
	P2-NB	Non-Nuclear Safety Systems Cables

* Radiological measurement data is lost if this equipment is damaged.

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Cable Trays	P4-NB	Non-Nuclear Safety Systems Cables
	C2-NB	Non-Nuclear Safety Systems Cables
	C4-NB	Non-Nuclear Safety Systems Cables
	L2-NB	Non-Nuclear Safety Systems Cables
	V2-NB	Non-Nuclear Safety Systems Cables
Auxiliary Boiler Deaerator	(Later)	(Later)
Pumps	(Later)	Auxiliary Boiler Recirculation Pump Auxiliary Boiler Feed Pump
Boiler Control Panel	(Later)	(Later)
428 kW 13.8 kV	(Later)	(Later)
Auxiliary Electrode Boiler	(Later)	(Later)
Fire Zone T-3.1, Unloading Bay		
None		
Fire Zone T-4.1, Switchgear Room A		
13.8 kV Bus	A1	(Later)
4.16 kV Bus	A2	(Later)
480V MCC	A211	(Later)
	A221	(Later)
480V PC	A21	(Later)
	A22	(Later)
Panel	NA	Uninterruptible Power Source
Cable Trays	P3-NA	Non-Nuclear Safety Systems Cables
	C7-NA	Non-Nuclear Safety Systems Cables
	C9-NA	Non-Nuclear Safety Systems Cables
	C1-NA	Non-Nuclear Safety Systems Cables
	L3-NA	Non-Nuclear Safety Systems Cables
	P5-NA	Non-Nuclear Safety Systems Cables
Fire Zone T-4.2, Switchgear Room B		
13.8 kV Bus	B1	(Later)
4.16 kV Bus	B2	(Later)

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
480V MCC	B221	(Later)
	B222	(Later)
480V PC	B22	(Later)
Panel	NB	Uninterruptible Power Source
Cable Trays	P4-NB	Non-Nuclear Safety Systems Cables
	C8-NB	Non-Nuclear Safety Systems Cables
	C10-NB	Non-Nuclear Safety Systems Cables
	C2-NB	Non-Nuclear Safety Systems Cables
	L4-NB	Non-Nuclear Safety Systems Cables
	P6-NB	Non-Nuclear Safety Systems Cables

Fire Zone T-5, Reheater Drain Tank Area

Equipment or Component	ID No. and Division	Function
Tanks	A1, B1	1st Stage Reheater Drain Tank
	A1, B1	2nd Stage Reheater Drain Tank
	A1, B1	Moisture Separator Drain Tank
High Pressure Heaters	1A	(Later)
	1B	(Later)
Oil Separator	(Later)	(Later)
Pumps	(Later)	Water Pump
	(Later)	Oil Pump
Gland Steam Condenser	(Later)	(Later)
Turbine Electric	(Later)	(Later)
Hydraulic	(Later)	(Later)
Fluid Unit	(Later)	(Later)
Accumulators	(Later)	(Later)
MCC	A114	(Later)
Instrument Racks	RT-3	(Later)
	RT-6	(Later)
	RT-7	(Later)
	RT-10	(Later)
Cable Trays	C7-NA	Non-Nuclear Safety Systems Cables
	C9-NA	Non-Nuclear Safety Systems Cables
	C1-NA	Non-Nuclear Safety Systems Cables
	L3-NA	Non-Nuclear Safety Systems Cables

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Cable Trays	P3-NA	Non-Nuclear Safety Systems Cables
	P5-NA	Non-Nuclear Safety Systems Cables
	C8-NB	Non-Nuclear Safety Systems Cables
	C10-NB	Non-Nuclear Safety Systems Cables
	C2-NB	Non-Nuclear Safety Systems Cables
	L4-NB	Non-Nuclear Safety Systems Cables
	P4-NB	Non-Nuclear Safety Systems Cables
	P6-NB	Non-Nuclear Safety Systems Cables
Fire Zone T-6, Turbine Exhaust Area		
Tanks	B2	1st Stage Reheater Drain Tank
	B2	2nd Stage Reheater Drain Tank
	B2	Moisture Separator Drain Tank
Low Pressure Heaters	5A, B, C	(Later)
	6A, B, C	(Later)
Covered Resin Hopper	(Later)	(Later)
Generator Main Leads Cooling Unit	(Later)	(Later)
Load Breaker Cooler	(Later)	(Later)
Instrument Rack	RT-20	(Later)
Cable Trays	P5-NA	Non-Nuclear Safety Systems Cables
	P3-NA	Non-Nuclear Safety Systems Cables
	C7-NA	Non-Nuclear Safety Systems Cables
	C9-NA	Non-Nuclear Safety Systems Cables
	L3-NA	Non-Nuclear Safety Systems Cables
	P6-NB	Non-Nuclear Safety Systems Cables
	P4-NB	Non-Nuclear Safety Systems Cables
	C8-NB	Non-Nuclear Safety Systems Cables
	C10-NB	Non-Nuclear Safety Systems Cables
	L4-NB	Non-Nuclear Safety Systems Cables
Fire Zone T-7, High Pressure Heater Area		
Pumps	(Later)	Feed Pump Turbine Exhaust A
		Feed Pump Turbine Exhaust B
High Pressure Heaters	1A, B	(Later)
Low Pressure Heaters	2A, B	(Later)
Tanks	A2	1st Stage Reheater Drain Tank
	A2	2nd Stage Reheater Drain Tank
	A2	Moisture Separator Drain Tank

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Generator Neutral Grounding Transformer & Resistor	(Later)	(Later)
MCC	B114	(Later)
Instrument Racks	RT-8	(Later)
	RT-11	(Later)
Gas Dryer	(Later)	(Later)
Generator Auxiliary Panel	(Later)	(Later)
Cable Trays	P4-NA	Non-Nuclear Safety Systems Cables
	P3-NA	Non-Nuclear Safety Systems Cables
	C7-NA	Non-Nuclear Safety Systems Cables
	C9-NA	Non-Nuclear Safety Systems Cables
	L3-NA	Non-Nuclear Safety Systems Cables
	P5-NA	Non-Nuclear Safety Systems Cables
	P6-NA	Non-Nuclear Safety Systems Cables
	P4-NA	Non-Nuclear Safety Systems Cables
	C8-NA	Non-Nuclear Safety Systems Cables
	C10-NA	Non-Nuclear Safety Systems Cables
	L4-NA	Non-Nuclear Safety Systems Cables

Fire Zone T-8, Hydrogen Seal Oil Unit Area

Hydrogen Oil Unit	(Later)	(Later)

Fire Zone T-9.1, Motor Driven Steam Generator Pump Area

Pump	(Later)	Motor Driven Steam Generator Pump
Cable Trays	(Later)	(Later) N/A

Fire Zone T-9.2, Turbine Driven Steam Generator Pump A Area

Pump	A	Turbine Driven Steam Generator Pump
Cable Trays	(Later)	(Later) N/A

Fire Zone T-9.3, Turbine Driven Steam Generator Pump B Area

Pump	B	Turbine Driven Steam Generator Pump
Cable Trays	(Later)	(Later) N/A

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zones 10.1 Thru 10.6, Turbine Bearing Areas		
Turbine Bearings	(Later)	(Later)
Fire Zone T-11, Operating Floor		
High Pressure Turbine	(Later)	(Later)
Low Pressure Turbines	1A 2B 3C	(Later)
Moisture Separator Reheaters	A, B	(Later)
Low Pressure Heaters	3A, B, C 4A, B, C	(Later)
Air Handling Units	EAC-2A, B, C, D	(Later)
Instrument Rack	RT-12 RT-13 RT-15 RT-16 RT-14 RT-17	(Later) (Later) (Later) (Later) (Later) (Later)
MCC's	A212 A213 A223 A224	(Later) (Later) (Later) (Later)
HVAC Control Panel	HCP-5	(Later)
Cable Trays	P3-NA C7-NA L3-NA P4-NB C8-NB L4-NB	Non-Nuclear Safety Systems Cables Non-Nuclear Safety Systems Cables Non-Nuclear Safety Systems Cables Non-Nuclear Safety Systems Cables Non-Nuclear Safety Systems Cables Non-Nuclear Safety Systems Cables
Fire Zone T-20, Lube Oil Tank Room		
De Laval Oil Purifier Unit	(Later)	(Later)
Lube Oil Transfer Pump	(Later)	(Later) (Later)

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Dirty L O Batch Tank	(Later)	(Later)
Clean L O Batch Tank	(Later)	(Later)
Cable Trays	P3-NA C1-NA	Non-Nuclear Safety Systems Cables
Fire Zone T-21, Lube Oil Room		
Main Turbine Lube Oil Reservoir	(Later)	(Later)
Pump	(Later)	Oil Pump
Fire Zone T-22 Auxiliary Boiler Room		
Auxiliary Boiler Deaerator	(Later)	
Pumps	(Later) (Later)	Auxiliary Boiler Recirculation Pump Auxiliary Boiler Feed Pump
Boiler Control Panel	(Later)	(Later)
428 kW 13.8 kV	(Later)	(Later)
Auxiliary Electrode Boiler	(Later)	(Later)
Fire Zone T-23, Electrical Switchgear Room		
Electrical Switchgear	(Later)	(Later)

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment in Fire Area T may be damaged or lose function due to fire. The fires and associated effects are contained by three-hour rated fire resistive barriers enclosing Zones T-4.1 and T.20. The equipment exposed by the fire is not required for safe reactor shutdown. Any safe shutdown equipment in any adjacent fire areas is beyond the influence of the postulated fire and continues to remain available as required.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fire (Cont'd)

f. Equipment in Area - Safety and non-safety-related equipment are listed below.

Equipment or Component	ID No. and Division	Function
Fire Area UH-1 Electric Area A		
Power Center	A331-SA	Power for CCW Dry Cooling Towers A & B
Cable Trays	(Later)	
Fire Area UH-4 Electric Area B		
Power Center	A332-SA	Power for CCW Dry Cooling Towers A & B
Cable Trays	(Later)	

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment in each fire area could be damaged or lose function due to a fire. The fire and its associated effects are contained by the three-hour rated fire resistive barriers enclosing each fire area. Since the fire areas contain redundant equipment safe shutdown will not be impacted by the same fire.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

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10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Zone FH-7, Spent Fuel Handling Area		
Transmitters	RE-HV-5071AS	(Later)
	RE-HV-5072AS	(Later)
	RE-HV-5071BS	(Later)
	RE-HV-5072BS	(Later)
	RLH-HV-5071S	(Later)
	RLH-HV-5072S	(Later)
Power Center	(Later)	(Later)
Spent Fuel Handling Machine	(Later)	(Later)
Jib Crane	(Later)	(Later)
5 Ton Trolley Hoist	(Later)	(Later)

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety equipment may be damaged or lose function due to a fire. The postulated fire and associated effects are contained by three-hour rated fire resistive barriers enclosing the fire area. No safe shutdown equipment is involved so that safe shutdown may be achieved and there is no loss of control of radioactive release.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to the design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fire (Cont'd)

Equipment or Component	ID No. and Division	Function
Fire Area FX, CCW Heat Exchanger and Pump		
Fire Zone FH-1, CCW Pump Room A-SA		
Pumps	1A-SA 2A-SA	CCW CCW
Unit Cooler & 2A	UC-8A-SA	CCW Heat Exchanger A and CCW Pumps 1A (CCW Area A)
Transmitters	TE-HV-4963AS TCV-HV-4963AS	CCW Pump A Area Temp CCW Pump A Area Temp
Fire Zone FH-2, CCW Heat Exchanger Room A-SA		
Heat Exchanger	A-SA	Remove Heat from CCW System
Instrument Rack	RF-3SA/SN	(later)

11. Assessment of Postulated Fire Consequences

The above listed safety and non safety equipment in the fire area FY may be damaged or lose function due to fire. The postulated fire and the associated effects are contained by the three-hour rated fire resistive barriers which enclose the fire area. Redundant safe shutdown equipment is not involved so that safe shutdown may be achieved and there is no loss of control of radioactive release.

12. Safety Evaluation

The safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to the design will be incorporated into the FHA at that time.

10. Analysis of Effects of Postulated Fire (Cont'd)

c. Damage Limitation - The extent of damage within and beyond the fire zone is limited by fire resistive barriers which separate trains of safety related equipment.

d. Fire Postulation - Fire Area UH-3 - A fire is postulated from an exposure fire in the storage tank dike area. The fire could achieve a maximum development that directly exposes the diesel oil storage tank and pumping equipment.

e. Fire Effect - Fire Area - UH-3 - The fire area is enclosed within three hour rated fire resistive barriers and therefore the redundant fire area (UH-6) is not affected.

f. Equipment in Area - Safety-related and non-safety-related equipment are listed below:

Equipment or Component	ID No. and Division	Function
Fire Area Active Area A		
Fire Area Cooling Tower A		
Fire Area UH-3 Diesel Oil Storage Tank and Pump A		
Diesel Oil Storage Tank Transfer Pump A	SA	Emergency generator A diesel fuel oil supply
Cable Trays	(Later)	
Fire Area Electric Area B		
Fire Area Dry Cooling Tower B		
Fire Area UH-6 Diesel Oil Storage Tank and Pump B		
Diesel Oil Storage Tank and Transfer Pump B	SB	Emergency generator and diesel fuel oil supply

11. Assessment of Postulated Fire Consequences

The above listed safety-related and non-safety-related equipment in each fire area could be damaged or lose function due to a fire. The fire and its associated effects are contained by the three-hour rated fire resistive barriers enclosing each fire area. Since the fire areas contain redundant equipment safe shutdown will not be affected by the same fire.

12. Safety Evaluation

Safety evaluation will be presented later as discussed in Subsection 9.5.1.3.1. Additional changes to design will be incorporated into the FHA at that time.

Attachment 1-45
2266A

Question No.

280.5 (9.5.1.5) Discuss the extent of conformance to the guidance given in NFPA 27.

Response

At the time our fire brigade training program is finalized and training commences, we will be in full compliance with the Recommendations for Organization, Training and Equipment of Private Fire Brigades, NFPA No. 27-1975. We anticipate that there will be no exceptions required.

Attachment 1-46
2266A

<u>Question No.</u>

281.1 (6.1.1.2) For all postulated design basis accidents involving release of water into the containment building, estimate the time-history of the pH of the aqueous phase in each drainage area of the building. Identify and quantify all soluble acids and bases within the containment.

<u>Response</u>

A complete response to this question will be available by December 1982.

Attachment 1-47
2266A

Question No.

281.2 (9.3.2) Subsection 9.3.2 does not include requirements to minimize, to the extent possible, hazards to plant personnel; provide these requirements or a date by which they will be supplied.

Response

The following safety features described in FSAR Subsection 9.3.2 are provided to minimize the hazards to plant personnel:

1. Sampling system is designed to limit radioactivity releases below the 10CFR20 limits under normal and failure conditions. An example of this is the samples from the Reactor Coolant System hot leg that must be provided with a contaminant delay time of 90 seconds to permit sufficient delay of nitrogen - 16. Due to the length of pipe inside containment and the sample flow rate, 135 seconds of delay will actually exist.

2. Instrument is provided to monitor the temperature and pressure of the samples to be taken. The instrumentation is provided to insure the samples are cooled by heat exchangers and their respective pressure reduced by pressure reducing valves to values less than 120F and 30 PSIG. This is to minimize the possibility of local airborne activity.

3. Samples are normally taken only when the sample hood fan is operating to minimize the possibility of local airborne activity. In addition, the Reactor Auxiliary Building Ventilation System provides a back-up means of maintaining low airborne activity levels.

4. The sample lines penetrating the containment are equipped with two isolation valves which close on actuation of the containment isolation actuation signal. In addition, should any of the remotely operated valves fail to close after a sample has been taken, back-up manual valves in the sampling room may be closed to limit the spread of contamination.

Attachment 1-48
2266A

Question No.

281.3 (10.4.6) As per Regulatory Guide 1.70, Revision 3, include a discussion of the control of chloride ions and other contaminants in the condensate cleanup system.

Response

Subsection 10.4.6.1 (Design Bases) states the following: "The purpose of the condensate demineralizer is to remove dissolved and suspended solids from the condensate in order to maintain a high quality of the feedwater being supplied to the steam generators under all normal plant conditions (start-up, shutdown, hot standby, power operation)."

The phrase "dissolved and suspended solids" includes chloride ions. The WNP-3 condensers are cooled using fresh water with a maximum chloride ion concentration of 50 ppm. Should the plant be operating with a postulated circulating water inleakage of five gpm due to condenser tube leakage, the maximum expected chloride ion concentration in the condensate will be 10 ppb. As the condensate cleanup system will be in operation during a condenser leak, any chloride ions in the condensate will be removed.

The FSAR will be amended to reflect the response to this question.

Q281.3
1 of 2

10.4.6 CONDENSATE CLEANUP SYSTEM

10.4.6.1 Design Bases

The purpose of the Condensate Demineralizer is to remove dissolved and suspended solids from the condensate in order to maintain a high quality of the feedwater being supplied to the steam generators under all normal plant conditions (startup, shutdown, hot standby, power operation). A flow diagram for the Condensate Cleanup System is provided on Figures 10.4-2a through g. This is accomplished by directing the full flow of condensate to a set of mixed bed demineralizer units. Since the demineralizers need periodic resin regeneration, spare units are provided in the system to replace units taken out of service. The system provides final polishing of the secondary cycle condensate water.

The system is designed to process the full flow of condensate corresponding to 100 percent power level with one heater drain pump out of service. This equals 28,506 gpm, and it is the highest condensate flow expected under all operating conditions.

The flow is split into 10 parallel units of mixed bed demineralizers, each designed to process 2,851 gpm.

Each demineralizer unit maintains the outlet water quality within the limits shown in Table 10.4.6-1 with a postulated circulating water in-leakage of five gpm due to condenser tube leakage. Circulating water quality is also described in Subsection 10.4.5 and Table 10.4.5-2.

See Insert 1

Two standby units are provided to replace units under maintenance or being regenerated.

Since only one demineralizer unit is expected to be exhausted at one time, the regeneration system is designed to handle the amount of resin of one mixed bed vessel at one time.

Since the system is located on the discharge of the low pressure condensate pumps, all related equipment is designed to withstand the pump's discharge pressure at shutoff conditions. The regeneration portion of the system is designed to withstand the maximum discharge pressure of the sluice water pumps, which are used to transfer resin to and from different tanks.

The design temperature which is 150F is determined by the maximum condensate temperature expected during plant operation, which is 126.7F at 4.14 in. Hg condenser pressure for power operation and 134F at five in. Hg for the feedwater cleanup cycle.

Since supplying feedwater to the steam generators at all times is necessary to ensure continuity of plant operation, the system is provided with a bypass line which can be used to bypass all or part of the condensate flow when some condensate demineralizer units are not available or their operation becomes impaired.

Insert 1

The WNP-3 condensers are cooled using fresh water with a maximum chloride ion concentration of 50 ppm. Should the plant be operating with a postulated circulating water in - leakage of five gpm due to condenser tube leakage, the maximum expected chloride ion concentration in the condensate will be 10 ppb. As the condensate cleanup system will be in operation during a condenser leak, any chloride ions in the condensate will be removed.

Attachment 1-49
2266A

Question No.

281.4 (10.4.8.1)	Subsection 10.4.8.1-e discusses the blowdown demineralizer systems removal of impurities from the blowdown and references Subsection 10.4.11. Subsection 10.4.11 states this system has been deleted from WNP-3. If this is correct, the referencing paragraph should also be deleted from the FSAR.

Response

The Steam Generator Blowdown Demineralizer System was deleted from the original design. This decision is now being reconsidered. We will supply additional information concerning this decision by January, 1983

Attachment 1-50
2266A

Question No.

311.1 (2.1.1.2) As per Regulatory Guide 1.70, the site area map should include the site boundary lines and if they are the same as the plant property lines, this should be stated.

Response

The site area map, FSAR Figure 2.1-1, provides the site boundary lines as well as the plant property lines. The plant property is indicated by the shaded area on Figure 2.1-1 and entitled "Fee Owned Property". The site boundary lines are defined on Figure 2.1-1 by the combination of the "Fee Owned Property" and the "Exclusion Zone Easement". The required exclusion boundary (4300 ft radius) is also shown on the figure.

Attachment 1-51
2266A

Question No.

311.2 (2.1.2.2)	As per Regulatory Guide 1.70, an estimate should be provided of the time required to evacuate all personnel from the exclusion area.

Response

Evacuation time estimates are provided in Section 12.5 of the WNP-3 Emergency Preparedness Plan. Table 12-5 (attached) provides evacuation times for the two, five, and 10-mile zones. Analysis of the one mile zone around WNP-3 indicates evacuation times within 10 minutes of that calculated for the two mile zone. The exclusion area (0.8 miles) can be approximated conservatively by the one mile zone data found on the attached table.

Q311.2

EP. 12-16

DESCRIPTION	TOTAL WITHIN 2 MILES*	AREAS WITHIN 5 MILES					AREAS WITHIN 10 MILES				
		I	II	III	IV	TOTAL	I	II	III	IV	TOTAL
PERMANENT POPULATION	109	4,098	152	—	1,617	5,867	6,207	1,100	20	7,838	15,165
PERMANENT POPULATION VEHICLES	55	2,049	76	—	809	2,934	3,104	550	10	3,919	7,583
TRANSIENT POPULATION	1,829	1,160	361	257	1,260	3,038	2,173	1,580	1,075	2,096	6,924
TRANSIENT POPULATION VEHICLES	1,165	780	181	129	880	1,970	1,437	790	538	1,298	4,063
GENERAL POPULATION	1,938	5,258	513	257	2,877	8,905	8,380	2,680	1,095	9,934	25,706
TOTAL VEHICLES	1,220	2,829	257	129	1,689	4,904	4,541	1,340	548	5,217	11,646
NOTIFICATION TIME MINUTES	30	30	30	30	30	30	30	30	30	30	30
PERMANENT POPULATION EVAC. TIME NORMAL CONDITIONS	1:05 (1:00)	1:10	1:00	—	1:10	1:10	2:00	1:20	—	1:35	2:00
GENERAL POPULATION EVAC. TIME NORMAL CONDITIONS	1:10 (1:05)	1:15	1:10	1:05	1:15	1:15	2:10	1:25	1:20	1:35	2:10
PERMANENT POPULATION EVAC. TIME ADVERSE CONDITIONS	1:15 (1:05)	1:30	1:05	—	1:45	1:45	3:05	1:50	—	2:25	3:05
GENERAL POPULATION EVAC. TIME ADVERSE CONDITIONS	2:20 (2:10)	1:35	1:30	1:25	2:50	2:50	3:25	2:15	2:10	2:55	3:25
CONFIRMATION TIME MINUTES	60	60	60	60	60	60	60	60	60	60	60
SPECIAL POPULATION EVAC. TIME—NORMAL CONDITIONS	—	1:40	—	—	3:30	3:30	1:40	—	—	3:30	3:30
SPECIAL POPULATION EVAC. TIME—ADVERSE CONDITIONS	—	2:50	—	—	6:00	6:00	2:50	—	—	6:00	6:00

HRS:MINS

* 1-mile data = ( )

TABLE 12-5 SUMMARY OF RESULT OF EVACUATION TIME ANALYSIS

From WNP-3 Emergency Preparedness Plan

Attachment 1-52
2266A

Question No.

311.3 (3.5.1.6)	You state that your analysis for aircraft hazards is forthcoming. Provide a schedule for furnishing this information.

Response

The aircraft hazards analysis results will be supplied by December 30, 1982.

Attachment 1-53
2266A

Question No.

410.1 (3.4.1.2) Provide or reference a discussion of the testing and inspection to be performed to verify that the groundwater drainage system capability and reliability are met and the instrumentation and control necessary for proper operation of the system are adequate.

Response

A discussion of the test performed to verify the design adequacy of the groundwater drainage system (GWDS) and establish the basis for the frequency of inspection required is provided in Subsection 3.4.2.

The in-service surveillance requirements of the groundwater drainage system is specified in Subsection 3.4.2.2.

Attachment 1-54
2266A

Question No.

410.2 (3.5.1.1)	Table 3.5.1-1 has several columns that have "under investigation" listed instead of the necessary data. Provide this data or a schedule for providing it.

Response

The missile evaluation for WNP-3 is currently being reviewed by the Supply System. This information will be available in January 1983.

Attachment 1-55
2266A

Question No.

410.3 (3.5.1.1) Provide or reference the following as specified in Regulatory Guide 1.70:

A tabulation showing the safety-related structures, systems, and components outside containment required for safe shutdown of the reactor under all conditions of plant operation should be provided and, as a minimum, should include the following:

1. Locations of the structures, systems, or components.

2. Applicable seismic category and quality group classifications (may be referenced from Section 3.2).

3. Sections in the SAR where descriptions of the items may be found.

4. Reference drawings or piping and instrumentation diagrams where applicable (may be referenced from other sections of the SAR).

5. Identification of missiles to be protected against, their source, and the bases for selection.

6. Missile protection provided.

The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles should be evaluated.

Response

The evaluation of WNP-3 for missiles outside containment is currently being reviewed by the Supply System. This information will be available by January 1983.

Attachment 1-56
2266A

Question No.

410.4 (3.5.1.2) Provide or reference the following as specified in Regulatory Guide 1.70:

A tabulation showing the safety-related structures, systems, and components inside containment required for safe shutdown of the reactor under all conditions of the plant operation, including operational transients and postulated accident conditions, should be provided and, as a minimum, should include the following:

1. Location of the structure, system, or component.

2. Identification of missiles to be protected against, their source, and the bases for selection.

3. Missile protection provided.

The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles should be evaluated.

Response

The evaluation of WNP-3 for missiles inside containment is currently being reviewed by the Supply System. This information will be available for January 1983.

Question No.

410.5 (3.5.1.4) Table 3.5.1-3 states the information for the dry cooling tower enclosure will be provided later. Provide this information or a schedule for furnishing it.

Response

The information indicated as later in Table 3.5.1-3 is:

Dry Cooling Tower Enclosure

Structure	Minimum Thickness	28 day Concrete Strength
Wall	2'-0"	50000 psi
Roof	Steel Grating	N/A

FSAR Table 3.5.1-3 will be amended to reflect this information.

TABLE 3.5.1-3

TORNADO MISSILE CONCRETE BARRIER

Structure(1)	Minimum Thickness (2)	28-Day Concrete Strength
Shield Building		
Cylindrical Wall	3'-0"	5000 psi
Dome	2'-6"	5000 psi
Reactor Auxiliary Building		
Wall	3'-6"	5000 psi
Roof	2'-0"	5000 psi
Fuel Handling Building		
Wall	3'-6"	5000 psi
Roof Slab	2'-0"	5000 psi
Condensate Storage Tank Enclosure		
Wall	3'-0"	4000 psi
Roof	Steel Grating	N/A
Dry Cooling Tower Enclosure		
Wall	Later 2'-0"	Later 5000 psi
Roof	Later Steel Grating	Later N/A

Notes:

1) The systems and components protected by these structures are identified in Table 3.2-1.

2) See Subsection 3.5.3 for barrier design details.

Attachment 1-58
2266A

Question No.

410.6 (3.5.1.4) The externally generated missile protection analyses should take into account the effect on ventilation openings in the various facility buildings housing essential shutdown equipment. Reference or provide a discussion addressing this subject.

Response

The ventilation openings in the Reactor Auxiliary Building and Dry Cooling Tower Control Building housing essential shutdown equipment are all provided with steel grating for protection from externally generated missiles. The type of steel grating used for missile protection of ventilation openings is identified by that described in Subsection 3.5.3.1.2. The barrier design procedure using steel grating is discussed in Subsection 3.5.3. Section 9.4 also contains discussions of missile protection for the ventilation openings.

Attachment 1-59
2268A

Question No.

410.7 (3.5.2) This section does not provide the detail required by Regulatory Guide 1.70 which states that it should be demonstrated that safety-related structures, systems and components are adequately protected against very low probability missile strikes by physical barriers or protective structures. According to the Standard Review Plan (NUREG-0800) this should even include such elements as essential service water intakes, buried components, and access openings and penetrations in structure. Provide or reference this level of detail for this FSAR section.

Response

A complete response to this question will be available by December 1982.

Attachment 1-60
2268A

Question No.

410.8 (9.2.5) Subsection 9.2.5 does not define the number of cells per cooling tower train, nor does it contain Figures 9.2.5-2a through 9.2.5-2d. Confirm the date by which you intend to supply this information.

Response

Each cooling tower train is divided into 10 cells. Figures 9.2.5-2a through 9.2.5-2d will be provided by February 1983.

The FSAR will be updated to reflect this response.

Q410.8
1of 2

9.2.5 ULTIMATE HEAT SINK

9.2.5.1 Design Basis

The Ultimate Heat Sink (UHS) provides heat rejection from the Component Cooling Water System (CCWS) to the atmosphere during safe plant shutdown or accident conditions. The description of the Component Cooling Water System is given in Subsection 9.2.2. The UHS operates in conjunction with the CCW Heat Exchangers, which rejects heat through the Service Water System (see Subsection 9.2.1) during normal operation and refueling. The UHS complies with regulatory positions defined in NRC Regulatory Guide 1.27, GDC 2, 4 and 44 and BTP ASB 9-2.

The UHS is capable of providing sufficient heat rejection during the most unfavorable anticipated meteorological conditions (as discussed in Subsection 2.3.2.1) for an indefinite period of time in order to:

a) Permit safe shutdown and cooldown of the plant and maintain the plant in a safe shutdown condition.

b) In the event of an accident, permit control of that accident safely.

The UHS is designed to ASME III Safety Class 3, seismic Category I, WPPSS Quality Class I requirements. The UHS is capable of withstanding the effects of natural phenomena as well as loss of offsite power coincident with a single failure. UHS component design data is given in Table 9.2.5-3

The failure of any single mechanical or electrical component of the UHS will not result in a loss of the heat sink safety function.

9.2.5.2 System Description

The preliminary design of the UHS is comprised of two independent 100 percent capacity trains. Each train has a cooling tower of an air-cooled heat exchanger type with CCWS fluid passing through the tube side and air over the extended surface of the tubes. Concrete structures provide seismic and missile protection as well as structural support for heat transfer equipment. As shown on Figure 9.2.5-1 Train A and Train B towers are separated by a concrete shear wall and respective electrical equipment rooms.

Each cooling tower train is divided into (later) 10 cells. Each cell is separated by a concrete shear wall and includes tube bundles, heat retention doors or louvers and induced draft fans. Fans are controlled to maintain appropriate CCWS water temperatures. Major component design data is presented in Table 9.2.5-3.

The UHS is capable of operating under a range of heat loads during normal and emergency conditions. At end of life conditions and 101.5F dry bulb ambient temperature, each train has a design heat rejection capability of 180 x 10^6 Btu/hr with 11,000 gpm CCWS flow entering at 153F and leaving at 120F. This capability exceeds the maximum expected heat rejection requirements as discussed below.

During normal operation the UHS in conjunction with the CCWS heat exchanger, can reject the maximum normal heat loads while maintaining CCWS temperature at or below 95F. Table 9.2.5-1 presents expected tower performance requirements during normal conditions. Parametric performance expectations for a range of ambient temperatures with and without heat rejection to the SWS under normal and emergency conditions is presented in Table 9.2.5-2. During emergency operation the UHS provides sufficient cooling to safety-related heat loads identified in Subsection 9.2.2. In the event of the lors of one train for any reason, the redundant train can reject the maximum instantaneous heat load. Non-essential heat loads can be manually aligned as reactor decay heat affords spare UHS capacity. The appropriate controls as discussed in Section 7.3 permit the operator to monitor and select these heat load alignments.

9.2.5.3 Safety Evaluation

The UHS with one tower is designed to meet maximum heat load operating alone following an accident. Its long term heat rejection capacity is sufficient to mitigate a postulated LOCA and return the containment pressure and temperature to ambient conditions.

The results of an analysis of the 30-day period following a design basis accident are found in Tables 9.2.5-4 and 9.2.5-5 and Figures 9.2.5-2a through 9.2.5-2d~~*~~. This analysis has determined the total heat rejected, the sensible heat rejected, the station auxiliary system heat rejected, and the decay heat release from the reactors.

As per Branch Technical Position ASB-9-2, the decay heat curves for fission products and for heavy elements were obtained using the assumptions and uncertainties set forth in the October 1973 draft proposed ANS standard, "Decay Energy Release Rates Following A Shutdown Of Uranium-Fueled Thermal Reactors" (ANS-5), to establish the heat input due to decay of radioactive material. An equilibrium fuel cycle and an increase in the calculated heat inputs were assumed as follows:

a) For the time interval 0 to 10^3 seconds, 20 percent was added to the heat released by the fission products to account for uncertainties in their nuclear properties.

b) For the time interval 10^3 to 10^7 seconds, 10 percent was added to the heat released by the fission products to account for uncertainties in their nuclear properties.

c) For the time interval 0 to 10^7 seconds, the heat released by the heavy elements was calculated (using the best estimate of the production rate for each unit) and 10 percent was added to account for uncertainties in their nuclear properties.

~~*Later~~

Attachment 1-61
2268A

Question No.

410.9 (9.2.6)	Subsection 9.2.6 does not contain a storage facility failure analysis. Provide this analysis or a date by which it will be supplied.

Response

The condensate storage facility failure analysis has been included in the Auxiliary Feedwater System Availability Analysis provided in FSAR Appendix 10.4.9A.

Attachment 1-62
2268A

Question No.

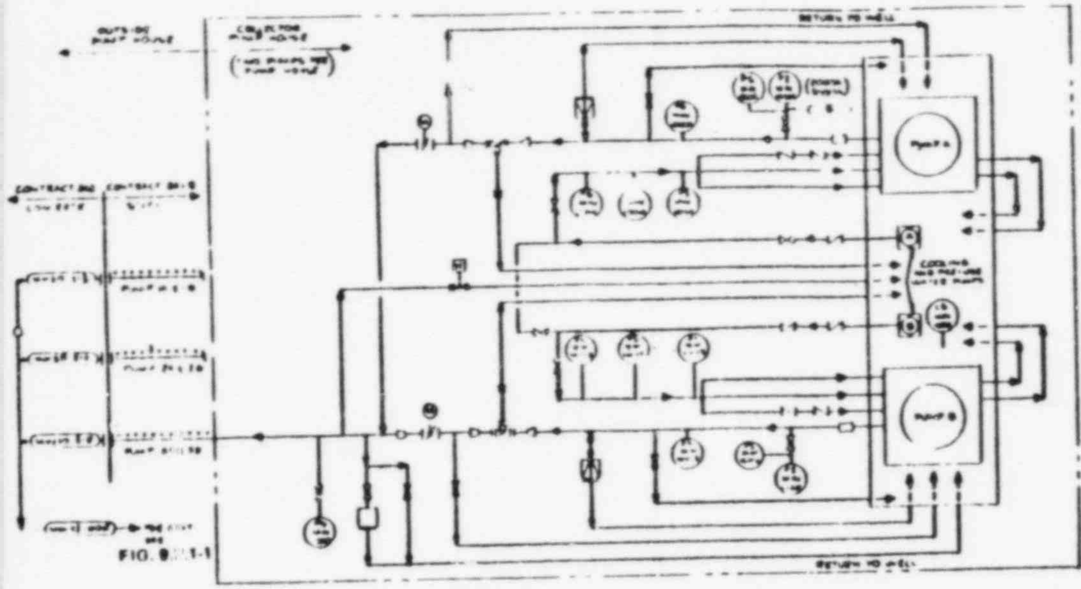
410.10 (9.2.8) Subsection 9.2.8 does not contain Figure 9.2.8-1; provide this drawing or a date by which it will be supplied.

Response

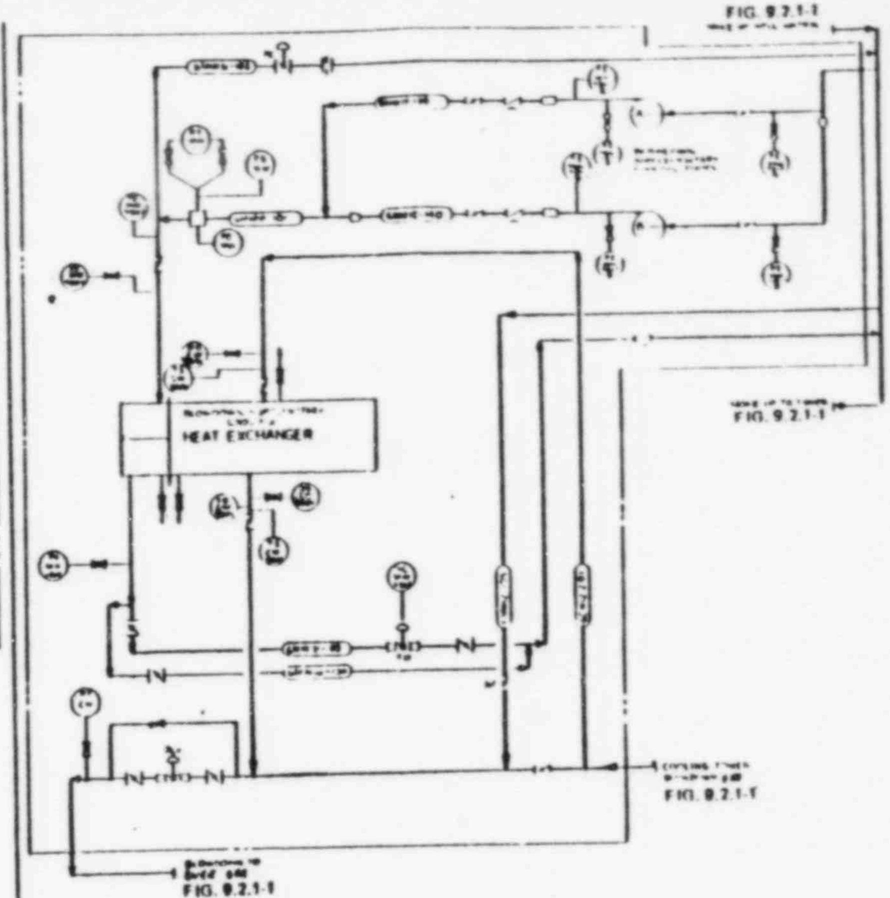
The Plant Makeup Water System flow diagram was included in the FSAR as Figure 9.2.1-3. This figure number will be revised to Figure number 9.2.8-1.

The FSAR will be updated to reflect this response.

Q410.10
1 of 1



MAKE-UP WELL SYSTEM



BLOWDOWN SUPPLEMENTARY COOLING

REF DWG: WPPS-3240-G-1098 303

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
Nuclear Projects 3 & 5
FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM – CIRCULATING, SERVICE AND PLANT MAKE-UP WATER SYSTEM
FIGURE 9.2.1-[illegible]

9.2.8-1

Attachment 1-63
2268A

Question No.

410.11 (9.4) Section 9.4 does not include piping and instrumentation diagrams for any of the systems discussed; provide these diagrams or a date by which they will be supplied.

Response

The piping and instrumentation diagrams for the HVAC Systems discussed in Section 9.4 are the air flow diagrams shown on Figures 9.4.1-1 through 9.4.7-1 and the respective instrument schematic-logic diagrams are shown in Section 7.3 on Figures 7.3-4 through 7.3-92.

This FSAR will be amended to include the above response.

9.4 AIR CONDITIONING, HEATING, COOLING AND VENTILATION SYSTEMS

The following sections describe the heating, ventilating and air conditioning (HVAC) systems serving the plant during normal and emergency operating conditions. Table 9.4.1-4 provides an FSAR location guide for the various HVAC and related systems. Systems are designed to provide a suitable environment for equipment and personnel. Ventilation zones and air distribution are arranged so that the ventilation air is drawn from clean areas to areas of potentially greater radioactive contamination to final filtration and exhaust. The ventilation systems air flow diagrams are shown on Figures 9.4.1-1 through 9.4.7-1 and the respective instrument schematic-logic diagrams are shown in Section 7.3 on Figures 7.3-4 through 7.3-92. Table 9.4.1-1 indicates the space temperatures in the plant during normal plant operation. System's equipment classifications are listed in Table 3.2-1.

INSERT

9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM

The Control Room HVAC System is designed to control the environment of the Control Room Envelope as defined in Subsection 6.4.2.1. The Control Room Envelope will be referred to as the "Control Room". The Control Room Area Ventilation System is in compliance with Regulatory Guide 1.78 and 1.95 requirements. The Control Room Emergency Air Cleaning Units are designed and constructed in accordance with the recommendations of Regulatory Guide 1.52, with exception as delineated in Table 6.5-2.

9.4.1.1 Design Bases

The Control Room HVAC System is designed to:

a) Exclude entry of airborne radioactivity into the Control Room and remove radioactive material from the Control Room environment such that inhalation dose to Control Room personnel is within General Design Criterion 19 (Appendix A of 10CFR50) limits.

b) Maintain the Control Room ambient temperature to assure personnel comfort during normal plant operation as shown in Table 9.4.1-1, equipment qualification temperatures are addressed in Section 3.11.

c) Permit personnel occupancy and proper functioning of instrumentation and controls during all normal and design basis accident conditions assuming a single active failure.

d) Withstand the effects of tornado and tornado missile. The HVAC system, with the exception of some ductwork located in office areas and emergency living quarters, is designed to seismic Category I requirements.

e) Provide Control Room personnel protection by detection and limit of the introduction of smoke and chlorine into the Control Room.

f) Provide accessibility for adjustment and periodic inspections and testing of the system components to assure continuous functional reliability.

INSERT

and the respective instrument schematic-logic diagrams are shown in Section 7.3 on Figures 7.3-4 through 7.3-92.

Attachment 1-64
2268A

Question No.

410.12 (9.4.7) Subsection 9.4.7 states that the CCWS Dry Cooling Towers Electrical Equipment Room Ventilation System design is conceptual only. Confirm the date by which you intend to supply this information.

Response

The CCWS Dry Cooling Towers Electrical Equipment Rooms Ventilation System design is not final at this time. Subsection 9.4.7 will be amended after completion of the design and will be supplied in January 1983.

Question No.

410.13 (10.4.7) As per Regulatory Guide 1.70, Revision 3, provide a discussion of the piping analysis, including forcing function, or results of test programs performed to verify that the uncovering of the feedwater lines could not occur or that the uncovering would not result in unacceptable damage to the system.

Response

The information requested in Regulatory Guide 1.70, Revision 3 applies to steam generators utilizing a top feed design. WNP-3 utilizes an integral economizer type steam generator. Therefore, this item is not addressed.

Attachment 1-66
2268A

Question No.

410.14 (10.4.9.3) Subsection 10.4.9.3 discusses the design to prevent water hammer in the pipe routing of the auxiliary feedwater system to the steam generators. It further states that tests acceptable to the NRC will be performed to verify unacceptable water hammer will not occur. Describe the proposed tests, how they will be conducted and when they will be conducted.

Response

A description of the proposed tests to verify acceptable water hammer mitigation in pipe routing of the auxiliary feedwater system to the steam generators will be available after July of 1983. Results of testing at other utility projects are being closely followed to determine test requirements. At this time it is believed that any necessary testing would occur during pre-core hot functional.

Attachment 1-67
2268A

Question No.

421.1 (7.3) Table 7.3-20 is shown as "to be supplied later". Provide either the table or a date by which it will be supplied.

Response

Table 7.3-20 is part of the description of the Dry Cooling Tower Electrical Equipment Room HVAC System. The design of the system is not finalized at this time. Upon completion of the design the following steps will be taken to update Chapter 7:

a) Complete Subsection 7.3.1.1.4.8, "CCWS Dry Cooling Towers Electrical Equipment Rooms HVAC"; The description of the system will be given including initiating circuits, logic, bypasses, interlocks, sequencing, redundancy, diversity and actuated devices.

b) Add Subsection 7.3.1.3.19 - The Final System Drawings including Instrument Schematics and Control Logic Diagrams and Instrument Location Arrangement Drawings will be given.

c) Complete Table 7.3-20, "CCWS Dry Cooling Tower Electrical Equipment Room HVAC Failure Mode and Effects Analysis".

This information will be supplied in January 1983.

Q421.1
1 of 3

The control logic associated with the Diesel Generator Area HVAC System and typical control wiring diagrams are provided in Subsection 7.3.1.3.

Bypasses

Components of the Diesel Generator Areas HVAC System are monitored continuously for bypassed or inoperable conditions and their status is indicated on both a system and a component basis on the Bypassed Inoperable Status Panels in the Control Room. System status is alarmed in the Main Control Board (CB-1). See Subsection 7.5.1.8 for a complete discussion of the Bypassed-Inoperable Status Panels and conformance to Regulatory Guide 1.97.

Redundancy

Each diesel generator area has its own air handling unit HV2A(B). Upon failure of the operating Diesel Generator Area HVAC System, an alarm is sounded in the Main Control Room.

Sequencing

The Diesel Generator Areas HVAC System is automatically loaded onto Load Block V of its associated diesel generator following a loss of offsite power. See Section 8.3 for discussion of the diesel generator load sequencing.

The non-safety electrical reheat coils have normal 120V ac power supply only and are not utilized following LOOSP.

Supporting Systems

The following systems support the Diesel Generator Areas HVAC System:

1) Emergency AC Diesel Generators
2) Reactor Auxiliary Building Main Ventilation System
3) 480V AC System
4) Vital AC System

7.3.1.1.4.8 CCWS Dry Cooling Towers Electrical Equipment Rooms HVAC

~~LATER~~ Upon completion of the design, the description of the system will be given including initiating circuits, logic, bypasses, interlocks, sequencing, redundancy, diversity and actuated devices.

7.3.1.1.4.9 Component Cooling Water System

The Component Cooling Water System (CCWS) removes heat from reactor auxiliary systems and the reactor coolant system during normal operation, normal shutdown and post-accident conditions. The CCWS is divided into essential and

2

7.3.1.3.17 Safety-Related 125 Volt DC System

Refer to Chapter 8.

7.3.1.3.18 Typical Control Wiring Diagrams (CWD)

Typical CWD's for system equipment and/or components are:

		Figure No.
1)	Solenoid Valve	7.3-86
2)	Air Operated Valve	7.3-87
3)	Motor Operated Valve	7.3-88
4)	Pump or Fan (>100 Horsepower)	7.3-89
5)	Pump or Fan (< 100 Horsepower)	7.3-90

→ add 7.3.1.3.19 Final System Drawings

Upon completion of the design, the final system drawing's including Instrument schematics and Control Logic Diagrams and Instrument Arrangement Drawings will be given.

TABLE 7.3-20

CCW DRY COOLING TOWER ELECTRIC EQUIPMENT ROOM HVAC
FAILURE MODES AND EFFECTS ANALYSIS

(~~TO BE SUPPLIED LATER~~)

to be supplied by January 1983

3 of 3

Attachment 1-68
2268A

Question No.

421.2 (7.6.2) Section 7.6.2 contains no analyses of instrumentation installed to prevent or mitigate the consequences of cold water slug injections and overpressurization of low-pressure systems; reference or provide these analyses or show these analyses are not applicable to WNP-3.

Response

Instrumentation and equipment to protect NSSS systems from cold water slug injections and overpressurization of low-pressure systems are within the CESSAR-F scope as noted in Section 7.6.2.1.6 of the WNP-3 FSAR. Therefore, this issue has been reviewed on the CESSAR-F docket and accepted in NUREG-0852, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80".

Attachment 1-69
2268A

Question No.

430.1 (80) The Utility Grid Description (Subsection 8.1.1) and the Offsite Power System (Section 8.2) should be revised to reflect the cancellation of WNP-5 and any consequential changes in the BPA grid structure.

Response

The Utility Grid Description (Subsection 8.1.1) and the Offsite Power System (Section 8.2) text and figures have been revised to reflect the termination of WNP-5 and provided in Amendment No. 1 to the WNP-3 FSAR.

Question No.

430.2 (10.4.1) As per Regulatory Guide 1.70, Revision 3, specifically provide or reference discussions of the following:

1) The anticipated inventory of radioactive contaminants in the main condensers during operation and during shutdown,

2) anticipated air leakage limits,

3) control functions that could influence operation of the primary coolant or secondary systems,

4) protection of safety-related equipment from flooding resulting from failure of the condenser,

5) a procedure to repair condensate leaks and

6) the length of time the condenser can operate with degraded conditions without affecting the condensate/feedwater quality for safe operation.

Response

The main condenser serves no safety function. During normal operation and shutdown, the main condenser does not contain radioactive contaminants. Radioactive contaminants can only be present through primary-to-secondary system leakage due to a steam generator tube leak. Therefore, the six items as required per Regulatory Guide 1.70, Revision 3, are not applicable to the normal operation of the main condenser, as described in Subsection 10.4.1.

The following are responses to each of the specific items:

1. "The anticipated inventory of radioactive contaminants in the main condensers during operation and during shutdown."

   Subsection 10.4.1.3, "Safety Evaluation" indicates that "During startup, normal operation and shutdown, the main condenser contains no radioactive contaminants. Noncondensible gases will be monitored for radioactivity prior to being discharged to the atmosphere as discussed in Subsection 10.4.2. A full discussion of the radiological aspects of primary to secondary leakage, including anticipated operation concentrations of radioactive contaminants is described in Section 11.1."

Question No.

430.2 (contd.)

2. "Anticipated air leakage limits"

   The Main Condenser Air Evacuation System, described in Subsection 10.4.2, evacuates the main condenser and turbine steam space during plant startup and thereafter maintains the condenser design vacuum of 2.24/3.08/4.14 in. of Hg. absolute in LP/IP/HP condenser shell during normal plant operation. The system maintains this vacuum by continuously removing non-condensible gases and air in leakage from the three condenser shells. The Condenser Air Evacuation System, as well as the Main Condenser, do not perform a safety function.

3. "Control functions that could influence operation of the primary coolant or secondary systems."

   There are no controls in the operation of the condenser that could influence the operation of the primary coolant system. Operation of the condenser in relation to secondary systems is discussed in Subsections 10.4.1.2 and 10.4.1.3.

4. "Protection of safety-related equipment from flooding resulting from failure of the condenser."

   No protection of safety-related equipment from flooding resulting from failure of the condenser is required. There is no safety-related equipment located in the Turbine Building.

5. "A procedure to repair condensate leaks"

   Subsection 10.4.1.5 describes methods of detecting and locating condenser tube leaks. The following is the procedure to be followed to repair condensate leaks: In case of inleakage to the condenser, the affected condenser section is removed from service, drained of cooling water and the leaking tubes are located and plugged. The affected condenser section is returned to service when repairs are completed.

   Subsection 10.4.1.2 will be amended to reflect this response.

6. "The length of time the condenser can operate with degraded conditions without affecting the condensate/feedwater quality for safe operations."

Question No.

430.2 (contd.) The main condenser is not required to maintain the integrity of the Reactor Coolant Pressure Boundary, and the ability to shutdown the reactor and maintain it in a safe shutdown condition. Condenser operation with degraded conditions, such as inleakage, loss of vacuum, etc. are discussed in Subsection 10.4.1.3.

the reaction of Hydrazine with free oxygen. By maintaining the proper chemistry, concentrations of impurities will not occur. Thus, no harmful impurities will form in any scratch, nondetectable weld imperfection or dent. Corrosion is also controlled by the addition of chemicals and ion exchange in the condensate demineralizers. Stress corrosion cracking will not occur because of the low operating temperatures, low chloride level and the use of stress relieved tubes. The tube material is stainless steel which has the highest resistance to erosion.

The main condenser hotwell serves as a reservoir which supplies the low pressure condensate pumps. Minimum hotwell storage capacity is sufficient for providing adequate volume for condensate and feedwater system surge protection. The storage capacity is about 116,000 gallons of water, equivalent to five minutes of condensate flow at full load operation. The condenser hotwell level is maintained by makeup from the condensate system (Subsection 9.2.6). The main condenser is serviced by an automatic makeup and letdown system to sustain a normal level in the hotwell. At low hotwell level a control valve opens to admit condensate from the demineralized water storage tank. At high hotwell level another control valve opens to discharge excess condensate from the system directly to the demineralized water storage tank via the low pressure condensate pump.

The main condenser is cooled by circulating water (Subsection 10.4.5) which flows in series through tubes in the three condenser shells.

The use of divided water boxes on each shell permits isolation of one-half of the total circulating water flow through each shell. This permits access to the isolated water box on each shell for repair and/or inspection while one-half of the circulating water flows through the other water box.

insert 1

Equalizing pipes between hotwells are provided so that only one hotwell level control is required.

Crossover ducts are provided between condenser shells for equalizing the exhaust steam flow.

10.4.1.3 Safety Evaluation

During startup, normal operation and shutdown, the main condenser contains no radioactive contaminants. In the event of primary to secondary tube leakage, radioactive contaminants will be present in the shell side of the steam generator. This will eventually enter the condenser. Noncondensible gases will be monitored for radioactivity prior to being discharged to the atmosphere as discussed in Subsection 10.4.2. A full discussion of the radiological aspects of primary to secondary leakage, including anticipated operating concentrations of radioactive contaminants is described in Section 11.1.

The main condenser is not required to maintain the integrity of the Reactor Coolant Pressure Boundary, and the ability to shutdown the reactor and maintain it in a safe shutdown condition.

insert 1

In case of inleakage to the condenser, the affected condenser section is removed from service, drained of cooling water and the leaking tubes are located and plugged. The affected condenser section is returned to service when repairs are completed.

Attachment 1-73
2268A

Question No.

440.1 (5.4.7.1.3) Provide the post-LOCA design heat load for the shutdown cooling heat exchangers. Also, Note 2 in Table 5.4.7-2 should be replaced by specific information rather than a general reference to the applicant's SAR.

Response

The Supply System, in conjunction with Combustion Engineering and Ebasco Services Inc., is preparing this information for submittal. The response should be complete by December 1982.

Attachment 1-74
2268A

Question No.

451.1 (2.3.1.2) As per Regulatory Guide 1.70, provide estimates of the weight of the 100-year return period snowpack and weight of the 48-hour Probable Maximum Winter Precipitation for the site vicinity. Using these estimates, provide the weight of snow and ice on the roof of each safety-related structure.

Response

The weight of the 100-year return period snowpack is estimated to be 19.7 psf corresponding to a snow depth of 25.2 inches. The weight of the 48-hour Probable Maximum Winter Precipitation (PMWP) is estimated to be 147.1 psf corresponding to a water depth of 28.28 inches. These estimates together with the effects on the roof of the safety-related structures are presented in Subsection 2.4.2.3.

TABLE 15-2 (Cont'd)

15.6.2 - Double-Ended Break of Letdown Line Outside Containment

	No Spike	Pre-existing Spike	Event Induced Spike
Kr-85m	2.4(1)	same	same
Kr-85	5.2(-1)	same	same
Kr-87	1.3(1)	same	same
Kr-88	3.6(1)	same	same
Xe-131m	3.2(0)	same	same
Xe-133	3.4(2)	same	same
Xe-135	5.7(1)	same	same
Xe-138	8.7(0)	same	same
I-131	5.3(0)	6.7(1)	1.2(2) DEC* I-131
I-132	1.1(0)	1.4(1)	— later
I-133	5.7(0)	7.3(1)	—
I-134	8.7(-1)	1.1(1)	—
I-135	3.9(0)	5.0(1)	—

15.6.3 - Steam Generator Tube Rupture With Loss of Offsite Power (2 hour)

	No Spike	Pre-existing Spike	Event Induced Spike
Kr-85m	9.3(-1)	same	same
Kr-85	3.5(-1)	same	same
Kr-87	8.5(0)	same	same
Kr-88	2.5(1)	same	same
Xe-131m	2.2(1)	same	same
Xe-133	2.2(2)	same	same
Xe-135	3.9(1)	same	same
Xe-138	5.9(0)	same	same
I-131	9.1(0)	1.2(2)	1.5(2) DEC* I-131
I-132	1.8(0)	2.4(1)	—
I-133	9.9(0)	1.3(2)	— later
I-134	1.5(0)	1.9(1)	—
I-134	6.7(0)	8.6(1)	—

* Dose Equivalent Curie

TABLE 15-2 (Cont'd)

15.6.3 - Steam Generator Tube Rupture (8 hour)

	No Spike	Pre-existing Spike	Event Induced Spike
Kr-85m	2.3(1)	same	same
Kr-85	9.0(-1)	same	same
Kr-87	9.7(1)	same	same
Kr-88	[illegible].5(1)	same	same
Xe-131m	2.5(0)	same	same
Xe-133	2.6(2)	same	same
Xe-135	4.4(1)	same	same
Xe-138	6.7(0)	same	same
I-131	9.1(0)	1.2(2)	1.5(2) DE=* I-131
I-132	1.8(0)	2.4(2)	- later
I-133	1.0(1)	1.3(2)	—
I-134	1.5(0)	1.9(1)	—
I-135	6.7(0)	8.6(1)	—

15.6.5 - Loss of Coolant Accident

Nuclide	Curies Containment Atmosphere	2 hr Release	8 hr Release	30 Day Release
Kr-85m	2.95(7)	3.7785(3)	1.1210(4)	1.5202(4)
Kr-85	9.36(5)	1.3890(2)	6.4622(2)	2.8514(4)
Kr-87	5.41(7)	5.0429(3)	8.0183(3)	8.1223(3)
Kr-88	7.73(7)	9.1264(3)	2.2101(4)	2.5322(4)
Xe-131m	8.24(5)	1.2201(2)	5.6319(2)	1.2423(4)
Xe-133	2.37(8)	3.4987(4)	1.5993(5)	1.9821(6)
Xe-135m	4.78(7)	1.3879(3)	1.3941(3)	1.3941(3)
Xe-135	4.24(7)	5.8591(3)	2.1623(4)	4.2706(4)
Xe-13[illegible]	2.09(8)	-	-	-
Xe-138	1.89(8)	5.0640(3)	5.6779(3)	5.0782(3)
I-131	2.93(7)	1.7476(2)	4.5884(2)	6.1862(3)
I-132	4.28(7)	2.5461(2)	6.6189(2)	4.4326(3)
I-133	5.90([illegible])	3.4607(2)	8.4428(2)	2.2769(3)
I-134	6.38(7)	2.5614(2)	2.8299(2)	2.8320(2)
I-135	5.50(7)	3.0921(2)	6.4080(2)	9.5351(2)

* Dose Equivalent Curie

TABLE 15-3 (Cont'd)

15.6.2 Double-Ended Break of a Letdown Line Outside Containment

No. Iodine Spike

	Whole Body	Thyroid
EAB (2 hr)	2.7(-2)	8.4(0)
LPZ (8 hr)	3.7(-4)	1.2(0)

Pre-existing Spike

	Whole Body	Thyroid
EAB (2 hr)	1.3(-1)	1.1(2)
LPZ (8 hr)	1.8(-2)	1.5(1)

Accident Induced Spike

~~Later~~

EAB (2 hr)	2.7(-2)	4.5(1)
LPZ (8 hr)	3.8(-3)	1.0(1)

15.6.3 Steam Generator Tube Rupture With LOP

No. Iodine Spike

	Whole Body	Thyroid
EAB (2 hr)	1.62(-2)	4.0(0)
LPZ (8 hr)	2.50(-3)	5.7(-1)

Pre-existing Spike

	Whole Body	Thyroid
EAB (2 hr)	6.7(-2)	5.2(1)
LPZ (8 hr)	9.5(-3)	7.3(0)

Accident Induced Spike

~~Later~~

EAB (2 hr)	2.0(-2)	6.1(1)
LPZ (8 hr)	3.0(-3)	1.5(1)

15.6.5 Loss of Coolant Accident (LOCA)

	Thyroid	Whole Body
EAB (2 hr)	2.5(2)	1.4(1)
LPZ (30 day)	1.7(2)	5.4(0)

15.6.5 30-Day Control Room Doses Following LOCA

Thyroid	Whole Body	Skin
1.6(0)	1.2(-1)	2.5(0)

Attachment 1-76
2268A

Question No.

460.1 (11.5.2.4.2) Supply information relating to the effluent radiation monitors for steam generator blowdown flash tank vent and steam seal gland steam condenser ventilation which the FSAR indicates as later or provide a schedule for submittal of this information.

Response

The following is the information indicated as later in Subsection 11.5.2.4.2:

Steam Generator Blowdown Flash Tank Vent Radiation Monitor

The steam generator blowdown flash tank vent radiation monitor provides plant operations personnel with an indication and record of contamination of the Secondary Steam System and the potential for release via the steam generator blowdown flash tank vent. This contamination could occur due to leakage of primary reactor coolant into the secondary coolant through a steam generator.

This monitor is an ambient type monitor located next to the steam generator blowdown, flash tank vent line 6BD12-200 at EL. 423 ft. in the Turbine Building. The monitor is collimated with a lead shield to reduce the effect of background. The unshielded portion of the detector has an unobstructed view of the vent line. The ambient monitor is described in Subsection 11.5.2.3.

The measured activity level is automatically transmitted to the system computer where it is recorded and available for display. If the activity exceeds setpoints an annunciation is made through the system CRTs and event typer. Receipt of these alarms will alert the operators to the possibility of contamination of the Secondary Steam System and indicate the need for additional sampling and further action.

The alarm setpoints are selected above plant background to give the greatest sensitivity for possible contamination without causing frequent false alarms. These setpoints may be adjusted continuously over the entire range of the monitor.

Steam Seal Gland Steam Condenser Exhaust Radiation Monitor

The steam seal gland steam condenser exhaust radiation monitor provides plant operations personnel with an indication and record of contamination of the Secondary Steam System and the potential for release via the steam seal gland steam condenser vent. This contamination could occur due to leakage of primary reactor coolant into the secondary coolant through a steam generator.

Attachment 1-77
2268A

Question No.

460.1 (11.5.2.4.2) (contd.)

The monitor is an ambient type monitor located next to the vent line 6AE10-022 on EL. 455 ft. of the Turbine Building.

The monitor is collimated with a lead shield to reduce the effect of background. The unshielded portion of the detector has an unobstructed view of the vent line. The ambient monitor is described in Subsection 11.5.2.3.

The measured activity level is automatically transmitted to the system computer where it is recorded and available for display. If the activity exceeds setpoints an annunciation is made through the system CRTs and event typer. Receipt of these alarms will alert the operators to the possibility of contamination of the Secondary Steam System and indicate the need for additional sampling and further action.

The alarm setpoints are selected above plant background to give the greatest sensitivity for possible contamination without causing frequent false alarms. These setpoints may be adjusted continuously over the entire range of the monitor.

FSAR Subsection 11.5.2.4.2 will be amended to reflect the addition of this information.

Information pertaining to the sensitivity of these monitors, now missing from Table 11.5-1, will be available by February 1983.

Q460.1
1 of 4

pre-established setpoints an annunciation is made through the system CRTs and event typer. If the activity exceeds the high radiation alarm setpoint, or if the monitor fails, as determined by the local microprocessor, a contact closure is made at the local microprocessor which is used to automatically terminate the waste gas discharge.

The receipt of these alarms will alert the operators to analyze additional gas samples to determine the reason for the alarm. The records of the total quantity of radioactive material released is used in writing the reports required by Regulatory Guide 1.21. The alarm setpoints are selected in consideration of the requirement to prevent activity concentrations at the plant boundary or beyond from exceeding 10CFR20 limits, and to support the release limits set in the plant technical specification. The setpoints may be adjusted continuously over the entire range of the monitor. The range of this monitor was selected to span the expected range of radioactive gas concentrations expected in the waste gas.

e) Steam Generator Blowdown Flash Tank Vent Radiation Monitor

~~(LATER)~~ see Insert 1

f) Auxiliary Condensate Flash Tank Radiation Monitor

The auxiliary condensate flash tank radiation monitor provides plant operations personnel with an indication and record of contamination of the Auxiliary Steam System and the potential for release via various vents in the Auxiliary Steam and Condensate System. This contamination could occur due to in-leakage from the various radioactive systems that are serviced by the Auxiliary Steam System.

This monitor is an ambient type monitor located next to the auxiliary condensate flash tank in the Reactor Auxiliary Building next to line 6BD12-200 on the 335 ft. level. The monitor is collimated with a lead shield to reduce the effect of background. The unshielded portion of the detector has an unobstructed view of the flash tank. The ambient monitor is described in Subsection 11.5.2.3.

The measured activity level is automatically transmitted to the system computer where it is recorded and available for display. If the activity exceeds setpoints an annunciation is made through the system CRTs and event typer. Receipt of these alarms will alert the operators to the possibility of contamination of the Auxiliary Steam and Condensate System and indicate the need for additional sampling and further action.

The alarm setpoints are selected above plant background to give the greatest sensitivity for possible contamination without causing frequent false alarms. These setpoints may be adjusted continuously over the entire range of the monitor.

g) Steam Seal Gland Steam Condenser ~~Ventilation~~ Exhaust Radiation Monitor

~~(LATER)~~
see Insert 2

£

Insert 1

The steam generator blowdown flash tank vent radiation monitor provides plant operations personnel with an indication and record of contamination of the Secondary Steam System and the potential for release via the steam generator blowdown flash tank vent. This contamination could occur due to leakage of primary reactor coolant into the secondary coolant through a steam generator.

This monitor is an ambient type monitor located next to the steam generator blowdown flash tank vent line 6BD12-200 at El. 423 ft. in the Turbine Building. The monitor is collimated with a lead shield to reduce the effect of backround. The unshielded portion of the detector has an unobstructed view of the vent line. The ambient monitor is described in Subsection 11.5.2.3.

The measured activity level is automatically transmitted to the system computer where it is recorded and available for display. If the activity exceeds setpoints an annunciation is made through the system CRTs and event typer. Receipt of these alarms will alert the operators to the possibility of contamination of the Secondary Steam System and indicate the need for additional sampling and further action.

The alarm setpoints are selected above plant backround to give the greatest sensitivity for possible contamination without causing frequent false alarms. These setpoints may be adjusted continuously over the entire range of the monitor.

Insert 2

The steam seal gland steam condenser radiation monitor provides plant operations personnel with an indication and record of contamination of the Secondary Steam System and the potential for release via the steam seal gland steam condenser vent. This contamination could occur due to leakage of primary reactor coolant into the secondary coolant through a steam generator.

The monitor is an ambient type monitor located next to the vent line 6AE10-022 on El. 455ft of the Turbine Building.

The monitor is collimated with a lead shield to reduce the effect of background. The unshielded portion of the detector has an unobstructed view of the vent line. The ambient monitor is described in Subsection 11.5.2.3.

The measured activity level is automatically transmitted to the system computer where it is recorded and available for display. If the activity exceeds setpoints an annunciation is made through the system CRTs and event typer. Receipt of these alarms will alert the operators to the possibility of contamination of the Secondary Steam System and indicate the need for additional sampling and further action.

The alarm setpoints are selected above plant background to give the greatest sensitivity for possible contamination without causing frequent false alarms. These setpoints may be adjusted continuously over the entire range of the monitor.

TABLE 11.5-1 (Cont'd)

Name (Instrument Tag Number)	Q'ty	Design Background (mR/hr Co-60)	Sampler Type	Activity Measured	Sensitivity @ Background	Range $\mu Ci/cc$	Typical Alarm Set-points $\mu Ci/cc$	Automatic Actions Initiated	Location	Duty
Administration Building Discharge Radiation Monitor (RE-RM-0008 & RE-RM-0009)	1	2.5	Two Stage Airborne	Gross β Particulate & Gas	$1x10^{-9}$ $\mu Ci/cc$ Sr-90 @ 1 mR/hr Co-60(2) $1x10^{-6}$ $\mu Ci/cc$ Kr-85 @ 1 mR/hr Co-60(3)	$1x10^{-10}$ to $1x10^{-5}$ $1x10^{-7}$ to $1x10^{-2}$	$3x10^{-10}$ $3x10^{-9}$ $5x10^{-7}$ $5x10^{-6}$	Alarm Only	Inside WR-51	Continuous
Condenser Mechanical Vacuum Pump Discharge Radiation Monitor (RE-AE-1400)	1	2.5	One Stage Airborne	Gross β Gas	$1x10^{-6}$ $\mu Ci/cc$ Xe-133 @ 1 mR/hr Co-60(3)	$1x10^{-9}$ to $1x10^{-2}$	$1x10^{-4}$ $1x10^{-3}$	Alarm Only	Line 6AK16-021	Continuous
Waste Gas Discharge Radiation Monitor (RE-WM-0648)	1	2.5	One Stage Airborne	Gross β Gas	$1x10^{-2}$ $\mu Ci/cc$ Kr-85 mR/hr Co-60(3)	$1x10^{-3}$ to $1x10^{+3}$	$2x10^{-1}$ $2x10^{0}$	Alarm Only	6GW1-137R	Batch
Steam Generator Blowdown Flash Tank Vent Radiation Monitor	1 ~~[illegible]~~		Ambient Pb Columnated	Gross γ	* ~~Later~~	$1x10^{-1}$ to $1x10^{4}$ mR/hr	Above ambient background	Alarm Only	Line 6BD12-200	Continuous
Auxiliary Condensate Flash Tank Radiation Monitor	1	2.5	Ambient Pb ~~collimated~~ columnated	Gross γ	* ~~Later~~	$1x10^{-1}$ to $1x10^{4}$ mR/hr	Above ambient background	Alarm Only	Near Aux. Cond. Tank Flash	Continuous
Steam Gland Seal Steam Exhaust ~~Discharge~~ Radiation Monitor	1 ~~[illegible]~~ 2.5		Ambient Pb Columnated	Gross γ	* ~~Later~~	$1x10^{-1}$ to $1x10^{4}$ mR/hr	Above ambient background	Alarm Only	Line 6AE10-022	Continuous
Hot Machine Shop Discharge Sampler (RE-RM-0010)	1	2.5	Sampler	None	N.A.	N.A.	N.A.	None	Inside CU-51	Continuous
Waste Management System Discharge Radiation Monitor (RE-WM-6213)	1	2.5	Liquid	Gross γ	$1x10^{-6}$ $\mu Ci/cc$ Cs-137 @ 1 mR/hr Co-60(1)	$5x10^{-7}$ to $5x10^{-2}$	$5x10^{-5}$ $5x10^{-4}$	Alarm Termination of Discharge	Line 6LS3-080 WMF-3 6LS3-101 WMF-5	Batch
Common Plant Effluent Radiation Monitor (RE-CW-8830)	1	2.5	Liquid	Gross γ	$1x10^{-6}$ $\mu Ci/cc$ C -137 @ 1 mR/hr Co-60(1)	$5x10^{-7}$ to $5x10^{-2}$	$5x10^{-6}$ $5x10^{-5}$	Alarm Only	Line 6CW21-056 via 6CW 1 1/2-123	Continuous

* information to be provided by february 1983

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Question No.

471.1 (12.3.4) As per Regulatory Guide 1.70 indicate whether, and if so how, the guidance provided by Regulatory Guide 1.97 has been followed concerning area radiation and airborne radioactivity monitoring instrumentation. Reference or provide this information.

Response

The guidance provided by Regulatory Guide 1.97 concerning area radiation and airborne radioactivity monitoring instrumentation will be addressed in the WNP-3 design.

The appropriate information will be supplied by February 1983.

f) To provide the capability to alarm, initiate isolation of the Control Room Normal Ventilation System, and actuate emergency ventilation.

g) To provide Control Room Operators with information regarding airborne radioactivity at the two Control Rooms outside air intakes.

h) To provide the capacity to alarm and initiate isolation of the containment purge and/or Fuel Handling Building.

i) To provide post-accident monitoring of conditions inside the containment.

12.3.4.2.2 Criteria for Location of Monitors

Considerations for locating the airborne radiation monitoring system monitors are based on the following:

a) Areas in which the airborne radioactivity can abruptly increase and in which personnel normally have access.

b) Inside the containment for the purpose of monitoring unidentified leaks.

c) In Control Room outside air intake ducts for post-accident habitability monitoring purposes.

d) Ventilation exhausts from areas in which spent fuel is handled.

e) The consolidated ventilation exhaust from sections of the Reactor Auxiliary Building.

f) The consolidated ventilation exhausted from the plant.

g) Monitors are located as close as possible to sample points to insure representative air samples.

h) In-plant airborne monitors have their sample points upstream of any HEPA filters.

12.3.4.2.3 Descriptions of Airborne Radiation Monitors

Airborne radiation monitors are used when the concentration of airborne radioactivity must be observed. A number of these monitors are formally part of the process and effluent radiation monitoring systems and are referenced below but are described elsewhere in this FSAR. The rest of these monitors are defined as in-plant airborne radiation monitors and are described in this section and are listed in Table 12.3.4-2. The airborne radiation monitors will be used to alert the operator and health physics personnel to the presence of an unusual level of airborne radioactivity. This will trigger health physics personnel to perform additional sampling and analysis to locate the source of the increased airborne radioactivity.

12.3.4.2.3.1 In-plant Airborne Radiation Monitors

12.3.4.2.3.1.1 Auxiliary Building Airborne Radiation Monitor

The auxiliary building airborne radiation monitors provides a measurement and a record of the airborne activity present in the Reactor Auxiliary Building. These monitors draw samples and monitor them for radioactive particulates, and radioactive gases and collects samples of the halogen that are in the exhaust air duct. The samples are available for later laboratory analysis.

Attachment 1-80
2268A

Question No.

471.3 (12.3.4)	Regulatory Guide 1.70 states that information on the auxiliary and/or emergency power supply should be provided. This information has not been provided for the general area radiation monitors. Provide or reference this information.

Response

The General Area Radiation Monitors are not safety-related and thus no emergency power supply is provided for them in the case of loss of offsite power. Protection of the memory of the microprocessors associated with the General Area Radiation Monitors is accomplished in the manner described in Subsection 11.5.2.1.

FSAR Subsection 12.3.4.1.3.1 will be changed to reflect the information on the auxiliary and/or emergency power supply to General Area Radiation Monitors.

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1 of 2

1647W-3

12.3.4.1.3.1 General Area Radiation Monitors

The area radiation monitors provide an indication to plant personnel of the radiation dose rate in selected areas of the plant both through the radiation monitoring system CRTs and local displays. The measurements made by these monitors are used primarily for personnel protection, with their use for the surveillance of equipment being incidental.

There are 70 G.M. radiation detectors serving as area radiation detectors, each with an associated set of local displays. The locations of these detectors and their displays are shown on Figures 12.3-1a through 12.3-28a and are listed in Table 12.3.4-1. The 70 radiation detectors are arranged in fours, threes or pairs, each group of which report to a single microprocessor. There are a total of 23 microprocessors for these detectors. Except for the multiple detectors and displays, the area monitors are physically similar to the ambient monitor described in Subsection 11.5.2.3.1.

Insert 1

The local displays associated with each radiation detector consists of an analog meter, a rotating beacon light and a howler (audible alarm). The analog meter is in a wall mounted box continuously displaying the measured radiation dose rate on a five decade scale. The rotating beacon is a wall mounted unit which has an amber color and is activated by the lower of the two radiation alarm set points, the radiation alert setpoint. The howler is a wall mounted unit which produces a loud distinctive sound and is activated by the upper of the two radiation alarm setpoints; the high radiation setpoint.

These displays are placed as follows, with respect to their associated radiation detector. The analog meter is placed along the access path to the area of the detector at a point before the detector has been reached. The rotating beacon is located in the vicinity of the radiation detector so that it is visible in the immediate area of the detector. The howler is located in the vicinity of the radiation detector so that its sound emanates from that area.

The measured dose rate for each of the radiation detectors is automatically transmitted to the system's computer where it is recorded and available for display through the system's CRTs in the Main Control Room and in the Health Physics office. The dose rate is also displayed locally through the analog meter. If the measured dose rate exceeds pre-established setpoints, an annunciation is made through the system CRT, event logger and the particular radiation detector's beacon and howler as appropriate.

The receipt of these alarms will alert the operator and plant personnel to the presence of an elevated radiation level so that surveys, investigations and other actions as appropriate may be taken. The high radiation and radiation alert alarm setpoints control the howler and flashing light displays respectively. The alarm setpoints are selected primarily for personnel protection and denote significant levels or changes in the measured close rate. The alert alarm setpoints are selected for each monitor on the basis of individual conditions. Typically the high alarm setpoint will be set at approximately 1R/hr.

The two area radiation monitors (RE-RM0028A,B) provide the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of 10CFR70, Section 70.24 for the storage area of the new fuel.

Insert 1

The General Area Radiation Monitors are not safety-related and thus no emergency power supply is provided in the case of loss of off-site power. Protection of the memory of the microprocessors associated with these monitors is accomplished as described in Subsection 11.5.2.1.

Attachment 1-81
2268A

Question No.

471.4 (12.4.3) The Annual Whole Body Dose Table is not complete. Furnish this information or provide a schedule for furnishing it.

Response

The missing information is given as follows:

ANNUAL WHOLE BODY DOSE (mrem/year)

	Maximum Point Restricted Area Boundary 1670 ft. SE	Maximum Point Exclusion Area Boundary 0.8 Miles SE
Contained Source	5.4	0.8
Submersion in cloud	3.2(-2)	2.6(-2)
Inhalation	7.2(-2)	7.2(-2)

Also, the information pertaining to construction of WNP-5 was deleted as no longer applicable.

FSAR Subsection 12.4.3 will be amended to reflect this information.

Q471.4
1/1

controls, described in Section 12.5, will be used to maintain occupational exposures to airborne radioactivity as low as is reasonably achievable. All individual exposures in excess of two MPC-hour/day or 10 MPC-hour/week will be assessed. If an individual's exposure exceeds 40 MPC-hours in any seven consecutive days, an evaluation will be made and necessary actions taken to assure against recurrence. Records will be kept of internal exposure assessments, evaluations and actions.

12.4.3 ~~NON~~ Annual OCCUPATIONAL EXPOSURE

Estimated annual whole body doses at the restricted area boundary, ~~WNP-5 construction site,~~ and exclusion area boundary are given below. These doses are based on occupancy of 50 weeks per year, 40 hours per week, and include doses from contained sources, submersion in the airborne cloud, and inhalation of airborne material.

ANNUAL WHOLE BODY DOSE (mrem/year)

	Maximum Point Restricted Area Boundary 1670 ft. SE	~~WNP-5 Construction Site 485 ft. W~~	Maximum Point Exclusion Area Boundary 0.8 Miles SE
Contained Source	5.4	~~49~~	0.8
Submersion In Cloud	~~(later)~~ 3.2(-2)	~~(later)~~	~~(later)~~ 2.6(-2)
Inhalation	~~(later)~~ 7.2(-2)	~~(later)~~	~~(later)~~ 7.2(-2)

Attachment 1-82
2268A

Question No.

471.5 (12.5.1) The SRP (NUREG-0800) and Regulatory Guide 1.70 state that the applicant should indicate whether, and if so how, the guidance of Regulatory Guides 8.2, 8.8, 8.10 and 1.8 has been followed and where applicable, describe the specific alternative approaches used. Provide or reference a discussion of your specific conformance or non-conformance to the guidelines in these Regulatory Guides.

Response

FSAR Subsection 12.5.1.1 states that Regulatory Guides 8.2, 8.8, 8.10 and 1.8 (as modified by the position statement in Section 17.2) were utilized as guidance in establishing the WNP-3 Health Physics Program.

Regulatory Guide 8.2, "Guide For Administrative Practices In Radiation Monitoring," provides rudimentary guidance in radiation monitoring practices for administrative and management personnel who may not be trained or experienced in radiation protection. The administrative and management personnel of the WNP-3 Health Physics Program are experienced radiation protection professionals. While the general guidance of Regulatory Guide 8.2 is followed, it contributes little to the organization of the WNP-3 Radiation Protection Program.

Regulatory Guides 8.8, "Information Relevant To Ensuring That Occupational Radiation Exposure At Nuclear Power Station Will Be As Low As Is Reasonably Achievable," provides guidance on organization, personnel and responsibilities to maintain occupational radiation exposures ALARA. WNP-3 FSAR Subsection 12.1.1.2 provides specific information on implementation of Regulatory Guide 8.8 with regard to organization, personnel and responsibilities.

Regulatory Guide 8.10, "Operating Philosophy For Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable," like Regulatory Guide 8.8, states that the qualifications of the Radiation Protection Manager (RPM) should be those specified by Regulatory Guide 1.8 and that the RPM should be given sufficient authority to enforce safe plant operation. FSAR Subsection 12.5.1.2 states that the Health Physics/Chemistry Manager is the Radiation Protection Manager of the plant. He reports directly to the Plant Manager who has the authority to control all plant activities (FSAR Subsection 13.1.2.2.1). Conformance to Regulatory Guide 1.8 is discussed below.

Attachment 1-83
2268A

Question No.

471.5 (12.5.1) (contd.)	Regulatory Guide 1.8 "Personnel Selection And Training," references ANSI N18.1-1971 for criteria for the selection and training of personnel, except for the Radiation Protection Manager (RPM). ANSI N18.1-1971 has been superseded, first by ANSI/ANS-3.1-1978 and more recently by ANSI/ANS-3.1-1981. The WNP-3 position on compliance with Regulatory Guide 1.8 with regard to selection and training of radiation protection personnel will be stated in FSAR Section 17.2.

Attachment 1-84
2268A

Question No.

471.6 (12.5.2) Regulatory Guide 1.70 and the SRP (NUREG-0800) state that the description of the health physics instrumentation should include the instruments sensitivity. You provided the type of radiation the instrument detects and not the instrument sensitivity in Table 12.5-1. Provide the requested information.

Response

Table 12.5-1 will be revised by June 1983 to add instrument sensitivity.

Attachment 1-85
2268A

Question No.

471.7 (12.5.2) Regulatory Guide 1.70 states that it should be indicated whether, and if so how, the guidance provided by Regulatory Guides 8.3, 8.4, 8.8, 8.9, 8.12, 8.14, 8.15, and 1.97 has been followed. If this guidance has not been followed, the specific alternative methods used should be described. Provide or reference a discussion of this information.

Reponse

Regulatory Guide 8.3, "Film Badge Performance Criteria," is not applicable to WNP-3 since WNP-3 will use thermoluminescent dosimeters (TLD) for personnel monitoring.

Conformance to Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters," will be discussed in Subsection 12.5.2.2.4 in a future amendment as indicated by the attached markup.

Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions For A Bioassay Program," is not directly applicable to the specification of health physics equipment, instrumentation or facilities. Regulatory Guide 8.9 is used for guidance in establishing the WNP-3 bioassay program as described in Subsection 12.5.3 as amended by FSAR Amendment #1.

Regulatory Guide 8.12, "Criticality Accident Alarm Systems," is not applicable to Subsection 12.5.2 and is not included in the SRP (NUREG-0800) review criteria.

Regulatory Guide 8.14, "Personnel Neutron Dosimeters," was used for guidance in evaluating neutron dosimeters for use at WNP-3. Subsection 12.5.2.2.4 will be revised to discuss conformance with Regulatory Guide 8.14 as indicated by the attached markup.

Regulatory guide 8.15, "Acceptable Programs For Respiratory Protection," has been superseded by incorporation of the requirements for respiratory protection programs into 10CFR20.103. WNP-3 complies with 10CFR20.103 in the selection of respiratory protective equipment as stated in Subsection 12.5.2.3.

Regulatory Guide 8.8, "Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Reasonably Achievable," was utilized as guidance in specifying WNP-3 health physics equipment, instrumentation, and facilities. Subsection 12.5.2 will be revised to more specifically address conformance with Regulatory Guide 8.8 as indicated by the attached markup.

Regulatory Guide 1.8 has been followed in the selection of personnel for the health physics organization and in the development of training programs for plant personnel. Supply System pre-employment practices include screening to determine that plant employees are responsible, conscientious and qualified to perform their duties safely.

12.5.1.3 Personnel Qualification and Training

The experience and qualifications of the supervisory personnel initially assigned to the Health Physics/Chemistry Department are outlined in Chapter 13. General employee training, specific formal training and on-the-job training programs are described in Section 13.2. The Health Physics/Chemistry Technicians are trained in health physics and chemistry to levels commensurate with the duties they are assigned. In addition to the above training, technician training may consist of industry sponsored schools, seminars and training at other Supply System nuclear plants. Other training programs are established based on the needs of the individual technician.

12.5.2 EQUIPMENT, INSTRUMENTATION AND FACILITIES

12.5.2.1 Health Physics Facilities

The WNP-3 health physics and support facilities conform to the requirements of Reg. Guide 8.8 (6/78).

The WNP-3 health physics facilities include a single health physics access control point for each unit with auxiliary control points that can be established during refueling and equipment outages. Main locker/change rooms are located in each Administration and Service Building (ASB) with separate facilities for both male and female employees. Auxiliary change rooms, personnel and equipment decontamination facilities, laboratories, dosimetry station, clothing storage area, lab instrument shops, a hot machine shop and health physics office/work area are located in the ASB. The majority of the health physics facilities are located on the EL. 390 ft. in the vicinity of the access control point. See Figures 1.2-21 and 1.2-22 for locations of health physics facilities and personnel traffic patterns through the main access control area.

12.5.2.1.1 Access Control and Auxiliary Control Points

Health physics access control is designed and equipped to maintain positive control over access to controlled areas (see Subsection 12.5.3.1.4 for the definition of controlled areas) and to prevent the spread of radioactive contamination to uncontrolled areas. Contamination control provisions include adequate area to separate the flow of personnel entering and exiting the controlled area. Personnel monitoring stations and facilities for storage of potentially contaminated small tools will be located in the access control area. Receptacles for protective clothing will be provided at exit points from boundaried areas within the controlled area.

Auxiliary control points can be established at other locations within controlled areas for the purpose of controlling contamination as close to the source as possible.

c) Portable radiation monitoring instrumentation.

d) Portable air sampling equipment.

e) Personnel friskers and miniscalers.

f) Working files and records.

g) Status log of radiation contaminated areas and areas with airborne radioactivity.

12.5.2.1.7 Laboratories

Laboratory facilities are located on the 405 ft. levels of the ASB and the RAB. The facilities include a shielded low background counting room, hot and cold laboratories for radiochemical and chemical analyses, hot and cold sample preparation rooms and counting room where samples will be analyzed qualitatively and/or quantitatively. The laboratories are equipped to perform routine analyses required for personnel protection, surveys and related health physics functions. The counting room (low background) is equipped with necessary instrumentation to perform routine counting on all plant radioactivity samples (water, air, swipe survey, etc.). Laboratory detection equipment is described in Subsection 12.5.2.2.

12.5.2.1.8 Hot Shops and Decontamination Facility

A separate complex is provided within the ASB for decontamination, repair, service and calibration of contaminated equipment, instrumentation and tools. The decontamination facility, hot ~~electrical and~~ electronic shop, hot machine shop, shielded calibration room and hot storage room are located on EL. 390 ft. (see Figure 1.2-21 for details). All facilities are within the controlled area.

12.5.2.1.9 Calibration Facilities

Facilities for calibration of all plant health physics instruments are provided by the plant staff, a Supply System central services organization or by a qualified contractor. Calibration equipment will include pulse generators, radiation sources, and electronic test equipment. Radiation sources used for calibrations will be traceable to National Bureau of Standards (NBS). Periodic checks and maintenance are performed on all test equipment at a prescribed interval to maintain reliability.

Instrument calibration records are maintained to insure that routine calibrations are performed at the specified frequencies. The calibration records are also used as a tracking mechanism on instrument maintenance to determine factors causative to reduced reliability. The instrument maintenance facilities include an instrument shop and a contaminated instrument shop.

The portable instrument calibration facilities are designed and located such that radiation in the calibration area will not interfere with low level monitoring or counting systems.

Storage of calibrated instruments is provided in the access control area with additional instruments stored at prescribed locations. Storage locations are selected to provide ready access for normal plant operations and to facilitate access to an adequate instrument supply during emergency or off normal situations.

12.5.2.1.10 Whole Body Counter

Whole body counting equipment will be maintained at a nearby facility. The unit will be housed in a low background area and operated by trained Health Physics/Chemistry personnel.

12.5.2.1.11 Storage

A radioactive material storage area will be maintained on the 390 ft. elevation of the RAB. This area is adjacent to the hot chemistry laboratory (see Figure 1.2-22 for details).

12.5.2.2 Health Physics Instrumentation

A summary of the quantities, types and ranges of WNP-3/5 health physics instrumentation is provided ~~by~~ in Table 12.5-1. Reg. Guides 8.8 (6/78) and 1.97 (R2) were used as guidance in selection of health physics instrumentation.

12.5.2.2.1 Laboratory Radiation Detection Instrumentation

The laboratory radiation detection instrumentation, located and used in the counting room, access control and laboratories, includes the following:

a) A Ge (Li) or HPGe gamma spectrometer system with conventional lead shielding and ~~a~~ multi-channel analyzer.

b) A liquid scintillation counter.

c) Two ~~gas flow~~ low background alpha-beta scintillation or proportional counters.

d) End window G-M type counting system.

e) A NaI scintillation spectrometer system.

Laboratory instrumentation is calibrated in accordance with manufacturer's recommendations using prepared liquid and sealed sources which are traceable to the National Bureau of Standards.

12.5.2.2.2 Portable Radiation Detection Instrumentation

The portable radiation detection instrumentation consists of the following:

a) Neutron dose equivalent rate meters.

b) Alpha ~~detectors.~~ scintillation or proportional countrate meters.

High range portable beta instrumentation is not commercially available. As the instrumentation becomes available, it will be evaluated against Reg Guide 1.97 (R2).

c) Low and high range gamma dose rate meters.

d) Remote probe G-M type survey meters (Friskers)

Portable radiation detection instruments are calibrated every six months, after repair, or any time a field check indicates an out of calibration condition, using appropriate calibration sources traceable to the National Bureau of Standards. Pulse type instruments may be calibrated with an electronic pulser and response to radiation verified with a radiation source. Instruments used to monitor radiography operations shall have been calibrated within the past three months. Mobile airborne monitors are discussed in Subsection 12.3.4.

12.5.2.2.3 Portable Air Sampling Equipment

Portable air sampling equipment includes high and low volume air samplers capable of accepting particulate filters and charcoal cartridges for grab samples of radioactive particulates and halogens. Air sampling instruments are calibrated periodically in accordance with an established schedule.

12.5.2.2.4 Personnel Monitoring Instruments

Reg. Guides 8.4(2/73), 8.8(6/78) & 8.14(8/77) were utilized as guidance in selecting personnel monitoring instruments.

Personnel dosimetry badges containing thermoluminescent dosimeters (TLD) provide the primary ~~legal record~~ measure of occupational radiation exposure incurred by personnel during normal and accident conditions. Each person entering plant controlled areas is assigned a TLD badge, which is ~~recorded~~ marked with the wearer's identification and exposure period. Results of the badge evaluation and the period of exposure are recorded on a document kept as an official record. Badges used will be capable of recording ~~exposure~~ dose equivalent over a range of at least 10 mrem to greater than 1,000 rem.

Neutron exposures are assigned by determining the dose equivalent rate versus the time spent in any areas where significant neutron exposure is present, or by use of state of the art TLD albedo neutron ~~badges~~ dosimeters. The total neutron dose equivalent for each exposed individual will be added to his permanent record at least quarterly. The performance of the TLD albedo neutron dosimeters exceeds the requirements of Reg. Guide 8.14.

In addition to the dosimeter badge, persons entering the restricted area may be required to wear other dosimetry assigned by the health physics staff such as direct reading pocket ion chambers, integrating dose meters, alarming electronic dosimeters, extremity badges or finger rings.

Pocket ion chamber dosimeters are tested for calibration/response and leak rate as required in Regulatory Guide 8.4, February 26, 1973. TLD's and TLD readers are calibrated prior to initial use and periodically thereafter using radiation sources traceable to the National Bureau of Standards.

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12.5.2.2.5 Emergency Radiation Detection Instrumentation

Reg. Guide 1.97 was used as guidance in selecting emergency radiation detection instrumentation.

Portable instruments are available for immediate access and consists of the following:

a) Wide-range ionization chamber beta-gamma survey instruments or instruments capable of measuring radiation fields in the nominal range of 1 mR/hr to 1,000 R/hr.

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'A'

G-M 'Friskers' with audible output for detecting low levels of radioactive materials will be available at the primary and auxiliary access control points for personal contamination monitoring. High sensitivity portal monitors located at the primary access control points will provide an additional check for radioactive contamination on personnel exiting the controlled areas of the plant.

b) Portable G-M type survey instruments for use in monitoring surface contamination and counting emergency air samples.

c) Portable air samplers.

d) Direct reading dosimeters with nominal ranges of 0-500 mR, 0-5R, and 0-100R.

e) Extendable probe high range gamma dose rate instruments with nominal range of 1mr/hr to 10^4 R/hr.

Instruments available for emergency use are checked regularly for proper operation and exchanged for calibration every six months.

12.5.2.2.6 Installed Radiation Detection Instrumentation

Installed Area Radiation Monitors (ARM) and Airborne Activity Monitors (AAM) are described in Subsection 12.3.4, Effluent Liquid Radioactivity Monitors (ELM), Effluent Gas Monitors (EGM) and Process Radiation Monitors (PRM) are described in Section 11.5.

12.5.2.3 Protective Clothing and Equipment

Personnel protective clothing and equipment are stored in the clean clothing storage area and in the vicinity of the access control and auxiliary control points. The types of equipment and clothing available include the following:

a) Anti-contamination clothing - coveralls, hoods, boots, gloves, glove liners, and lab coats.

b) Plastic gloves and plastic shoe covers.

c) Plastic waterproof suits and hoods.

d) Continuous airflow two-piece plastic suits.

Respiratory protective equipment is provided and required for personnel when levels of airborne radioactive materials approach or exceed applicable limits or when a potential for this condition exists. Respiratory protection procedures are described in Subsection 12.5.3.3.4. The respiratory protection program is conducted within the requirements of 10CFR20.103 and exposure is limited to average concentrations less than the values specified in 10CFR20, Appendix B, Table 1. Allowance is made for use of respiratory protective equipment, as prescribed in 10CFR20.103, in determining an individual's inhalation of airborne radioactive materials. The following types of equipment are used:

a) Full facepiece air purifying respirators with high efficiency filters.

b) Full facepiece pressure demand air line respirators.

c) Full facepiece pressure demand self-contained breathing apparatus.

Attachment 1-86
2268A

Question No.

471.7 (12.5.2) (contd.)	Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following An Accident," was used for guidance in specifying emergency radiation detection instrumentation. The portable emergency radiation detection instrumentation conforms to the requirements of Regulatory Guide 1.97 except that high range portable beta does rate measuring instruments are not commercially available. Subsection 12.5.2 will be revised to more specifically address conformance with Regulatory Guide 1.97 as indicated by the attached markup.

Attachment 1-87
2268A

Question No.

471.8 (12.5.3) The SRP (NUREG-0800) requires information describing the implementation of Regulatory Guides 1.8, 8.2, 8.7, 8.8, 8.9, 8.10, 8.26, 8.27, and 8.29. This information is not completely discussed. Provide this material, including a specific discussion of the implementation of Regulatory Guides 1.8, 8.9, 8.26, 8.27, and 8.29.

Response

Regulatory Guide 1.8, "Personnel Selection and Training" refers to ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel" which has recently been superseded by ANSI 3.1-1981, "Standard for Selection, Qualification and Training of Personnel for Nuclear Power Plants". The WNP-3 position on compliance with Regulatory Guide 1.8 will be stated in FSAR Section 17.2.

Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring" refers to ANSI N13.2-1969, "Guide for Administrative Practices in Radiation Monitoring". Subsections 12.5.3.1.2 and 12.5.3.3.1 both state that Regulatory Guide 8.2 was used in developing radiation protection procedures. In addition, Subsection 12.5.3.8.1 has been revised with FSAR amendment No. 1 to state that Regulatory Guide 8.2 was used as guidance, in part, for developing Radiation Exposure Records System.

Regulatory Guide 8.7, "Occupational Radiation Exposure Records Systems" refers to ANSI N13.6-1966 (R1972), "American National Standard Practice for Occupational Radiation Exposure Records Systems" for guidance. Subsection 12.5.3.8 provides specific information and states that Regulatory Guide 8.7 was used to develop the Radiation Exposure Records System.

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable" was used as guidance in developing the Radiation Work Permit as discussed in Subsection 12.5.3.1.2. In addition, operation, maintenance, repair, surveillance and refueling procedures are reviewed, as applicable, to ensure occupational exposures are ALARA. Subsection 12.5.3 will be amended to reflect this as indicated by the attached.

Regulatory Guide 8.9, "Acceptable Concept, Models, Equations, and Assumptions for a Bioassay Program" refers to several ICRP publications for guidance on developing an acceptable bioassay program. Subsection 12.5.3.4.2 has been revised by FSAR amendment No. 1 to include a discussion as to how Regulatory Guide 8.9 has been implemented. Both in vivo and in vitro bioassay programs have been developed using the methodologies recommended by the ICRP.

12.5.3 PROCEDURES

12.5.3.1 Exposure Control

12.5.3.1.1 Maintaining Exposures ALARA

The occupational radiation exposure of personnel at WNP-3/5 is maintained ALARA by a combination of monitoring, protection, access control, training and planning. Monitoring consists of personnel exposure monitoring to maintain control over total radiation doses received and radiation, contamination, and airborne radioactivity surveys to identify and control radiological hazards. Protection involves the use of temporary shielding, where necessary, to reduce radiation levels and the use of anti-contamination clothing and respiratory protective equipment to protect personnel from actual and possible sources of contamination and airborne radioactivity. Access control is employed to minimize exposure of personnel by limiting access to controlled areas commensurate with both the need for entry and the degree of radiological hazard involved. Training is used to keep personnel aware of radiological hazards and on the alert for methods of reducing their exposure. Operating, maintenance, surveillance and refueling procedures will be reviewed as applicable, to insure that prescribed actions conform with ALARA concepts, in conformance with Regulatory Guides 8.8, dated June 1978, and 8.10 dated May, 1977. Planning is utilized when high exposures are expected. It consists of job prebriefing, preparation and review of written procedures for work in controlled areas, and "cold" practice of work to be performed. Subsection 12.1.3 describes a process that was incorporated into the preparation and revision of all plant procedures which provides a positive method of assuring health physics input and ALARA consideration into all radiation exposure related activities.

12.5.3.1.2 Personnel Control Procedures

The WNP-3/5 Plant Administrative Procedures and Health Physics Procedures contain the administrative control procedures for entry into radiation and high radiation areas. The procedures limit the entry and time spent in radiation areas to the time necessary to perform routine operations, maintenance and surveillance activities. The Radiation Work Permit (RWP) is used as the primary tool to insure the control at WNP-3/5. Health Physics/Chemistry personnel or personnel who are provided direct coverage by Health Physics/Chemistry Technicians may be exempt from the issuance of a RWP on a case to case basis as approved by plant management. Provision will be included on an individual basis for exposure tracking by job category and function.

The radiation work permit is issued for a particular task or function, and is required before entering a radiation area. This permit provides current data on radiation levels within the area of interest, any restrictions on allowable work time, protective clothing and respiratory protective requirements, information on special tools or equipment needed, special radiation safety and personnel monitoring requirements and any other special instructions

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12.5.3.8.3 Radioactive Materials

Records of radiation and contamination surveys upon receipt of shipments of radioactive materials are retained for the time specified in 10CFR20.401. Records on inventory and sealed source leak tests of the by-product materials not produced by operation of WNP-3/5 are retained until the activities of the by-product materials decay to less than the exempt quantities. Records of disposal of the by-product materials not produced by operation of WNP-3/5 are retained for the time specified in 10CFR20.401.

12.5.3.8.4 Radioactive Effluent Releases and Solid Radioactive Material Shipments

Records of batch releases of radioactive effluent and shipments of radioactive material for disposal are retained for the time specified in 10CFR20.401. Reports to the NRC pursuant to the radioactive effluent releases and the shipment of radioactive material shall be made in accordance with the requirements in Regulatory Guide 1.21, Revision 1, June 1974.

12.5.3.9 Health Physics Program Review and Audits

The WNP-3/5 Health Physics Procedures are reviewed annually by members of the central and/or plant health physics staff to determine areas where improvements may be desirable. The Radiological Assessment and Audit Section annually audits portions of each plant's radiation protection program. The areas to be audited are selected so that the entire radiation protection program of each plant will be audited over a period of five years.

Health Physics Procedures will be reviewed as required by the Technical Specification.

in accordance with Regulatory Guide 8.10 dated May, 1977.

Evaluations to determine where significant occupational radiation exposures are occuring are conducted continuously as a part of the operational ALARA program. These evaluations are conducted jointly by the Health Physics/Chemistry Department and the Radiological Assessment and Audit Section. The Radiation Exposure Records (RER) computer programs permit analysis of radiation exposures by craft, job, component or system. The ALARA program uses the methodology of NUREG/CR-0446 to identify and evaluate potential areas of exposure reduction.

Attachment 1-88
2268A

Question No.

471.8 (12.5.3) (contd.)

Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable" gives guidance with respect to modifying procedures and giving training to ensure exposures are ALARA. Procedures are reviewed, as applicable, to ensure occupational exposures are ALARA. Subsection 12.5.3 will be amended to reflect this as indicated by the attached. Training is discussed in Section 13.2.

Regulatory Guide 8.26, "Applications of Bioassay for Fission and Activation Products" refers to ANSI N343 which was used in developing the bioassay program for WNP-3/5. Subsection 12.5.3.4.2 has been revised with FSAR amendment No. 1 to include a discussion of how Regulatory Guidc 8.26 is being implemented.

Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants" was used in the development of training and retraining programs consistent with the ALARA objective to meet the requirements of 10CFR19, except that persons permanently assigned to nuclear facilities will be provided refresher advanced radiological training associated with their specialty every two years. A detailed discussion with respect to the implementation of Regulatory Guide 8.27 can be found in Subsection 12.5.3.5 as amended by FSAR amendment No. 1.

Regulatory Guide 8.29, "Instructions Concerning Risks From Occupational Radiation Exposure" has been incorporated into the general employee radiation protection training. Discussion concerning conformance with Regulatory Guide 8.29 can be found in Subsection 12.5.3.5 as amended by FSAR amendment No. 1.

Attachment 1-89
2268A

Question No.

480.1 (6.2.1.1.2) Identify the locations in the containment where water may be trapped and prevented from returning to the containment sump. The quantity of water involved should be specified. Discuss how the static head for recirculation pumps may be affected.

Response

The only dead volume in the Containment where water may be trapped and prevented from returning to the containment SIS recirculation sump, is the reactor cavity and incore guide tube chase (up to EL. 375 ft). The total volume of water that may be trapped is approximately 96,700 gallons. Effects of dead volumes have been considered in the containment flood level elevation analysis. The available NPSH of the containment spray pumps is based on the static head generated from the containment flood level. Discussion of the NPSH calculation is presented in Subsection 6.2.2.

Attachment 1-90
2268A

Question No.

480.2 (6.2.1.1.3) Reference or provide a discussion of the administrative controls and/or electrical interlocks that would prevent the inadvertent operation of the containment heat removal system or other systems that could result in pressures lower than the external design pressure of the containment structure. Identify the worst single failure that could result in the inadvertent operation of the containment heat removal system.

Response

A discussion of the administrative controls and/or electrical interlocks that would prevent the inadvertent operation of the containment heat removal system or other systems that could result in pressures lower than the external design pressure of the containment structure will be supplied by December 31, 1982.

Attachment 1-91
2268A

Question No.

480.3 (6.2.1.5) Provide the results of the confirmatory review of the containment pressure analysis for emergency core cooling system capability studies.

Response

Final design calculations for the WNP-3 containment backpressure analysis indicate that WNP-3 will not conform to the pressure curve identified on Figure 6.2.1-24 of CESSAR-F because containment heat sinks were found to be considerably greater than assumed at the preliminary design stage. Therefore, the large break LOCA analyses results in Subsection 6.3.3.2 of CESSAR-F will be replaced with WNP-3 specific results in the WNP-3 FSAR. The results will be provided by June 1983.

Attachment 1-92
2268A

Question No.

480.4 (6.2.2.3)

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

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

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

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

3. Evaluate the extent to which the containment sumps satisfy each of the positions of Regulatory Guide 1.82. The following additional guidance is provided for this evaluation:

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

   b. Estimate the extent to which debris could block the trash rack or screens. If a blockage problem is identified, describe the corrective actions you plan to take.

   c. For each type of thermal insulation used in the containment, provide the following information:

      (i) type of material including composition and density,

      (ii) manufacturer and brand name,

      (iii) method of attachment,

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Question No.

480.4
(6.2.2.3)
(contd.)

(iv) location and quantity in containment of each type,

(v) an estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

d. Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

Response

The staff concerns regarding containment sump designs and their effect on long term cooling following a Loss of Coolant Accident (LOCA) will be addressed as part of Enclosure 4, item (10), "Effects of Containment Coatings and Sump Debris on ECCS and Containment Spray Operation."

Attachment 1-94
2268A

Question No.

480.5 (6.2.4)	Provide an evaluation of your conformance to Branch Technical Position CSB 6-4. Identify and justify any deviations.

Response

The Supply System is reviewing the evaluation of conformance to BTP-CSB 6-4 at this time. This process will be complete by November 1982.

Attachment 1-95
2268A

Question No.

480.6 Provide instrument lines containment penetration information in Table 6.2.4-1 and Figure 6.2-36 which is labeled as "later".

Response

The following is a new writeup for the present FSAR Subsection 6.2.3.1.2 "Design Criteria for Instrument Lines".

Instrument lines penetrating the primary reactor containment and that are connected directly to the containment atmosphere meet the requirements of General Design Criterion (GDC) 56. There are no instrument lines which penetrate the primary reactor containment and form part of the reactor coolant pressure boundary.

Compliance with the requirements of GDC 56 is achieved through conformance with Regulatory Position C.1.a through e and C.2.a of Regulatory Guide 1.11.

Those instrument lines penetrating the primary reactor containment and connected to the Containment Vacuum Relief System, meet the requirements for redundancy, independence and testability in that Train A instruments are connected to penetration 84 as shown on Figures 7.3-8 and 7.3-9.

All instrument lines penetrating the primary reactor containment (Penetrations 84, 86, 87) are provided with a self-actuated excess flow check valve, located on the instrument line outside of the containment as close to the penetration as possible.

The excess flow check valves are self-actuating and will close on excess flow due to loss of integrity of the instrument line outside of containment. The check valves are manual reset type and are provided with position indication for open/close and an alarm for closed position, in the control room.

The valves are designed in accordance with ASME Section III Class II requirements and Quality Class I. The response time of instruments connected to those penetrations has been determined to be acceptable with the excess flow check valves installed.

Table 6.2.4-1 and Figures 6.2-36 through 6.2-36r provide details of the instrument line penetrations.

The instrument line penetration information shown in Table 6.2.4-1 as later in Attachment 1 will be provided by August 1983.

The instrument line penetration information shown on Figures 6.2-36 through 6.2-36r is given in Attachment 2.

Attachment 1

Penetration No.	Service	CDC Isolation Criteria	System Category	Valve No.	Electrical Division	Electrical Power	Valve Size (Inches)	Type Of Valve Operator	Valve Type	Location with Respect to RB	Normal Valve Position	Shutdown Valve Position	Post-Accident Valve Position	Failure Mode	Length from Penetration (Feet)	Valve Closure Signal	Valve Open Signal	Closure Time (Secs.)	Override	Fluid	
67	SB Vent 'A'/Cont. Vacuum Relief A Instrument	SB	ESF	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	
67	Instrument Line Spare	SB	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-		
77	Cont. Purge/Annulus Vacuum Maint Instr.	SB	NDN	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	
77	Instrument Line Spare	SB	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	
84	Containment Vacuum Relief B Instrument	SB RC 1.11	ESFREL	ZHV-EFD03	SN	-	½	-	EF	O	O	O	O	-	* ~~Later~~	-	Manual Reset	-	No	A	Y
84	Instrument Line Spare	-																			
85	Instrument Line Spare	-																			
85	Instrument Line Spare	-																			
86	Containment Vacuum Relief A Instrument	SB RC 1.11	ESFREL	ZHV-EFD02	SN	-	½	-	EF	O	C	O	O	-	* ~~Later~~	-	Manual Reset	-	No	A	Y
86	Instrument Line Spare	-																			
87	Containment Purge Instrument Line	SB RC 1.11	NDN	ZHV-EFD01	SN	-	½	-	EF	O	C	O	O	-	* ~~Later~~	-	Manual Reset	-	No	A	Y
87	Instrument Line Spare	-																			
88	Containment Vacuum Relief 'B'	SB	ESFREL	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	NA	
88	Instrument Line Spare	-																			
89	Annulus Vacuum Maintenance Instr.	SB	NDN	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	NA	
9	Instrument Line Spare	-																			

*Information will be supplied by August 1983.

Attachment 2

VALVE TYPES

GLOBE

GATE

CHECK

RELIEF

BUTTERFLY

STOP CHECK

EXCESS FLOW CHECK VALVE

OPERATORS

AO AIR

MO MOTOR

S SOLENOID

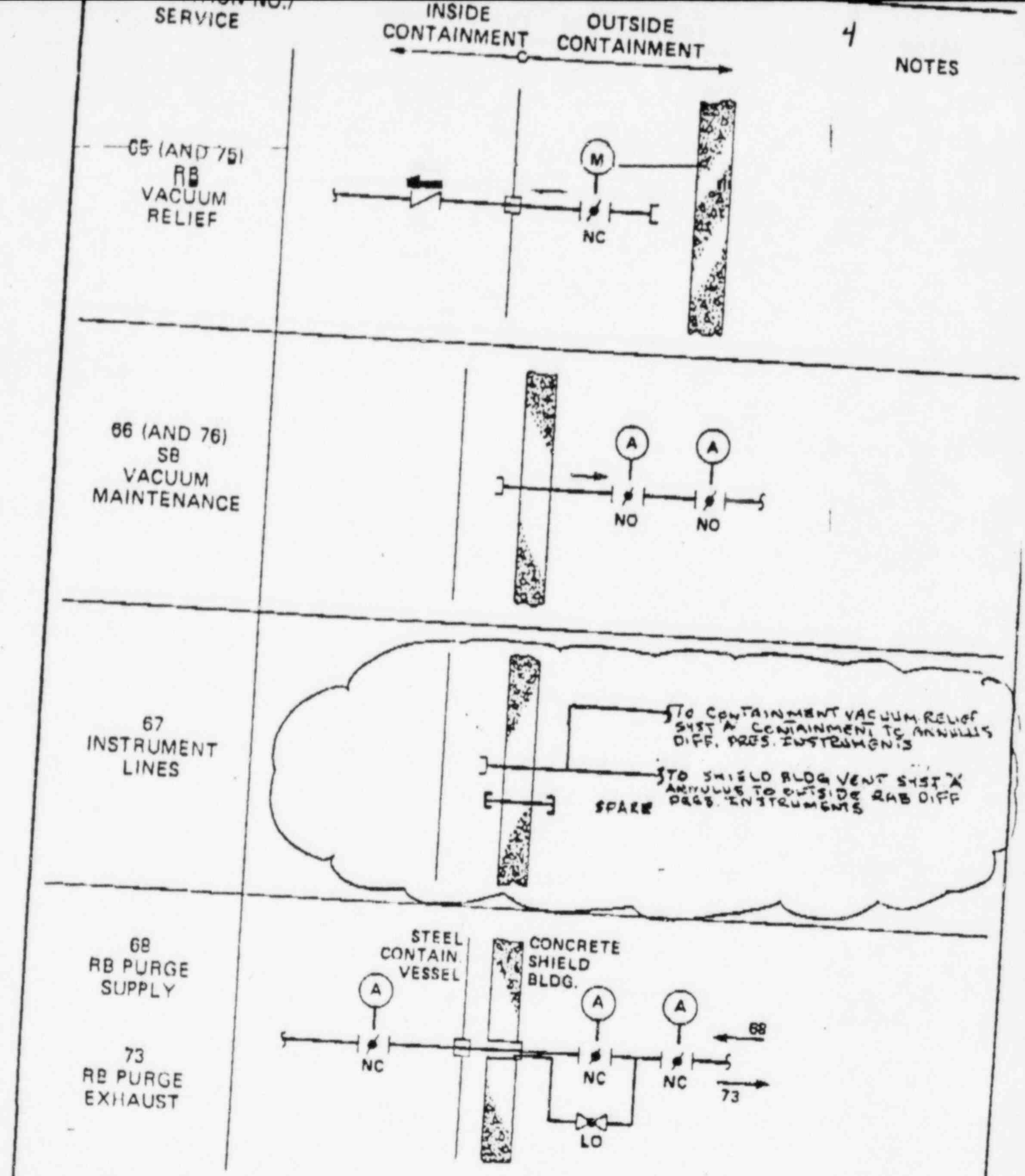
E ELECTRO-HYDRAULIC

POSITION

NORMALLY CLOSED

NORMALLY OPEN

WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES ARRANGEMENT	FIGURE 6.2-36

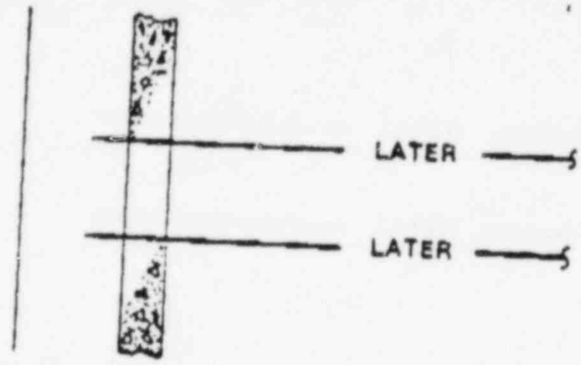
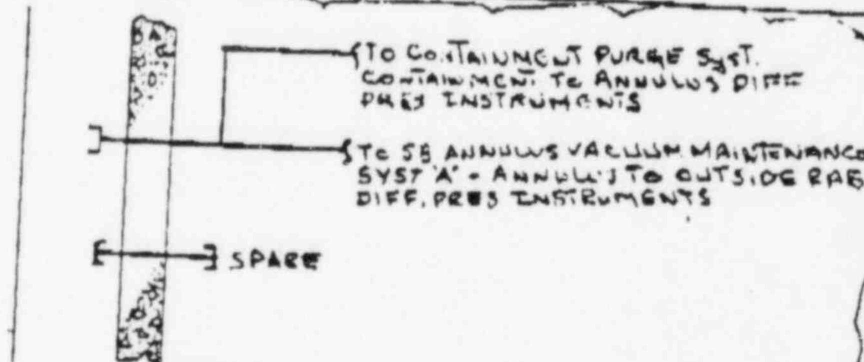
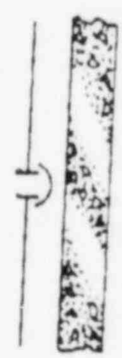


WASHINGTON PUBLIC POWER SUPPLY SYSTEM

Nuclear Projects 3 & 5
FINAL SAFETY ANALYSIS REPORT

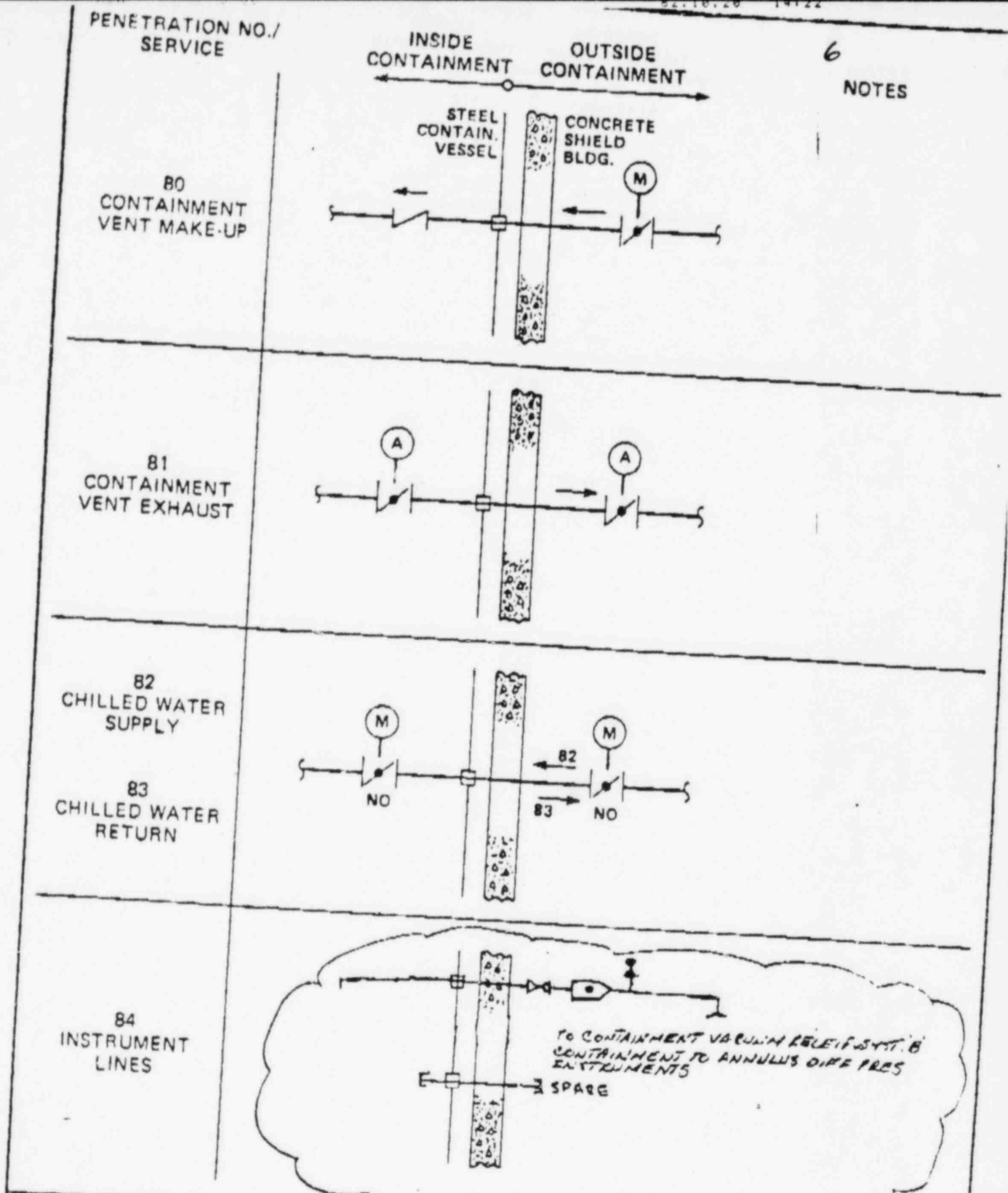
CONTAINMENT ISOLATION VALVES ARRANGEMENT

FIGURE 6.2-36k

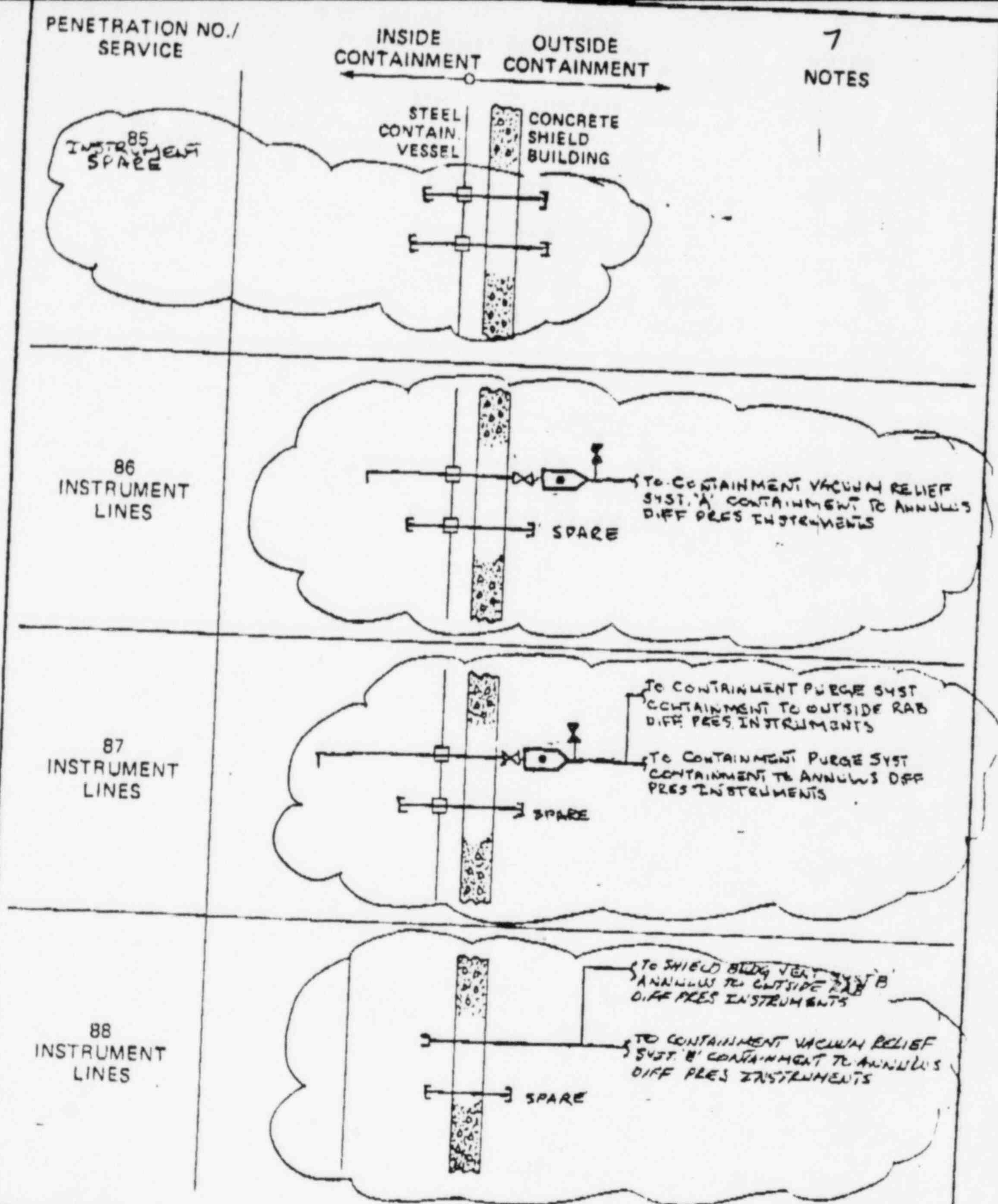
PENETRATION NO./ SERVICE	INSIDE CONTAINMENT ← ○ → OUTSIDE CONTAINMENT	NOTES
69 INSTRUMENT AIR SUPPLY TO CONTAINMENT VACUUM RELIEF SYSTEM		
70	SEE PENETRATION 60	
71	SEE PENETRATION 61	
72	SEE PENETRATION 62	
73	SEE PENETRATION 68	
75	SEE PENETRATION 65	
76	SEE PENETRATION 66	
77 INSTRUMENT LINES		
78 SPARE		DOES NOT HAVE A SB PENETRATION.

5

WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES ARRANGEMENT	FIGURE 6.2-361



WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES ARRANGEMENT	FIGURE 6.2-36m

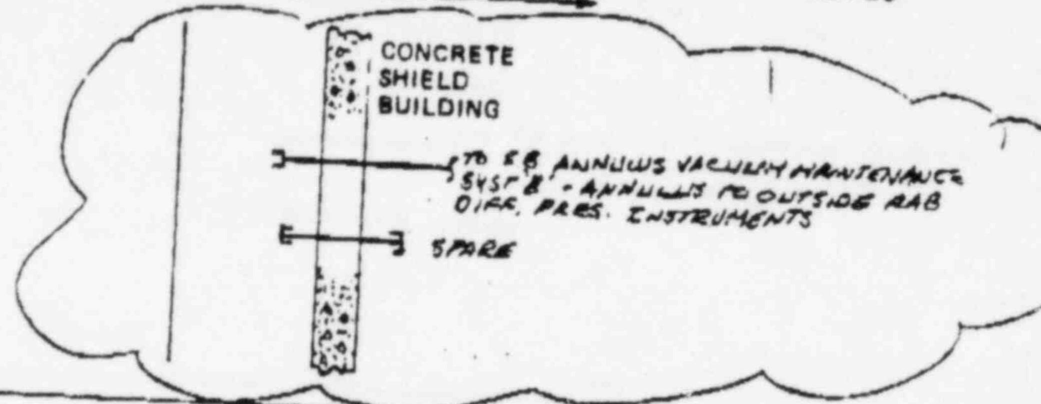


WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES ARRANGEMENT	FIGURE 6.2-36n

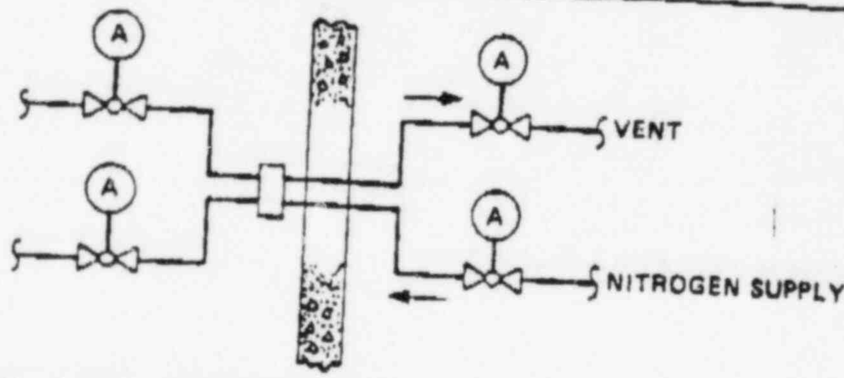
PENETRATION NO./ SERVICE | INSIDE CONTAINMENT ← → OUTSIDE CONTAINMENT | NOTES

8

89 INSTRUMENT LINES



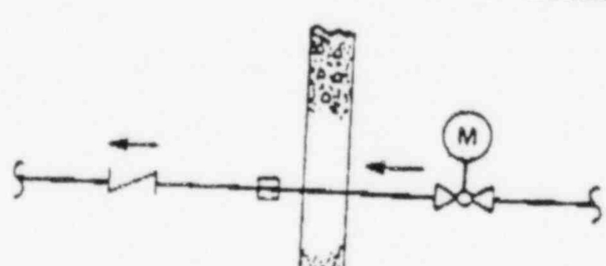
90 REACTOR DRAIN TANK VENT, AND REACTOR DRAIN TANK NITROGEN SUPPLY



91 — SEE PENETRATION 48

92 — SEE PENETRATION 25

93 SEAL INJECTION HEADER



WASHINGTON PUBLIC POWER SUPPLY SYSTEM Nuclear Projects 3 & 5 FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES ARRANGEMENT	FIGURE 6.2-36o

9 See Insert 1

6.2.4.1.2 Design Criteria for Instrument Lines

Instrument lines penetrating the containment will be designed to R.G. 1.11 criteria. Table 6.2.4-1 and Figures 6.2-36 through 6.2-36r provide details of the instrument line penetrations.

6.2.4.1.3 Design Requirements for Isolation Barriers

All isolation barriers are designed to perform their function taking into account the accident transient, servere natural phenomena, and loss of normal power sources. Specifically, all isolation barriers are designed to seismic Category I, Safety Class 2, LOCA transient and environment, and containment design temperature. All penetrations and isolation valves are designed, as a minimum, to the containment structural acceptance test pressure. All ESF systems are designed, as a minimum to the containment design pressure. All closed systems inside containment are designed, as a minimum, for an external pressure equal to the containment structural acceptance test pressure and have thermal relief protection through the steam generator relief valves.

Detailed information on the design requirements is given or referenced in Subsection 6.2.4.2.3.

6.2.4.1.6 Containment Isolation Dependability Design Bases

The Containment Isolation System meets the requirements of NUREG-0737 Task Action Plan II.E.4.2. The design criteria are as follows:

a) The CIS complies with the requirements of Standard Review Plan 6.2.4, specifically that there be diversity in the parameters sensed for the initiation of containment isolation (refer to Subsection 7.3.1.1.2.3).

b) Control systems for the automatic containment isolation actuation signal do not automatically reopen the containment isolation valves upon reset of the CIAS but require deliberate operator action.

c) The containment setpoint pressure for nonessential systems lines is reduced to the minimum compatible with normal operating conditions.

d) Containment vent and purge isolation systems are isolated upon a receipt of the high radiation signal.

A detailed discussion of the specific design requirements is given or referenced in Subsection 6.2.4.2.

6.2.4.2 System Design

Personnel air locks and the design of containment penetrations are discussed in Subsection 3.8.2. Material requirements for the CIS are discussed in Section 6.1. Guard pipes are discussed in Subsection 3.6.2.4.

Table 6.2.4-1 tabulates all the specific design information for the Containment Isolation System (CIS) on a penetration-by-penetration basis. The discussion below provides additional, general information and addresses any exceptions to the system's design bases. Subsection 6.2.4.2.3 elaborates how the design requirements for isolation barriers are met.

Insert 1

Instrument lines penetrating the primary reactor containment and that are connected directly to the containment atmosphere meet the requirements of General Design Criterion (GDC) 56. There are no instrument lines which penetrate the primary reactor containment and form part of the reactor coolant pressure boundary.

Compliance with the requirements of GDC 56 is achieved through conformance with Regulatory Position C.1.a through e and C.2.a of Regulatory Guide 1.11.

Those instrument lines penetrating the primary reactor containment and connected to the Containment Vacuum Relief System, meet the requirements for redundancy, independence and testability in that Train A instruments are connected to penetration 84 as shown in Figures 7.3-8 and 7.3-9.

All instrument lines penetrating the primary reactor containment (Penetrations 84, 86, 87) are provided with a self-actuated excess flow check valve, located on the instrument line outside of the containment as close to the penetration as possible.

The excess flow check valves are self-actuating and will close on excess flow due to loss of integrity of the instrument line outside of containment. The check valves are manual reset type and are provided with position indication for open/close and an alarm for closed position, in the control room.

The valves are designed in accordance with ASME Section III Class II requirements and Quality Class I. The response time of instruments connected to those penetrations has been determined to be acceptable with the excess flow check valves installed.

Table 6.2.4-1 and Figures 6.2-36 through 6.2-36r provide details of the instrument line penetrations.

TABLE 6.2.4-1 (Cont'd)

Penetration No.	Service	GDC Isolation Criteria	System Category	Valve No.	Electrical Division	Electrical Power	Valve Size (Inches)	Type of Valve Operator	Valve Type	Location with Respect to CB	Normal Valve Position	Shutdown Valve Position	Post-Accident Valve Position	Failure Mode	Length from Penetration (Feet)	Valve Closure Signal	Valve Open Signal	Closure Time (Seconds)	Override	Field
67	[illegible]	55	P&C	—	—	—	—	—	—	—	LATER	—	—	—	[illegible]	—	—	—	—	A
67	Instrument line [illegible]	55	—	—	—	—	—	—	—	—	LATER	—	—	—	—	—	—	—	—	[illegible]
68	SB Purge Supply	56	NNS	3PV-0018	SB	-	48	AO		O	C	O	C	FC	5	CIAS, HR	-	5	No	A
		C		3PV-0017	SB	-	48	AO	Bfly	O	C	O	C	FC	5	CIAS, HR	-	5	No	
				3PV-0016	SA	-	48	AO	Bfly	I	C	O	C	FC	-	CIAS, HR	-	5	No	
69	Instrument Air to Shield Bldg	56		LATER																
70	Hydrogen Purge Exhaust	56	ESFNSL	3PV-0110	SB	460V ac	6	MO	Bfly	O	C	C	C	FAI	3	CIAS	-	10	No	A
		C		3PV-0109	SA	460V ac	6	MO	Bfly	I	C	C	C	FAI	-	CIAS	-	10	No	
71	Shield Bldg Vent Supply	56	ESF	3PV-0125	SB	460V ac	30	MO	Bfly	O	C	C	O	FAI	2	-	-	10	No	A
72	Shield Bldg Vent Exhaust	56	ESF	3PV-0101	SB	460V ac	30	MO	Bfly	O	C	C	O	FAI	3	-	HF	10	No	A
73	SB Purge Exhaust	56	NNS	3PV-0132	SB	-	48	AO	Bfly	O	C	O	C	FC	5	CIAS, HR	-	5	No	A
		C		3PV-0132	SA	-	48	AO	Bfly	I	C	O	C	FC	-	CIAS, HR	-	5	No	
				3PV-0113	SB	-	48	AO	Bfly	O	C	O	C	FC	9	CIAS, HR	-	5	No	
75	SB Vacuum Relief	56	NNS	3PV-0114	SB	460V ac	42	MO	Bfly	O	C	C	C	FAI	3	-	HF	10	No	A
		C		3PV-0115	SB	-	42	-	Check	I	C	C	C	-	-	-	-	-	-	

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TABLE 6.2.4-1 (Cont'd)

Penetration No.	Service	GDC Isolation Criteria	System Category	Valve No.	Electrical Division	Electrical Power	Valve Size (Inches)	Type of Valve Operator	Valve Type	Location with Respect to CB	Normal Valve Position	Shutdown Valve Position	Post-Accident Valve Position	Failure Mode	Length from Penetration (Feet)	Valve Closure Signal	Valve Open Signal	Closure Time (Seconds)	Override	Field	Type C Test
	[illegible]	[illegible]	[illegible]	1PV-0123	[illegible]	–	10	AO	Btrfly	O	O	O	C	FC	1	CIAS	–	5	No	A	NA
				1PV-0124	SA	–	10	AO	Btrfly	O	O	O	C	FC	6	CIAS	–	5	No		
	[illegible]	[illegible]	[illegible]	—	—	—	—	—	—	LATER	—	—	—	—							
	Instrument Line / SPARE	[illegible]	—	—	—	—	—	—	—	LATER	—	—	—	—	—	—	—	—	—	A	N/A
	Spare Penetration	-	-																	—	N/A
	Containment Vent Makeup	56 C	[illegible]	1PV-0019	SA	460V ac	8	MO	Btrfly	O	O/C	C	C	FAI	3	CIAS, HR	–	10	No	A	Yes
				1PV-0031	SA	–	8	-	Check	I	O/C	C	C	–	–	–	–	–	–		
	Containment Exhaust	56 C	[illegible]	2PV-0104	SB	–	8	AO	Btrfly	O	O/C	C	C	FC	3	CIAS, HR	–	3	No	A	Yes
				0PV-0064	SA	–	8	AO	Btrfly	I	O/C	C	C	FC	–	CIAS, HR	–	3	No		
	Chilled Water Supply	56 C	[illegible]	2BC-0009	SB	460V ac	8	MO	Btrfly	O	O	O	C	FAI	2	CIAS	–	10	No	[illegible]	Yes
				2BC-0010	SA	460V ac	8	MO	Btrfly	I	O	O	C	FAI	–	CIAS	–	10	No		
	Chilled Water Return	56 C	[illegible]	2BC-0011	SB	460V ac	8	MO	Btrfly	O	O	O	C	FAI	2	CIAS	–	10	No	[illegible]	Yes
				2BC-0012	SA	460V ac	8	MO	Btrfly	I	O	O	C	FAI	–	CIAS	–	10	No		

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TABLE 6.2.4-1 (Cont'd)

Penetration No.	Service	GDC Isolation Criteria	System Category	Valve No.	Electrical Division	Electrical Power	Valve Size (Inches)	Type of Valve Operator	Valve Type	Location with Respect to [illegible]	Normal Valve Position	Shutdown Valve Position	Post-Accident Valve Position	Failure Mode	Length from Penetration (Feet)	Valve Closure Signal	Valve Open Signal	Closure Time (Seconds)	Override	Fluid	Type Test
[illegible]	Containment [illegible] Channel A Isolation	EF [illegible] 1.11	[illegible]	[illegible]	SN	—	1/2	—	EF	O	O	O	O	—	LATER	—	MANUAL RESET	—	[illegible]	A	Yes
[illegible]	Instrument Line Non-ESF SPARE	[illegible]																			
[illegible]	Instrument Line Channel B Spare	[illegible]																			
[illegible]	Instrument Line Non-ESF Spare	[illegible]																			
[illegible]	Containment [illegible] Channel B Isolation	EF 1.11	[illegible]	[illegible]	SN	—	1/2	—	EF	O	O	O	O	—	LATER	—	MANUAL RESET	—	No	A	Yes
[illegible]	Instrument Line Non-ESF Spare	[illegible]																			
[illegible]	Containment [illegible] Channel [illegible]	EF 1.11	NON	[illegible]	SN	—	1/2	—	EF	O	O	●	O	—	LATER	—	MANUAL RESET	—	No	A	Yes
[illegible]	Instrument Line Non-ESF Spare	[illegible]																			
[illegible]	[illegible] Non-ESF [illegible] B	[illegible]	[illegible]	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	N/A
[illegible]	Instrument Line Non-ESF SPARE	[illegible]																			
[illegible]	[illegible]	[illegible]	NON	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	N/A
[illegible]	Instrument Line Non-ESF SPARE	[illegible]																			

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Attachment 1-96
2268A

Question No.

490.1 Chapter 4 states: "The initial fuel cycle for WNP-3 is not consistent with the extended fuel cycle described in Chapter 4 of CESSAR-F through Amendment 6. Although the fuel design parameters in CESSAR-F may envelope the WNP-3 specific fuel design this has not been confirmed. WNP-3 specific evaluations are currently in process and if it is determined that Chapter 4 or portions thereof are not applicable, an amendment will be filed with WNP-3 specific "information".

Provide a schedule for completion of these evaluations. Chapter 4.0 data is used in other chapters of the FSAR and/or CESSAR-F. Provide an evaluation of the effects of the WNP-3 fuel cycle data on the results in other chapters (e.g., accident analyses, instrumentation setpoints, etc.)

Response

The WNP-3 specific fuel cycle necessitates an evaluation of the fuel design parameters, and safety analysis results as presented in CESSAR-F. Section 4.3, which addresses the nuclear design, will change substantially. Other sections of Chapter 4, notably 4.2 and 4.4 will require some modification.

All other areas of the NSSS scope impacted by the WNP-3 specific fuel cycle will be evaluated. We expect the specific fuel cycle design to be bounded by the envelope presented in the CESSAR-F safety analysis.

The evaluation of the WNP-3 specific fuel design and amendments to Chapter 4 and all other sections of the FSAR, requiring revisions due to the specific fuel cycle, will be provided by June 1983.

Attachment 1-97
2268A

Question No.

620.1 (18) Provide a Section 18 discussion regarding a detailed Control Room Design Review per NUREG-0660, NUREG-0737 and SECY-82-111.

Response

In accordance with NUREG-0660 and 0737 Item 1.D.1 licensees and applicants for operating license are required to conduct a detailed Control Room Design Review in an effort to identify and correct human engineering design deficiencies. The Supply System shall submit for NRC review within two months prior to the start of the Control Room Review a progress plan that describes how the Control Room Design Review will be accomplished. The guidance provided in SECY-82-111 will be utilized in the development of the approach to Control Room Design Reviews.

In addition to the above, acceptance criteria which is being developed in conjunction with the development of each subsection of Section 18.0 of the Standard Review Plan will be addressed by the Supply System upon finalization by the staff. For use as interim guidance the Supply System will utilize the guidelines for Control Room Design Review as set forth in NUREG-0700.

Attachment 1-98
2268A

Question No.

630.1 (13.2.2) This section should include a chart to show the schedule of, or each part of, the reactor operator training program. The time scale should be relative to expected fuel loading and should also display the preoperational test period, expected time for examinations for licensed operators prior to criticality, and expected time for examinations for licensed operators after criticality. This section should delineate clearly the extent to which the training program has been accomplished at the approximate time of the submittal of the FSAR.

Response

The extent of training accomplished at the approximate time of submittal of the FSAR and a chart depicting the schedule for the Reactor Operator training program will be provided in a subsequent amendment to the WNP-3 FSAR as shown by the attached pages.

f) Command responsibility and limits

g) Administrative requirements for the particular SRO position.

License Review Training (4 weeks)

A comprehensive examination is given to the license candidates to determine their knowledge of the plant (written examination) and ability to safely operate (simulator examination). Based on the results of the examination, review training and/or individual tutoring may be provided. Instruction on plant design and operating problems at similar plants will be provided.

Training Program Evaluation

The performance of employees participating in the Cold License Training Program are monitored and evaluated throughout the program. Frequent examinations are given to license candidates in order to determine the effectiveness of the training and the knowledge of the trainees. Records will be maintained on an individual basis. In the event the scheduled fuel loading date is substantially delayed, the cold license candidates will continue to maintain proficiency through participation in training similar in scope to the retraining program described in (Section 13.2.3.1).

Training Schedule

The training schedule for license operator candidates is based upon three training groups. The three groups include initial license candidates and replacements. The schedule for the three groups is shown in Figure 13.2-2.

At the time of FSAR submittal the first group was in training and had completed the following courses:

COURSE	APPROXIMATE CONTACT HOURS
Pre-Calculus Math (Math 103, 105)	90
Trigonometry	25
Chemistry	45

COURSE (CONT'D)	HOURS (CONT'D)
Calculus I, II, III	135
General Physics I, II, III	135
Fluid Mechanics	45
Material Science	45

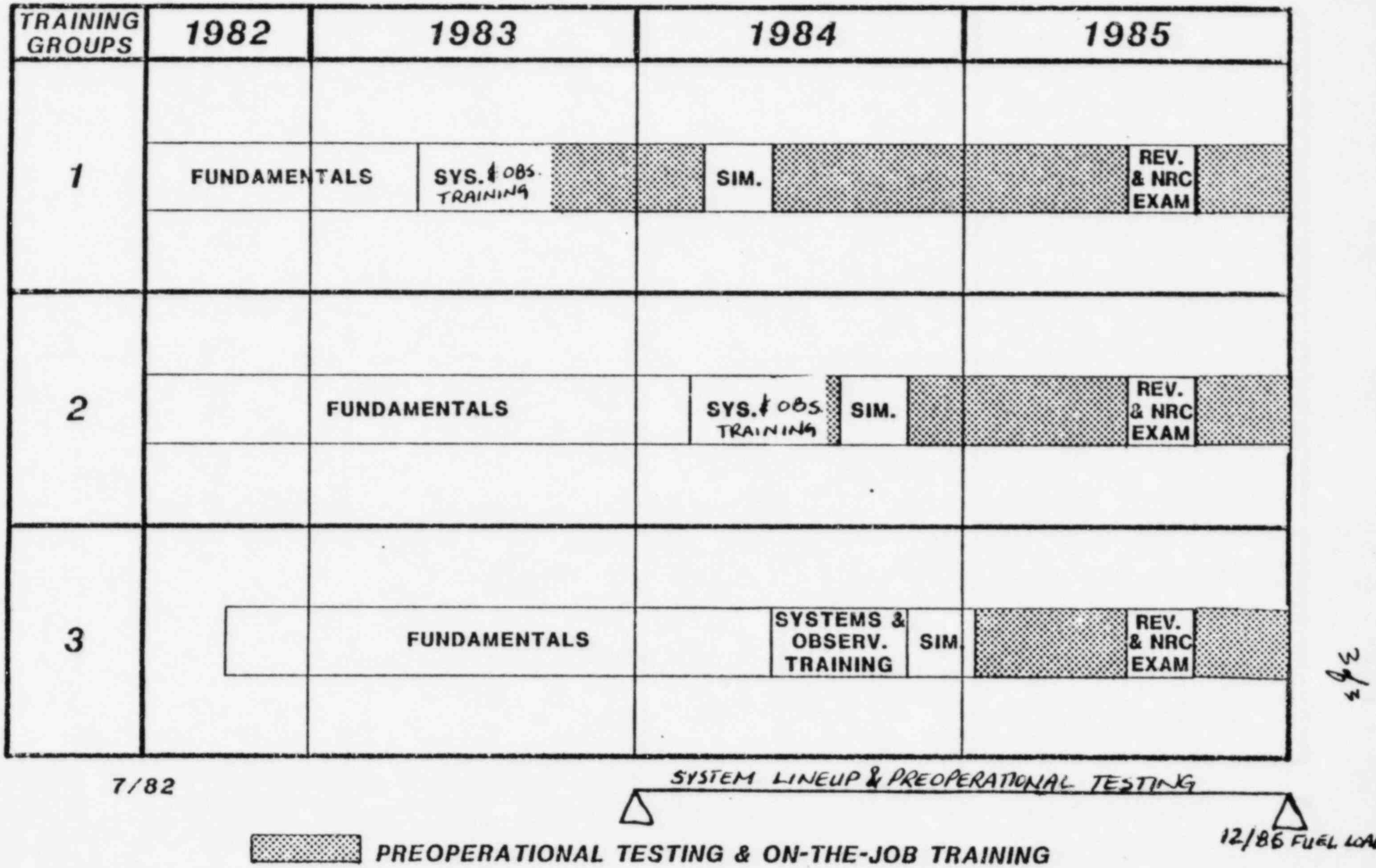

FIGURE 13.2-2 WNP-3 TRAINING SCHEDULE

Attachment 1-99
2268A

Question No.

630.2 (13.2.5) Subsection 13.2.5 states that Regulatory Guide 1.101 is no longer active. This was withdrawn at one time but has been reinstated as Revision 2 in October of 1981 and should be addressed in the FSAR. Subsection 13.2.5 also indicates that information regarding compliance with Regulatory Guides 8.2, 8.8, and 8.10 will be supplied later. Either supply this information or provide a schedule for its submittal.

Response

Subsection 13.2.5 will be revised in a subsequent FSAR amendment as indicated by the following:

Regulatory Guide 8.2 dated February 1973, "Guide for Administrative Practices in Radiation Monitoring" was utilized in establishing the Health Physics program (see Subsection 12.5.1.1).

Regulatory Guide 8.8 dated June 1978, "Information Relevant to Ensuring that Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" was utilized in developing radiation protection training programs. Training programs address Provisions of the Guide. Regulatory Guide 1.101 dated October 1981, "Emergency Planning and Preparedness for Nuclear Power Reactors" establishes NUREG 0654/ FEMA-REP-1 as an acceptable method for compliance. WNP-3 will meet the intent of NUREG/FEMA-REP-1 thereby complying with Regulatory Guide 1.101 (see Subsection 13.3). Regulatory Guide 8.10 dated May 1977, "Operating Philosophy for Maintaining Operational Radiation Exposure As Low As Reasonably Achievable" was used in establishing radiation protection training. The only exception taken to the guide is to provide training every two years instead of testing every year. (This was discussed with the NRC in conjunction with WNP-2 SER. The WNP-2 sumbittal was found acceptable.)

13.2.4.3 Senior Operator

Senior operator candidates will be selected from the Licensed Reactor Operators or from experienced and qualified plant staff.

Academic and Nuclear Plant Fundamentals

Senior Operator candidates will receive an Academic/Nuclear Fundamentals Program described in Table 13.2-1.

Senior Operators and Shift Managers Duties (3 weeks)

Senior Operators and Shift Managers will have instruction in subjects relating to their duties.

13.2.4.4 Replacement Superintendents Managers and Supervisors (Same as 13.2.2.3)

13.2.4.5 Replacement Technical Department Support Personnel (Same as 13.2.2.4)

13.2.4.6 Replacement Maintenance Department Personnel (Same as 13.2.2.5)

13.2.5 APPLICABLE NRC DOCUMENTS

The Supply System will comply with the below documents or take exception to documents as identified below:

a) 10CFR Part 50, "Licensing of Production and Utilization Facilities". No exceptions.

b) 10CFR Part 55, "Operators Licenses". No exceptions.

c) 10CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections". No exceptions.

d) Regulatory Guide 1.8, "Personnel Qualification and Training", see FSAR Section 17.2.

The Supply System will comply with the below documents or take exception to documents as identified below:

e) Regulatory Guide 1.101, October 1981 "Emergency Planning for Nuclear Power Plants". ~~This is no longer an active Regulatory Guide~~. establishes NUREG 0654/FEMA-REP-1 as an acceptable method for compliance. WNP-3 will meet the intent of NUREG 0654/FEMA-REP-1 thereby complying with Regulatory Guide 1.101 (see Section 13.3).

f) Regulatory Guide 8.2, February 1973 "Guide for Administrative Practices in Radiation Monitoring" ~~(latest)~~ was utilized in establishing the Health Physics program (see section 12.5.1.1)

g) Regulatory Guide 8.8, June 1978 "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable" ~~(latest)~~ was utilized in developing radiation protection training programs.

h) Regulatory Guide 8.10, May 1977 "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable" ~~(latest)~~ was utilized in establishing radiation protection training. The only exception taken to the Regulatory Guide is to provide training every two years instead of testing every year.

i) Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure".

j) "Utility Staffing and Training for Nuclear Power", WASH-1130, Revised June, 1973. Contains advisory information and was considered in establishing staffing training.

k) NRC Operator Licensing Guide, NUREG-0094, July, 1976. Contains information "intended to assist" and was considered in establishing training programs.

Attachment 1-100
2268A

Question No.

640.1 (14.2.4) State whether all tests at each given power test plateau will be performed before increasing power to the next plateau (power level).

Response

Add to FSAR Subsection 14.2.4.2.2-Startup Test Scheduling and Sequencing:

Those tests which are deemed necessary to ensure the safe operation of the plant will be conducted at all incremental power plateaus during Power Ascension Physics Testing (PAPT). These tests will be identified in the PAPT program procedure and will include such tests as verification of negative power coefficient of reactivity and steady state core performance including calorimetric NI calibrations. Certain other tests may be waived at intermediate power levels or rescheduled to a higher power level if WNP-3 is found to behave in an acceptable manner relative to the lead CESSAR Standard Design plant (CESSAR-F, Chapter 14) and any test results obtained at lower power levels are found to be acceptable. Tests may be waived or rescheduled at the discretion of the Plant Manager.

Q 640.1
1 of 1

14.2.4.2.2 Startup Test Scheduling and Sequencing

Scheduling and sequencing of testing during startup is performed under the direction of the Plant Manager.

The test sequence and individual Startup Test Procedure identify hold points for data review and authorization to proceed and establishes the general plant condition for each group of tests.

14.2.4.2.3 Startup Test Performance

Before starting each test, the assigned test engineer will review the test procedure to assure that prerequisite activities or conditions have been satisfied as described in Subsection 14.2.4.3.

The test will be stopped or curtailed if it cannot be performed safely or in accordance with the approved test procedure. Required test procedure deviations or changes may be effected in accordance with a "Test Change Notice" as described in Subsection 14.2.4.4.

Should apparent deviations of test results from performance requirements or acceptance criteria be revealed, or should other anomalies develop, the plant will be placed in a safe condition and relevant test data reviewed by the test engineer and Shift Manager. If the apparent discrepancy or anomaly is substantiated, the situation will be reviewed by the Plant Operations Committee to ascertain if a plant safety question is involved. Control of any identified nonconformance or noncompliance will be in accordance with the plant administrative procedures.

Evaluation of the effect of the discrepancy or anomaly on plant safety will be performed at the appropriate level of review and appropriate corrective action will be taken before resumption of the test or test conditions at which the problem was revealed.

At the completion of an entire test procedure, the test engineer will assemble all of the data and supporting information, nonconformance documentation and test results evaluations for review by the Plant Operating Committee. Any data reduction or analysis required will be done as soon after the data is available as is practical so that the results of the analysis may be included in the completed test package.

Test records will be maintained as described in Subsection 14.2.6.

14.2.4.3 Control of Test Prerequisites

Conditions and activities prerequisite to a given test will be identified in the applicable test procedure. Prior to commencement of the particular test.

¶ Those tests which are deemed necessary to ensure the safe operation of the plant will be conducted at all incremental power plateaus during Power Ascension Physics Testing (PAPT). These tests will be identified in the PAPT program procedure and will include such tests as verification of negative power coefficient of reactivity and steady state core performance including calorimetric NI calibrations. Certain other tests may be waived at intermediate power levels or rescheduled to a higher power level if WNP-3 is found to behave in an acceptable manner relative to the lead CESSAR Standard Design plant (CESSAR-F, Chapter 14) and any test results obtained at lower power levels are found to be acceptable. Tests may be waived or rescheduled at the discretion of the Plant Manager.

Question No.

640.2 (14.2.5) Clarify the description of the review process for preoperational and startup test results. Include: (1) requirements for review and approval of the test data for each major test phase before beginning the next phase, and (2) a description of the requirements for review and approval of the startup test data at each major power test plateau before raising power to the next test plateau during the power ascension phase.

Response

All test data and results will be reviewed by personnel formally judged to be qualified for such evaluation, as noted in Subsection 14.2.5.1. Based on that review, all test results will be approved, as necessary, by the Startup Manager and, for post-core tests, the Plant Manager. This approval reflects a judgment that the test data and results adequately satisfy the test objectives and criteria specified in the test procedure. Instances of deficiency or incompleteness will be dispositioned formally, as noted in Subsection 14.2.5.2. In general, testing may proceed from SLT to Preoperational Phase to the Startup Phase as permitted by the procedure-specified test prerequisites. However, prior to proceeding with fuel loading and major power test plateaus, a formal assessment may be required, as noted in Subsection 14.2.5.3, to consider the readiness of the plant equipment, procedures, and personnel for this next major phase. This readiness judgment will be made within the framework of intention to comply with all regulatory commitments, preparation to cope with unexpected plant performance, and avoidance of plant damage.

Attachment 1-102
2268A

Question No.

640.3 (14.2.12) Subsection 14.2.12 indicates that test descriptions for other post-core hot functional and power ascension tests will be provided later. Provide a schedule for submittal of this information or submit the test descriptions.

Response

Test descriptions for other post-core hot functional and power ascension tests noted in Subsection 14.2.12 will be developed and approved three months prior to initiation of fuel loading.

Attachment 1-103
2268A

Question No.

810.1 (13.3.2) The Washington Nuclear Project 3 Emergency Preparedness Plan indicates that additional information on some facilities in the design phase and one County and State emergency response plans will be provided later. Provide a schedule for submittal of this information.

Response

1. Additional information comparing Supply System facilities to NUREG-0696 will be available October 31, 1983.

2. Washington State Plan - submitted to FEMA in July. FEMA review expected Fall of 1982. Revised during Summer of 1983. The revised plan will be available September 30, 1983.

3. Grays Harbor County Plan - schedule same as Washington State.

4. Mason County Plan - to be submitted to FEMA by the end of 1982. The revised plan will be available September 30, 1983.

Attachment 1-104
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

1) ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

Commission Memorandum and Order of May 23, 1980 defines the current staff requirements for qualification of this equipment. Additional guidance on this matter was provided in a subsequent NRR order dated November 26, 1980 (concerning record requirements), Supplements 2 and 3 dated September 30, 1980 and October 24, 1980, respectively to IE Bulletin No. 70-OIB, and a generic letter dated October 1, 1980 to all holder of CPs and OLs.

Response

The environmental qualification of safety-related and electrical equipment is discussed in Section 3.11. Tables 3.11-1 and 3.11-2 "Environmental Qualification of Safety-Related and Electrical Equipment" will be updated quarterly beginning December 1982 with a scheduled completion date of June 1984. Additional information concerning documentation will be available by January 1983.

Attachment 1-105
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

2) EMERGENCY PREPAREDNESS

Guidance on the preparation of emergency plans is presented in NUREG-0654 (FEMA-REP-1) "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants". The requirements for the emergency response facilities are included in NUREG-0696 "Functional Criteria for Emergency Response Facilities." Further guidance on emergency preparedness is provided in the revised Appendix E to 10CFR Part 50.

Response

The WNP-3 Emergency Preparedness Plan was written using NUREG-0654/FEMA-REP-1 as guidance. Additional information with respect to the compliance with this document can be found in the WNP-3 Emergency Preparedness Plan, Appendix 8, "cross reference to NUREG-0654". The WNP-3 Emergency Response Facilities were designed using NUREG-0696 as guidance. A detailed document comparing the design of WNP-3 Emergency Response Facilities to NUREG-0696 will be available October 31, 1983.

Appendix E to 10CFR Part 50 refers to NUREG-0654 for guidance in developing Emergency Preparedness Plans. NUREG-0654 was used in developing the WNP-3 Emergency Preparedness Plan.

Attachment 1-106
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

3) SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS (Q-LIST) CONTROLLED BY THE QA PROGRAM

Staff requests for additional information regarding this issue have been sent to a number of OL applicants. A request from the Diablo Canyon review is provided as Enclosure 5.

Response

The Supply System is in the process of developing the Q List. We expect to have finalized this list by February 1983.

Attachment 1-107
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

4) FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY (GDC-51)

GDC-51 requires that under operating, maintenance, testing and postulated accident conditions, (1) the Ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Ferritic materials of the contaiment pressure boundary which are assessed by the staff are those of components such as freestanding containment vessel, equipment hatches, personnel airlocks, primary containment drywell head, heads containment penetration sleeves, process pipes, and closure caps and flued heads and penetrating piping systems downstream of penetration process pipes extending to and including the system isolation valves.

The acceptability of these materials within the context of GDC-51 is determined in accordance with the fracture toughness criteria identified for Class 2 materials by the Summer 1977 Addenda to ASME Code Section III.

Response

The FSAR has been reviewed and it has been determined that the ferritic materials of the containment pressure boundary have been sufficiently identified in the FSAR to allow the NRC to perform their review. Refer to FSAR Subsection 3.1.44 for discussion regarding GDC-51. CESSAR-F was not reviewed in this regard and is not considered necessary to be reviewed.

Attachment 1-108
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION

In accordance with Subsection 9.5.1., Branch Technical Position ASB 9.5-1, Position C.4.a(1) of NRC Standard Review Plan and Section III.G of new Appendix R to 10CFR Part 50, it is the staff's position that cabling for redundant safe shutdown systems should be separated by walls having a three-hour fire rating or equivalent protection (see Section III.6.2 of Appendix R). That is, cabling required for or associated with the primary method of shutdown, should be physically separated by the equivalent of a three-hour rated fire barrier from cabling required for/or associated with the redundant or alternate method of shutdown. To assure that redundant shutdown cable systems and all other cable systems that are associated with the shutdown cable systems are separated from each other so that both are not subject to damage from a single fire hazard, we require the following information for each system needed to bring the plant to a safe shutdown.

Response

Safe Shutdown Steps and related equipment lists have been prepared for use in a safe shutdown analysis. An analysis of all essential equipment is being performed in order to insure that no single fire can prevent WPPSS Unit 3 from achieving a safe cold shutdown. In this regard a list of all equipment and power sources required to bring the plant to cold shutdown has been compiled. This list differentiates between equipment required to maintain hot standby and that required for cold shutdown only.

To facilitate this review a computer program was developed which lists all essential equipment that appear in a fire area. This report will be analyzed by fire area, in conjunction with the physical drawings, verify that the operation of redundant essential components are not impaired by a single fire in any fire area. The results of this analysis will be presented by fire area and will include one acceptable means of separation as described in Appendix R to 10CFR50, Section III.G.2. This analysis will demonstrate that no single fire could prevent the plant from being brought safely to cold shutdown.

Question No.

040.75 Provide a table that lists all equipment including instrumentation and vital support system equipment required to achieve and maintain hot and/or cold shutdown. For each equipment listed:

a. Differentiate between equipment required to achieve and maintain hot shutdown and equipment required to achieve and maintain cold shutdown,

b. Define each equipment's location by fire area,

c. Define each equipment's redundant counterpart,

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION (contd.)

d. Identify each equipment's essential cabling (instrumentation, control, and power). For each cable identified: (1) Describe the cable routing (by fire area) from source to termination, and (2) Identify each fire area location where the cables are separated by less than a wall having a three-hour fire rating from cables for any redundant shutdown system, and

e. List any problem areas identified by Item 1.d.(2) above that will be corrected in accordance with Section III.6.3 of Appendix R (i.e., alternate or dedicated shutdown capability).

Response

A list of all equipment and their power sources required to achieve and maintain hot and/or cold shutdown will be provided by May, 1983. This list will differentiate between equipment required for hot standby and cold shutdown, define components location by fire area and list the redundant components of the essential systems. The listing will be developed assuming off-site power not available.

In areas where the function of essential redundant components could be damaged by a single fire, the equipment and cables will be either relocated or one means of insuring that one of the redundant trains is free of fire damage to meet the requirements of Appendix R to 10CFR so Section III.G.2 will be provided.

The Supply System is taking exception to providing a listing of essential cabling for safe shutdown equipment. It is our position that all safety class electrical trays, conduits and pull boxes for redundant systems will be sufficiently separated to meet separation requirements of Appendix R. For areas where associated cables are separated by less than a barrier having a three-hour rating from cables required for/or associated with a redundant shutdown system, these cables will be protected at the power source by electrical devices (breakers, fuses). In the event of fire damage to these cables, they will become electrically isolated, thus preventing the propagation of damage to other cables in that raceway section. Instrumentation cables are self-protected by virture of low energy levels they convey.

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION (contd.)

Question No.

040.76 Provide a table that lists Class 1E and Non-Class IE cables that are associated with the essential safe shutdown systems identified in Item 1 above. For each cable listed: (*See note on page 3.)

a. Define the cables' association to the safe shutdown system (common power source, common raceway, separation less than IEEE Standard-384 guidelines, cables for equipment whose spurious operation will adversely affect shutdown systems, etc.).

b. Describe each associated cable routing (by fire area) from source to termination, and

c. Identify each location where the associated cables are separated by less than a wall having a three-hour fire rating from cables required for/or associated with any redundant shutdown system.

Response

The Supply System is taking exception to providing a listing of Class IE and Non-Class IE cables that are associated with the essential safe shutdown systems. It is our position that these associated cables whether safety (Class IE) or non-safety (Non-Class IE) are both qualified as Class IE cables.

For areas where associated cables are separated by less than a barrier having a three-hour rating from cables required for/or associated with a redundant shutdown system, these cables will be protected at the power source by electrical devices (breakers, fuses). In the event of fire damage to these cables, they will become electrically isolated, thus preventing the propagation of damage to other cables in that raceway section. Instrumentation cables are self-protected by virtue of low energy levels they convey.

Question No.

040.77 Provide one of the following for each of the circuits identified in Item 2.c above:

Attachment 1-111
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION (contd.)

Question No.

040.77 (contd.)

(a) The results of an analysis that demonstrates that failure caused by open, ground, or hot short of cables will not affect it's associated shutdown system, *Note*

(b) Identify each circuit requiring a solution in accordance with Section III.6.3 of Appendix R,

or

Identify each circuit meeting or that will be modified to meet the requirements of Section III.G.2 of Appendix R (i.e., three-hour wall, 20 feet of clear space with automatic fire suppression, or one-hour barrier with automatic fire suppression).

Response

The Supply System is taking exception to providing a listing of Class IE and Non-Class IE cables that are associated with the essential safe shutdown systems. It is our position that these associated cables whether safety (Class IE) or non-safety (Non-Class IE) are both qualified as Class IE cables.

*NOTE

Option 3a is considered to be one method of meeting the requirements of Section II.G.3 Appendix R. If Option 3a is selected the information requested in Items 2a and 2c above should be provided in general terms and the information requested by 2b need not be provided.

For areas where associated cables are separated by less than a barrier having a three-hour rating from cables required for/or associated with a redundant shutdown system, these cables will be protected at the power source by electrical devices (breakers, fuses). In the event of fire damage to these cables, they will become electrically isolated, thus preventing the propagation of damage to other cables in that raceway section. Instrumentation cables are self-protected by virtue of low energy levels they convey.

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION (contd.)

Question No.

040.78 To assure compliance with GDC 19, we require the following information be provided for the Control Room. If credit is to be taken for an alternate or dedicated shutdown method for other fire areas (as identified by Item 1.a or 3.b above) in accordance with Section III.G.3 of new Appendix R to 10CFR Part 50, the following information will also be required for each of these plant areas.

a. A table that lists all equipment including instrumentation and vital support system equipment that are required by the primary method of achieving and maintaining hot and/or cold shutdown.

b. A table that lists all equipment including instrumentation and vital support system equipment that are required by the alternate, dedicated, or remote method of achieving and maintaining hot and/or cold shutdown.

c. Identify each alternate shutdown equipment listed in Item 4.b above with essential cables (instrumentation, control, and power) that are located in the fire area containing the primary shutdown equipment. For each equipment listed provide one of the following:

(1) Detailed electrical schematic drawings that show the essential cables that are duplicated elsewhere and are electrically isolated from the subject fire areas, or

(2) The results of an analysis that demonstrates that failure (open, ground, or hot short) of each cable identified will not affect the capability to achieve and maintain hot or cold shutdown.

d. Provide a table that lists Class IE and Non-Class IE cables that are associated with the alternate, dedicated, or remote method of shutdown. For each cable so identified provide the results of an analysis that demonstrates that failure (open, ground, or hot short) of the associated cable will not adversely affect the alternate, dedicated, or remote method of shutdown.

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION (contd.)

Response

040.78 (contd.) The Remote Shutdown Panel (RSP), local Diesel Generator (DG) Control panels and Local Turbine Driven AFW Control panels located in the Reactor Auxiliary Building provides an alternative location from the Control Room, which allows the operator to: 1) Achieve prompt hot shutdown of the reactor; 2) Maintain the unit in a safe condition during hot shutdown; 3) Achieve cold shutdown of the reactor through the use of suitable procedures. The Remote Shutdown Panel is fully described in FSAR Subsection 7.4.1.1. The RSP would be utilized as the alternate means of plant shutdown in the event that the Control Room becomes uninhabitable.

The Supply System is taken exception to providing a listing of essential cabling for safe shutdown equipment. It is our position that all safety class electrical trays, conduits and pull boxes for redundant systems are sufficiently separated to meet the separation requirements of Appendix R. For areas where associated cables are separated by less than a barrier having a three-hour rating from cables required for/or associated with a redundant shutdown system, these cables will be protected at the power source by electrical devices (breakers, fuses). In the event of fire damage to these cables, they will become electrically isolated, thus preventing the propagation of damage to other cables in that raceway section. Instrumentation cables are self-protected by virtue of low energy levels they convey.

Question No.

040.79 The Residual Heat Removal System is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, this interface most likely consists of two redundant and independent motor operated valves with diverse interlocks in accordance with Branch Technical Position ICSB 3. These two motor operated valves and their associated cable may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

5) FIRE PROTECTION (contd.)

Question No.

040.78 (contd.)

a. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

b. Identify each device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

c. Identify each locaton where the identified cables are separated by less than a wall having a three-hour fire rating from cables from the redundant device.

d. For the areas identified in Item 5.c above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications.

Response

The assurance that high-low pressure interfaces are adequately protected from the effects of a single fire will be demonstrated in the safe shutdown analysis which will be completed by April 1983.

Attachment 1-115
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

6) EFFECTS OF MASONRY WALLS ON CLASS I STRUCTURES

1. Are there any concrete masonry walls being used in any of the Category I structures of your plant? If the answer is "No" to this question there is no need to answer the following questions.

2. Indicate the loads and load combinations to which the walls were designed to resist. If load factors other than one (1) have been employed, please indicate their magnitudes.

3. In addition to complying with the applicable requirements of the SRP Sections 3.5, 3.7 and 3.8, is there any other code, such as the "Uniform Building Code" or the "Building Code Requirements for Concrete Masonry Structures" (Proposed by the American Concrete Institute) which was or is being used to guide the design of these walls? Please identify and discuss any exceptions or deviations from the SRP requirements or the aforementioned codes.

4. Indicate the method that you used to calculate the dynamic forces in masonry walls due to earthquake, i.e., whether it is a code's method such as Uniform Building Code, or a dynamic analysis. Identify the code and its effective date if the code's method has been used. Indicate the input motion if a dynamic analysis has been performed.

5. How were the masonry walls and the piping/equipment supports attached to them designed? Provide enough numerical examples including details of reinforcement and attachments to illustrate the methods and procedures used to analyze and design the walls and the anchors needed for supporting piping/equipment (as applicable).

6. Provide plan and elevation views of the plant structures showing the location of all masonry walls for your facility.

Response

A complete response to this request for information will be available by December 1982.

Attachment 1-116
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

7) INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

(TMI Action Item II.F.2 in NUREG-0737)- Discussion of this item should address how core thermocouple readouts are provided in the control room including location and rate of printout (see Part (4) of Attachment 1 to item II.F.2).

Response

Instrumentation for detection of inadequate core cooling are within the CESSAR-F scope. This issue is being reviewed on the CESSAR-F Docket as discussed in NUREG-0852 "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System, CESSAR System 80".

Attachment 1-117
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

8) PRESERVICE AND INSERVICE INSPECTIONS

Staff guidance in this review area has been sent to a number of pending OL applicants. A copy of that guidance is provided as Enclosure 8.

Response

The Preservice Inspection Program Plan will be submitted as a separate document by September 1983 and will be consistent with the requirements of ASME Section XI, 1977 edition with addenda through Summer 1978. In accordance with 10CFR50, the WNP-3 Inservice Inspection Plan will be submitted six months prior to start of commercial operation.

Attachment 1-118
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

9) PRESERVICE INSPECTION AND TESTING OF SNUBBERS

The staff has recently established requirements to ensure snubber operability which have been transmitted to pending OL applicants. A copy of those requirements is provided as enclosure 9 (Attachment I).

Response

The Supply System has reviewed the Preservice Examination and Pre-Operational Testing requirements discussed in the referenced Enclosure 9 (Attached). It is the intent of WNP-3 to fully comply with the requirements as stated in Enclosure 9. The Preservice Examination for snubbers on safety-related systems will be detailed in the "WNP-3 Preservice Inspection Program Plan" which will be submitted for NRC staff review by August 1983. The Pre-Operational testing requirements will be included in the Pre-Operational and Startup Tests described in Chapter 14 of the FSAR by July 1983.

Attachment 1-119
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

10) EFFECTS OF CONTAINMENT COATINGS AND SUMP DEBRIS ON ECCS AND CONTAINMENT SPRAY OPERATION

A copy of the staff concerns on this issue, including a request for additional information which has been sent to a number of OL applicants, is provided as Enclosure 10 (Attachment I).

Response

The Supply System is evaluating WNP-3 with respect to the Staff concerns mentioned above. Further information will be available by November 1982.

Attachment 1-120
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

11) SEISMIC QUALIFICATION

A staff request for additional information in this review area has been sent to a number of pending OL applicants. A copy of that request is provided as Enclosure 11 (Attachment I).

Response

The Supply System will have further information available concerning Seismic Qualification by December 1982.

Attachment 1-121
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

12) SPECIAL LOW POWER TEST PROGRAM (TASK ACTION PLAN ITEM I.G.1)

The staff has established guidance on this matter for transmittal to all pending and prospective OL applicants. A copy of that guidance is provided as enclosure 12 (Attachment 1).

Response

As noted in Subsection 14.2.12.6 of the WNP-3 FSAR, we propose to conduct the WNP-3 Low Power Testing in accordance with tests outlined in CESSAR-F Subsection 14.2.12.4. That Subsection recommends natural circulation tests in the first-of-a-kind NSSS plant, in this instance Palo Verde. It is our intention to monitor the conduct and results of that testing prior to finalizing test requirements for Low Power Testing at WNP-3. As noted previously, we intend to have completed preparation of all postcore test procedures at least three months prior to fuel loading.

Attachment 1-122
2268A

REQUEST FOR ADDITIONAL INFORMATION - ENCLOSURE 4

13) INITIAL TEST PROGRAM DESCRIPTION (CHAPTER 14)

Staff review of near term OL applications has revealed a number of concerns which are common to pending applications. The nature of these concerns are typically expressed in the questions the staff has raised in its review of the Summer and the San Onofre 2 & 3 applications.

Response

Staff concerns regarding startup testing as expressed in the questions raised in its review of the Summer and San Onofre 2 and 3 applications will be reviewed as part of the further development of the WNP-3 startup test program.

enclosure 4
item 14
1 of 1

TABLE 13.5-5

TYPICAL EMERGENCY PROCEDURES

Reactor Trip/Turbine Trip

Anticipated Transient Without Scram

Loss of Feedwater

Loss of Forced Reactor Coolant Flow

Loss of Coolant Accident

Steam Generator Tube Rupture

Steamline Break

Inadequate Core Cooling

Station Blackout