File this instruction sheet in the front of Volume 1 as a record of changes.

The following information and check list are furnished as a guide for the insertion of new sheets for Amendment 24 into the Preliminary Safety Analysis Report for the Skagit/Hanford Nuclear Project. This material is denoted by use of the amendment date in the upper right hand corner of the page.

New sheets should be inserted as listed below:

Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)
CHAPTER 1	
Figure 1.2-2	Figure 1.2-2
1B-ii/blank	1B-ii/blank
--	1B-80a/80b
	1B-80c/blank
1B-87/88	1B-87/88
CHAPTER 2	
2-v/vi	2-v/vi
Figure 2.1-16	Figure 2.1-16
2.2-11/12	2.2-11/12
2.3-1/2	2.3-1/2
Table 2.3-1 Sht 1/Sht 2 through Table 2.3-2/ Table 2.3-3 Sht 1	Table 2.3-1 Sht 1/Sht 2 through Table 2.3-2/ Table 2.3-3 Sht 1
Table 2.3-5 Sht 3/Sht 4	Table 2.3-5 Sht 3/Sht 4
2.4-11/12 through 2.4-13/14	2.4-11/12 through 2.4-13/14
Table 2.4-4/Table 2.4-5	Table 2.4-4/Table 2.4-5
Table 2.4-18/Table 2.4-19	Table 2.4-18/Table 2.4-19
Table 2.4-20/Table 2.4-21	Table 2.4-20/Table 2.4-21
Figure 2.4-5	Figure 2.4-5
2.5-19/20 through 2.5-23/24	2.5-9/20 through 2.5-23/24
2.5-27/28	2.5-27/28
--	2.5-28a/blank
2.5-29/30	2.5-29/30
Figure 2.5-11	Figure 2.5-11
Figure 2.5-15	Figure 2.5-15
2K-23/24	2K-23/24
2K-29/30 through 2K-31/32	2K-29/30 through 2K-31/32

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--	2K-32a/blank
Figure 2K-2	Figure 2K-2
Figure 2K-13	Figure 2K-13
Figure 2K-14	Figure 2K-14
Figure 2K-37	Figure 2K-37
Figure 2K-52	Figure 2K-52
2L-1/2	2L-1/2
2L-5/6	2L-5/6
Figure 2L-A9	Figure 2L-A9
Table 2O-6/Table 2O-7	Table 2O-6/Table 2O-7
Figure 2Q A-1	Figure 2Q A-1
Figure 2Q A-3	Figure 2Q A-3
Figure 2Q A-18	Figure 2Q A-18
Figure 2Q A-25	Figure 2Q A-25
Figure 2Q A-30	Figure 2Q A-30
Figure 2Q A-35	Figure 2Q A-35
Table 2Q B-1 (7 pages)	Table 2Q B-1 (7 pages)
Table 2R-3/blank	Table 2R-3/blank
Figures 2R-7 through 2R-18	Figures 2R-7 through 2R-18

CHAPTER 3

3.2-1/2	3.2-1/2
Table 3.2-1 (Sht 23 of 29)/ (Sht 24 of 29)	Table 3.2-1 (Sht 23 of 29)/ (Sht 24 of 29)
3.4-1/2 through 3.4-3/blank	3.4-1/2 through 3.4-3/blank
3.5-13/14	3.5-13/14
3.7-3/4 through 3.7-7/8	3.7-3/4 through 3.7-7/8
3.7-13/14 through 3.7-15/16	3.7-13/14 through 3.7-15/16
3.8-5/6	3.8-5/6
--	3.8-6a/blank
3.8-19/20	3.8-19/20
3.8-27/28 through 3.8-31/32	3.8-27/28 through 3.8-31/32
--	3.8-32a/blank
3.8-35/36	3.8-35/36
3.8-45/46 through 3.8-51/52	3.8-45/46 through 3.8-51/52
Table 3.8-1/Table 3.8-2	Table 3.8-1/Table 3.8-2
3.10-1/2 through 3.10-3/4	3.10-1/2 through 3.10-3/4

CHAPTER 5

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Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)
CHAPTER 6	
6.5-5/6	6.5-5/6
Table 6.5-2 (Sht 3 of 4)/ (Sht 4 of 4)	Table 6.5-2 (Sht 3 of 4)/ (Sht 4 of 4)
CHAPTER 7	
Figure 7.1-2	Figure 7.1-2
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--	7.3-30a/blank
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--	7.3-36a/blank
Table 7.3-2/Table 7.3-3	Table 7.3-2/Table 7.3-3
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CHAPTER 9	
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Figure 9.2-17 (Sht 1 of 3)	Figure 9.2-17 (Sht 1 of 3)

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CHAPTER 10	
--	Tab 10.4 (dark blue)
Table 10.4-1/Table 10.4-2	Table 10.4-1/Table 10.4-2
Figure 10.4-1	Figure 10.4-1
CHAPTER 11	
Table 11.3-1/Table 11.3-2 (Sht 1 of 2)	Table 11.3-1/Table 11.3-2 (Sht 1 of 2)
Table 11.3-2 (Sht 2 of 2)/ Table 11.3-3	Table 11.3-2 (Sht 2 of 2)/ Table 11.3-3
Tables 11.3-5 through 11.3-14 not used/ Table 11.3-15	Tables 11.3-5 through 11.3-14 not used/ Table 11.3-15
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Table 12.1-14 (Sht 4 of 4)/ Table 12.1-15	Table 12.1-14 (Sht 4 of 4)/ Table 12.1-15
CHAPTER 13	
(APPENDIX 13A)	
13A-v/vi	13A-v/vi
13A-9/10	13A-9/10
13A-17/18 through 13A-25/26	13A-17/18 through 13A-25/26
--	13A-26a/blank
13A-33/34 through 13A-41/42	13A-33/34 through 13A-41/42
13A-45/46 through 13A-53/54	13A-45/46 through 13A-53/54

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Table 1/Table 2 (Sht 1 of 3)	Table 1/Table 2 (Sht 1 of 3)
Table 2 (Sht 2 of 3)/ (Sht 3 of 3)	Table 2 (Sht 2 of 3)/ (Sht 3 of 3)
Table 3/Table 4	Table 3/Table 4
Appendix 13A Appendix A, Letters of Agreement to be provided subsequently sheet	Appendix 13A Appendix A

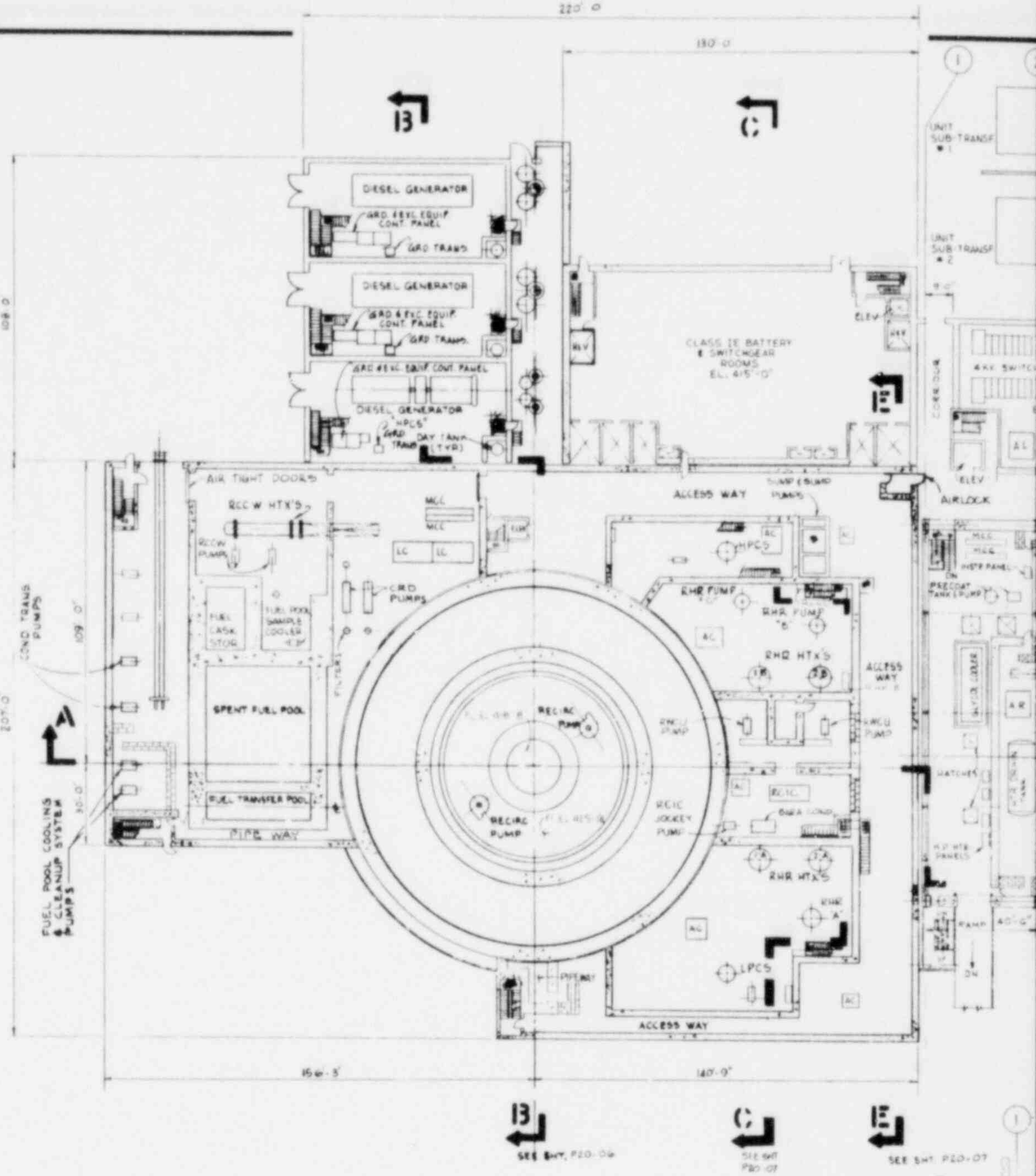
CHAPTER 15

15-i/ii	15-i/ii
15.2-1/2	15.2-1/2
15.2-5/6	15.2-5/6
15.2-13/blank	15.2-13/blank
15.4-3/4	15.4-3/4
15.6-5/6	15.6-5/6
Table 15.6-1/Table 15.6-2	Table 15.6-1/Table 15.6-2
Table 15.6-9/Table 15.6-10	Table 15.6-9/Table 15.6-10
Table 15.6-25/Table 15.6-26 through Table 15.6-29/ Table 15.6-30	Table 15.6-25/Table 15.6-26 through Table 15.6-29/ Table 15.6-30
15.7-13/14	15.7-13/14
Table 15.7-17/Table 15.7-18	Table 15.7-17/Table 15.7-18
15A-3/4	15A-3/4
Table 15A-1/Table 15A-2	Table 15A-1/Table 15A-2

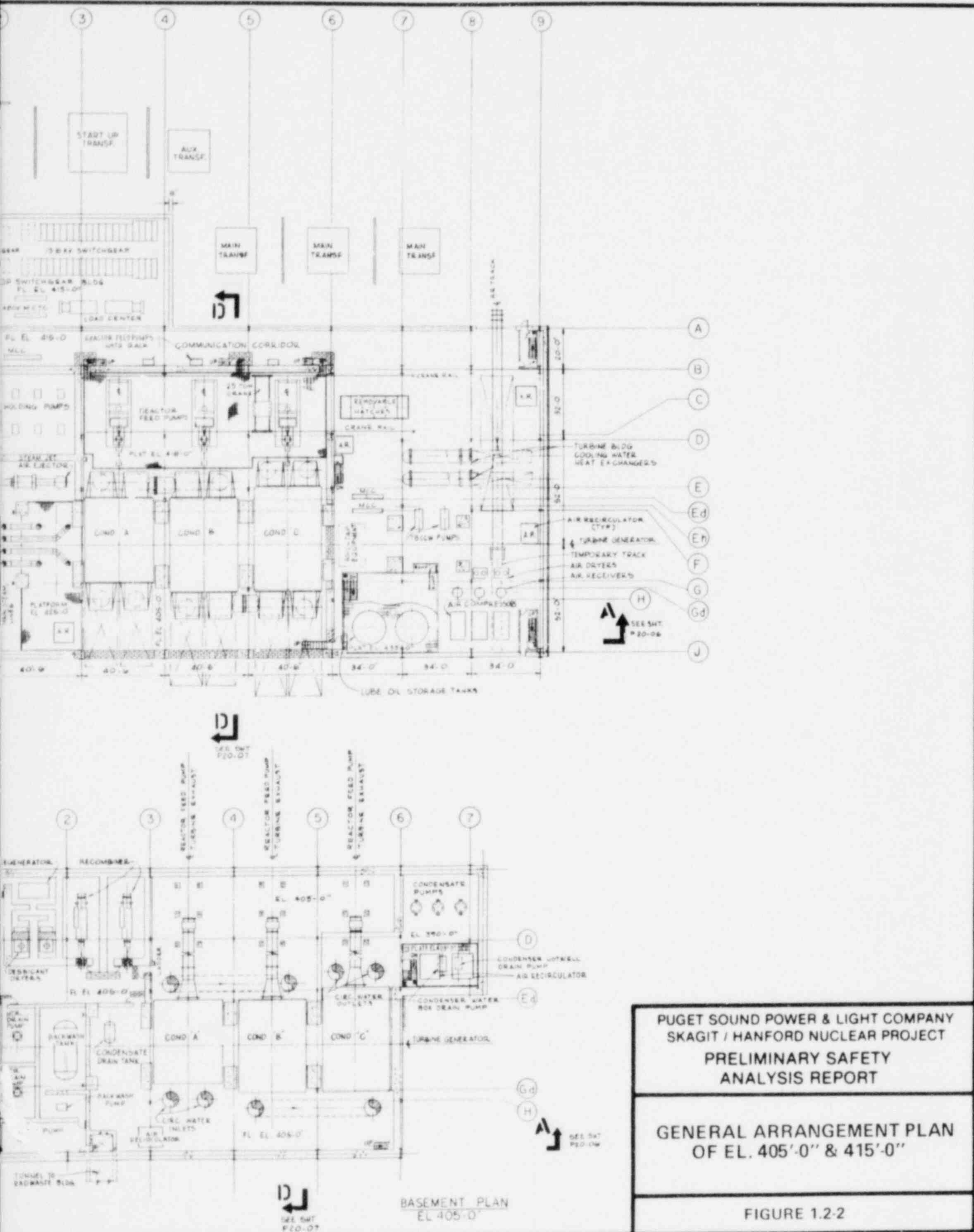
QUESTIONS AND RESPONSES

--	Tab Skagit Site (dark blue) at front of Questions and Responses Section)
--	Tab Skagit/Hanford Site (dark blue) at end of existing Questions and Responses Section
--	S/HNP Questions and Responses Introduction page/blank
--	Question 220.01/220.02 through Question 220.19/ blank
--	Question 230.1/230.1 (Cont'd) through Question 230.3/blank

Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)
--	Question 231.1/231.1 (Cont'd) through Question 231.3.b.1 (Cont'd)/ Question 231.3.b.3
--	Table 231.3-1/Table 231.3-2
--	Figures 231.1-1 through 231.1-3
--	Figure 231.3-1
--	Question 240.1/240.2 through Question 240.3/ 240.3 (Cont'd)
--	Question 241.1/241.1 (Cont'd) through Question 241.7/241.8
--	Question 271.1/blank
--	Question 403.1/403.2 through Question 403.3/403.4
--	Table 403.1-1
--	Figures 403.1-1 through 403.1-3
--	Figures 403.2-1 through 403.2-3
--	Figures 403.3-1 through 403.3-4
--	Question 410.1/410.1 (Cont'd) through Question 410.1 (Cont'd)/410.2
--	Question 420.1/blank
--	Question 421.2/blank
--	Question 460.1/460.2 through Question 460.4/ blank
--	Question 471.1/471.2-2(a) through Question 471.2-2(d)/blank
--	Question 810.1/blank



PLAN OF EL 415'-0"
UNLESS OTHERWISE NOTED



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GENERAL ARRANGEMENT PLAN OF EL. 405'-0" & 415'-0"

FIGURE 1.2-2

APPENDIX 1B

The following pages identify the Applicant's commitments regarding the design, construction, and operation of the S/HNP in response to the review of the incident at Three Mile Island Unit 2.

Commitments in this Appendix supersede any conflicting statements elsewhere in the PSAR where such conflicting statements were made earlier than the date of the current revision of this Appendix.

The following text consists of responses to NUREG-0718, Rev. 1, entitled "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License", dated June, 1981. These responses meet the requirements of the proposed amendment to 10 CFR 50, entitled "Licensing Requirements for Pending Construction Permits and Manufacturing License Applications", dated July 10, 1981, as sent to all parties to pending construction permit proceedings by Generic Letter No. 81-26.

The requirements of the Final Rule on Licensing Requirements for Pending Construction Permit and Manufacturing License Application, 10 CFR 50.34(f) Additional TMI-Related Requirements, effective February 16, 1982, have been compared with the information submitted in the S/HNP PSAR. This comparison indicates that S/HNP meets all requirements, with the inclusion of the response to 50.34(f)(3)(v)(B) (see page 1B-80a of this Appendix). 24

Response to 10 CFR 50.34(f)(3)(v)(B)

Requirement

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.

(v)(B) Containment structure loadings, produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A limits except that evaluation of instability is not required (for concrete containments the loading specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category, (2). The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting. 24

Response

In PSAR Appendix 1B, Section II.B.8(3), p. 1B-66, S/HNP has committed to a distributed ignition system for hydrogen control. A commitment has also been made to meet Containment integrity requirements during an accident that releases hydrogen generated from a 100% fuel-clad metal water reaction accompanied by hydrogen burning, including as a minimum, meeting the specified code requirement with an internal pressure of 45 psig as stated in Appendix 1B, Section II.B.8(4), p. 1B-68.

Section II.B.8(4) indicates that:

(1) The Containment general cylindrical shell, away from major discontinuities, was verified to be adequate to withstand the 45 psig internal pressure and accompanying thermal effects.

(2) An investigation will be made to verify the adequacy of the Containment shell adjacent to major discontinuities.

(3) The dome will be designed to the 45 psig criteria by increasing the amount of reinforcing if necessary.

(4) The base mat will be analyzed and designed for the 45 psig criteria prior to construction.

(5) Preliminary analysis performed to date at 45 psig indicates that the liner strain is well within the allowable limit.

In the event that a post-accident inerting system (assuming carbon dioxide) is ultimately required for hydrogen 24
control, it is expected that the pressure would be approximately 26.5 psig and the temperature would not exceed the design criteria already committed to in Appendix 1B, Section II.B.8(4). The pressure of 26.5 psig is based on the results of the analysis performed by the Allens Creek Project. (See Allens Creek PSAR Appendix O, p. O-167, Amendment 59).

Preliminary evaluation, based on results obtained in response to item II.B.8(4) indicates that for S/HNP, the Containment structure loading produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce strains in the Containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category (2), and that the Containment will have the capability to safely withstand pressure tests of 1.15 times the pressure calculated to result from carbon dioxide inerting. The pressure and temperature loads due to inadvertent actuation of a post-accident inerting system are expected to be less than the hydrogen burn pressures and are therefore not expected to control the design of the Containment.

Section II.B.8(4) of Appendix 1B describes a two phase post CP program for hydrogen control studies. Within 6 months following issuance of the Construction Permit, a report will be provided to the NRC including among other things, evaluation of alternate methods for accommodating hydrogen releases. In the event that as a result of this study, S/HNP elects to adopt a post-accident inerting system in place of the distributed ignition system, specific design criteria will be established and analysis specified for Containment integrity to demonstrate conformance with the specific requirement of 10 CFR 50.34(f)(3)(v)(B). At the completion of the full two-year program, analysis of Containment structure response to inadvertent actuation will be reported to the NRC in the event a post-accident inerting system is chosen for design. If design limits for inadvertent actuation of the post-accident inerting system are exceeded, design modification to the reinforced concrete Containment design will be initiated. 24

Our preliminary evaluation of Containment capability discussed above, ensures that the post-accident inerting method of hydrogen control remains a viable option until final selection of the method for hydrogen control is made.

pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification.

All systems that provide a path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal.

Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, Item II.3(f) during operational conditions 1, 2, 3 and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit state of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

The Containment Isolation System (CIS) is discussed in Section 6.2.4. The criteria for the design of the CIS Control System are listed in Section 7.1.2.1.2. The following are the responses to NUREG-0718, Item II.E.4.2:

1. Compliance with SRP 6.2.4, Rev. 1

   The design of the S/HNP Containment Isolation System will meet the recommendations of Standard Review Plan Section 6.2.4, Rev. 1. The details of how these requirements will be met will be described in the S/HNP FSAR. The present Containment Isolation System has been reviewed and accepted by the NRC (Reference: Skagit Nuclear Power Project SER, NUREG-0309, Section 6.2.15).

2. Identification of Essential and Nonessential Systems

PSAR Table 6.2-11 lists the systems which penetrate the containment. These systems will be categorized as essential, intermediate or nonessential in the FSAR. The following definitions will be applied in categorizing the systems. 21

Essential

Essential systems are those critical to the immediate mitigation of the consequences of a LOCA. Essential systems are not automatically isolated by accident signals. 22

Intermediate

Intermediate systems are those which could be useful (although not critical) in mitigating an accident which results in containment isolation. Intermediate systems are automatically isolated by accident signals. If automatically isolated, the operator may choose selectively to reopen the valves as they are needed, while the accident signal is still present. This permits the operator to use all available systems to cope with an accident, while still maintaining the effectiveness of the containment.

In summary, the isolation provisions for intermediate systems have the same essential features as nonessential systems (double barrier isolation, automatic isolation on diverse accident signals). The main difference is that intermediate systems can be manually re-opened by the operator while the accident signal is still present. 22 24

Nonessential

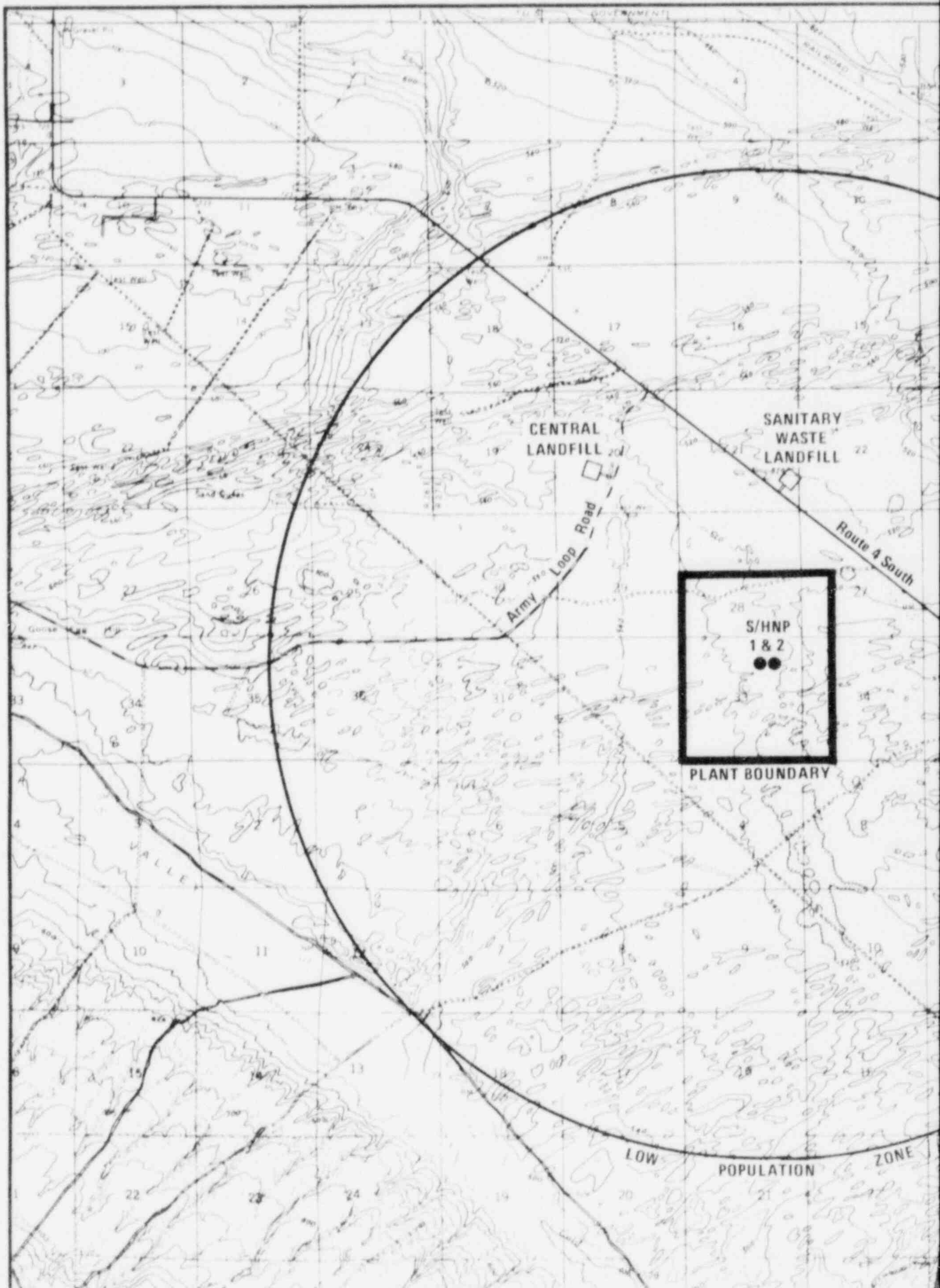
Nonessental systems are those which are not required or used in the mitigation of an accident which results in containment isolation. All nonessential systems are automatically isolated by the Containment Isolation Actuation Signal and cannot be reopened by the operator while the accident signal is still present. 21

3. Isolation of Nonessential Systems

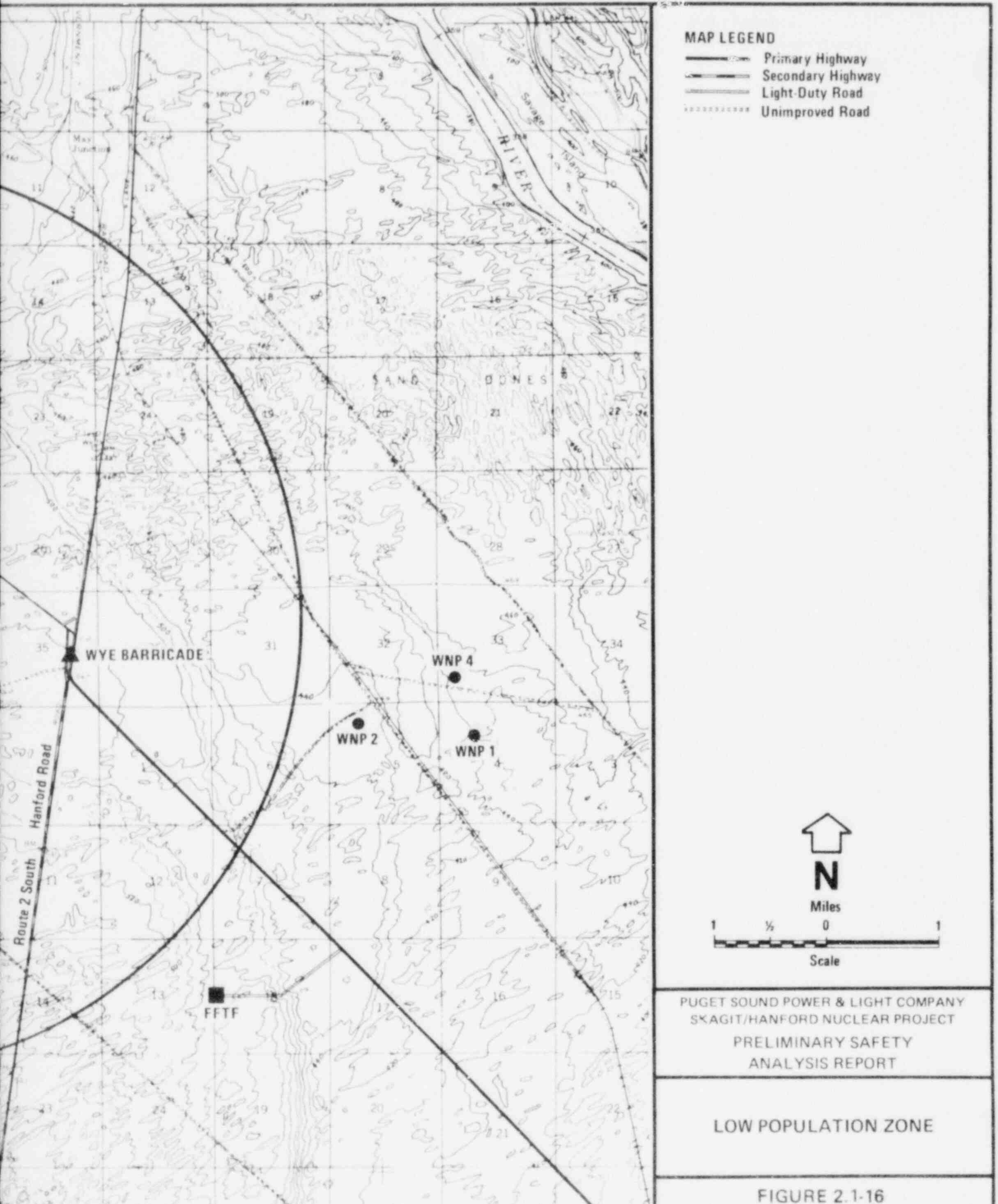
As required for post-accident situations, each nonessential penetration (except instrument lines)

APPENDICES

NUMBER	TITLE
2A through 2J	Not Used
2K	Geophysical Investigations Umtanum Ridge to Southeast Anticline Hanford Site, Washington
2L	Geophysical Investigations Skagit/Hanford Nuclear Project Site Hanford Site, Washington
2N	Geologic Structure of Umtanum Ridge: Priest Rapids Dam to Sourdough Canyon
2O	Gable Mountain: Structural Investigations and Analyses
2P	Geohydrologic Investigations
2Q	Foundation Investigation and Analysis
2QA	Field Investigations and Results
2QB	Laboratory Test Procedures and Results
2R	Stratigraphic Investigation of the Skagit/Hanford Nuclear Project
2S	The Origin of the Umtanum Ridge-Gable Mountain Structural Trend and Implications for a Regional Tectonic Model



CENTRAL
LANDFILL
SANITARY
WASTE
LANDFILL
Army Loop Road
Route 4 South
S/HNP
1 & 2
PLANT BOUNDARY
LOW
POPULATION
ZONE



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LOW POPULATION ZONE

FIGURE 2.1-16

A preliminary analysis of the chemicals identified above was performed to determine which chemicals could pose a hazard to the control room operators. The concentrations of toxic chemicals in the control room are calculated similarly to the analysis presented in Section 2.2.3.1.4.1. The results, shown on Table 2.2-5, indicate that anhydrous ammonia and truck shipments of chlorine could pose a hazard to the control room operators if transported on the roadway near the S/HNP Site. 23 24

A survey performed to determine shipment frequencies of chlorine and anhydrous ammonia indicates that chlorine is not shipped by trucks near the S/HNP Site (Refs 17, 18). Anhydrous ammonia is shipped in quantities of 2000-3000 gallons 8-10 times annually (Ref 17). Therefore only anhydrous ammonia could pose a hazard to the control room operators.

The Control Room habitability requirements are addressed in Section 6.4.

2.2.3.1.4.3 Chemicals stored on-Site. Chemicals stored on the S/HNP Site that could pose a hazard to the control room operators have been identified and are listed on Table 2.2-6.

The chemicals listed on Table 2.2-6 were analyzed to determine the impacts of an accidental spill on the control room habitability using the methodology described in Section 2.2.3.1.4.1. Results of this analysis as shown on Table 2.2-7 indicate that no chemical stored on the S/HNP Site could pose a hazard to the control room operators.

2.2.3.1.5 Vapor Clouds

Flammable vapor clouds at the control room air intake are not a problem at the S/HNP Plant. Liquid petroleum gas (LPG), the probable source for flammable vapor clouds, has a low usage rate at the Hanford Reservation areas north of S/HNP. A small quantity of LPG is stored in the 200 West Area, and it is assumed that a small number of delivery vehicles pass by the Plant. There is no LPG stored within 5 miles of the S/HNP. No other potential sources of flammable vapor clouds have been located.

2.2.3.1.6 Ground Fires

Range fires occur in the desert surrounding the S/HNP Site. There was an average of 12.2 fires per year during a recent

10 year period, and the median fire covered an area of 6 acres. Every 3.3 years there is a fire greater than 1000 acres (Ref 5). The brush-type fires present no problem to the safety-related concrete structures. Landscaping around the Plant provides a firebreak. A perimeter fence and road serve the same purpose. Smoke induction to the control room is prevented by smoke detectors and automatic isolation of the ventilation system.

2.2.3.2 Effects of Design Basis Events

23

Based on the information given in the preceding sections for the S/HNP and the evaluations described in Section 2.2.3.1, explosions (Section 2.2.3.1.1), barge traffic accidents (Section 2.2.3.1.2), water contamination (2.2.3.1.3), flammable vapor clouds (Section 2.2.3.1.5) and ground fires (Section 2.2.3.1.6) do not constitute hazards to the Plant and are therefore not considered as design basis events.

Accidents involving transportation of anhydrous ammonia near the S/HNP Site, as described in Section 2.2.3.1.4.2, could pose hazards to the control room operators and are therefore considered as design basis events. Control room habitability requirements for this chemical are discussed in Section 6.4.

2.3 METEOROLOGY

Because of the proximity of the S/HNP Site to the Supply System WNP 1, 2 and 4 units and absence of significant terrain differences, the WNP-2 FSAR Amendments 1 to 16, Section 2.3, Docket No. 50-397, meteorological data have been determined to be applicable to the S/HNP Site and are incorporated herein by reference. The data were determined applicable for the following reasons:

Location

The S/HNP Site is located 4.5 miles WNW of the WNP-2 meteorological tower.

Elevation

The S/HNP Site is approximately 527 ft MSL in elevation.

The WNP-2 meteorological tower elevations are 441 ft MSL base and 686 ft MSL top.

The WNP-2 meteorological instrument elevations are:

Surface	-	precipitation 23
473 ft MSL	-	wind direction and speed, dry bulb and dew point temperatures
685 ft MSL	-	wind direction and speed, dry bulb and dew point temperatures

Temperature difference is obtained from 685-473 ft.

Topography

Terrain between the S/HNP Site and the WNP-2 meteorological tower is composed of gently rolling, sandy hillocks. The S/HNP Site is located approximately 4.5 miles WNW from the west boundary of the Supply System WNP-2 site with about 10 miles of similar, continuous terrain to the north, west, and southwest. The Columbia River is about 8 miles from the S/HNP Site. The White Bluffs rise 200-400 feet above the east bank of the Columbia River to a maximum height of 920 ft MSL.

Figure 2.3-1 shows the detailed topographic features within a 5-mile radius of the S/HNP. The topographic cross sections plotted by sector from the S/HNP Site are shown in Figure 2.3-2.

Vegetation:

The intervening vegetation between the S/HNP and WNP-2 tower sites is chiefly composed of 3-6 foot high scattered sagebrush and cheatgrass providing a lower groundcover. Growth occurs mainly in the fall and spring as moisture availability increases. 23

Tower Location

The WNP-2 meteorological tower is located about 2,500 ft west of the WNP-2 facility. The terrain around the tower is flat and vegetated with sagebrush and cheatgrass. The land around the tower is undeveloped with the exception of a single lane, unpaved access road to the WNP-2 plant site.

Analysis of Possible Differences in Meteorological Data if Collected at S/HNP Site

There are no significant terrain, soil or vegetation differences, airflow patterns, advection paths, stability conditions, local moisture sources, dust, or population centers which would affect local meteorology or its extremes at the S/HNP Site differently than at the WNP-2 site. Diffusion of effluents would continue to be dominated by regional topographic features. Therefore, the use of the WNP-2 meteorological data at the S/HNP Site is justifiable.

Present Use of WNP-2 Meteorological Data

Data are currently being used to support licensing activities of WNP 1, 2 and 4 plants and the requirements of the DOE-FFTF.

Quality Assurance Considerations

All tower data used by S/HNP for licensing/permitting actions are WNP-2 data.

2.3.1 REGIONAL CLIMATOLOGY

See Supply System WNP-2 Section 2.3.1 for regional climatology, except 2.3.1.2.3. The meteorological data used for evaluating the performance of the Ultimate Heat Sink with respect to (1) maximum evaporation and drift loss and (2) minimum water cooling are discussed in Section 9.2.5, Ultimte Heat Sink Complex. 24

Although the data for this description are for the period up to 1970, the data up through 1980 (Ref 1) were examined and it has been concluded that no modifications to the description are required.

TABLE 2.3-1

ANNUAL JOINT FREQUENCY DISTRIBUTIONS*

Stability Class A

24

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	0
0.75- 3.49	2	1	0	0	0	0	1	0	0	0	0	0	0	1	1	2	8
3.50- 7.49	22	17	8	7	7	3	16	15	9	11	12	4	12	9	15	26	193
7.50-12.49	20	20	12	8	1	3	7	13	52	8	6	5	10	9	20	19	213
12.50-18.49	7	5	1	0	0	0	2	4	21	21	8	9	9	7	13	6	113
18.50-23.99	0	0	0	0	0	0	0	0	6	0	6	6	12	8	20	1	59
>23.99	1	0	0	0	0	0	0	0	0	0	1	5	6	5	9	1	28
Total	52	43	21	15	8	6	26	32	88	40	33	29	49	39	78	55	614

Stability Class B

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Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	0
0.75- 3.49	2	3	0	2	1	0	1	0	2	3	1	1	3	0	2	0	21
3.50- 7.49	41	25	17	7	6	21	20	30	21	22	25	15	29	21	21	38	359
7.50-12.49	17	15	6	2	4	11	9	12	68	19	12	13	10	3	8	14	223
12.50-18.49	11	4	0	0	0	0	0	1	14	17	12	11	11	12	9	6	108
18.50-23.99	0	0	0	0	0	0	0	0	1	4	8	7	6	3	10	2	41
>23.99	0	0	0	0	0	0	0	0	0	0	2	1	2	2	6	0	13
Total	71	47	23	11	11	32	30	43	106	65	60	48	61	41	56	60	765

Stability Class C

24

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	0
0.75- 3.49	5	5	3	1	2	3	1	1	5	1	3	4	1	5	10	10	60
3.50- 7.49	58	36	15	13	6	12	31	53	49	31	22	22	21	18	29	43	459
7.50-12.49	36	27	12	14	6	6	11	26	68	49	16	14	22	12	12	16	347
12.50-18.49	13	7	2	0	0	0	1	2	8	25	26	16	14	6	13	9	142
18.50-23.99	1	0	0	0	0	0	0	0	3	5	7	8	7	3	7	1	42
>23.99	0	0	0	0	0	0	0	0	0	1	5	3	4	2	6	0	21
Total	113	75	32	28	14	21	44	82	133	112	79	67	69	46	77	79	1071

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Stability Class D

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	4
0.75- 3.49	62	37	32	29	32	52	53	58	39	55	58	41	43	66	68	78	803
3.50- 7.49	99	97	46	65	50	59	122	169	148	119	71	58	62	91	161	159	1576
7.50-12.49	74	48	19	22	15	8	27	75	165	183	74	56	53	78	106	82	1085
12.50-18.49	23	12	4	0	0	0	8	15	70	154	103	50	48	75	81	24	667
18.50-23.99	0	1	2	0	0	0	1	0	6	39	47	25	13	50	47	4	235
>23.99	0	2	0	0	0	0	0	0	0	18	30	9	8	11	25	0	109
Total	258	203	103	116	97	119	211	317	428	568	383	239	227	371	488	347	4479

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Stability Class E

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	12
0.75- 3.49	67	45	35	28	37	48	56	53	66	50	50	55	47	90	88	78	893
3.50- 7.49	68	32	53	52	37	36	83	156	141	108	100	98	100	169	238	156	1627
7.50-12.49	25	13	9	10	4	8	45	147	140	102	66	69	77	216	220	86	1237
12.50-18.49	1	4	1	1	0	0	11	33	71	127	82	26	26	126	86	13	608
18.50-23.99	0	7	0	0	0	0	1	3	7	52	28	11	8	25	12	0	154
>23.99	0	13	0	0	0	0	0	0	1	26	14	6	2	2	0	0	64
Total	161	114	98	91	78	92	196	392	426	465	340	265	260	628	644	333	4595

|24

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Stability Class F

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	5
0.75- 3.49	68	56	35	37	32	31	32	30	41	54	39	53	34	66	62	72	742
3.50- 7.49	65	64	55	31	14	24	71	157	144	113	64	54	54	86	136	127	1259
7.50-12.49	7	8	3	10	1	0	17	98	96	64	20	16	43	80	66	16	545
12.50-18.49	0	0	3	0	0	0	0	7	14	19	6	0	2	1	2	0	54
18.50-23.99	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
>23.99	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Total	140	128	96	78	47	55	120	292	295	250	129	123	133	233	266	215	2605

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Stability Class G

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	11
0.75- 3.49	108	97	100	67	71	54	41	42	44	35	44	37	36	52	81	108	1017
3.50- 7.49	87	65	67	37	8	13	47	122	125	56	31	22	25	44	119	119	987
7.50-12.49	2	1	3	3	0	0	2	42	71	16	4	8	^	9	27	9	201
12.50-18.49	0	0	0	0	0	0	0	0	8	3	0	0	[illegible]	0	0	0	11
18.50-23.99	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
>23.99	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Total	197	163	170	107	79	67	90	206	248	110	79	67	65	105	227	236	2227

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Stability Class All

Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
Calm																	32
0.75- 3.49	314	244	205	164	175	188	185	184	197	198	195	191	164	280	312	348	3544
3.50- 7.49	440	336	261	212	128	168	390	702	637	460	325	273	303	438	719	668	6460
7.50-12.49	181	132	64	69	31	36	118	413	660	441	198	181	219	407	459	242	3851
12.50-18.49	55	32	11	1	0	0	22	62	206	366	237	112	110	227	204	58	1703
18.50-23.99	1	8	2	0	0	0	2	3	23	100	96	57	46	89	96	8	531
>23.99	1	21	0	0	0	0	0	0	1	45	52	24	22	22	46	1	235
Total	992	773	543	446	334	392	717	1364	1724	1610	1103	838	864	1463	1836	1325	16356

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Annual
Percentage Occurrence by Stability Classes

A	B	C	D	E	F	G
3.75	4.68	6.55	27.38	28.09	15.93	13.62

Annual
Distribution of Wind Direction Vs Stability

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
A	52	43	21	15	8	6	26	32	88	40	33	29	49	39	78	55	0
B	71	47	23	11	11	32	30	43	106	65	60	48	61	41	56	60	0
C	113	75	32	28	14	21	44	82	133	112	79	67	69	46	77	79	0
D	258	203	103	116	97	119	211	317	428	568	383	239	227	371	488	347	4
E	161	114	98	91	78	92	196	392	426	465	340	265	260	628	644	333	12
F	140	128	96	78	47	55	120	292	295	250	129	123	133	233	266	215	5
G	197	163	170	107	79	67	90	206	248	110	79	67	65	105	227	236	11
Total	992	773	543	446	334	392	717	1364	1724	1610	1103	838	864	1463	1836	1325	32

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Annual
Distribution of Variable Winds
(These Values Are Not Included in the Previous JFD's)

Speed (mph)	A	B	C	D	E	F	G	TOTAL
Calm	0	0	0	0	0	0	0	0
0.75-3.49	1	5	15	48	30	41	43	183
3.50- 7.49	22	31	32	42	19	11	8	165
7.50-12.49	3	1	3	3	3	0	0	13
12.50-18.49	0	0	0	0	1	0	0	1
18.50-23.99	0	0	0	0	0	0	0	0
23.99	0	0	0	0	0	0	0	0
Total	26	37	50	93	53	52	51	362

*Based on WNP-2 data with the 33-ft wind and delta T (245-33 ft) stability.
Total Number of Observations: 17544
Total Number of Valid Observations: 16356
Total Number of Missing Observations: 1188
Percent Data Recovery for this Period: 93.2 %
Mean Wind Speed for this Period: 7.5 mph
Total Number of Observations with Backup Data: 0
Wind Measured At: 33.0 Feet
Wind Threshold at: 0.75 mph

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TABLE 2.3-2 23

CONSERVATIVE χ/Q VALUES FOR SHORT-TERM (ACCIDENT) ASSESSMENT AT S/HNP

Accident Period	Distance (m)	Maximum Sector χ/Q (sec/m^3)	
2 hours	3058(EAB)	8.7E-5(SSE)	
8 hours	6437(LPZ)	2.1E-5(SSE)	24
16 hours	6437(LPZ)	1.4E-5(SSE)	
72 hours (3 days)	6437(LPZ)	5.7E-6(SSE)	
624 hours (26 days)	6437(LPZ)	1.6E-6(SSE)	

Notes:

1. Relative concentrations are for a ground-level release to a ground-level receptor including credit for plume meander and building wake effects.

2. Based on WNP-2 meteorological data for the period April 1, 1974, to March 31, 1976: 33-ft wind and delta T (245-33 ft).

TABLE 2.3-3 Sheet 1 of 2

ANNUAL AVERAGE ATMOSPHERIC DISPERSION AND DEPOSITION PARAMETERS FOR S/HNP

Site Boundary: Unit 1

Dir	Distance (meters)	Chi/Q (sec/m3)	Chi/Q Decayed (sec/m3)	Chi/Q Decayed, Depleted (sec/m3)	D/Q (m^{-2})
N	1150.	1.043E-05	1.040E-05	9.308E-06	4.571E-08
NNE	1175.	8.661E-06	8.632E-06	7.722E-06	4.112E-08
NE	1095.	7.276E-06	7.252E-06	6.513E-06	3.177E-08
ENE	930.	9.820E-06	9.780E-06	8.876E-06	3.185E-08
E	910.	8.727E-06	8.699E-06	7.900E-06	3.383E-08
ESE	930.	1.504E-05	1.499E-05	1.360E-05	5.545E-08
SE	1095.	1.400E-05	1.396E-05	1.253E-05	5.311E-08
SSE	1290.	1.012E-05	1.007E-05	8.970E-06	2.780E-08
S	1265.	8.321E-06	8.281E-06	7.385E-06	2.202E-08
SSW	1290.	6.341E-06	6.310E-06	5.621E-06	1.626E-08
SW	1325.	4.941E-06	4.918E-06	4.373E-06	1.061E-08
WSW	1125.	5.499E-06	5.474E-06	4.913E-06	1.242E-08
W	1100.	4.439E-06	4.423E-06	3.972E-06	9.598E-09
WNW	1120.	5.175E-06	5.148E-06	4.624E-06	1.106E-08
NW	1325.	4.921E-06	4.896E-06	4.355E-06	1.397E-08
NNW	1175.	9.362E-06	9.334E-06	8.348E-06	3.502E-08

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NOTES:

1. Relative concentrations are for a ground-level release to a ground-level receptor, are undepleted and undecayed, and incorporate Pasquill-Gifford dispersion coefficients, building height wake, and open terrain correction factors.

2. Based on WNP-2 meteorological data for the period April 1, 1974 to March 31, 1976: 33-ft wind and delta T (245-33 ft).

3. Distances are from the center of each Containment.

8.000 Day Decay, Depleted
Corrected for Open Terrain Recirculation

Distance in Miles

Sector	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	5.223E-05	1.602E-05	8.222E-06	4.141E-06	1.627E-06	8.658E-07	5.408E-07	3.725E-07	2.741E-07	2.114E-07	1.689E-07
SSW	4.217E-05	1.278E-05	6.557E-06	3.318E-06	1.312E-06	7.011E-07	4.392E-07	3.033E-07	2.236E-07	1.727E-07	1.382E-07
SW	3.526E-05	1.062E-05	5.430E-06	2.766E-06	1.103E-06	5.923E-07	3.725E-07	2.580E-07	1.907E-07	1.476E-07	1.183E-07
WSW	2.826E-05	8.623E-06	4.433E-06	2.247E-06	8.895E-07	4.758E-07	2.982E-07	2.060E-07	1.519E-07	1.174E-07	9.391E-08
W	2.210E-05	6.760E-06	3.467E-06	1.756E-06	6.948E-07	3.715E-07	2.328E-07	1.608E-07	1.186E-07	9.160E-08	7.329E-08
WNW	2.561E-05	8.027E-06	4.132E-06	2.070E-06	8.079E-07	4.280E-07	2.663E-07	1.829E-07	1.343E-07	1.033E-07	8.240E-08
NW	3.335E-05	1.054E-05	5.442E-06	2.720E-06	1.058E-06	5.593E-07	3.476E-07	2.385E-07	1.748E-07	1.345E-07	1.071E-07
NNW	4.859E-05	1.545E-05	8.016E-06	4.004E-06	1.557E-06	8.226E-07	5.111E-07	3.506E-07	2.571E-07	1.977E-07	1.576E-07
N	5.309E-05	1.673E-05	8.651E-06	4.325E-06	1.683E-06	8.904E-07	5.537E-07	3.801E-07	2.789E-07	2.146E-07	1.711E-07
NNE	4.411E-05	1.422E-05	7.410E-06	3.676E-06	1.415E-06	7.430E-07	4.593E-07	3.138E-07	2.293E-07	1.758E-07	1.397E-07
NE	3.413E-05	1.086E-05	5.612E-06	2.794E-06	1.082E-06	5.701E-07	3.534E-07	2.420E-07	1.772E-07	1.361E-07	1.083E-07
ENE	3.502E-05	1.123E-05	5.856E-06	2.919E-06	1.132E-06	5.968E-07	3.702E-07	2.536E-07	1.857E-07	1.427E-07	1.136E-07
E	3.020E-05	9.664E-06	5.016E-06	2.496E-06	9.656E-07	5.087E-07	3.153E-07	2.159E-07	1.580E-07	1.214E-07	9.660E-08
ESE	5.309E-05	1.721E-05	8.968E-06	4.466E-06	1.729E-06	9.113E-07	5.650E-07	3.869E-07	2.833E-07	2.175E-07	1.731E-07
SE	6.545E-05	2.086E-05	1.081E-05	5.404E-06	2.102E-06	1.111E-06	6.907E-07	4.740E-07	3.476E-07	2.674E-07	2.131E-07
SSE	6.539E-05	2.031E-05	1.048E-05	5.271E-06	2.067E-06	1.099E-06	6.857E-07	4.720E-07	3.471E-07	2.675E-07	2.136E-07

Sector	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	1.387E-07	6.909E-08	4.341E-08	2.357E-08	1.521E-08	1.078E-08	8.098E-09	6.339E-09	5.112E-09	4.218E-09	3.543E-09
SSW	1.136E-07	5.687E-08	3.584E-08	1.954E-08	1.264E-08	8.972E-09	6.752E-09	5.291E-09	4.271E-09	3.526E-09	2.964E-09
SW	9.743E-08	4.902E-08	3.101E-08	1.700E-08	1.103E-08	7.850E-09	5.921E-09	4.649E-09	3.760E-09	3.109E-09	2.618E-09
WSW	7.721E-08	3.862E-08	2.432E-08	1.325E-08	8.555E-09	6.063E-09	4.557E-09	3.567E-09	2.876E-09	2.372E-09	1.992E-09
W	6.026E-08	3.014E-08	1.899E-08	1.035E-08	6.689E-09	4.744E-09	3.569E-09	2.796E-09	2.256E-09	1.862E-09	1.565E-09
WNW	6.755E-08	3.341E-08	2.088E-08	1.125E-08	7.214E-09	5.083E-09	3.803E-09	2.965E-09	2.382E-09	1.959E-09	1.640E-09
NW	8.777E-08	4.331E-08	2.702E-08	1.453E-08	9.304E-09	6.551E-09	4.898E-09	3.817E-09	3.066E-09	2.521E-09	2.111E-09
NNW	1.291E-07	6.375E-08	3.983E-08	2.148E-08	1.380E-08	9.747E-09	7.312E-09	5.718E-09	4.609E-09	3.802E-09	3.195E-09
N	1.403E-07	6.941E-08	4.343E-08	2.347E-08	1.510E-08	1.068E-08	8.022E-09	6.278E-09	5.063E-09	4.179E-09	3.513E-09
NNE	1.142E-07	5.590E-08	3.469E-08	1.853E-08	1.183E-08	8.308E-09	6.203E-09	4.831E-09	3.879E-09	3.189E-09	2.671E-09
NE	8.868E-08	4.363E-08	2.718E-08	1.459E-08	9.339E-09	6.576E-09	4.918E-09	3.834E-09	3.082E-09	2.535E-09	2.124E-09
ENE	9.296E-08	4.570E-08	2.844E-08	1.522E-08	9.714E-09	6.818E-09	5.083E-09	3.951E-09	3.166E-09	2.597E-09	2.169E-09
E	7.906E-08	3.889E-08	2.422E-08	1.299E-08	8.306E-09	5.843E-09	4.366E-09	3.401E-09	2.732E-09	2.246E-09	1.880E-09
ESE	1.417E-07	6.961E-08	4.330E-08	2.318E-08	1.479E-08	1.039E-08	7.750E-09	6.029E-09	.836E-09	3.971E-09	3.322E-09
SE	1.746E-07	8.623E-08	5.385E-08	2.902E-08	1.862E-08	1.314E-08	9.843E-09	7.687E-09	6.189E-09	5.099E-09	4.279E-09
SSE	1.753E-07	8.714E-08	5.465E-08	2.960E-08	1.906E-08	1.347E-08	1.011E-08	7.904E-09	6.367E-09	5.248E-09	4.405E-09

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Corrected for Open Terrain Recirculation

Distance in Miles

Sector	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	1.425E-07	4.819E-08	2.474E-08	1.176E-08	4.226E-09	2.096E-09	1.234E-09	8.079E-10	5.685E-10	4.213E-10	3.247E-10
SSW	1.108E-07	3.747E-08	1.924E-08	9.146E-09	3.285E-09	1.629E-09	9.594E-10	6.282E-10	4.420E-10	3.276E-10	2.524E-10
SW	7.756E-08	2.623E-08	1.347E-08	6.403E-09	2.300E-09	1.141E-09	6.716E-10	4.397E-10	3.094E-10	2.293E-10	1.767E-10
WSW	6.354E-08	2.149E-08	1.103E-08	5.245E-09	1.884E-09	9.343E-10	5.502E-10	3.602E-10	2.535E-10	1.879E-10	1.448E-10
W	4.730E-08	1.599E-08	8.212E-09	3.904E-09	1.402E-09	6.955E-10	4.095E-10	2.681E-10	1.887E-10	1.398E-10	1.078E-10
WNW	5.618E-08	1.900E-08	9.754E-09	4.637E-09	1.666E-09	8.260E-10	4.864E-10	3.185E-10	2.241E-10	1.661E-10	1.280E-10
NW	1.021E-07	3.453E-08	1.773E-08	8.430E-09	3.028E-09	1.502E-09	8.842E-10	5.790E-10	4.074E-10	3.019E-10	2.327E-10
NNW	1.928E-07	6.520E-08	3.348E-08	1.592E-08	5.717E-09	2.835E-09	1.669E-09	1.093E-09	7.692E-10	5.700E-10	4.393E-10
N	2.428E-07	8.209E-08	4.215E-08	2.004E-08	7.198E-09	3.570E-09	2.102E-09	1.376E-09	9.684E-10	7.177E-10	5.531E-10
NNE	2.264E-07	7.657E-08	3.931E-08	1.869E-08	6.714E-09	3.329E-09	1.960E-09	1.284E-09	9.033E-10	6.694E-10	5.159E-10
NE	1.554E-07	5.254E-08	2.698E-08	1.282E-08	4.607E-09	2.285E-09	1.345E-09	8.808E-10	6.198E-10	4.593E-10	3.540E-10
ENE	1.193E-07	4.034E-08	2.071E-08	9.848E-09	3.537E-09	1.754E-09	1.033E-09	6.764E-10	4.759E-10	3.527E-10	2.718E-10
E	1.223E-07	4.135E-08	2.123E-08	1.009E-08	3.625E-09	1.798E-09	1.059E-09	6.932E-10	4.877E-10	3.615E-10	2.786E-10
ESE	2.077E-07	7.023E-08	3.606E-08	1.714E-08	6.158E-09	3.054E-09	1.798E-09	1.177E-09	8.285E-10	6.140E-10	4.732E-10
SE	2.597E-07	8.784E-08	4.510E-08	2.144E-08	7.702E-09	3.819E-09	2.249E-09	1.473E-09	1.036E-09	7.679E-10	5.918E-10
SSE	1.895E-07	6.407E-08	3.289E-08	1.564E-08	5.617E-09	2.786E-09	1.640E-09	1.074E-09	7.558E-10	5.601E-10	4.316E-10

	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	2.579E-10	1.146E-10	6.941E-11	3.508E-11	2.123E-11	1.424E-11	1.020E-11	7.660E-12	5.956E-12	4.758E-12	3.883E-12
SSW	2.005E-10	8.909E-11	5.397E-11	2.728E-11	1.651E-11	1.107E-11	7.932E-12	5.956E-12	4.631E-12	3.699E-12	3.019E-12
SW	1.404E-10	6.236E-11	3.778E-11	1.909E-11	1.156E-11	7.749E-12	5.552E-12	4.169E-12	3.242E-12	2.589E-12	2.114E-12
WSW	1.150E-10	5.109E-11	3.095E-11	1.564E-11	9.468E-12	6.348E-12	4.549E-12	3.415E-12	2.656E-12	2.121E-12	1.731E-12
W	8.560E-11	3.803E-11	2.304E-11	1.164E-11	7.047E-12	4.725E-12	3.386E-12	2.542E-12	1.977E-12	1.579E-12	1.289E-12
WNW	1.017E-10	4.517E-11	2.736E-11	1.383E-11	8.370E-12	5.612E-12	4.021E-12	3.020E-12	2.348E-12	1.875E-12	1.531E-12
NW	1.848E-10	8.211E-11	4.974E-11	2.514E-11	1.522E-11	1.020E-11	7.310E-12	5.489E-12	4.268E-12	3.409E-12	2.783E-12
NNW	3.490E-10	1.550E-10	9.391E-11	4.747E-11	2.873E-11	1.926E-11	1.380E-11	1.036E-11	8.059E-12	6.437E-12	5.254E-12
N	4.394E-10	1.952E-10	1.182E-10	5.976E-11	3.617E-11	2.425E-11	1.738E-11	1.305E-11	1.015E-11	8.105E-12	6.615E-12
NNE	4.098E-10	1.821E-10	1.103E-10	5.574E-11	3.374E-11	2.262E-11	1.621E 11	1.217E-11	9.463E-12	7.559E-12	6.170E-12
NE	2.812E-10	1.249E-10	7.567E-11	3.825E-11	2.315E-11	1.552E-11	1.112E-11	8.351E-12	6.493E-12	5.187E-12	4.234E-12
ENE	2.159E-10	9.592E-11	5.811E-11	2.937E-11	1.778E-11	1.192E-11	8.540E-12	6.413E-12	4.986E-12	3.983E-12	3.251E-12
E	2.213E-10	9.831E-11	5.955E-11	3.010E-11	1.822E-11	1.221E-11	8.752E-12	6.572E-12	5.110E-12	4.082E-12	3.332E-12
ESE	3.759E-10	1.670E-10	1.012E-10	5.113E-11	3.095E-11	2.075E-11	1.487E-11	1.116E-11	8.680E-12	6.934E-12	5.660E-12
SE	4.701E-10	2.088E-10	1.265E-10	6.394E-11	3.870E-11	2.595E-11	1.859E-11	1.396E-11	1.086E-11	8.672E-12	7.078E-12
SSE	3.429E-10	1.523E-10	9.227E-11	4.664E-11	2.823E-11	1.893E-11	1.356E-11	1.018E-11	7.918E-12	6.325E-12	5.162E-12

*33 ft winds, delta T (254'-33'). Data Period 4/74-3/76.

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24- and 48-hour general storm PMPs of 9.0 and 11.7 in., respectively. 23

The thunderstorm PMP, which is considerably more severe than the general storm PMP was used to determine the PMF hydrograph. The critically arranged precipitation for the 1- and 6-hour PMPs are listed in Tables 2.4-14 and 2.4-15, respectively. The Bureau of Reclamation(Ref 29) critical sequence was used for the 1-hour thunderstorm PMP and the U.S. Army Corps of Engineers(Ref 32) critical sequence was used for the 6-hour thunderstorm PMP.

2.4.3.2.2 Precipitation Losses

The thunderstorm precipitation losses were estimated by the method developed by the Soil Conservation Service.(Ref 29) The method involves the determination of the direct runoff curve number (CN) from the hydrologic soil group, cover type, and cover density. For thunderstorm precipitation, the type and density of vegetative cover on the watershed, as well as the time sequence of precipitation have more influence on thunderstorm runoff than does the antecedent moisture condition or the soil group.

The soil within the S/HNP watershed is classified as Hydrologic Soil Group B, and the area has a thunderstorm cover index of brush, sage, grass, or a combination thereof.(Ref 13) Based on these data, a curve number (CN) of 85 was estimated for the watershed. When this CN was used in the runoff model described in Section 2.4.3.2.4, total losses for the 1- and 6-hour thunderstorm PMFs were 1.7 and 1.9 in., respectively. Because these losses are small compared to the precipitation input, for the calculations of the PMP hydrograph, it was conservatively assumed that the basin was 100 percent impervious, that is, that there were no precipitation losses.

2.4.3.2.3 Runoff and Stream Course Model

The 5 minute unit hydrographs for the 1-hour and 6-hour 24
thunderstorm PMP were determined by the Soil Conservation Service(Ref 29) method for small, ungaged watersheds. The calculations were made with the HEC-1 computer program.(Ref 33)

The Skagit/Hanford Nuclear Project watershed has a drainage area of 1.3 square miles. The drainage area length is 1 mi. The elevation at the upstream boundary is 530 ft MSL and at

the basin exit it is 516 ft MSL. The time of concentration, Tc, is given by:

$$T_C = \left[\frac{11.9L^3}{H}\right]^{0.385} \qquad (2.4.3\text{-}1)$$

where:

L = drainage area length (mi.)
H = drainage area elevation difference (ft)

The Tc for the watershed is 0.94 hour. The corresponding lag time is approximately 0.6 Tc, or 0.6 hour. The basin hydrologic characteristics used in HEC-1 are shown in Table 2.4-16. The unit hydrograph used for the 1- and 6-hour thunderstorm PMPs are listed in Tables 2.4-17 and 2.4-18, respectively, and shown on Figures 2.4-14 and 2.4-15.

2.4.3.2.4 Probable Maximum Flood Flow

The PMF hydrograph for the 1- and 6-hour thunderstorm PMPs were determined from the incremental precipitation, the direct runoff curve number and the unit hydrograph.(Ref 29) The calculations were made with the HEC-1 computer code. (2.4-33) 23

Direct runoff increments were computed from the critically arranged incremental precipitation listed in Tables 2.4-14 and 2.4-15 using the following equations:

$$Q = \frac{(P-0.2S)^2}{(P+0.8S)} \qquad (2.4.3\text{-}2)$$

$$S = \frac{1000 - 10}{CN} \qquad (2.4.3\text{-}3)$$

where:

Q = direct runoff (in.)
P = incremental precipitation (in.)
CN = curve number = 85

Because the losses are small, it was conservatively assumed that the basin was 100 percent impervious. The direct runoff increments were convoluted with the 5-minute and 10-minute unit hydrographs to yield the 1-hour and 6-hour thunderstorm PMF peak discharge of 4,724 cfs and 3,240 cfs, respectively. The peak discharge of the 1-hour PMP was used to calculate the PMF water surface elevation because it was higher than the 6-hour PMP peak discharge. The hydrograph for the 1-hour PMP is shown in Table 2.4-19 and on Figure 2.4-16. The peak discharge occurs approximately 70 minutes after the start of rainfall. 23 24

2.4.3.2.5 Water Surface Elevations

The water surface elevations for the peak discharge of the 1-hour thunderstorm PMF were determined with the HEC-2 computer code.(Ref 31) It was assumed that the discharge was constant and had a peak value of 4,724 cfs.

The locations of the sections used in the backwater calculations are shown on Figure 2.4-17. The geometry of the sections is shown on Figure 2.4-18. The backwater calculations were started at Section 1 as it constricts the flow and acts as a natural control section. It was difficult to distinguish between channel and overbank areas for the natural sections downstream of the graded areas. Consequently, a single Manning's roughness coefficient was chosen to characterize the natural sections. The Manning's roughness coefficient for the natural sections was estimated (Ref 13) as 0.035. The final surface will consist of 6 inches of stabilizing gravel over the entire rough graded area. The Manning's roughness coefficient for the latter areas was also estimated as 0.035.(Ref 34) The water surface profile for 1-hour thunderstorm PMF is shown on Figure 2.4-19. The area inundated is shown on Figure 2.4-20. The maximum water surface elevation at section number 15, immediately to the north of the power block, is 524.2 ft MSL; and that at section number 17, immediately to the south of the power block, is 525.1 ft MSL. The water surface elevations and water depths are shown in Table 2.4-20.

2.4.3.2.6 Coincident Wind Wave Activity

Wind wave effects coincident with the thunderstorm PMF were evaluated using the method developed by the U.S. Army Corps of Engineers.(Ref 35)

The 100-year 1-hour average windspeed for coincident wind wave activity calculations at the Skagit/Hanford Nuclear Project is 64 mph. It is the direction-independent, 33-feet above ground, overland windspeed. The windspeed estimate appears conservative relative to the maximum 1-hour average of 47 mph observed on the Hanford Meteorological Station Tower on January 11, 1972, and corrected to 33 feet with the 1/7-th power law wind profile. It is consistent with the fastest windspeed reported in Section 2.3.1. The overwater wind speed was estimated as 1.1 times the overland windspeed,(Ref 35) or 70 mph. 23

The stillwater area and the radials used to estimate the length of the effective fetch, F_e, are shown on Figure 2.4-20. Wind setup at the power block was estimated for winds from the northeast and the southeast. The latter produced the higher wind setup. At a 1-hr thunderstorm PMF elevation at the southeast corner of the power block of 525.1 ft MSL, the effective fetch length was estimated at 1300 ft (0.25 mi). The average water depth, d, along the central radial (Figure 2.4-21) was estimated at 2.3 ft.

The slope at the southeast corner of the power block is about 1:400.

The wind setup can be estimated from(Ref 35)

$$S = \frac{\Delta S}{2} = \frac{(1.165 \times 10^{-3})\, U^2 F \cos\theta}{2d} \qquad (2.4.3\text{-}4)$$

where:

- S = wind setup (ft) 24
- ΔS = difference in water level between the two ends of the fetch (ft)
- U = windspeed (mph)
- F = Fetch length (mi), assumed to equal $2F_e$.
- d = average water depth in fetch (ft)
- θ = angle between wind and fetch.

TABLE 2.4-4

HISTORICAL AVERAGE MONTHLY DISCHARGE[a]
(See Table 2.4-3 for Period of Record)

Month	Average Monthly Discharge (cfs) Columbia River below Priest Rapids Dam	Yakima River at Kiona	Snake River below Ice Harbor Dam
Oct	68,500	2,157	27,560
Nov	65,950	3,136	31,560
Dec	64,620	4,092	36,380
Jan	66,180	3,848	42,390
Feb	68,150	4,057	47,030
Mar	70,460	4,583	55,020
Apr	97,910	5,234	77,250
May	204,100	6,265	114,300
Jun	303,200	5,817	118,500
Jul	221,300	2,136	45,270
Aug	125,200	1,433	22,510
Sep	82,960	1,670	23,540

(a) Ref 14

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TABLE 2.4-5

REGULATED AVERAGE MONTHLY DISCHARGE[a]
(1980 LEVEL OF IRRIGATION DEPLETION AND DAM REGULATION)

23

Month	Regulated Average Monthly Discharge (cfs) Columbia River below Priest Rapids Dam	Yakima River at Kiona	Snake River below Ice Harbor Dam
Oct	83,671	2,154	26,645
Nov	87,562	3,280	33,582
Dec	114,180	3,619	35,071
Jan	190,952	3,256	41,888
Feb	140,581	3,909	42,853
Mar	109,839	3,747	48,467
Apr	109,501	3,043	70,227
May	169,160	4,556	94,690
Jun	134,613	4,435	88,965
Jul	92,394	1,199	34,901
Aug	96,584	1,023	20,383
Sept	77,773	1,286	20,568

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(a) Ref 19

TABLE 2.4-18

10 MINUTE UNIT HYDROGRAPH* 23

Time (min)	Discharge (cfs)	Time (min)	Discharge (cfs)
0	0	110	87
10	129	120	85
20	415	130	39
30	788	140	26
40	919	150	18
50	844	160	12
60	660	170	9
70	419	180	6
80	280	190	3
90	192	200	2
100	128	210	1
	220	0	

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*Used for S/HNP watershed for 6-hour thunderstorm PMP Convolution

TABLE 2.4-19

HYDROGRAPH FOR 1-HOUR THUNDERSTORM PMP 23

Time (min)	Discharge (cfs)	Time (min)	Discharge (cfs)	
0	0	125	860	
5	17	130	696	
10	68	135	564	24
15	167	140	457	
20	342	145	369	
25	606	150	298	
30	943	155	241	
35	1,358	160	196	
40	1,851	165	159	
45	2,420	170	129	
50	3,035	175	105	
55	3,656	180	85	
60	4,205	185	69	
65	4,579	190	44	
70	4,724	195	35	
75	4,652	200	27	
80	4,395	205	21	
85	4,000	210	15	
90	3,507	215	10	
95	2,973	220	6	
100	2,454	225	3	
105	2,000	230	1	
110	1,627			
115	1,315			
120	1,063			

TABLE 2.4-20

WATER SURFACE ELEVATIONS AND WATER DEPTHS FOR THE 1-HOUR THUNDERSTORM PMP ALONG SECTION AA

Section Number	Distance (ft)	Elevation (ft MSL)	Depth (ft)
1	0	520.1	2.5
2	200	520.7	1.9
3	400	520.9	2.9
4	600	521.1	3.1
5	800	521.3	3.3
6	1000	521.4	3.4
7	1200	521.4	1.4
8	1450	521.7	1.7
9	1850	522.2	2.2
10	1960	522.3	2.3
11	2250	522.8	1.8
12	2450	523.2	2.2
13	2650	523.7	2.7
14	2850	524.0	3.0
15	3050	524.2	2.2
16	3300	524.6	2.6
17	3730	525.1	2.1
18	3950	525.3	2.3
19	4090	525.3	2.3
20	4220	525.3	2.3
21	4320	525.3	2.3
22	4670	525.6	2.6
23	5020	525.7	2.7

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TABLE 2.4-21

LOW FLOW DATA FOR COLUMBIA RIVER BELOW PRIEST RAPIDS DAM[a]

Water Year	Minimum Average Daily Flow		Minimum Instantaneous Flow	
	Discharge (cfs)	Date	Discharge (cfs)	Date
1960	52,100	03/12/60	40,100	03/15/60
1961	44,100	11/09/60	37,200	09/25/61
1962	36,400	01/01/62	35,300	01/01/62
1963	51,100	09/07/63	37,300	03/18/63
1964	38,700	09/20/64	36,900	12/15/63-12/16/63
1965	38,100	12/13/64	34,800	12/31/64-01/01/65
1966	37,600	04/03/66	36,300	01/17/66
1967	38,000	10/16/66	36,000	11/15/66
1968	38,200	02/24/67	35,600	04/29/68
1969	38,500	09/07/69	36,100	10/16/68
1970	38,400	02/08/70	36,100	10/04/69
1971	37,100	01/23/71	34,800	12/25/70
1972	45,900	11/20/71	36,100	01/09/71
1973	38,000	12/25/72	35,800	12/26/72
1974	37,800	11/11/73	36,400	11/30/73
1975	43,500	08/16/75	38,300	11/24/74
1976	39,300	04/10/76	32,100	04/11/76
1977	38,400	07/10/77	35,500	07/02/77
1978	39,000	12/25/77	28,300	11/09/77
1979	50,500	08/19/79	34,800	08/17/79

(a) Ref 55

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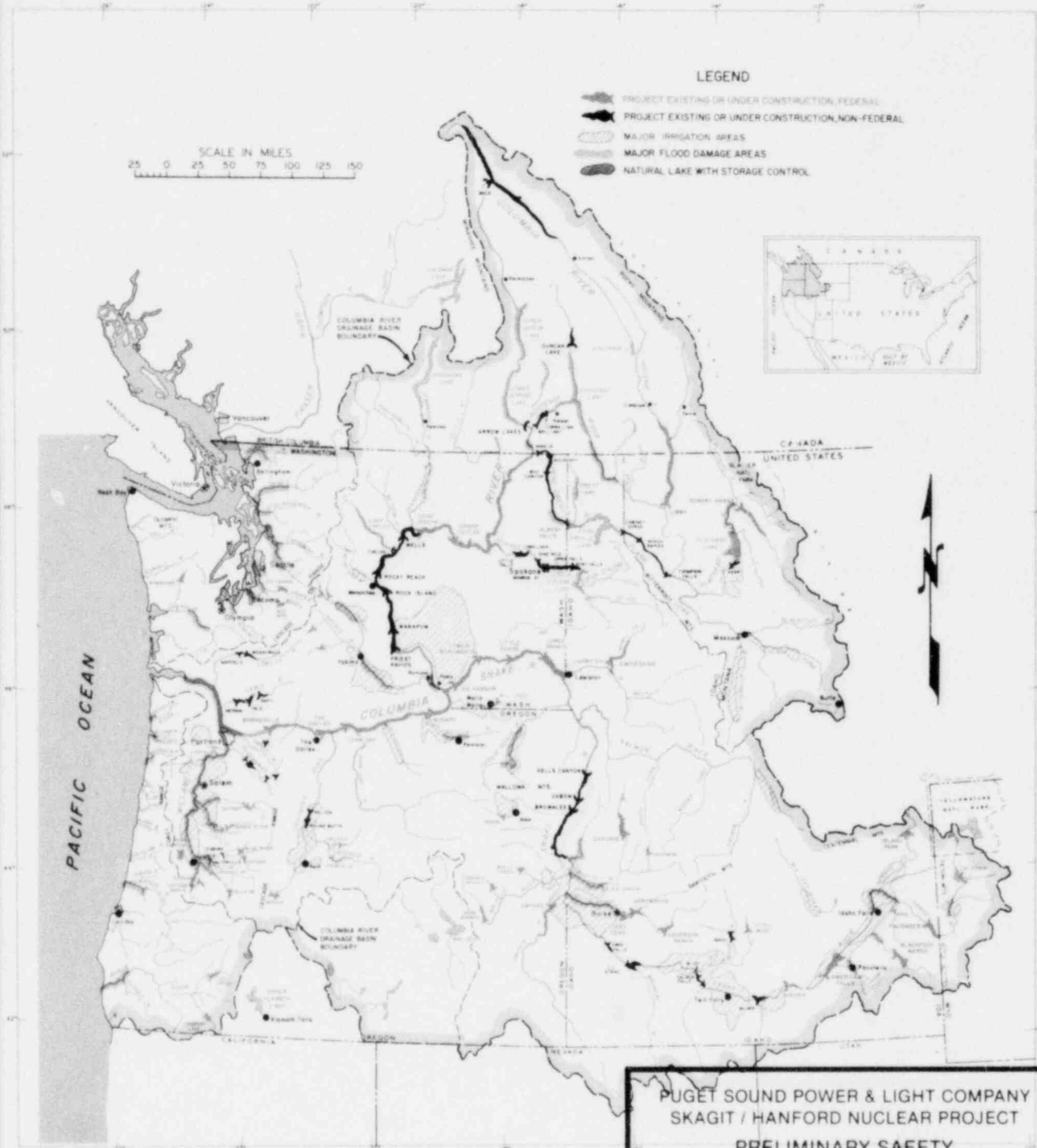
COLUMBIA RIVER AND COASTAL BASINS



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SKAGIT / HANFORD NUCLEAR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

COLUMBIA RIVER DRAINAGE BASIN AND MAJOR DAMS

FIGURE 2.4-5

2.5.2.1 Seismicity 23

Reference Section 2.5.2.1, WNP-2 FSAR.

2.5.2.2 Geologic Structures and Tectonic Activity

Reference Section 2.5.2.2, WNP-2 FSAR.

2.5.2.3 Correlation of Earthquake Activity with Geologic Structure or Tectonic Provinces

Reference Section 2.5.2.3, WNP-2 FSAR.

2.5.2.4 Maximum Earthquake Potential

Reference Section 2.5.2.4, WNP-2 FSAR.

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

In-situ velocity measurements by crosshole and downhole techniques and surface refraction studies have been conducted in the vicinity of the S/HNP Site (Appendix 2L). The velocity column beneath the S/HNP Site from ground surface to the Elephant Mountain Basalt is shown on Figure 2.5A. The velocity column has been compiled from crosshole velocity measurements at the west reactor site to a depth of approximately 200 ft, from downhole velocity measurements to a depth of 570 ft in borehole S-15, and from surface refraction data for the basalts. Velocity values between a depth of 570 ft and the top of basalt at a depth of 704 ft have been estimated from downhole velocity measurements at other locations as correlated with the coarse and fine materials indicated by the geophysical logs. The coarse materials within the Ringold and lower pre-Missoula section are generally cemented, producing the high velocity (8,000 fps or greater) layers shown on Figure 2.5-10. Shear wave velocities within the Ringold section below a depth of 230 ft are estimated based on velocity measurements at other locations in the Hanford Reservation area. Compressional and shear wave velocities of the materials above basalt have also been measured at WNP-1/4 and WNP-2 (Appendix 2L of the WNP-1/4 PSAR). The 24

changes in seismic wave velocities with depth at WNP-1/4 and WNP-2 are similar to those described above for the S/HNP.

The sonic velocities of the basalt flows and interbeds below the Elephant Mountain Basalt have been measured in the Rattlesnake Hills No. 1 well (Ref 16) to a depth of 3,230 m (10,600 ft). The sonic log from this well shows that the compressional wave velocity varies. Relatively high velocities of 5.0 to 5.7 km/s (16,400 to 18,700 fps) were measured for the competent basalt flows. Lower velocities of 4.0 to 4.5 km/s (13,000 to 14,800 fps) were measured for the interbeds. Shear wave velocities were not measured.

2.5.2.6 Safe Shutdown Earthquake

The maximum acceleration at the Site resulting from historical or instrumental earthquakes is estimated to have been 0.015g (see Section 2.5.2.6 WNP-2 FSAR).

A peak acceleration of 0.25g at ground surface is an appropriate "zero-period" anchoring point for a Reg. Guide 1.60 response spectra to define the Safe Shutdown Earthquake (SSE). This spectrum is consistent with the design criteria previously adopted for the Hanford Reservation (Atomic Energy Commission, 1972) and with accelerations associated with an intensity (MM) VIII earthquake. This intensity is larger than any historical earthquake in the Columbia Plateau. 23

This spectrum anchored at 0.25g is herein referred to as the Site SSE. It must also be recognized that many structures and components of the Plant had previously been designed for an SSE anchored at 0.35g associated with the Skagit Site. Consequently, this 0.35g spectrum has been retained as the Plant design basis and is herein referred to as the Design SSE. The requirements of this design basis exceed all of those of the Site SSE. Unless otherwise specified, references to the SSE in Chapter 2 of this SAR refer to the Site SSE and references to the SSE in Chapters 3 through 15 refer to the Design SSE.

2.5.2.6.1 Evaluation of the Probability of Exceedance of the Vibratory Ground Motion of the SSE

A seismic exposure analysis to estimate the probability of exceeding the vibratory ground motions of the Site SSE is presented in the WNP-2 FSAR.

2.5.2.7 Operating Basis Earthquake

A peak acceleration of 0.125g or one-half that of the Site SSE is assigned for the Site Operating Basis Earthquake (Site OBE). A peak acceleration of 0.175g or one-half that of the Design SSE is assigned for the Design Operating Basis Earthquake (Design OBE). Unless otherwise specified, references to the OBE in Chapter 2 of this SAR refer to the Site OBE and references in Chapters 3 through 15 refer to the Design OBE.

2.5.3 SURFACE FAULTING

All available geologic and geophysical information was evaluated to determine whether any evidence suggested that surface faulting might occur within 5 miles of the Site. Available information was supplemented with detailed, site-specific geologic and geophysical surveys extending beyond 5 miles in some directions and concentrated in a 2 mile radius of the Site. These surveys have included ground gravity and magnetic surveys along closely spaced lines and a seismic refraction survey. The results of the geophysical investigations are described in Appendices 2K and 2L. Geologic investigations undertaken specifically to supplement available information included photogeology, field mapping, rotary and core drilling, and stratigraphic analysis. The results of these investigations are described 23
in Section 2.5.1.2 and Appendix 2R.

Geologic and geophysical studies have shown that the basalt bedrock underlying the Site within a radius of at least 2 miles shows slopes with only gentle relief (generally less than 5 degrees). Sedimentary units within the Miocene-Pliocene Ringold Formation which overlies bedrock are generally horizontal or show some minor warping (slopes less than 5 degrees). Sediments overlying the uppermost Ringold Formation (generally considered to be part of the Hanford Formation) are Pleistocene or older in age and contain a refracting horizon of 8,000 ft/sec velocity which is flat-lying within a radius of 2 miles of the Site. This velocity horizon has also been found to be flat-lying over an area of approximately 28 square miles within the vicinity of the Site. There are no photolinears within the Site Area which are structurally controlled. On the basis of these data, there is no evidence that suggests potential for surface faulting; therefore, Sections 2.5.3.1 through 2.5.3.8 do not apply.

2.5.4 STABILITY OF SUBSURFACE MATERIALS

Beneath a surficial layer of loose silty sand, the central 23
Plant facilities are underlain by geologic strata which will
provide suitable founding materials. The loose surficial
sands have an average thickness of 6 to 8 feet across the
Site (mean approximate surface elevation 525 feet). These
sands are underlain by medium dense to dense sands of late
Pleistocene age to approximate elevation 490 feet (MSL),
very dense sands and gravels of late Pliocene (?) to
Pleistocene age to approximate elevation 320 feet, and
lacustrine and fluvial very dense sands and gravels and hard
silts and clayey silts of late Miocene to Pliocene age
(Ringold Formation) through to basalt bedrock at approximate
elevation -200 feet. The present groundwater table is at
elevation 400 feet.

A majority of the central Plant structures, with the excep-
tion of the Ultimate Heat Sinks and Radwaste Buildings which
will be founded directly on the very dense sands below Eleva- 24
tion 490', will be supported directly on structural backfill
(see Figure 2.5-15).

Because of the use of large mat foundations and the nature
of the foundation materials, there is no possibility of
large scale movements associated with bearing capacity
failure. Structure permissible total and differential
settlements control the allowable bearing pressures.
Permanent settlements will occur under the static loads
applied by the surface structures and their equipment. 24
These settlements are not expected to cause adverse effects
on the structures and operating equipment.

During design basis earthquake shaking, the surface
structures will be subjected to dynamic pseudo-elastic
(recoverable) movements. The dynamic movements are not 24
expected to have adverse effects on the Plant and equipment
during the Design SSE.

Because of the great depth of the water table and the nature
of the geologic strata at the Site, there is no potential
for liquefaction of the structure foundations. In addition,
there are no other foundation conditions (e.g. zones of
alteration or weathering, collapse features, poorly
consolidated strata or soluble zones) which could impact
foundation stability.

2.5.4.1 Geologic Features

The geologic features at the Site are discussed in detail in Section 2.5.1.2.

2.5.4.1.1 Areas of Potential Surface or Subsurface Subsidence

There are no areas of actual or potential surface or subsurface subsidence, uplift, or collapse at the Site.

2.5.4.1.2 Previous Loading History

Several hundred feet of Ringold Formation sediments were eroded from the Site prior to or during Pleistocene time. Consolidation testing of silty units within the Ringold (Figures 2Q B-28 and 2Q B-29) has confirmed effective preload pressures of these units on the order of 8 to 10 tons per square foot greater than existing loads. There is no evidence that the glaciofluvial sediments subsequently deposited on the erosion surface of the Ringold Formation have been subjected to significant pre-loading.

2.5.4.1.3 Zones of Alteration, Weathering and Structural Weakness 23

There are no zones of alteration or irregular weathering, no zones of structural weakness, and no shears, joints, fractures, faults or folds which will influence structural foundations at the Site.

2.5.4.1.4 Unrelieved Residual Stresses in Bedrock

Basalt bedrock at the Site is at approximate depth 700 feet and unrelieved residual stresses in the basalt will not impact structural foundations.

2.5.4.1.5 Potentially Unstable Rocks or Soils

At the Site there are no rocks or soils which are potentially unstable due to lack of consolidation, high

water content, solubility or undesirable response to natural or induced site conditions. 23

With the exception of the loose surficial materials, the in-place soils are competent to provide sound foundations for the structures. However, the major structures will be founded on compacted structural backfill down to the very dense sands at approximately Elevation 490' (see Section 24 2.5.4.5).

The near surface Missoula sands (above average elevation 490 feet) could be expected to undergo some dynamic compaction settlements during the postulated design basis earthquake; these settlements are estimated in Appendix 2Q (Section 8.2.4.2) and their magnitudes are such that the central Plant structures and operating equipment are not expected to be adversely impacted. These materials will, however, be 24 removed and replaced with structural backfill. The very dense pre-Missoula sands and gravels (down to average elevation 360-380 feet) would be expected to undergo negligible settlements during the design basis earthquake. The lower gravels of this unit are below the water table (average elevation 400 feet) but the very dense nature of the gravels and the relative depth of the saturated zone ensure that significant excess pore pressure development and liquefaction would not occur as a result of Design SSE shaking (Appendix 2Q, Section 8.2.4.1).

Below average elevation 360-380 feet (MSL), the very dense lower Pre-Missoula gravels and the gravels, sands and silts of the Ringold Formation exhibit high in situ seismic velocities (Table 2.5-3). The gravels are cemented (see 2.5.1.2) and stiff (see Menard pressuremeter test results--Table 2Q A-4), and the silts are overconsolidated to very hard and compact materials (see consolidation test results and Menard pressuremeter test results--Figures 2Q B-28 and 2Q B-29 and Table 2Q A-4). There is no possibility of a loss of stability or of a significant deterioration of the engineering properties of the Site materials during motions associated with the Design SSE.

2.5.4.2 Properties of Subsurface Materials

Static and dynamic engineering properties of the subsurface materials underlying the Site were determined by field and laboratory investigations as described in Appendix 2Q (Sections 4.0 and 5.0). All testing was carried out in accordance with the applicable standards published by the American Society for Testing and Materials (ASTM) (Ref 17, 18) with the exception of laboratory dynamic tests (cyclic

taken within each of the borings, and Menard pressure-meter tests were performed in 12 of the holes. Four monitoring wells and four multiple completion standpipe piezometers were installed around the central plant facilities area for groundwater pressure monitoring and water sampling purposes. In addition, two cross-hole geophysical seismic surveys were carried out to determine compression and shear wave velocity profiles for the foundation soils. No unanticipated foundation conditions were encountered during the Site investigation program.

The field exploration program provided some of the data necessary to characterize the static and dynamic properties of the foundation materials and to permit an evaluation of foundation stability conditions to be undertaken. Details of the field exploration methods are given in Appendices 2P, 2L and 2Q (Section 4.0).

2.5.4.3.2 Location Plans and Logs of Explorations

Locations of the field explorations in relation to the central plant facilities structures are shown on Figure 2.5-12. Detailed logs of all boreholes are given in Appendix 2Q, on Figures 2Q A-1 through 2Q A-37. Trench logs and photographs are shown on Figures 2Q A-52 through 2Q A-54. The coordinates, collar elevations, depths and drilling methods for each of the boreholes are given in Table 2.5-4. 23

The log of the deep corehole at the Site is given in Appendix 2R, and the results of the Site cross-hole geophysical seismic surveys are presented in Appendix 2L and summarized in Table 2.5-3. Water level measurements within the monitoring wells and piezometric holes are presented in Table 2P-1.

2.5.4.3.3 Subsurface Soil Profiles

Subsurface soil profiles for the Site, showing boring locations, groundwater elevations and final foundation grades, are given in Figure 2.5-13.

2.5.4.4 Geophysical Surveys

Seismic profiles in the S/HNP Site Area, including compressional ("P") wave velocity values, are presented in

Appendix 2L. A contour map of the top of the 8,000-10,000 fps refracting horizon within the Pre-Missoula gravels, and 23 a contour map of the top of basalt based on seismic profiles as supported by gravity and test boring data, are also included in Appendix 2L.

Table 2.5-3 and Appendix 2L provide measured in situ compressional ("P") and shear ("S") wave velocity values, along with corresponding calculated elastic moduli values.

2.5.4.5 Excavations and Backfill

2.5.4.5.1 Extent of Excavations and Backfill

The upper six to eight feet of loose silty aeolian sand does 24 not provide suitable founding material and as a minimum will be removed beneath all major structures. A plan and profiles showing the extent of Category 1 excavations at the Site are presented in Figures 2.5-14 and 2.5-15. These drawings also show the required excavation slopes. Chemical or other types of slope protection materials will be applied as necessary on exposed excavated or fill slopes to prevent wind erosion. Mud mats will be provided to protect prepared foundation soils. Methods for controlling runoff from rainfall on the foundation mats and mud mats will be included in 24 the construction specification in order to protect foundation soils from erosion.

2.5.4.5.2 Sources of Backfill

The clean black medium sands, which occur beneath the upper silty materials and above the very dense Pre-Missoula sediments, will be used for Category 1 backfill. Much of the upper silty sand material will also be used, but only in area fills outside the limits of structures and roads.

2.5.4.5.3 Compaction Criteria

Structural backfill material will be compacted to an average relative density of 85 percent with a minimum relative density of 75 percent, determined in accordance with ASTM D 2049. Prior to the start of backfilling, a test program will be carried out to provide a correlation between relative density and relative compaction using ASTM D 1557 or ASTM D 2049, whichever results in the higher density. At that time practical procedures for developing rapid methods 24 of field control will be developed which will be

conservatively correlated with relative density. The latter test will then be used for quality control. 24

2.5.4.5.4 Engineering Properties of Backfill 23

Grain size distributions of the clean black medium sands which will be suitable as Category 1 backfill are given in Figures 2Q B-3 and 2Q B-9. Static and dynamic material properties were evaluated for samples of the backfill material taken from the test trenches and prepared to a relative density of 75 percent (structural backfill will be placed to a minimum relative density of 75 percent). The maximum and minimum densities presented in Table 2Q B-6 were used to determine the required sample density corresponding to 75 percent relative density.

The static shear strength of 75 percent relative density backfill was evaluated by a program of triaxial compression testing. These data are given in Figure 2Q B-39 and summarized in Figure 2Q-6.

Strain-dependent dynamic elastic moduli data for 75 percent relative density backfill were generated by cyclic triaxial testing. The dynamic stress-strain data are given in Figures 2Q B-35 and 2Q B-36, and summarized in Figure 2Q-7.

2.5.4.5.5 Quality Control Program

In situ density of compacted fill will be measured. At least one density test for quality control will be performed for each one-foot thickness of compacted fill for every 22,500 square feet of material placed. All field densities will be compared with the maximum density determined by ASTM D 1557 or ASTM D 2049, whichever results in the higher 24 density, and from that and the comparison between relative density and relative compaction, verification that the relative density of the fill complies with the criteria in Section 2.5.4.5.3 will be obtained.

2.5.4.5.6 Control of Groundwater

Because of the great depth to the water table in relation to the depths of required excavation, dry conditions will exist in all excavations.

2.5.4.5.7 Category I Piping and Electrical Duct Banks 24

Category I piping and electrical duct banks will be supported by backfill for structures, fill for rough grade, or native material to an elevtion of one foot below the bottom of the pipe or duct bank. The material compaction requirements will be in accordance with Sections 2.5.4.5.2 and 2.5.4.5.3 respectively. Category I piping and electrical duct banks will not be founded in near-surface loose silty aeolian sands. The quality control program will conform to Section 2.5.4.5.5.

Bedding material will be placed a minimum distance of one foot below the bottom, one foot above the top, and five feet on each side of the pipe or duct bank. The bedding material will be in accordance with Section 2.5.4.5.2, except that 100 percent of the material will pass a one-half inch screen with no more than five percent passing the No. 200 size screen. Compaction of the bedding material will meet the requirements of Section 2.5.4.5.3.

2.5.4.6 Groundwater Conditions 23

Groundwater conditions at the Site are described in Sections 2.4.13 and 2.5.4.5.6 and in Appendix 2P.

2.5.4.6.1 Stability

The depth of the water table ensures that groundwater will not impact the stability of the safety related facilities (Sections 2.5.4, 2.5.4.1.5, 2.5.4.7, 2.5.4.8 and 2.5.4.10).

2.5.4.6.2 Control of Water Levels and Seepage

Because of the great depth of the water table, all structures will be located above the water table and there will be no special requirement for the collection and control of seepage or for the control of groundwater levels.

2.5.4.6.3 Construction Dewatering

There will be no requirement to dewater construction excavations (Section 2.5.4.5.6).

2.5.4.6.4 Permeability

Permeability values are discussed in Section 2.4.13.1 and in Appendix 2P.

23

2.5.4.6.5 Groundwater Fluctuations

Past and projected fluctuations in groundwater conditions are discussed in Sections 2.4.13.1 and 2.4.13.2.

2.5.4.6.6 Monitoring of Wells and Piezometers

Monitoring of local wells and piezometers is considered in Section 2.4.13.1 and 2.4.13.2 and in Appendix 2P.

2.5.4.6.7 Direction of Groundwater Flow

Direction of groundwater flow is discussed in Sections 2.4.13.1 and 2.4.13.2 and in Appendix 2P.

Depth (feet)	Description	In Situ Unit Weight	Relative Density	Average Water Content
0–8	Loose silty fine sand			
8–35	Medium dense to dense clean medium sand	110 p.c.f.	75%	3%
35–80	Very dense fine to medium sand, silty in parts	105 p.c.f.	85%	3%
80–125	Very dense clean gravelly sand	130 p.c.f.		3%
125–165	and sandy gravel	140 p.c.f.		12%
165–725	Very dense gravels and sands and hard silts	Gravels: 150 p.c.f. Sands: 125 p.c.f. Silts: 125 p.c.f.		Gravel: 7% Sands: 22% Silts: 25%
725–	Basalt			

Note: Average Surface Elevation 525' (MSL)

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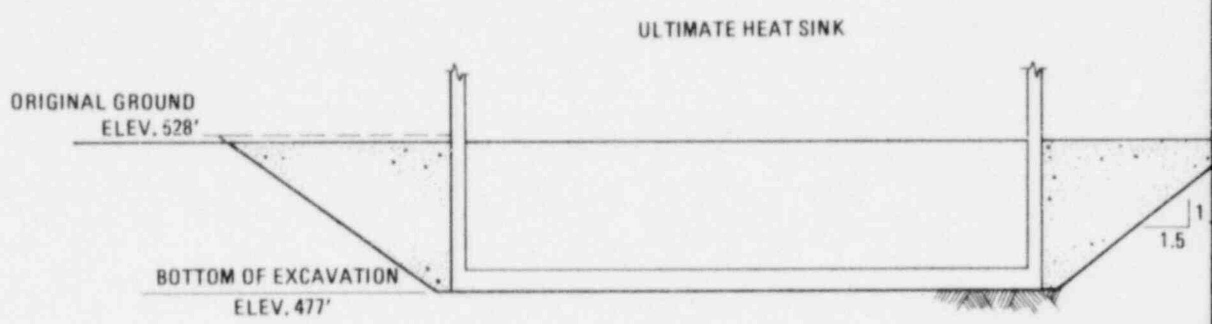
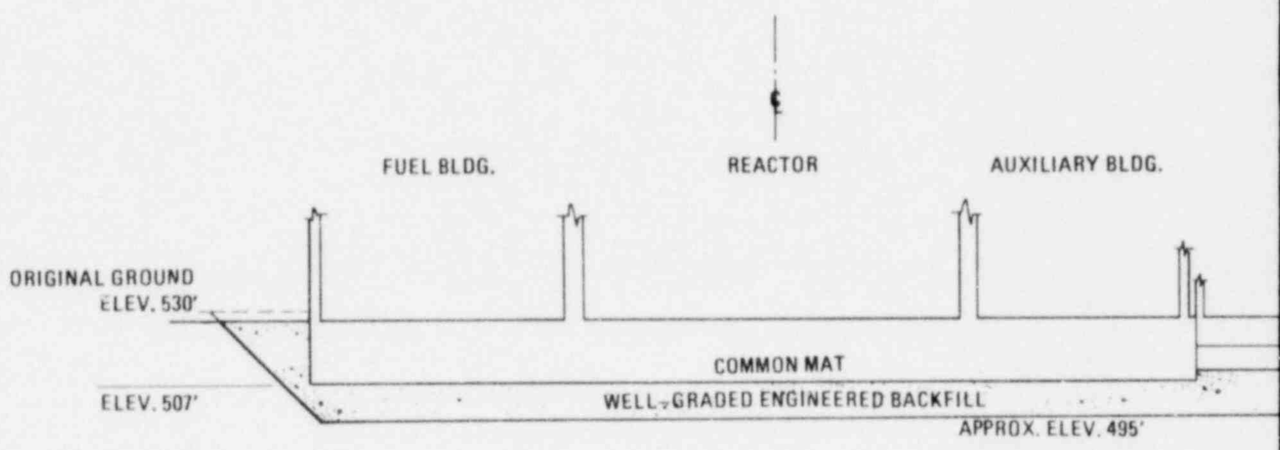
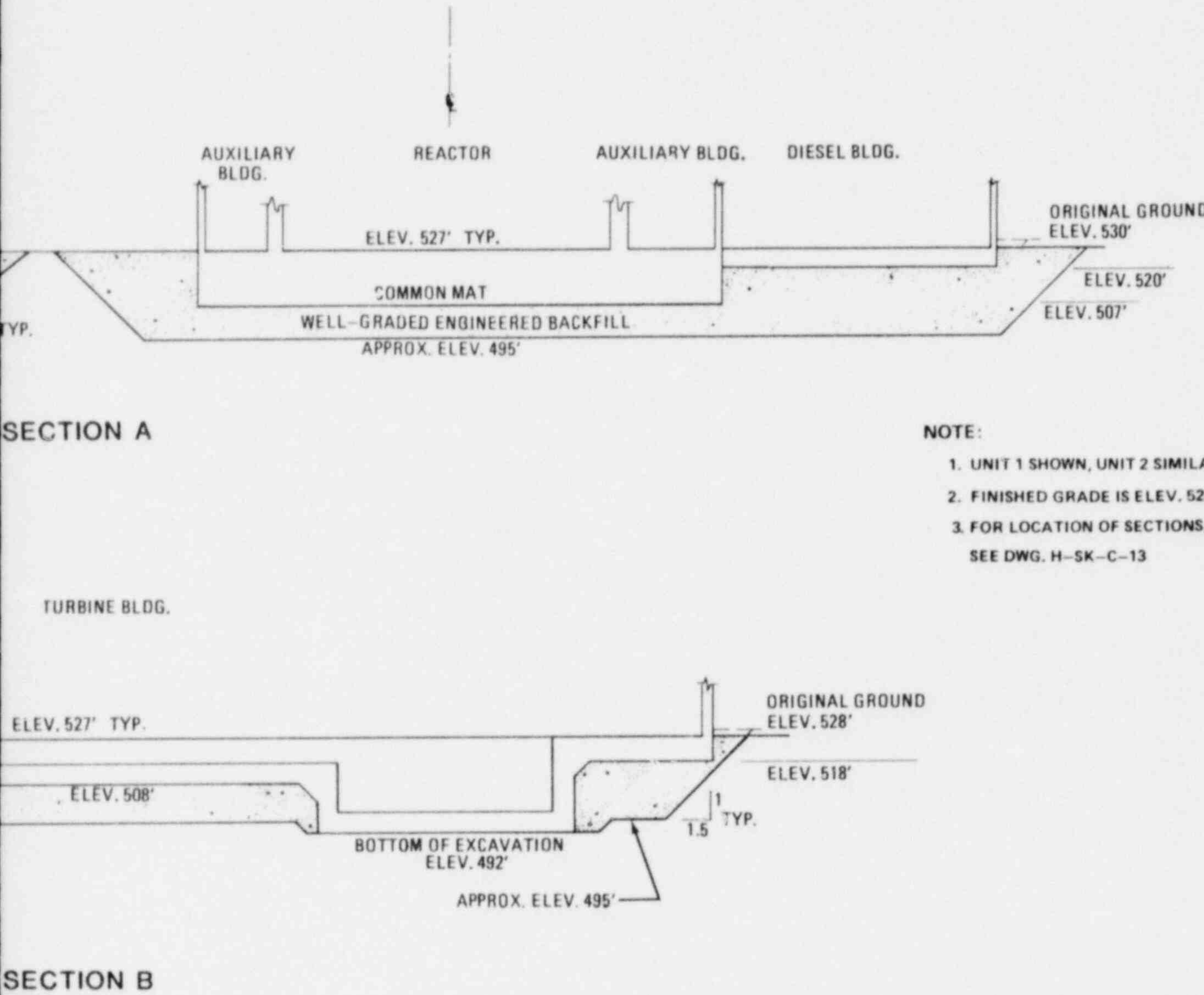
GENERALIZED SOIL PROFILE

FIGURE 2.5-11


ULTIMATE HEAT SINK
ORIGINAL GROUND
ELEV. 528'
BOTTOM OF EXCAVATION
ELEV. 477'
1
1.5


FUEL BLDG.
REACTOR
AUXILIARY BLDG.
ORIGINAL GROUND
ELEV. 530'
COMMON MAT
ELEV. 507'
WELL-GRADED ENGINEERED BACKFILL
APPROX. ELEV. 495'



NOTE:

1. UNIT 1 SHOWN, UNIT 2 SIMILAR.
2. FINISHED GRADE IS ELEV. 526.5
3. FOR LOCATION OF SECTIONS, SEE DWG. H-SK-C-13

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EXCAVATION AND BACKFILL SECTIONS

FIGURE 2.5-15

5.2.3.2 Discussion of Results

5.2.3.2.1 Seismic Data

Seismic refraction Profiles DB-10-1 through DB-10-8 (Figures 2K-34 through 2K-36 and Attachment C generally show higher, near-surface overburden velocities than are found elsewhere in the Hanford Reservation; the typical 1,000-2,000 ft/sec. shallowest material is less than 20 feet thick and is underlain by material with velocities ranging from 3,000 to 8,000 ft/sec. An analysis of materials excavated from test pits (Table 2K-1) along the seismic refraction lines indicate a lateral variation in the composition of the gravels. These compositional changes define a northwest-trending transition zone which is indicated by the northwest-trending dotted line on Figure 2K-33. Lower velocities of 3,000-5,000 ft/sec. in the gravels found northeast of the transition zone correlate with a matrix of basaltic sand. The higher velocity material (6,000-8,000 ft/sec.) encountered southwest of the zone has a matrix of silt and clay with weak calcareous cementation.

The seismic refraction data in the DB-10 area clearly define the northwest-trending basalt ridge indicated by the gravity data, as well as the drillhole data obtained by Golder Associates. Top of basalt elevations in the area of seismic lines DB-10-1 through DB-10-8 are between 460 feet and 260 feet. The seismic velocities for bedrock vary from 10,000 to 14,000 ft/sec.

Seismic lines DB-10-3A, DB-10-5, DB-10-6 and DB-10-7 were located to explore the lateral extent of the 8,000 ft/sec. material that appear at the west end of Line DB-10-3. Shallow, competent basalt (12,000 - 14,000 ft/sec.) with a localized weathered zone (6,000-9,000 ft/sec.) in the vicinity of Test Pit 6 was found to extend across the entire DB-10 area.

Profile DB-10-4 (Figure 2K-34) shows an anomalous low velocity zone in the basalt bedrock between Stations 12+00 and 16+00. A low velocity zone in the bedrock was encountered between Stations 6+00 and 10+00 on Profile DB-10-8 (Figure 2K-35) and at Station 15+50 on Line DB-10-3 (Figure 2K-36).

Seismic Profiles 7 and 8 (Figures 2K-37 and 2K-38) show an overburden sequence of 2,000-4,500 ft/sec. material underlain by 8,000-10,000 ft/sec. material. Isolated zones within the overburden have intermediate velocities that

range from 5,000-8,000 ft/sec. The velocity of the basalt varies from 13,000 to 14,500 ft/sec. in areas overlain by 100 feet or more of the gravels. In areas where the basalt is less than 100 feet below the surface the velocities range from 11,000 to 13,000 ft/sec. The seismic refraction data were used to contour the bedrock surface (Figure 2K-39) and bedrock velocity values (Figures 2K-40).

5.2.3.2.2 Gravity Data

The Bouguer gravity map shown in Figure 2K-41 was processed with a density of 2.67 g/cm^3. Because of the contrast in density between basalt and sediments (0.2 to 0.5 g/cm^3), the map is controlled mainly by the topography of the basalt. The immediate DB-10 area is characterized by an elongate, northwest-trending, gravity high located approximately one mile south of Gable Mountain and one mile west of the north-south trending May Junction monocline.

The residual gravity map for the DB-10 area is shown in Figure 2K-42. It is characterized by two major features: a northwest-trending elongate gravity high and the gravity gradient indicating the location of the May Junction monocline. The elongate, gravity high, A, coincides with the bedrock ridge which trends through the site of the DB-10 drill hole. The crest of the bedrock ridge appears to be disrupted by a north-south to northeast-trending "saddle". The dips on the flanks of the ridge are less than 20°. 24

5.2.3.3 Interpretations

The bedrock high, A, was interpreted by Myers and Price (1979) to be an anticlinal ridge. Their interpretation was based upon a single drill hole, DB-10 and their aeromagnetic data. Subsequent drilling by Golder Associates across the southeastern nose of the bedrock ridge (Figure 2K-32) has confirmed the presence of anticlinal folding (Figure 2K-43). This drill hole profile crosses the northwest-trending ridge where it intersects the May Junction monocline. The structural section of Figure 2K-43 therefore contains interference from two different features, the DB-10 anticline and the May Junction monocline. The change in the elevations between DH-97 and DH-93 contains two components. Furthermore, the line of drill holes is oriented at 45° to the trend of the bedrock high.

5.3 SOUTHEAST ANTICLINE

5.3.1 INTRODUCTION

The aeromagnetic data acquired by Washington Public Power Supply System (Figure 2K-49) show a magnetic high trending southeasterly from the eastern end of Gable Mountain. Two aeromagnetic survey blocks are joined along the axis of this magnetic high. Those individual flight lines which overlap from one survey block to another have been evaluated and confirm that the magnetic high is real and not an artifact of merging the two aeromagnetic survey blocks. Rockwell's 1980 aeromagnetic survey of the Hanford Reservation area further confirms the existence of this aeromagnetic high. Extensive seismic refraction, gravity and land magnetic data were acquired to characterize this anticlinal ridge and to define the structural relationships between the Southeast Anticline and the first-, second-, and third-order folds of the Umtanum Ridge-Gable Mountain structural trend.

5.3.2 DISCUSSION OF RESULTS

5.3.2.1 Magnetic Data

An aeromagnetic high, generally symmetrical in shape, trends in a southeasterly direction from the eastern end of Gable Mountain to the vicinity of Line 4C. At this location the anomaly decreases in amplitude and appears to be offset to the southwest. This lower amplitude magnetic high continues trending southeasterly and then easterly in the vicinity of Lines 4E and 4F.

The individual land magnetic profiles (Figure 2K-50) indicate a feature which may be more complex than the aeromagnetic data would indicate. A sharp anomaly (A) trends in a S60°E direction from Line 3 to Line 1-A but is not traceable south of Line 1-A. The single peaked, magnetic anomaly on Line 1 broadens and divides into [illegible] more subdued peaks (B and b) in the vicinity of Line 4[illegible]. The northeasterly of the two southeast trends decreases in amplitude to a magnetic low (C) on Line 4D. The southwesterly of the two southeast-trending highs appears to continue southeast of Line 4D but is then offset in an en echelon manner, similar to the aeromagnetic data, to a southeasterly-trending lower amplitude magnetic high on Lines 4E and 4F (D). The ground magnetic data are 24

consistent with the aeromagnetic data and provide greater detail.

5.3.2.2 Gravity Data

The gravity data processed at a density of 2.67 g/cm^3 (Figure 2K-51) define a gravity high trending southeasterly from the Gable Mountain area. Detailed gravity coverage (Figure 2K-6) shows that the northwest portion of this gravity feature is quite linear and appears to extend one mile northwest of the eastern end of Gable Mountain.

The southeast-trending gravity high is generally symmetrical in shape and decreases in amplitude toward the southeast. Assuming that 1 milligal gravity is equal to about 150 feet of basalt relief, the basalt surface slopes at angles ranging from 5 to 13 degrees. The gravity data clearly indicate that the southeast-trending feature changes trend to east-northeast in the vicinity of Lines 4C and 4D.

5.3.2.3 Seismic Refraction Data

Seismic refraction data across the Southeast Anticline have been acquired and profiled along Lines 3 (Figure 2K-52), 1 (Figure 2K-54), 4A (Figure 2K-55), and 4B (Figure 2K-56) and on the southwesterly side of the feature on Line 2 (Figure 2K-53). Seismic data were also obtained for portions of Lines 4C (Figure 2K-57), 6 (Figure 2K-58), and 6A (Figure 2K-59) to provide more information on the configuration of the bedrock surface in the area where the gravity and magnetic data indicated a change in the orientation of the feature. Seismic data were acquired on Line 6B (2K-60) to explore the northeast flank of the Southeast Anticline.

Overburden seismic velocities in the area of the Southeast Anticline are, in general, typical of those encountered elsewhere in the Hanford Reservation. The low velocity (2,500-3,000 ft/sec.) overburden has a uniform thickness of approximately 100 feet except at the northeast limit of the area near the Columbia River where it thins to 50 feet. Higher velocity overburden materials (9,500-10,000 ft/sec.) underlie the lower velocity material southwest of the bedrock high. The seismic velocity of this material changes to 6,500-7,500 ft/sec. northeast of the bedrock ridge.

The seismic velocity of competent basalt in the vicinity of the bedrock ridge is 15,000-16,000 ft/sec. Highly fractured basalt above a depth of 250 feet in Boring 125 (Line 4A, Figure 2K-55) correlates with a seismic velocity of 7,500 ft/sec. The higher velocity overburden materials also have a velocity of 7,200-7,500 ft/sec. over the anticline, precluding a determination of the lateral extent of the fractured basalt along Line 4A. To the southeast of Boring 125, the velocity of 9,000-9,500 ft/sec. is indicative of cemented overburden materials as identified in Boring 122A. In Boring 109, northeast of Boring 125, the 7,200-7,500 ft/sec. material has been identified as overburden.

The fractured basalt encountered on Line 4A in the vicinity of Boring 125 probably extends along strike of the ridge. Differences between the seismic top of rock elevations and borehole bedrock elevations also occur on Lines 1, 3 and 4B along the southwest side of the bedrock ridge. Basalt elevations in Boreholes 105 (Line 1) and 37 (Line 3) are 50 to 100 feet above the seismic top of high velocity bedrock (16,000 ft/sec.). The materials above the 16,000 ft/sec. horizon have a seismic velocity of 6,800-7,500 ft/sec. and are described in the boring logs as "weathered basalt." To the southeast, on Line 4B, "extremely weathered, fractured basalt" was logged in Boring 101 at elevation 234, 75 feet above the top of seismic high velocity basalt.

The profiles for Lines 1 (Figure 2K-54), 3 (Figure 2K-52), and 4A (Figure 2K-57) show slopes on the high velocity bedrock ranging from 5° to 9° on each side of the bedrock high. The profile of the southwestern side of the bedrock ridge on Line 2 (Figure 2K-53) also exhibits a bedrock slope of approximately 10°. All of the bedrock slopes described above are smooth.

The top of high velocity basalt contour map (Figure 2K-61) compiled from the seismic refraction data for the Southeast Anticline defines a southeast-trending, broadly asymmetrical anticline feature. The anticline extends from the vicinity of Line A to Line 4B where it changes trend from a southeasterly to an east-northeasterly direction. The southwest flank of the anticline has a slightly steeper gradient than the northeast flank. The feature becomes symmetrical as it changes trend to the east-northeast; the maximum slopes on either flank of the ridge decrease to 8°.

5.3.3 MODELING STUDIES

5.3.3.1 Two-Dimensional Modeling of Land Magnetic Data

The faulting along the southwest flank of the Southeast Anticline as evidenced in borehole 125 prompted further study of the land magnetic data. The magnetic anomalies that appear on Lines 2 and 4A were modeled by a two-dimensional method developed by Talwani and Heirtzler (1964). Reduction of a geologic structure to a simple geometric shape is required for the numerical modeling method applied in this study. The sequence of basalt flows and the overlying gravels of the site area were modeled as a layered sheet structure. NRM and susceptibility values for the stratigraphic units modeled (Weston Geophysical, 1978b) are summarized in Table 2K-2. Lines 2 and 4A are approximately perpendicular to the northwest trending Southeast Anticline.

It is recognized that the geologic column used for magnetic modeling analysis is not consistent with current knowledge of the units present and the thicknes of those units. In particular, the Esquatzel and Asotin members are not represented and the Umatilla member is shown 150 to 175 feet thicker than has been found in nearby boreholes. Personal communication of preliminary paleomagnetic data for the Esquatzel and Asotin members from Rockwell Hanford Operations during 1981 indicated that the natural remnant magnetization values of the Esquatzel and Asotin members are similar to that of the Umatilla member. Initial modelling of the revised stratigraphic column using these preliminary paleomagnetic data for the Esquatzel and Asotin members does not significantly change the results of modelling presented in S/HNP PSAR Amendment 23. Therefore, the Umatilla members as depicted in Figures 2K-62 through 2K-64 can be considered to be a composite of the Esquatzel, Asotin and Umatilla members. 24

The presence or absence of faults was investigated through comparison of faulted and unfaulted models of each profile. Faults modeled for this investigation included normal, northeast dipping thrust faults and southwest dipping thrust faults. The Elephant Mountain Member of the Columbia River Basalt unit is modeled as the top of basalt bedrock horizon to conform to XRF identification of this unit in several boreholes (108, 110, 117, and 125) along profile Lines 2 and 4A. The Rattlesnake Ridge interbed and five additional basalt horizons that underlie the Elephant Mountain member are included in each model. Basalt unit

thicknesses summarized in Table 2K-2 are based on information from Swanson et al. (1979) and Myers and Price (1979). Ground surface elevations are taken from survey data and the layer overlying the basalt bedrock is modeled as alluvium.

Top of basalt bedrock elevations of models for Lines 2 and 4A conform to drilling data (Appendix 2R). Shown in Figure 2K-62 are the unfaulted models for Lines 2 and 4A and the corresponding calculated anomalies. The location of borehole data is indicated on each profile modeled. Also shown in Figure 2K-62 is a version of the unfaulted model for Line 2 in which the top of bedrock has been eroded. The Elephant Mountain Basalt unit is thinned to 25 feet at the ridge crest and the underlying flows maintain their typical thicknesses in this model. The increased thickness of Elephant Mountain Basalt in Drillhole 125 of Line 4A did not allow the possibility of an erosional model for this line.

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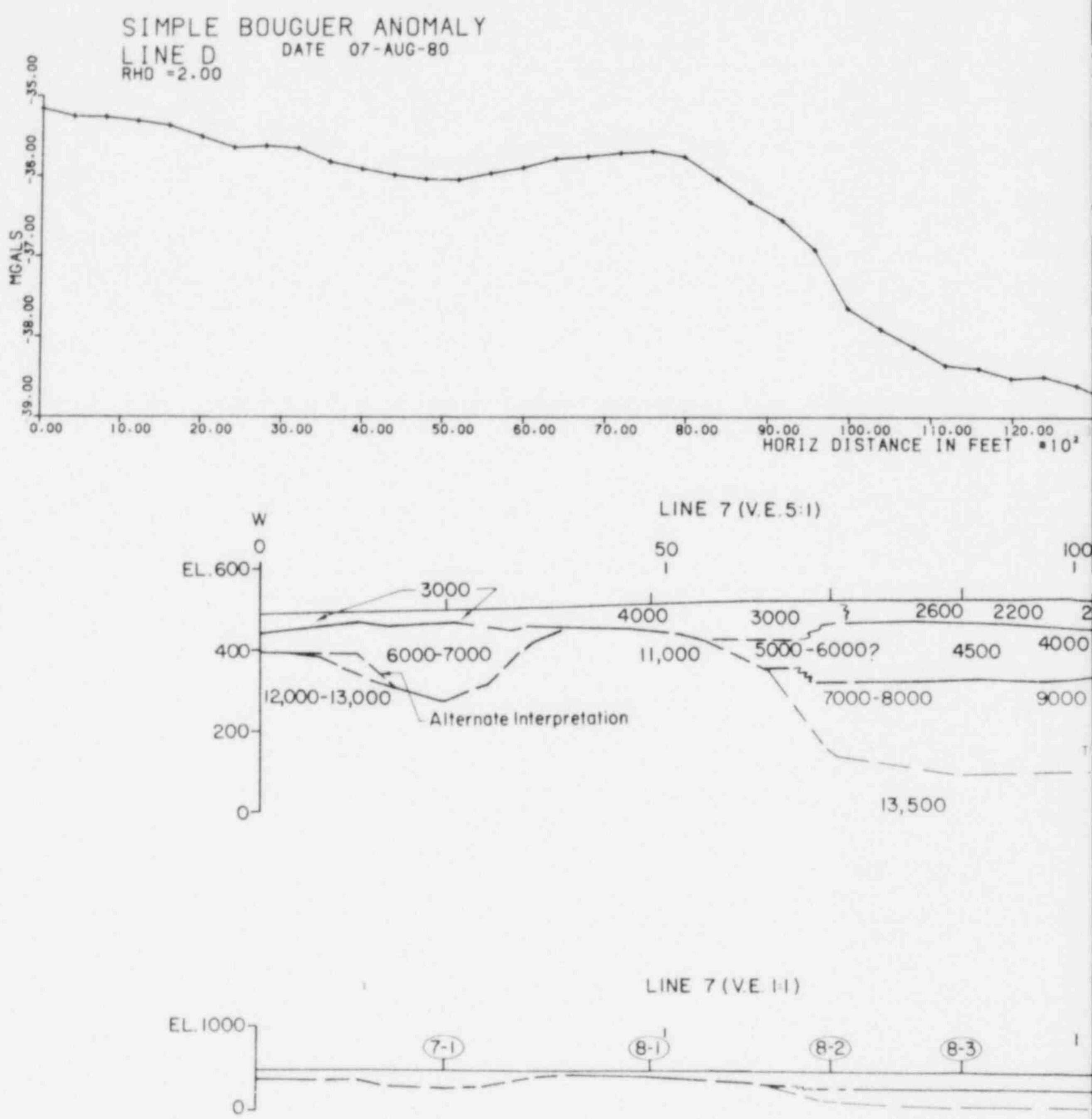
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-36.00
-37.00
-38.00
-39.00
0.00 10.00 20.00 30.00 40.00 50.00 60.00 70.00 80.00 90.00 100.00 110.00 120.00
HORIZ DISTANCE IN FEET *10²
LINE 7 (V.E. 5:1)
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0
50
100
EL. 600
400
200
0
3000
4000
3000
2600
2200
6000-7000
11,000
5000-6000?
4500
4000
12,000-13,000
7000-8000
9000
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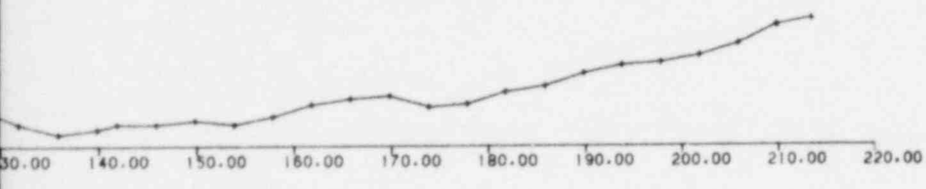

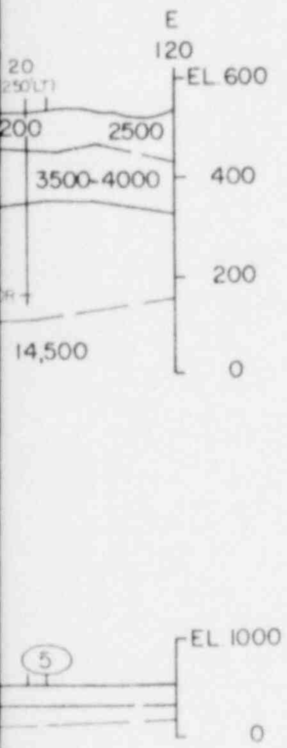

EXPLANATION

0+00	LINE STATION
5	LINE INTERSECTION
10	BOREHOLE LOCATION
	ROTARY BORING
TOP OF CORE, TOR	CORE BORING
TOR	TOP OF BASALT BEDROCK
6000	SEISMIC VELOCITY IN FT/SEC
1500-2000, 6000±	INFERRED SEISMIC VELOCITY IN FT/SEC
	SEISMIC VELOCITY INTERFACE
	INFERRED SEISMIC VELOCITY INTERFACE
	LOCATION OF LATERAL CHANGE IN SEISMIC VELOCITY

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

SEISMIC PROFILE LINE 7 AND GRAVITY PROFILE LINE D.

FIGURE 2K-37

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1.0 INTRODUCTION

As part of the siting investigations for the Skagit/Hanford Nuclear Project, geophysical field studies were conducted for Northwest Energy Services Company (NESCO) from February 1980 to August 1981. The geophysical investigations delineated the bedrock configuration of the Hanford Site surrounding the Umtanum Ridge-Gable Mountain structural trend (Appendix 2K) and the region surrounding the proposed Plant locations (Figure 2L-1), and determined the characteristics of the foundation materials beneath the proposed Plant Site.

The bedrock units of the Hanford Site, the Miocene Columbia River Basalt Group, are overlain by as much as 700 feet of Late Tertiary and Quaternary sediments and sedimentary rocks. The drilling, performed under the direction of Golder Associates, encountered the Elephant Mountain Member of the Saddle Mountains Basalt as the uppermost bedrock unit in all boreholes. Consequently, the configuration of the top of bedrock can be considered a structure contour map on the top of the Elephant Mountain Member. These conditions are favorable for the use of geophysical methods to delineate subsurface structure as revealed through topography. The density contrast between the overburden and the basalt (0.3 to 0.7 g/cm^3) is such that a gravity contour map is a quasi-topographic map of the top of basalt. In addition, seismic refraction and downhole surveys have been used to profile overburden horizons and the basalt surface through delineation of seismic velocities.

1.1 SUBREGIONAL SETTING

The Skagit/Hanford Nuclear Project Site is located in an area of low gravity gradients and low magnetic gradients. The geophysical data indicate an area of low bedrock relief and gradual bedrock slopes beneath the Site area.

The Gable Butte-Gable Mountain segment of the Umtanum Ridge-Gable Mountain structural trend (Appendix 2K), is located north of the S/HNP Site. Gable Butte, Gable Mountain, the DB-10 faults and anticline and the May Junction monocline are all structures contained within or bounding the Gable Butte-Gable Mountain segment.

The Southeast Anticline (Appendix 2K), a subsurface anticline which extends for one mile to the northwest of the eastern end of Gable Mountain, is located northeast of the

S/HNP Site. This northwest-southeast trending, low amplitude anticline changes trend to east-northeast in the vicinity of the Columbia River. 24

During the course of siting investigations for the Skagit/Hanford Nuclear Project, crosshole in-situ velocity measurements were made at four locations shown in Figure 2L-2. Measurements were made in the southeast corner of Section 34 (T.13N, R.27E) and near Corehole 3 during preliminary site location evaluations. As part of the final site location studies, measurements were made at the location of the western unit and at station 100+00 on line W.

Each seismic crosshole study used an array of five bore-holes. The locations of the two arrays of boreholes used in the crosshole velocity measurements in the west unit area and the array of boreholes at station 100+00 on line W are shown on Figures 2L-3.

3.1.1.3.1 Data Acquisition and Processing

Crosshole velocity measurements were made using geophones containing three orthogonal elements (one vertical, two horizontal). Recordings were obtained using a 12-channel seismograph with visual time marks at two-millisecond intervals. The seismic signal was generated with the energy source in one hole, and is detected in the geophone holes with the source and receivers at the same elevation (Figure 2L-4). Seismic energy was either generated by a specially designed borehole air-gun or by small explosive charges.

The "P" wave and "S" wave velocity data were obtained at 10-foot intervals in each borehole array. Each array of boreholes was drilled to sufficient depth to penetrate the upper most high velocity refractor (velocity of 8,000 to 10,000 ft/sec). | 24

Verticality measurements were made in each of the boreholes used for the in-situ velocity study. These measurements are necessary to compensate for inter-hole distance changes with depth that arise due to deviations of the boreholes from the vertical. Incorporation of these data permits accurate velocity value determinations from individual travel time plots.

3.1.2 GRAVITY DATA

Approximately 10,000 gravity stations were established during the siting studies for the S/HNP Site along the traverse lines shown in Appendix 2K, Figure 2K-6. In the

Site Area, 2,250 closely spaced gravity stations were established along the traverse lines shown on Figure 2L-5. The gravity method, including data acquisition and processing techniques, is described in Appendix 2K.

3.1.3 LAND MAGNETIC DATA

Land magnetic data were acquired along five hundred miles of traverse lines during the S/HNP siting study (Appendix 2K, Figure 2K-6), including 145 miles of traverse lines within the Site Area (Figure 2L-5). The magnetic method, including data acquisition and processing techniques, is described in Appendix 2K.

3.2 SUPPLEMENTAL GEOPHYSICAL DATA

3.2.1 DATA SUPPLIED BY WASHINGTON PUBLIC POWER SUPPLY SYSTEM

The Washington Public Power Supply System provided aeromagnetic data (Figure 2L-6) to augment the data acquired in the Site Area for Northwest Energy Services Company. Details of the aeromagnetic survey, as well as previous interpretations, can be found in Washington Public Power Supply System (1977, Appendix 2R-I) and Weston Geophysical (1978a).

3.2.2 DATA SUPPLIED BY ROCKWELL HANFORD OPERATIONS

Rockwell Hanford Operations provided land magnetic and gravity data, aeromagnetic contour maps of a multi-level survey, and prints of processed reflection data. The locations of the Rockwell reflection profiles and those Rockwell gravity data utilized in the Site Area are shown on Figure 2L-7.

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TABLE 20-6

CORRELATION OF GLACIOFLUVIAL DEPOSITS
BASED ON LITHOLOGY*

Very Good Correlation**

1. Between samples with:

High basalt content (>65%)
Low to moderate "granitic" content (7% to 20%)
Low sandstone content (0 to 11%)

E1-W1	CT-2-W4	GT-4-W1	CD-4-W6
E1-W2	GT-3-W2	GT-4-W4	CD-8B-W2
E1-W4	GT-3-W3	CD-4-W4	

2. Between samples with:

Moderate basalt content (43% to 57%)
Moderate sandstone content (22% to 33%)
Low to moderate "granitic" content (6% to 19%)

GT-2-W3
GT-4-W2
E1-W3

3. Between samples with:

Low to moderate basalt content (20% to 48%)
Moderate "granitic" content (48% to 66%)
Very low sandstone content (1% to 2%)

CD-4-W1	CD-6-W1
CD-4-W2	CD-6-W3
CD-4-W3	

No Correlation with Other Samples

1.	CD-6-W2	Low basalt content (9%) Moderate "granitic" content (33%) Moderate sandstone content (48%)
2.	GT-2-W5	Low basalt content (10%) Low "granitic" content (7%) Very high sandstone content (83%)
3.	CD-6-W6	Low basalt content (8%) High "granitic" content (77%) Moderate sandstone content (15%)

*Semi-quantitative correlation based on a scatter analysis.

**All samples are coarse-grained sand (0.5Ø) collected from bedded sediments.
No clastic dikes were analyzed.

TABLE 20-7 Sheet 1 of 1

AMOUNTS AND RATES OF DISPLACEMENT
FOR THE CENTRAL FAULT

DISPLACEMENTS

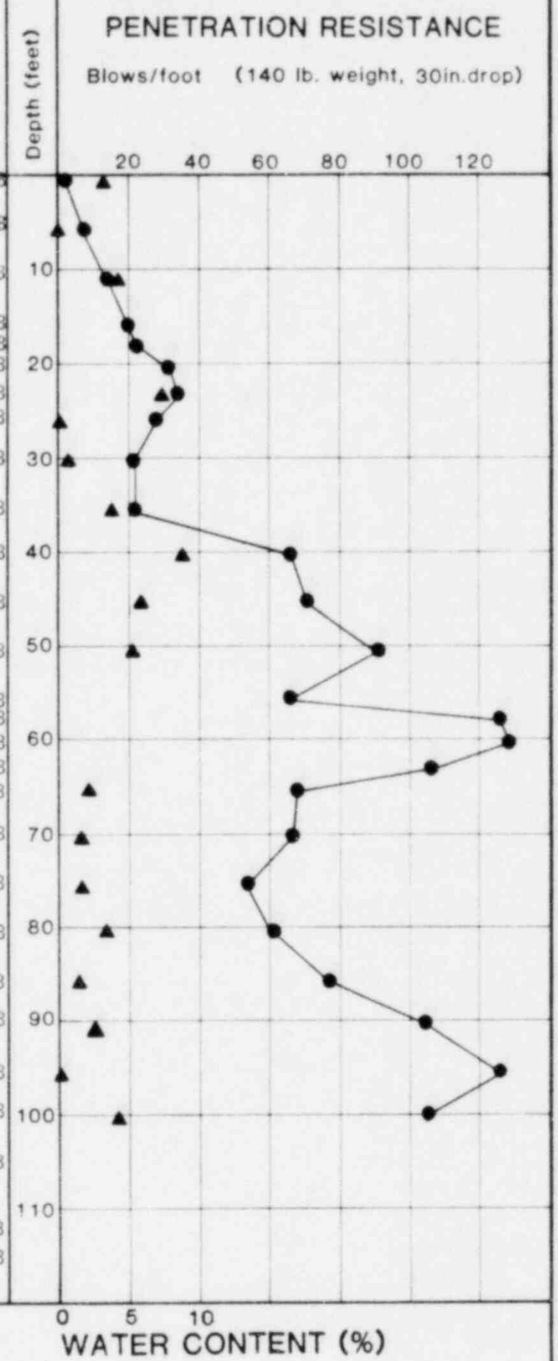
Base of Rattlesnake Ridge Interbed	Maximum feet (meters)	Minimum feet (meters)
Vertical	123 (38)	112 (34)
Dip Slip	182 (56)	148 (45)
Base of Selah Interbed		
Vertical	165 (50)	150 (46)
Dip Slip	210 (64)	206 (63)
Base of Glaciofluvial Deposits		
Dip Slip	0.2 (0.05)	---

RATES OF DISPLACEMENT

Base of Rattlesnake Ridge Interbed (11 m.y.)	Maximum inches/yr (mm/yr)	Minimum inches/yr (mm/yr)
Vertical	1.34×10^{-4} (3.5×10^{-3})	1.2×10^{-4} (3.0×10^{-3})
Dip Slip	2.0×10^{-4} (5.1×10^{-3})	1.6×10^{-4} (4.0×10^{-3})
Base of Selah Interbed (12.5 m.y.)		
Vertical	1.6×10^{-4} (4.0×10^{-3})	1.4×10^{-4} (3.6×10^{-3})
Dip Slip	2.0×10^{-4} (5.1×10^{-3})	2.0×10^{-4} (5.1×10^{-3})
Base of Glaciofluvial Deposits (13,000 to 15,000 yr/b.p.)		
Dip Slip	1.8×10^{-4} (4.6×10^{-3})	1.3×10^{-4} (3.3×10^{-3})

SOIL PROFILE / SAMPLES

Elev / Depth	Description	Number	Type	Recovery
532.3 / 0	Very loose to loose dark yellowish brown silty fine SAND	1	I	18/18
		2	I	16/18
524.3 / 8	Medium dense olive gray silty fine SAND	3	I	18/18
517.3 / 15	Medium dense to dense varicolored to olive black clean fine to medium SAND	4	I	0/18
		5	I	0/18
		6	I	1/18
		7	I	18/18
		8	I	18/18
		9	I	18/18
		10	I	18/18
492.3 / 40	Very dense dark yellowish brown silty fine SAND	11	I	18/18
		12	I	16/18
		13	I	17/18
480.3 / 52	Very dense varicolored to olive black clean fine to medium SAND	14	I	0/18
		15	I	6/18
		16	I	0/18
		17	I	18/18
		18	I	15/18
		19	I	12/18
		20	I	12/18
		21	I	12/18
447.3 / 85	Very dense varicolored to olive black clean gravelly fine to coarse SAND	22	I	6/18
		23	I	12/18
		24	I	10/18
		25	I	16/18
		26	II	18/18
		27	II	18/18
417.3 / 115	Very dense varicolored to olive black clean sandy fine to coarse GRAVEL	28	II	12/18



Boring Method: Hollow Stem Auger
Top Drive Rotary

Sample Type: I – 2" (O.D.) Drive Open
II – 4 1/2" (O.D.) Drive Open
III – 3" Thin Walled, Push Open

Standard Penetration Resistance ●
Water Content ▲
Pressuremeter Test →

Drill Hole Coordinates N 422698+92 E 2268389+43

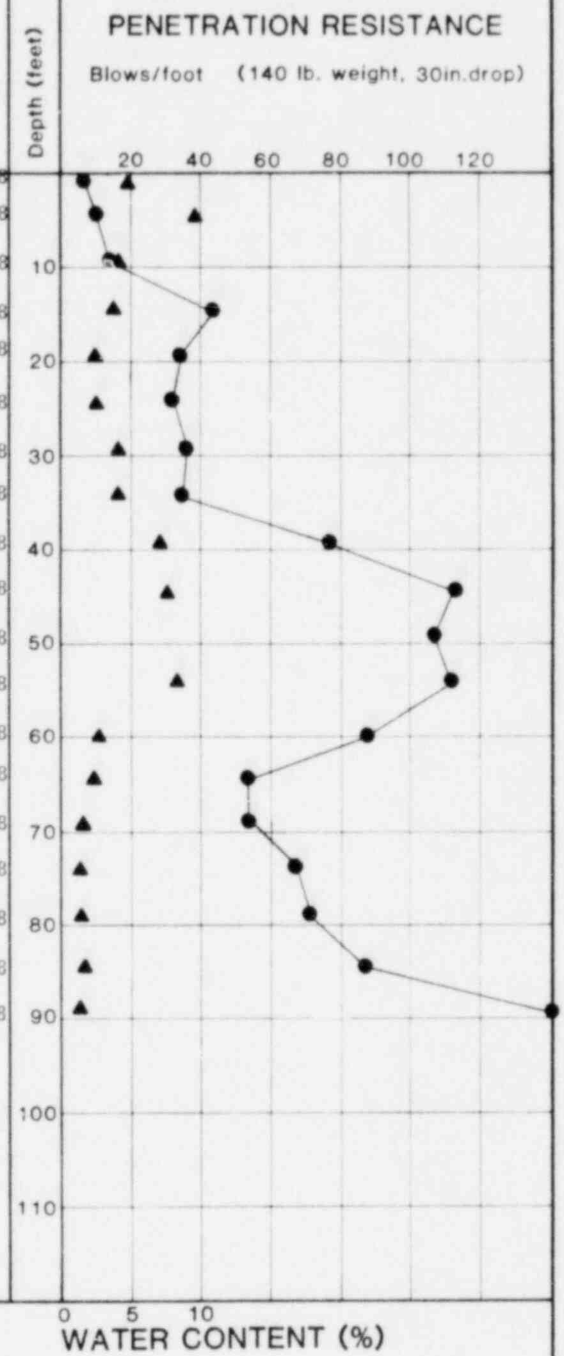
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

LOG OF DRILL HOLE W-1

FIGURE 2Q A-1

SOIL PROFILE

Elev / Depth	Description	Samples: Number	Type	Recovery
527.7 / 0	Loose dark yellowish brown silty fine SAND	1	I	14/18
		2	I	15/18
519.7 / 8	Medium dense dark yellowish brown silty fine to medium SAND	3	I	10/18
514.7 / 13	Dense varicolored to olive black clean fine to medium SAND	4	I	15/18
		5	I	12/18
		6	I	18/18
		7	I	10/18
		8	I	14/18
489.7 / 38	Very dense dark yellowish brown slightly silty fine SAND	9	I	18/18
		10	I	18/18
479.7 / 48	Very dense varicolored to olive black clean fine to medium SAND	11	I	1/18
		12	I	1/18
		13	I	18/18
		14	I	18/18
		15	I	18/18
		16	I	16/18
		17	I	13/18
447.7 / 80	Very dense varicolored to olive black clean gravelly fine to coarse SAND	18	I	12/18
		19	I	11/18
437.7 / 90	End of Hole			



Boring Method: Hollow Stem Auger

Sample Type: I - 2" (O.D.) Drive Open
II - 4 1/2" (O.D.) Drive Open
III - 3" Thin Walled, Push Open

Drill Hole Coordinates N 422870 E 2268500

Standard Penetration Resistance ●
Water Content ▲
Pressuremeter Test →

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

LOG OF DRILL HOLE W-3

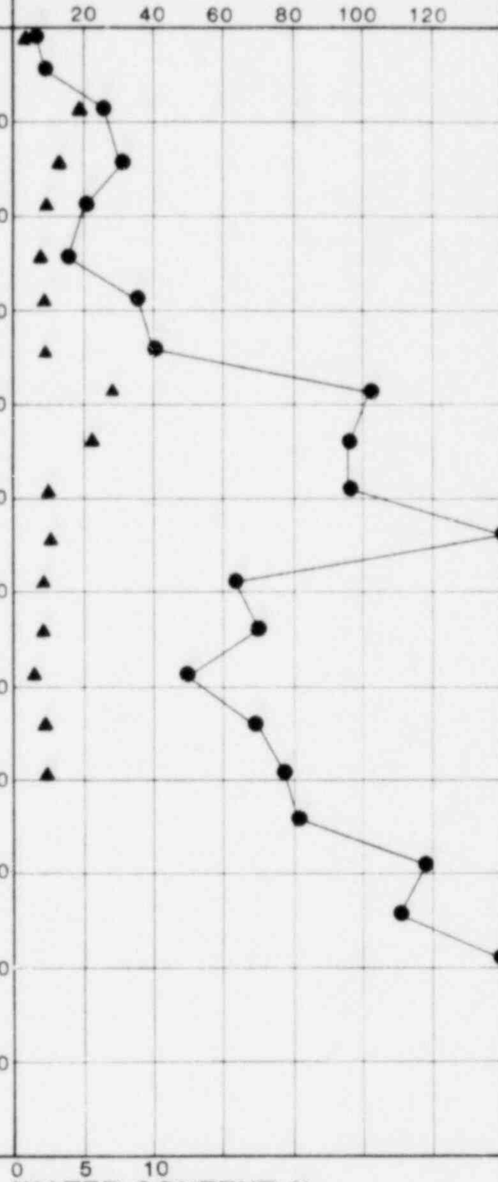
FIGURE 2Q A-3

4/2/82

SOIL PROFILE			SAMPLES			
Elev / Depth	DESCRIPTION	Stratigraphic Plot	Number	Type	Recovery	Depth (feet)
527.4 / 0	Loose dark yellowish brown to olive gray silty fine SAND		1	I	15/18	
			2	I	15/18	
519.4 / 8	Medium dense varicolored to olive black silty fine and clean medium SANDS		3	I	17/18	10
			4	I	18/18	
512.4 / 15	Medium dense to dense varicolored to olive black clean fine to coarse SAND with scattered gravel		5	I	12/10	20
			6	I	11/18	
			7	I	12/18	30
			8	I	8/18	
489.4 / 38	Very dense varicolored to olive gray silty fine to medium SAND		9	I	14/18	40
			10	I	17/18	
479.4 / 48	Very dense varicolored to olive black clean fine to medium SAND with scattered gravel		11	I	14/18	50
			12	I	18/18	
			13	I	15/18	60
			14	I	6/18	
			15	I	14/18	70
			16	I	14/18	
			17	I	14/18	80
445.4 / 82	Very dense varicolored to olive black slightly silty gravelly fine to coarse SAND		18	I	13/18	
			19	I	14/18	90
	Very dense olive gray silty fine to medium SAND		20	I	16/18	
428.0 / 99.4			21	I	6/8	100
	End of Hole					110

PENETRATION RESISTANCE

Blows/foot (140 lb. weight, 30in.drop)



WATER CONTENT %

Boring Method: Hollow Stem Auger

Sample Type: I - 2" (O.D.) Drive Open

II - 4 1/2" (O.D.) Drive Open

III - 3" Thin Walled, Push Open

Standard Penetration Resistance ●

Water Content ▲

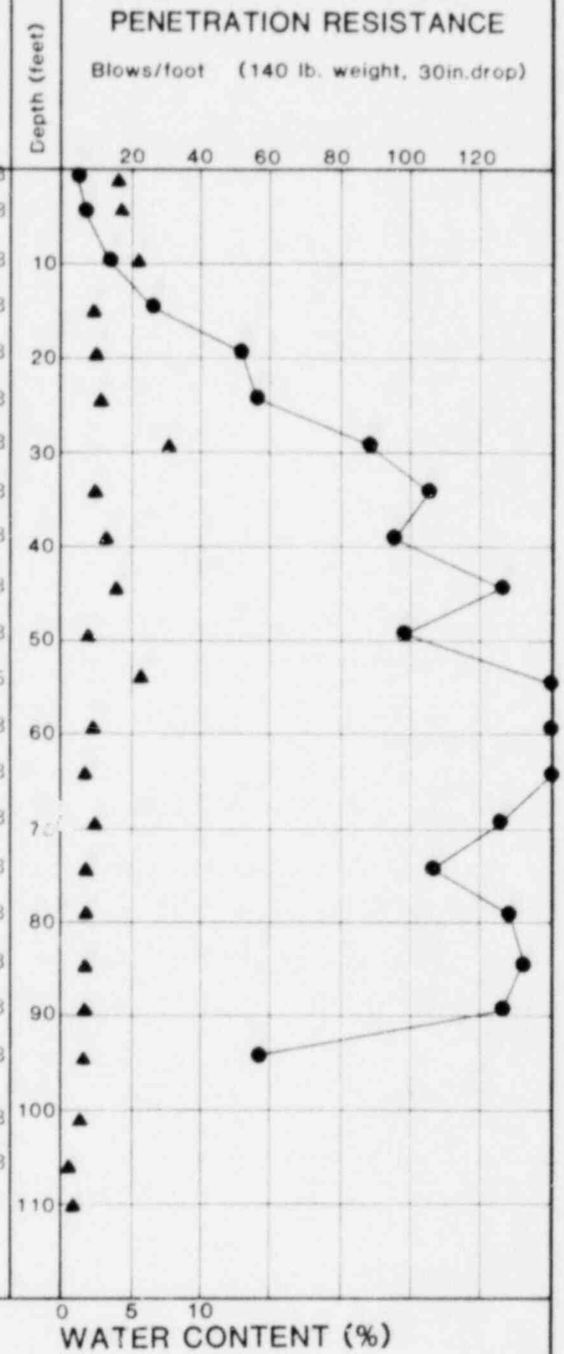
Pressuremeter Test →

Drill Hole Coordinates N 422460 E 2268660

PUGET SOUND POWER & LIGHT COMPANY SKAGIT / HANFORD NUCLEAR PROJECT	LOG OF DRILL HOLE W-18	FIGURE 2Q A-18

SOIL PROFILE / SAMPLES

Elev / Depth	Description	Stratigraphic Plot	Sample Number	Type	Recovery
519.1 / 0	Loose dark yellowish brown slightly silty fine SAND		1	I	16/18
			2	I	18/18
511.1 / 8	Medium dense varicolored to olive gray silty fine to medium SAND		3	I	12/18
506.1 / 13	Medium dense to very dense varicolored to olive black clean fine to coarse SAND with scattered gravel		4	I	8/18
			5	I	16/18
			6	I	15/18
491.1 / 28	Very dense varicolored to olive black slightly silty fine to medium SAND		7	I	18/18
			8	I	18/18
			9	I	18/18
			10	I	18/18
			11	I	18/18
	Very dense varicolored to olive black clean fine to medium SAND with scattered gravel		12	I	16/16
			13	I	18/18
			14	I	18/18
			15	I	18/18
446.1 / 73	Very dense varicolored to olive black clean gravelly fine to coarse SAND		16	I	16/18
			17	I	17/18
			18	I	19/18
			19	I	14/18
426.1 / 93	Very dense varicolored to olive black clean sandy fine to coarse GRAVEL		20	I	11/18
			21	II	16/18
			22	II	12/18
			23	II	8/9
			24	II	0/3



Boring Method: Hollow Stem Auger
Top Drive Rotary

Sample Type: I - 2" (O.D.) Drive Open
II - 4 1/2" (O.D.) Drive Open
III - 3" Thin Walled, Push Open

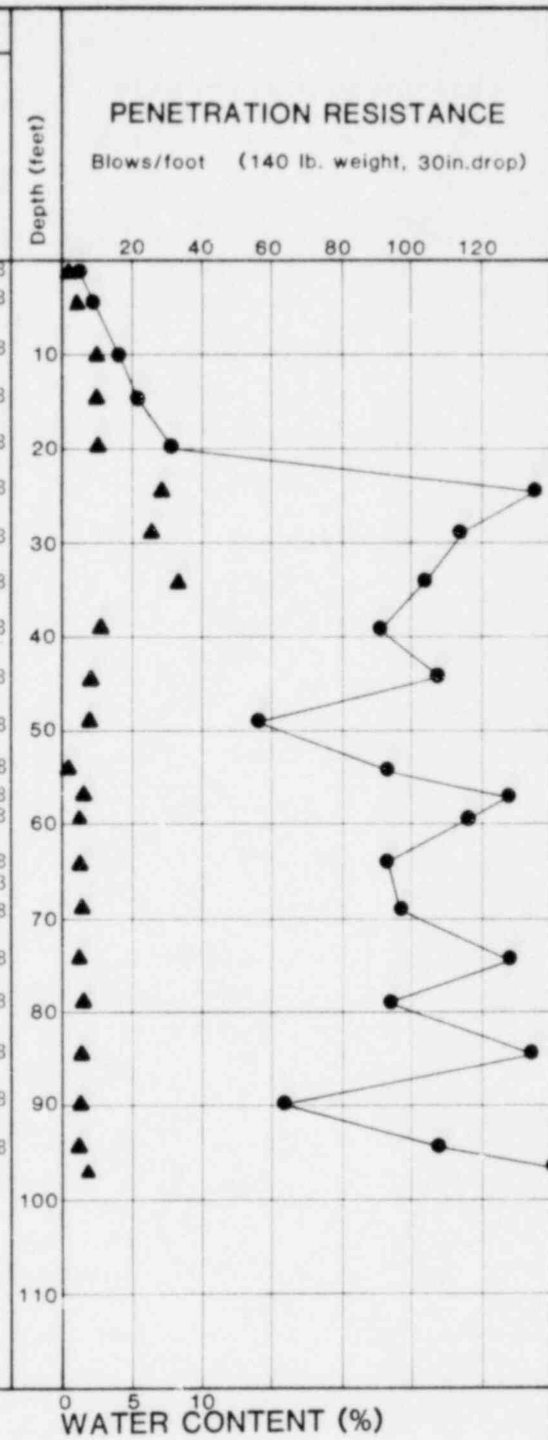
Standard Penetration Resistance ●
Water Content ▲
Pressuremeter Test →

Drill Hole Coordinates N 422340 E 2269560

PUGET SOUND POWER & LIGHT COMPANY SKAGIT / HANFORD NUCLEAR PROJECT	LOG OF DRILL HOLE E-6	FIGURE 2Q A-25

SOIL PROFILE / SAMPLES

Elev / Depth	Description	Stratigraphic Plot	Number	Type	Recovery
518.4 / 0	Loose dark yellowish brown and olive gray silty fine SAND		1	I	9/18
			2	I	16/18
510.4 / 8	Medium dense to dense clean medium to coarse SAND with scattered gravel		3	I	12/18
			4	I	12/18
			5	I	8/18
495.4 / 23	Very dense varicolored to olive black silty fine to medium SAND		6	I	18/18
			7	I	18/18
			8	I	17/18
480.4 / 38	Very dense varicolored to olive black clean fine to medium SAND		9	I	16/18
			10	I	18/18
			11	I	15/18
			12	I	0/18
			13	I	13/18
			14	I	15/18
			14A	III	7/7
	Very dense varicolored to olive gray slightly silty fine to medium SAND		15	I	12/18
			15A	III	13/13
			16	I	17/18
446.4 / 72	Very dense varicolored to olive black clean gravelly fine to coarse SAND		17	I	13/18
			18	I	13/18
			19	I	15/18
			20	I	13/18
			21	I	14/18
421.3			22	I	4/4
97.1	End of Hole				



Boring Method: Hollow Stem Auger

Sample Type: I - 2" (O.D.) Drive Open

II - 4 1/2" (O.D.) Drive Open

III - 3" Thin Walled, Push Open

Drill Hole Coordinates N 422740 E 2269400

Standard Penetration Resistance ●

Water Content ▲

Pressuremeter Test →

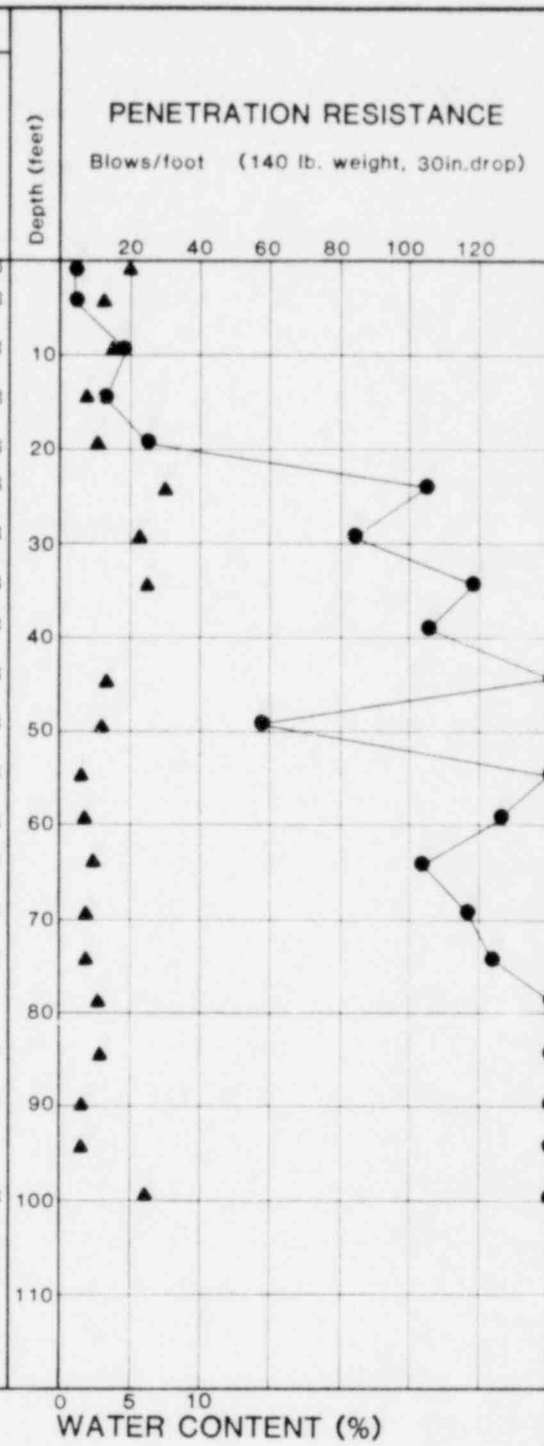
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

LOG OF DRILL HOLE E-11

FIGURE 2Q A-30

SOIL PROFILE

Elev / Depth	Description	Samples: Number	Samples: Type	Samples: Recovery
519.8 / 0	Loose dark yellowish brown clean fine SAND	1	I	14/18
		2	I	14/18
510.8 / 9	Medium dense varicolored to olive black clean fine to coarse SAND	3	I	16/18
		4	I	5/18
		5	I	13/18
495.8 / 24	Very dense varicolored to olive gray silty fine to medium SAND	6	I	15/18
		7	I	18/18
		8	I	17/18
		9	I	1/18
479.8 / 40	Very dense varicolored to olive black clean fine to medium SAND	10	I	16/18
		11	I	16/18
	Very dense varicolored to olive black clean gravelly fine to coarse SAND	12	I	18/18
		13	I	16/18
		14	I	6/18
	Very dense varicolored to olive gray slightly silty fine SAND	15	I	18/18
		16	I	18/18
443.8 / 76	Very dense varicolored to olive black clean gravelly fine to coarse SAND with silty layers	17	I	8/8
		18	I	15/18
		19	I	15/18
		20	I	14/18
		21	I	13/18
419.6 / 100.2	End of Hole			



Boring Method: Hollow Stem Auger

Sample Type: I - 2" (O.D.) Drive Open
II - 4 1/2" (O.D.) Drive Open
III - 3" Thin Walled, Push Open

Drill Hole Coordinates N 422440 E 2269120

Standard Penetration Resistance ●
Water Content ▲
Pressuremeter Test →

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

LOG OF DRILL HOLE E-16

FIGURE 2Q A-35

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING W-1

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION
S-11	40.0-41.5	66	8.1	M.A. S.C.	Silty SAND, poorly graded. Maximum size about 6mm. About 80% subangular to subrounded, fine sand and 20% non-plastic fines. Dark yellowish brown. Dry. Very dense. Weak reaction to HCl.
S-12	45.0-46.5	71	6.3	M.A.	Slightly silty SAND, poorly graded. Maximum size about 6mm. About 90% subangular to rounded, fine to medium sand and 10% non-plastic fines. Dark yellowish brown. Dry. Very dense. Weak reaction to HCl.
S-13	50.0-51.5	92	5.7	M.A.	Slightly silty SAND, poorly graded. Maximum size about 5mm. About 90% subangular to rounded, fine to medium sand and 10% non-plastic fines. Varicolored to dark yellowish brown. Dry. Very dense. Weak reaction to HCl.
S-14	55.0-56.5	66	-	-	No recovery
S-15	57.5-59.0	126	-	S.C.	Clean SAND, poorly graded. Maximum size about 6mm. Subangular to rounded, fine to medium sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-16	60.0-61.5	128	-	-	No recovery
S-17	63.0-64.5	116	-	M.A.	Slightly silty SAND, poorly graded. Maximum size about 5mm. About 90% subangular to rounded, fine to medium sand and 10% non-plastic fines. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-18	65.0-66.5	68	2.2	M.A.	Clean SAND, poorly graded. Maximum size about 10mm. Subangular to rounded, fine to coarse sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-19	70.0-71.5	66	2.0	-	Clean SAND, poorly graded. Maximum size about 10mm. Subangular to rounded, fine to coarse sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.

24

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING W-1

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION
S-20	75.0-76.5	54	1.8	M.A. S.G.	Clean SAND, poorly graded. Maximum size about 5mm. Subangular to rounded, fine to medium sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-21	80.0-81.5	62	3.2	M.A.	Clean SAND, poorly graded. Maximum size about 10mm. Subangular to rounded, fine to coarse sand. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-22	85.0-86.5	79	1.8	S.G.	Clean gravelly SAND, well graded. Maximum size about 10mm. About 70% subangular to rounded, fine to coarse sand and 30% fine gravel. Varicolored. Dry. Very dense. Weak reaction to HC1.
S-23	90.0-91.5	105	2.5	-	Clean gravelly SAND, well graded. Maximum size about 25mm. About 75% subangular to rounded, fine to coarse sand and 25% fine to coarse gravel. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl. 24
S-24	95.0-96.5	126	0.2	-	Clean gravelly SAND, well graded. Maximum size about 25mm. About 70% subangular to rounded, fine to coarse sand and 30% fine to coarse gravel. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-25	100.0-101.5	106	4.2	-	Slightly silty SAND, poorly graded. Maximum size about 25mm. About 90% subrounded, fine to coarse sand and 10% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HC1.
S-26	104.5-106.0	-	-	M.A.	Slightly silty gravelly SAND, poorly graded. Maximum size about 50mm. About 75% subangular to rounded, fine to coarse sand, 15% fine to coarse gravel and 10% non-plastic fines. Varicolored to dark yellowish brown. Dry. Weak reaction to HC1.
S-27	111.5-113.0	-	-	M.A.	Clean gravelly SAND, well graded. Maximum size about 40mm. About 55% subangular to rounded, fine to coarse sand and 45% fine to coarse gravel. Varicolored to olive gray. Dry. Weak reaction to HCl.

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING W-3

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION	
S-1	0.0-1.5	6	4.8	-	Silty SAND, poorly graded. Maximum size about 2mm. About 85% subrounded, fine sand and 15% non-plastic fines. Dark yellowish brown. Dry. Loose. No reaction to HCl.	24
S-2	3.5-5.0	10	9.8	-	Silty SAND. Maximum size about 2mm. About 65% subrounded, fine sand and 35% non-plastic fines. Dark yellowish brown. Dry. Loose. No reaction to HCl.	24
S-3	8.5-10.0	14	4.2	-	Silty SAND, poorly graded. Maximum size about 20mm. About 85% subrounded, fine to medium sand and 15% non-plastic fines. Dark yellowish brown. Dry. Medium dense. No reaction to HCl.	
S-4	13.5-15.0	44	3.9	M.A.	Slightly silty SAND, poorly graded. Maximum size about 10mm. About 95% subangular to subrounded, fine to coarse sand and 5% non-plastic fines. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.	
S-5	18.5-20.0	35	2.6	-	Clean SAND, poorly graded. Maximum size about 25mm. Subangular to subrounded, fine to medium sand. Varicolored to olive gray. Dry. Dense. Weak reaction to HCl.	
S-6	23.5-25.0	32	2.8	-	Clean SAND, poorly graded. Maximum size about 30mm. Subangular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.	
S-7	28.5-30.0	37	4.3	-	Clean SAND, poorly graded. Maximum size about 35mm. Subangular to subrounded fine to coarse sand. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.	
S-8	33.5-35.0	35	4.4	-	Slightly silty SAND, poorly graded. Maximum size about 20mm. About 95% subangular to subrounded, fine to coarse sand and 5% non plastic fines. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.	
S-9	38.5-40.0	79	7.1	-	Slightly silty SAND, poorly graded. Maximum size about 1mm. About 90% subangular to rounded, fine to medium sand and 10% non-plastic fines. Dark yellowish brown. Dry. Very dense. Weak reaction HCl.	

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING W-18

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION
S-1	0.0-1.5	6	1.6	-	Silty SAND, poorly graded. Maximum size about 3mm. About 85% angular to subrounded fine to medium sand and 15% non-plastic fines. Dark yellowish brown. Dry. Loose. No reaction to HCl.
S-2	3.2-4.7	9	-	-	Silty SAND . Maximum size about 4mm. About 60% angular to subrounded, fine to medium sand and 40% non-plastic fines. Varicolored to olive gray. Dry. Loose. No reaction to HCl.
S-3	8.2-9.7	25	3.5	-	Slightly silty SAND, poorly graded. Maximum size about 8mm. About 95% angular to subrounded, fine to medium sand and 5% non-plastic fines. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl. (8.2-9.5)
			7.1	-	Silty SAND . Maximum size about 2mm. About 60% angular to subrounded fine sand and 40% non-plastic fines. Olive gray. Dry. Medium dense. No reaction to HCl. (9.5-9.7)
S-4	13.2-14.7	31	2.4	-	Clean SAND, poorly graded. Maximum size about 5mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.
			4.3	-	Silty SAND. Maximum size about 4mm. About 75% angular to subrounded, fine to medium sand and 2[illegible]% non-plastic fines. Varicolored to olive black. Dry. Dense. Weak reaction to HCl. (13.6-14.7)
S-5	18.2-19.7	21	2.7	-	Slightly silty SAND, poorly graded. Maximum size about 10mm. About 95% angular to subrounded, fine to coarse sand and 5% non-plastic fines. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl.
S-6	23.2-24.7	16	2.2	-	Clean SAND, poorly graded. Maximum size about 10mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl.
S-7	28.2-29.7	35	2.4	-	Clean gravelly SAND, well graded. Maximum size about 30mm. About 80% angular to subrounded, fine to coarse sand and 20% fine to coarse gravel. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.
S-8	33.2-34.7	41	2.7	-	Slightly silty gravelly SAND, poorly graded. Maximum size about 20mm. About 85% angular to subrounded, fine to coarse sand, 10% fine gravel and 5% non-plastic fines. Varicolored to olive black. Dry. Dense. Weak reaction to HCl.
S-9	38.2-39.7	103	7.2	-	Silty SAND, poorly graded. Maximum size about 3mm. About 85% angular to subrounded, fine to medium sand and 15% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-10	43.2-44.7	97	5.9	-	Silty SAND, poorly graded. Maximum size about 5mm. About 85% angular to subrounded, fine to coarse sand and 15% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl 24
S-11	48.2-49.7	97	2.5	-	Slightly silty SAND, poorly graded. Maximum size about 7mm. About 90% angular to subrounded, fine to medium sand and 10% non-plastic fines. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING E-6

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION
S-1	0.0-1.5	5	4.4	-	Slightly silty SAND, poorly graded. Maximum size about 3mm. About 95% angular to subrounded, fine to medium sand and 5% non-plastic fines. Dark yellowish brown. Dry. Loose. No reaction to HCl.
S-2	3.5-5.0	7	4.6	-	Slightly silty SAND, poorly graded. Maximum size about 4mm. About 90% angular to subrounded, fine to medium sand and 10% non-plastic fines. Dark yellowish brown. Dry. Loose. No reaction to HCl.
S-3	8.5-10.0	14	5.7	-	Silty SAND, poorly graded. Maximum size about 6mm. About 85% angular to subrounded, fine to medium sand and 15% non-plastic fines. Varicolored to olive gray. Dry. Medium dense. Weak reaction to HCl.
S-4	13.5-15.0	26	2.9	-	Clean SAND, poorly graded. Maximum size about 10mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl.
S-5	18.5-20.0	51	2.6	-	Clean gravelly SAND, poorly graded. Maximum size about 20mm. About 85% angular to subrounded, fine to coarse sand and 15% fine gravel. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-6	23.5-25.0	57	3.0	-	Slightly silty gravelly SAND, well graded. Maximum size about 25mm. About 80% angular to subrounded, fine to coarse sand, 15% fine to coarse gravel and 5% non-plastic fines. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-7	28.5-30.0	88	-	-	Slightly silty SAND, poorly graded. Maximum size about 3mm. About 90% angular to subrounded, fine to medium sand and 10% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-8	33.5-35.0	106	2.6	-	Slightly silty SAND, poorly graded. Maximum size about 20mm. About 95% angular to subrounded, fine to medium sand and 5% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-9	38.5-40.0	96	3.5	-	Slightly silty SAND, poorly graded. Maximum size about 4mm. About 90% angular to subrounded, fine to medium sand and 10% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-10	43.5-45.0	127	4.0	-	Slightly silty SAND, poorly graded. Maximum size about 3mm. About 90% angular to subrounded, fine to medium sand and 10% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-11	48.5-50.0	99	2.3	-	Slightly silty SAND, poorly graded. Maximum size about 3mm. About 95% angular to subrounded, fine to medium sand and 5% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-12	53.5-54.8	163/10"	6.4	-	Slightly silty gravelly SAND, well graded. Maximum size about 25mm. About 85% angular to subrounded, fine to coarse sand, 10% fine to coarse gravel and 5% non-plastic fines. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl. 24

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING E-11

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION
S-14	58.6-60.1	117	1.4	-	Clean SAND, poorly graded. Maximum size about 20mm. Angular to subrounded, fine to medium sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-15	63.6-65.1	93	1.4	-	Clean SAND, poorly graded. Maximum size about 12mm. Angular to subrounded, fine to medium sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.
S-16	68.6-70.1	98	1.4	-	Slightly silty SAND, poorly graded. Maximum size about 7mm. About 90% angular to subrounded, fine to medium sand and 10% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-17	73.6-75.1	127	1.7	-	Slightly silty gravelly SAND, well graded. Maximum size about 35mm. About 75% angular to subrounded, fine to coarse sand, 20% fine to coarse gravel and 5% non-plastic fines. Varicolored Dry. Very dense. Weak reaction to HCl.
S-18	78.6-80.1	95	1.7	-	Clean gravelly SAND, well graded. Maximum size about 30mm. About 65% angular to subrounded, fine to coarse sand and 35% fine to coarse gravel. Varicolored. Dry. Very dense. Weak reaction to HCl.
S-19	83.6-85.1	135	1.4	-	Clean gravelly SAND, well graded. Maximum size about 25mm. About 75% angular to subrounded, fine to coarse sand and 25% fine to coarse gravel. Varicolored. Dry. Very dense. Weak reaction to HCl.
S-20	88.6-90.1	64	1.5	-	Clean sandy GRAVEL, well graded. Maximum size about 35mm. About 60% angular to subrounded, fine to coarse gravel and 40% fine to coarse sand. Varicolored. Dry. Very dense. Weak reaction to HCl.
S-21	93.6-95.1	108	1.7	-	Clean gravelly SAND, well graded. Maximum size about 30mm. About 55% angular to rounded, fine to coarse sand and 45% fine to coarse gravel. Varicolored. Dry. Very dense. Weak reaction to HCl.
S-22	96.8-97.1	86/4"	2.0	-	Slightly silty SAND, well graded. Maximum size about 20mm. About 90% angular to subrounded, fine to coarse sand and 10% non-plastic fines. Varicolored. Dry. Very dense. Weak reaction to HCl. 24

TABLE 2Q B-1

SOIL SAMPLE DESCRIPTIONS

BORING E-16

SAMPLE NUMBER	DEPTH (FEET)	SPT (BLOWS/FT)	NATURAL WATER CONTENT	OTHER TESTS	SAMPLE DESCRIPTION
S-1	0.0-1.5	5	5.1	-	Clean SAND, poorly graded. Maximum size about 3mm. Angular to subrounded, fine sand. Varicolored to dark yellowish brown. Dry. Loose. No reaction to HCl.
S-2	3.5-5.0	5	3.7	-	Clean SAND, poorly graded. Maximum size about 3mm. Angular to subrounded, fine sand. Varicolored to dark yellowish brown. Dry. Loose. No reaction to HCl.
S-3	8.5-10.0	19	3.3	-	Clean SAND, poorly graded. Maximum size about 2mm. Angular to subrounded, fine sand. Varicolored to dark yellowish brown. Dry. Medium dense. No reaction to HCl. (8.5-8.8)
			4.7	-	Silty SAND, poorly graded. Maximum size about 15mm. About 85% angular to subrounded, fine to medium sand and 15% non-plastic fines. Varicolored to olive gray. Dry. Medium dense. Weak reaction to HCl. (8.8-9.6)
			2.0	-	Clean SAND, poorly graded. Maximum size about 10mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl. (9.6-9.8)
S-4	13.5-15.0	13	2.1	-	Clean SAND, poorly graded. Maximum size about 10mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl.
S-5	18.5-20.0	25	3.0	-	Clean SAND, poorly graded. Maximum size about 15mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Medium dense. Weak reaction to HCl.
S-6	23.5-25.0	105	3.4	-	Clean SAND, poorly graded. Maximum size about 10mm. Angular to subrounded, fine to coarse sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl. (23.5-23.9)
			7.6	-	Silty SAND, poorly graded. Maximum size about 3mm. About 85% angular to subrounded, fine to medium sand and 15% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl. (23.9-24.7) 24
S-7	28.6-30.1	84	5.2	-	Silty SAND. Maximum size about 2mm. About 70% angular to subrounded, fine sand and 30% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl. 24
S-8	33.6-35.1	118	6.7	-	Silty SAND. Maximum size about 12mm. About 80% angular to subrounded, fine to medium sand and 20% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-9	38.6-40.1	106	-	-	Silty SAND. Maximum size about 1mm. About 80% angular to subrounded, fine to medium sand and 20% non-plastic fines. Varicolored to olive gray. Dry. Very dense. Weak reaction to HCl.
S-10	43.6-45.1	155	3.7	-	Clean SAND, poorly graded. Maximum size about 10mm. Angular to subrounded, fine to medium sand. Varicolored to olive black. Dry. Very dense. Weak reaction to HCl.

Unit I	Commonly low neutron count rates in upper part and moderate to high gamma activity. Normally includes high gamma spike of A-horizon at top. Defined to include basal conglomeratic unit with high neutron count rates and low gamma count rates. Basal unit may include a fine-grained section with low neutron counts and high gamma activity.
Unit II	Normally a section with low neutron count rates and moderate gamma activity with some gamma spikes. A gravel bed displaying increased neutron and gamma count rates often occurs near base. Low neutron sections difficult to distinguish from low neutron sections of Unit I.
Unit III	Generally displays alternating low and high neutron count rates associated with low-to-moderate and high gamma count rates, respectively. These resonses correspond to the alternating zones of fine clastics and medium-to-coarse clastics. A gravel or sand frequently appears at the base and displays high neutron and gamma activity.
Unit IV	Ordinarily displays a relatively thick zone of high neutron and low-to-moderate gamma activity at the base. An upper section which displays low or moderate neutron activity and moderate gamma activity may or may not be present, depending on post-depositional erosion. The top of Unit IV is taken at the reduction in gamma activity which is normally found at the base of the pre-Missoula Flood Gravels. 24 If this response is not pronounced, and it may not be, the contact is taken at the nearest significant change in neutron activity.

Table 2R-3. Typical Geophysical Characteristics of Units

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3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

24

(See 251 NSSS GESSAR for a complete discussion of NSSS equipment classification.)

3.2.0 STRUCTURES, COMPONENTS AND SYSTEMS IMPORTANT TO SAFETY

See 251 NSSS GESSAR, and Tables 3.2-3 throughout 3.2-22. 11

3.2.1 SEISMIC CLASSIFICATION

Plant structures, systems and components important to safety are designed to withstand the effects of a Safe Shutdown Earthquake (SSE) and remain functional if they are necessary to assure those three items outlined below.

Seismic Category I structures, components, and systems are defined in accordance with USAEC Regulatory Guide 1.29 as those necessary to assure:

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe condition or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential off-Site exposures comparable to the guideline exposures of 10 CFR 100.

Plant structures, systems and components, including their foundations and supports, designed to remain functional in the event of a Safe Shutdown Earthquake, are designated as Seismic Category I, as indicated in Table 3.2-1.

Structures, components, equipment and systems designated as Safety Class 1, Safety Class 2, or Safety Class 3 (see Section 3.2.3 for a discussion of Safety Classes) are classified as Seismic Category I with the exception of:

a. Those portions of the radioactive waste treatment, handling, and disposal systems whose postulated

simultaneous failure would not result in conservatively calculated potential off-Site exposures comparable to the guideline exposures of 10 CFR 100. 11

All Seismic Category I structures, systems and components will be analyzed under the loading conditions of the SSE and OBE. Since the two earthquakes vary in intensity, the design of Seismic Category I structures, components, equipment and systems to resist each earthquake and other loads will be based on levels of material stress or load factors, whichever is applicable, and will yield margins of safety appropriate for each earthquake. The margin of safety provided for Safety Class structures, components, equipment and systems for the SSE will be sufficiently large to assure that their design functions are not jeopardized.

Seismic Category I structures, components, and systems will be designed to withstand the appropriate seismic loads as discussed in Section 3.7, Seismic Design; Section 3.10, Seismic Design of Electrical Equipment; and other applicable loads (as discussed in Sections 3.3, 3.4, 3.5, and 3.6) without loss of function. In accordance with AEC Regulatory Guide 1.29, Seismic Category I structures will be isolated or protected from other structures to ensure that their integrity will be maintained at all times. Seismic Category I requirements will extend to the first seismic restraint at the defined boundaries. Those structures, components, and systems that are Non-Seismic Category I are designated Seismic Category II. Seismic Category II components (and their supporting structures), whose collapse could result in a loss of required function through impact on or flooding of Seismic Category I structures, equipment, or systems, will be checked analytically to determine that the Seismic Category II components will not collapse when subjected to the various loadings such as seismic, tornado, flood, missile, etc. Structures, components or systems which form interfaces between Seismic Category I and Seismic Category II features will be designed to Seismic Category I requirements.

By definition IEEE Class I and Class IE components and systems (and their supports) will be Seismic Category I.

The seismic classification indicated in Table 3.2-1 meets the requirements of AEC Safety Guide 29 except as otherwise noted in the table. Where only portions of systems are identified as Seismic Category I on Table 3.2-1, the boundaries of the Seismic Category I portions of the system are shown on the Piping and Instrumentation Diagrams (P&ID's) in appropriate sections of this report.

P = Puget Sound Power & Light Company

(c) 1,2,3, OTHER = SAFETY CLASSES DEFINED IN SECTION 3.2.3.

(d) A = Auxiliary Building
B = Control Building
C = part of, or within, Containment
F = Fuel Building
M = Any Other Location
O = Outdoors On-Site
P = Pump House
R = Diesel Generator Building
T = Turbine Building
W = Radwaste Building

(e) The equipment shall be constructed in accordance with the codes listed on Table 3.2-2.

(f) Notations for principal construction codes:

(The earliest editions or revisions of Codes and Standards to be used for the design and construction of specific structures, systems, and components are listed in the appropriate PSAR sections. More recent revisions issued during design and construction of the S/HNP will be evaluated and adopted if practicable. A list of Codes and Standards, including revision dates, applied during the design and construction of the S/HNP will be supplied in the FSAR.) 23

AFBMA = Anti-Friction Bearing Manufacturer's Association

AGMA = American Gear Manufacturer's Association

AISC = American Institute of Steel Construction

AMCA = Air Moving and Conditioning Association Publication 211A, "AMCA Certified Ratings Program for Air Performance" or AMCA Standard 210, "Test Codes for Air Moving Devices" will be utilized for blower design purposes.

ANI = American Nuclear Insurers (formerly NEL-PIA)

API-620 = American Petroleum Institute, Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks

API-650 = American Petroleum Institute, Welded Steel Tanks for Oil Storage

ARI = Air Conditioning and Refrigeration Institute, "Standards for Application and Ratings of Centrifugal Water Chilling Packages", ARI 550-69, "Standard Method of Testing for Rating Forced Circulation Air Coolers and Air Heating Code", ARI 420-64.

ASHRAE = American Society of Heating, Refrigerating and Air Conditioning Engineers "Test Standards for Air Cleaning Devices Used in General Ventilation for Removing Particulate Matter", ASHRAE Standard 52-68.

AWS D1.1 = American Welding Society Structural Welding Code

A14.3 = ANSI A14.3, Safety Code for Fixed Ladders

B9.1	= ANSI B9.1, Safety Code for Mechanical Refrigeration
B30.2	= ANSI B30.2-1973, Overhead and Gantry Cranes
B30.10	= ANSI B30.10, Cableways, Cranes, Derricks, Hoists, Hooks, Jacks, and Slings
B31.1	= Code for Power Piping - 1973
DEMA	= Diesel Engine Manufacturers Association Standards
CMAA-70	= Crane Manufacturer's Association of America Specification for Electric Overhead Travelling Cranes
D100	= American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage AWWA D100
HEI	= Heat Exchange Institute
HSI-306	= Health and Safety Information, USAEC, Revised Minimal Specification for the High-Efficiency Particulate Air Filter, Issue No. 306
Hyd. I	= Hydraulic Institute
IEEE-279	= IEEE-279, Criteria for Protection Systems for Nuclear Power Generating Stations - 1971
IEEE-308	= IEEE-308, Standard Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations - 1974
IEEE-317	= IEEE-317, Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations - 1972
IEEE-323	= IEEE-323, Standard for Qualifying Class IE Electric Equipment Nuclear Power Generating Stations - 1974 | 23
IEEE-344	= IEEE-344, Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations - 1975 | 24
IEEE-379	= IEEE-379, Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE-387	= IEEE-387, Criteria for Diesel-Generator Units applied as Standby Power Supplies for Nuclear Power Generating Stations - 1972
IPCEA	= Insulated Power Cable Engineers Association
MFR. STD.	= Manufacturer's Standard
NEMA	= National Electrical Manufacturers Association
NFPA	= National Fire Protection Association
ORNL-NSIC	= Oak Ridge National Laboratory - Nuclear Information Center

3.4 WATER LEVEL (FLOOD DESIGN)

23

3.4.1 FLOOD PROTECTION MEASURES FOR SEISMIC CATEGORY I STRUCTURES

Structures housing safety-related components of the Skagit/
Hanford Nuclear Project include the Reactor Building, the
Auxiliary Building, the Control Building, the Fuel
Building, and the Ultimate Heat Sink. The location of
these structures is shown on Figure 2.1-2. Finished Plant 24
grade will be at elevation 526.5'. The lowest elevation of
the bottom of any opening into the structures will be at
elevation 527'0".

Flooding of the Site, as a result of floods on the Columbia
River, is addressed in Section 2.4.3. Probable maximum
floods were considered in accordance with NRC Reg. Guide
1.59. The maximum flood elevation at River Mile 361.5,
(location of raw water pumphouse) generated by a Probable
Maximum Flood, would be 404.4 ft for the existing level of
regulation on the Columbia River. A dam break flood on the
Columbia River is considered in Section 2.4.2. Maximum
flood elevation at River mile 361.5, produced by the dam- 24
break flood, will be 447.6. Both hypothesized flood levels
due to Columbia floods are well below the lowest opening at
elevation 527'0", indicating that flooding due to the 24
Columbia River is not a concern.

Section 2.4.3 considered a flood generated by a Probable
Maximum Precipitation (PMP) event on the watershed
surrounding the Plant Site, an area of 1.3 square miles.
The maximum flood elevation of the Plant Site, produced by
this PMP generated flood and including possible wind
effects, would be 526'0". Thus, the critical elevation of
527'0" is 1 foot above the flood level and no safety- 24
related structures or systems will be affected by local
flooding.

The area around the Plant Site is semi-arid (see Section
2.4.1). Recorded ground water levels at the Site are 125
feet below the ground surface (see Section 2.5.4.6). No
part of the plant structures is far enough below ground
surface to be affected by ground water. Local flooding,
generated by severe storms, would be of short duration
since the drainage area covers only 1.3 square miles.
Thus, temporary rises of the groundwater table cannot occur
as a result of storms. Grading of the Plant Site, shown on
Figure 2.4-3, will produce surface drainage away from the 24
Plant in all directions.

Because flooding will not impact any of the structures housing safety-related components or systems, no special flood protection requirements will be required. 23

The pumphouse, which provides makeup water for the Plant will be located on the west bank of the Columbia River at approximately River Mile 361.5. This pumphouse is not safety-related and, thus, if it were impacted by a PMF or a dam-break flood on the Columbia River, the safety of the Plant would not be affected.

3.4.1.2 Permanent Dewatering System

Because the ground water table of the Site is 125 feet below ground surface (see Section 2.5.4.6) groundwater will not affect the operation of any of the safety related systems or components. No permanent dewatering systems will be required.

3.4.2 INTERNAL FLOODING 23

The maximum postulated internal flood would result from a failure of the Circulating Water System in conjunction with a failure of the circulating water pumps to trip.

The ECCS pump rooms will be separate watertight compartments with their entrances located well above the level that the Circulating Water System flood could reach, and the water systems inside these rooms will be Seismic Category I systems. 020.21 020.22 042.14

There will be no connection between the Diesel Generator Building and other Plant buildings. The large water systems in the Diesel Generator Building will be Seismic Category I and the three generator rooms will be separated from each other.

The UHS Complex will be separated from the other Plant buildings, and each of the safety-related pumps will be located in separate rooms containing only Seismic Category I piping.

The Control Building, which is accessed from the Turbine Building at Elevation 527' through a corridor and lobby, will be protected against the maximum postulated internal flood by use of pressure relief panels in the railroad access door at Elevation 527' in the Turbine Building and by elevating the access door to the Control Buildir to provide a sill location at Elevation 531'. In the unlikely event that the Turbine Building floods to approximately 23 24

Elevation 531', water pressure will cause the railroad door
panels to open, releasing the flood water to the outside. 24
The lobby and corridor are separated by a water resistant
door as further protection.

Flooding in the Auxiliary Building caused by a failure of 23
the Circulating Water System will be detected by use of
flood level detectors which trigger Control Room alarms.
Redundant Class IE detectors will be located 3 inches above
the floor in the Fuel and Auxiliary Buildings. Operator
action is then relied upon to limit the effects of Auxiliary 24
and Fuel Building flooding.

The condensate and refueling water storage tanks will be
surrounded by a curb about 12 inches in height to provide 11
collection and a drainage path for tank leakage. Other
tanks located outside the main Plant buildings as well as
tanks in the Turbine, Auxiliary and Fuel Buildings will not 020.21
cause a flood of sufficient depth to endanger any safety- 020.22
related equipment. The tanks in the Control Building will 042.14
be small enough that any water lost from them will be within
the capacity of the floor drain system.

b. For missiles which are generated by release of stored strain energy, the strain energy is equated to kinetic energy in determining missile velocity. The ultimate stress of the material is used resulting in a larger amount of energy than would be present at fracture. Losses due to heating, friction, and the relaxation of the material are ignored.

c. For missiles acted upon by a single phase liquid stream for a certain distance, kinetic energy for these missiles is determined by converting work energy into kinetic energy. The calculational technique used in obtaining missile velocity is found in ORNL-NSIC-22, Subsection 4.1.1.

3.5.4 BARRIER DESIGN PROCEDURES

The tornado-generated missiles considered in the design of Seismic Category I structures are listed in Table 3.5-2.

The wall and roof thicknesses provided to resist the effects of tornado-generated missiles are considered to be more than adequate. At least 21 in. of reinforced concrete is provided for the missile resisting walls. At least 16 in. of reinforced concrete ($f_c' = 4000$ psi) is provided for roof slabs with removable forming, or above the flutes where metal form deck is used. However, the design will consider the contribution of the permanent presence of the metal form deck to prevent spalling. If it is shown that the metal form deck is effective, the concrete thickness above the flutes will be reduced accordingly, but will not be less than 12 in. (Figure 3.5-4). 310.6,33

In general, protection for internal missiles is provided by barriers. The procedures and calculations employed in design of missile-resistant barriers for both internal and external missiles are described in Bechtel Topical Report "Design of Structures for Missile Impact" (BC-TOP-9A) (Ref 10), with the following modifications: 220.17

a. In general the ductility ratio for flexure in steel will be limited to 10.0. Justification to the satisfaction of NRC staff will be provided when ductility ratios will be greater than 10. h220.1

b. The ductility ratio of steel columns will be limited to 1.0 when the slenderness ratio is more than 20.0. For columns with a slenderness ratio

equal to or less than 20.0, a ductility ratio less than or equal to 1.3 will be used. H220.1

A typical analysis to determine structural response due to impact by an automobile (unconventional missile) will include the following steps: The force-time history of the automobile is obtained from Ref 10. The area on which the force acts is assumed to be equal to the missile frontal area. The fundamental frequency of the combined wall and missile is calculated and the dynamic load factor is determined. An equivalent static force (the dynamic load factor multiplied by the maximum force from the force-time history) is applied to the structure and the maximum reactions, such as moments and shears, are determined. The capacity of the wall to resist punching shear caused by the automobile is analyzed by the conventional procedures of the ACI Building Code (ACI-318), using an allowable punching shear stress of $4\sqrt{f'_c}$.

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3.5.5 MISSILE BARRIER FEATURES

The preliminary layout and principal design features of structures serving primarily as missile-resistant barriers are shown on Figures 3.5-2 and 3.5-3. Detailed sketches will be provided in the FSAR.

3.7.1.4 Supporting Media for Category I Structures

Seismic Category I structures will have concrete mat foundations founded on soil. Information on soil depth, layering, and physical characteristics may be found in Section 2.5 of this PSAR.

The Reactor Building, Auxiliary Building, Control Building and Fuel Building are all located on a common basemat. The basemat will be approximately 20 ft thick, with the top surface at or near the finished grade. The plan dimensions will be approximately 280 ft by 290 ft.

The Ultimate Heat Sink is a box-like structure, approximately 200 ft by 200 ft in plan, with an embedment depth of about 45 ft to the bottom of the basemat. The Standby Service Water Pump Structure is part of the Ultimate Heat Sink.

The Diesel Generator Building is a surface founded structure, approximately 94 ft by 115 ft in plan, with a basemat approximately 7 ft thick.

The Diesel Generator Fuel Oil Tank Vault will be described in the project FSAR. 23

A more complete description of all these structures is given in Section 3.8.4 of this PSAR.

3.7.2 SEISMIC SYSTEMS

3.7.2.1 Seismic Analysis Methods

Seismic Category I structures are listed in Section 3.2 of this PSAR. These structures are analyzed for both the OBE and the SSE conditions. The procedures used to create the analytical models are described in Section 3.7.2.3.

The mathematical method used to solve the equations of motion for a particular structure will be the modal superposition method or the complex response (frequency domain) method.

For modal superposition the fixed-base mode shapes and frequencies are calculated first. When soil-structure interaction is considered, the soil-structure mode shapes, frequencies, and composite modal damping are also calculated. The modal responses are then computed, using

either the time history method or the response spectrum method. For the time history method the modal responses are combined algebraically. For the response spectrum method the modal responses are combined as described in Section 3.7.2.7 of this PSAR. 23

In the complex response method, the input forcing function time history is separated into its frequency components by means of the Fourier transform. The structural responses are calculated in the frequency domain, and the inverse Fourier transform then gives the reponse time histories. Some variations of this method use modal properties to describe portions of the model.

When a modal analysis is performed, the significant modes will be chosen on the basis of frequency, participation factor and generalized mass. Sufficient mass points will be used to adequately define the mode shapes.

Consideration will be given to rocking and translational responses of structures and foundations in the dynamic seismic analysis. Maximum relative displacements among supports of structures, systems and components will also be considered. The computer code to be used for this analysis will be a code approved for use by the NRC. H220.4 H220.5

3.7.2.2 Natural Frequencies and Response Loads

Seismic loads for Category I structures will be summarized in the FSAR. If modal superposition analysis is used, the significant mode shapes and frequencies will be given. If complex response analysis is used, relevant transfer functions will be given. In addition, the response spectra at major Category I equipment locations will be provided.

3.7.2.3 Procedures Used for Analytical Modeling

Analytical models are developed for all Category I structures. The type of superstructure model used will be determined by the characteristics of the structure itself.

For symmetrical structures such as the Containment and Drywell, 2-dimensional lumped-mass stick models will be used. Structures which are highly complex or asymmetrical, such as the Auxiliary-Fuel-Control Building complex, will be represented by 3-dimensional finite element models.

Sufficient refinement will be provided in the models to adequately define significant mode shapes.

Subsystems are assumed rigid and their masses lumped into the supporting structural system whenever significant coupling between the primary (supporting) system and the secondary (supported) system does not occur. The decoupled subsystems are later analyzed using the response spectra generated at the supporting levels. 23

3.7.2.4 Soil-Structure Interaction

The input motion, as given in Section 3.7.1, is defined at the surface level in the free field. Because the presence of the Plant structures modifies this motion, a soil-structure interaction analysis will be performed.

The major Category I structures (the Auxiliary, Fuel, Control, and Reactor Buildings) will be constructed on a common basemat, approximately 20 ft thick. The soil-structure interaction analysis for these structures will be done with a combined model, using the "lumped parameter" approach. (This approach is also known as the "sub-structure," the "foundation impedance," and the "multistep" approach.) The decision to use this approach is based on the shallow embedment, relative to horizontal size, of the common basemat.

The analysis will consist of the following steps:

1) A free-field soil column analysis is performed, using the input acceleration time history defined in Section 3.7.1 as the surface control motion. Strain dependency of stiffness and damping will be considered, using an iterative equivalent linear method.

2) The soil impedance functions are calculated, using the soil stiffness and damping derived from the free-field analysis. Soil layering will be explicitly considered.

3) The base and structural responses are calculated using substructuring techniques.

Soil parameter variations will be accounted for by multiple analyses, using a realistic range of soil parameters.

A finite element analysis will be performed as a confirmation of the lumped-parameter approach. The analysis will utilize the FLUSH program or other program approved by the NRC. The size of the soil elements and locations of the H220.6

transmitting lateral boundary and the rigid base will be chosen in accordance with standard accepted procedures for that program. The structure will be modelled by beam and plane strain elements, with sufficient detail to simulate the dynamic properties of the system. 24

3.7.2.5 Development of Floor Response Spectra

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Floor response spectra will be developed using time histories of significant support points within the structures. The effects of the three components of ground motion will be combined as recommended in Regulatory Guide 1.122, Section C, Paragraph 2 or 3. Widening of spectral curves is described in Section 3.7.2.9 of this PSAR.

3.7.2.6 Three Components of Earthquake Motion

When the maximum response to the three components of earthquake motion have been calculated separately, the maximum response to the total motion will be taken as the square root of the sum of the squares of the component responses.

When three time histories are applied to a model simultaneously, the maximum responses will be taken as the maximum of the algebraic combinations of the responses to the three components.

3.7.2.7 Combination of Modal Responses

Where the response spectrum method of analysis is used, the modal responses will be combined by the "grouping method" described in Section C, Paragraph 1.2.1 of Regulatory Guide 1.92.

3.7.2.8 Interaction of Non-Category I Structures with Category I Structures

Non-Category I structures whose collapse could result in the loss of required function of Category I structures, equipment or systems required for safe shutdown after an earthquake will be analytically checked to determine that they will not collapse when subjected to a Safe Shutdown Earthquake.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra 23

The effects of parameter variations on floor response spectra shall be considered by widening the spectra, using the following procedure:

Let f_j be the structural frequency, which is determined by using the most probable material and section properties in formulating the structural model. The variation in the structural frequency is determined by evaluating the individual frequency due to the most probable variation in each parameter that is of significant effect, such as soil modulus, material density, material stiffness, etc. The total frequency variation, $\pm\Delta f_j$, is then determined by taking the square root of the sum of squares of a minimum variation of $0.05f_j$ and the individual frequency variation $(\Delta f_j)_n$, that is:

$$\Delta f_j = \left[(0.05f_j)^2 + \sum_n (\Delta f_j)_n^2 \right]^{1/2} \qquad (5\text{-}1)$$ 24

A value of $0.1f_j$ is used if the actually computed value of Δf_j is less than $0.10f_j$.

3.7.2.10 Use of Constant Vertical Static Factors

Constant vertical static factors will not be used for Category I structures.

3.7.2.11 Methods Used to Account for Torsional Effects

Generally Category I structures with low eccentricity, such as containment, will be analyzed using 2-dimensional stick models. A static factor will be used to account for this eccentricity in design. H220.7

Structures with significant 3-dimensional properties will be modelled using finite element models which will account explicitly for torsion.

To account for accidental torsion in both of the above cases an additional eccentricity, applied statically, of 15% of the maximum building dimension at the level under consideration shall be assumed for structural design.

3.7.2.12 Comparison of Responses

When different analysis methods are used, a comparison of the responses will be provided for the operating license review.

3.7.2.13 Methods for Seismic Analysis of Category I Dams

This project has no Category I dams.

3.7.2.14 Determination of Category I Structure Overturning Moments

Overturning moments will be determined using the results of the dynamic analyses. Three components of input motion will be included, as well as a conservative evaluation of vertical and lateral seismic forces.

3.7.2.15 Analysis Procedure for Damping

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For cases where modal analysis is used, one of three techniques may be used to account for damping in different elements of the models: mass weighting, stiffness weighting, or dissipating energy technique. These techniques will produce composite modal damping values. Where modal analysis is used for a soil-structure system, Tsai's method (Ref 5) may be used.

For cases where complex response (frequency domain) analysis is used, damping is considered by forming a complex-valued stiffness matrix.

3.7.2.16 Seismic Analysis of Radwaste Building

A modified seismic analysis will be used for the foundation and walls of the Radwaste Building, at least up to a height sufficient to contain the liquid inventory in the building. This modified analysis will comply with Regulatory Guide 1.143, Revision 1, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

3.7.4 SEISMIC INSTRUMENTATION PROGRAM

3.7.4.1 Comparison with NRC Regulatory Guide 1.12 12

Seismic instrumentation will be provided to furnish at least as much data about the response of Seismic Category I structures and equipment as the instrumentation suggested in NRC Regulatory Guide 1.12 for sites with SSE ground acceleration of 0.3g or greater. This instrumentation will provide a means to obtain at least the following information: 12

a. Records of the acceleration time history in three orthogonal directions at the following locations:

(1) Free Field

(2) Containment Base

(3) Containment Operating Floor

(4) Auxiliary Building Base

(5) One of the floors of the Auxiliary Building where major Seismic Category I equipment or piping is supported.

b. Triaxial peak accelerations at the following locations: 23

(1) One location on reactor equipment (using peak strain gages)

(2) One location on reactor piping (using peak strain gages)

(3) One location on Seismic Category I equipment in the Auxiliary Building

(4) One location on Seismic Category I piping in the Auxiliary Building.

c. Triaxial response spectrum recorders at the following locations:

(1) Reactor equipment support (or piping support)

(2) Auxiliary Building base

(3) One of the floors of the Auxiliary Building where Seismic Category I equipment is supported 23

(4) One of the Seismic Category I piping supports in the Auxiliary Building

(5) Containment Base, with indication in the control room.

d. Triaxial seismic switches, with indication in the control room when OBE acceleration has been exceeded, at the following locations:

(1) Containment Base

(2) Reactor equipment support (or piping support).

Instruments will provide sufficiently accurate data for the subsequent analyses of the Plant components.

3.7.4.2 Location and Description of Instrumentation

The types, location, basis for selection of location, and operational capability of seismic instrumentation that will be installed for Seismic Category I structures and components will be described in the FSAR. It is intended to provide seismic instrumentation which, when used with the Plant operating instrumentation, will provide sufficient information to determine the Plant's capability for continued use following the occurrence of an earthquake.

3.7.4.3 Control Room Operator Notification

When the acceleration at the base of the Containment or at a reactor equipment support (or piping support) exceeds the comparable OBE acceleration both an audible and visual annunciation will be made in the control room. In addition, the triaxial time-history accelerograph located in the containment foundation or in the free field will be connected to the control room, for indication of acceleration level to the control room operator. The response spectrum recorder in the reactor containment foundation will also be connected to the control room to indicate if the design response spectra values for discrete frequencies are exceeded during an earthquake. 12 H220.8

3.7.4.4 Comparison of Measured and Predicted Responses

Response spectra and peak accelerations obtained by means of instrumentation will be compared with the design response spectra and calculated peak accelerations for the same location. If the recorded responses exceed the OBE responses used in the design of the Plant, the Plant will be shut down and a detailed analysis of the earthquake motion will be undertaken.

3.7.4.5 Inservice Surveillance

Each of the seismic instruments will be periodically demonstrated operable in accordance with the Plant Technical Specification requirements. Seismic instruments will be designed to ensure that channel checks, channel calibration and channel functional tests can be performed to a frequency consistent with that of NUREG 0800 SRP Table 3.7.4-2. H220.9

3.7.5 SEISMIC DESIGN CONTROL MEASURES

3.7.5.1 Seismic Category I Systems and Components other than NSS System

The seismic input data are provided to the suppliers of Seismic Category I equipment by means of detailed equipment specifications. One part of the specifications contains input data in the form of floor response curves (Subsection 3.7.2.5). The floor response curves are prepared by the responsible civil/structural engineering group. The detailed equipment specifications are prepared by the responsible mechanical, electrical, or instrumentation engineering group. The specification designates the particular floor response spectra curve(s) for the floor(s) on which the equipment or component is located.

The detailed equipment specifications require that the supplier submit test data and/or seismic analyses for review, as a condition of acceptance of the equipment for the intended function. The supplier is permitted to use:

a. Test reports of the particular component

b. Reports of tests with applicable data from a previously tested comparable component which during normal operating conditions has been subjected to equal or greater loadings, or

c. Suitable analytical results.

The report from the supplier is reviewed by the responsible engineering group. The group reviews the methods, procedures, and results for compliance with the criteria. The submittal and review procedures are repeated, if necessary, when questions are raised as to conformance with the criteria.

The design control measures will conform to the requirements of Chapter 17, Quality Assurance, of the PSAR.

3.7.5.2 NSS System

The NSS system supplier is provided with the floor response spectra. The specifications for the NSS system components are prepared by the NSS system vendor's engineering group to assure that seismic input for equipment design is appropriate for each component.

The seismic qualifications of NSSS equipment submitted by the equipment suppliers are reviewed and approved by the cognizant engineering unit of the NSSS supplier to assure that the loadings and the applications are appropriate.

The design control measures will conform to the requirements of Chapter 17, Quality Assurance, of the PSAR.

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Criterion Nos. 1, 2, 4, 16, 50, 51, 52, 53, 54, 55, 56, and 57. H220.13

b. NRC Regulatory Guides (compliance is discussed in Appendix 3A of this PSAR). 12

Regulatory Guide 1.10 - Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

Regulatory Guide 1.15 - Testing of Reinforcing Bars for Category I Concrete Structures

Regulatory Guide 1.18 - Structural Acceptance Test for Concrete Primary Reactor Containments

Regulatory Guide 1.19 - Non-Destructive Examination of Primary Containment Liner Welds

Regulatory Guide 1.29 - Seismic Design Classification

Regulatory Guide 1.46 - Protection against Pipe Whip Inside Containment

Regulatory Guide 1.54 - Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

Regulatory Guide 1.55 - Concrete Placement in Category I Structures

Regulatory Guide 1.57 - Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.

Regulatory Guide 1.63 - Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants.

Regulatory Guide 1.94 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Steel During the Construction Phase of Nuclear Power Plants. H220.12

c. Industry Standards

Nationally recognized industry standards, such as those published by the American Society for Testing and Materials (ASTM), are used whenever possible to describe material properties, testing procedures, and fabrication and construction methods. The

applicable ASTM specifications are listed in Section 3.8.1.7 of this PSAR.

d. Bechtel Power Corporation Topical Reports

BC-TOP-1[5]	Containment Building Liner Plate Design Report, Revision One, December, 1972 with Supplement and Conditions per letter dated February, 1974. 12
BC-TOP-3-A[1]	Tornado and Extreme Wind Design Criteria for Nuclear Power Plants, Revision Three, August, 1974.
BC-TOP-4-A[3]	Seismic Analysis of Structures and Equipment for Nuclear Power Plants, Revision Three, November 1974. 12
BC-TOP-5A[4]	Prestressed Concrete Nuclear Reactor Containment Structures, Revision Three, Feb, 1975. Only Section 6.0, Design, Section 7.0, Analysis and parts of Section 2.0, Physical Description, of this Topical Report apply to the Containment.
BC-TOP-9-A[2]	Design of Structures for Missile Impact, Revision Two, September, 1974. 12
BN-TOP-1[8]	Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants, Revision One, November, 1972.
BN-TOP-2[9]	Design for Pipe Break Effects, Revision Two, May, 1974.

3.8.1.3.4 Structural Specifications

Structural specifications are prepared to cover the areas related to design and construction of the Containment. These specifications are prepared by Bechtel Power Corporation specifically for this Containment. These specifications emphasize important points of the industry standards for this Containment and reduce options such as would otherwise be permitted by the industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards and as

such need not be included in the PSAR. These specifications cover the following areas:

a. Furnishing and delivery of concrete

b. Forming, placing, finishing, and curing of concrete

c. Furnishing, detailing, fabricating, delivering, and placing of reinforcing steel

d. Furnishing, delivery, and installation of exothermic splicing

e. Furnishing, delivery, and erection of liner plate and penetration assemblies

b. ACI 306-66 - Recommended Practice for Cold-Weather Concreting is used except as follows:

During cold weather concreting members shall be enclosed in an ambient temperature within 10°F of the tempreature of the concrete as placed.

c. ACI 347-68 - Recommended Practice for Concrete Form- 12
work is used augmented as follows:

Tolerance of surfaces from the theoretical radius:

1) Outside surfaces of containment wall & dome ± 1 inch.

2) Inside and outside surfaces of the drywell 12
wall: ± 1/2 inch.

Tolerances from plumb:

1) Outside surface of the containment wall: 1/2 inch in 10 feet.

2) Inside and outside surfaces of the drywell wall: 1/2 inch in 10 feet.

d. ACI 305-72 - Recommended Practice for Hot-Weather Concreting is used except as follows:

All members 2-1/2 feet thick or larger shall have a placing temperature less than 70°F. All other concrete shall have a maximum placing temperature of 85°F.

e. ASTM C 94-74a - Ready-Mixed Concrete is used with- 12
out exception.

3.8.1.7.5.2 Steel construction.

a. AWS D1.1-75 - Structural Welding Code is used except:

1) As an alternative to AWS D1.1 paragraph 4.9.2
for the issuance of low hydrogen low alloy 24
electrodes, in lieu of the time limits of 4.9.2, the electrodes may be issued in portable warmers and as long as the warmers are energized at the work location, the electrodes in warmers may be out of the storage ovens

indefinitely, but shall be returned to the storage oven at the end of each shift.

2) Welding procedure specifications and personnel (welders, welding operators, and tackers) may be qualified to ASME Section IX in lieu of AWS D1.1 Section 5. 24

b. AISC - Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Sections 1.23 and 1.25, February 1969, including Supplement Nos. 1 and 2 is used without exception. Exceptions are taken to the limitations on depth of metal deck and stud spacing recommended in Supplement No. 3.

c. AISC - Specification for Structural Joints Using ASTM A 325-74 or A 490-75 Bolts is used without exception, load indicating washers may be used. 12

3.8.1.7.5.3 Linear plate erection. Vertical and dome liner plates are used forms and erection will precede the concrete placemen e in accordance with ASME Code Subarticle CC-450. 12

3.8.3.2 Applicable Codes, Standards, and Specifications

The following regulations, codes, standards, and specifications are used in the design of the containment internal structures:

3.8.3.2.1 Regulations

a. Code of Federal Regulations, Title 10-Atomic Energy Part 50, "Licensing of Production and Utilization Facilities."

b. Code of Federal Regulations, Title 29-Labor, Part 1910, "Occupational Safety and Health Standards."

3.8.3.2.2 Codes and Standard Specifications

Acceptance of the following codes and standards for design or for design bases does not constitute full compliance with them. Exceptions to these codes and standards are given in Subsection 3.8.1.7.5.

a. Uniform Building Code (UBC), 1973 edition, (applicable) and 1975 Supplements (applicable portions) 12

b. American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures" (ACI 349-76) H220.19

c. American Institute for Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," adopted February 12, 1969 and Supplement Nos. 1, 2 and 3

d. American Welding Society, "Structural Welding Code" (AWS D1.1-74).

e. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code "Nuclear Power Plant Components", Section III, Division 2 (1975 Edition) Subsection CC-3000. 12

f. American National Standards Institute, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural

Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (ANSI N45.2.5).

3.8.3.2.3 General Design Criteria, Regulatory Guides, Industry Standards, and Topical Reports

a. 10 CFR 50, Appendix A - General Design Criteria for Nuclear Power Plants. (Conformance is discussed in Section 3.1)

Criterion Nos. 1, 2, 3, 4 and 16. H220.13

b. NRC Regulatory Guides (conformance is discussed in Appendix 3A of this PSAR). 12

- o Regulatory Guide 1.10 - Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Containments
- o Regulatory Guide 1.15 - Testing Reinforcing Bars for Category I Concrete Structures
- o Regulatory Guide 1.29 - Seismic Design Classification
- o Regulatory Guide 1.46 - Protection against Pipe Whip Inside Containment
- o Regulatory Guide 1.55 - Concrete Placement in Seismic Category I Structures
- o Regulatory Guide 1.57 - Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
- o Regulatory Guide 1.94 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Steel During the Construction Phase of Nuclear Power Plants H220.12

c. Industry Standards

- o Nationally recognized industry standards, such as those published by the American Society for Testing and Materials (ASTM), are used whenever possible to describe material properties, testing procedures, fabrication methods, and construction methods. The ASTM specifications listed in Subsection 3.8.1.7 of

this PSAR are applicable to the internal structures.

d. Bechtel Power Corporation Topical Reports

oBC-TOP-4-A	Seismic Analysis of Structures and Equipment for Nuclear Power Plants, Revision Three, November, 1974	12
oBC-TOP-9-A	Design of Structures for Missile Impact, Revision Two, November, 1974	12
oBN-TOP-2[9]	Design for Pipe Break Effects, Revision Two, May 1974.	12

3.8.3.2.4 Structural Specifications

Structural specifications are prepared to cover the areas related to design and construction of the Plant structures. These specifications are prepared by Bechtel Power Corporation specifically for these structures. The specifications emphasize important points of the industry standards for these structures and reduce options such as would otherwise be permitted by the industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards and as such are not included in the PSAR. The specifications cover the following areas:

a. Furnishing and delivery of concrete

b. Purchasing, forming, placing, and curing of concrete

c. Furnishing, detailing, fabricating, delivery, and placing of reinforcing steel

d. Furnishing, delivery, and installation of exothermic splicing

e. Furnishing, delivery, and erection of structural steel

f. Furnishing, delivery, and erection of stainless steel liner plate for water-filled cavities (refueling canal, etc.).

3.8.3.3 Loads and Loading Combinations

With the exception of the drywell, the internal structures are designed for the loads and loading combinations given in Subsection 3.8.6, Structural Design Criteria for Category I Structures Other than Containment and Drywell. The loading combinations involving extreme wind, tornado, or flood forces are not applicable to the Containment internal structures. The concrete portions of the drywell and the drywell vent structure are designed for the loading combinations shown in Table 3.8-1 in accordance with the ASME Code Section III, Div. 2, Article CC-3000. For load definitions and nomenclature see Subsection 3.8.1.4. The steel portions of the drywell structure (Class MC components) are designed for the same combinations except that the load factor for all loads is 1.0 in accordance with the applicable requirements of the ASME Code Section III. Div. 1 Article NE-3000 and the guidelines of Regulatory Guide 1.57. The steel liner portions of the drywell vent structure shall be designed for the loading combinations in Table 3.8-1. Allowable stresses under service load conditions are as per AISC Specification, Part 1. The allowable load limits under factored load conditions are 90% of those in AISC Specification, Part 2. The drywell design accident pressure loads are 30 psig for a large pipe break and 5 psig for a small pipe break. For details of pressure and temperature transients, see Section 6.2.

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3.8.3.4 Design and Analysis Procedures

The basic techniques of analyzing the internal structures can be broadly classified into two groups:

a. Conventional methods involving simplifying assumptions such as those found in beam theory, and

b. Those based on plate and shell theories of different degrees of approximation.

The strength methods given in the ACI 349-76 code are used for design. The internal structures are provided with connections capable of transmitting axial and lateral loads to the Containment base slab. Table 3.8-2 lists the computer programs used for analysis. H220.19

The following computer programs may be used to evaluate the effect of radiation-generated heat on the shield structure of the internals:

a. Grace II[6] (NE-348) - This program solves multigroup, multiregion, gamma ray attenuation problems for gamma ray heating and also dose rates in infinite or semi-infinite slab shields with movable source regions.

b. Heating II[7] (ME-611) - This program solves transient and/or steady state heat transfer problems in three dimensions. (Cartesian, cylindrical, or spherical coordinates system).

In the final stages of design of the internal structures, the proportioning of reinforcing steel in concrete structures is based upon the specified codes of practice. The reinforcing steel is distributed according to common detailing methods. Likewise, the selection of structural steel sections and the methods of fabrication and connection are in accordance with engineering codes and accepted industry practices.

3.8.3.5 Structural Acceptance Criteria

Internal structures are designed for structural acceptance criteria as outlined in Subsection 3.8.3.2.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The internal structures are constructed of concrete and steel using proven methods common to heavy industrial construction. There will be no safety related masonry walls. Material properties and characteristics assumed in design are given in Table 3.8-3. H220.17

3.8.3.6.1 Concrete

Concrete is the same as that described in Subsection 3.8.1.7.1.a. High density concrete aggregates where used, conform to ASTM C 638-73, "Descriptive Nomenclature of Constituents of Aggregates for Radiation Shielding," and to ASTM C 637-73, "Specifications for Aggregates for Radiation Shielding Concrete." 12

3.8.3.6.2 Reinforcing Steel

Reinforcing steel is the same as that described in Subsection 3.8.1.7.2.

3.8.3.6.3 Structural Steel

Structural steel is the same as that described in Subsection 3.8.1.7.3.

3.8.3.6.4 Construction Procedures

The construction procedures are the same as those described in Subsection 3.8.1.7.5.

3.8.3.6.5 Quality Control

The quality control requirements are met as described in Subsection 3.8.1.7.5.6 and Chapter 17 of this PSAR.

3.8.3.7 Testing and Inservice Surveillance Requirements

With the exception of the drywell, a formal program of testing and inservice surveillance is not planned for the internal structures. The internal structures are not directly related to the functioning of the containment concept; hence, no testing or surveillance is required. Both drywells will be leak-rate and structurally tested. Alternate means of satisfying the high pressure test objectives will be considered as more experience with this test program becomes available in the industry. Periodic leak-rate tests will be conducted at a lower pressure. 130.15 130.32,28

A discussion of the preliminary drywell structural integrity test is presented in the following section. See Section 6.2.1.4 for a description of leak-rate testing. 130.26

3.8.3.7.1 General Test Procedure

As late as practical in the construction sequence but before initial fuel load, each drywell will be subjected to a 130.32

structural proof test at the design pressure of 30 psig. The drywell will be pressurized and depressurized in at least three approximately equal increments. During pressurization and depressurization of the Unit 1 drywell, the pressure at each stage will be held constant for about one hour before deflection and strain measurements are taken. Measurements will not be taken on the Unit 2 drywell, subject to the stipulations given below.

Radial deflections will be measured on the Unit 1 drywell at a minimum of three points along three or more meridians equally spaced around the drywell, including locations with varying stiffness characteristics. Radial deflections will be measured at the lower vent region, at about mid-height and near the top of the cylindrical wall. The measurement points may be varied depending on the anticipated distribution of stresses and deformation. 130.32

Strain measurements will be taken on the Unit 1 drywell, at the bottom of the wall and at the mid-height of the wall, on at least two opposing meridians. At each of these locations, strains will be measured at the center of the wall section and near the inner and outer faces of the wall section.

Deformations will be measured with internal taut wire devices. Uniaxial strains will be measured with Carlson Strain meters and with embedded reinforcing bars instrumented with strain gauges.

Strain levels will be correlated with deflection measurements during the test on the Unit 1 drywell. If these correlations are within predefined tolerances for predicted response, deflection and strain measurements will not be taken during the test on the Unit 2 drywell.

Pretest data for the Unit 1 drywell test will consist of predicted responses and allowable tolerances on the predicted responses.

Auxiliary Building and the Containment structure will allow free lateral movement during a seismic event in accordance with Section 4.1 of BC-TOP-4A. 12,23 H220.16

Emergency core cooling systems (ECCS), Reactor Water Cleanup System (RWCU) and Reactor Core Isolation Cooling System (RCIC) equipment are supported at the foundation level of the building in compartments with an elevated door. A controlled entrance for personnel is provided to this level. 12

Hatches in the concrete floor slab provide access to the RHR heat exchangers and to the secondary isolation valves for the main steam and reactor feedwater lines. The RHR heat exchangers are situated in vertical compartments on either side of the steam tunnel.

The only rooms in the Auxiliary Building which need to be designed to handle the consequences of high-energy pipe breaks are the RCIC and RWCU rooms and the main steam tunnel. 042.6 The RCIC and RWCU rooms will be designed for high-energy pipe breaks including the associated pressure, temperature and jet forces. Main steam and feedwater line breaks are not postulated in the main steam tunnel. However the tunnel vent size has conservatively been based on the energy released from a non-mechanistic blowdown of a main steam line as discussed in Section 3.6.1.4.

The other rooms need not provide for high-energy pipe breaks since the RHR system contains high-energy steam less than 2 percent of the time the Plant is in operation. 23 There are no other sources of internal pressurization in the Auxiliary Building and the highest pressure that can occur is atmospheric (except for RCIC & RWCU Room and the main steam tunnel discussed above).

As described in Section 6.5 and 9.4, the Standby Gas Treatment (SGTS) and Auxiliary Building HVAC Systems normally maintain the Auxiliary Building air pressure at 1/4 inch of water vacuum. Following an accident, one of the Standby Gas Treatment System (SGTS) fans controls the pressure in the Auxiliary Building at 1/4 inch of water vacuum by exhausting at a flow rate equal to the inleakage rate of one volume per day in the Auxiliary Building and the Enclosure Building. In the unlikely event that the SGTS exhaust fan recirculating damper V005 (Figure 6.5-1) fails closed, one of the 042.65 SGTS fans will draw air from these buildings (without recirculation) at a reduced fan flow rate due to higher system pressure drop. Consequently, a maximum vacuum of 1.7 inches of water in the Auxiliary Building will be produced using flow-pressure relationships:

$$\frac{P_1}{P_2} = \frac{Q_1}{Q_2}^{2}$$

Table 6.5-2 shows the operator action on the control failure mode. Except for the RCIC and RWCU room and the main steam 042.6 tunnel discussed earlier, the maximum and minimum design pressures for the Auxiliary Building are 0 and 5 in. of water vacuum, respectively. This represents a margin of 300 12 percent above the maximum vacuum that the SGTS can produce. The Auxiliary Building HVAC System cannot produce as high a negative pressure as the SGTS.

3.8.4.1.3 Fuel Building

The Fuel Building is located adjacent to the Containment and opposite to the Auxiliary Building (refer to the general arrangement drawings of Section 1.0). The building's principal function is housing the equipment and facilities for receiving, storing, shielding, shipping and handling of fuel. The building is a Seismic Category I concrete structure designed for tornado and missile protection. Figures 1.2-2 through 1.2-9 show the main structural features of the building.

The Fuel Building will be supported on the common power 23 block mat. The building is enclosed by reinforced concrete walls which support the floor framing. The central part of the building is occupied by the fuel pool and equipment com- 12 partments formed by concrete walls and slabs. Stainless steel liner plates seal the interior pool surfaces.

Liner plate joints are fitted with leak chases draining to a sump thus allowing testing and monitoring of leaktightness. Concrete floors surrounding the pool are supported on steel beams which frame to concrete walls or to columns bearing on the foundation. Weatherproof joints will allow lateral 12,23 seismic movement between the Containment and Fuel Building walls. Wall separation between the Fuel Building and the Containment structure will allow free lateral movement during a seismic event in accordance with Section 4.1 of BC- H220.16 TOP-4A.

The Fuel Building will normally be maintained at 1/4 inch of 042.6 water negative pressure by the H&V System. 12

As described in Sections 6.5 and 9.4, the Standby Gas Treatment (SGTS) and Fuel Building HVAC Systems normally maintain 042.6 the Fuel Building air pressure at 1/4 inch of water vacuum. Following an accident, one of the SGTS fans controls the pressure in the Fuel Building at 1/4 inch of water vacuum by exhausting at a flow rate equal to the inleakage rate of one volume per day in the Fuel Building. In the unlikely event 042.65 that the SGTS exhaust fan recirculating damper V005 (Figure 6.5-1) fails closed, one of the SGTS fans will draw air from this building (without recirculation) at a reduced fan flow rate due to higher system pressure drop. Consequently, a

R_a = Pipe reactions under thermal conditions generated by the postulated break and including R_o.

Y_r = Load on a structure generated by the reaction on a ruptured high-energy pipe during the postulated event. The time-dependent nature of the load and the ability of the structure to deform beyond yield is considered in establishing the structural capacity necessary to resist the effects of Y_r.

Y_j = Load on a structure generated by the jet impingement from a ruptured high-energy pipe during the postulated event. The time-dependent nature of the load and the ability of the structure to deform beyond yield is considered in establishing the structural capacity necessary to resist the impact.

Y_m = The energy resulting from the impact of a ruptured high-energy pipe on a structure or a pipe restraint during the postulated event. The type of impact, ie, plastic, elastic, etc., together with the ability of the structure to deform beyond yield is considered in establishing the structural capacity necessary to resist the impact.

3.8.6.1.5 Other Definitions

S = For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel Buildings," February 12, 1969. H220.19

U = For concrete structures, U is the section strength required to resist design loads, based on the strength design methods described in ACI 349-76. H220.19

Y = For structural steel, Y is in the section strength required to resist design loads, based on plastic design methods described in Part 2 of AISC "Specification for the Design, Fabrication and Erection of Structural Steel Buildings," February 12, 1969.

3.8.6.2 Load Combinations and Criteria for Seismic Category I Concrete Structures

The following presents a set of load combinations and allowable design limits used for Seismic Category I concrete structures. To assure that the structural integrity will be maintained, limits on the resulting stresses and the required strength capacities are considered for service loads, including earthquake (OBE) and wind loads, and for factored loads, including earthquake (SSE), tornado, and pipe break effects and various combinations thereof.

3.8.6.2.1 Load Combinations for Service Load Conditions

The strength design method is used, and the following load combinations are considered:

$$U = 1.4D + 1.7L \quad (3.8\text{-}1)$$

$$U = 1.4D + 1.7L_o + 1.9E_o \quad (3.8\text{-}2)$$

$$U = 1.4D + 1.7L + 1.7W \quad (3.8\text{-}3)$$

H220.19

If thermal stresses due to T_o and R_o are present, the following combinations are also used:

$$U = (0.75)(1.4D + 1.7L + 1.7T_o + 1.7R_o) \quad (3.8\text{-}4)$$

$$U = (0.75)(1.4D + 1.7L_o + 1.9E_o + 1.7T_o + 1.7R_o) \quad (3.8\text{-}5)$$

23

$$U = (0.75)(1.4D + 1.7L + 1.7W + 1.7T_o + 1.7R_o) \quad (3.8\text{-}6)$$

The cases of L having its full value or being completely absent are both checked and the following calculations are also satisfied:

$$U = 1.2D + 1.9E_o \quad (3.8\text{-}7)$$

$$U = 1.2D + 1.7W \quad (3.8\text{-}8)$$

3.8.6.2.2 Load Combinations for Factored Load Conditions

For these conditions, which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/-extreme environmental conditions, respectively, the strength design method is used and the following load combinations are considered:

$$U = D + L_o + T_o + R_o + (E_{ss} \text{ or } W_t \text{ or } V) \quad (3.8\text{-}9)$$

$$U = D + L + T_a + R_a + 1.5\ P_a \quad (3.8\text{-}10)$$

$$U = D + L_o + T_a + R_a + 1.25\ P_a + (Y_r + Y_j + Y_m) + 1.25\ (E_o \text{ or } W_t \text{ or } V) \quad (3.8\text{-}11)$$

H220.19

$$U = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + (E_{ss} \text{ or } W_t \text{ or } V) \quad (3.8\text{-}12)$$

In combinations (3.8-10), (3.8-11) and (3.8-12), the maximum effects of P_a, T_a, R_a, Y_j, Y_r, and Y_m are considered unless a time-history analysis is performed to justify otherwise.

For combinations (3.8-9) to(3.8-12), strains due to T_a and due to the dynamic effects of W_t (tornado missile impact), P_a, Y_r, Y_j, and Y_m may exceed the allowable strains, provided there will be no loss of function of any safety-related system. 16 24

In combination (3.8-10), to account for the effect of SRV loads on containment internals, the load factor of L shall be increased to 1.25. H220.15

Whenever strains are permitted to exceed yield due to a certain type of load, the structure is checked to satisfy that its ability to carry other loads is not jeopardized.

The cases of L having its full value or being completely absent are both checked.

The effects of tornado-generated differential pressures and missiles are combined in accordance with BC-TOP-3-A (Ref 1). 12

3.8.6.2.3 Concrete Temperatures

The limitations listed below are considered applicable only to concrete structural components:

a. The following temperature limitations are for normal operation or any other long-term period. The temperatures are not allowed to exceed 150°F, except for local areas which may be allowed increased temperatures not exceeding 200°F.

b. The following temperature limitations are for accident or any other short-term period. The temperatures are not allowed to exceed 350°F for the interior surface. However, local areas may be allowed to reach 650°F from steam and/or water jets in the event of a pipe failure.

c. Higher temperatures than given in items a. and b. may be allowed in concrete, if test data can be provided to evaluate the reduction in strength. Such a reduction can be applied to the design allowable values. Also, evidence will be provided which verifies that the increased temperatures do not cause deterioration of concrete, either with or without load.

3.8.6.3 Load Combinations and Acceptance Criteria for Seismic Category I Steel Structures

The following presents a set of load combinations and allowable design limits used for Seismic Category I steel structures. To assure that the structural integrity will be maintained, limits on the resulting stresses and the required strength capacities are considered for service loads and for factored loads.

3.8.6.3.1 Load Combinations for Service Load Conditions

Either the working stress design methods of Part 1 of AISC, or the plastic design methods of Part 2 of AISC will be used. 23

a. If the working stress design methods are used, the following load combinations are considered: 23

$S = D + L$ 12

$S = D + L_o + E_o$

$S = D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations are also used:

$$S = D + L + R_o + T_o \quad (3.8\text{-}13)$$

$$S = D + L_o + E_o + R_o + T_o \quad (3.8\text{-}14)$$

$$S = D + L + W + R_o + T_o \quad (3.8\text{-}15)$$

H220.19

No increase in allowable stress is permitted for load combinations (3.8-13), (3.8-14) and (3.8-15), except as indicated below.

If the thermal stresses due to T_o and R_o are secondary and self relieving, the value of S may be increased by 50 percent.

The cases of L having its full value or being completely absent are both checked.

b. If plastic design methods are used, the following load combinations are considered:

$$Y = 1.7D + 1.7L \quad (3.8\text{-}16)$$

$$Y = 1.7D + 1.7L_o + 1.7E_o \quad (3.8\text{-}17)$$

$$Y = 1.7D + 1.7L + 1.7W \quad (3.8\text{-}18)$$

H220.19

The cases of L having its full value or being completely absent are both checked.

If thermal stresses due to T_o and R_o are present, the following combinations are also to be satisfied:

$$Y = 1.3(D + L + T_o + R_o) \quad (3.8\text{-}20)$$

$$Y = 1.3(D + L_o + E_o + T_o + R_o) \quad (3.8\text{-}21)$$

$$Y = 1.3(D + L + W + T_o + R_o) \quad (3.8\text{-}22)$$

H220.19

3.8.6.3.2 Load Combinations for Factored Load Conditions

The following load combinations are considered:

a. If working stress design methods are used, the applicable load combinations are: 23

$$1.6S = D + L_o + T_o + R_o + (E_{ss} \text{ or } W_t \text{ or } V) \quad (3.8\text{-}22)$$

$$1.6S = D + L + T_a + R_a + P_a \quad (3.8\text{-}23)$$

$$1.6S = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + E_o \quad (3.8\text{-}24)$$

$$1.7S = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + (E_{ss} \text{ or } W_t \text{ or } V) \quad (3.8\text{-}25)$$

H220.19

b. If plastic design methods are used, the applicable load combinations are:

$$Y = D + L_o + T_o + R_o + (E_{ss} \text{ or } W_t \text{ or } V) \quad (3.8\text{-}26)$$

$$Y = D + L + T_a + R_a + 1.5\ P_a \quad (3.8\text{-}27)$$

$$Y = D + L_o + T_a + R_a + 1.25\ P_a + (Y_r + Y_j + Y_m) + 1.25\ E_o \quad (3.8\text{-}28)$$

$$Y = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + (E_{ss} \text{ or } W_t \text{ or } V) \quad (3.8\text{-}29)$$

H220.19

In combinations (3.8-22) to (3.8-29), thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material being designed for is ductile.

In combinations (3.8-27), to account for the effect of SRV loads on containment internals, the load fctor for L shall be increased to 1.25. H220.15

In combinations (3.8-23) through (3.8-25) and (3.8-27) through (3.8-29), the maximum effects of P_a, T_a, R_a, Y_j, Y_r, and Y_m are used unless a time-history analysis is performed to justify otherwise. H220.14

For combinations (3.8-22) through (3.8-29) strains due to T_a and the dynamic effects of W_t (tornado missile impact), P_a, Y_r, Y_j, and Y_m may exceed the allowables provided there will be no loss of function of any safety-related system. H220.19

Whenever strains are permitted to exceed yield due to a certain type of load, the structure is checked to satisfy that its ability to carry other loads is not jeopardized.

When computing the required section strength, S, for combinations (3.8-22) through (3.8-25), the plastic section modulus may be used if it meets the AISC criteria for compact sections. H220.19

The effects of tornado-generated differential pressures and missiles shall be combined in accordance with BC-TOP-3-A (Ref 1). 12

3.8.6.3.3 Steel Temperatures

For structural steel elements, the maximum temperatures are limited to 700°F and the allowable values are reduced by 5 percent for each 100°F increase in temperature using 100°F as the base for the allowables.

3.8.6.4 Procedures for Determination of the Effects of Missile Impact on Concrete and Steel Structures

Missile barriers, whether of concrete or steel, are designed with sufficient strength to stop the postulated missiles in accordance with BC-TOP-9-A Section 3.5. To accomplish this objective a prediction of local and overall damage due to the missile impact is necessary. 12

Local damage prediction in the immediate vicinity of the impacted area includes estimation of the depth of penetration and determination of secondary missiles that might be generated by spalling in the case of concrete targets. Overall damage prediction includes estimation of structural response of the target to the missile impact, including structural stability and deformations.

In general, missiles are characterized by impact velocity, missile mass, and impact area. Procedures used in determining these parameters are discussed in Section 3.5, Missile Protection.

3.8.6.4.1 Local Damage Prediction

BC-TOP-9-A is used to estimate missile penetration, perforation, and spalling.

3.8.6.4.2 Overall Damage Prediction

The response of structures to a missile impact depends largely on the location of impact, eg, midspan of a slab or near the support, on the dynamic properties of the target and missile, and on the kinetic energy of the missile. Energy losses due to missile deformation, local penetration, and type of impact are accounted for. The techniques given in BC-TOP-9-A are used to determine an analytical approach, 12 ductility factors, strength increase due to high strain rates, and methods for determining yield displacement. For local effects, yield line theory may be used to determine the capacity of concrete members, and the method of collapse mechanisms may be used to determine the capacity of structural steel members.

3.8.6.5 Procedures for Design of Structural Pipe Restraints

Protection of Seismic Category I structures, systems and 12 components from the dynamic effects of postulated high-energy pipe ruptures is discussed in BC-TOP-9-A PSAR, Section 3.6, and BN-TOP-2.

TABLE 3.8-1

LOAD COMBINATIONS AND LOAD FACTORS

CATEGORY	No.	D	L(1)	H_q	P_t	P_a	T_t	T_o	T_a	E_o	E_{ss}	W	W_t	R_o	R_a	R_r	P_v	P_{mw}	T_{mw}
Service:																			
Test	1.	1.0	1.0	---	1.0	---	1.0	---	---	---	---	---	---	---	---	---	---	---	---
Construction	1.	1.0	1.0	---	---	---	---	1.0	---	---	---	1.0	---	---	---	---	---	---	---
Normal	1.	1.0	1.0	---	---	---	---	1.0	---	---	---	---	---	1.0	---	---	1.0	---	---
Severe Environmental	1.	1.0	1.0	---	---	---	---	1.0	---	1.0	---	---	---	1.0	---	---	1.0	---	---
	2.	1.0	1.0	---	---	---	---	1.0	---	---	---	1.0	---	1.0	---	---	1.0	---	---
Factored:																			
Severe Environmental	1.	1.0(2)	1.3	---	---	---	---	1.0	---	1.5	---	---	---	1.0	---	---	1.0	---	---
	2.	1.0(2)	1.3	---	---	---	---	1.0	---	---	---	1.5	---	1.0	---	---	1.0	---	---
Extreme Environmental	1.	1.0	1.0	---	---	---	---	1.0	---	---	1.0	---	---	1.0	---	---	1.0	---	---
	2.	1.0	1.0	---	---	---	---	1.0	---	---	---	---	1.0	1.0	---	---	1.0	---	---
Abnormal	1.	1.0	1.0***	---	---	1.5	---	---	1.0	---	---	---	---	---	1.0	---	---	---	---
	2.	1.0	1.0	---	---	1.0	---	---	1.0	---	---	---	---	---	1.25	---	---	1.0	1.0
	3.	1.0	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---
Abnormal/ Severe Environmental	1.	1.0	1.0	---	---	1.25	---	---	1.0	1.25	---	---	---	---	1.0	---	---	---	---
	2.	1.0	1.0	---	---	1.25	---	---	1.0	---	---	1.25	---	---	1.0	---	---	---	---
	3.	1.0	1.0	1.0	---	---	---	1.0	---	1.0	or	1.0	---	---	---	---	---	---	---
Abnormal/ Extreme Environmental	1.	1.0	1.0	---	---	1.0	---	---	1.0	---	1.0	---	---	---	1.0	1.0	---	---	---

H220.14

H220.14

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H220.14

(1) Includes all temporary construction loading during and after construction of containment.

(2) Since this load combination is also specified under service loads, it need not be checked under the factored loads.

H220.14

* Concrete tangential shear not to exceed 40 psi for the containment structure described in Subsection 3.8.1

** Concrete tangential shear not to exceed 60 psi for the containment structure described in Subsection 3.8.1

*** For Main Steam Safety Relief Valve Loads, a load factor of 1.25 shall be used. For all other live loads, a load factor of unity shall be used.

TABLE 3.8-2

COMPUTER PROGRAMS FOR USE ON SEISMIC CATEGORY I STRUCTURES OTHER THAN CONTAINMENT

SAR IDENT NO.	CODE NO.	NAME	DOCUMENTATION TRACEABILITY	REMARKS
1	None	Classical Methods	a. Roark, Formulas for Stress and Strain, McGraw-Hill b. M. Hentenyi, "Beams on Elastic Foundation, The Univ. of Michigan Press, 1946. c. ACI-Standard 318-71 d. AISC-Steel Construction Manual, 1970.	The classical methods are for use in analyses of beams, plated, frame and shells. They are given in the standard text book and reference handbooks as use universities and engineering practice.
2	CE309	Structural Engineering System Solver	Pacific International Computer Corporation (PICC)	A method formulated for digital computer solution and based on a computer program widely known as SMISS.
3	CE548	Symbolic Matric Interpretive System	PICC	A problem solving method formulated digital computer solution and based on program widely known as SMISS.
4	CE779	Structural Analysis Program	PICC	A method formulated for digital computer solution and based on a program commonly called SAP as developed at the University of California, Berkeley.
5	CE901	Structural Design Language	PICC	A method formulated for digital computer solution and based on a program commonly called ICES-STRUDL
6	ME620	Heat Conduction	PICC	A heat transfer analysis method formulated for digital computer solution using finite elements.
7	None	MRI/STARDYNS	Control Data Corporation	A multipurpose method formulated for digital computer solution.
8	None	Marc CDC	Control Data Corporation	Formulated for digital computer solution.
9	None	Ease	Control Data Corporation	Formulated for digital computer solution.
10	CE668	Plate Beinding 3 Deg.	PICC	A linear elastic analysis of plates on elastic subgrade formulated for digital computer solution using finite elements.

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3.10 SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND CLASS IE ELECTRICAL EQUIPMENT

(See 251 NSSS GESSAR for complete discussion of NSSS equipment seismic design.)

(For electrical motors driving mechanical equipment, see Section 3.9.) 6

Seismic Category I instrumentation and Class IE electrical equipment will be designed to operate during and after an Operating Basis Earthquake (OBE) and a Safe Shutdown Earthquake (SSE). This will be demonstrated by either one or a combination of both of the following two methods:

a. Prediction of the instrument or electrical equipment performance by mathematical analysis

b. Test under simulated seismic conditions.

An exact description of the methodology is provided in IEEE 344-1975. "IEEE Recommended Practices for Seismic Qualification for Class IE Equipment for Nuclear Power Generation Stations." H271.1

Testing will be the principal qualification method. Analysis without testing will be acceptable only in those cases where structural integrity alone can assure the intended function; where electrical equipment must function, testing will be performed. When testing alone is impracticable, a combination of test and analysis will be used. 110.21

3.10.1 SEISMIC DESIGN CRITERIA

3.10.1.1 Seismic Category I Equipment Identification

Refer to Section 3.2.1 for a listing of all Seismic Category I Instrumentation and Class IE Electrical Equipment requiring seismic qualification.

3.10.1.2 General Seismic Design Criteria

All the Plant Seismic Category I Instrumentation and Electrical Equipment will be designed to resist and withstand the effects of the postulated earthquakes. For the

Safe Shutdown Earthquake (SSE) defined in Section 3.7.1, Seismic Category I Instrumentation and Electrical Equipment will be designed to withstand the effects of the earthquake without functional impairment.

From the basic input ground motion data, a series of response curves at various building elevations will be developed after the building layout is completed. This information will be included in the purchase specifications for Seismic Category I equipment and systems. Suppliers of equipment such as batteries and racks, instrument racks, control consoles, etc., will be required to submit test data, operating experience and/or calculations to substantiate that their components, systems, etc., will not suffer loss of function during or after seismic loadings due to the SSE. The magnitude and frequency of the SSE loadings which each component will experience will be determined by its location within the Plant.

Where applicable, i.e. for construction of racks and panels, the structural requirements will be in accordance with AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings", or with AISI "Specifications for Design of Light Gage Cold Formed Structural Members".

3.10.1.2.1 Reactor Protection, Engineered Safety Features, and Standby Power Circuits

See 251 NSSS GESSAR.

3.10.1.2.2 Cable Tray and Bus Duct Supports Criteria

The following criteria will be used in the design of Class IE Trays and Bus Duct Supports:

a. Regardless of cable tray or bus duct function, all supports are designed to meet the requirements of Seismic Category I structures by dynamic analysis using the appropriate seismic response spectra

b. The most probable maximum values are obtained by taking the square root of the sum of the squares of the stresses and reactions of all significant modes

c. Cable tray loading will be in accordance with NEMA Standard VE-1, 1971 i.e. 50 lbs/linear ft, with a

maximum of 200 lbs concentrated load and a Safety Factor of 1.5.

3.10.1.3 Compliance with Seismic Qualification Requirements

See 251 NSSS GESSAR.

3.10.1.3.1 Equipment Supplied other than by GE

Qualification and documentation procedures used for Seismic Category I electrical and instrumentation equipment, supplied by other than GE, will meet the provisions of IEEE Standard 344-1975 including the provisions of NRC Regulatory Guide 1.100. 220.1 H271.1

3.10.2 SEISMIC ANALYSES, TESTING PROCEDURES, AND RESTRAINT MEASURES

The following sections outline the seismic analyses, testing procedures, and restraint measures for the Seismic Category I instrumentation and electrical equipment.

3.10.2.1 Seismic Category I Equipment

The following procedures will be applicable to the analysis of seismic design adequacy of Seismic Category I instrumentation and electrical equipment, including supports such as cable tray supports, battery racks, instrument and control consoles. 110.8

Seismic specification will be provided to the vendor with appropriate response-spectrum curves at the related floor elevations and instructions on their use in qualifying the specified equipment and components.

The general approach employed in the dynamic analysis of Seismic Category I equipment and component design will be based on the response-spectrum technique, where applicable. The time-history analysis of Seismic Category I structures generates in-structure response-spectrum curves and time histories at various support elevations for use in the analysis of systems and equipment.

At each level of the structure where vital items are located, horizontal spectra for each of the two major axes of the structure and a vertical response spectrum will be developed.

Simplified analytical models will be used for analysis of systems and equipment; however, where one or two degrees-of-freedom models do not provide a suitable representation of the systems or equipment under consideration, multi-mass models will be used in accordance with the lumped parameter modeling techniques and normal model theory described in "Nuclear Reactors and Earthquakes", AEC publication, TID-7024.

110.8

The above procedures will be applicable to the analysis of seismic design adequacy of equipment, including supports such as cable tray supports, batteries and racks, instruments, control consoles, and switchgear.

Suppliers of such equipment will be required to submit test data and/or calculations to substantiate that their components and systems will not suffer loss of function before, during, or after seismic loadings due to the SSE.

All safety-related cable tray and instrument tubing supports will be designed by the response-spectrum method. Analysis and seismic restraint measures for tray and tubing supports will be based on combined limiting values for static load, span length, and computed seismic response.

The following bases will be used in the seismic analysis of Class IE cable tray and instrument tubing supports:

a. All safety-related cable tray and instrument tubing supports will be designed to meet the requirement by dynamic analysis (first mode), using the appropriate seismic response spectra

b. Conservative loading will be assumed

c. The support system will be designed to exclude all natural frequencies in a band covering the peak or peaks of the response-spectrum curve

d. Maximum stress will be limited to 90 percent of minimum yield to compensate for effects of higher modes and minor inaccuracies in method of analysis

The design of instrument racks and tubing supports provides that code-allowable stresses will not be exceeded during the SSE.

will be chosen with a steep rising pressure-flow characteristic to maintain a reasonably constant air flow over the full filter train life. Fan and motor materials will be suitable for operation under conditions of maximum radiation and humidity exposures resulting from the design basis accidents in conformance with Regulatory Position C.3.1 of Regulatory Guide 1.52.

h. Ductwork

Ductwork and dampers of the SGTS will be designed to Seismic Category I requirements and in accordance with the recommendation of Section 2.8 of ORNL-NSIC-65, consistent with Regulatory Position C.3.m of Regulatory Guide 1.52.

6.5.2.3 System Operation

The SGTS will start to operate in response to any of the following signals:

a. SGTS automatic start

- Radiation level exceeding the preset value of the monitors in the exhaust duct of the Fuel Building fuel pool area due to a fuel handling accident 23

- High radiation level in the Containment High Purge System exhaust duct 23

- Low reactor water level or excessive drywell pressure (LOCA)

b. SGTS manual start

- Local radiation level exceeding the preset value of the area monitors in the Auxiliary Building and Fuel Building

- Radiation level exceeding the preset value of the monitors in the exhaust duct of the Auxiliary Building

- Manual initiation from the control room.

Following any of the SGTS actuation signals, the fail closed building pressure control valves leading to the SGTS from the area or areas where the signals originated will 23

open and modulate to maintain the building at a design 23
pressure of (-)0.25 in. wg as discussed in Sections 9.4.2,
Auxiliary Building HVAC System and 9.4.6, Fuel Building
HVAC System. The intake and discharge valves of one of the
two SGTS filter trains will both open, and the SGTS filter
train exhaust fan will be automatically placed into operation. The building isolation valves in the Auxiliary
Building or the Fuel Building HVAC outside air intake duct 23
and the building exhaust duct, from the area or areas where
the SGTS draws air flow, will automatically close with the
SGTS actuation signal. Closure of the building isolation
valves will automatically stop their central a/c units and 23
central exhaust fans. The potentially contaminated air
will be filtered in the operating filter train before being
released to outdoors (within the guidelines of 10 CFR 100).

The design negative pressure in the inlet duct header of the SGTS filter train will be controlled by operation of the pressure controller which will modulate the recirculating damper of the operating exhaust fan to maintain the design negative pressure.

The standby filter train will start automatically in response to a low air flow signal from the operating filter train. The filter train automatic intake and discharge isolation valves will operate in conjunction with their fan.

Each filter train will be sized to handle the design airflow at 100 percent relative humidity. The entering air temperature will be 50°F minimum and 148°F maximum. The electric heater will reduce the relative humidity of the entering air to 70 percent, with maximum temperature rise across the heater of 16°F, and 164°F maximum air temperature entering the filter section.

The carbon adsorber beds will be all welded, gasketless design, to assure the integrity of the carbon adsorbed bed frame-to-support joint, thus eliminating the possibility of charcoal bypass flow.

Two temperature sensors will be provided for each SGTS
carbon adsorber bed to sense bed temperature. In the
unlikely event that a sufficiently large quantity of 23
radioactive iodine is trapped in the shutdown filter train
to cause carbon adsorber bed temperature to reach the first
preset value, a high temperature alarm will be actuated in 23
the control room. The decay heat removal air supply and
exhaust valves of the shutdown filter train will be
automtically opened to cool off the carbon adsorber bed to 24
prevent the carbon from reaching the desorption or ignition
temperatures of 250°F and 640°F,

TABLE 6.5-2 Sheet 3 of 4

Component	Malfunction	Comments	
Carbon adsorber	High temperature in carbon adsorber bed	Two temperature switches provided in each carbon bed to actuate valves and alarms on rising charcoal temperature. The first temperature switch will initiate an alarm in the control room when carbon temperature reaches first preset value. The decay heat removal air supply valve and the filter train air exhaust valve will automatically open to cool off carbon adsorber bed. If, despite air flowing through the bed, the temperature continues to rise to the second preset value, the other temperature switch will actuate an alarm in the control room and automatically isolate the filter train. The Plant operator may manually actuate the water deluge fire protection system.	24 23

TABLE 6.5-2 Sheet 4 of 4

Component	Malfunction	Comments
Exhaust fan recirculating damper	1. Fail open	Lack of airflow automatically starts redundant SGTS.
	2. Fail closed	Excessive negative pressure will exist in the SGTS inlet duct. The building pressure control valves connected to the SGTS will automatically throttle towards a closing position to maintain the design building pressure at (-)0.25 in. wg. The high pressure differential switch, sensing the inlet duct pressure, will actuate an alarm in the control room and the operator may switch to redundant filter train, and restore the defective damper. 23
Differential pressure sensor/ transmitter/ switch	Failure of instruments	Redundant controls are provided.
Filter train isolation valves	Fail closed	Failure to obtain required air flow will automatically start redundant SGTS.

	Standby Service Water	Standby PWR	FPC & C	Suppression Makeup	Standby Gas Treat	Control Room HVAC	Combustible Gas Control (a)	Aux. Bldg. HVAC	Diesel Gen. Bldg HV	Containment Ventil. Isolation	Enclosure (d) Bldg Exhaust
IEEF 279-1971	X	X		X	X	X	X	X	X	X	–
IEEE 308-1971	X	X		X	X	X	X	X	X	X	–
IEEE 323-1974 (c)	X	X		X	X	X	X	X	X	X	–
IEEE 334-1974	X	X		X	X	X	X	X	X		–
IEEE 336-1971	X	X		X	X	X	X	X	X	X	–
IEEE 338-1971	X	X		X	X	X	X	X	X	X	–
IEEE 344-1975	X	X		X	X	X	X	X	X	X	–
IEEE 379-1972	X	X		X	X	X	X	X	X	X	–
IEEE 384-1974	X	X		X	X	X	X	X	X	X	–
IEEE 387-1972		X									–
RG 1.3					X	X			X		–
RG 1.6		X									–
RG 1.7							X				–
RG 1.9 (c)		X									–
RG 11				X							–
RG 1.22	X	X		X	X	X	X	X	X	X	–
RG 1.25					X						–
RG 1.29	X	X		X	X	X	X	X	X	X	–
RG 1.30				X			X				–
RG 1.32	X	X		X	X	X	X	X	X	X	–
RG 1.47	X	X		X	X	X	X		X		–
RG 1.52					X	X		X			–
RG 1.53	X	X		X	X	X	X	X	X	X	–
RG 1.62	X				X	X	X		X	X	–
RG 1.68	X			X			X				–
RG 1.73				X			X			X	–
RG 1.75 (c)	X	X		X	X	X	X	X	X	X	–
RG 1.78						X					–
RG 1.80								X		X	–
RG 1.81	X										–
											–
											–
GDC 13	X	X		X	X	X	X	X			–
GDC 17		X									–
GDC 18		X									–
GDC 19		X				X					–
GDC 20	X	X		X	X	X			X	X	–
GDC 21	X	X		X	X	X	X		X	X	–
GDC 22	X			X	X	X	X		X	X	–
GDC 23	X			X	X	X	X			X	–
GDC 24	X			X	X	X	X		X	X	–
GDC 29	X			X	X	X	X		X	X	–
GDC 61			X								–
IEEE 382-1972	X			X	X	X	X	X	X	X	–
RG-1.40							X				–
RG-1.41		X									–

H271.1

223.14

(a)
Combustible Gas Control includes both the Mixing System and the Flammability Control System (in lieu of the optional system shown in Figure 7.1-2 of the NSSS 251 GESSAR).

(c)
Partial Compliance see section 8.3.1.3.1 for exceptions. 223.14

(b)
For systems supplied by the NSSS supplier this information is given in Figure 7.1-2 of the NSSS 251 GESSAR.

(d)
Normal use: see Aux. Bldg. HVAC. POST-LOCA Operation: See Standby Gas Treat. Also see Section 6.5.2.3 223.14

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CODES AND STANDARDS APPLICABILITY MATRIX

FIGURE 7.1-2

7.3.2.3.4 Qualifying Class I Electric Equipment

IEEE-308 (1971), "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations." Refer to Subsection 8.3.1.2.1.

IEEE-323 (1971), "IEEE Standard for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations." Refer to Section 8.3.1.3. 6

Compliance with IEEE-344 (1975), "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations" is described in Section 3.10. H271.1

7.3.2.3.5 Conformance to 10 CFR 50 Appendix A

a. Criterion 13 - Cooling water to essential components during reactor normal shutdown isolation modes and following a LOCA is assured by monitoring appropriate signals which start the SSWS when required.

b. Criterion 20 - The SSWS Control System will automatically initiate appropriate action with no operation action required.

c. Criterion 21 - The high functional reliability, redundancy, and inservice testability of the two separate trip systems assures that the SSWS Control System will function as required.

d. Criterion 22 - The two redundant separate trip systems are physically and electrically separated so that no single failure can prevent an initiation of the SSWS.

e. Criterion 23 - The system logic and actuator signals are designed to ensure that the SSWS Control System will fail to a safe state. Motor operated valves will fail as-is on loss of power.

f. Criterion 24 - The process control system and SSWS Control System will be physically and electrically separated so that failure in the process system will not cause failure in the SSWS Control System.

g. Criterion 29 - Separation, redundancy, functional reliability, and physical and electrical independency ensure that no anticipated operational occurrence will prevent initiation of the SSWS.

7.3.2.3.6 Conformance to 10 CFR 50 Appendix B

High reliability components are employed throughout the system. Testing of components and panels will be performed prior to shipment to the reactor site. The assembled systems are tested prior to operation. Provisions for testing assure that quality will be maintained throughout plant life. Further discussion is provided in Chapter 17.0.

7.3.2.3.7 Safety Guides Conformance

Safety Guide No. 22 - Periodic Testing of Protection System Actuation Functions. All components may be tested from sensor to actuator during plant operation. (See Paragraph 7.3.1.4.3, "Testability").

7.3.2.3.8 Conformance to Information Guide No. 2

1. a. Refer to Subsection 7.1.2.1.

   b. Suppliers other than General Electric have not been identified at this date. If other suppliers are selected to design and/or build safety-related equipment, they will be identified at the time of commitment.

   c. Refer to Subsection 7.3.2.3.

2. a. Refer to Subsection 7.1.1.2.1.

   b. Refer to Subsection 7.1.1.2.1.

3. Refer to Section 3.10.

4. Refer to Chapter 17.0.

5. Refer to Section 3.12 and Subsection 8.3.1.2.5.

6. Refer to Section 3.11.

7. Refer to Section 3.11.

8. Refer to Subsection 7.1.1.3.4.

9. Refer to Section 3.12 and Subsection 8.3.1.2.5.

10. The method of periodic testing is described under Subsection

<u>System Repair (IEEE-279 Par. 4.21)</u>. The Combustible Gas Control System is designed to permit repair or replacement of components.

All devices are designed for a 40-year lifetime, subject to replacement of limited identified parts, under the imposed duty cycles. Since this duty cycle is composed mainly of periodic testing rather than operation, lifetime is more a matter of shelf life than active life. However, all components are selected for continuous duty plus thousands of cycles of operations, far beyond that anticipated in actual service.

Recognition and location of a failed component will be accomplished during periodic testing. The simplicity of the logic will make the detection and location relatively easy, and components are mounted in such a way that they can be conveniently replaced in a short time.

<u>Identification (IEEE-279 Par. 4.22)</u>. A nameplate identified each control panel and instrument panel that is part of the Combustible Gas Control System. The nameplate shows the division to which each panel is assigned, and also identifies the function in the system of each item on the control panel. The system to which each relay belongs is identified on the relay panels.

<u>Conformance to IEEE-338</u>. The system will be testable during reactor operation. The test will completely test each logic through to the final actuators and demonstrate independence of channels and bare any credible failures while not neglecting its safety function.

<u>Qualifying Class IE Electrical Equipment</u>. IEEE-308 (1971), "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations". Refer to Subsection 8.3.1.2.1(5).

IEEE-323 (1974), IEEE Standard for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations. See Section 8.3.1.3. 6

Compliance with IEEE-344 (1975), "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations" is described in Section 3.10. H271.1

<u>Conformance to 10 CFR 50 Appendix A</u>

a. <u>Criterion 13</u>. Any concentration of hydrogen within the drywell and containment following a loss of coolant accident will be monitored and can

be recirculated by initiation of the Hydrogen Mixing Control System.

b. Criterion 21. The high functional reliability, redundancy, and in-service testability of the two separate control systems assures that the Combustible Gas Control System will function as required.

c. Criterion 22. The two redundant separate control systems are physically and electrically separated so that no single failure can prevent the operation of the Combustible Gas Control System.

d. Criterion 23. Separate and redundant subsystems ensure that the Combustible Gas Control System can sustain a channel failure without preventing minimum function. Motor operated valves will fail close on loss of power.

e. Criterion 24. The process control system and the Combustible Gas Control System will be physically and electrically separated so that failure in the process system will not cause failure in the Hydrogen Mixing Control System.

f. Criterion 29. Separation, redundancy, functional reliability, and physical and electrical independency ensure that no anticipated operational occurrence will prevent operation of the Combustible Gas Control System.

Conformance to 10 CFR 50 Appendix B. High reliability components are employed throughout the system. Testing of components and panel will be performed prior to shipment to the Plant Site. The assembled systems are tested prior to operation. Provisions for testing assure that quality will be maintained throughout Plant life.

Safety Guides Conformance. Safety Guide No. 22 - Periodic Testing of Protection System Actuation Functions. All valves may be tested from sensor signal to actuator during Plant cperation.

7.3.2.6 Standby Power System

The analysis of this system is presented in Chapter 8.3.

7.3.2.7 Auxiliary Building, Heating, Ventilating and Air Conditioning (HVAC) System

The safety analysis of this system and the associated instrumentation and control are discussed in Section 9.4.2.

7.3.2.8 Enclosure Building Exhaust System

control room. For instance, wires can be disconnected in an energize-to-operate component without giving indication. Nor is the so called "fail-safe" system immune from the equally disabling action of jumpering of normally closed contacts so their action will not be seen by the system. Instrument valve shutoff is another disabling mechanism which is not directly indicated in the control room, but such action is under the operator's procedural control and cannot be taken without defeating seals or locks.

Access to Means for Bypassing (IEEE-279 Par. 4.14). The instrument valves cannot be operated without removing seals or locks that are procedurally controlled by operating personnel.

The switches used for placing an SGTS train in standby are located on the panel in the control room associated with the instrumentation and control for the particular SGTS unit.

Multipoint Set Point (IEEE-279 Par. 4.15). Not applicable to this system because all set points are fixed.

Completion of Protection Action Once Initiated (IEEE-279 Par. 4.16). All initiation decisions are sealed in downstream of the decision-making logic, so dampers go to the proper position and the SGTS starts and remains in operation.

Manual Actuation (IEEE-279 Par. 4.17). Each system is capable of being initiated manually at the system level from the control room.

Access to Set Point Adjustments (IEEE-279 Par. 4.18). Set point adjustments for the level switches, pressure switches and flow switches are integral with the sensors on the local instrument racks and cannot be changed without the use of tools to remove covers over these adjustments.

Access to sensor adjustments is under administrative control of Plant personnel.

Identification of Protective Actions (IEEE-279 Par. 4.19). Initiation of the SGTS is directly indicated and identified by annunciator operation and sensor relay indicator lights, convenient, visible verification of the relay position.

Information Readout (IEEE-279 Par. 4.20). Sufficient information is provided on a continuous basis so that the operator can have a high degree of confidence that the SGTS Control System function is available and/or operating properly.

System Repair (IEEE-279 Par. 4.21). The SGTS Control System is designed to permit repair or replacement of components.

All devices in the system are designed for a long lifetime under the imposed duty cycles. Since this duty cycle is composed mainly of periodic testing rather than operation, lifetime is more a matter of shelf life than active life. However, all components are selected for continuous duty plus thousands of cycles of operation far beyond that anticipated in actual service.

Recognition and location of a failed component will be accomplished during periodic testing. The simplicity of the logic will make the detection and location relatively easy, and components are mounted in such a way that they can be conveniently replaced in a short time.

The design of the SGTS Isolation Control System facilitates rapid diagnosis and repair. Provisions have been made to facilitate repair of the radiation monitors and replacement of the gamma detectors during reactor operation.

Identification (IEEE-279 Par. 4.22). A nameplate identifies each control and instrument panel that is part of the SGTS. The nameplate shows the division to which each panel is assigned, and also identifies the function in the system of each item on the panel. The system to which each relay belongs is identified on the relay panels.

Conformance to IEEE-338. The system will be testable during reactor operation. The test will completely check each sensor through to the final actuators and demonstrate independence of channels and bare any credible failures while not neglecting its safety function.

Qualifying Class I Electric Equipment. IEEE-308 (1971), "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations". Refer to Section 8.3.

IEEE-323 (April 1971), "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generation Stations". Refer to NEDO-10698.

Compliance with IEEE-344 (1975), "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generation Stations" is described in Section 3.10. H271.1

Conformance to 10 CFR 50 Appendix A

a. Criterion 13. The release of radioactive materials to the environment will be prevented by the monitoring of appropriate plant variables and, upon detection of abnormal conditions, close the appropriate dampers and activate the SGTS.

TABLE 7.3-2

BALANCE OF PLANT ESF INSTRUMENTATION PANELS 23

Panel Number	Panel Name	BOP-ESF System
H13-P849	BOP ESF Systems Div. 1	Standby Gas Treatment Essential Chilled Water Control Room HVAC Combustible Gas Control
H13-P850	BOP ESF Systems Div. 2	Standby Gas Treatment Essential Chilled Water Control Room HVAC Combustible Gas Control
H13-P863	HVAC Control Panel	Control Room HVAC Auxiliary Building HVAC Standby Gas Treatment
H13-P870	BOP Longterm Response Board	ADS & SRV Air Supply Containment Isolation
H13-P877	Diesel Generators No. 1 & No. 2 Control Board	Standby Power Supply
H13-P879	ESF System Auxil. CH. A Relay Panel	Standby Service Water
H13-P880	ESF System Auxil. Relay Panel CH. B	Standby Service Water

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Notes: 1. See Figure 7.3-2 for Panel Locations

2. NSS-ESF Panel H13-P601, Reactor Core Cooling Benchboard, also includes the following BOP-ESF System:

   o Standby Service Water

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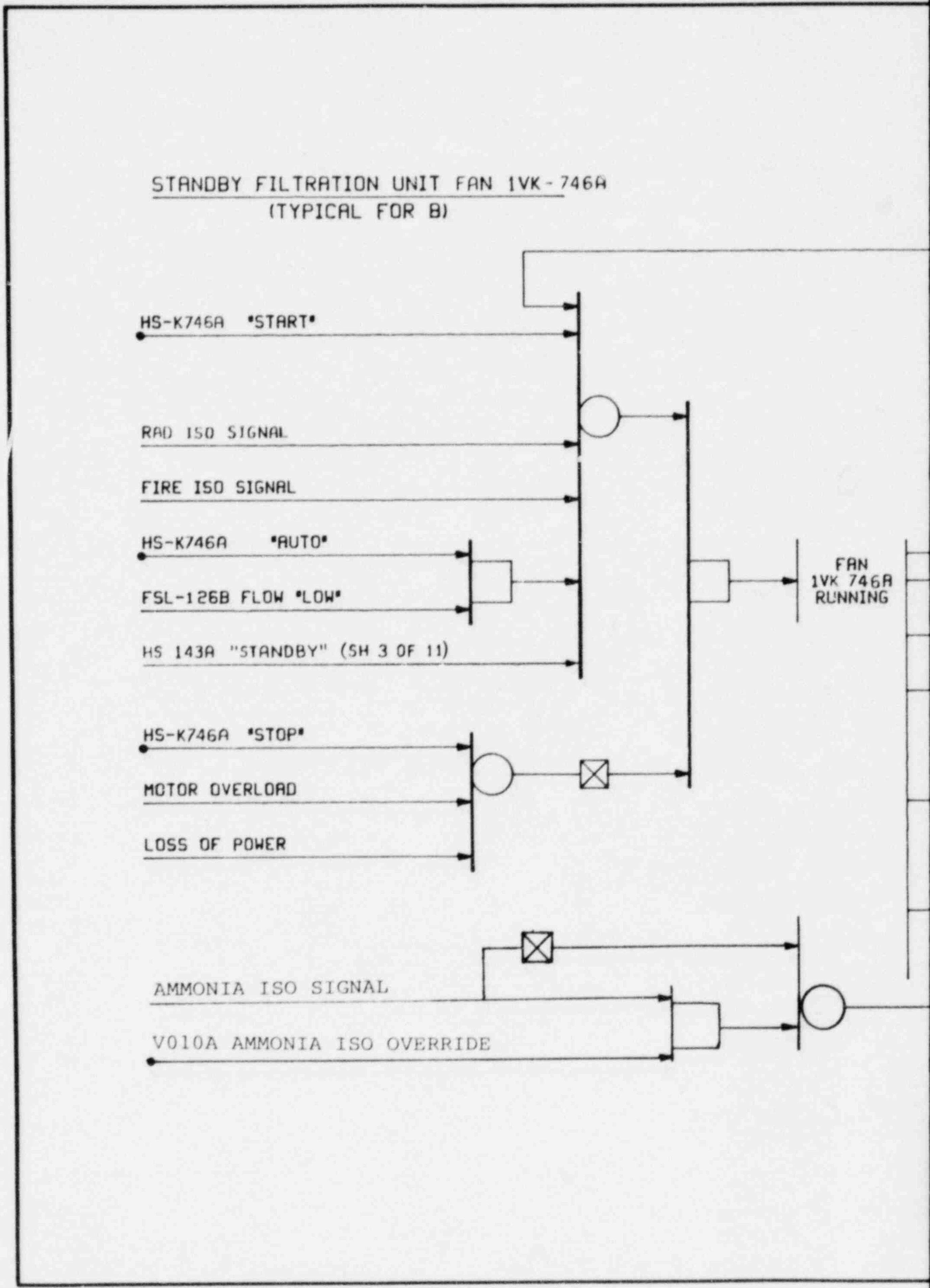
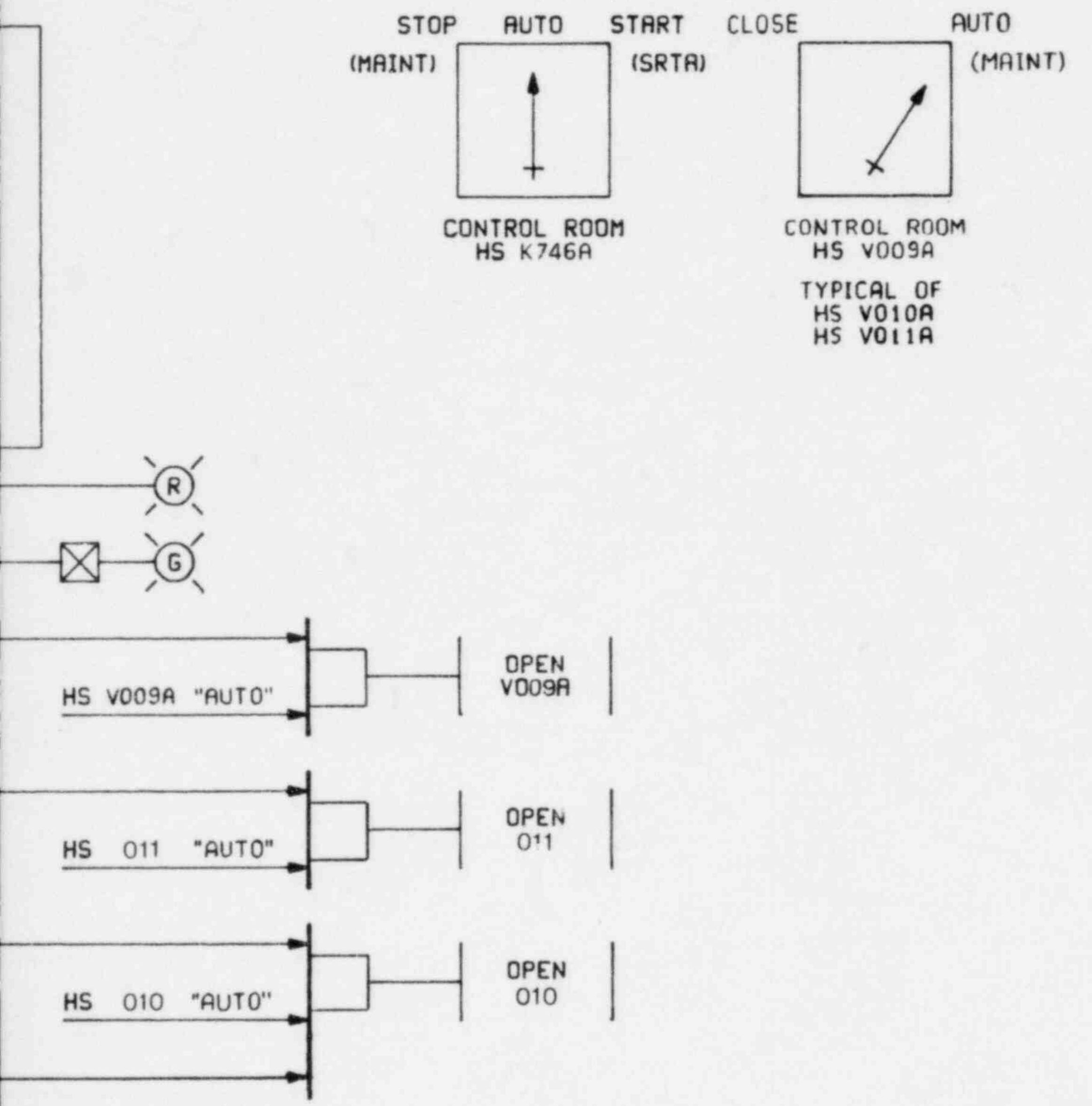
TABLE 7.3-3

BALANCE OF PLANT ESF SYSTEM SUMMARY

	System	Logic Figure No.	P&ID Figure No.
a.	Standby Service Water	7.3-3	9.2-17
b.	Control Room HVAC	7.3-4	9.4-1
c.	Essential Chilled Water	7.3-5	9.2-20
d.	Combustible Gas Control	7.3-6	6.2-29
e.	Standby Power Supply	7.3-7	8.3-2 (Single line)
f.	Auxiliary Building HVAC	7.3-8	9.4-3
g.	Standby Gas Treatment	7.3-9	6.5-1
h.	Fuel Oil Storage & Transfer	7.3-10	9.5-2
i.	Fuel Building HVAC	7.3-11	9.4-8
j.	Diesel Generator Building HVAC	7.3-12	9.4-9


STANDBY FILTRATION UNIT FAN 1VK-746A
(TYPICAL FOR B)
HS-K746A "START"
RAD ISO SIGNAL
FIRE ISO SIGNAL
HS-K746A "AUTO"
FSL-126B FLOW "LOW"
HS 143A "STANDBY" (SH 3 OF 11)
HS-K746A "STOP"
MOTOR OVERLOAD
LOSS OF POWER
FAN 1VK 746A RUNNING
AMMONIA ISO SIGNAL
V010A AMMONIA ISO OVERRIDE



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CONTROL ROOM HVAC LOGIC DIAGRAM

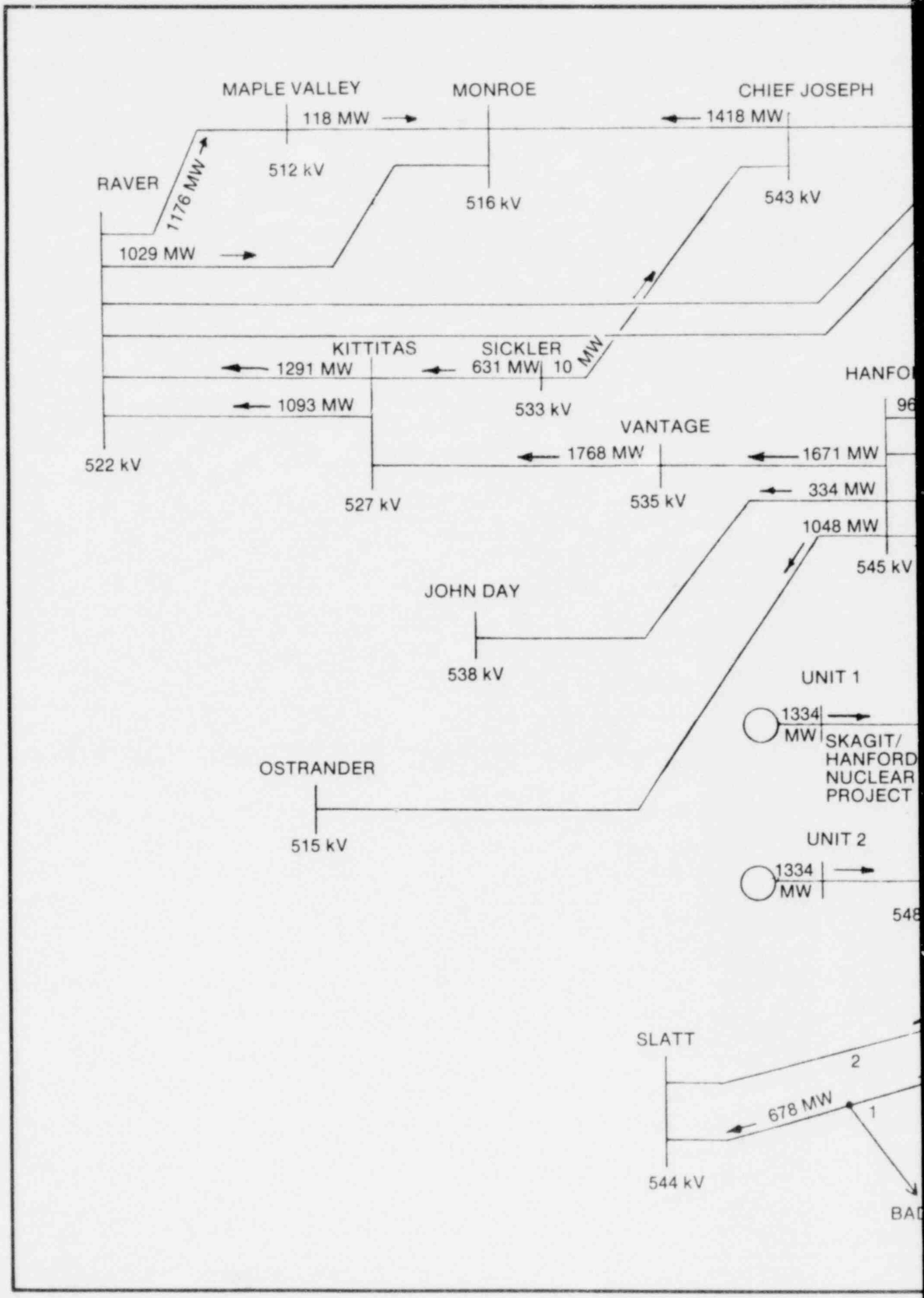
FIGURE 7.3-4 (9 OF 11)

CHAPTER 8.0

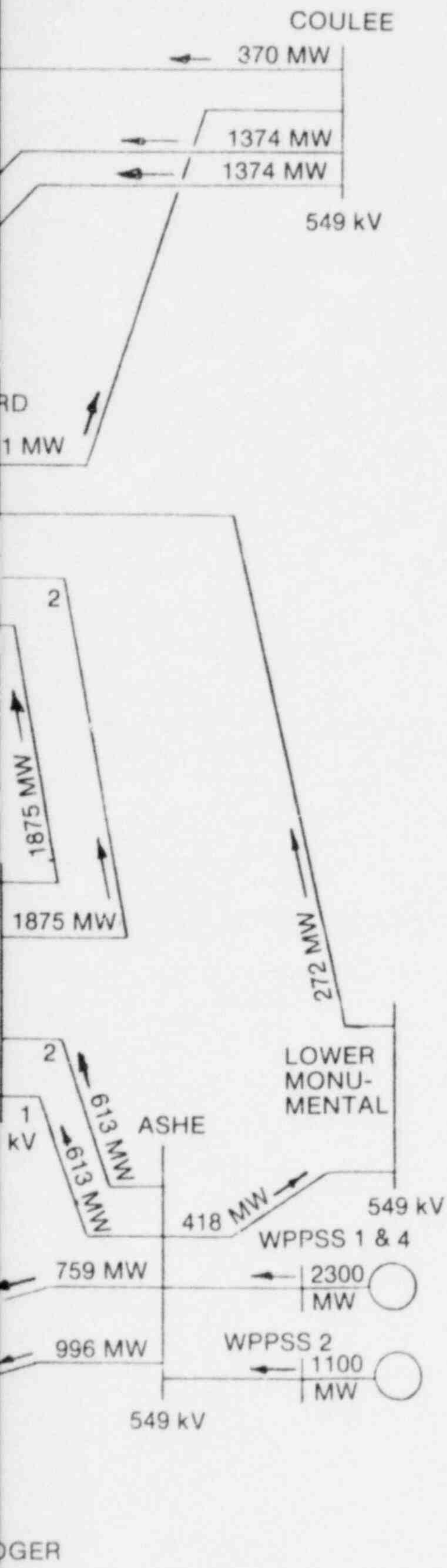
ELECTRIC POWER

FIGURES

24



MAPLE VALLEY
MONROE
CHIEF JOSEPH
118 MW
1418 MW
RAVER
1176 MW
512 kV
516 kV
543 kV
1029 MW
KITTITAS
SICKLER
1291 MW
631 MW
10 MW
HANFO
1093 MW
533 kV
96
VANTAGE
522 kV
1768 MW
1671 MW
527 kV
535 kV
334 MW
1048 MW
545 kV
JOHN DAY
538 kV
UNIT 1
1334 MW
SKAGIT/
HANFORD
NUCLEAR
PROJECT
OSTRANDER
515 kV
UNIT 2
1334 MW
548
SLATT
2
678 MW
1
544 kV
BA



DYNAMIC STABILITY SWING CURVE CASE NUMBER	DESCRIPTION OF DYNAMIC STABILITY TEST	RESULTS
8.2-8, Sheet 3	3-Phase fault on the Project's 500 kV bus, clear Project-to-Hanford #1 500 kV line in 4 cycles, continue run for 10 seconds.	Stable
8.2-8, Sheet 4	3-Phase fault on the Project's 500 kV bus, clear Project-to-Ashe #1 500 kV line in 4 cycles, continue run for 10 seconds.	Stable
8.2-8, Sheet 9	3-Phase fault on the Hanford 500 kV bus, clear Hanford-to-John Day 500 kV line in 4 cycles, continue run for 10 seconds.	Stable
8.2-8, Sheet 10	3-Phase fault on the Ashe 500 kV bus, clear Ashe-to-Slatt #1 500 kV line in 4 cycles, continue run for 10 seconds.	Stable

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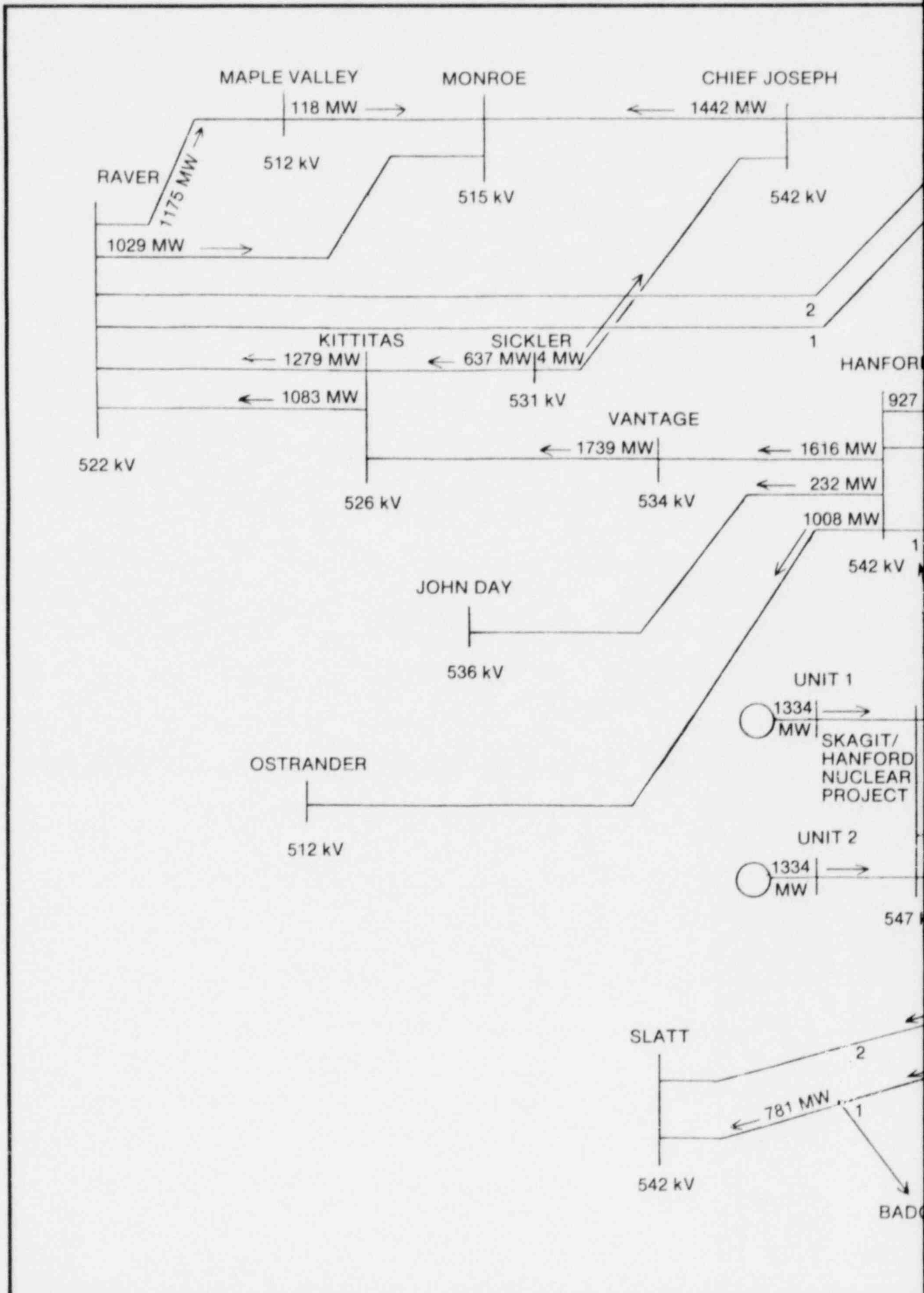
PRELIMINARY SAFETY ANALYSIS REPORT

POWER FLOW DIAGRAM AND DYNAMIC STABILITY TEST

FIGURE 8.2-7 SHEET (2A OF 8)



MAPLE VALLEY
MONROE
CHIEF JOSEPH
118 MW
1442 MW
512 kV
RAVER
1175 MW
515 kV
542 kV
1029 MW
2
1
KITTITAS
SICKLER
1279 MW
637 MW
4 MW
HANFOR
531 kV
1083 MW
927
VANTAGE
522 kV
1739 MW
1616 MW
526 kV
534 kV
232 MW
1008 MW
542 kV
JOHN DAY
536 kV
UNIT 1
1334 MW
SKAGIT/
HANFORD
NUCLEAR
PROJECT
OSTRANDER
UNIT 2
512 kV
1334 MW
547
SLATT
2
781 MW
1
542 kV
BAD

DYNAMIC STABILITY SWING CURVE CASE NUMBER	DESCRIPTION OF DYNAMIC STABILITY TEST	RESULTS
8.2-8, Sheet 5	Project-to-Hanford #2 500 kV line out of service, 3-phase fault on the Project's 500 kV bus, clear Project-to-Hanford #1 line in 4 cycles, continue run for 10 seconds.	Stable

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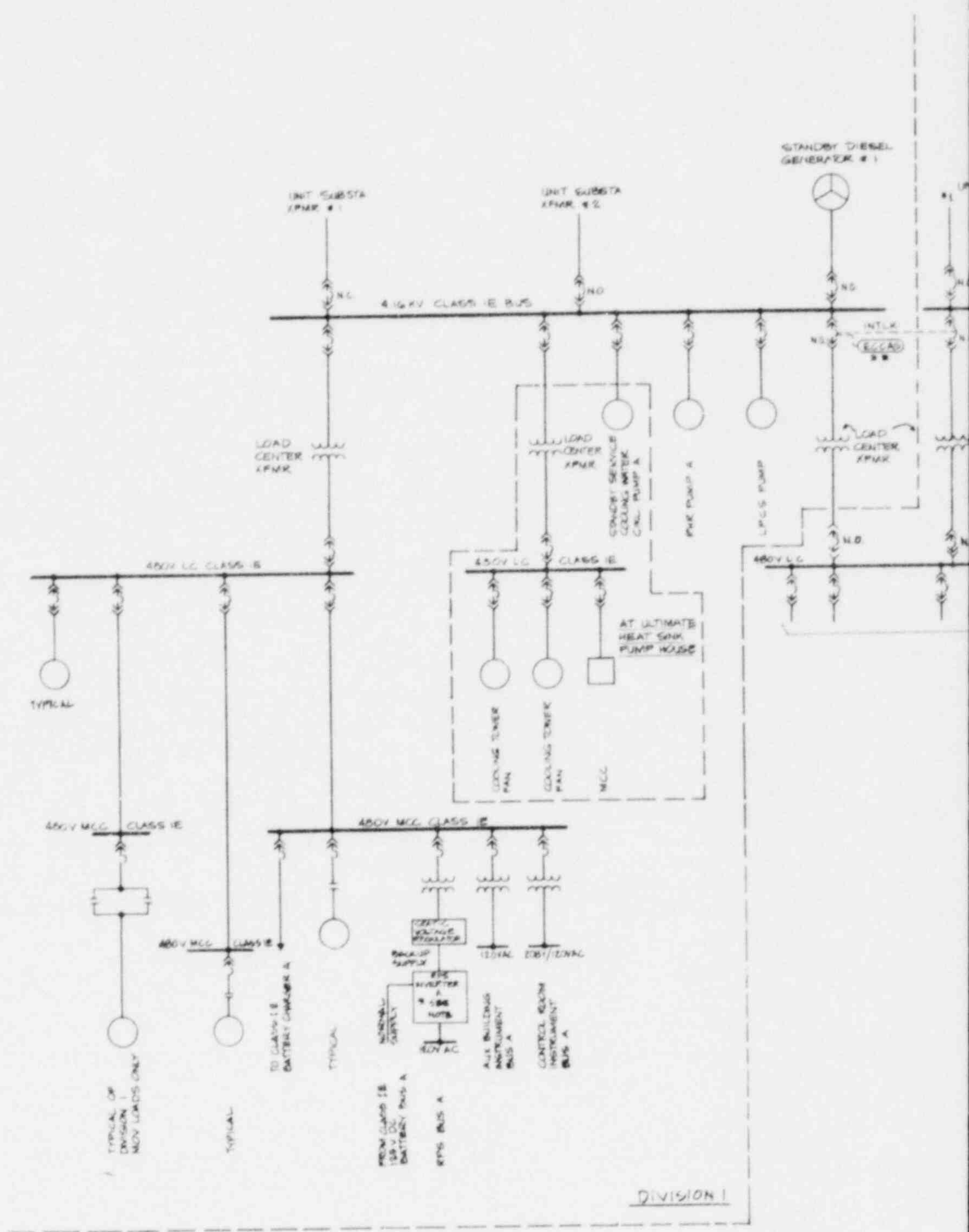
POWER FLOW DIAGRAM AND DYNAMIC STABILITY TEST

FIGURE 8.2-7 SHEET (3 OF 8)



NOTES:

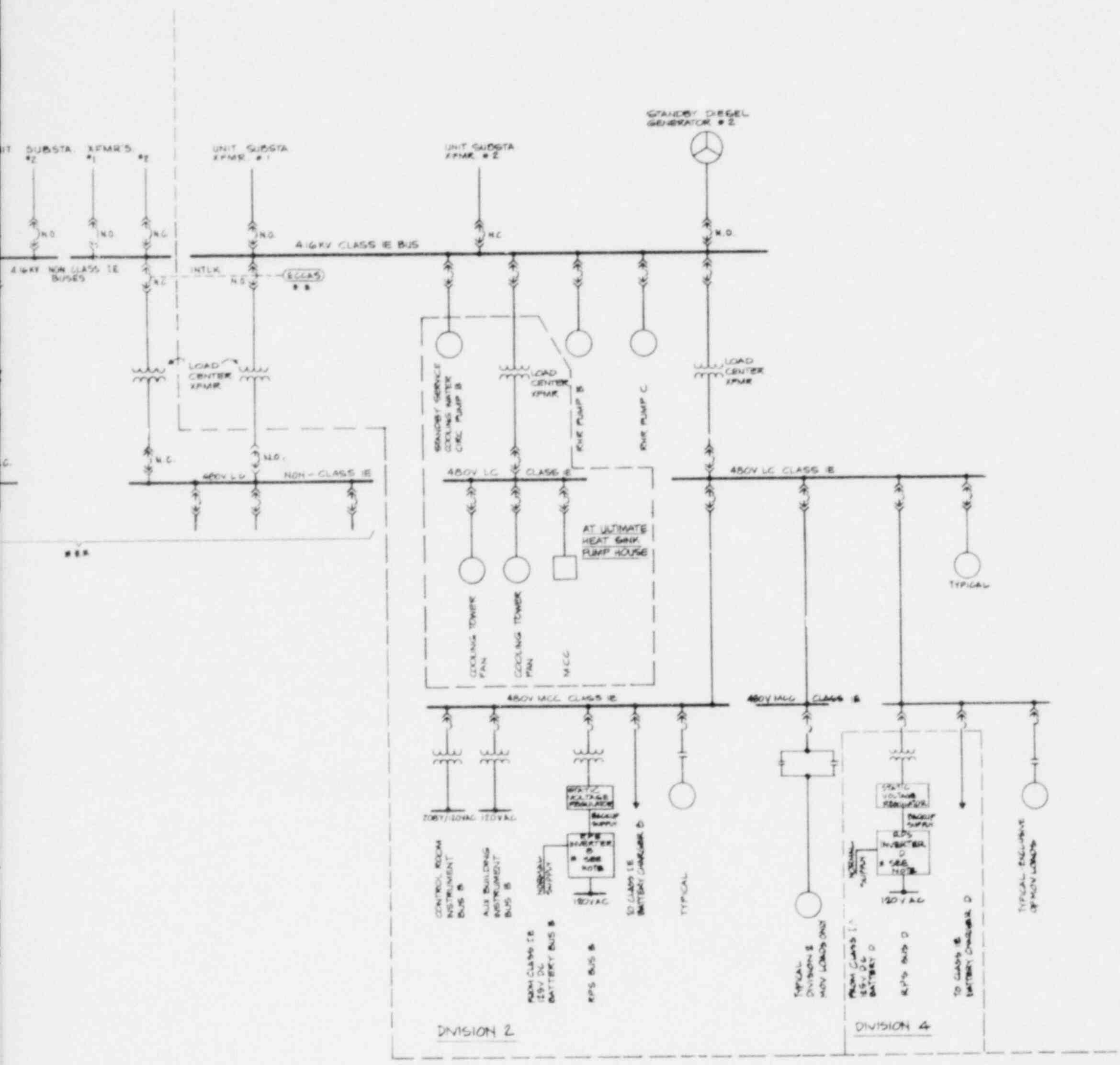
* AND ASSOCIATED STATIC AND MANUAL TRANSFER SWITCHES.

** BREAKER (ISOLATION DEVICE) TRIPPED UNDER DESIGN BASIS ACCIDENT CONDITIONS BY EMERGENCY CORE COOLING ACTUATION SYSTEM (ECCAS) SIGNAL.

*** SIMPLIFIED REPRESENTATION. EACH OF THE TWO NON-CLASS 1E LOAD CENTERS FED THROUGH AN ISOLATION SYSTEM WILL SUPPLY SEVERAL NON-CLASS 1E MOTOR CONTROL CENTERS SERVING LOADS WHICH ARE IMPORTANT TO POWER GENERATION EQUIPMENT INTEGRITY OR TO ITS ORDERLY SHUTDOWN. EXAMPLES ARE:

- NON-CLASS 1E BATTERY CHARGERS
- PLANT COMPUTER
- TURBINE/GENERATOR ESSENTIAL AC AUXILIARIES
- ESSENTIAL LIGHTING
- NON-CLASS 1E INSTRUMENTATION
- PLANT P.A. SYSTEM

THESE LOADS WILL BE ENERGIZED FROM THE NON-CLASS 1E BUS IF OFFSITE PREFERRED POWER IS AVAILABLE.



PUGET SOUND POWER & LIGHT COMPANY
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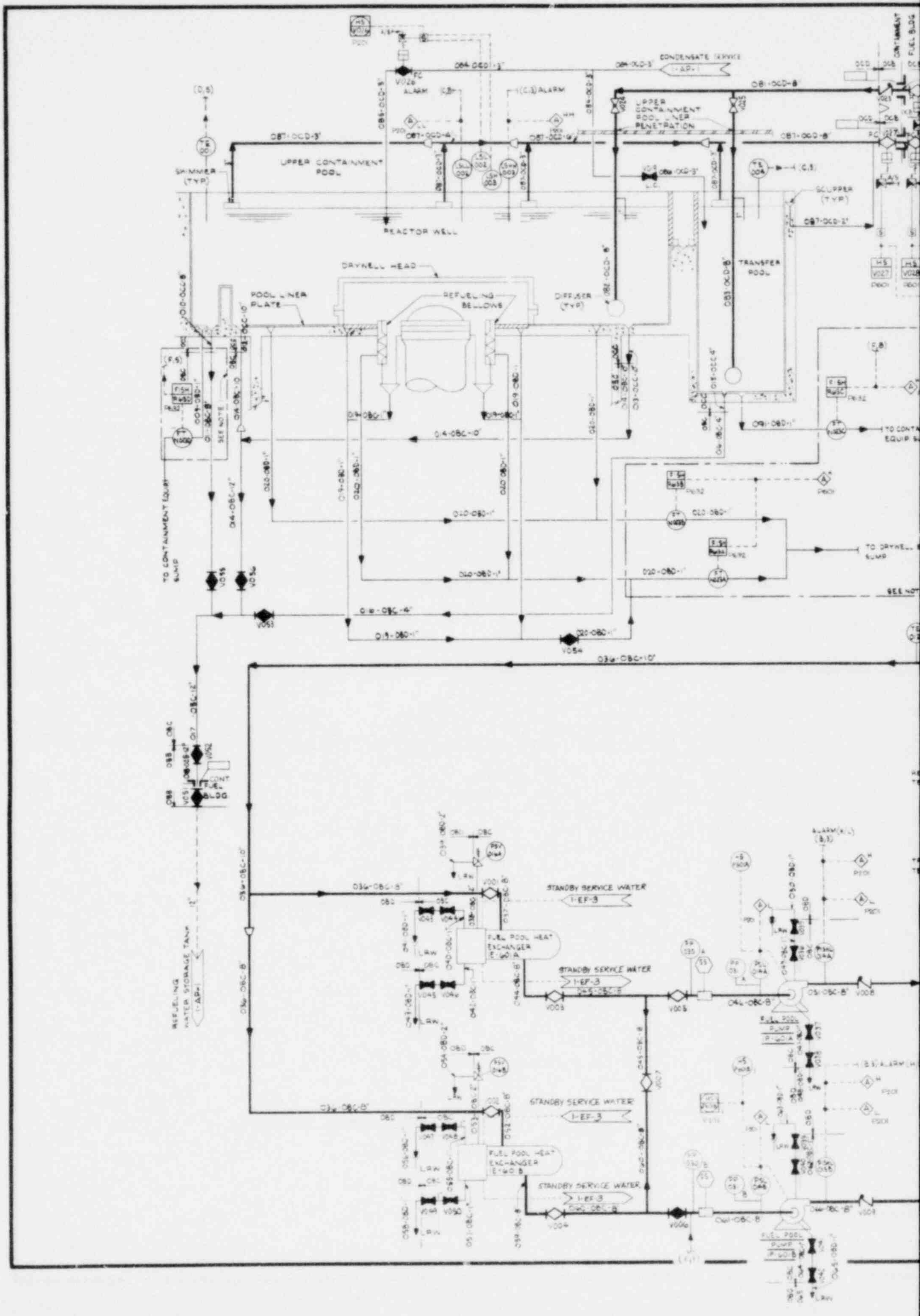
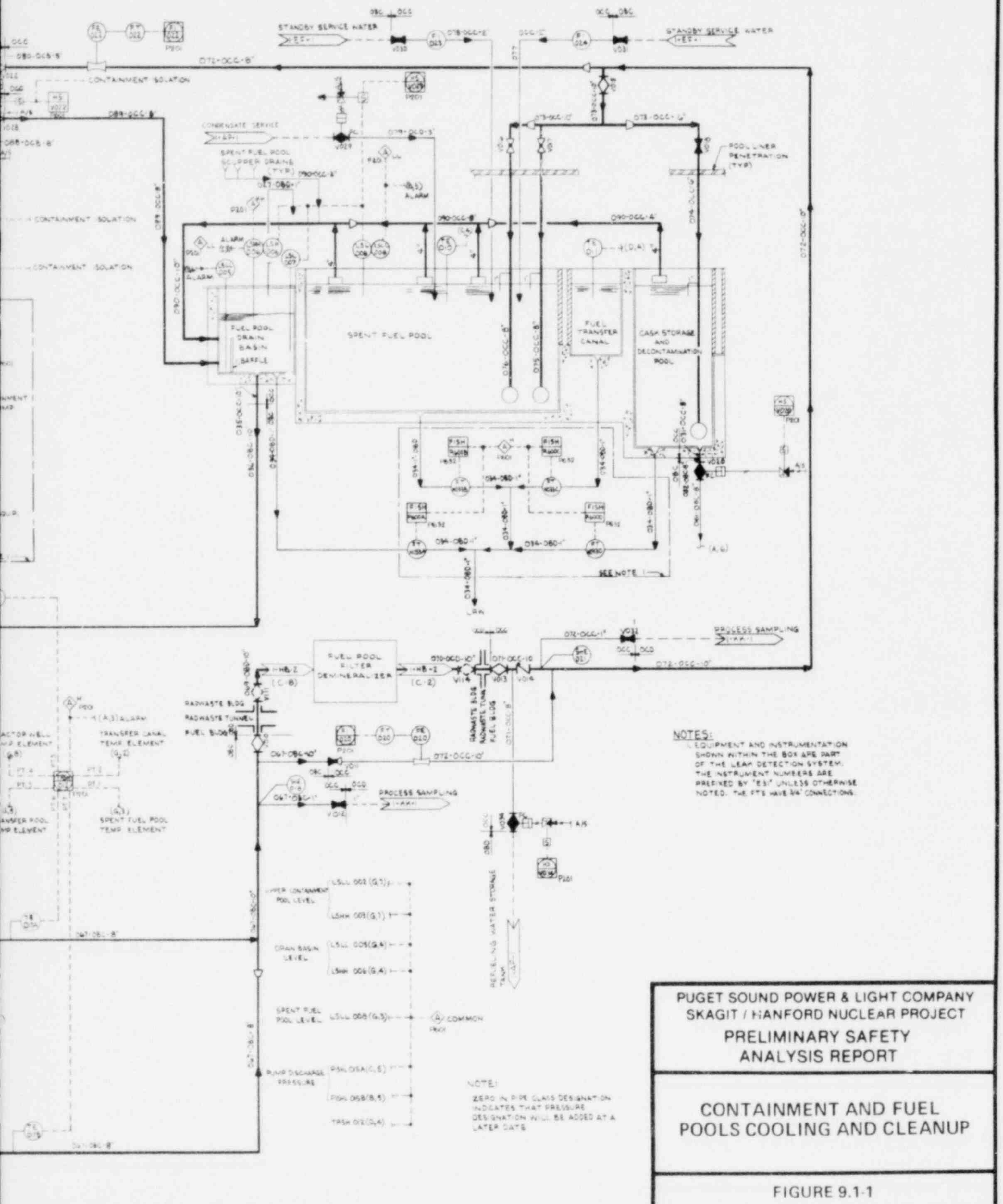
PRELIMINARY SAFETY ANALYSIS REPORT

SIMPLIFIED SINGLE LINE DIAGRAM
STANDBY AC POWER SYSTEM
CLASS IE, DIVISIONS 1, 2 & 4

FIGURE 8.3-2


CONDENSATE SERVICE
1-AP-1
UPPER CONTAINMENT POOL LINER PENETRATION
UPPER CONTAINMENT POOL
SKIMMER (TYP)
REACTOR WELL
SCUPPER (TYP)
DRYWELL HEAD
POOL LINER PLATE
REFUELING BELLOWS
DIFFUSER (TYP)
TRANSFER POOL
TO CONTAINMENT EQUIP SUMP
TO DRYWELL SUMP
SEE NOTE
REFUELING WATER STORAGE TANK
CONT
FUEL BLDG
STANDBY SERVICE WATER
1-EF-3
FUEL POOL HEAT EXCHANGER E-601A
FUEL POOL HEAT EXCHANGER E-601B
FUEL POOL PUMP P-601A
FUEL POOL PUMP P-601B
LRW
CONTAINMENT
FUEL BLDG



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CONTAINMENT AND FUEL POOLS COOLING AND CLEANUP

FIGURE 9.1-1

The system closed loop provides cooling water to non-essential equipment which has the potential to carry radioactive fluids, and is designed to take into account and minimize long-term corrosion which may degrade system performance.

The service water in the heat exchanger tube side is maintained by pump operation at higher pressure than that of the contaminated closed loop on the heat exchanger shell side. In the event of tube failure, the service water will leak into the closed system precluding the possibility of radioactive releases to the environment.

9.2.2.2 System Description

The RCCW System consists of two 100 percent capacity circulating pumps, two 100 percent capacity heat exchangers, a chemical addition tank, a head tank and associated valves, piping and controls, as shown in Figure 9.2-2. System containment penetrations and isolation valves are designed to Seismic Category I and ASME Code Section III, Class 2 requirements, a pressure of 150 psig, and a temperature of 500°F. The system is designed to ANSI B31.1 requirements.

The RCCW System is designed to supply water to the following components:

a. Drywell Coolers

b. RWCU Pump "A" Seals Coolers

c. RWCU Pump "B" Seals Coolers

d. Fuel Pool Sample Cooler

e. CRD Pump "A" Seals Cooler

f. CRD Pump "B" Seals Cooler

g. RWCU Non-Regenerative Heat Exchangers

h. RWCU Sample Coolers

i. Containment Equipment Drain Sump Heat Exchangers

j. Drywell Equipment Drain Sump Heat Exchangers

k. Recirculation Pump "A" Winding Cooler

l. Recirculation Pump "A" Lower Bearing Cooler

m. Recirculation Pump "A" Upper Bearing Cooler

n. Recirculation Pump "A" Seals Cooler

o. Recirculation Pump "B" Winding Cooler

p. Recirculation Pump "B" Lower Bearing Cooler

q. Recirculation Pump "B" Upper Bearing Cooler

r. Recirculation Pump "B" Seals Cooler

During normal operation one RCCW pump and one heat exchanger are in service. The other pump and heat exchanger are on standby. The tube side flow to the heat exchanger is supplied by the Circulating Water Booster System during normal operation and by the Standby Service Water System during loss of off-Site power. H410.2

The RCCW System is a balanced closed loop system. The water is circulated throughout the closed loop by the pump and the capacity required by each individual component is set by a manual control valve located on the discharge side of each component.

The water is discharged from the RCCW heat exchangers to the low inlet temperature components at a temperature of 77°F and discharged from these components at approximately 85°F. This tempered water is then circulated through other components having an acceptable inlet temperature in the range of 95°F to 105°F. The water discharged from these components, at a temperature of approximately 117°F, is then returned to the RCCW heat exchanger. In the heat exchangers the heat is transferred to the service water flow on the tube side.

The RCCW System pumps, heat exchangers, and chemical addition tank will be located in the Fuel Building. The head tank will be the highest point in the loop.

9.2.2.3 Safety Evaluation

The RCCW System has no safety-ralated function. Failure of the system will not compromise any safety-related system or component or prevent a safe reactor shutdown. Piping for the RCCW System is routed so that a pipe break will not flood or damage any safety-related equipment. The System will remain operable during all modes of normal operation including startup and shutdown of the Plant.

Chilled water flow into and out of the Containment will be controlled by isolation valves which will close automatically on isolation signal. For complete discussion on containment isolation, refer to Section 6.2.4, Containment Isolation.

Temperature and/or pressure indicators and relief valves will be provided throughout the system for monitoring and protection.

9.2.9 ESSENTIAL CHILLED WATER SYSTEM

The Essential Chilled Water (ECW) System will be designed to provide a heat sink for the safety-related air conditioning systems requiring chilled water.

9.2.9.1 Design Bases

The ECW System is safety-related and will be designed to meet Seismic Category I (SC-I) requirements.

The ECW System will be designed with two independent 100 percent capacity loops to ensure that the failure of an active or passive component, or of one source of electric power or cooling water, shall not impair the system's ability to perform its safety function.

The ECW System will be designed to supply 45°F chilled water to the cooling coils of the following heating, ventilating, and air-conditioning (HVAC) equipment dur.ng all modes of Plant operation:

a. Control room air-conditioning (A/C) units (SC-1)

b. Control Building A/C units (SC-1)

c. Switchgear rooms air recirculating units (SC-1).

9.2.9.2 System Description

Major components of the ECW System will be located in the Control Building (SC-I). The process and instrumentation diagram for the ECW System is shown on Figure 9.2-20 and will consist of the following major components: 020.10

a. Two, 100 percent capacity water chillers

b. Two, 100 percent capacity chilled water pumps

c. Two, open type expansion tanks.

During reactor shutdown, cooling water to the water chiller condensers will be provided with standby service water (SC-I), discussed in Section 9.2.11, Standby Service Water System. During normal operation the water chiller condensers are cooled by the Circulating Water Booster System. H410.2 H410.2

Normal makeup water to each of the two chilled water loops will be supplied with (SC-II) demineralized makeup water, discussed in Section 9.2.3, Demineralized Water Makeup, Storage and Transfer System. Standby service water will be provided as backup makeup water, when demineralized water is not available.

Chemical treatment will be provided in the ECW System to prevent pipe corrosion and scale buildup. The type of chemicals and the methods used to determine the frequency of chemical treatment will be provided in the FSAR.

9.2.9.2.1 Component Description

Design parameters for the major components of the ECW System are presented in Table 9.2-7. Major components are described as follows:

a. Water Chiller Package - Each water chiller will be the factory assembled and tested centrifugal hermetic type, complete with evaporator, water cooled shell and tube condenser, and complete with safety and refrigeration controls, standard accessories such as high-low pressure cutouts, low water temperature switch, and piping connections.

   Each water chiller will meet the Air Conditioning and Refrigeration Institute (ARI) Standard 550-72, "Standards for Application and Ratings of Centrifugal Water Chilling Packages" and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Class 3, "Nuclear Power Plant Components."

b. Chilled Water Pumps - Each chilled water pump will be the centrifugal type with direct-coupled motor and base.

h. Fuel Pool Heat Exchanger B

i. Essential Chilled Water Condenser

j. Reactor Component Cooling Water Heat Exchanger B.

DIVISION III, THE HPCS SERVICE WATER SYSTEM

a. HPCS Diesel Generator Heat Exchangers:

(1) Turbocharger Aftercooler
(2) Lube Oil Heat Exchanger
(3) Jacket Water Heat Exchanger

b. HPCS Pump Room Cooler

c. HPCS Pump Bearing Cooler.

The SSW System operates only during reactor shutdown. During normal operation the following SSW System components are cooled by the Circulating Water (CW) Booster System via an intertie to the SSW System headers: H410.2

a. Reactor Component Cooling Water System

b. Essential Chilled Water Condenser

c. Fuel Pool Heat Exchangers.

The interties between the SSW and CW Booster Systems supply and discharge headers, shown in Figure 9.2-17, are provided with motor operated isolation valves. The valves required to be open during normal Plant operation automatically close during SSW System operation initiated by a LOCA and/or loss of off-Site power. H410.2

The SW pumps draw water from the Ultimate Heat Sink basin during normal Plant operation. Drawdown of the water level in the basin is precluded by operator action in response to safety-related level indicators and in response to alarms actuated by safety-related level switches.

A conservative time in excess of 30 minutes has been allowed between reaching the alarm for basin low water level and the 30 day water level assuming failure of RWS makeup water and continuous loss of basin water at a rate equal to the flow of both SW pumps. This time will be sufficient for the operator to take action to preclude depletion of the level below that required for 30 days of shutdown heat removal. 020.54

The interties between the SSW System and the following are provided with double (redundant) isolation valves:

a. To Residual Heat Removal (RHR) System, (for post-accident containment flooding)

b. To Containment and Fuel Pool Cooling and Cleanup (CFPCC) System, (spent fuel pool makeup).

The interface of the SSW System and RWS System shown in Figures 9.2-14 and 9.2-17 is designed to insure adequate basin water level. The RWS System discharge pipe into and the overflow pipe from the SSW System cooling tower basins are located above the water level elevation required for sufficient cooling for at least 30 days.

To minimize the transients during system startup, the system piping not in service during normal Plant operation is pressurized.

To prevent the accumulation of algae and other fouling agents, sodium hypochlorite additions will be utilized.

9.2.11.3 Safety Evaluation

The SSW System will provide a reliable source of cooling for Plant auxiliaries which are essential to a safe reactor shutdown following a design-basis accident. Either Division I, II or III provides adequate cooling water to meet safe shutdown requirements. The entire system will be adequately protected to withstand adverse environmental occurrences.

The three divisions will each be separated from each other and protected to the extent necessary to assure that, in the event of any one of the following events, sufficient equipment would remain operational to permit safe shutdown of the unit:

a. Earthquake

b. Flooding or steam release from equipment failure such as pipe or tank rupture

c. Pipe whip and jet forces resulting from pipe rupture

d. Missiles which may result from tornado or equipment failure, and

e. Fire.

9.2.12 MAIN COOLING TOWER MAKEUP AND BLOWDOWN SYSTEM (MCTMBS)

9.2.12.1 Design Bases

The Main Cooling Tower makeup will be supplied to replenish water lost in the Cooling Tower due to evaporation, drift, and blowdown.

9.2.12.2 System Description

The sources of water for the Main Cooling Tower makeup are
the Raw Water Supply System and the discharge from the 24
Service Water System. See Figure 9.2-18 for flow diagram
and Figure 1.2-1 for plot plan. The constant SW flow not required for Main Cooling Tower makeup will be recirculated to the SW System via the SSW Cooling Tower basins. The makeup flow is controlled from the level controller in the Cooling Tower basin. The recirculation flow is controlled by a pressure control valve located in the recirculation line to the SSW Cooling Tower basin. The blowdown rate is manually controlled. In the event of loss of makeup water for an extended period of time, the unit will be shut down.

The blowdown from the Circulating Water System will be through a takeoff on the Circulating Water System piping.
The maximum rate of blowdown is approximately 2750 gpm per 24
unit. The daily average maximum and minimum temperatures of
blowdown will be 80.2°F and 57.0°F respectively. 23

The blowdown from the Cooling Towers flows to the gravity
head structure and into the Project discharge pipeline. The 23
Project discharge will be monitored for radioactivity in accordance with Section 11.4.

9.2.12.3 Safety Evaluation

The MCTMBS has no safety-related function. Failure of the system will not compromise any safety-related system or component and prevent safe reactor shutdown.

9.2.12.4 Tests and Inspections

The entire MCTMBS is proven operable by the use of the system components during normal Plant operations. Those portions of the system closed to flow can be tested to ensure their operability and the integrity of the system.

9.2.12.5 Instrumentation Application

The MCTMBS will operate during normal Plant operation. Pressure indicators located at various points throughout the system will inform the operator of pump performance and/or line integrity.

9.2.13 TURBINE BUILDING CLOSED COOLING WATER SYSTEM

9.2.13.1 Design Basis

The Turbine Building Closed Cooling Water (TBCCW) System is a closed loop system designed to continuously supply cooling water to components listed in Subsection 9.2.13.2.

The TBCCW System provides cooling water to the Turbine Building HVAC local air recirculation units and selected components that are required to operate during normal Plant operation. 23

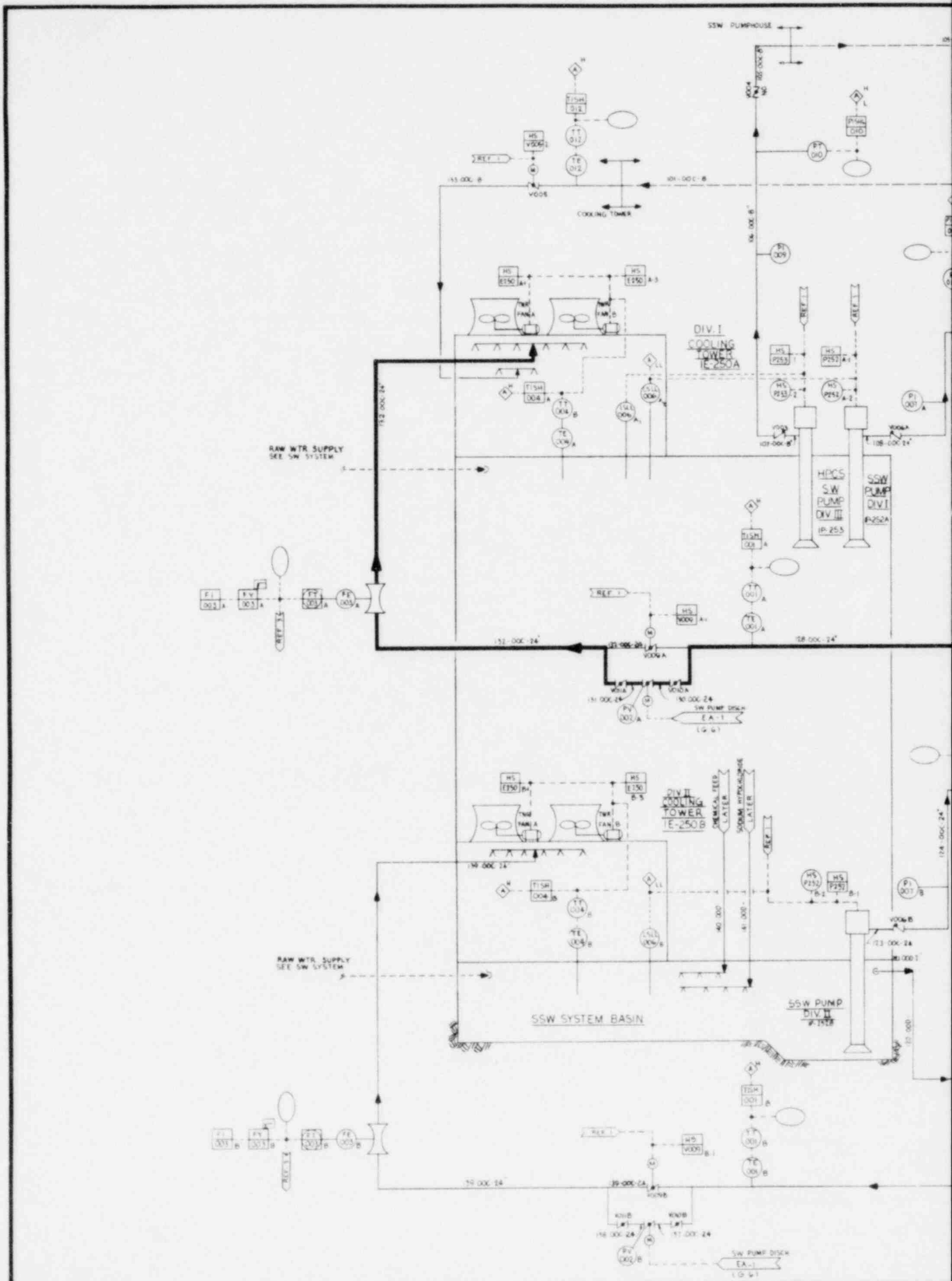
The heat exchangers' shell side is designed for the flow equal to two-pump system operation experienced during pump transfer.

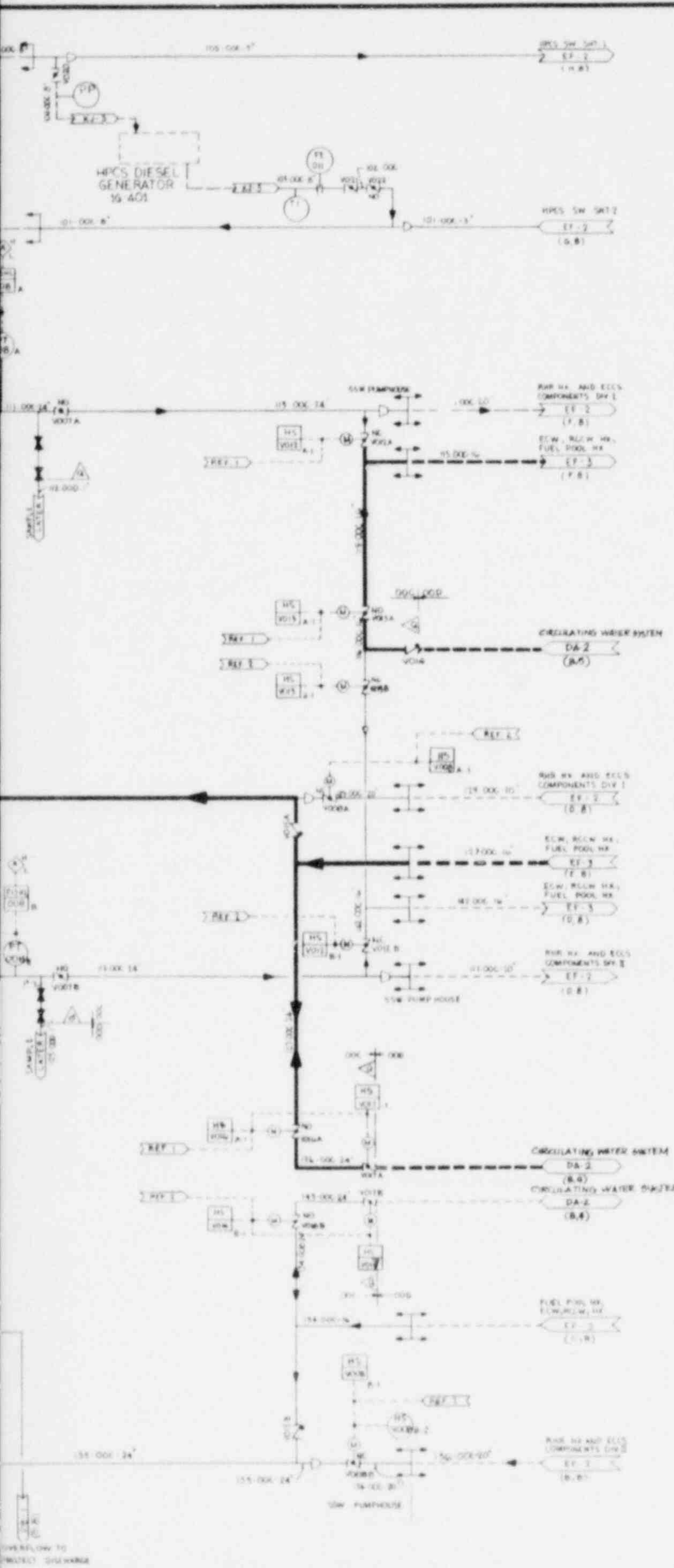
9.2.13.2 System Description

The TBCCW System consists of two 100 percent capacity circulating pumps, two 100 percent heat exchangers, air separator, an expansion tank and associated valves, piping and controls as shown in Figure 9.2-19. 23

The system is designed to ANSI B31.1 requirements.

During normal Plant operation, the TBCCW pump discharges into two headers: 23


SSW PUMPHOUSE
COOLING TOWER
DIV. I
COOLING
TOWER
TE-250A
HPCS
SW
PUMP
DIV III
1P-253
SSW
PUMP
DIV I
1P-252A
RAW WTR SUPPLY
SEE SW SYSTEM
SW PUMP DISCH
EA-1
DIV II
COOLING
TOWER
TE-250B
CHEMICAL FEED
LATER
SODIUM HYPOCHLORIDE
LATER
RAW WTR SUPPLY
SEE SW SYSTEM
SSW SYSTEM BASIN
SSW PUMP
DIV II
SW PUMP DISCH
EA-1



REF. NO.	TITLE	DRAWING NO.
1	SSW SYSTEM CONTROL, DIV I	2-EF1-
2	SSW SYSTEM CONTROL, DIV II	2-EF1-
3	RAW WATER SUPPLY PUMP CONTROL	2-EF1-

NOTE

ALL LINE SIZES ARE PRELIMINARY

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STANDBY SERVICE WATER SYSTEM

FIGURE 9.2-17 (1 OF 3)

TABLE 10.4-1

CONDENSER PERFORMANCE PARAMETERS

Data	Condenser Shell L. P.	I. P.	H. P.
Condenser pressure, inches of hg. absolute	1.51	2.07	2.92
Condenser duty, Btu/hr x 10^6	2956	2918	2980
Condenser surface, sq ft	350,927	395,000	439,073
Condenser circulating water inlet design temperature, ^{o}F	66	79.5	93
normal rise	13.5	13.5	13.5
Flow at design temperature, gpm	438,800	438,800	438,800
Shell Material	C.S.	C.S.	C.S.
Tube Material	304 S.S.	304 S.S.	304 S.S.

TABLE 10.4-2

CIRCULATING WATER SYSTEM COMPONENT DESIGN DATA

Cooling Tower

The round mechanical draft cooling towers are designed based on the following parameters: 24

Number	3 (per unit)
Design wet bulb, °F	80
Relative Humidity, %	50
Design Range, °F	38.5
Design Flow, gpm	156,000 (per tower)
Makeup Flow, gpm maximum	19,450 (per unit)
Blowdown, gpm maximum	2750 (per unit)
Dimensions, ft	
Height	60
Diameter	253

Circulating Water Pumps, Motors and Pump Discharge Values

The circulating water pumps, motors and pump discharge valves is designed based on the following parameters:

1. PUMPS

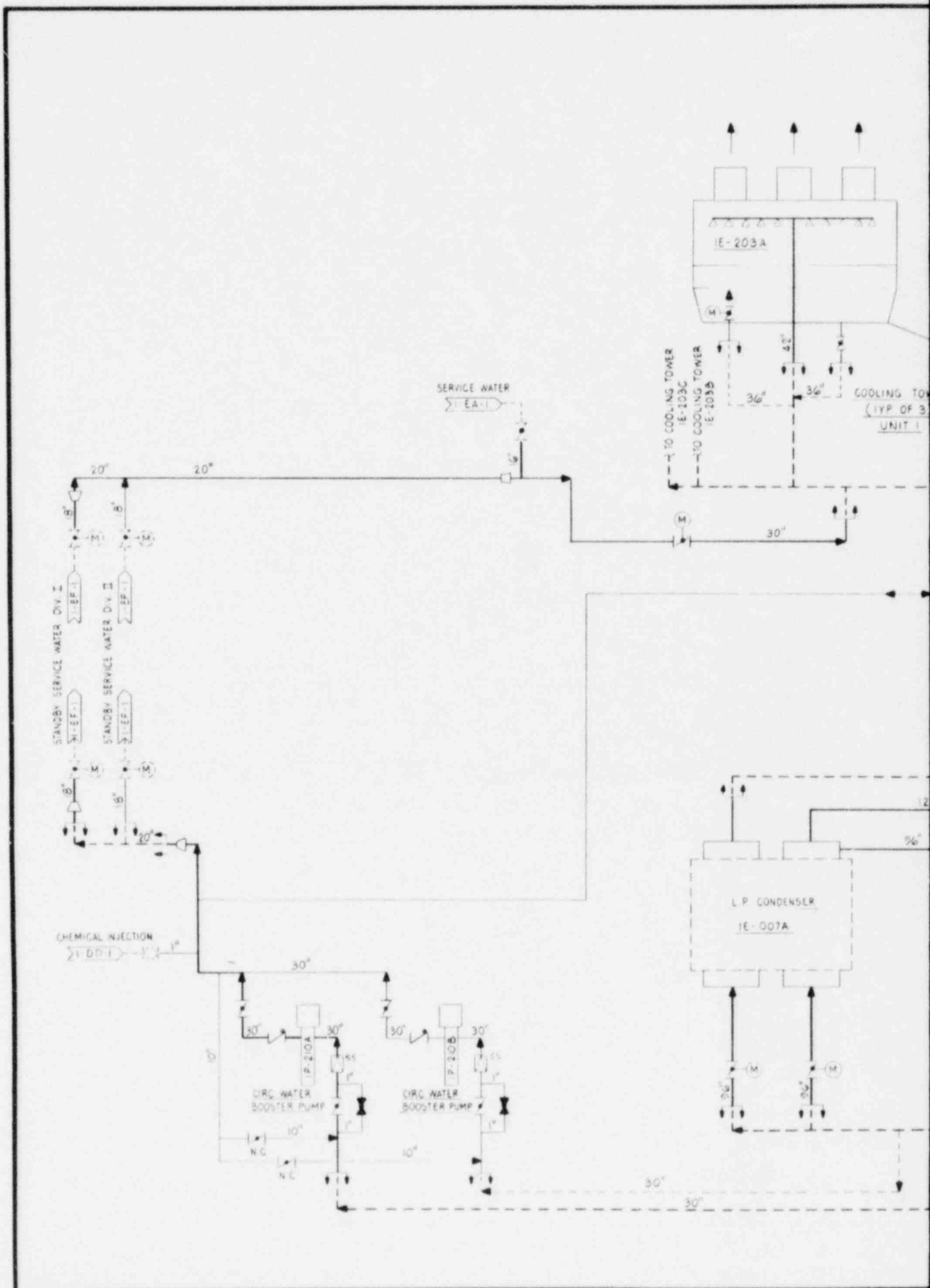
Number,	Two	
Type,	Vertical, wet pit type, closed line shaft water lubricated and below surface discharge	4
Capacity, (50 percent of total flow)	231,000 gpm	4
Discharge Head, TDH	105 Feet	4
Speed,	356 RPM	

2. MOTORS

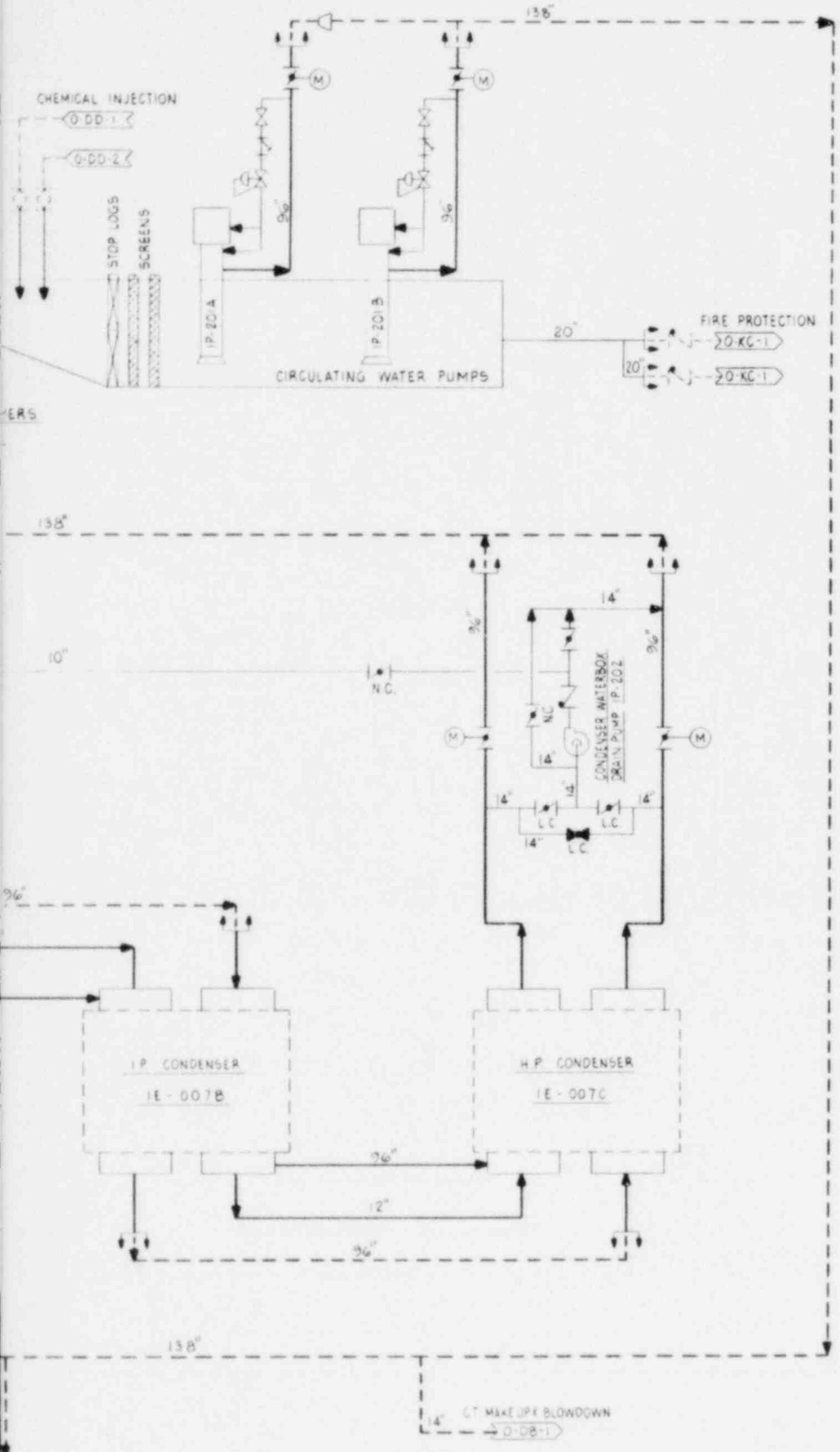
Number,	Two	4
Type,	Induction	

3. PUMP DISCHARGE VALVES

Number,	Two
Type,	Butterfly, rubber seated, motor operated
Valve Body	Fabricated Steel



IE-203A
TO COOLING TOWER IE-203C
TO COOLING TOWER IE-203B
COOLING TOW
(TYP OF 3
UNIT 1
SERVICE WATER
1-EA-1
STANDBY SERVICE WATER DIV. I
STANDBY SERVICE WATER DIV. II
1-EF-1
CHEMICAL INJECTION
1-DD-1
L.P. CONDENSER
IE-007A
P-210A
P-210B
CIRC WATER BOOSTER PUMP
CIRC WATER BOOSTER PUMP
N.C.
N.C.
SS
20"
20"
30"
30"
30"
30"
36"
36"
42"
96"
96"
96"
8"
8"
8"
8"
10"
10"
10"
6"
1"



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CIRCULATING WATER SYSTEM

FIGURE 10.4-1

TABLE 11.3-1

EXPECTED ANNUAL ACTIVITY RELEASED FROM GASEOUS EFFLUENT TREATMENT SYSTEM[1] (CURIES/YEAR/UNIT) USED FOR EVALUATION OF COMPLIANCE WITH APPENDIX I OF 10CFR50

NUCLIDE	COOLANT CONC. (MICROCURIES/G)	CONTAINMENT BLDG.	TURBINE BLDG.	AUXILIARY BLDG.	RADWASTE BLDG.	GLAND SEAL	AIR EJECTOR	MECB VAC PUMP	TOTAL
Ar-41	0.000	3.9E+00	*	1.1E+01	*	*	2.1E+01	*	3.6E+01
Kr-83M	9.1E-03	*	*	*	*	*	*	*	*
Kr-85M	1.6E-03	*	2.5E+01	3.7E+00	*	*	2.0E+00	*	3.1E+01
Kr-85	5.0E-06	*	*	*	*	*	2.6E+02	*	2.6E+02
Kr-87	5.5E-03	*	6.1E+01	2.0E+00	*	*	*	*	6.3E+01
Kr-88	5.5E-03	*	9.1E+01	3.7E+00	*	*	*	*	9.5E+01
Kr-89	3.4E-02	*	5.8E+02	2.0E+00	2.9E+01	*	*	*	6.1E+02
Xe-131M	3.9E-06	*	*	*	*	*	4.8E+00	*	4.8E+00
Xe-133M	7.5E-05	*	*	*	*	*	*	*	*
Xe-133	2.1E-03	7.0E+00	1.5E+02	1.0E+02	2.2E+02	*	2.5E+01	1.3E+03	1.8E+03
Xe-135M	7.0E-03	3.9E+00	4.0E+02	5.6E+01	5.3E+02	*	*	*	9.9E+02
Xe-135	6.0E-03	8.6E+00	3.3E+02	1.2E+02	2.8E+02	*	*	5.0E+02	1.2E+03
Xe-137	3.9E-02	1.2E+01	1.0E+03	1.6E+02	8.3E+01	*	*	*	1.3E+03
Xe-138	2.3E-02	*	1.0E+03	7.5E+00	2.0E+00	*	*	*	1.0E+03
TOTAL NOBLE GASES									7.4E+03
I-131	3.719E-03	5.7E-04	6.3E-03	5.9E-02	2.2E-03	.0	.0	4.6E-03	7.3E-02
I-133	5.085E-02	7.8E-03	8.6E-02	8.1E-01	3.1E-02	.0	.0	5.0E-02	9.8E-01

H-3 released from Turbine Bldg. Ventilation System 5.0E+01
H-3 released from Containment Bldg. Ventilation System 5.0E+01
Total H-3 released via gaseous pathway 1.0E+02
C-14 released via Main Condenser Off-Gas System 9.5 CI/YR

* Less than 1 Ci/yr
(1) Estimated Releases Based on NUREG-0016, Rev. 1, GALE Code Evaluation

23
24
24

TABLE 11.3-2

EXPECTED ANNUAL PARTICULATE ACTIVITY RELEASED FROM GASEOUS EFFLUENT TREATMENT SYSTEM[1] (CURIES PER YEAR/UNIT)
USED FOR EVALUATION OF COMPLIANCE WITH APPENDIX I OF 10CFR 50

Nuclide	Containment Bldg.	Turbine Bldg.	Auxiliary Bldg.	Radwaste Bldg.	Mech Vac. Pump	Total
Cr-51	5.2E-07	9.0E-06	1.0E-03	7.0E-06	1.0E-08	1.0E-03
Mn-54	1.0E-06	6.0E-06	1.3E-03	4.0E-05	.0	1.3E-03
Co-58	2.6E-07	1.0E-05	2.7E-04	2.0E-06	.0	2.8E-04
Fe-59	2.3E-07	1.0E-06	3.7E-04	3.0E-06	.0	3.7E-04
Co-60	2.6E-06	1.0E-05	4.7E-03	7.0E-05	5.6E-09	4.8E-03
Zn-65	2.6E-06	6.0E-05	4.7E-03	3.0E-06	3.4E-09	4.8E-03
Sr-89	7.8E-08	6.0E-05	4.2E-05	.0	.0	1.0E-04
Sr-90	7.8E-09	2.0E-07	9.2E-06	.0	.0	9.4E-06
Nb-95	2.6E-06	6.0E-08	9.7E-03	4.0E-08	.0	9.7E-03
Zr-95	7.8E-07	4.0E-07	9.2E-04	8.0E-06	.0	9.3E-04
Mo-99	1.6E-05	2.0E-05	6.4E-02	3.0E-08	.0	6.4E-02
Ru-103	5.2E-07	5.0E-07	4.1E-03	1.0E-08	.0	4.1E-03

24

Nuclide	Containment Bldg.	Turbine Bldg.	Auxiliary Bldg.	Radwaste Bldg.	Mech Vac. Pump	Total
Ag-110M	1.0E-09	.0	2.3E-06	.0	.0	2.3E-06
Sb-124	5.2E-08	1.0E-06	4.5E-05	7.0E-07	.0	4.7E-05
Cs-134	1.8E-06	2.0E-06	4.5E-03	2.4E-05	3.2E-08	4.5E-03
Cs-136	2.6E-07	1.0E-06	4.7E-04	.0	1.9E-08	4.7E-04
Cs-137	2.6E-06	1.0E-05	5.7E-03	4.0E-05	8.9E-08	5.8E-03
Ba-140	5.2E-06	1.0E-04	2.1E-02	4.0E-08	1.1E-07	2.1E-02
Ce-141	5.2E-07	1.0E-04	8.5E-04	7.0E-08	.0	9.5E-04
Total Particulates						1.2E-01

(1) Estimated releases based on NUREG-0016, Rev. 1, GALE Code Evaluation

23

TABLE 11.3-3 23

MAXIMUM SITE BOUNDARY CONCENTRATIONS COMPARED TO 10 CFR 20 LIMITS

Radionuclide	Maximum Site Boundary Ground Level Air Concentration (pCi/m³)[a]	10 CFR 20 Table II Limits (pCi/m³)	Maximum Site Boundary Ground Surface Concentration (pCi/m²)[b]	
Ar-41	2.99E+01[c]	4.0E+04	(d)	
Kr-85m	2.57E+01	1.0E+05	(d)	
Kr-85	2.16E+02	3.0E+05	(d)	
Kr-87	5.23E+01	2.0E+04	(d)	
Kr-88	7.89E+01	2.0E+04	(d)	
Kr-89	5.07E+02	(e)	(d)	
Xe-131m	3.99	4.0E+05	(d)	
Xe-133	1.49E+03	3.0E+05	(d)	
Xe-135m	8.22E+02	(e)	(d)	
Xe-135	9.97E+02	1.0E+05	(d)	
Xe-137	1.08E+03	(e)	(d)	
Xe-138	8.31E+02	(e)	(d)	
H-3	8.31E+01	2.0E+05	(d)	
C-14	7.89	1.0E+05	(d)	
I-131	6.06E-02	1.0E+02	2.29E+02	
I-133	8.14E-01	4.0E+02	3.32E+02	24
Cr-51	8.31E-04	8.0E+04	1.08E+01	
Mn-54	1.08E-03	1.0E+03	1.58E+02	
Co-58	2.33E-04	2.0E+03	7.76	
Fe-59	3.07E-04	2.0E+03	6.44	
Co-60	3.99E-03	3.0E+02	3.10E+03	
Zn-65	3.99E-03	2.0E+03	4.57E+02	
Sr-89	8.31E-05	3.0E+02	1.97	
Sr-90	7.81E-06	3.0E+01	1.17E+01	
Nb-95	8.06E-03	3.0E+03	1.33E+02	
Zr-95	[illegible].72E-04	1.0E+03	2.33E+01	
Mo-99	5.32E-02	7.0E+03	6.86E+01	
Ru-103	3.41E-03	3.0E+03	6.29E+01	
Ag-110m	1.91E-06	3.0E+02	2.25E-01	
Sb-124	3.90E-05	7.0E+02	1.11	
Cs-134	3.74E-03	4.0E+02	1.31E+03	
Cs-136	3.90E-04	6.0E+03	2.41	
Cs-137	4.82E-03	5.0E+02	7.33E+03	
Ba-140	1.74E-02	1.0E+03	1.05E+02	
Ce-141	7.89E-04	5.0E+03	1.20E+01	

(a) Maximum Site Boundary annual average X/Q = 2.62×10^{-5} sec/m³ (see Table 2.3-3).

(b) Maximum Site Boundary annual average D/Q = 9.88×10^{-8} m⁻² (see Table 2.3-3).

(c) 2.99E+01 = $2.99 \times 10^{+1}$.

(d) Negligible.

(e) Not reported.

Tables 11.3-5 through 11.3-14
not used

TABLE 11.3-15

FILTER TRAINS USED TO CONTROL GASEOUS EFFLUENTS 23

Filter	CFM	Inlet Temp, °F	Filter Components	Iodine Removal Efficiency (Elemental & Organic)
Standby Gas Treatment Filter Train A & B	6,000	50 Min 150 Max	MS; EH; HIGH; HEPA; CHAR; HEPA	99%
Containment Low Purge Filter Train	6,000	50 Min 150 Max	MS; EH; HIGH; HEPA; CHAR; HEPA	90%
Turbine Bldg. Equipment Compartment Exh. Filter Train	12,000	50 Min 120 Max	MS; EH; HIGH; HEPA; CHAR; HEPA	90%
Control Room Standby Filtration Unit A & B	3,000	50 Min 104 Max	HIGH; HEPA; CHAR; HEPA	99%
Radwaste Bldg. Tanks Vent Filter Train	1,000	50 Min 120 Max	MS; EH; HEPA; HIGH; CHAR; HEPA	90%
Radwaste Bldg. Equipment Compartment Exh. Filter Train	30,000	50 Min 120 Max	MOD; HEPA	--
Administration Bldg. Lab Hood Exh. Filter Unit	6,000	50 Min 120 Max	HEPA; CHAR	90%

24

Legend:

MOD - Moderate Efficiency Filter 20% to 60% Efficiency by ASHRAE Stain Test

HIGH - High Efficiency Filter 85% Efficiency by ASHRAE Stain Test to 95% Efficiency

MS - Moisture Separator

HEPA - High Efficiency Particulate Air Filter 99.97% Efficiency for 0.30 Micrometer DOP

CHAR - Activated Charcoal Adsorber Tray or Deep-Bed Type

EH - Electric Heater

The process is completed by the Solid Radwaste Processing
Subsystem, a concrete solidification process of proven
reliability in many earlier radwaste systems. Filter sludge
waste is discharged from the liquid radwaste filter(s), the
fuel pool filter/demineralizer and from the suppression pool 23
filter/demineralizers (these filters will be centrifugal-
disc, dry discharge type) to their respective sludge
batching tanks, where detergent waste, chemical waste,
and/or condensate is added to make the mixture a pumpable
slurry. The slurry and dry cement are fed at controlled
rates to one of the redundant in-line mixers which mixes the
waste and dry cement and pumps the homogeneous mixture
through a vented and sealed fillport, where sodium silicate
is added, and into a shipping container. The solidification
system is provided with an interlock which ensures that an
adequate amount of cement is added to the slurry during
mixing. The absence of free liquids in a container
following solidification will be verified by preoperational 010.6
testing and a process control program. The shipping 010.13
container is then capped, swiped and decontaminated if 23
necessary. The solid radwaste collection and processing
sub-systems are operated from remote control panels. The
processing subsystem is provided with safety features which
ensure adequate flushing to prevent solidification in the
pipe lines and pumps in case of a power failure. Capacities
of major components are presented on the Subsystem P&ID's,
Figures 11.5-1 and 11.5-2.

A compacter is used to process dry waste as described
previously. This compacted waste is packaged in 55-gallon 23
drums for off-Site disposal. Provisions will be made to
collect and filter all airborne radioactivity generated 010.14
during operation of the compacter. 331.41

The Solid Radwaste System is located in the Radwaste
Building. The non-Seismic Category I Solid Radwaste System
will be designed in accordance with Regulatory Guide 1.143,
Rev. 1, "Design Guidance for Radioactive Waste Management
Systems, Structures, and Components Installed in 010.1
Light-Water-Cooled Nuclear Power Plants." A modified 23
seismic analysis in accordance with Regulatory Guide 1.143
will be used for the design of the foundation and walls of
the Radwaste Building, at least up to a height sufficient to
contain the liquid inventory in the building.

11.5.4 EXPECTED VOLUMES

Tables 11.5-1 and 11.5-2 show the expected nuclide
distributions associated with each type of solid waste and
the expected rate of shipment of these wastes from the

Plant Site. The solid waste activities shown on this table were derived from the design basis source rates as described in Section 11.1.

11.5.5 PACKAGING

The Solid Radwaste System is designed to package wastes in 55-gal drums, 50 cu ft shipping containers, and 100-200 cu ft shipping containers. The shipping containers will be of welded construction with standard type 17 fillspouts and lids. The shipping containers will conform to the applicable NRC (10 CFR 71) and DOT (49 CFR 171-178) regulations. 23

11.5.6 STORAGE FACILITIES

The solidified radwaste storage area will be located in the Radwaste Building. The storage area will be 41 ft long and 17.5 ft wide and enclosed by a 19 ft shield wall. Normally, solidified radwaste containers will be stored with concrete shield covers. 23

The amount of waste that can be stored in this area will depend on the container size and the packaging efficiency (i.e., the ratio of volume of waste before and after solidification). Table 11.5-5 shows the volume of solidified waste that will result at various packaging efficiencies and the volume of waste that can be stored utilizing two likely container sizes. This table also shows the storage capacity in terms of months of normal waste generation for combinations of the two variables discussed above. 010.5 010.16

This storage capacity should be sufficient to accommodate the normal waste generation with a reasonable margin. However, in the event of an emergency, such as a prolonged transportation strike or long periods of high waste generation, emergency shielded storage areas can be utilized. In this event containers could be stored on top of the concrete covers inside the shielded storage area, in the monitoring area, or in the solidification area. Table 11.5-5 shows the storage capacity that can be utilized under such conditions.

When the container storage area is full and all system components are filled with wastes, the units must shut down until wastes can be shipped for burial.

CHAPTER 12.0

RADIATION PROTECTION

CONTENTS

12.1.2.4.6 Turbine Building

Fission product gases and activation product gases, principally nitrogen-16, enter the turbine and condenser with the primary steam from the reactor. Approximately 100 percent of the non-condensible activity and 80 percent of the nitrogen are assumed to be discharged via the air ejector to the offgas system while the remaining 20 percent of the nitrogen follows the condensate liquid through the condensate system.

Areas within these shield walls will have high radiation levels and will have limited access.

The dose to the Site boundary as a result of direct and scattered radiation from the turbine and associated equipment will be considered during final Plant design and will be included in the FSAR.

12.1.2.4.7 Control Room Shielding

Figure 12.1-16 presents an isometric drawing of the control room layout showing its relationship to the Containment. | 331.3

The design basis LOCA dictates the shielding requirements for the control room. Shielding is provided to permit access and continuous occupancy of the control room under Loss-of-Coolant Accident (LOCA) conditions with radiation exposures limited to 5 rem whole body dose for the duration of the accident in accordance with the 10 CFR 50, Appendix A, Criterion 19.

The design basis LOCA is described in Section 15.6.5 and is based on Regulatory Guide 1.3. The direct radiation from the airborne fission products inside the Containment would contribute less than .03 rem to personnel inside the control room for the 30-day period following the LOCA. | 24

The parameters used in the demonstration of the control room habitability, in addition to Regulatory Guide 1.3, are listed below. (The ventilation system parameters are listed in Section 12.2.).

a. For all isotopes which escape from the drywell to the Containment, no credit is taken for shielding by the internal structures in the Containment. Shielding credit is taken for the Containment wall.

b. Credit is taken for walls between the Containment and control room

c. Credit is taken for the control room walls

d. Credit is taken for radiological decay and any recirculation filtration

e. Credit is taken for iodine species distribution as listed in Section 15.6.5. | 24

12.1.2.4.8 Diesel Generator Buildings

The diesel generators are located in a separate building. It is expected that radiation sources will not be present at any time, therefore restrictions on access will not be imposed. The dose rate in all areas of the Diesel Generator Building is expected to be less than the Zone I level of 0.5 mrem/hr.

12.1.2.4.9 Miscellaneous Plant and Plant Yard Areas

Sufficient shielding is provided for all Plant buildings containing potential radiation areas to yield radiation levels at the outside surfaces of the buildings below Zone I levels. The Plant yard areas will be fully accessible during normal operation and shutdown.

12.1.2.5 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with Plant radiation zoning and to minimize Plant personnel exposure were based on maximum equipment activities under the Plant operating conditions described in Section 12.1.3 rather than annual average activities. The thickness of each shield wall surrounding radioactive equipment was determined by approximating as closely as possible the actual geometry and physical condition of the source or sources. The isotopic concentrations were converted to energy group sources using data from the Table of Isotopes (Ref 1).

The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, demineralizers, evaporators, and the Containment was a finite cylindrical volume source. For shielding evaluation of piping, the geometric model was an infinite shielded cylinder. In cases where corrosion products are deposited on surfaces such as a pipe, the latter was treated as an annular cylindrical surface source.

for the shielding calculations for this system. The shielding will be based on the reactor steam N-16 activities in Table 11.1.4 (251 NSSS GESSAR). 331.5

12.1.3.8 Fuel Building

12.1.3.8.1 Spent Fuel Transfer and Storage

The primary sources in the Spent Fuel Transfer and Storage areas are the spent fuel elements. The spent fuel element sources are discussed in 251 NSSS GESSAR Section 12.1.3.2.4. 6

The isotopic composition of spent fuel in μCi/watt is given by Table 12.1-20 for 0 decay time. Fuel is transferred after 2 days' decay. The average power per assembly is 4.52 MW_T. Two assemblies may be present in the transfer tube simultaneously. Normally, one-third of the total core of 848 assemblies will be replaced during a refueling operation. The volume of an assembly is 6.8126×10^4cc. 331.17

12.1.3.8.2 Fuel Pool Cooling and Cleanup (FPCC) System

The following equipment will be potential radiation sources due to radioisotopes which leak from the spent fuel and radioisotopes which diffuse from the reactor vessel into the spent fuel pool and are subsequently pumped through the FPCC System:

a FPCC heat exchangers

b. FPCC pumps

c. Associated valves and piping.

The FPCC filter-demineralizers will be located in the Radwaste Building.

The specific activity of the fuel pool water is assumed to be that of seven day old reactor water diluted to a total isotopic concentration of 1.25×10^{-3} μCi/cc. The basis for this assumption is discussed in Section 12.1.2.4.4. The specific emission spectrum for this source is given in Table 12.1-21. The emission spectrum was obtained based on data presented in Ref 2. The volume of water in the fuel pool is estimated at 75,000 ft^3. The isotopic inventory of the fuel pool filter is given in Table 12.1-22. 331.17

12.1.3.9 Turbine Shine Dose

The N-16 present in the reactor steam in the primary steam
lines, turbines, and moisture separators can contribute to
the Exclusion Area Boundary dose as a result of the high
energy gammas which it emits as it decays. 23

Turbine shine doses are calculated using the SKYSHINE
computer program described in Table 12.1-3. Point sources 331.3
are used to represent the components on the turbine deck.
Table 12.1-15 provides the estimated N-16 inventories of
equipment in the Turbine Building. The equipment and piping
located above the main turbine deck were included in the
turbine shine dose calculation. These are:

a. A portion of the main steam piping (40 ft)
b. The high pressure turbine
c. A portion of the crossunder piping (100 ft) H471.2
d. The moisture separator/reheaters
e. The crossover piping
f. The low pressure turbines

The estimated inventory of N-16 is 195 Ci. After adjusting
for self absorption in the components, the equivalent
inventory was found to be 117 Ci of N-16. The sources are
surrounded by 24'-6" high walls on the north, south, and 23
east and a 31'-0" high wall on the west. The center of the
mid-LP turbine is 60'-10" from the east wall and 50'-0" from
the north wall. The area enclosed by the walls is 100'-0"
in the north-south direction and 204' in the east-west
direction.

The expected turbine shine dose at the Exclusion Area 23
Boundary (EAB), which is approximately 1.9 miles from the
Turbine Building, is conservatively estimated to be less
than 0.5 mrem/yr. The use of the EAB is conservative
because it is closer to the Plant than the nearest H471.2
unrestricted access road (WYE Barricade, approximately 2
miles). This estimate is based on operation of two units
with an availability factor of 80 percent. 23

The dose rates due to radiation scattered by air and the
walls surrounding the components on the turbine deck were
evaluated at several locations within the Turbine Building
but outside the shield walls. The maximum cumulative 331.32
contribution from these sources of exposure was found to be
less than 1.0 mrem/hr. The shield walls will be designed to
maintain the Zone II maximum of 2.5 mrem/hr.

12.1.3.10 Field Run Pipe Routing

The procedures for routing of field run piping are discussed in Section 12.1.2.3.2.

12.1.4 AREA RADIATION MONITORING

See Appendix 1A, Section 1A.1 and PSAR Table 12.1-4. 16

12.1.5 OPERATING PROCEDURES

The health physics program and access control described in Section 12.3 will ensure that Plant personnel exposures are kept as low as practicable during Plant operation and maintenance.

Operating experience of other BWR Plants will be continually evaluated to determine radiation levels present. Any high levels in areas not previously considered will be noted. Doses received by Plant personnel will also be noted. Procedures will be developed to ensure personnel exposure to radiation is maintained ALAP in accordance with Regulatory Guide 8.8. A description of these procedures will be provided in the FSAR. 6

12.1.6 ESTIMATES OF EXPOSURE

12.1.6.1 Anticipated Doses 331.6

The peak external doses within each area in the Plant will be considered as the maximum dose for which the area is zoned (Section 12.1.2.1). These doses are not expected to occur during normal operation because the Plant shielding is based on maximum coolant activities while the average isotopic concentrations will be considerably less than the maximum. The highest dose rates will occur in Zone V areas such as inside the drywell, in the turbine-condenser area, and in rooms containing equipment and piping handling highly radioactive fluids.

The direct radiation doses to the control room will be less than the zoned maximum dose rate of 0.5 mr/hr because of the 2'-0" thick concrete wall surrounding the control room. The Control Building roof is also 2'-0" thick concrete. Total shielding thickness above the control room including the ceiling, is 2'-9" of concrete. H471.1 The areas of other buildings adjacent to the control room are designated as Zone II, so the dose rate in them will not exceed 2.5 mr/hr. Conservatively assuming this dose rate is due solely to N-16 gamma radiation, the 2'-0" thick concrete wall will reduce the radiation level inside the control room to less than 0.05 mr/hr. Hence, the dose will be well within the design basis of 0.5 mr/hr. 23

The annual dose to the construction workers employed in Unit 2 while Unit 1 is in operation has been estimated for various points in the Unit 2 construction area; the locations of these dose points are shown on Figure 12.1-17. Radiation dose to construction workers will be due mostly to
the N-16 sources present in the operating Unit 1 turbine 23
system. Dose due to airborne effluents from Unit 1 will be 331.8
small in comparison with the dose from N-16 sources. The results of these estimated annual doses at ground level are listed in Table 12.1-30.

The doses from turbine shine were calculated in the manner 331.34
described in Section 12.1.3.9 based on 2,000 hr per yr. The resultant dose includes the direct, as well as air scattered, contribution. No credit was taken for the
shielding which will be afforded by the partially erected 10
Unit 2 structures. The radioactive wastes will be processed and stored in the Radwaste Building where shielding will be provided to insure that the dose outside the building will
be less than 0.5 mrem/hr. With an allowance for distance 331.8
between the Radwaste Building and the Unit 2 construction area, the estimated direct shine dose will be less than 0.01 mrem/hr.

The exposure for Unit 2 construction workers has been estimated based on the following assumptions: 331.36

a. The current schedule will be met.

b. Airborne doses are small compared to turbine shine doses.

c. Doses to personnel in the Unit 2 structures are negligible once the exterior walls and slabs have been fully erected. 10

d. Manual laborers spend 80% of their time inside the Unit 2 building and 20% of their time in yard areas. The contractor's nonmanual personnel spend 70% of their time in the field office and the remainder in Unit 2 structures. Construction management personnel spend 80% of their time in the field office and the remainder in Unit 2 structures.

e. The average dose rate in the yard areas is the
average of the dose rates at points A, C, D and E 23
of Figure 12.1-17, or .075 mrem/hr. The average 24
dose rate in the field office is .05 mrem/hr.

f. The availability factor for Unit 1 is 80%. Exposure to personnel in various categories and locations is summarized in Table 12.1-25 which gives the total estimated exposure to Unit 2 construction workers as 258 man-rem. 23

TABLE 12.1-14

Equipment Number	Description
1VS-705A	Auxiliary Bldg Central A/C Unit A
1VS-705B	Auxiliary Bldg Central A/C Unit B
1VS-709A	CTMT Press Control and Low Purge A/C Unit A
1VS-709B	CTMT Press Control and Low Purge A/C Unit B
1VS-721	Containment Low Purge Filter Train
1VS-722A	Standby Gas Treatment Filter Train A
1VS-722B	Standby Gas Treatment Filter Train B
1VS-728	Auxiliary Bldg LPCS & RHR Pump Room Air Rec Unit
1VS-729	Auxiliary Bldg RCIC & RWCU Pump Room Air Rec Unit
1VS-730	Aux. Bldg. RHR Pump Room C Air Rec Unit
1VS-732	Auxiliary Bldg. HPCS Pump Room Air Rec Unit
1VS-741A	Containment Air Rec Unit A
1VS-741B	Containment Air Rec Unit B
1VS-741C	Containment Air Rec Unit C
1VS-741D	Containment Air Rec Unit D
1VS-741E	Containment Air Rec Unit E
1VS-741F	Containment Air Rec Unit F
1VS-742A	Drywell Air Rec Unit A
1VS-742B	Drywell Air Rec Unit B
1VS-753A	Central Chilled Water Packaged Chiller A
1VS-753B	Central Chilled Water Packaged Chiller B
1VS-763A	CTMT Press Control and High Purge A/C Unit A
1VS-763B	CTMT Press Control and High Purge A/C Unit B
1VS-764	Aux. Bldg. RHR Pump Room A Air Rec Unit

TABLE 12.1-15

N-16 INVENTORIES IN EQUIPMENT IN THE TURBINE BUILDING (1)

Component	Estimated Volume (ft^3)	Estimated Transit Time to Component (sec)	N-16 Inventories (Ci)
Main Steam Piping (2)	4448	0	201.6
High Pressure Turbine	1023	2.04	25.37
Crossunder Piping	7008	2.357	61.84
Moisture Separato's	18,400	3.273	118.25
Crossover Piping	3719	5.291	17.13
Low Pressure Turbines	44,441	5.672	19.74
Gland Steam Evaporator	862	2.357	2.37(3)
High Pressure Heaters	3000	2.357	86.67(3)

(1) Based on 251 NSSS GESSAR Table 11.1.4 N-16 activities, Figure 10.1-2, and the assumption that the N-16 is uniformly partitioned.

(2) Includes piping in the Containment and the Auxiliary Building.

(3) Saturation value. Represents total N-16 inventory present beyond the extraction point.

consequences. Some of these events could, however, indicate a potential degradation in the level of Plant safety and/or could escalate to a more severe condition if appropriate on-Site action is not taken.

The primary purpose of notifying off-Site agencies of an Unusual Event is to place the off-Site response organization on standby, provide them with current information and provide unscheduled testing of the off-Site communication link. An Unusual Event classification will initiate augmentation of the on-Site response resources in order to assist in the assessment and mitigation of the event. Recommendations will specify that no off-Site actions are necessary. 23

4.2 ALERT

Events within the Alert class will indicate an actual or potential degradation in the level of Plant safety. The purpose of declaring this emergency class will be to assure that off-Site emergency personnel and monitoring teams are ready to respond if needed. This class may also serve as an unscheduled test of the activation of the on-Site and off-Site emergency response facilities and the related communication systems. The response of off-Site agencies will be to bring key elements of the emergency response organization into standby status, including off-Site monitoring teams.

This class of emergency also will initiate the activation of the Technical Support Center, the Operations Support Center and partial activation of the Emergency Operations Facility. 24

In addition to requiring manning of the on-Site response facilities, this emergency class may require radiological and meteorological assessments to be made and reported to the off-Site agencies. No public action will be recommended in this emergency class.

4.3 SITE AREA EMERGENCY

Events within the Site Area Emergency class involve actual or probable major failures of Plant functions needed for protection of the public. The purpose of declaring the Site Area Emergency class is to assure the manning of all emergency response facilities, the dispatching of monitoring teams and the assembling of personnel required for

evacuation, if such action becomes necessary. Declaration of this emergency class will also be used as a means of informing the off-Site agencies and the public that significant events are taking place. The response actions of off-Site agencies following such a declaration will include considering the implementation of protective actions and assessing information from S/HNP and off-Site monitoring teams. Protective action may include notification of the public within the plume exposure EPZ and instructions to seek shelter or evacuate. Meteorological conditions and dose estimates will be made and provided to the off-Site agencies. Assessment of Plant safety and release projections will be compiled for use in consultation with off-Site agencies.

4.4 GENERAL EMERGENCY

Events within the General Emergency class involve actual or imminent substantial core degradation or melting, with potential loss of containment integrity and/or releases of large quantities of radioactive material to the environment. 23
The purpose of declaring a General Emergency class is to initiate predetermined protective actions for the public and additional measures as indicated by event releases or potential releases. Declaration of this emergency class also provides for continuous assessment and information on events to off-Site agencies. Upon declaration of a General Emergency, all local, State and Federal response organizations will be fully activated. These agencies will use on-Site assessments and dose estimates, in addition to off-Site monitoring data, to implement the necessary protective actions. Public notification will be recommended. Protective actions may include instructions to seek shelter or evacuate persons within the plume exposure EPZ.

Declaration of a General Emergency will require that all of the S/HNP emergency response facilities be manned, appropriate on-Site protective actions be taken and monitoring teams be dispatched. Meteorological conditions and dose estimates will be provided to the off-Site agencies. Assessments of Plant safety and release projections will be compiled for use in consultation with off-Site agencies.

a. Transient analysis and system interaction

b. Nuclear engineering and fuel management

c. Core physics, design and control

d. Electrical power systems

e. Process computers

f. Instrumentation and control systems

g. Refueling

h. Engineering mechanics of systems and components

i. Thermal-hydraulics

j. Plant structural and containment design

k. Metallurgy

5.3.3 SITE SUPPORT MANAGEMENT

A staff composed of Plant personnel will be assigned the function of providing liaison between the Site and the recovery team in order to minimize the number of organizations and individuals communicating directly with the Site. The Site support staff is expected to contain personnel with expertise in the following areas: 23

a. Plant operations and maintenance

b. Radiation control and health physics

c. Radwaste management

d. Decontamination

e. Radiochemistry and chemical engineering

f. Fire protection

5.3.4 RADIOLOGICAL SUPPORT MANAGEMENT

Radiological support to recovery operations will be provided by the Radiological Emergency Manager, whose activities and responsibilities are described in Section 5.2.5.

5.3.5 PUBLIC INFORMATION SUPPORT MANAGEMENT

Personnel will be assigned to provide public affairs support to recovery operations by preparing and disseminating public information and interfacing with the news media.

5.3.6 ADMINISTRATIVE SUPPORT MANAGEMENT

Personnel will be assigned to provide necessary administrative and logistics support to recovery activities. Specific support activities essentially are the same as those 23
provided to the Emergency Director and are described in Section 5.2.8.

5.4 OFF-SITE ORGANIZATIONS

This section presents preliminary information on emergency response resources available outside of Puget. These include fire, medical, law enforcement, and radiological assistance. These external resources will be available through separate formal agreements or in the State and County emergency plans; copies of the formal agreements will be contained in the final S/HNP Emergency Program and State and County emergency plans as applicable. Letters of agreement indicating capability to provide future services are provided in Appendix A.

5.4.1 LOCAL SUPPORT SERVICES

Rapid response can be supplied to Puget by local fire, medical, law enforcement, and radiological organizations. Agreements will be made with response agencies, either through agreement letters or through the County and State emergency plans. In cooperation with Puget, Benton County Department of Emergency Services, the Department of Energy 24
and the Supply System, a training program will be established to familiarize responding agencies with their roles at the Plant Site during an emergency. Puget will provide escorts, assistance, and dosimetry for all persons entering the Plant Site.

5.4.1.1 Fire

Fire Department response on the Hanford Reservation is provided by the Rockwell Hanford Operations Fire Department. Lease agreements will be established between the DOE and Puget to provide for fire department response to S/HNP facilities on the Hanford Reservation. Response is provided by the 300-Area fire station located approximately 11 miles from the Site. Two other fire stations on the Reservation provide backup capability. Additional fire department response, if needed, is available through mutual aid agreements between county/municipal fire organizations. The Rockwell Hanford Operations Fire Department personnel are specifically trained to handle nuclear facility fire incidents and have specially equipped vehicles and personnel for rescue and other emergencies. 23

5.4.1.2 Medical

Kadlec Hospital in Richland is the closest hospital for handling patients from the S/HNP. Adjacent to Kadlec Hospital is the Emergency Decontamination Center, operated by the Hanford Environmental Health Foundation. Agreements will be maintained with the Kadlec Hospital and other Tri-City hospitals and Yakima Valley Memorial Hospital in order to provide for adequate medical capability including handling of radiation accident cases in the event of a large number of injured personnel. 24

5.4.1.3 Law Enforcement

The Rockwell Hanford Operations Patrol has jurisdiction on the Hanford Reservation highways. Arrangements will be made for this organization to provide traffic control or to limit access to the Hanford Reservation during an S/HNP emergency. This support can be activated on a 24 hr basis by phone or radio through the Hanford Patrol Emergency Officer in the Federal Building in Richland. 24

Members of the Rockwell Hanford Operations Patrol also are deputies of the Benton County Sheriff. The Benton County Sheriff has jurisdiction at the S/HNP Site for all civil disturbances or threats. Requests for assistance to the Benton County Sheriff can be made directly to the Sheriff's office or through the 24-hour Emergency Dispatch Center located in Kennewick. The S/HNP Communication Centers will

be capable of communication with responding law enforcement agencies.

In the event evacuation is required for areas off the Hanford Reservation, the Benton County Sheriff and the Franklin County Sheriff are the lead agencies for their respective counties.

The Washington State Patrol provides assistance to the Sheriff concerning traffic control and area access. The local Washington State Patrol headquarters is located in Kennewick and has direct radio communications with the Benton and Franklin sheriffs and the Emergency Dispatch Center. 23

5.4.1.4 Radiological

Extensive nuclear activities by the DOE on the Hanford Reservation result in a large number of trained personnel capable of assisting during a radiological emergency.

Because the DOE office is located in Richland, Federal response is available within two hours. The DOE, through the Federal Radiological Monitoring and Assessment Program (FRMAP), will provide field monitoring teams and dose assessment capability. To the extent available, assistance from the DOE will be provided upon notification and request by the Recovery Manager or Emergency Director. 24

5.4.2 PRIVATE RESPONSE ORGANIZATIONS

5.4.2.1 Other Utilities

Agreements will be negotiated with other utilities with operating nuclear power plants such as Portland General Electric Company and the Washington Public Power Supply System to provide field monitoring teams, equipment, health physics and technical support. Effective use of the response personnel would be to provide relief to Puget personnel during an extended emergency. Provisions will be made to include these organizations in Puget's training program.

5.4.2.2 General Electric Company

The General Electric Company (GE), supplier of the S/HNP Nuclear Steam Supply System, provides a support program utilizing the resources of their Nuclear Energy Group in San Jose, California. An Emergency Response Team can be dispatched to the S/HNP within 24 hours to provide technical assistance. A Technical Support Team with experts in appropriate technical disciplines will be activated at GE headquarters and will establish telephone communications with the on-Site GE Emergency Response Team. GE personnel may be used in the Plant as needed. 23

5.4.2.3 Institute of Nuclear Power Operations

The Institute of Nuclear Power Operations (INPO) was organized by the electric utility industry in response to the needs of the utility industry following the Three Mile Island accident. INPO is working in the areas of training, operating experience evaluation, criteria development, assistance to the utilities, and emergency preparedness. Assistance in emergency preparedness is provided to utilities to increase their ability to respond to an accident. During an accident resources such as equipment and manpower can be coordinated through INPO, particularly during the recovery phase. Representatives will be dispatched to the S/HNP, if requested, to provide assistance. Estimated response time is less than 24 hours. Assistance may include emergency resources search, accident analysis, and technical consultation.

5.4.2.4 American Nuclear Insurers

American Nuclear Insurers (ANI) provides insurance coverage for Puget's nuclear facilities. If an accident involving the general public occurs, ANI can provide financial assistance to persons who are required to relocate.

5.4.2.5 Exxon Nuclear

Exxon Nuclear maintains radiological staff to support Exxon activities. Assistance in monitoring radiological conditions along the southern Hanford Reservation boundary can be provided. 24

5.4.3 COUNTY ORGANIZATIONS

5.4.3.1 Counties within the Plume Exposure Emergency Planning Zone

Benton and Franklin Counties are located within the plume 23
exposure EPZ. A joint planning program between the two
counties has provided a consolidation of effort, resources,
and facilities into one plan for responding to a nuclear
emergency. Pre-emergency responsibilities include
coordinating local radiological response planning and
exchanging emergency preparedness information with Puget;
assisting local agencies and officials to develop
appropriate emergency capabilities; and promoting the 24
development of effective communications and other emergency
resources.

Emergency responsibilities of the County Emergency Services
Directors include providing communications support to the
Sheriff, participating in local public information
activities, establishing assistance centers and emergency 24
operations centers, assisting welfare services, and
coordinating local support of recovery efforts.

The Benton County Emergency Operations Center is located in
the Kennewick City Hall basement, 210 West 6th Avenue,
Kennewick, Washington. The Benton County Emergency
Operations Center is the facility in which the Benton and
Franklin Counties Boards of County Commissioners; Directors
of Emergency Services; Sheriffs; City Police 24
representatives; Fire Services Coordinators; City
Executives; and representatives of other Benton and Franklin
Counties emergency response organizations shall assemble to
direct and control implementation of protective actions in
Benton and Franklin Counties.

Notification of an emergency will be received by the Kennewick-Richland (Benton County) Emergency Dispatch Center from the S/HNP. This Emergency Dispatch Center will notify the Pasco (Franklin County) Emergency Dispatch Center. The Counties' required response actions will depend upon the emergency classification.

5.4.3.2 Counties within the Ingestion Exposure Emergency Planning Zone

Ten counties are located within the ingestion exposure EPZ, and the State of Washington will provide emergency

notification to these counties. If resources are needed, specific requests will be coordinated through the State Department of Emergency Services. Local governments will be kept apprised of all pertinent information regarding the Plant and will recommend protective actions.

5.4.4 STATE ORGANIZATIONS

The States of Washington and Oregon are located within the 50-mile EPZ. 23

5.4.4.1 State of Washington

The Washington State Fixed Nuclear Facility Emergency Response Plan provides for detailed emergency response for all State agencies. The following outlines the response actions of several of the State agencies which will respond during an emergency at S/HNP.

Department of Emergency Services

The Department of Emergency Services (DES) is the lead State agency for radiological response planning and operations. Responsibilities of this agency include:

a. Develop and maintain the Washington State Fixed Nuclear Facility Emergency Response Plan and assist the counties in developing their individual emergency response plans.

b. Arrange training programs for State and local agencies designed to promote effective response to radiological incidents.

c. Provide affected communities with warning confirmation, upon request.

d. Coordinate communications and other available support to affected local governments. 24

e. Coordinate the support of other State agencies and political subdivisions near the affected area and obtain assistance of Federal agencies as required. 24

A representative of the State DES may proceed to the Emergency Operations Facility during a Site Area Emergency or General Emergency, and if conditions warrant, during an Alert. Dedicated communications with the State Emergency 24

Operations Facility in Olympia will be available. Response time is estimated to be less than five hours. 24

Department of Social and Health Services

The Department of Social and Health Services, Health Services Division, is responsible for administering and directing radiation control programs and activities within the State. The State Radiation Control (Radcon) Field Team provides direct radiological emergency response capability and will be activated for Site Area or General Emergencies. 23
The Team's responsibilities are to:

a. Move immediately to the affected area and perform radiological monitoring, population dose estimates, and decontamination.

b. Determine and report the nature and scope of the hazard.

c. Provide local authorities with technical guidance, recommend appropriate emergency countermeasures and recovery actions, and otherwise assist the affected community.

The Department of Social and Health Services Health Division Director or his designee, is responsible for providing the public with health hazard evaluations, guidance on protective actions, and other pertinent information concerning radiological incidents. 24

A representative of the Washington State Department of Social and Health Services will respond to the Emergency Operations Facility. Response time is expected to be less than six hours for fully equipped teams.

Washington State Patrol

The Washington State Patrol will provide, upon request, emergency traffic control, area access control, escort services, communications, and other law enforcement assistance. 24

Energy Facility Site Evaluation Council

The Energy Facility Site Evaluation Council is the lead state organization for siting power generating facilities in the State of Washington. The Council is responsible for assessing the overall safety of the facility and ensuring the safety and well-being of the public.

Provisions will be made for a representative of the Energy Facility Site Evaluation Council to proceed to the Emergency Operations Facility to assess the situation.

5.4.4.2 State of Oregon

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The Oregon State Emergency Response Program provides the
detailed emergency response for the Oregon State agencies.
The Oregon State Health Division is responsible for
evaluating off-Site radiological hazards. The S/HNP
facilities are located approximately 35 miles from the
Oregon border. Two Oregon Counties, Morrow and Umatilla,
are within the ingestion exposure EPZ. For a General
Emergency, Oregon State will be notified and precautionary
measures such as food and milk analysis may be implemented
if wind conditions dictate. An Oregon State representative 24
may elect to proceed to the Emergency Operations Facility in
the event of a General Emergency to assess the level of
response which should be implemented by the State of Oregon.
Dedicated communications with the Oregon State Emergency 24
Operations Facility in Salem will be available.

5.4.5 FEDERAL ORGANIZATIONS

5.4.5.1 United States Nuclear Regulatory Commission

Personnel from the Nuclear Regulatory Commission (NRC) may elect to respond to any emergency. The NRC Resident Inspector assigned to the S/HNP is available to proceed rapidly to the Plant Site. He will be notified of any emergency condition on-Site. Facilities are available for NRC personnel in the on-Site TSC and the Emergency Operations Facility. Dedicated communications are available between these facilities and the NRC.

5.4.5.2 Department of Energy - Richland Operations

Activation of the DOE FRMAP will be requested by the 24
Emergency Director or Recovery Manager. A representative
from the DOE will proceed to the Emergency Operations
Facility to coordinate the effort. For a Site Area
Emergency or General Emergency, response actions will be
taken upon request of the Emergency Director. These will
include dispatching field survey teams.

The DOE - Richland Operations has a large available resource of equipment and manpower. Personnel from Pacific Northwest

Laboratories, Rockwell Hanford Operations, Westinghouse Hanford, United Nuclear Corporation and other Hanford Reservation contractors can provide assistance in the event of an emergency. Technical personnel and laboratory facilities from these organizations can be made available rapidly. 23

Assistance, which can be requested from the DOE, includes:

a. Emergency Decontamination Center for handling contaminated patients. 24

b. Rockwell Hanford Operations' ambulances as backup to Supply System's ambulances.

c. Rockwell Hanford Operations' fire department to respond to fires on the Hanford Reservation.

d. Rockwell Hanford Operations to provide additional respirators, protective clothing and equipment for personnel decontamination.

e. Rockwell Hanford Patrol to control access to the Hanford Reservation.

f. Pacific Northwest Laboratories to assist with field monitoring, dose assessment, meteorological data and laboratory analysis.

5.4.5.3 Federal Emergency Management Agency

The Federal Emergency Management Agency (FEMA) is assigned lead responsibility for non-technical Federal off-Site nuclear emergency planning and response. The Federal government will provide technical and/or logistical resource support at the request of the Washington State Department of Emergency Services.

Functions to be performed by FEMA include:

1. To jointly participate with the licensee, NRC, and State and local governments in affecting the coordination of emergency public information. 24

2. To assist with the logistical support to deal with the technical aspects of the technical response of the NRC and DOE.

3. To provide a unified set of Federal recommendations to the State and local government officials. 24

Federal emergency response will consist of technical and non-technical components. The NRC/DOE and FEMA jointly will coordinate Federal emergency response actions. The NRC/DOE and FEMA will coordinate the technical and non-technical aspects of Federal response, respectively. 23

Provisions will be made for a representative at the Emergency Operations Facility to work with the State and County representatives.

feasibility investigation on the joint use of their near-Site EOF. This feasibility study will be submitted to the NRC for review prior to proceeding with construction of the S/HNP portion of the joint EOF, if joint use of the Supply System near-site EOF is proposed. In the event that this study indicates joint use of the Supply System near-Site EOF is not feasible, a facility, meeting the requirements of NUREG-0696 (2/81), either will be constructed near the S/HNP Site, similar to the Supply System near-Site EOF, or will be located in Richland, Washington, approximately 12 to 15 miles from the S/HNP Site. Design information will be submitted to the NRC for review prior to proceeding with construction of the facility.

Should a backup EOF not be required at the time of EOF construction, Puget reserves the option of deleting the commitment to provide a backup EOF.

Based on discussions with the Supply System, the tentative use of the Supply System near-Site EOF is described below.

6.4.1 FUNCTION

The EOF is a near-Site support facility for the management of the overall Puget emergency response (including
coordination with Federal, State and local officials), 23
coordination of radiological and environmental assessments, and determination of recommended public protective actions. The EOF will have appropriate technical data displays and Plant records to assist in the diagnosis of Plant conditions to evaluate the potential or actual release of radioactive materials to the environment. A senior Puget official in the EOF will organize and manage Puget off-Site resources to support the TSC and the control room operators.

Based on conceptual planning, joint Puget and Supply System utilization of the near-Site EOF will: (a) improve emergency communications between Puget, Supply System and emergency response organizations; (b) facilitate coordination of off-Site radiological monitoring; (c) improve joint dose assessment, and (d) consolidate off-Site response organizations and Puget and Supply System decision centers. This conceptual planning has indicated that an S/HNP emergency should not interfere with the Supply System emergency activities due to an accident at WNP 1, 2 or 4.

Utilization of the near-Site EOF in the highly unlikely event of coincident WNP 1, 2 or 4 and S/HNP emergencies will be addressed in the feasibility study and described in the report to be provided to the NRC for review. Preliminary

information on joint use of the near-Site EOF during the highly unlikely event of coincident WNP 1, 2 or 4 and S/HNP emergencies is provided below.

6.4.2 JOINT SUPPLY SYSTEM AND PUGET UTILIZATION OF EOF

A preliminary floor plan of the near-Site EOF, indicating the joint utilization of the facility, is shown in Figures 14 and 15. An elevation view of the Plant Support Facility, showing the location of the EOF, is provided in Figure 16. A preliminary description of the joint use of the near-Site EOF facilities is provided below: 23

a. Supply System and Puget Utilization of Near-Site EOF

The Supply System and Puget will have separate Decision Centers to coordinate the individual activities of each company. The Decision Centers will maintain up-to-date information on scheduling of personnel, status of Plant conditions and radiological conditions.

b. Off-Site Agency Coordination Center 24

The Supply System and Puget will utilize a joint Off-Site Agency Coordination Center which will be used by representatives of the various agencies to coordinate activities, resolve problems, maintain duty rosters and provide periodic status reports to their respective agencies. This center receives information from the Meteorology and Unified Dose 24
Assessment and Communications Centers and initiates off-Site protective actions.

Periodic briefings will be provided by the Supply System and Puget in the off-Site Agency Decision Center to keep agency personnel up-to-date on Plant conditions and emergency measures underway.

c. Meteorology and Unified Dose Assessment Center 24

The Supply System and Puget will utilize a joint Meteorology and Unified Dose Assessment Center. 24
The Meteorology and Unified Dose Assessment Center will be utilized to coordinate Supply System, Puget and other radiological field monitoring teams to perform joint projected dose rate assessments, based on the Supply System and Puget facilities

source terms, and to record actual measured dose
rates. Actual and projected doses will be
available for use in making decisions concerning 23
protective actions for the general population.

Outside agencies, particularly the Washington State
Department of Social and Health Services, Oregon 24
State Health Division, and the DOE will work
closely with the Supply System and Puget to
evaluate dose projections. Washington State, the
DOE, the Supply System and Puget will utilize this
information to dispatch teams and work jointly in
evaluating field conditions.

d. Communications Center

The Supply System and Puget will utilize a joint Communications Center to provide radio contact with field monitoring teams, assisting agencies and other emergency centers.

e. Media Briefing Area

The Supply System and Puget will utilize a joint Media Briefing Area. The Media Briefing Area will be provided in a classroom in the unprotected portion of the Supply System Plant Support Facility Building. The primary Supply System and Puget media centers will be located at their respective corporate offices in Richland, Washington. Groups of media personnel will be escorted to the near-Site EOF, as conditions permit, for briefings and tours.

f. Media Briefing Preparation Area

The Supply System and Puget will utilize a joint Media Briefing Preparation Area with sufficient equipment for the Supply System and Puget public relations personnel and off-Site agency public information personnel to develop public information media releases.

g. Nuclear Regulatory Commission Work Area

The Supply System and Puget will utilize a joint work area for NRC personnel. The area will be limited to NRC personnel and will have adequate communications, including dedicated circuits to the NRC headquarters in Bethesda, Maryland.

h. Health Physics Center

The Supply System and Puget will utilize a joint Health Physics Center. The Health Physics Center will include external dosimetry, internal dose assessment, a radiological laboratory and respiratory testing facilities.

The external dosimetry area provides automated thermoluminescent dosimeter (TLD) readers which are sufficient to process the increased numbers of TLDs required during an emergency. Results are recorded in a Radiation Exposure Records System, which can be accessed for information from both Supply System and Puget Plant health physics areas.

The internal dosimetry area provides whole body counting facilities and a computerized internal dose assessment system. The facility is capable of accommodating about twelve whole body counts per hour.

The respiratory testing area consists of fitting booths and testing equipment. A large supply of respirators is available, if required, for emergency response operations. 23

The Radiological Laboratory and adjoining counting room provide for radiological analysis of environmental samples, as well as backup capability for Plant laboratories. If Plant analytical capabilities become unusable, Plant samples can be transported rapidly to the near-Site EOF for chemical and/or radiological analysis. The laboratory is directly accessible from the outside of the building, and can be isolated from the remainder of the building.

i. First Aid and Decontamination Area

The Supply System and Puget will utilize a joint First Aid and Decontamination Area which will be located in the unprotected portion of the Plant Support Facility adjacent to the ambulance garage. The layout of this area provides for simultaneous treatment and decontamination of injured personnel, as well as independent operation of the first aid and decontamination facilities.

6.4.3 LOCATIONS

The Supply System near-Site EOF is located approximately five miles from the S/HNP as shown in Figure 17. The S/HNP backup EOF will be located in Richland, Washington, approximately 12 to 15 miles from the S/HNP Site as shown in Figure 17.

6.4.4 STRUCTURE

The Supply System near-Site EOF will meet the requirements of the Uniform Building Code. In addition, it will be able to withstand adverse conditions of high winds (other than tornadoes) and floods with a 100-year recurrence frequency.

6.4.5 HABITABILITY

The Supply System near-Site EOF has special shielding and ventilation to maintain habitability requirements. Two feet of concrete equivalent shielding is provided to ensure that the total dose to occupying personnel is less than the Environmental Protection Agency Protective Action Guide limit of 5 rem whole body for the duration of the postulated accident. The ventilation system is designed to provide maximum habitability during an accidental radiological release. HEPA filters condition entering air during emergency conditions. Radiation detectors are strategically located in the ventilation system to detect impending infiltration of radioactive air, thus allowing reconfiguration of airflows from replenishment to recirculation modes. 23

6.4.6 EMERGENCY TECHNICAL INFORMATION

Equipment will be provided in the near-Site EOF as shown in Figure 14 to gather, store and display S/HNP data needed in the EOF in order to analyze and exchange information on Plant conditions with the designated senior Puget manager in charge of the S/HNP TSC. The EOF technical data system will receive, store, process and display information sufficient to perform assessments of the actual and potential on-Site and off-Site environmental consequences of an emergency condition. Data providing information on the general condition of the Plant also will be available for display in the EOF for Puget resource management. The EOF data set

will include radiological, meteorological and other environmental data needed to:

a. Assess environmental conditions

b. Coordinate radiological monitoring activities

c. Recommend implementation of off-Site emergency plans

The SPDS will be available for display in the EOF. This duplication will provide Puget management and NRC representatives information about the current S/HNP reactor systems status, and will facilitate communications among the control room, TSC and EOF.

The EOF will have ready access to up-to-date Plant records, 23
procedures, and emergency plans needed to exercise overall management of Puget emergency response resources. The S/HNP EOF records will include:

a. Plant technical specifications

b. Plant operating procedures

c. Emergency operating procedures

d. Final Safety Analysis Report

e. Up-to-date records related to Puget, State and local emergency response plans

f. Off-Site population distribution data

g. Evacuation plans

h. Environs radiological monitoring records

i. Puget employee radiation exposure histories

j. Up-to-date drawings, schematics and diagrams showing conditions of Plant structures and systems down to the component level and in-Plant locations of these systems.

6.5 JOINT INFORMATION CENTER

Puget will establish a joint information center (JIC) in its corporate offices in Richland, Washington for a Site Area
Emergency or General Emergency. The JIC will be utilized to 24
interface with Federal, State and local governments for the dissemination of emergency public information.

7.0 EMERGENCY MEASURES

The S/HNP will have adequate systems, equipment, facilities and procedures to promptly identify and monitor actual or potential consequences of an emergency condition within and outside the Site boundary. Emergency action levels will be utilized to determine the event classification and need for protective actions. The final S/HNP Emergency Plan and implementing procedures will describe in detail accident assessment methods and criteria.

7.1 EMERGENCY ACTION LEVELS

The S/HNP will utilize a system of Emergency Action Levels (EALs) for accident assessment. EALs are particular in-Plant conditions, instrument readings and on-Site and off-Site monitoring results that indicate an emergency event has occurred. The EALs provide a basis for categorizing the event into one of the following classifications: Unusual Event, Alert, Site Area Emergency or General Emergency. The EALs also provide the basis for determining the need for notification and participation of off-Site organizations and for determining when and what type of protective measures should be implemented. The EALs for the S/HNP will be agreed upon by Puget and State and local response organizations and will be approved by the NRC. EALs will be presented in the final S/HNP Emergency Plan. 23

7.2 ASSESSMENT CAPABILITY

The S/HNP will have the assessment capability and resources to provide: (a) initial evaluation and notification of the off-Site authorities within 15 minutes of the initiating event; and (b) continuing assessment of an emergency event throughout the course of an accident. This capability will include post-accident sampling capability, radiation and effluent monitors, in-Plant iodine instrumentation and containment radiation monitoring. Assessment of in-Plant conditions will be coordinated in the control room with the assistance of the TSC. Information provided by the Safety Parameter Display System (see S/HNP PSAR Appendix 1B, Item I.D.2) and the real-time meteorological instrumentation will be major tools in the accident assessment process. This information will be available in the control room and TSC, as well as in the near-Site EOF. Other sources of meteorological information, such as the Hanford meteorological network and the National Weather Service, may be assessed by telephone.

The S/HNP will have the capability to monitor the area within and outside the Site boundary, and to estimate and project radiation doses. Under the direction of the Radiological Emergency Manager (see Section 5.2.5) operating out of the EOF (see Section 6.4), radiological monitoring teams will be dispatched within the plume exposure EPZ. The data gathered by the monitoring teams will aid in determining the severity of the accident. The EOF will have available data from real-time meteorological monitoring instrumentation for use in computerized dose projection. The capability to manually project doses will be provided. 23

7.3 NOTIFICATION OF EMERGENCY ORGANIZATIONS

Notification is the initial contact of response organizations and the public, as necessary, for the purpose of alerting them that an emergency has occurred at the Plant. The notification process proceeds promptly, is of short duration and transmits a brief, but essential, amount of information. Notification is followed by messages providing more information, instructions to the public and/or specific recommendations.

Emergency planning for the S/HNP will include provisions for the prompt notification of appropriate Puget, State, local and Federal response personnel and organizations by the S/HNP. Sufficient communication capabilities, including backup systems, will be in place to effect prompt notification of all response organizations. In this regard, S/HNP shall have the assessment and communication capability needed to notify local governments within 15 minutes after declaring an emergency. 24 Plans and procedures will be established which govern the primary contact within response organizations and the means of contact, backup communications, the content of initial and follow-up messages and message authentication. Puget will work with appropriate State and local agencies to ensure that the physical and administrative means are in place to provide early and understandable warnings and instructions to the population within the plume exposure EPZ. Specific notification means, methods and procedures will be described in the final S/HNP Emergency Plan.

7.3.1 METHODS OF NOTIFICATION

7.3.1.1 Puget Emergency Organization

When an emergency has been classified, the Plant Emergency Director initiates notification of emergency personnel through use of the Communications Center and the in-Plant paging system.

The extent of the notification will depend upon the emergency classification declared; however, the Plant Emergency Director may call anyone he deems necessary to support the emergency effort. Table 2 lists the response organizations which will be notified for each emergency class.

7.3.1.2 Nearby Facilities

Upon declaration of any class of emergency, S/HNP will notify the WNP 1, 2 and 4 facilities. Both phone lines and radio communications are available for this notification.

In the event of an emergency requiring protective measures to be taken at WNP 1, 2 or 4, the Plant Emergency Director 23
will contact the WNP 1, 2 and 4 facilities and give specific instructions.

Initial notification to the Fast Flux Test Facility control room will be made by phone for an Alert, Site Area Emergency, or General Emergency. If protective measures are needed, notification will be made by the Plant Emergency Director and specific actions to be taken will be recommended.

The DOE will notify other facilities on the Hanford Reservation.

7.3.1.3 Support Organizations

Notification of the Emergency Dispatch Center and Benton County Emergency Operations Center in Kennewick, the DOE and the Washington State Department of Emergency Services (State Emergency Operations Center) will be made by dedicated phone lines. Notification of the NRC in Bethesda, Maryland, will be made by automatic ring-down telephone from the control room. The NRC offices in Bethesda will patch the call

through to the offices in Region V, Walnut Creek,
California, thus ensuring that both NRC offices receive
immediate notification. Notification of all other off-Site
emergency agencies will be made by telephone. Backup radio 23
communications to the Emergency Dispatch Center and DOE have
been established. A means for verifying the authenticity of
the notification also will be established.

7.3.1.4 General Public

The Sheriffs of Benton and Franklin Counties are responsible 24
for implementing emergency notification of the public,
including both transients and residents. This notification
will be based on information received from Puget. 24

Initial notification of an emergency at the Plant is made to the Emergency Dispatch Center. Follow-up messages and updates to the County Emergency Operations Center will be made by the Plant Emergency Director over the dedicated phone system. Radio communications also are provided as a backup. A decision is then made at the Benton County Emergency Operations Center to notify the public, based upon county emergency implementing procedures and recommendations of the Plant and State.

It is anticipated that two means will be provided by Puget for notification of the public in the plume exposure EPZ. Sirens also are located along the Columbia and Yakima Rivers in the plume exposure EPZ. The main function of the siren system is to notify the transient population, including construction workers.

For residents within the plume exposure EPZ, it is anticipated that tone-activated radios will provide the primary means of notification. These radios are activated by the Emergency Broadcast System signal and provide instructions for protective actions transmitted by the local Emergency Broadcast System stations.

The combined system is designed to provide both an alert signal and information to the population on an area-wide basis, throughout the plume exposure EPZ, within 15 minutes of a decision that the public is to be notified.

County Sheriff's Departments. Protective measures will be implemented initially in the areas directly affected by the plume from the Plant, as well as areas which would be affected if the plume changed direction.

The protective measures for the public will include notification of affected individuals in the plume exposure EPZ. As described in Section 7.3.1.4, Benton and Franklin Counties will have the capability to notify and instruct the public of emergency conditions at S/HNP. Instructions may include seeking of shelter, such as remaining indoors with windows and doors closed, or, under severe conditions, 23
evacuation.

If evacuation is deemed necessary, it will be carried out in accordance with detailed evacuation plans which will be contained in the Benton and Franklin Counties Emergency Response Plans. These plans will include the identification of evacuation routes, traffic control points, relocation centers, mass care centers and the individuals and groups responsible for performing evacuation-related functions.

A preliminary analysis has been conducted of the time required for both transient and permanent populations to evacuate various sectors within the S/HNP plume exposure EPZ. The analysis, provided in Appendix B, considered evacuation under a variety of circumstances. The analysis considered good and adverse weather conditions, transient populations, special facilities such as schools and nursing homes, and projected populations. The analysis demonstrated that there are no significant impediments to evacuation in the 10-mile area around the S/HNP.

The counties will coordinate all activities off the Hanford Reservation during the initial phases of the emergency. The Benton County Emergency Operations Center will coordinate activities with Puget and the Department of Energy through a county representative at the Nearsite Emergency Operations Facility. Puget will provide dose assessment and protective action recommendations to state and local governments.

On arrival of monitoring teams from the Washington State 24
Department of Social and Health Services, emergency assessment and protective actions within the plume exposure EPZ outside the Hanford Reservation will be directed by the State. Coordination of monitoring activities will be provided from the Nearsite Emergency Operations Facility.

7.4.2.2 Protective Measures for the Ingestion Pathway Emergency Planning Zone

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The ingestion EPZ is an area about 50 miles in radius around the S/HNP. The principal exposure source is from ingestion of contaminated water or food. The principal protective action is the control of food and water pathways. The States of Washington and Oregon have the responsibility to maintain communication with counties in the 50-mile 24 ingestion pathway EPZ. Dose projections and environmental sampling are also the States' responsibilities and will be coordinated from the Nearsite Emergency Operations Facility by representatives from the two states.

The Washington State Department of Social and Health Services, as the lead Washington State agency for recommending protective measures in the ingestion pathway EPZ, will specify the appropriate Washington State protective measures. 24 The Washington State Department of Agriculture is the lead agency for implementing food and milk protection measures. These measures will include, as necessary, controlling drinking water supplies, impounding milk, crops and other human or animal foodstuffs, and 23 establishing criteria for continuing consumption of these items.

The Governor of Oregon is responsible for directing protective measures to be used for the 50-mile ingestion pathway EPZ in the State of Oregon, including methods for protecting the public from consumption of contaminated 24 foodstuffs. The Health Division is responsible for implementing protective actions concerning water supplies. The Department of Agriculture is responsible for implementing protective actions concerning foods. Ingestion pathway EPZ protective action decisions and public information releases will be coordinated between the State of Washington and the State of Oregon.

Implementation of protective measures within the 50-mile ingestion pathway EPZ will be carried out by responsible agencies, including agricultural agencies, within the affected counties.

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7.5 AID TO AFFECTED PERSONNEL

Emergency planning for the S/HNP will include provisions for the treatment of individuals injured as a result of licensed activities at the S/HNP. These provisions will consist of limited on-Site facilities and arrangements with ambulance services and hospitals. Since it is possible that personal

injuries could be complicated by the presence of radioactive contamination, off-Site treatment facilities and personnel will be prepared to handle contaminated individuals. All off-Site treatment facilities and personnel with whom arrangements have been made will be involved in emergency 23
exercises and drills (see Section 8.3) and S/HNP sponsored training (see Section 8.2).

7.5.1 ON-SITE FACILITIES

On-Site decontamination facilities and provisions will be available in the Plant as described in PSAR Section 12.3.2.7. A first aid treatment center, equipped with necessary supplies, is located in the Service Building at elevation 562 as shown in Figure 12.1-15 and Figure 13 of 24
Appendix 13A. This center is in close proximity to the personnel decontamination facilities and the health physics area. These facilities will not be staffed full-time; rather, employees certified in first aid and trained in decontamination procedures will be available on all shifts to provide care. Further description of the facilities will be provided in the FSAR.

In addition, first aid and decontamination facilities are available in the Near-Site EOF, as described in Section 6.4.2.i.

7.5.2 OFF-SITE FACILITIES

7.5.2.1 Local Hospitals

Kadlec Hospital in Richland is the primary hospital for handling patients from the S/HNP facilities. Kadlec Hospital has 135 beds and 4 general surgery rooms. In 24
addition, two specialized surgical rooms are available for a large catastrophe. Approximately 90 physicians are on staff, with coverage in the emergency room on a 24-hour basis.

Kennewick General Hospital in Kennewick and Our Lady of Lourdes Hospital in Pasco are alternates to provide additional hospital and medical services. Both hospitals have emergency room coverage 24 hours per day. Yakima Valley Memorial Hospital in Yakima is a backup to the three Tri-City hospitals because of its alternate direction from S/HNP.

Puget will assist each hospital in developing procedures for the care of radiation accident patients. Training and assistance in drills to test procedures will be offered by Puget to each of these hospitals on an annual basis, or as requested.

7.5.2.2 Special Medical Facilities

23

If high levels of radioactive contamination or exposure are involved, injuries will be pretreated at the specially equipped Emergency Decontamination Center located adjacent to Kadlec Hospital. This facility is operated by the Hanford Environmental Health Foundation for the DOE in support of nuclear activities on the Hanford Reservation.

24

The Emergency Decontamination Center is a sophisticated unit specifically designed for the handling and treatment of radiation/contamination accident victims. Provisions include decontamination equipment, shielded surgery tables, remote TV viewing, contaminated water retention, air filtration, radiation monitoring instrumentation, whole body counting, and communications equipment. The physicians responding to the Emergency Decontamination Center are prepared and qualified to make radiation exposure evaluations.

The Hanford Environmental Health Foundation maintains an agreement with the University of Washington for treatment of radiation accident cases requiring specialized facilities and care at the Fred Hutchinson Cancer Research Center located in Seattle. This facility has a team of physicians and nurses skilled in the management of patients who lack bone marrow function and immunological competence and has the following features:

a. Complete tissue typing facilities for identification of compatible blood and marrow donors.

b. Facilities for collection of multiple units of platelets from a single donor.

c. Continuous flow centrifuges and leukofiltration facilities for the collection of large quantities of granulocytes.

d. Laminar air flow rooms for ultra isolation.

e. A research dietician and dietary team, skilled in preparation of sterile diets.

7.5.3 MEDICAL TRANSPORTATION

The Supply System maintains two radio dispatched ambulances at the Plant Support Facility which contain special equipment for transporting contaminated patients. Additional medical emergency transportation can be provided by the DOE. These ambulances are specially equipped for the handling of contaminated patients. If additional assistance is required, Tri-City ambulance services may be used through existing mutual assistance agreements. 23

7.6 REENTRY AND RECOVERY

Recovery operations will be carried out by the organization described in Section 5.0. Closely related to recovery is the reentry into damaged or contaminated areas of the Plant or Site. Recovery and reentry operations will be conducted in such a manner as to minimize exposure to workers and with total regard for the health and safety of the public. Plans for recovery and reentry activities will be described in detail in the final S/HNP Emergency Plan.

8.0 MAINTAINING EMERGENCY PREPAREDNESS

This section describes the means to be employed to ensure the S/HNP Emergency Program will be effective throughout the lifetime of the Plant.

8.1 PLAN PREPARATION AND UPDATING

Overall responsibility and authority for emergency preparation lies with the Manager, Licensing and Regulation. He assures the plan is prepared in accordance with applicable State and Federal regulations. The FSAR will describe the annual review, updating, and controlled distribution processes and the associated responsibilities of designated individuals.

8.2 TRAINING

23

Puget will establish an organized and comprehensive emergency response training program for S/HNP personnel, Puget headquarters support personnel, and participating local organizations. Personnel will receive initial training before the S/HNP begins operation, followed by annual retraining. A qualified person on the S/HNP staff will be designated as a training coordinator to manage the training program, develop programs, maintain documents and records and schedule training sessions.

8.2.1 PUGET PERSONNEL TRAINING

Members of the S/HNP Plant staff will receive detailed specialized training relative to their specific assignments at the Plant and their roles in emergency response in such areas as radiation monitoring, first aid, rescue, and damage control and repair. Individuals who will be in special positions of authority and significant responsibility, such as the Emergency Director and the Recovery Manager, will receive extensive training in all aspects of emergency planning and implementing procedures.

Persons working at the S/HNP Site, but outside the Plant itself, such as Security Guards, warehouse personnel, etc., will receive instructions on warning signals, assembly areas, evacuation routes and procedures.

Puget headquarters personnel having assignments in the emergency response and recovery organizations (see Section 5.0) will receive orientation in the content of the Emergency Plan and appropriate implementing procedures, as well as task-specific training for their assigned functions.

Unescorted personnel within the protected area will receive orientation in the content of the S/HNP Emergency Plan and implementing procedures. Plant personnel will receive instructions regarding the protection of escorted personnel, including visitors. 23

8.2.2 OFF-SITE PERSONNEL TRAINING

Puget's emergency training program makes provisions for the training of personnel from local, State and Federal organizations who may be requested to assist during an emergency. These organizations will be invited to participate in the training program. Tours of the areas in the facilities in which their assistance may be needed will be included. 24

Provisions will be established for the periodic retraining of these personnel. Every attempt will be made to utilize training offered by various Federal agencies.

8.3 DRILLS AND EXERCISES

Exercises will be conducted to assure that an adequate level of emergency preparedness at S/HNP is maintained. The exercises also will evaluate major portions of the on-Site and off-Site emergency response capabilities, will test the adequacy of the emergency plans and implementing procedures, and will ensure that personnel are familiar with their designated emergency response functions. The exercises also will serve as a test of the adequacy of emergency facilities, equipment and communication networks. Exercises will include participation and mobilization of local response and support organizations to verify their ability to respond to emergency conditions at S/HNP. Federal and State evaluators will critique the exercise, noting any deficiencies requiring correction. Exercises, involving the participation of appropriate State and Federal agencies, will be held. A preliminary exercise frequency schedule is provided in Table 3.

Drills will be conducted to test, develop and maintain the key skills necessary for effective response to emergency conditions at S/HNP. The drills consist of supervised

instructional periods directed toward particular operations in emergency response. The evaluation of the drills will be performed by a drill instructor who will use the experience to further develop the emergency response training programs for the on-Site and support organizations. The drills will cover the areas of communication, medical emergencies, radiological monitoring and emergency health physics activities. These drills will involve, to varying extents, the participation of State and local agencies and off-Site support organizations, as well as S/HNP personnel. A preliminary drill frequency schedule is provided in Table 3. 23

8.4 PUBLIC EDUCATION

Puget will establish, in conjunction with Federal, State and local organizations, a program for the annual dissemination of information to the public in the plume exposure EPZ. 24
This program will delineate the manner in which the public will be notified and the actions that should be taken in the event of an emergency. The information will include radiation, respiratory protection, sheltering, evacuation routes and the sources of additional information. Puget also will establish a regular program to annually acquaint news media personnel with emergency plans, radiation and sources of public information in an emergency. 24
The programs and means of implementation will be described in the final S/HNP Emergency Plan.

9.0 COMPATIBILITY

In conjunction with the preparation of the foregoing preliminary plans for responding to emergencies and the attached preliminary evacuation analysis, Puget has examined the compatibility of emergency plans with the S/HNP Site layout, Site location and facility design features. No factors have been identified which would indicate that the S/HNP will be incompatible with emergency planning. 23

The preliminary evacuation analysis demonstrates that access routes are sufficient to carry out a timely evacuation of the population within the plume exposure EPZ. The population density and distribution in the Site vicinity pose no problem to carrying out a timely evacuation or other lesser emergency action. Similarly, land use and local jurisdictional boundaries do not present any unique or difficult problems for carrying out emergency response plans. Special facilities within the plume exposure EPZ have been identified, and these facilities also pose no serious problems for emergency response planning. Finally, as Section 10.0 demonstrates, the S/HNP will satisfy the emergency planning standards of 10 CFR 50.47(b).

10.0 COMPLIANCE WITH REGULATIONS

10 CFR 50, Appendix E, Part II and 10 CFR 50.47(b) define the Federal regulations pertaining to emergency planning for nuclear power plants. Tables 4 and 5 provide a cross reference between these regulations, respectively, and the particular parts of this document wherein they are addressed. 23

TABLE 1

LOCATION OF KEY PUGET EMERGENCY RESPONSE PERSONNEL DURING ALERT OR GREATER EMERGENCIES

		Emergency Class	
Personnel	Alert	Site Area Emergency	General Emergency
Emergency Director	TSC	TSC	TSC
Radiation Protection Manager	TSC	TSC	TSC
Recovery Manager	TSC	TSC	TSC
Shift Technical Advisor	TSC	TSC	TSC
Emergency Communications Coordinator	TSC	EOF	EOF
TSC Supervisor	TSC	TSC	TSC
OSC Supervisor	OSC	OSC	OSC
On-Site Emergency Teams	OSC	OSC	OSC
Public Affairs Spokesperson	TSC	EOF,JIC	EOF,JIC
Off-Site Agency Liaison	TSC	EOF	EOF
Radiological Emergency Manager	TSC	EOF	EOF

23

TABLE 2 Sheet 1 of 3

NOTIFICATION AND ACTIVATION OF PRINCIPAL EMERGENCY RESPONSE ORGANIZATIONS

Organization	Unusual Event	Alert	Site Area Emergency	General Emergency	
Puget:					
On-Site Technical Support Center		A	A	A	23
On-Site Operations Support Center		A	A	A	
Communications Center	N,R	A	A	A	
Emergency Operations Facility		N,R	A	A	
Puget Headquarters		N,R	A	A	
County:					
Benton/Franklin Department of Emergency Services	N	A	A	A	
Emergency Dispatch Center	N	N	N	N	24
Benton and Franklin County Sheriffs	R	A	N,R	A	
State:					
Washington Department of Emergency Services	N	N,R	A	A	
Washington State Department of Social and Health Services	N	N,R	A	A	
Energy Facility Site Evaluation Council	N	N	N	N	

TABLE 2 Sheet 2 of 3

Organization	Unusual Event	Alert	Site Area Emergency	General Emergency	
Washington State Patrol	R	N,R	N,R	A	
Oregon State	N	N,R	A	A	24
Federal:					
Nuclear Regulatory Commission	N	N	A	A	
U.S. Coast Guard		N,R	N,R	A	
Federal Emergency Management Agency		N	A	A	
Department of Energy, Richland (will notify and activate DOE contractors)	N	N,R	A	A	23
Support Services:					
Supply System	R	R	A	A	
Exxon Nuclear	R	R	A	A	
General Electric		R	N,R	N,R	
Institute of Nuclear Power Operations	N	N	N,R	N,R	
American Nuclear Insurers	N	N	N,R	A	
Hanford Environmental Health Foundation	R	R	R	R	

24

TABLE 2 Sheet 3 of 3

Organization	Unusual Event	Alert	Site Area Emergency	General Emergency
Hospitals	R	R	N,R	N,R
BPA Substation		N	N	N

A = Activate N = Notify R = Respond if needed

23

TABLE 3

S/HNP EMERGENCY RESPONSE ORGANIZATION DRILLS AND EXERCISES

Drill/Exercise	Participants	Frequency	
			23
Joint emergency response exercises	S/HNP, local & off-Site support groups	Annually (plus or minus 3 months). First exercise about 6 months prior to low power license.	24
	S/HNP, local, off-Site support groups & State	At least once every 3 years	
	S/HNP, local, off-Site support groups & State & Federal	At least once every 5 years	
Communications Drills	S/HNP, State & local within plume exposure EPZ	Monthly	
	S/HNP, Federal & State organizations within ingestion EPZ	Quarterly	
	S/HNP, State & local EOC, and field assessment teams	Annually	
Medical Emergency Drills	S/HNP & off-Site support groups	Annually	
Radiological Monitoring Drills	S/HNP & State	Annually	
Health Physics Drills	S/HNP & State	Semiannually	

TABLE 4

S/HNP EMERGENCY PROGRAM: COMPLIANCE WITH 10 CFR 50, APPENDIX E, PART II

Regulatory Requirement	Section
Paragraph A	5.0 7.3 Table 1 Table 2
Paragraph B	5.4 7.3 7.4 Table 2
Paragraph C	7.1 7.2 7.3 7.4 7.6
Paragraph D	6.4.2.i 7.5.1 7.5.3
Paragraph E	7.5.2
Paragraph F	8.2 8.3 Table 3
Paragraph G	7.3 7.4 Appendix B
Paragraph H	5.2 7.4

23

APPENDIX A

LETTERS OF AGREEMENT

1. Benton County Department of Emergency Services (February 19, 1982)
2. Northwest Health Services Division of Preventative Medicine (March 11, 1982)
3. Our Lady of Lourdes Hospital (February 22, 1982)
4. Exxon Nuclear Company, Inc. (February 3, 1982)
5. Institute of Nuclear Power Operations (February 5, 1982)
6. Washington Public Power Supply System (February 25, 1982)
7. Department of Energy, State of Oregon (April 13, 1982)
8. Department of Emergency Services, State of Washington (March 3, 1982)
9. Department of Social and Health Services, State of Washington (March 5, 1982)
10. Energy Facility Site Evaluation Council, State of Washington (March 18, 1982)
11. Federal Emergency Management Agency, Region X (February 26, 1982)
12. Department of Energy, Richland Operations Office (March 2, 1982)
13. Department of Transportation, United States Coast Guard (March 8, 1982)

BENTON COUNTY
DEPARTMENT OF EMERGENCY SERVICES
~~JOHN D. DUNCAN, Director~~
Kennewick City Hall
P. O. Box 6144
Kennewick, Washington 99336-0144

Telephones:
Emergency: 911
Office: (509) 586-1451
Home: (509) 588-3188

February 19, 1982

Mr. M.V. Stimac, Manager
Licensing & Regulation
Puget Sound Power & Light Co.
Puget Power Building
Bellevue, WA 98009

SUBJECT: SKAGIT/HANFORD NUCLEAR PROJECT
PRELIMINARY EMERGENCY PLAN

Dear Mr. Stimac:

This is in response to the verbal request of Terry Grebel, made at the February 18th conference in this office, on the Skagit/Hanford Nuclear Project Preliminary Emergency Plan review.

Our comments on the plan were adequately addressed.

We understand that the final emergency plan will be a part of the Final Safety Analysis Report, and that prior to filing of that document we will have the opportunity to review and comment on the final plan. We also understand that appropriate contractual arrangements will be concluded as part of the final plan.

We appreciate the opportunity to participate in your development of plans for coping with emergencies, are looking forward to further dealings in this endeavor, and concur in the Preliminary Emergency Plan.

Sincerely,

Paulette H. Vopalensky

Paulette H. Vopalensky
Director

PHV/clc

NORTHWEST HEALTH SERVICES

MAR 1 5 1982

March 11, 1982

M. V. Stimac, Manager
Licensing and Regulation
Puget Power & Light Company
Puget Power Building
Bellevue, WA 98009

Dear Mike:

In reference to our telephone call of about two weeks ago I would like to call your attention to two places in your Preliminary Emergency Plan.

In Amendment 23, Table 2, Northwest Health Services was deleted as an emergency response organization. That was proper, as Northwest Health Services is our commercial branch and as such is not authorized to activate the Emergency Decontamination Facility (a DOE facility). However, HEHF is a DOE contract organization and has the proper authority. We feel that we should be listed on the notification and activation list.

You and I discussed the possibility of paragraph 7.5.2.2, page 13A-47 referring to a "Criticality" incident with extremely high induced radiation from a patient and the risk of handling them in the Emergency Decontamination Facility. I have reviewed this with our radiation physics people and they tell me that our facility could handle any person who survived a criticality long enough to be brought in. (See enclosure).

So I guess no change needs to be made in that paragraph. If any more information turns up I will contact you.

Very truly yours,

John F. Fulton, MD
Manager, Occupational Medicine

cd

enc:

TABLE 2 Sheet 2 of 3

Organization	Unusual Event	Alert	Site Area Emergency	General Emergency
Washington State Patrol	R	N,R	N,R	A
Oregon Health Division			N	A
Federal:				
Nuclear Regulatory Commission	N	N	A	A
U.S. Coast Guard		N,R	N,R	A
Federal Emergency Management Agency		N	A	A
Department of Energy, Richland (will notify and activate DOE contractors)	N	N,R	A	A
Support Services: 23				
Supply System	R	R	A	A
Exxon Nuclear	R	R	A	A
General Electric		R	N,R	N,R
Institute of Nuclear Power Operations	N	N	N,R	N,R
American Nuclear Insurers	N	N	N,R	A
Hanford Environmental Health Foundation ~~Northwest Health Services~~	R	R	R	R
~~U.S. Testing~~			R	R
~~Pacific Northwest Lab~~	R	R	R	R

OUR LADY OF LOURDES HOSPITAL

RECEIVED
FEB 25 1982
ENVIRONMENTAL & RESOURCE SERVICES

February 22, 1982

M. V. Stimac, Manager
Licensing & Regulation
Puget Sound Power & Light Company
Puget Power Building
Bellevue, WA 98009

RE: Skagit/Hanford Nuclear Project
Preliminary Emergency Plan

Dear Mr. Stimac:

We have reviewed your Preliminary Emergency Plan dated 12/21/81 and recommended changes by a number of agencies at Kennewick, Washington on February 18, 1982.

We accept the Preliminary Plan as amended on February 18, 1982 as it pertains to Our Lady of Lourdes Hospital in Pasco.

We welcome the opportunity to participate in this portion of the Final Safety Analysis Report and look forward to being able to review and comment on the final document.

Sincerely,

Sister Anthony Marie

Sister Anthony Marie
Administrator

SAM:kj

520 NORTH FOURTH AVENUE • POST OFFICE BOX 2568 • PASCO, WASHINGTON 99302 • TELEPHONE (509) 547-7704

EXXON NUCLEAR COMPANY, Inc.

2101 Horn Rapids Road
P. O. Box 130, Richland, Washington 99352
Phone (509) 375-8100 Telex 15-2878

February 3, 1982

Mr. M.V. Stimac, Manager
Licensing and Regulation
Puget Sound Power and Light Company
Puget Power Building
Bellevue, WA 98009

Dear Mr. Stimac:

Your letter to Mr. Purcell transmitting a copy of your Preliminary Emergency Plan has been passed to me for review and comment.

Exxon Nuclear Company, Inc. had not received prior contact from Puget Power with respect to emergency assistance. As a member of the nuclear community, we do have an interest in lending assistance in the case of an emergency. A mutual assistance agreement has been negotiated between Exxon Nuclear and Washington Public Power Supply System. I interpret Section 5.4 of the Preliminary Emergency Plan to mean that Puget Power intends to negotiate mutual assistance agreements with Exxon Nuclear and others prior to issuance of the final plan. If this interpretation is correct, we therefore have no objection to your including Exxon Nuclear as a provider of assistance in your preliminary emergency plan.

With respect to Section 5.4 of the draft plan, it would appear that Exxon Nuclear should be listed under 5.4.2 Private Response Organizations rather than 5.4.1 Local Support Services. Exxon Nuclear does not maintain radiological staff on weekends and holidays. Some staff could normally be supplied on those days by call out.

We do not intend to attend the comment meeting currently scheduled for February 18, 1982.

Further communications on this subject should be directed to my attention.

Sincerely,

C W Malody

C. W. Malody, Manager
Licensing & Compliance,
Operating Facilities

CWM:clc

INPO

Institute of
Nuclear Power
Operations

1820 Water Place
Atlanta, Georgia 30339
Telephone 404 953-3600

February 5, 1982

Mr. M. V. Stimac
Manager, Licensing and Regulation
Puget Sound Power and Light Company
Bellevue, WA 98009

Subject: Skagit/Hanford Nuclear Project Preliminary Emergency Plan

Dear Mr. Stimac:

INPO's Emergency Preparedness department has reviewed your Preliminary Emergency Plan as submitted. It is the opinion of the EP department that your plan is well done and we have no comments or suggestions for improvement at this time. We would appreciate being kept informed as your facilities and procedures progress through the design, construction, and implementation stages so that we may provide constructive criticism as your program evolves.

Sincerely,

P. W. Lyon

P. W. Lyon, Director
Radiological Protection
& Emergency Preparedness
Division

PWL:ACS/jky

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

February 25, 1982

Mr. M. V. Stimac, Manager
Licensing and Regulations
Puget Sound Power and Light Company
Puget Power Building
Bellevue, WA 98009

Dear Mr. Stimac:

The Supply System appreciates the opportunity to comment on the Puget Sound Power and Light Company preliminary emergency plan for the Skagit/Hanford Nuclear Projects. In reviewing the emergency plan and attending the review meeting of February 18, 1982, I feel confident that all areas of interface between the Supply System and Puget Power have been adequately addressed. The effort to work together in the area of emergency planning is to the benefit of both companies, and I feel we will be successful in this effort.

As stated in the Skagit/Hanford emergency plan, the Supply System will participate in an assessment of the joint use of the Supply System's Emergency Operations Facility to support the Skagit/Hanford nuclear plants during an emergency. In future years, I am sure there will be many areas of joint cooperation and utilization of resources in which the Supply System and Puget Power will work together.

If the Supply System can be of further assistance in your effort to construct and license the Skagit/Hanford nuclear facilities, please feel free to contact me.

Very truly yours,

Vincent Everett

J. V. Everett, Manager
Emergency Preparedness

JVE/cw

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APR 1 : 1982
ENVIRONMENTAL RESOURCE

STATE OF OREGON 1859
VICTOR ATIYEH
GOVERNOR

Department of Energy

LABOR & INDUSTRIES BUILDING, ROOM 102, SALEM, OREGON 97310 PHONE 378-4040

April 13, 1982

Mr. M.V. Stimac, Manager
Licensing and Regulation
Puget Power
Puget Power Building
Bellevue, WA 98009

Dear Mr. Stimac:

Puget Power has prepared a Preliminary Emergency Plan for the Skagit/Hanford Nuclear Project. On April 8, 1982, the Oregon Department of Energy commented on the proposed plan. Your letter of April 12, 1982 agreed to incorporate the Department's comments.

By incorporating ODOE's comments in the Preliminary Emergency Plan, you have resolved the Department's concerns. ODOE concurrs in the proposed plan.

Sincerely,

Lynn Frank
Director

LF/DWG:kg
1355C(D1/F1)

JOHN SPELLMAN
Governor

STATE OF WASHINGTON

DEPARTMENT OF EMERGENCY SERVICES

4220 E Martin Way • Olympia Washington 98504 • (206) 753-5255

March 3, 1982

Mr. M. V. Stimac, Manager
Licensing and Regulation
Puget Sound Power and Light Company
Puget Power Building
BELLEVUE WA 98009

Dear Mr. Stimac:

We have reviewed Appendix 13A "Preliminary Emergency Plan " of the Skagit/Hanford Nuclear Project (S/HNP) Preliminary Safety Analysis Report for coordination with the Washington State FNF Response Plan. Comments were forwarded to your office February 10, 1982. These comments were favorably resolved on February 18, 1982.

We understand that the "Final Emergency Plan" will be a part of the Final Safety Analysis Report (FSAR), and that this Department will review and comment on the final plan prior to submittal of the plan to NRC and the plan becoming part of the FSAR filing.

We appreciate your acknowledgement of the importance of this Department's participation in the development of plans for coping with radiological emergencies. This Department agrees that the Preliminary Emergency Plan correctly describes the state of Washington's emergency organization and capabilities.

Sincerely,

James M. Thomas
Assistant Director
Plans and Preparedness

JMT:ll

c: Mike Mills
Dick Donovan
Terry Grebel

JOHN SPELLMAN
Governor

RECEIVED
MAR 9 198

STATE OF WASHINGTON

DEPARTMENT OF SOCIAL AND HEALTH SERVICES

March 5, 1982

Mr. M. V. Stimac, Manager
Licensing and Regulation
Puget Sound Power & Light Company
Puget Power Building
Bellevue, Washington 98009

Dear Mr. Stimac:

This is in reference to the meeting on February 18, 1982 in Kennewick during which comments on the Skagit/Hanford Nuclear Project Preliminary Emergency Plan were discussed.

Based on discussions and changes made during the course of the meeting, we have determined that the plan correctly describes our organization and facilities. We understand that the final emergency plan will be a part of the Final Safety Analysis Report, and that prior to filing of that document we will have the opportunity to coordinate further development and to subsequently review and comment on the final plan. We also understand that appropriate memoranda of understanding and agreements will be part of the final plan.

We appreciate the opportunity to participate with you in developing plans for coping with emergencies, and concur in the preliminary emergency plan as changed.

Sincerely,

Jim

James F. Self, Jr., Manager
Emergency Response Program

Robert R. Mooney

Robert R. Mooney, Supervisor
Environmental Radiation and
Emergency Response Unit

JFS:RRM:jd

JOHN SPELLMAN
Governor

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ENVIRONMENTAL &
RESOURCE SERVICES

NICHOLAS D. LEWIS
Chairman

STATE OF WASHINGTON
ENERGY FACILITY SITE EVALUATION COUNCIL
Mail Stop PY-11 • Olympia, Washington 98504 • (206) 459-6490 • (SCAN) 585-6490

March 18, 1982

Mr. M. V. Stimac, Manager
Licensing & Regulation
Puget Sound Power & Light Company
Puget Power Building
Bellevue, WA 98009

Dear Mr. Stimac:

This is in response to your letter of January 22, 1982, requesting comments on the Skagit/Hanford Nuclear Project Preliminary Emergency Plan.

We have had an opportunity to review the preliminary plan and find that it correctly identifies the Council as an agency with off-site responsibilities in the event of an emergency. Council staff also had an opportunity to participate in the February 18 meeting in Kennewick where all of the comments received on this preliminary document were discussed. It is my understanding that the reviewing agencies comments were satifactorily resolved at that meeting and that you are proceeding to incorporate the agreed upon changes into a revised version of the plan for submittal to the Nuclear Regulatory Commission.

It is further understood that the final emergency plan for the project will be a part of the Final Safety Analysis Report, and that prior to filing of that document we will have additional opportunities to coordinate the development of the final plan with you. Your efforts to involve interested organizations and facilities in the development of the plan at this early stage in the licensing process will help to ensure that the requirements for coordination with response agencies are met.

Thank you for the opportunity to participate in the review of the preliminary plan and we look forward to our continued involvement in the development of the project's Emergency Plan.

Sincerely,

Bill Fitch

William L. Fitch
Executive Secretary

WLF:mg

Federal Emergency Management Agency
Region X Federal Regional Center Bothell, Washington 98011

February 26, 1982

Mr. M. B. Stimac
Manager, Licensing & Regulations
Puget Sound Power & Light Company
Puget Power Building
Bellevue, Washington 98009

Dear Mr. Stimac:

I have previously forwarded to you my comments on the Skagit/Hanford Nuclear Project Preliminary Emergency Plan.

I thought the February 18, 1982 meeting and joint rewrite session addressed my concerns and those of the participating organizations.

I understand that the final emergency plan will be a part of the FSAR, and that prior to the filing of that document, I will have the opportunity to review the plan.

Sincerely,

Richard W. Donovan

Richard W. Donovan
RAC Chairman

Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

March 2, 1982

Mr. M. V. Stimac, Manager
Licensing and Regulation
Puget Sound Power and Light Company
Puget Power Building
Bellevue, Washington 98009

Dear Mr. Stimac:

SKAGIT/HANFORD NUCLEAR PROJECT PRELIMINARY EMERGENCY PLAN

This is in response to your letter of January 22, 1982, requesting comments on the Skagit/Hanford Nuclear Project Preliminary Emergency Plan.

We have reviewed the Plan and have determined that it correctly describes our organization, facilities, and the support functions that we can provide. Minor editorial changes were provided to you during our meeting on February 18, 1982. It is our understanding that the final emergency plan will be a part of the Final Safety Analysis Report, and that prior to your filing of that document we will have the opportunity to review and comment on the final plan. We also understand that appropriate contractual arrangements will be included as part of the final plan.

We appreciate the opportunity to work with you in integrating our mutual emergency preparedness programs and to provide a coordinated emergency preparedness program in response to Federal, State, and local needs and requirements. We look forward to our continued good working relationships with your emergency preparedness planning staff.

Very truly yours,

R. E. Gerton, Director
Safety and Quality Assurance
Division

SQA:PHT

DEPARTMENT OF TRANSPORTATION
UNITED STATES COAST GUARD

Captain of the Port
Marine Safety Office
6767 N. Basin Avenue
Portland, Oregon 97217
Phone: (503) 221-6326

RECEIVED
MAR 1 8 1982
ENVIRONMENTAL & RESOURCE SERVICES

8000
8 March 1981

Mr. M. V. Stimac
Manager, Licensing and Regulation
Puget Sound Power and Light Company
Puget Power Building
Bellevue, Washington 98009

Dear Mr. Stimac:

I offer the following comments regarding the "S/HNP Preliminary Emergency Plan":

1. Section 5.4.5.4 states that the estimated response time to effect a river closure is one and one half hours. I find this to be a reasonable time frame to accomplish the task. I also approve of the provision to notify Coast Guard Station Kennewick directly as the radiological situation dictates.

Sincerely,

G. K. GREINER, Jr.
Captain, U.S. Coast Guard
Captain of the Port

S/HNP PSAR
CHAPTER 15
ACCIDENT ANALYSIS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

15.2 INCREASE IN REACTOR PRESSURE

15.2.1 PRESSURE REGULATOR FAILURE (CLOSED)

See 251 NSSS GESSAR Section 15.1.6.

15.2.2 GENERATOR LOAD REJECTION - TURBINE CONTROL VALVE (TCV) FAST CLOSURE

See 251 NSSS GESSAR Section 15.1.1.

15.2.3 TURBINE TRIP

See 251 NSSS GESSAR Section 15.1.2.

15.2.4 MAIN STEAM ISOLATION VALVE CLOSURE

15.2.4.1 Identification of Causes

See 251 NSSS GESSAR Section 15.1.4.1. 23

15.2.4.2 Analysis of Effects And Consequences

15.2.4.2.1 Methods

See 251 NSSS GESSAR Section 15.1.4.2.1.

15.2.4.2.2 Assumptions and Conditions

See 251 NSSS GESSAR Section 15.1.4.2.2.

15.2.4.2.2.1 Radiological consideration. As noted in 251 NSSS GESSAR Section 15.1, due to the smaller dose effect for the Type I (Turbine Trip) transient, only the effects for the Type II (MSIV closure) transient will be discussed. The radiological exposures evaluated are the external whole body and skin exposures and the internal exposure from

inhalation. These evaluations are performed for both on-Site and off-Site personnel and are based on the source terms presented in Section 12.2.3. 23

15.2.4.2.2.1.1 On-Site. Considering an equilibrium off gas release rate of 10^5 µCi/sec after 30 minutes prior to actuation of the relief valve and based on the transitory conditions existing as a consequence of turbine trip (or similar occurrence) and MSIV closure, the resultant activity airborne in the Containment is presented in 251 NSSS GESSAR Figure 12.2-3. Considering an average occupancy in the Containment after actuation of the relief valve of 4 minutes and uniform mixing within the annular area immediately above the suppression pool (ie, 270,000 cubic ft), the resultant dose effects are presented in Table 15.2-2. The whole body dose effects are determined by dividing the annular area into a very large number of differential volume elements and approximating the activity in the volume element by a point source. The total dose effect is therefore the sum of the individual dose effects and takes into consideration the energy spectrum associated with the released activity, as well as attenuation between the point source and the dose recipient. Due to the short range of beta particles, the beta dose, as well as the inhalation dose, is primarily dependent upon activity immediately surrounding the dose recipient and is therefore based upon those models specified in Regulatory Guide 1.3. The inhalation dose is presented for both the thyroid, which is iodine controlled, and the lung, which is based on the noble gas daughter products. 24

Tests have been performed under simulated conditions to determine realistic operator egress times from the Mark III Containment. The distance and elevation an operator would be required to travel in a Mark III Containment were simulated as closely as possible. In all cases the operators were able to travel the required distance in less than two minutes. Therefore a four minute egress time is conservative.

15.2.4.2.2.1.2 Off-Site. The radiological exposures which are calculated to occur as a consequence of releasing the airborne activity identified in Section 12.2.3 are based on the assumption that the release occurs during average annual meteorological conditions from an effective release height of zero meters. The realistic X/Q values described in Section 2.3.4 are used to determine the off-Site radiological exposures.

15.2.8 FEEDWATER SYSTEM PIPING BREAK 23

15.2.8.1 Identification Of Causes

15.2.8.1.1 Starting Conditions And Assumptions

Prior to this event the reactor is operating at normal Plant operating full power.

15.2.8.1.2 Accident Description

Accidents that result in the release of radioactive materials outside the Containment are the results of postulated breaches in the reactor coolant pressure boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the Design Basis Accident for breaks outside the Containment is a complete severence of one of the main steam lines as described in Section 15.6.4. The feedwater system piping break is less severe than the main steam line break.

Sequence of Events	Approximate Elapsed Time
a. Feedwater pipe circumferentially breaks between the last high pressure heater and the outboard feedwater check valve.	0.0
b. Feedwater flow into vessel reaches zero and feedwater check valves in the broken line isolate the reactor from the break.	4.0 secs
c. Low reactor vessel water level scrams the reactor and the main turbine trips from load mismatch.	8 secs
d. The feedwater pipe break has reduced the reactor feed pump suction pressure or the condensate pump discharge pressure sufficiently to start standby condensate pumps. The increased differential pressure across the condensate demineralizers automatically opens the bypass around the demineralizers. 24	

e. Low water level in reactor closes the main steam line isolation valves. — 30 secs

f. Steam for the turbine-driven feed pumps has been exhausted either from the main turbine cross-around piping or the steamlines between the main steam line isolation valves and the main turbine stop valves. Reactor feedpumps will continue to windmill with flow from the condensate pumps.

g. Inventory of water in the main condenser hotwell is completely pumped out of the break by the condensate pumps. — 7 mins

h. The feedwater lines between the last feedwater heater and the break complete draining out of the break. — 15 mins

i. Event ends.

15.2.8.1.3 Identification Of Operator Actions

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The operator maintains adequate reactor coolant inventory with RCIC and/or HPCS. The feedwater line check valves isolate the reactor from the break; no operator actions are necessary to effect reactor isolation.

Sequence Of Operator Actions	Approximate Elapsed Time
a. Event begins - failure occurs.	0
b. The operator determines that line break has occurred and evacuates that area of the Turbine Building. The operator shuts down the condensate and heater drain pumps.	10 mins
c. The operator is not required to take any action to prevent primary reactor system mass loss, but should insure reactor is shut down and that RCIC and/or HPCS are operating normally.	

15.2.8.2.2.2.4 Radiological effects. Due to the type of activity released, the radiological exposures are primarily limited to an inhalation dose to the thyroid gland, with secondary effects to the whole body. The radiological consequences for this event are evaluated in accordance with those assumptions and conditions specified in Section 15.6.4 for the steam line break accident. 23

The radiological exposures calculated for this event are presented in Table 15.2-4. It should be noted that the maximum off-Site exposures are orders of magnitude below the guidelines set forth in 10 CFR 100.

15.2.8.2.2.3 Consideration of uncertainties. This event was conservatively analyzed. Due to this approach no uncertainties were evaluated.

15.2.8.4 Comparison of Realistic vs. Conservative Evaluation

As noted above, an NRC guided evaluation of this accident was not made. Therefore, no comparison can be made. However, those parameters of significance in evaluating the consequences of this event are presented in Table 15.2-5.

15.2.8.5 References

N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors" APED 5756, March 1969.

15.2.9 CORE COOLANT TEMPERATURE INCREASE 24

See 251 NSSS GESSAR Section 15.1.27.

15.4.9.5.1.2.1 through 15.4.9.5.1.2.2. See 251 NSSS GESSAR Section 15.1.38.5.2.2.1 through 15.1.38.5.2.2.2.

15.4.9.5.1.2.3 Condenser activity. Based on the failure and transport mechanisms defined above, and assuming a partition factor of 100 in the condenser for iodines, the activity airborne in the condenser is presented in Table 15.4-4. 23

15.4.9.5.1.2.4 Activity released to environment. The fission product activity released to the environment is dependent upon the activity airborne in the condenser, the condenser leak rate, and the Turbine Building leak rate. For the purpose of this analysis it is assumed that the condenser leak rate is 1.0% per day and the Turbine Building leak rate is infinite. 24 Based on the airborne activity presented in the previous sections and the above leakage rates, the noble gas and iodine releases to the environment are presented in Table 15.4-5.

15.4.9.5.1.2.5 Radiological effects.

a. Off-Site. While no specific guidelines have been written by the NRC for this accident, it is assumed that the meteorological conditions of 5 percent probability level for an equivalent ground level release apply for this event. Consideration of the fission product releases in Table 15.4-5 and the above meteorology results in the radiological exposures presented in Figures 15.4-3 and 15.4-4. It should be noted that these exposures are well below the guidelines set forth in 10 CFR 100.

b. Control Room. Potential dose to the personnel in the control room has been evaluated for the assumptions used by the NRC in the past. Potential dose contributions from direct shine, overhead clouds, and inhalation are considered. Direct shine conditions are evaluated for a case where steam carrying the released activity is located in the piping equipment above the operating floor. A second direct shine condition assumes that all of the activity released from the vessel is in the main condensers. The maximum dose rate occurs while the activity is being transported through the piping on the operating floor. This condition is a factor in control room design. The cloud dose is not significant. The inhalation dose is limited to acceptable levels by the control room ventilation system. The control room personnel doses are presented in Table 15.4-6

15.4.9.5.2 Realistic Evaluation Methods 23

See 251 NSSS GESSAR Section 15.1.38.5.1.

15.4.9.5.2.1 Methods, assumptions and conditions. See 251 NSSS GESSAR Section 15.1.38.5.1.1.

15.4.9.5.2.2 Results and consequences.

15.4.9.5.2.2.1 through 15.4.9.5.2.2.2. See 251 NSSS GESSAR Section 15.1.38.5.1.2.1 through 15.1.38.5.1.2.2.

15.4.9.5.2.2.3 Condenser activity. The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

15.6.2 OFF DESIGN OPERATIONAL TRANSIENT AS A CONSEQUENCE OF INSTRUMENT LINE FAILURE

15.6.2.1 through 15.6.2.4

See 251 NSSS GESSAR Sections 15.1.35.1 through 15.1.35.4. 23

15.6.2.5 Analysis of Effects and Consequences

15.6.2.5.1 Conservative (NRC) Licensing Basis Evaluation Methods 24

The design basis analysis is based on NRC Standard Review Plan 15.6.2 and NRC Regulatory Guide 1.5. The specific models, assumptions, and the program used for computer evaluation are described in Appendix 15A. Values of parameters used in the evaluation are presented in Table 15.6-16.

The assumptions and calculation methodology used are as follows:

a. Spiking factor

The activity released from the fuel to the coolant as a consequence of reactor scram and vessel depressurization was based on measurements during plant shutdowns (Ref 2). It was shown that for a 95 percentile probability, a total of 7 Ci of I-131 is released to the coolant for every 1 μCi/sec of prespike I-131 release. This conservative ratio was applied for all the iodine isotopes for the dose analysis. The prespike iodine releases were those that correspond to a 0.35 Ci/sec noble gases release after 30 min. decay, a design basis accident assumption.

b. Iodine concentration in coolant

The total iodine released from the fuel to the coolant was assumed to take place in a span of 5 hours, resulting in continued buildup of coolant activity during that period. The coolant activity during 0-2 hours was assumed to be constant and equal to that at the end of the first hour. The coolant activity during 2 to 5 hours was assumed to be equal to that at the end of 3-1/2 hours. This is a conservative

assumption, since the rate of increase in coolant activity decreases with time.

c. Partition factor

It was assumed that 100% of the activity in the coolant that flashed into steam remains airborne and that 10% of the activity carried by the coolant water into the secondary containment becomes airborne (corresponding to a conservative partition factor of 0.1).

d. Activities in the containment are released to the environment.

Air in the containment is vented directly to the atmosphere for the first ten minutes; then the containment is isolated and air is routed to the Secondary Containment (Enclosure Building) prior to release to the atmosphere via the Standby Gas Treatment System (SGTS). The SGTS filter has an efficiency of 99%.

The activity airborne in the containment and in the secondary containment are presented in Tables 15.6-8 and 15.6-9. The activity released to the environment is presented in Table 15.6-10. 23

e. The 5% probability level X/Q values are used for this analysis (see Table 15.6-17).

The calculated exposure at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are presented in Table 15.6-15.

15.6.2.5.2 Realistic Evaluation Methods

See 251 NSSS GESSAR Section 15.1.35.5.1.

15.6.2.5.2.1 Methods, assumptions and conditions. See 251 NSSS GESSAR Section 15.1.35.5.1.1.

15.6.2.5.2.2 Results and consequences. See 251 NSSS GESSAR Section 15.1.35.5.1.2.

15.6.2.5.2.2.1 through 15.6.2.5.2.2.2 See 251 NSSS GESSAR Sections 15.1.35.5.1.2.1 through 15.1.35.5.1.2.2.

TABLE 15.6-1

TYPE III AND IV S/R VALVE TRANSIENT PARAMETERS
TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	NA	4100 MWt
B. Burn-up	NA	NA
C. Fuel damaged	NA	None
D. Release of activity by nuclide	NA	Sect. 12.2.3
E. Iodine fractions	NA	
(1) Organic	NA	0
(2) Elemental	NA	1.0
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	NA
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	Infinite [a]
B. Secondary conainment leak rate (%/day)	NA	NA
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	99%
(2) Elemental iodine	NA	99%
(3) Particulate iodine	NA	99%
(4) Particulate fission products	NA	99%
E. Recirculation system parameters		
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	15.6.1.2.1
III. Dispersion Data		
A. EAB and LPZ distances (m)	NA	3058/6437
B. X/Q values in sec/m^3	NA	$1.2x10^{-5}$/ $3.0x10^{-6}$
IV. Dose Data		
A. Method of dose calculation	NA	Sec 15.6.1.2.1
B. Dose conversion assumptions	NA	Sect. 15.6.1.2.1
C. Activity in containment	NA	Sect 12.2.3
D. Doses	NA	Tables 15.6-2, 15.6-3, 15.6-4

(a) Applicable 8 hours after S/R valve transient commences

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TABLE 15.6-2

TYPE III TRANSIENT ON-SITE DOSE

Organ Evaluated	Dose Effect (mrem)
Whole Body	43
Skin	119

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TABLE 15.6-9

INSTRUMENT LINE FAILURE
ACTIVITY AIRBORNE IN THE ENCLOSURE BUILDING, CURIES 23
(CONSERVATIVE ANALYSIS)

Isotope	1 Min	1 Hr	2 Hr	8 Hr	1 Day	4 Days	30 Days	
I-131	-0-	6.50E-5	2.58E-4	5.55E-3	1.29E-2	1.19E-2	1.20E-3	
132	-0-	6.00E-4	1.79E-3	5.65E-3	1.21E-4	7.29E-14	0	
133	-0-	2.95E-4	1.14E-3	2.01E-2	2.90E-2	3.21E-3	3.19E-12	24
134	-0-	6.98E-4	1.31E-3	1.74E-4	1.38E-9	0	0	
135	-0-	3.40E-4	1.23E-3	1.36E-2	6.42E-3	4.65E-6	0	
TOTAL	-0-	2.0E-3	5.72E-3	4.50E-2	4.84E-2	1.52E-2	1.20E-3	

TABLE 15.6-10

INSTRUMENT LINE FAILURE
ACTIVITY RELEASED TO THE ENVIRONMENT, CURIES
(CONSERVATIVE ANALYSIS)

Isotope	1 Min	1 Hr	2 Hr	8 Hr	1 Day	4 Days	30 Days
I-131	3.77E-5	3.69E-3	1.26E-7	1.30E-5	1.35E-4	7.95E-4	2.43E-3
132	4.77E-4	4.51E-2	9.84E-7	2.52E-5	2.26E-5	3.55E-7	0
133	1.76E-4	1.72E-2	5.63E-7	4.98E-5	3.78E-4	7.57E-4	8.06E-5
134	9.30E-4	8.31E-2	8.80E-7	3.88E-6	2.19E-7	0	0
135	2.20E-4	2.13E-2	6.23E-7	3.87E-5	1.50E-4	5.69E-5	3.75E-8
Total	1.84E-3	1.70E-1	3.18E-6	1.31E-4	6.86E-4	1.61E-3	2.51E-3

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TABLE 15.6-25

LOSS-OF-COOLANT ACCIDENT
ACTIVITY AIRBORNE INSIDE PRIMARY CONTAINMENT(1)
CONSERVATIVE ANALYSIS

Isotope	0.0 Hr	2.0 Hr	8.0 Hr	24.0 Hr	96.0 Hr	720.0 Hr
I-131	2.57E07	6.13E06	5.96E06	3.38E06	1.40E06	3.90E05
132	3.90E07	5.17E06	8.63E05	4.44E03	1.20E-06	0
133	5.76E07	1.30E07	1.06E07	3.72E06	1.85E04	1.85E-04
134	6.74E07	3.32E06	2.85E04	5.36E-02	0	0
135	5.23E07	1.02E07	5.48E06	6.32E05	2.05E02	0
Kr-83m	1.70E07	8.23E06	9.28E05	2.76E03	1.17E-08	0
85m	5.32E07	3.87E07	1.49E07	1.17E06	1.23E01	0
85	2.72E05	2.72E05	2.72E05	2.72E05	2.70E05	2.51E05
87	9.57E07	3.30E07	1.35E06	2.67E02	0	0
88	1.31E08	7.95E07	1.77E07	3.23E05	4.81E-03	0
89	1.63E08	7.24E-04	0	0	0	0
Xe-131m	1.06E06	1.06E06	1.04E06	1.00E06	8.36E05	1.75E05
133m	5.67E06	5.53E06	5.13E06	4.19E06	1.68E06	6.22E02
133	2.31E08	2.28E08	2.20E08	2.02E08	1.35E08	4.16E06
135	2.20E08	1.89E08	1.20E08	3.54E07	1.48E05	0
135m	6.38E07	3.10E05	3.54E-02	0	0	0
137	2.09E08	1.15E-01	0	0	0	0
138	1.96E08	1.47E06	6.21E-01	0	0	0

(1) Units for activities are in curies

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TABLE 15.6-26

LOSS-OF-COOLANT ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT[1]
CONSERVATIVE ANALYSIS

23

Isotope	0-2 Hr	2-8 Hr	8-24 Hr	24-96 Hr	96-720 Hr
I-131	1.77E02	2.52E02	5.65E02	9.42E02	2.48E03
132	2.25E02	1.00E02	1.97E01	9.37E-02	0
133	3.90E02	4.89E02	8.15E02	4.88E02	4.04E01
134	3.03E02	2.90E01	2.51E-01	0	0
135	3.39E02	3.18E02	2.78E02	3.39E01	1.44E-02
Kr-83m	2.52E03	2.09E03	2.64E02	7.87E-01	0
85m	9.47E03	1.55E04	8.95E03	7.60E02	7.81E-03
85	5.67E01	1.70E02	4.52E02	2.02E03	1.69E04
87	1.22E04	6.17E03	2.63E02	5.20E-02	0
88	2.15E04	2.57E04	7.22E03	1.34E02	0
89	1.30E03	0	0	0	0
Xe-131m	2.21E02	6.55E02	1.70E03	6.86E03	2.74E04
133m	1.17E03	3.32E03	7.71E03	2.05E04	1.38E04
133	4.77E04	1.40E05	3.51E05	1.24E06	2.44E06
135	4.24E04	9.46E04	1.15E05	4.82E04	2.03E02
135m	2.48E03	1.21E01	0	0	0
137	2.04E03	0	0	0	0
138	8.26E03	6.25E01	0	0	0

24

(1) Units for activities are in curies

TABLE 15.6-27

LOSS-OF-COOLANT ACCIDENT
ACTIVITY AIRBORNE INSIDE PRIMARY CONTAINMENT*
REALISTIC ANALYSIS 23

Isotope	0.0 Hr	2.0 Hr	8.0 Hr	24.0 Hr	96.0 Hr	720.0 Hr	
I-131	2.73×10^{1}	5.85	5.71	2.93	9.49×10^{-1}	9.63×10^{-2}	
132	1.38×10^{1}	1.64	2.75×10^{-1}	1.27×10^{-3}	0	0	
133	9.35×10^{1}	1.89	1.55×10^{1}	4.93	1.91×10^{-1}	0	
134	5.41	$2.40 \times 10^{-}$[illegible]	2.06×10^{-3}	3.51×10^{-9}	0	0	
135	2.68×10^{1}	4.70	2.53	2.64×10^{-1}	6.66×10^{-5}	0	
Kr-83m	3.50×10^{2}	1.69×10^{2}	1.91×10^{1}	5.66×10^{-2}	0	0	24
85m	1.94×10^{2}	1.41×10^{2}	5.42×10^{1}	4.24	4.45×10^{-5}	0	
85	3.20×10^{2}	3.20×10^{2}	3.20×10^{2}	3.20×10^{2}	3.17×10^{2}	2.96×10^{2}	
87	1.72×10^{2}	5.91×10^{1}	2.42	4.79×10^{-4}	0	0	
88	3.41×10^{2}	2.07×10^{1}	4.61×10^{1}	8.40×10^{-1}	1.25×10^{-8}	0	24
89	5.50×10^{1}	0	0	0	0	0	
Xe-131m	2.4×10^{1}	2.39×10^{1}	2.35×10^{1}	2.26×10^{1}	1.89×10^{1}	3.95	
133m	1.04×10^{2}	1.01×10^{2}	9.39×10^{1}	7.67×10^{1}	3.09×10^{1}	1.14×10^{-2}	
133	5.00×10^{3}	4.95×10^{3}	4.78×10^{3}	4.38×10^{3}	2.93×10^{3}	9.03×10^{1}	
135	1.05×10^{3}	9.03×10^{2}	5.72×10^{2}	1.70×10^{2}	7.09×10^{-1}	0	
135m	1.6×10^{1}	7.77×10^{-2}	8.88×10^{-5}	0	0	0	
137	6.98×10^{1}	3.84×10^{-8}	0	0	0	0	
138	1.94×10^{2}	1.45	6.15×10^{-7}	0	0	0	

* Units for activities are in curies

TABLE 15.6-28

LOSS-OF-COOLANT ACCIDENT
ACTIVITY AIRBORNE IN ENCLOSURE BUILDING*
REALISTIC ANALYSIS

Isotope	2.0 Hr	8.0 Hr	24.0 Hr	96.0 Hr	720.0 Hr
I-131	3.36×10^{-3}	9.78×10^{-3}	1.27×10^{-2}	9.21×10^{-4}	8.61×10^{-5}
132	9.41×10^{-4}	4.70×10^{-4}	5.54×10^{-6}	0	0
133	1.08×10^{-2}	2.64×10^{-2}	2.14×10^{-2}	1.85×10^{-4}	1.72×10^{-13}
134	1.38×10^{-4}	3.53×10^{-6}	1.52×10^{-11}	0	0
135	2.70×10^{-3}	4.33×10^{-3}	1.14×10^{-3}	6.46×10^{-8}	0
Kr-83m	2.98×10^{-2}	1.06×10^{-2}	5.61×10^{-5}	0	0
85m	2.48×10^{-2}	3.02×10^{-2}	4.20×10^{-3}	5.11×10^{-8}	0
85	5.63×10^{-2}	1.78×10^{-1}	3.16×10^{-1}	3.63×10^{-1}	3.38×10^{-1}
87	1.04×10^{-2}	1.35×10^{-3}	4.74×10^{-7}	0	0
88	3.64×10^{-2}	2.57×10^{-2}	8.32×10^{-4}	1.43×10^{-11}	0
89	4.29×10^{-14}	0	0	0	0
Xe-131m	4.30×10^{-3}	1.31×10^{-2}	2.24×10^{-2}	2.16×10^{-2}	4.52×10^{-3}
133m	1.78×10^{-2}	5.24×10^{-2}	7.60×10^{-2}	3.53×10^{-2}	1.31×10^{-5}
133	8.70×10^{-1}	2.67	4.33	3.35	1.03×10^{-1}
135	1.59×10^{-1}	3.19×10^{-1}	1.68×10^{-1}	8.52×10^{-4}	0
135m	1.37×10^{-5}	4.95×10^{-12}	0	0	0
137	6.76×10^{-12}	0	0	0	0
138	2.56×10^{-4}	3.43×10^{-10}	0	0	0

* Units for activities are in curies

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TABLE 15.6-29

LOSS-OF-COOLANT ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT*
REALISTIC ANALYSIS

Isotope	0-2 Hr	2-8 Hr	8-24 Hr	24-96 Hr	96-720 Hrs
I-131	1.41×10^{-4}	1.64×10^{-4}	4.62×10^{-4}	6.81×10^{-4}	1.37×10^{-3}
132	6.10×10^{-5}	2.10×10^{-5}	4.93×10^{-6}	2.94×10^{-8}	0
133	4.74×10^{-4}	4.83×10^{-4}	1.01×10^{-3}	6.01×10^{-4}	3.42×10^{-5}
134	1.94×10^{-5}	1.33×10^{-6}	1.38×10^{-8}	0	0
135	1.31×10^{-4}	9.82×10^{-5}	1.06×10^{-4}	1.47×10^{-5}	3.85×10^{-9}
Kr-83m	7.54×10^{-3}	1.44×10^{-2}	3.35×10^{-3}	1.46×10^{-5}	0
85m	5.19×10^{-3}	2.06×10^{-2}	2.19×10^{-2}	2.54×10^{-3}	2.89×10^{-8}
85	1.03×10^{-2}	7.75×10^{-2}	3.93×10^{-1}	2.33	1.99×10^{1}
87	3.11×10^{-3}	3.50×10^{-3}	2.79×10^{-4}	8.32×10^{-8}	0
88	8.29×10^{-3}	2.35×10^{-2}	1.21×10^{-2}	3.16×10^{-4}	0
89	3.83×10^{-5}	0	0	0	0
Xe-131m	7.68×10^{-4}	5.74×10^{-3}	2.83×10^{-2}	1.51×10^{-1}	6.19×10^{-1}
133m	3.29×10^{-3}	2.35×10^{-2}	1.04×10^{-1}	3.67×10^{-1}	2.53×10^{-1}
133	1.59×10^{-1}	1.18	5.61	$2.63 \times 10^{+1}$	5.29×10^{1}
135	3.09×10^{-2}	1.70×10^{-1}	3.88×10^{-1}	2.17×10^{-1}	[illegible].59 $\times 10^{-4}$
135m	6.78×10^{-5}	7.47×10^{-7}	0	0	0
137	6.05×10^{-5}	0	0	0	0
138	9.08×10^{-4}	1.54×10^{-5}	0	0	0

* Units for activities are in Curies

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TABLE 15.6-30

SGTS FILTER IODINE LOADINGS, CURIES
(CONSERVATIVE CASE)

23

	Time Period, hr					
Isotope	2	8	24	96	720	
I-131	3.51E 03	1.29E 04	3.19E 04	3.72E 04	1.15E 04	
132	2.96E 03	1.87E 03	4.19E 01	3.18E-08	0	
133	7.41E 03	2.28E 04	3.52E 04	4.90E 03	1.5E-05	24
134	1.90E 03	6.17E 01	5.06E-04	0	0	
135	5.85E 03	1.19E 04	5.97E 03	5.43	0	

b. 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is available for release. This assumption is based on fission product release data from defective fuel experiments (Ref 1).

c. Because of the negligible particulate activity available for release from the fuel, none of the solid fission products will be released. 23

d. It is assumed that 98 fuel rods fail. This is conservative because fewer than 98 rods would be damaged.

Using the above assumptions, the fission product activity released from each damaged fuel rod to the water in the fuel pool as a result of the dropped fuel assembly is:

Isotope	(Ci/Rod)	
I-131	2.50E1	
I-132	3.44E0	
I-133	6.70E0	
I-134	---	
I-135	8.05E-1	
Kr-83m	3.23E-3	
Kr-85	6.90E0	
Kr-85m	5.67E-2	
Kr-87	---	
Kr-88	1.94E-2	
Xe-131m	5.24E-1	24
Xe-133	1.02E+2	
Xe-133m	1.90E0	
Xe-135	3.00E+1	
Xe-135m	2.77E-2	

<u>15.7.4.5.2.2.3 Refueling building activity</u>. See 251 NSSS GESSAR Section 15.1.41.5.1.2.3 and PSAR Table 15.7-13.

<u>15.7.4.5.2.2.4 Activity released to environment</u>. The assumptions used in calculating the fission products released to the environment are:

a. High radiation levels in the Refueling Building ventilation exhaust will isolate the Refueling Building.

b. Because the refueling accident does not result in the release of any liquid or vapor to the building, the normal building environmental

conditions before the accident will also exist after the accident, except for the addition of the released fission products.

c. The ventilation rate is 1 air change per day with the effluent released via the Standby Gas Treatment System with a filter efficiency of 99 percent for all forms of iodine. Based on the preceding conditions, the fission product release is presented in Table 15.7-14.

15.7.4.5.2.2.5 Radiological effects. Based on the release in Table 15.7-14. The off-Site radiological exposure is presented in Table 15.7-15. It should be emphasized that the radiological exposures presented in Table 15.7-15 are based upon the assumption that the worst meteorological conditions exist for the duration under consideration and that the wind blows in one direction during the entire release period. These doses are well below the guidelines of 10 CFR 100. 23

15.7.4.5.2.3 Consideration of uncertainties. See 251 NSSS GESSAR Section 15.1.41.5.1.3.

15.7.4.5.3 Comparison of Realistic and Conservative Parameters

See 251 NSSS GESSAR Section 15.1.41.5.3 and PSAR Table 15.7-20.

15.7.4.6 References

See 251 NSSS GESSAR Section 15.1.41.6.

TABLE 15.7-17

FUEL HANDLING ACCIDENT

(Conservative Analysis)

Isotope	Fission Product Released to Environs, Ci, 0-2 hour
I-131	2.58E+0
I-132	4.00E-3
I-133	2.92E+0
I-134	-
I-135	5.00E-1
Kr-83M	6.50E-1
Kr-85M	2.82E+2
Kr-85	8.53E+2
Kr-87	4.80E-2
Kr-88	8.33E+1
Xe-131M	2.00E+2
Xe-133M	1.20E+3
Xe-133	5.62E+4
Xe-135	9.93E+3

23

TABLE 15.7-18

FUEL HANDLING ACCIDENT
OFF-SITE RADIOLOGICAL EXPOSURES

(Conservative Analysis) 23

Distance (meters)	Whole-Body (rem)	Thyroid (rem)
3058 (EAB)	5.16×10^{-3}	7.7×10^{-1}
6437 (LPZ)	1.24×10^{-3}	1.86×10^{-1}

24

$$\beta D = (X/Q) \sum_{i=1}^{N} DCF_i \; Q_i \qquad (15A\text{-}3)$$

where

βD = beta dose from a semi-infinite cloud (rem)
DCF_i = dose conversion factor given in Table 15A-5 for isotope i

15A.2 CONTROL ROOM DOSES 23

15A.2.1 CONTROL ROOM X/Q MODEL

In calculating X/Q for the control room, an exception was taken to the method recommended in SRP 6.4 and the 1974 Murphy-Campe (M-C) paper (Ref 10). A review of the recent literature on building wake X/Qs, models, wind tunnel tests and field measurements (References 11, 12, 13, 14, 15 and 16) concluded that the X/Q methodology recommended in the M-C paper was overly conservative and inappropriate for the S/HNP plant design principally due to the location of the intakes close to the building walls and to the complex configuration of the structures involved.

The method used in calculating the present control room X/Q is based upon the methodology presented in Meteorology and Atomic Energy (Ref 1) as developed by Halitsky. The equation presented to Halitsky is given as follows: 24

$$X/Q = K/A\bar{u}$$

where

A = cross sectional area, m^2
$\bar{u}$ = wind speed, m/sec
K = isopleth (concentration coefficient - dimensionless)

It is found in many cases that the Halitsky equation still provides a reasonable estimate of X/Q. Several correction factors can be applied to this equation to account for situation and plant specific features. Thus, the following modified Halitsky formulation for the X/Q is used.

$$X/Q = \frac{K}{A\bar{u}} \times f_1 \times f_2 \times f_3 \times f_4 \times f_5 \; (sec/m^3) \qquad (15A\text{-}4)$$

The choice of K factors and the suggested modifying factors, f_1, f_2 etc. are discussed below.

K factors: The choice of an appropriate K factor from the wind tunnel test data is critical for the X/Q estimate to be valid. Halitsky in Reference 1 has several sets of K isopleths for round topped containments (PWRs) and block buildings (BWRs). Multiple building complexes must be simulated by sing? equivalent structures. The effluent velocity to wind speed ratio of approximately 1 is valid for most power plant systems. Various angles of wind incidence are shown to account for vortexing which could result in worse conditions than a wind normal to the building face. K factors should be estimated for various combinations of wind incidence angle and the appropriate effective building cross-sectional area causing the wake (not just the containment area) to determine the peak value as was done by Walker (Ref 12).

K values and the appropriate building cross-sectional areas were evaluated for different wind incidence angles. The most conservative case for S/HNP was found to be for the wind blowing from the SGTS exhaust towards the control room air intake. A K-value of 2 was obtained (Ref 1) for the control room air intake 1/5 of the way up from grade elevation. The Auxiliary Building, Control Building and Diesel Generator Buildings would create the largest wake effects. The Turbine Building, which is higher than the Auxiliary Building, could create channeling effects, thus concentrating the effluents at the intake location. To account for the channeling effects, the contribution to the wake effect from the Enclosure Building is halved. 23

The cross sectional area orthogonal to the wind direction is 2461 m^2.

Wind speed, $\bar{u}$: Halitsky's K values are based on wind speeds measured at the top of the containment or building. Therefore, the M-C 5 percentile wind speed at a 10 meter height should be adjusted to the actual speed at the top of containment or release point. The 5 percentile wind speed is adjusted using the formulation presented by Wilson (reference 16) as follows:

TABLE 15A-1

ACCIDENT ATMOSPHERIC DILUTION FACTORS -- X/Q

Time Period (hrs)	Wind Speed Factor	Wind Direction Factor	X/Q (sec/m^3)
0-8	1	1	7.75×10^{-5}
8-24	0.67	0.67	4.57×10^{-5}
24-96	0.5	0.75	2.91×10^{-5}
96-720	0.33	0.5	1.28×10^{-5}

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TABLE 15A-2

PHYSICAL DATA FOR ISOTOPES

Isotope	Decay[(1)] Constant (Hr^{-1})	Gamma[(1)] Energy (Mev/Disint.)	Beta[(1)] Energy (Mev/Disint.)	
I-131	3.5856×10^{-3}	0.374	0.207	23
I-132	2.97×10^{-1}	2.30	0.420	
I-133	3.31×10^{-2}	0.513	0.432	
I-134	7.92×10^{-1}	2.53	0.603	
I-135	1.03×10^{-1}	1.89	0.302	
Xe-133	5.47×10^{-3}	0.0486	0.146	
Xe-133m	1.25×10^{-2}	0.0291	0.135	
Xe-135	7.60×10^{-2}	0.249	0.313	
Xe-135m	2.72×10^{0}	0.429	1.930	24
Xe-138	2.37×10^{0}	1.390	1.930	
Kr-85	7.95×10^{-6}	0.002	0.223	
Kr-85m	1.49×10^{-1}	0.160	0.233	
Kr-87	5.33×10^{-1}	1.48	0.804	
Kr-88	2.50×10^{-1}	1.743	0.297	

Notes: (1) Reference 6

SKAGIT/HANFORD NUCLEAR PROJECT

QUESTIONS AND RESPONSES

The questions and responses provided in this section are NRC Requests for Additional Information and Applicant Responses which result from the NRC review of S/HNP PSAR Amendment 23 and subsequent amendments. These amendments reflect the move of the Project from the Skagit Site in northwestern Washington State to a Site on the Hanford Reservation in south-central Washington State. The information presented in this section reflects S/HNP design at the Hanford Site. Sections of the PSAR amended in response to the questions in this section are designated by a dark vertical bar in the right margin of the page, identified with the NRC Request for Additional Information number, with the prefix "H", e.g., H220.1.

QUESTION 220.01 (3.5.3)

With respect to PSAR Section 3.5.4, missile barriers are designed as described in BC-TOP-9A, which shows the ductility ratio 20 for flexture in steel barriers. The staff had approved BC-TOP-9 in 1974 taking exception for ductility ratio over 10. The current staff position is included in SRP Section 3.5.3 Appendix "A" (Attachment 1). Confirm that you have not used ductility ratio more than 10 or justify deviation from above mentioned SRP criteria.

Also, for columns with slenderness ratio more than 20, the current SRP requires ductility ratio less than or equal to 1.0 while BC-TOP-9A mentions 1.3. Confirm that you will use all the ductility ratios per SRP criteria or justify the deviation, and revise the PSAR sections accordingly.

RESPONSE:

The analysis of structures, shields and barriers to determine the effects of missile impacts will be done in accordance with BC-TOP-9A. Justification to the satisfaction of the NRC staff will be provided when ductility ratios will be greater than 10.

See revised PSAR Section 3.5.4

QUESTION 220.02 (3.7.1)

Figure 3.7-3 and 3.7-4 show that more than 8 points of the time history response spectra fall below the design response spectra for 1%, 2%, 5%, and 7% damping. SRP Section 3.7.1 subsection II.1.b requires that no more than 5 points should fall below the design response spectra. Justify your deviation from the SRP position.

RESPONSE:

Figures 3.7-3 and 3.7-4 are taken from Bechtel Topical Report BC-TOP-4A, Revision 3, Chapter 2, wherein the synthetic time-histories are described in more detail. These time-histories have been extensively reviewed and approved by the NRC staff as part of the Topical Report. Please note that the time-histories remain unchanged from the original Skagit PSAR.

QUESTION 220.03 (3.7.1)

With respect to PSAR Section 3.7.1.2 you did not demonstrate that the frequency interval for calculation of response spectra are small enough that further reduction does not result in more than 10% change in computed spectra. This is one of the requirements of SRP Section 3.7.1. Confirm that you will meet the SRP criteria and revise the PSAR accordingly.

RESPONSE:

Please refer to the response to Question 220.02. The frequencies at which the response spectra have been computed are given in BC-TOP-4A, Revision 3, Chapter 2.

These frequencies are so chosen that most of the increments do not exceed 5 percent within the range of 1 to 15 cps, which is the usual range of power plant structure frequencies.

QUESTION 220.04 (3.7.2)

With respect to PSAR Section 3.7.2.11, you have mentioned that torsional effects will be considered in Category I structures using 3-dimensional models, or 2-dimensional models with static factors to account for torsional accelerations. Confirm that the dynamic analysis method you use, will also have the consideration of rocking, and translational responses of the structures and their foundations.

RESPONSE:

See revised Section 3.7.2.1.

A meeting will be scheduled with the NRC following receipt of a Construction Permit to specifically review in detail, and to receive NRC approval of the computer code to be utilized for this analysis.

QUESTION 220.05 (3.7.2)

In seismic analysis methods for Category I structures you have not mentioned the effect of differential support movement. Confirm that your analysis methods will have the consideration of maximum relative displacements among supports of Category I structures, systems, and components. Also discuss the extent to which your analysis method conforms with the applicable criteria of SRP Section 3.7.2.

RESPONSE:

See revised Section 3.7.2.1. The proposed analysis methods described in the S/HNP PSAR and BC-TOP-4A, Rev. 3, are in agreement with Section 3.7.2.1 of the SRP (NUREG 0800).

QUESTION 220.06 (3.7.2)

In PSAR Section 3.7.2.4, it is stated that for soil-structure interaction elastic half space method will be used and a "simplified finite element analysis" will be performed as a confirmation. The words "simplified finite element analysis" are unclear to the staff. Describe in detail the procedures, assumptions, and boundary conditions (specially affected by the site move amendment) that you intend to use in your finite element analysis. The current position of the staff about soil-structure interaction is outlined in REV 1 of SRP Section 3.7.2 issued in July 1981. Confirm that you will meet the SRP criteria, and revise the PSAR accordingly.

RESPONSE:

Section 3.7.2.4 has been revised to comply with the referenced SRP criteria.

QUESTION 220.07 (3.7.2)

In seismic analysis methods for Category I structures you did not address the accidental torsion. Confirm that, in your seismic analysis, you will account for accidental torsion by taking an additional eccentricity of ±5% of the maximum building dimension over and above the actual geometric eccentricity. This is the requirement of SRP Section 3.7.2 subsection II.11.

RESPONSE:

See revised PSAR Section 3.7.2.11.

QUESTION 220.08 (3.7.4)

PSAR Section 3.7.4 shows that triaxial response spectrum recorders and triaxial time history monitors are provided at specific locations, meeting the requirement of SRP Section 3.7.4. However, for control room operator notification you have mentioned that, whenever an acceleration time history is being or has been recorded, a visual annunciation will be made in the control room.

SRP Section 3.7.4 requires that triaxial time history monitor will readout peak acceleration in the control room and the response spectrum recorder will readout values at discrete frequencies. Just the visual annunciation is not sufficient. Confirm that you will meet the SRP criteria or justify the deviation.

RESPONSE:

Section 3.7.4.3 has been revised to comply with the referenced SRP criteria.

QUESTION 220.09 (3.7.4)

PSAR does not mention the inservice surveillance program for seismic instruments. SRP Section 3.7.4, subsection II.5 requires that each seismic instrument be demonstrated operable by the performance of test operations at specified intervals. Revise the PSAR to meet the SRP criteria or justify the deviation from same.

RESPONSE:

See new Section 3.7.4.5.

QUESTION 220.10 (3.8.1)

In your Amendment 23 of PSAR Section 3.8.1, concrete containment was not included. Please confirm that the site move does not affect your commitment in the PSAR including the design and analysis procedures of concrete containment, loads and loading combinations, structural acceptance procedures and the applicable codes, standards and specifications to comply with ASME Section III, Division 2 code and the related Regulatory Guides. Also, please indicate that in the containment loads and loading combinations the LOCA/SRV related hydrodynamic loads in suppression pools manifested as jet loads and/or pressure loads will be considered.

RESPONSE:

The site move does not affect the commitment in the PSAR. As stated in Section 3.8.1.3, design of the containment will comply with the provisions of the ASME Section III, Division 2, Code and related Regulatory Guides. Refer to the response to Question 220.12 for applicable Regulatory Guides.

LOCA and SRV hydrodynamic loads are considered: pool swell loads are included in the term "Pa", as defined in Section 3.8.1.4; SRV loads are included in the term "L", as defined also in Section 3.8.1.4.

QUESTION 220.11 (3.8.1)

Provide an ultimate capacity analysis of the containment responding to the internal pressure build-up due to accidents. The guideline and the staff position on this subject is enclosed (Attachment 2).

RESPONSE:

An ultimate capacity analysis of the Containment will be provided in the FSAR.

QUESTION 220.12 (3.8.1, 3.8.4)

Update and expand the list of Regulatory Guides that would be applied to all Category I structures (e.g., R.G. 1.94, 1.115, 1.117, 1.122, 1.136, 1.142, . . .). Address any exceptions and deviations from these Regulatory Guides and provide comments and explanations.

RESPONSE:

Sections 3.8.1.3.3 and 3.8.3.2.3 have been updated to show S/HNP compliance with Regulatory Guide 1.94.

In addition, the design and analysis of S/HNP Category I structures will comply, as clarified, with the Regulatory Guides listed below.

Regulatory Guide 1.115, Protection Against Low-Trajectory Turbine Missiles, Rev. 1:

Project Position: The analysis presented in Sections 3.5 and 10.2 comply with the guidance contained in Regulatory Guide 1.115.

Regulatory Guide 1.117, Tornado Design Classification, Rev. 1:

Project Position: Protection against the effects of a Design Basis Tornado will be provided in accordance with Regulatory Guide 1.117, Rev. 1, with the following exception:

Paragraph 3 of Section B of the Regulatory Guide states that "the physical separation of redundant or alternative structures or components required for the safe shutdown of the plant is generally not considered acceptable by itself for protecting against tornado effects, including tornado-generated missiles."

For the Standby Service Water cooling towers (UHS), separation of redundant trains is relied upon for protection from large missiles travelling in the vertical direction. Large missiles are defined as missiles C, D, F and G below, from PSAR Table 3.5-2. Separation is considered adequate protection because large missiles are expected to be much less numerous than small missiles.

Barriers will be provided over the top of the cooling towers to protect the mechanical draft fans from small missiles A, B and E below, from PSAR Table 3.5-2, travelling vertically.

QUESTION 220.12 (Cont'd)

TORNADO GENERATED MISSILES

	Missile	Weight W (lb)
A.	Wood Plank 4 in. x 12 in. x 12 ft.	200
B.	Steel Pipe 3 in. dia. x 10 ft. (Sch. 40)	75.8
C.	Steel Pipe 6 in. dia. x 15 ft. (Sch. 40)	285
D.	Steel Pipe 12 in. dia. x 15 ft. (Sch. 40)	744
E.	Steel rod 1 in. dia. x 3 ft.	8
F.	Utility pole 13.5 in. dia. x 35 ft.	1490
G.	Automobile	4000

(Reference Table 3.5-2)

Regulatory Guide 1.122, Development of Floor Response Spectra for Seismic Design of Floor-Supported Equipment or Components, Rev. 1:

Project Position: The requirements of Regulatory Guide 1.122 will be met. Section 3.7.2 describes a method of development of floor response spectra for S/HNP which complies with Regulatory Guide 1.122, Rev. 1.

Regulatory Guide 1.136, Materials, Construction, and Testing of Concrete Containments, Rev. 2:

Project Position: Materials, Construction, and Testing of concrete containments comply with Regulatory Guide 1.136 except as follows:

(1) Regulatory Position C.8: The Summer 1980 Addenda to the ASME Code, CC-4333, 5.2, permits both production and sister splice samples. The

QUESTION 220.12 (Cont'd)

requirement for all samples to be production splices does not ensure added integrity.

(2) Regulatory Position C.9: Staggering of splices is a design objective and will be specified wherever possible. At certain locations, such as a penetration and blackouts, staggering is impractical.

(3) Regulatory Position C.13: Design specifications will state that remedial measures shall be undertaken or a retest shall be conducted.

Regulatory Guide 1.142, Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments):

Project Position: Comply

The PSAR will be revised to reflect these commitments within six months of issuance of a Construction Permit.

QUESTION 220.13 (3.8.1)

In PSAR you have mentioned the use of 10 CFR 50 Appendix A, GDC 2, 4, 16, 50, etc. Why has GDC 1 been omitted? Include GDC 1 also and address its effect on your Quality Assurance Program.

RESPONSE:

The Quality Assurance Program described in Chapter 17 of this PSAR complies with GDC 1.

See revised Sections 3.8.1.3.3 and 3.8.3.2.3.

QUESTION 220.14 (3.8.1)

PSAR Table 3.8-1 shows load combinations and load factors. For service load at construction stage you have not included the wind loads. The current SRP refers to the Table CC-3230-1 of ASME Section III Division 2 Code for load combinations. The construction load in this table includes wind. Confirm that you will meet the SRP criteria and include the wind loads at construction stage and revise the PSAR section accordingly.

RESPONSE:

See revised Table 3.8-1.

QUESTION 220.15 (3.8.3)

In your Amendment 23 of PSAR Section 3.8.3, concrete and steel internal structures of steel or concrete containment, were not included. Please confirm that the site move would not affect your commitment in the PSAR including the design of containment internal structures, loads and loading combinations, structural acceptance procedures and the applicable codes, standards and specifications to comply with ASME Section III, Division 1 and 2, ACI 349, AISC and related Regulatory Guides. For structures or structural components subject to hydrodynamic loads resulting from LOCA and/or SRV actuation, the consideration of such loads should be included. Please refer to the Appendix to NUREG-800 SRP Section 3.8.1.

RESPONSE:

The site move does not affect the commitment in the PSAR to design the containment internal structures, loads and loading combinations, structural acceptance procedures and the applicable codes, standards and specifications to comply with ASME Section III, Division 1 and 2, Codes; ACI 349 as augmented by Regulatory Guide 1.142, AISC and related Regulatory Guides as described in the PSAR. Also, refer to the response to Question 220.12.

The internal structures will be designed for the loads and loading combinations given in Section 3.8.6. LOCA and SRV hydrodynamic loads are considered and will be defined using the methodology as described in GESSAR II to the satisfaction of the NRC staff: pool swell loads are included in the term "Pa", as defined in Section 3.8.6.1.4; SRV loads are included in the term "L", as defined in Section 3.8.6.1.1.

Load combinations and load factors are considered. See revised Sections 3.8.6.2.2 and 3.8.6.3.2.

QUESTION 220.16 (3.8.4)

With respect to PSAR Section 3.8.4.1.2, you only discussed wall separation between the Auxiliary Building and containment structure. There are other Category I structures adjacent to containment. Please indicate whether there are any wall separations between them. If any, please specify how much is the wall separation. Please provide this information and the technical bases to verify that adequate separation is provided.

RESPONSE:

Wall separation between the containment and other Category I structures will be in accordance with Section 4.1 of BC-TOP-4A. See revised Sections 3.8.4.1.2 and 3.8.4.1.3 (the Auxiliary and Fuel Buildings are the only Category I structures adjacent to containment).

QUESTION 220.17 (3.8.4)

With respect to PSAR Section 3.8.4, you didn't indicate whether masonry construction was utilized or not. Enclosed is a copy of design criteria for safety-related masonry wall evaluation (Attachment 3). Identify any difference in requirements of materials, testing analysis, design and construction between SKAGIT/HANFORD design, and staff position. Provide justification for these differences or indicate your compliance with them.

If no Category I masonry wall construction is planned, please so indicate and neglect the technical portion of this question.

RESPONSE:

In a generic letter dated April 21, 1980 to all Construction Permit and Operating License Applicants regarding an information request on Category I masonry walls employed by plants under CP and OL review the NRC requested information as to whether concrete masonry walls will be used in any of the Category I structures of the plant. There will be no concrete masonry walls used in any of the S/HNP Category I structures. See revised Section 3.8.3.6 (which is referenced by Section 3.8.4.6).

QUESTION 220.18 (3.8.4)

With respect to PSAR Section 3.8.4.1.3, Fuel Building, discuss, in detail, the design of spent fuel pool racks. Enclosed is a copy of staff position on the "minimum requirements for design of spent fuel pool racks" (Attachment 4). Modify your analysis and design, if necessary, to agree with this position.

RESPONSE:

The design of the spent fuel racks is described in PSAR Section 9.1.2 and has been accepted by the NRC in SER Section 9.1.2. The spent fuel rack and fuel pool designs have not changed.

QUESTION 220.19 (3.8.3, 3.8.4, 3.8.5)

With respect to PSAR Sections 3.8.3.2, 3.8.4.2, and 3.8.5.2 applicable codes standards and specifications, it is the staff's position that ACI 349-76 code should be used in conjunction with Regulatory Guide 1.142. Identify any deviations of Category I structural design from the requirements of the code and the Regulatory Guide and justify your deviations.

RESPONSE:

ACI 349-76 code will be used in conjunction with Regulatory Guide 1.142 for the design of safety related concrete structures.

See revised Sections 3.8.3.2.2, 3.8.3.4, 3.8.6.1.5, 3.8.6.2.1, 3.8.6.2.2, 3.8.6.3.1, and 3.8.6.3.2.

QUESTION 230.1

Provide a complete interpretation of all reflection and refraction lines within 5 miles of the site. Present the final processed section of the reflection data in a depth format next to your preferred interpretation. Also present the travel time curves of the refraction data from which you estimated velocity and thickness of the different layers. Compare the interpretation obtained from reflection and refraction lines covering the same area, and explain any discrepancy between the two sets if one exists and why.

RESPONSE:

Available Data

Seismic reflection data collected by Rockwell during fiscal years 1979 and 1980 along lines shown in Fig 2K-8 of the PSAR were reviewed by consultants to NESCO (Appendix 2L, Section 4.4.1.1, p 2L-10, S/HNP Amendment 23). Seismic refraction data were collected along lines shown in Figure 2K-3 of the PSAR by consultants to NESCO and were interpreted independently by them and correlated with data from drillholes and magnetic and gravity surveys.

Deficiencies of Processed Reflection Data

There are several reasons why the processed seismic reflection data have not been interpreted in detail nor reprocessed into a depth format, but remain in the time domain:

1. The processed seismic reflection data in their present form are of poor or marginal quality and only gross configurations of the basalt are suggested.

2. The processed data are unreliable because they show apparent reflections which are discontinuous and are not correlatable with the top of basalt or other geologic horizons recognized from drill holes and seismic refraction surveys.

3. Downhole velocity surveys and seismic refraction profiling indicate that lateral velocity changes in the post-basalt sediments are highly variable and not adequately defined to permit reliable reprocessing. Therefore, reprocessing of the reflection data to incorporate all of the velocity variations would be a costly and time consuming operation with limited chance of significantly improving the processed

QUESTION 230.1 (Cont'd)

records. Reducing the stack from 1200% to 300% or 600% would reduce possible interference between wide-angle reflections and refraction arrivals, but would probably not measurably improve the quality of the reflections due to the reduction in stack.

Furthermore, as described below, the reflection data do not correlate with refraction and test boring data indicating that the reflection data as processed by Rockwell are unreliable for determining top of basalt or structure within the basalt.

A. Reflection Line 2 Near Southeast Anticline

Reflection Line 2 crosses the Southeast Anticline in the vicinity of Station 300 to 340 and then extends southward towards Horn Rapids as shown in Figure 230.1-1 (provided separately by letter WGC-0306 dated March 5, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical). The location of seismic refraction data and test borings in the vicinity of Line 2 are also shown in Figure 230.1-1. The seismic refraction data and the test borings in the vicinity of the northern portion of reflection line 2 are shown as an overlay on the reflection profile (Figure 230.1-2, provided separately by letter WGC-0306 dated March 5, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical). Although this figure illustrates a general similarity in the configuration of the basalt surface as defined by reflection, refraction and test boring data, thereflecting horizons are not continuous and cannot be directly correlated with geologic horizons recognized by the other methods. Test Boring 4 and refraction line 1 generally confirm a rise in the basalt surface near Station 470, although the amplitude of the rise predicted by the reflection data cannot be determined because of a lack of continuity of the reflecting horizons.

B. Reflection Line 2, Horn Rapids Area

The southern end of reflection line 2 is located along Horn Rapids Road (see Figure 230.1-1) where seismic refraction investigations were conducted during studies for the WNP 2 Site. The seismic reflection data (Figure 230.1-3, provided separately by letter WGC-0308 dated March 8, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical) show flat-lying reflectors between Stations 1240 and 1340,

QUESTION 230.1 (Cont'd)

with reflectors rising to the south between Stations 1480 to 1540. Between Stations 1340 and 1480 the reflectors lack clarity and continuity, whereas the subsurface profile as determined by seismic refraction is continuous as shown on Figure 230.1-4 (provided separately by letter WGC-0308 dated March 8, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical) and an overlay to reflection line 2 (Figure 230.1-3). The refraction profile shows a zone of complex lateral and vertical velocity changes between Stations 1340 and 1480 where there are discontinuous reflecting horizons. These velocity changes render a proper interpretation of the processed reflection data difficult to impossible.

C. Reflection Line 3 with Refraction Line 8

A portion of Reflection Line 3 was specifically investigated by seismic refraction (Line 8), gravity and test borings in order to address the interpretation of faulting by Seismograph Service Company as described in Myers and Price (1979). As shown on Figure 2K-38 (S/HNP PSAR, Amendment 23, 1981) the seismic refraction, gravity and test boring data are consistent and define an anticlinal ridge. The seismic refraction profile and borings drilled along Line 8 are shown as an overlay on the reflection profile (Figure 230.1-5, provided separately by letter WGC-0306 dated March 5, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical). Also shown is the total bouguer gravity profile, which is consistent with the refraction profile. The refraction profile shown is descriptive of the lateral velocity changes, the greater of which occur where the reflections lack continuity. In summary, the refraction, test boring and gravity data provide a continuous profile of the basalt surface and define an anticlinal ridge; no faulting of the basalt surface is indicated and, as can be seen, the reflection profile does not accurately portray this ridge.

D. Reflection Line 10 with Refraction Line M

Reflection Line 10 (Stations 120 to 430) and Refraction Line M (Stations 206 to 360) are approximately the same location (Figure 230.1-1). Borings S-8, S-9, S-10 and S-15 were located on or immediately adjacent to the seismic lines. A gravity profile was also obtained along refraction line M. The seismic

QUESTION 230.1 (Cont'd)

refraction, gravity and test boring data are in good agreement (Figure 2L-18, S/HNP PSAR, Amendment 23, 1981) and define a smoothly varying basalt surface as shown on the overlay to the reflection data (Figure 230.1-6, provided separately by letter WGC-0306 dated March 5, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical). Downhole velocity values below the near surface low velocity zone range from 7040 ft/sec in Hole S-8 to 7650 ft/sec in Hole S-10. Although there is no direct match between reflectors and the bedrock surface as shown on the refraction profile, the discontinuous reflectors at a time (two way) of 0.25 to 0.30 seconds show a configuration similar to that indicated by the other techniques. The relatively strong reflecting horizon at 0.2 sec (two-way time) between Stations 230 and 360 is apparently a horizon within the Ringold Formation: however, there is no readily identifiable layer described on the boring logs that correlate with this reflector. The lack of continuous reflecting horizons, and the smoothly varying basalt profile defined by seismic refraction, gravity and test boring data, would appear to preclude any structure in the vicinity of Stations 410 to 415 as proposed by Rockwell in ST-14 (see Response 231.3.b.3).

E. Reflection Line 11 with Borings and Gravity Data

Although seismic refraction data have not been acquired along Reflection Line 11 (Figure 230.1-1), gravity and test boring data define a very gently sloping basalt surface. The gravity profiles constructed from the contour map and the test borings are shown on the overlay to the reflection data (Figure 230.1-7, provided separately by letter WGC-0306 dated March 5, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical). The depth to rock as identified in the test borings has been converted to time using an average downhole velocity value of 7,100 ft/sec as measured in nearby boreholes. There is no continuity of the reflectors nor is there any correlation with recognizable geologic horizons between the reflection data and the gravity and test boring data. The rise in the reflector at about 0.25 sec (two-way time) in the vicinity of Stations 440 to 450 amounts to approximately 125 feet (assuming an average downhole velocity of 7,100 ft/sec), which would correspond to nearly a one milligal positive gravity anomaly. Such a gravity anomaly is not present.

QUESTION 230.1 (Cont'd)

Seismic Refraction Travel Time Curves

Seismic refraction data have been acquired over a wide area of the Hanford Site as part of S/HNP investigations (see Figure 2K-3) and previously on the WNP 1, 2 and 4 sites as documented in Appendix 2L and 2M of the WNP 1 and 4 PSAR (Amendment 9, 1974). Transmitted by Letter WGC-0309 dated March 8, 1982 to M. W. Mallory, NRC, from Edward N. Levine of Weston Geophysical are one copy each of the interpreted and uninterpreted refraction time-distance plots for those lines, all or part of which are within five miles of the S/HNP Site as shown on Figure 230.1-1. The refraction data have been obtained and processed in accordance with the procedures described in Section 4.1.1, page 2K-9 and 2K-10 of S/HNP, Amendment 23, and in Appendix 2L of Amendment 9 to the WNP 1 and 4 PSAR.

QUESTION 230.2

The shear velocity profile under WPS 1, 2, 4 (Figure 361.1, Supply System) shows a velocity contrast at a depth of 100 ft. while the shear velocity profile for your site in the same region (Figure 2.5.10 PSAR) does not show the contrast. Explain why there is a difference.

RESPONSE:

Seismic refraction measurements throughout the Hanford Reservation have detected a persistent refracting horizon with a compressional velocity generally varying from 7,500 ft/sec to 12,000 ft/sec between elevations 350 to 400 feet above sea level. This type of lateral variation in velocity is common for cemented layers and is probably associated with variations in porosity which existed prior to cementation. Therefore, within a layer characterized by a velocity range the lower or higher ends of the range are correlatable to fine or coarse material.

Compressional velocities of 10,000 to 12,000 ft/sec have been detected in the referenced refraction horizon in the WNP 1, 2 and 4 plant site area and northward towards Line 1. In the area north and west of Line 1, including the S/HNP Site area, the velocity of this refracting horizon is generally in the range of 7,500 to 9,000 ft/sec. The depth to this horizon is approximately 100 feet (elevation 350 feet) in the WNP 1, 2 and 4 plant site area approximately 140 feet (elevation 390 feet) in the S/HNP Site area. The difference in elevation of 40 feet between the S/HNP site and the WPPSS site, a distance of 5 miles, is not significant.

The shear wave velocities of this refracting horizon have been measured by the crosshole technique at the S/HNP Site (Figure 2L-3), two other locations to the north and northeast of S/HNP (Figure 2L-2), and in the WNP 1, 2 and 4 plant site area. The shear wave velocity of this high compressional velocity refracting horizon is in the range of 2,500 to 3,000 ft/sec in the S/HNP area, 3,000 to 4,000 ft/sec north and northeast of S/HNP, and 4,000 to 5,000 ft/sec in the WNP 1, 2, and 4 site area. These differences, like the differences in compressional velocity, are probably due to normal lateral variations in cementation.

QUESTION 230.3

The bedrock contour map based on refraction data, southeast anticline (Figure 2K-61, PSAR) does not show any faulting yet the structural contour map (Figure 2R-7, PSAR) based on boring indicates the existence of a fault. Explain the reasons for the differences.

RESPONSE:

The structural contour map shown on PSAR Figure 2R-7 is based on an interpretation of all of the geologic data reported in PSAR Appendix 2R (Stratigraphic Investigation of the Skagit/Hanford Nuclear Project). These data indicate the presence of a fault at depth on line 4A. In particular, core 125 intersected an anomalous thickness of Elephant Mountain basalt and several shear zones within this flow. In Appendix 2R, the fault is conservatively inferred to extend northwest to line 3 and southeast to line 4C.

The seismic refraction profiling across the Southeast Anticline did not identify any abrupt offset suggestive of faulting in the 16,000 ft/sec refracting horizon identified by test borings as Elephant Mountain basalt. The maximum slope on the high-velocity basalt surface in the vicinity of the Southeast Anticline is less than 10 degrees (see Figure 2K-61 PSAR). The smooth and gentle slopes on the high-velocity basalt surface are not indicative of faulting in the basalt; accordingly, a fault is not shown on the Bedrock Contour Map Based on Refraction (Figure 2K-61).

QUESTION 231.1

Among the numerous remote sensing surveys conducted in the Columbia Basin area are reports by Slemmons and O'Malley (1979, Revised 1980) and by Shannon and Wilson, Inc. (1980). Many of the lineaments identified in the reports are within the immediate vicinity of the Skagit/-Hanford site. For the most part, the above reports simply identify lineaments and do not discuss their possible origin. With respect to these reports perform an analysis, including field-truth as required, of the origin and possible safety significance of the lineaments within an approximate 5 mile radius of the Skagit/Hanford site. Include in your analysis (1) lineaments interpreted from the side looking airborne radar which was flown for the Seattle District of the Corps of Engineers in the Fall of 1979 and (2) additional lineaments not included in the above two reports, but orally described to the NRC staff at Richland, Washington on February 2, 1982.

RESPONSE:

A photogeologic analysis has been made of the area within a 10-mile radius of the Skagit/Hanford Site. This analysis examined all lineaments within a 5-mile radius of the Site and all other linear features within a 10-mile radius that trend toward the Site. This analysis utilized the following imagery:

Imagery	Scale	Source
U-2, Color IR	1:135,000	NASA, 1978
U-2, Black & White	1:135,000	NASA, 1978
PSLG, Color	1:48,000	BPA, 1973
Radar Mosaic (SLAR-east & west looking)	1:250,000	Dept. of Army Seattle District, Corps of Engineers (Fall, 1979)

The analysis identified all linear features previously identified by Glass (1977) and Slemmons and O'Malley (1979 - 1980) within 10 miles of the Site, and several additional lineaments which had not been noted by previous workers (Figure 231.1-1).

QUESTION 231.1 (Cont'd)

The origin of these features was determined by grouping lineaments having similar characteristics on the imagery and then field checking lineaments representative of each group. Four groups of features were identified and attributed to either previously mapped geologic structures or to one of the three following origins:

- o Pleistocene glaciofluvial flood channeling or other glaciofluvial processes;
- o Eolian processes (e.g., they are the edges of dunes or the margins of stabilized dune fields); or
- o Wildfire burns.

Lineaments which could not, by inspection of the imagery, be assigned either to previously mapped geologic structure or to one of the three origins noted above (Figure 231.1-2) were specifically field checked. The characteristics and origins of these lineaments are discussed below.

Feature 1 is a 2 1/4 mile-long, generally north-northeast trending curvilinear tonal alignment crossing stable and active dune areas approximately 1 1/2 miles southeast of the 200E area. Field inspection indicated that the northern end of the feature is the escarpment of a Pleistocene flood channel and is thus of glaciofluvial origin.

Feature 2 is a north-northeast trending, linear tonal-alignment 3 1/4 miles northwest of the Wye Barricade. The tonal alignment represents an irregular vegetation change that crosses active and stabilized dune fields. At the northern end it coincides with a 15- to 20-ft high, west-facing embankment of dune sand. The feature is of eolian origin.

Feature 3 is a tonal contrast in an active dune field which trends northwest for approximately 2 miles south of the 200E area. The area to the northeast of the lineament is darker in tone than to the southwest. This feature is caused by a contrast in vegetation density in more and less active dune areas.

Feature 4 is a portion of the Cold Creek lineament defined in Shannon and Wilson (1980). The Washington Public Power Supply System has determined that this

QUESTION 231.1 (Cont'd)

lineament is a fortuitous alignment of various non-tectonic features with no common causal relationship.

Feature 5 is a 4 1/2 mile-long lineament that strikes approximately N35°E from 3/4 mile southeast of the 200E area to just south of Gable Mountain. This feature was detected only on east-looking SLAR imagery (scale 1:250,000) where it is expressed as a narrow, subtle alignment of weaker signal return which contrasts with areas of stronger signal return on either side. The alignment does not appear on any conventional aerial photography and does not correspond to any topographic, geomorphic, vegetational, or tonal alignments. Consequently, the feature is considered to be an artifact of SLAR instrumentation and processing.

Although examination of the imagery and field checking of specific features indicated no evidence of structural control for any of the photolineaments within a five-mile radius of the site (Figure 231.1-3), a comparison has also been made with bedrock contour maps (Figure 2R-7, Appendix 2R, S/HNP Amendment 23, 1981; Figure 8-3, Rockwell, 1981; and Figure 111-4a, Rockwell, 1979). With only three exceptions, there is no correspondence between the photolineaments and bedrock topography. These three exceptions occur in the vicinity of Werner solutions shown on Rockwell Figures B-11 and B-12 (1981), and are described below in terms of the location of these solutions.

In the area of Werner solution N-245, there are two photolineaments which also generally coincide with the trend of the Benson Ranch syncline as shown on Figure 8-3 (Rockwell, 1981). These photolineaments have been produced by a transverse dune and an alignment of vegetation within a trough in the dune. The bedrock map (Figure 8-3, Rockwell, 1981) shows only a very gently sloping bedrock surface in this area. Based on this information and the obvious surficial origin of the photolineaments, the coincidence of these lineaments with the bedrock trend and Werner solution is considered to be fortuitous. There is no causal relationship between the two.

One photolineament has been identifed in the vicinity of Werner solution D-29 (Figure 231.1-2, Feature 3), which corresponds in trend and location with the northern flank of the Yakima Ridge extension shown on Figure 8-3 of Rockwell (1981). The Werner solution has probably been

QUESTION 231.1 (Cont'd)

produced by the presence of the buried Yakima Ridge anticline. The photolineament in this vicinity has been produced by tonal contrasts across the boundary between active and less active parts of a dune field. This clearly surficial origin for the lineament and lack of evidence to suggest a relationship with Yakima Ridge extension indicate that the photolineament has no structural significance.

Some correspondence is shown between bedrock structure and a photolineament which crosses Werner solution N-63. Although solution N-63 appears to be subparallel to the bedrock contours on Figure 8-3 (Rockwell, 1981) and the gravity contours on Figure 2K-13 (Appendix 2K, S/HNP Amendment 23, 1981), both of these figures suggest that the surface of the basalt slopes gently to the southeast. The photolineament in this area is produced by the edge of a longitudinal dune and, therefore, despite its fortuitous correspondence with the Werner solution N-63 bedrock slope, has no structural significance.

A comparison was also made between the photolineaments identifiied within a five-mile radius of the site and Werner solutions shown on Figures B-11 and B-12 (Rockwell, 1981). This comparison suggests that in some areas, in addition to those discussed above, there is a coincidence, either in location or trend, between the lineaments and the Werner solutions. However, there is no manifestation of any structural origin for either the Werner solutions or the photolineaments in the bedrock surface in these areas. These photolineaments are discussed below by reference to the Werner solutions with which they correspond in location or trend.

The photolineaments in the area of Werner solution D-240 and SD-10 are produced by the edges of longitudinal dunes. The lineaments trend at an angle greater than 45 degrees to structural contours on the bedrock surface and in the overlying Ringold Formation (Figure 2R-7 through 2R-12, Appendix 2R, S/HNP Amendment 23, 1981). These lineaments are also oblique to the gravity contours on Figure 2K-13 (Appendix 2K, S/HNP Amendment 23, 1981).

A photolineament produced by tonal alignments and glaciofluvial erosion is located to the northwest of Werner solution D-22. Although only the southeastern end of the photolineament and northwestern end of the Werner solution come close to coinciding, the two features correspond to some degree in trend. The photolineament is

QUESTION 231.1 (Cont'd)

perpendicular to the trend of the bedrock contours as shown on Figure 8-3 (Rockwell, 1981) and the gravity contours on Figure 2K-13 (Appendix 2K, S/HNP PSAR, Amendment 23, 1981).

The trends of some photolineaments also parallel Werner solution N-53. These lineaments are produced by the edges of longitudinal dunes. The lineaments are oblique to the trend of the generally smooth bedrock surface shown on Figure 2R-7 (Appendix 2R, S/HNP PSAR, Amendment 23, 1981) and the gravity contours shown on Figure 2K-13 (Appendix 2K, S/HNP PSAR, Amendment 23, 1981). Structure contours on Figures 2R-8 through 2R-12 (Appendix 2R, S/HNP PSAR, Amendment 23, 1981) show that the Ringold Formation is flat-lying in the vicinity of these lineaments.

Photolineaments in the vicinity of Werner solution D-228 are also produced by the edges of longitudinal dunes. Figure III-4a (Rockwell, 1979) indicates that these lineaments are in an area of flat-lying basalt.

Photolineaments in the vicinity of Werner solution N-246A do not coincide with the Werner solution but approximately parallel its trend. These lineaments are produced by the edges of longitudinal dunes. Figure III-4a (Rockwell, 1979) shows that these lineaments are perpendicular to bedrock contours, which show only a gentle slope on the bedrock surface.

In summary, although some photolineaments may correspond in trend or location with bedrock features and/or Werner solutions, based on the obvious surficial origins of the features and the lack of evidence for structural control, where such a correspondences exists they are purely fortuitous and have no structural significance.

REFERENCES CITED

Glass, Charles E., 1977, Remote sensing analysis of the Columbia Plateau: Amendment 23, Appendix 2R-K; Preliminary Safety Analysis Report, WPPSS Nuclear Project No. 1.

Glass, Charles E., and Slemmons, David B., 1977, Imagery and topographic interpretation of geologic structures in central Washington: Amendment 23, Subappendix 2R-F, Preliminary Safety Analysis Report, WPPSS Nuclear Project No. 1.

QUESTION 231.1 (Cont'd)

Shannon and Wilson, Inc., 1980, Geologic evaluation of selected faults and lineaments, Pasco and Walla Walla Basins - Washington, by Farooqui, S. M. and Thomas, R. E.: Prepared for Washington Public Power Supply System.

Slemmons, D. B. and O'Malley, P., 1979 (Revised 1980), Fault and earthquake hazard evaluation of five U.S. Corps of Engineers dams in southeastern Washington: Prepared for Seattle District U.S. Corps of Engineers.

QUESTION 231.2

A northwest-trending, northeast-dipping feature has been interpreted on bedrock surface (see PSAR Appendix 2R, Figure 2R-7) within approximately one mile southwest of the Skagit/Hanford Site. Weston Geophysical Corporation views this feature (see Appendix 2R, page 2R-32) as the steeper-dipping southwestern limb of the northwest trending Cold Creek syncline. Based upon a staff review of the seismic coverage (see PSAR Figures 2L-14 and 2K-3), Weston's refraction data does not extend sufficient distance to the southwest of the Skagit/Hanford site to define the geometry of the syncline. On this basis, present all geologic and geophysical data, including appropriate diagrams and stratigraphic/structural cross sections supporting your interpretation, of the synclinal origin and extent of the feature. Include with your response a discussion of why the feature cannot be of fault origin. The cross-sections should extend to the southwest beyond the Cold Creek lineament area.

RESPONSE:

The geologic and geophysical data that bear on the origin and the extent of the northwest-trending slope southwest of the Site consist of drilling data (Rockwell, 1979; Rockwell, 1981; S/HNP PSAR Appendix 2R), downhole geophysical surveys (S/HNP PSAR Appendix 2R), and gravity and magnetic surveys (S/HNP PSAR Appendices 2K and 2L). The most complete data coverage of the S/HNP study across this feature consists of a gravity survey (Figure 231.2-1). Contour maps of the gravity data in this area are included as Figure 2K-13 in Appendix 2K and Figures 2L-8, 2L-9, 2L-10, and 2L-11 in Appendix 2L of S/HNP PSAR, Amendment 23. Gravity profiles for Lines 1, 4B and 4D in the S/HNP Site area (Figures 231.2-1, 2, and 3 provided separately by letter WGC-0306 dated March 5, 1982 to M. W. Mallory, NRC, from John P. Imse of Weston Geophysical) show the location of the Cold Creek syncline and the southwestern limit of the Cold Creek syncline, defined by a small gravity high (probably a small anticline) on Line 4D at Station -355 and on Line 4B at Station -330. This gravity high becomes broader and less pronounced on Line 1. Gravity gradients indicate that dips on the southwestern limb of the Cold Creek syncline are approximately 5 degrees.

Although data collected for the S/HNP Project cover only a part of the feature, they are compatible with all of the data available from the region. These data collectively indicate that the northwesterly-trending bedrock slope is the southwestern limb of the Cold Creek

QUESTION 231.2 (Cont'd)

syncline. In particular, the bedrock maps and structure cross sections included in reports by Rockwell (1979, Plates III-4 and III-5; 1981, Figures 8-3 and 8-7) show that the feature southwest of the Site conforms to regional trends in both strike and dip.

The gentle nature of the bedrock slope strongly suggests that the slope was produced by folding. No evidence suggests that the slope southwest of the Site may have been produced by faulting.

REFERENCES CITED

Rockwell Hanford Operations, 1979, Geologic studies of the Columbia Plateau - A status report, RHO-BWI-ST-4, Myers, C. W., Price S. M., and Others: Prepared for the U.S. Department of Energy.

Rockwell Hanford Operations, 1981, Subsurface geology of the Cold Creek syncline, RHO-BWI-ST-14, Myers, C. W., and Price, S. M., Editors: Prepared for the U.S. Department of Energy.

QUESTION 231.3

Provide an assessment of the information included in a recently released report by the Department of Energy[1] on the geologic and geophysical interpretations within 5 miles of the Skagit/Hanford Site. The report suggests that several structures and geophysical anomalies within the Pasco Basin may be faults. For any structures and anomalies within 5 miles of the site determine if these features are faults. For any feature which is determined to be a fault, determine whether it is capable. Provide all bases for determination including geophysics (magnetics, gravity, reflection and refraction seismology) and geology (maps, cross-sections, borehole logs, borehole correlations, remote sensing, etc.).

RESPONSE:

The applicant has not identified any information in the referenced report which might affect the geologic and geophysical interpretations within 5 miles of the S/HNP Site other than (1) those items specifically identified by the NRC staff in their informal comments providing additional detail to this question, received by the applicant on February 16, 1982; and (2) the dike-like geometric solutions determined by Werner deconvolution of aeromagnetic data presented in Figures B-11 and B-12 of the referenced report. The staff's informal comments have been addressed as Questions 231.3a and 231.3b. Both dike-like and fault-like Werner solutions from Figures B-11 and B-12 are addressed in the response to Question 231.3.a.

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[1]Rockwell Hanford Operations, 1981, Subsurface geology of the Cold Creek syncline, RHO-BWI-ST-14, Myers, C. W., and Price, S. M., Editors: Prepared for the U.S. Department of Energy.

QUESTION 231.3.a

A number of possible normal faults, based upon an interpretation of aeromagnetic data, have been identified in Figure B-11 and B-12 of a report recently released by the Department of Energy (DOE).[2] The location of one of the northwest-trending features (faults) is nearly coincident, both in trend and location, with the top of bedrock feature located about one mile southwest of the Skagit/Hanford Site. This bedrock feature is discussed in NRC Question 231.2 Provide an analysis of each of the aeromagnetically-interpreted faults(?) identified in the above report within at least a five mile radius of the Skagit/Hanford site. The analysis is to include a discussion, and the underlying bases, of the confirmation or rebuttal of the existence of each of the faults(?). Particular emphasis should be directed toward discussing the coincidence of the relationship of the top-of-bedrock feature (NRC question 231.2) and the aeromagnetic fault (?) one mile southwest of the Skagit/Hanford site as shown on DOE Figures B-11 and B-12. If your analysis confirms the existence of any of the aeromagnetically-identified faults(?) demonstrate, by appropriate text and figures, the non-capability of the feature. Your response should include relevant portions of the appropriate figures from the DOE report.

RESPONSE:

The "possible normal faults" identified on Figure B-11 and B-12 of the referenced report are, in fact, not identified as possible faults but rather as geometric solutions determined by Werner deconvolution of aeromagnetic data. These solutions are presented graphically in three forms: (1) fault-like solutions, (2) dike-like solutions, and (3) structural disturbances. It is noted on page B-22 of the referenced report that:

"It should be emphasized that a fault-like solution does not necessarily mean that an actual fault is present. Rather, the fault-like solution indicates

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[2]Holmes, G. E. and Mitchell, T. H. Seismic-reflection and aeromagnetic surveys in the Cold Creek syncline area, Appendix B, in Myers, C. W., and Price, S. M, Editors, Subsurface geology of the Cold Creek syncline, RHO-BWI-ST-14, prepared by Rockwell International for the United States Department of Energy, 1981.

QUESTION 231.3.a (Cont'd)

> that a horizontal magnetic source terminates at a particular location. In the Cold Creek syncline, horizontal termination of magnetic sources (lava flows) can be caused by flow pinchout, possible abrupt changes in the magnetic properties of a flow, steep anticlinal/synclinal flanks, as well as fault displacement."
>
> "Solutions of thin magnetic layers dipping vertically ($\pm 45^{\circ}$) are mapped as "dikes" on the interpretive maps (Fig. B-11 and B-12), although they may not represent true geologic dikes. A similar Werner deconvolution solution is obtained over anomalies caused by recognized anticlinal or synclinal structures. Structural disturbances, as noted on the interpretive maps, are generally solutions that do not meet the criteria for either dike-like or fault-like features. For example, structural disturbances may be mathematically resolved as a dike-like structure on one survey level, and possibly as two fault-like solutions from another flight level which represents the two edges of a dike-like body."

All of the Werner solutions within a 5-mile radius of the S/HNP Site are described along with other available geophysical data in Tables 231.3-1 and 231.3-2. All Werner solutions within 5 miles of the S/HNP Site which can be evaluated using other geophysical and geological data have been interpreted as either the axes or gently-dipping flanks of folds or as not being expressed in the basalt surface and, therefore, not due to post-Elephant Mountain Basalt deformation. Consequently these Werner solutions do not affect the applicants geologic interpretation of the S/HNP Site vicinity. The following discussion concerns the interpretation of Werner solutions which were not used in the structural synthesis on Figure 8-8 of Rockwell (1981). Those Werner solutions are discussed in the response to Question 231.3.b.1.

Werner Solution N-53

This east-west trending Werner solution is not interpreted by Rockwell (1981). The 760-meter solution for N-53 trends normal to the Total Bouguer Gravity contours shown on Figure 2K-13 (Appendix 2K, S/HNP Amendment 23, 1981) and where N-53 intersects Line 1 (Station 80+00) the seismic refraction, gravity and land magnetic data are interpreted as indicative of a smooth bedrock surface (Figure 2K-54, Appendix 2K, S/HNP Amendment 23, 1981).

QUESTION 231.3.a (Cont'd)

The 1220 meter solution for N-53 also trends normal to the gravity contours and projects into seismic refraction Line 1 (Station 65+00). The gravity and seismic refraction data indicate a smooth bedrock surface (Figure 2K-54, Appendix 2K, S/HNP Amendment 23, 1981). Structural contours derived from drilling and downhole geophysical data (Figures 2R-7 through 2R-12, Appendix 2R, S/HNP PSAR, Amendment 23, 1981) also indicate a smooth bedrock surface and flat-lying sedimentary units within the overlying Ringold formation.

The geologic source for Werner Solution N-53 is, therefore, not manifested in the bedrock surface or the overlying Ringold formation.

Werner Solutions D-240 and SD-10

These Werner solutions are located approximately 4.5-5 miles northeast of the S/HNP Site and are not interpreted by Rockwell (1981). At Station 120+00 on seismic refraction Line 5, a slight rise is indicated south of where the projected sources for D-240 and SD-10 would be located (Figure 2L-A4, Appendix 2L, S/HNP Amendment 23, 1981). Both solutions trend perpendicular to the gravity contours (Figure 2K-13, Appendix 2K, S/HNP Amendment 23, 1981) and the bedrock contours derived from seismic refraction data (Figure 2K-14, Appendix 2L, S/HNP Amendment 23, 1981). Structural contours derived from drilling and downhole geophysical data (Figures 2R-7 through 2R-12, Appendix 2R, S/HNP Amendment 23, 1981) also indicate a smooth bedrock surface and flat-lying sedimentary units within overlying Ringold formation.

The geologic source for Solutions D-240 and SD-10 is not manifested in the bedrock surface or the overlying Ringold formation.

Werner Solution N-54

Werner Solution N-54 is not interpreted by Rockwell (1981). The source for Solution N-54 is located in an area of gravity and land magnetic coverage and trends perpendicular to the gravity contours shown on Figure 2K-13 (Appendix 2K, S/HNP Amendment 23, 1981). The top-of-basalt contour map (Figure 8-3) of Rockwell (1981) indicates that this solution is perpendicular to the projection of the axis of a small anticline which plunges to the southeast. The potential field data are interpreted as indicative of a smoothly varying, low

QUESTION 231.3.a (Cont'd)

gradient bedrock surface. The individual geophysical profiles for Lines M, K, and 5 are compatible with a 3° southeastward-dipping bedrock surface.

This low gradient bedrock slope is a possible source for Solution N-54.

Werner Solutions N-246 and N-74

Although Solution N-246 (760 meter level) and Solution N-74 (1220 meter level) have different orientations, their locations are the same and probably have a common geologic source. Rockwell (1981, P. B-48) interprets the source of N-246 as the northern limb of the Yakima Ridge extension. Rockwell does not interpret Solution N-74. The land magnetic and gravity data in the S/HNP Site vicinity (Figures 2L-13 and 2L-8, Appendix 2L, S/HNP Amendment 23, 1981) are indicative of a slight rise in the basalt just northeast of the source location for the northern one-half of N-74. There is no indication of a source for the east-trending N-246. To the southwest of Solutions N-246 and N-74, the gravity and magnetic data (Appendix 2L, S/HNP Amendment 23, 1981) and geologic data (Figure 2R-7, Appendix 2R, S/HNP Amendment 23, 1981) are indicative of a gently-sloping (approx. 5° NE) bedrock surface which has been interpreted as the southwestern flank of the Cold Creek syncline.

The geologic source for Werner Solutions N-246 and N-74 may be the 5° northeastward-dipping flank of the Cold Creek syncline or a very small bedrock high within the syncline.

Werner Solution N-88

This Werner solution is not interpreted by Rockwell (1981). The northeastern end of the source for Solution N-88 would be located at Station -380+00 on Line 4B. The land magnetic data for this line are indicative of a smooth bedrock surface at this location (Figure 2L-A9, Appendix 2L, S/HNP Amendment 23, 1981). Solution N-88 trends perpendicular to the gravity contours (Figure 2L-8, Appendix 2L, S/HNP Amendment 23, 1981) and the top-of-basalt contours of Rockwell (Figure 8-3, 1981).

The geologic source for Werner Solution N-88 is not manifested in the bedrock surface.

QUESTION 231.3.a (Cont'd)

Werner Solution D-23

Rockwell Seismic Reflection Line 2 and Seismic Refraction WNP Line 1 intersect the source for Werner Solution D-23. These seismic data have been specifically discussed in the response to Question 230.1. As noted in that response, the refraction data show a flat lying basalt surface, no structural feature is indicated. Plate III-4a (Rockwell, 1979) also shows that this is an area of generally flat-lying basalt.

The geologic source for Werner Solution D-23 is not manifested in the bedrock surface.

Werner Solution D-227 and D-228

Werner Solutions D-227 and D-228 are designated on Rockwell (1981) Figure B-11 as occurring at the ground surface. They are coincident with the WNP-1 and WNP-2 nuclear plants. The physical sources of Werner Solutions D-227 and D-228 are interpreted to be the large masses of magnetic material contained in the WNP-1 and WNP-2 plants.

Werner Solution N-63

Although solution N-63 appears to be subparallel to the bedrock contours on Figure 8-3 (Rockwell, 1981), and the gravity contours on Figure 2K-13 (Appendix 2K, S/HNP Amendment 23, 1981), both of these figures suggest that the surface of the basalt slopes gently to the souheast. This gentle bedrock slope is a possible source for Solution N-63.

Werner Solution N-245

Solution N-245 is located in an area where Plate III-4a (Rockwell, 1979) indicates a very nearly flat bedrock surface. There is no indication that the geologic source of Werner Solution N-245 is manifested in the bedrock surface.

Werner Solution N-246-A

This solution is in an area where Plate III-4a (Rockwell, 1979) indicates a nearly flat basalt surface (slope less than 3 degrees). The contours on this surface are shown to trend perpendicular to the trend of the Werner solution. There is no indication that the geologic source of

QUESTION 231.3.a (Cont'd)

Werner Solution N-246-A is manifested in the bedrock surface.

Werner Solution N-248

Plate III-4a (Rockwell, 1979) shows that this solution is located in an area of flat-lying basalt. There is no indication that the geological source of Werner solution N-248 is manifested in the bedrock surface.

Werner Solution D-37

This solution is shown on Plate III-4a (Rockwell, 1979) to be perpendicular to the bedrock contours. This solution is located in an area where the bedrock surface is generally smooth, slopes less that 3 degrees. There is no indication that the geologic source for Werner solution D-37 is manifested in the bedrock surface.

QUESTION 231.3.b

Provide an assessment of the impact (including positive and, if applicable, negative aspects) of the 1981 Department of Energy report[3] as it relates to the geologic interpretation of the Skagit/Hanford Site vicinity (within 5 miles of the site). Several specific aspects of the report to be incorporated within the assessment include:

1. Inferred bedrock structure (Figure 8-8, Chapter 8)

   This figure shows several northwesterly-trending structures of undefined nature, one passing directly beneath the site.

2. Brecciated Core (Boreholes, DC 8 and DC 12, Chapter 6)

   Breccia, both tectonic and flow-top, has been noted at depths below 1,700 ft.

3. Seismic reflection anomalies (Figure B-5, Appendix 8)

   Bedrock feature anomalies have been identified on seismic reflection survey lines 10 and 11.

RESPONSE TO QUESTION 231.3.b.1:

The "structures" shown on Figure 8-8 of the referenced report have been tentatively interpreted by Rockwell (1981) not as structures but as the boundaries of "relatively intact volumes of bedrock." These boundaries are proposed on the bases of previously interpreted bedrock structure and geometric solutions determined by Werner deconvolutions of aeromagnetic data. Those Werner solutions within 5 miles of the S/HNP Site which have been used to define these boundaries have been interpreted by Rockwell and the applicant as the axes or gently-dipping flanks of low amplitude folds. Consequently, the "structures" shown on Figure 8-8 of the referenced report do not affect the applicants' previous geologic interpretation of the S/HNP Site vicinity. Those "structures" tentatively interpreted by Rockwell

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[3]Rockwell Hanford Operations, 1981, Subsurface geology of the Cold Creek syncline, RHO-BWI-ST-14, Myers, C. W., and Price, S. M., Editors: Prepared for the U.S. Department of Energy.

QUESTION 231.3.b.1 (Cont'd)

within 5 miles of the S/HNP Site and designated as features "A" through "E" on Figure 231.3-1 (attached) and are discussed individually below.

Feature "A"

The basis for this feature is Werner Solution D-22 which is shown on both Figures B-11 and B-12 (Rockwell, 1981). Rockwell interpreted this anomaly as a deeply buried, asymmetric anticline (Rockwell, 1981, p. B-53). The gravity and seismic refraction data acquired for the S/HNP Site along Line 4A-1 (Figure 2L-16, Appendix 2L, S/HNP Amendment 23, 1981) and the geologic infromation derived from drilling and downhole geophysics (Figure 2R-7, Appendix 2R, S/HNP Amendment 23, 1981) are consistent with the existence of a low-amplitude bedrock rise with gently sloping flanks. The central and northern portions of D-22 are also coincident with a northwest-trending, low-amplitude gravity high in the vicinity of Station -40+00 on Line 1 and Station -210+00 on Line 2 as illustrated on Figure 2L-8 (Appendix 2L, S/HNP Amendment 23, 1981).

Feature "A" is interpreted to be coincident with the crest of a low amplitude anticlinal high.

Feature "B"

Feature "B" (Figure 231.3-1) is based upon an alignment of Werner Solutions D-28, N-73 and N-269 (Rockwell, 1981, Figure 8-3 and p. B-53). There is no additional geophysical data over Solutions D-28 and N-269, but Rockwell interpreted this to be a small fold (Rockwell, 1981, Figure 8-3 and p. B-53). Solution N-73 is only present on the 1220 meter level and is partially coincident with Solutions D-28 and N-269, therefore, Solution N-73 probably has a common source with Solutions D-28 and N-269. The Total Bouguer Gravity data for this portion of the Hanford Site (Figure 2L-8, Appendix 2L, S/HNP Amendment 23, 1981) only include the southeasternmost end of Solution N-73. This Werner solution is coincident with a very slight gradient on the southwest flank of a small gravity high. The ground magnetic data for Line C are compatible with a small bedrock rise.

The ground geophysical data are consistent with the Rockwell interpretation that Solutions D-28, N-73 and N-269 are due to a small, low amplitude fold.

QUESTION 231.3.b.1 (Cont'd)

Feature "C"

The feature labeled "C" on Figure 231.3-1 is coincident with the May Junction Linear as first described by Myers and Price (1979). This bedrock structure was studied extensively during recent field studies for the S/HNP Project. Gravity, land magnetic and seismic refraction data (Appendix 2K, S/HNP Amendment 23, 1981) were acquired to investigate the May Junction linear.

The results of the geophysical studies characterize Feature C as an eastward dipping monoclinal fold.

Feature "D"

The bases for this feature are the interpreted continuation of the May Junction structure and Werner Solution N-242 (Rockwell, 1981, p. 8-23). Rockwell postulates a small fold as the source for Solution N-242 based upon a nearby reflection line (Rockwell, 1981, p. B-53). A low amplitude gravity high is indicated one-half mile southeast of the location for Solution N-242 (Figure 2L-8, Appendix 2L, S/HNP Amendment 23, 1981). The land magnetic data collected between Stations -60+00 and -90+00 on Line X and Stations -10+00 and -50+00 on Line Y (Figure 2L-A12, Appendix 2L, S/HNP Amendment 23, 1981) are compatible with the interpretation of a low amplitude, anticlinal fold.

The source for Solution N-242 is interpreted to be a northeast-trending, low amplitude fold, and appears to be a saddle in the Cold Creek syncline.

Feature "E"

Feature "E" (Figure 231.3-1) coincides with the interpreted extension of the Yakima Ridge structure (Rockwell, 1981, p 8-22) and Werner Solutions D-29 and N-75 (Rockwell, 1981, p B-48). Solution D-29 is concident with a gravity high located approximately 2 miles west-southwest of the S/HNP Site (Figure 2L-8, Appendix 2L, S/HNP Amendment 23, 1981). The Total Bouguer Gravity contours delineate a northwest-trending high in this area (Figure 2L-8, Appendix 2L, S/HNP Amendment 23, 1981) and Solution N-75 is coincident with the projected crest of this gravity high.

The only other geophysical data in the vicinity of Solution N-75 is Rockwell Reflection Line 1 (Table

QUESTION 231.3.b.1 (Cont'd)

231.3-2) (see response to Question 230.1). The reflection horizons on Line 1 lack continuity in the area of Solution N-75.

The gravity data support the interpretation that Solutions D-29 and N-75 are the result of a subsurface anticlinal ridge that appears to be asymmetric to the northeast. The steeper northeast limb is the southwest flank of the Cold Creek syncline.

RESPONSE TO QUESTION 231.3.b.2:

Brecciated core noted in boreholes DC-8 and DC-12 are considered by Rockwell (1981) to be minor features unrelated to large displacements. The following excerpt from Chapter 6 of the Rockwell report (p. 6-3) discusses these breccia zones:

> In general, tectonic breccias are infrequent in the thousands of feet (meters) of core drilled in the Cold Creek syncline area and elsewhere in the Pasco Basin. The breccia zones that were identified (Table 6-1) are generally intact and <4 in. (<10cm) in thickness, although some are slightly thicker (Figure 6-2). They appear in all deep boreholes within the Hanford Site and are principally in the Grande Ronde and Wanapum Basalts. As noted in the next chapter, such small tectonic breccia zones and their associated fractures are viewed as typical strain features of folded basalt and should be expected within the limbs of any of the Yakima folds, including the Cold Creek syncline. None of the tectonic breccias examined are judged as being associated with large displacements. This conclusion is based on comparisons with surface exposures of similar breccias and the lack of anomalously thick basalt flows that would be expected if the section were repeated.

These brecciated cores are, therefore, considered to have no implications regarding the interpreted geologic structure in the S/HNP Site vicinity.

REFERENCED ITEMS

Rockwell Hanford Operations, 1981, Subsurface Geology of the Cold Creek Syncline, RHO-BWI-ST-14, Myers, C. W., and Price, S. M., Editors: Prepared for the U.S. Department of Energy.

RESPONSE TO 231.3.b.3:

As discussed in the response to Question 230.1, Rockwell seismic reflection records collected during fiscal years 1979 and 1980 are of marginal quality for any structural interpretation. Other data in the vicinity of anomalies identified on seismic reflection Lines 10 and 11 would appear to preclude the existence of structures such as those proposed by Rockwell at Line 10 station 410 to 415 and Line 11 Stations 440 to 450.

TABLE 231.3-1

WERNER SOLUTIONS FROM 760m LEVEL WITHIN 5 MILES OF S/HNP SITE

Werner Solution Number	Only At This Elev.	Common to Both Elev.	Common[1] Location Different Solution	Ground Geophysical Coverage[2] WGC Refrac.	RHO Reflec.	WGC Gravity	RHO Gravity	WGC Ground Magnetics
N-269			X					
D-28		X						
N-242	X				(9)	(X,Y)		(X,Y)
D-29		X				(Z,1)		(Z,1)
D-22		X		(4A-1)	(11)	(2,1)		(2,1,4A-1 4A-2)
N-53		X		(1)	(10,11)	(1,M,L,K)		(1,M,L,K)
D-240			X	(5)	(2)	(5,U,V)		(5,U,V)
N-54		X				(M,K)		(M,K)
N-246			X		(9)	(4B,X)		(4B,X)
D-227	X							
N-75		X			(1)			
N-248	X							
N-246A[3]	X							
N-245	X				(1)			
D-37		X						

[1]This indicates that what is modeled as a dike on one figure may be modeled as a fault on the other, or it may be a case of a common source type with a different orientation.

[2]The geophysical profiles which intersect the inferred structures are indicated.

[3]The designation "A" has been added because there are two solutions on Figure B-11 labeled N-246.

TABLE 231.3-2

WERNER SOLUTIONS FROM 1220m LEVEL WITHIN 5 MILES OF S/HNP SITE

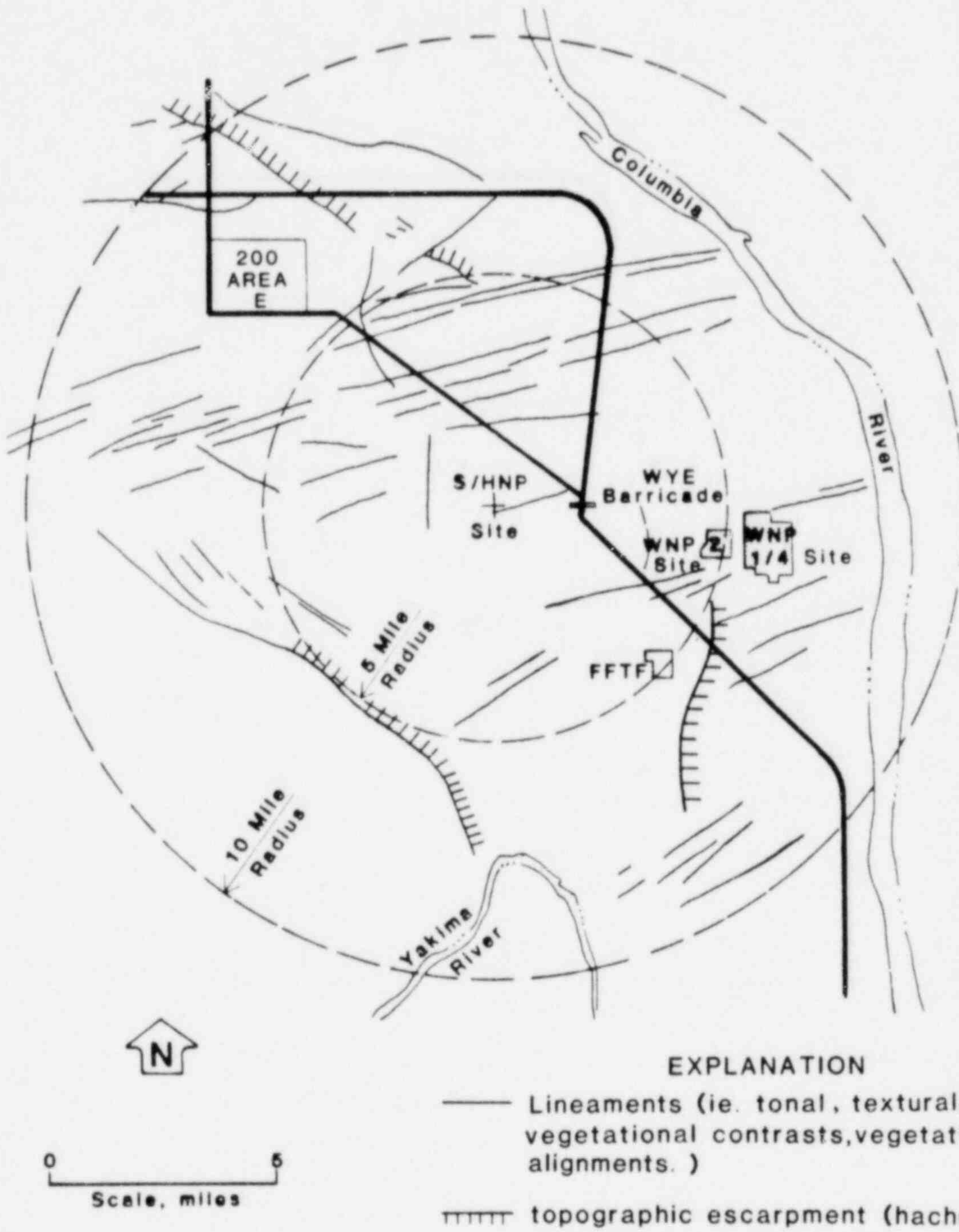
Werner Solution Number	Only At This Elev.	Common to Both Elev.	Common[1] Location Different Solution	Ground Geophysical Coverage[2] WGC Refrac.	RHO Reflec.	WGC Gravity	RHO Gravity	WGC Ground Magnetics
D-28		X						
N-73			X			(C)		(C)
D-29		X				(Z,1)		(Z,1)
N-63	X							
D-22		X		(4A-1)	(11)	(1,2)		(1,2,4A-1 4A-2)
N-74			X			(4A1-10 4B)		(4A1-10 4B)
N-53		X		(1)		(1,5A,K)		(1,5A,K)
SD-10			X	(5)	(2)	(5,U,V)		(5,U,V)
N-54		X				(M,K)		(M,K)
N-75		X			(1)			
D-23	X			WNP Line 1	(2)			
D-37		X						
N-88			X			(4B)		(4B)

[1]This indicates that what is modeled as a dike on one figure may be modeled as a fault on the other, or it may be a case of a common source type with a different orientation.

[2]The geophysical profiles which intersect the inferred structures are indicated.

COMPILATION OF LINEAMENTS DETERMINED FROM ALL IMAGERY

Figure 231.1-1



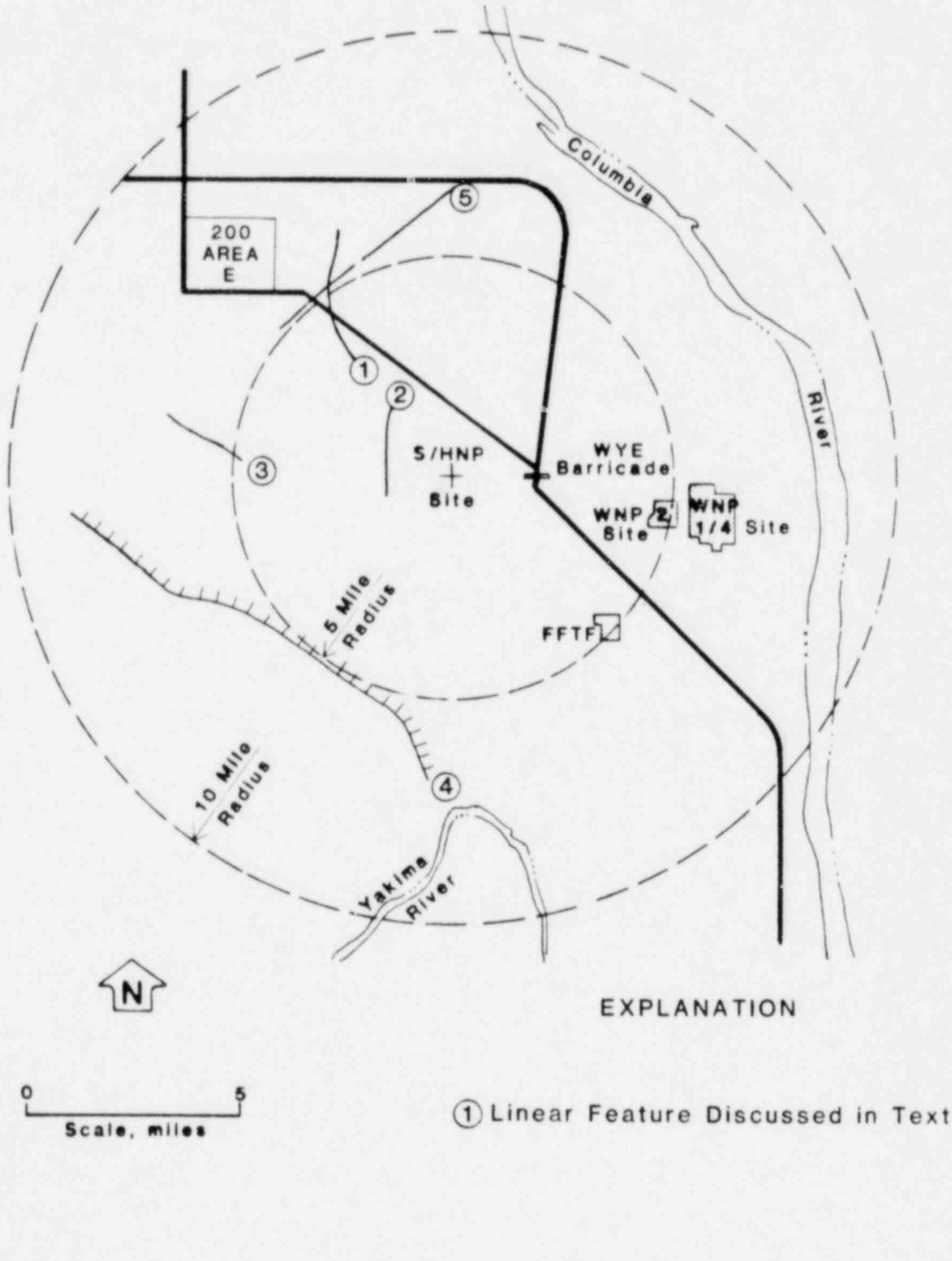
EXPLANATION

—— Lineaments (ie. tonal, textural, and vegetational contrasts, vegetational alignments.)

TTTTTT topographic escarpment (hachures indicating down hill side)

LINEAMENTS NOT ATTRIBUTED TO GLACIOFLUVIAL OR EOLIAN PROCESSES, WILDFIRE BURN SCARS, OR MAPPED GEOLOGIC STRUCTURE FROM INSPECTION OF IMAGERY

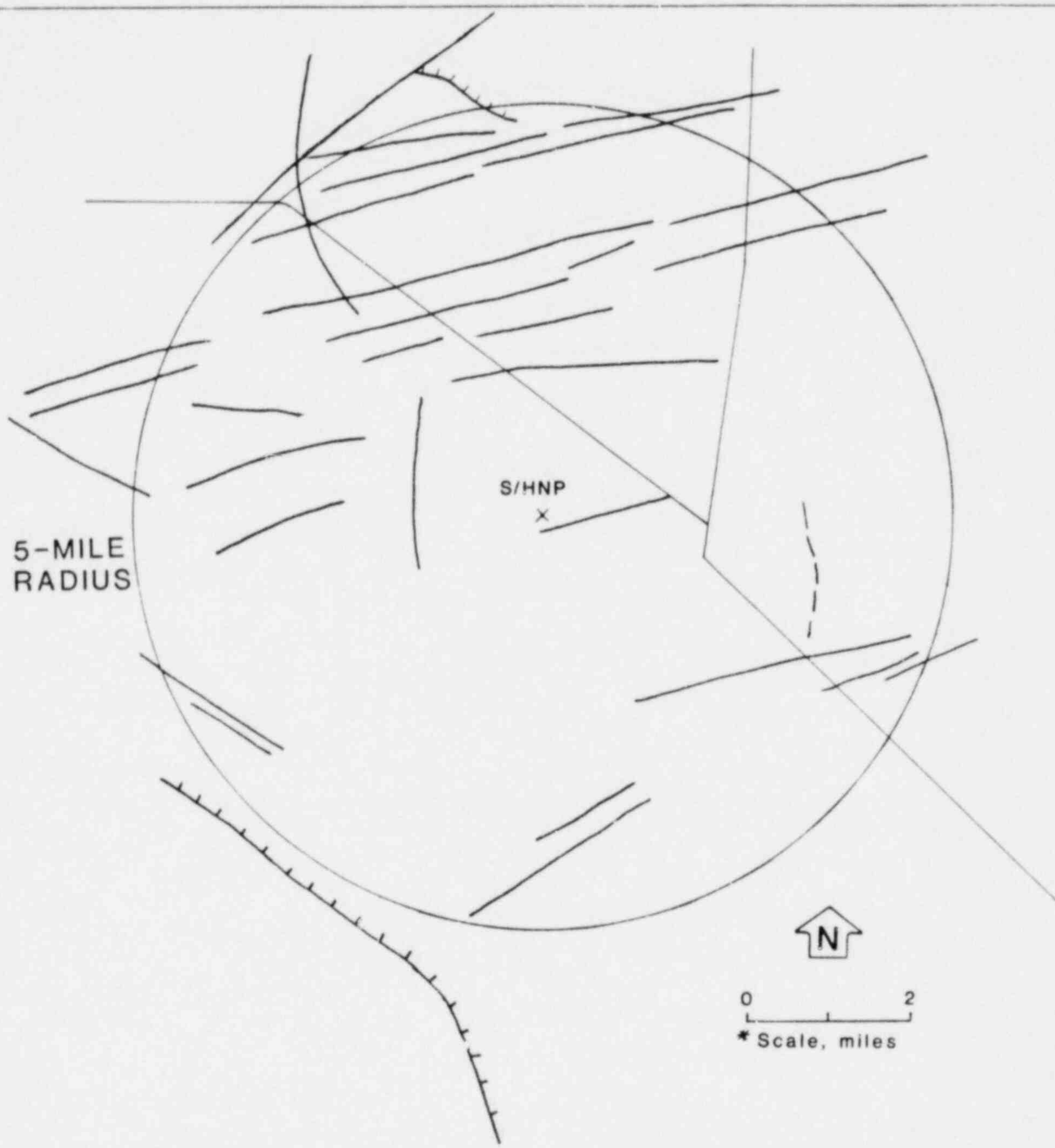
Figure 231.1-2



EXPLANATION

① Linear Feature Discussed in Text

COMPILATION OF LINEAMENTS WITHIN A 5-MILE RADIUS OF THE S/HNP SITE DETERMINED FROM ALL IMAGERY Figure 231.1-3



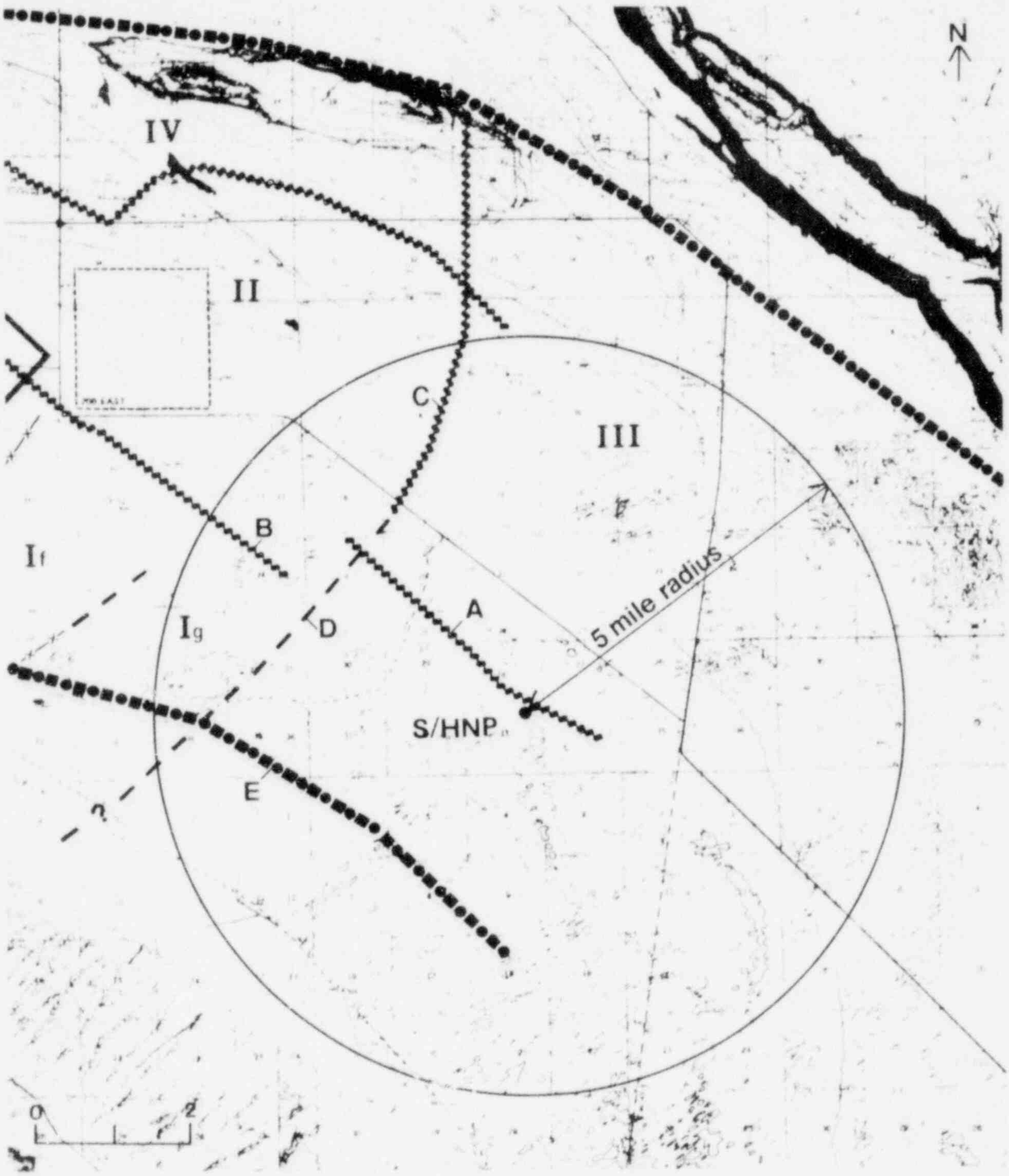
EXPLANATION

Lineaments (i.e., tonal, textural and vegetational contrasts: vegetational alignments).

Topographic escarpment (hachures indicating down hill side)

* NOTE: Scale of this map coincides with that of Figures 8-3, 8-8, B-11 and B-12 in Rockwell (1981).

Golder Associates



(AFTER ROCKWELL (1981) FIG. 8-8)

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

FEATURES FROM RHO-BWI-ST-14 (FIG.8-8)

FIGURE 231.3-1

QUESTION 240.1 (2.4)

Throughout Section 2.4 you refer to the top of basemat elevation. The staff is interested in access openings and access floor levels in regard to potential flood levels and potential flooding of safety related equipment. Please explain the relationship between the top of basemat elevation and plant grade and access floor elevations.

RESPONSE:

Top of basemat, elevation 527'-0", is the minimum floor elevation which has access openings into safety related structures, except for the diesel generator fuel oil tank vault, as noted in Table 2.4-1. Finished plant grade at safety related structures is elevation 526'-6" (see Figure 2.4-3). S/HNP flood design considerations will be as described in PSAR Section 2.4.2.2.

QUESTION 240.2

Discuss the potential for volcanic ash accumulation in the safety-related Mechanical Draft Cooling Tower Basins and provide an estimate of the maximum depth, and the basis for your estimate.

RESPONSE:

As indicated in Section 2.5.1.2.6.1 (page 2.5-103) of the WNP-2 FSAR, Amendment 18, the maximum volcanic ashfall rate at the Hanford Reservation is established to be 3 inches in 20 hours. The design basis ashfall scenario for the S/HNP is conservatively six inches in 24 hours.

As discussed in Section 9.2.5.2 of the S/HNP PSAR, the volume of the Ultimate Heat Sink basin required for at least 30 days cooling of the reactor includes a one foot depth for sedimentation resulting from suspended solids. Based on the conservative assumption that a 6" ash fall occurs in a 24 hour period, the airborne particulate concentration would be approximately 750 mg/m^3 which results in maximum cooling tower volcanic ash intake of about 66,000 lbs during the 24 hour period. With uniform distribution across the basin floor assumed, the height of the ash accumulation would be significantly less than the one foot allowance for sedimentation. This is considered to be an adequate allowance for volcanic ash accumulation.

QUESTION 240.3

Provide a commitment to operationally test the safety-related Mechanical Draft Cooling Towers to verify the parameters used for preliminary estimates of maximum temperature and water use.

RESPONSE:

The design of the UHS cooling towers for S/HNP has not been finalized to date as the towers have not yet been specified or purchased.

Design parameters used to construct mechanical draft cooling towers are very well understood today so that a high level of confidence exists that the towers' performance capability will meet or exceed cooling requirements for maximum conditions.

The S/HNP Standby Service Water System incorporates many conservative design assumptions, all of which will work to produce system components with extra margins of capability and a system that will generously exceed the requirements for handling the maximum heat loads at the maximum ambient conditions. Examples of these design conservatisms are listed below:

(1) Fuel pool heat loads used for determining the cooling requirements have been conservatively assumed to be over one-third greater than that actually calculated as necessary for maximum cooling conditions.

(2) Fuel pool decay heat and core decay heat are based upon a core power level of 4100 MWt whereas maximum licensed power will be 3800 MWt.

(3) Fuel pool initial (maximum) heat rate is assumed to be constant over the 30 day period, whereas heat rate actually decreases with time.

(4) UHS basin water volume is determined assuming all heat will be dissipated by evaporation, whereas other mechanisms beside evaporation will actually dissipate the heat loads. For instance, some heat will be dissipated by heating the stored basin water.

(5) The volume of water in the standby service water pump sump located below the UHS basin bottom required for sump submergence, was not included in calculating the 30 day water volume.

QUESTION 240.3 (Cont'd)

(6) The UHS basin storage adequacy analysis shows a significant portion of the total water inventory will remain at the end of the 30 day period.

(7) The 30-day water requirement corresponds to a water volume of 9.17×10^6 gallons. The transient analysis was performed based on a basin water volume of 8.24×10^6 gallons. The actual water volume of 9.17×10^6 gallons provides a conservative margin over that required.

Further details on UHS basin sizing and the method of analysis are provided in PSAR Section 9.2.5.

Based on the above discussion on the high level of confidence in the safety-related mechanical draft cooling towers' performance capability and Standby Service Water System design parameters' conservatism, committing to operationally test the Standby Service Water System mechanical draft cooling towers to verify preliminary estimate parameters will be discussed at the FSAR stage, should testing remain a requirement at that time.

QUESTION 241.1 (2.5.4)

In Amendment 23, section 2.5.4 to the PSAR, the applicant has proposed to support the common foundation mat for the seismic Category I structures about 20 ft below existing (and final) ground surface (plant grade) at elevation 507 (El 507), on Missoula sediments that occur between about El 520 and El 495. Plant grade is at El 527.

The Missoula sediments are described as medium dense to dense, clean medium sand, gray to black. Field tests in these soils showed Standard Penetration Test (SPT) values ranging from about 10 to 40 blows/ft (PSAR Fig. 2QA-1 through 37 and Table 2QB-1). Below El 507 the SPT values were generally greater than 20 blows/ft.

The underlying Pre-Missoula sediments, between about El 95 and El 480 are described as very dense, silty fine sand, dark yellowish-brown to olive gray. The SPT values in the Pre-Missoula sediment were generally on the order of 100 blows/ft. These soils are underlain by dense to very dense sand and gravel (SPT values greater than 45 blows/ft).

Based on our review of the applicant's submittals, it is our opinion that the Missoula sands in their present condition are not suitable for the direct support of seismic Category structures because the in-place densities are variable and, in some cases, too low to assure satisfactory structural support. Additionally, the proposed foundation support conditions are significantly inferior to the conditions adopted (and found by the staff to be acceptable) at the nearby Washington Public Power Supply System Unit 2 (WNP-2). The factors supporting our conclusion are as follows:

1. The lower SPT values recorded below proposed foundation level (17 to 23 blows/ft near Unit 1, 13 to 22 blows/ft near Unit 2) correlate to relative densities near or below 60% (PSAR Figs. 2QA-38 and 39). The in-place relative densities may, in fact, be near or below 50% according to recent studies by the Waterways Experiment Station (ASCE Journal, GT-11, November 1977, page 1295).

2. The applicant determined in-place relative densities of 9% to 53% in the exploratory trenches (PSAR Table 2QB-6). The applicant suggested that the tests were not representative because of soil layering. We believe that they are also indicative of loose, unsuitable in situ soil conditions.

QUESTION 241.1 (Cont'd)

At the WNP-2 site, shallow sands having SPT values generally in the range of 15 to 40 blows/ft were judged to have relative densities in the range of 30 to 50% (WNP-2 FSAR, App. 2.5F, Figs. 4 and A-3 through A-7). These soils were excavated to a depth of about 40 ft below grade and recompacted to about 80% relative density in order to provide suitable foundation support.

Based on the applicant's submittals and a telephone discussion between the staff and the applicant on March 2, 1981, we understand that the applicant concluded that the Missoula sands are suitable for foundation support based on the following factors:

1. The average SPT values of 25 to 30 blows/ft and corresponding relative densities of 75% to 80% are within ranges that will provide suitable foundation support and that the large, thick foundation mats will distribute structural loads over local, loose pockets.

2. The plate load tests in the exploratory trenches show relatively high elastic modulus values for the Missoula sands (10,000 psi to 20,000 psi) so that calculated settlements of structures under static loads are small.

3. The geophysical studies show relatively high shear wave velocities (900 ft/sec) in the Missoula sands so that calculated settlements of structures under earthquake loads are small.

4. The applicant beieves that removing and recompacting the Missoula sands may not produce improved densities in the bearing soils.

We find that the applicant's information and evaluation does not resolve our concern for the suitability of the in-place Missoula sand as a foundation bearing material. Thus, we ask that the applicant submit an alternative to the presently proposed plan that has seismic Category I foundations supported directly in the in situ Missoula sand. For guidance, the applicant should refer to the WNP-2 foundation construction wherein medium dense to dense sands were excavated to a depth of about 40 ft (down to dense sand) and foundation elevations were re-established by use of structural backfill. The staff found this procedure to be acceptable for the WNP-2.

QUESTION 241.1 (Cont'd)

RESPONSE:

This question asks the applicant submit an alternative to the presently proposed plan that has Seismic Category I foundations supported directly in the in-situ Missoula sands. During discussions with the staff on March 17, 1982, it was agreed that the applicant could instead submit, for Staff review and approval, a program of additional investigations and testing for the purpose of further demonstrating the suitability of the Missoula sands for support of Category I foundations.

The applicant now agrees to remove the medium-dense to dense Missoula sands underlying Category I foundations down to the very dense sands which occur at approximately Elevation 495 and to reestablish foundation elevations with structural backfill constructed in accordance with PSAR Sections 2.5.4.5.2 through 2.5.4.5.5. A revised Figure 2.5-15 will be provided with Amendment 24 to the PSAR. The applicant does, however, reserve the prerogative to preclude this program of overexcavation and backfill by demonstrating to the Staff's satisfaction, prior to construction of Category I foundations, that the in-situ Missoula sands are a suitable foundation material.

QUESTION 241.2 (App 2L)

Table 2L-5 shows "P" wave and "S" wave values and calculated Poisson's ration values that appear to be inconsistent. Provide a discussion of the bases for the acceptability of these design values.

RESPONSE:

In Table 2L-5, calculated Poisson's ratio values above elevation 425 range between 0.28 and 0.32, which is coincidentally very close to the assumed value of 0.30, although there is no good reason to except that Poisson's ratio values determined under dynamic conditions at small strain levels should necessarily be appropriate to static conditions at larger strain levels. Below elevation 400 (i.e. below the water table), the seismic velocities indicate high Poisson's ratio values which are indicative of nearly incompressible material. Since the seismic disturbance would be associated with virtually undrained (i.e. incompressible) soil behavior, under static (drained) loading conditions, the estimated values shown in Table 2Q-3 are considered more appropriate for the materials below the water table.

In fact, the only difference of any significance in the Poisson's ratio values of Tables 2Q-3 and 2L-5, occurs between elevations 400 and 425. The seismic survey suggests a Poisson's ratio of 0.39, compared with the estimated value of 0.3. As noted above the low strain values can reasonably be expected to differ somewhat from the high strain values, and the Poisson's ratio values to Table 2Q-3 are considered appropriate values to be used in settlement computations at the strain levels which occur beneath the loaded foundation.

Finally, reasonable variations in Poisson's ratio will have no engineering impact of any significance.

QUESTION 241.3 (2.5.4.5)

Provide a description of the anticipated bearing conditions and bedding details for soil-supported seismic Category I pipes and conduits. Provide a summary of the specifications for bedding and backfilling.

RESPONSE:

See new PSAR Section 2.5.4.5.7.

QUESTION 241.4 (2.5.4.5)

Provide a correlation between the seismic Category I structures listed under "B. Other Structures" on Sheet 22 of Table 3.2-1 and the numerical listing on Figure 1.2-1. Identify the proposed foundation elevations and conditions of any seismic Category I structures that are not shown on Figure 2.5-15.

RESPONSE:

Six Category I Structures are listed in Table 3.2-1, "B. Other Structures", as follows:

B.1 - Standby Service Water and Storage Basins
These are part of the Ultimate Heat Sink complex. (Building 10 on Fig 1.2-1).

B.3 - New Fuel Structure
New fuel is stored in the Fuel Building. (Building 1 on Fig. 1.2-1).

B.4 - Diesel-Generator Fuel Storage Facilities
These are the Diesel Fuel Storage Tanks. (Building 34 on Fig 1.2-1).

B.5 - Station Battery Rooms
These are in the Control Building. (Building 3 on Fig. 12-1).

B.6 - Spent Fuel Pool
This is in the Fuel Building. (Building 1 on Fig. 1.2-1).

B.7 - Ultimate Heat Sink
(Building 10 on Fig 1.2-1)

The foundation for the Ultimate Heat Sink and the common mat are shown on Fig. 2.5-15. The Fuel and Control Buildings are located on the common mat.

The Diesel-Generator Fuel Storage Tanks will be located in an underground structure, at a foundation elevation of approximately 506'.

Treatment of the foundation for the Diesel-Generator Fuel Storage Structure will be the same as for the other foundations for Category I structures.

QUESTION 241.5 (2.5.4.5)

Provide a commitment to notify the NRC staff in advance of the completion of foundation excavations so that the staff may inspect the excavations.

RESPONSE:

The NRC staff will be notified in advance of the completion of foundation excavations so that the staff may inspect the excavation.

QUESTION 241.6 (2.5.4.5)

Provide a description of the procedures that will be adopted to protect and maintain temporary soil slopes and to provide adequate drainage around structures so as to assure that foundation soils will not be damaged by local heavy rains and erosion during construction. Include a description of the periodic inspection procedures that will assure proper maintenance of temporary slopes and drainage facilities.

RESPONSE:

The site is in a desert area, where the 100 year precipitation is about 2 inches. The site soils are relatively pervious. The site area will be graded to drain, as shown on Fig 2.4-3 of the PSAR. Methods for controlling runoff on the foundation mats and mud mats and protective measures for wind erosion are discussed in PSAR Section 2.5.4.5.1.

QUESTION 241.7 (2.5.4.5)

Discuss the efficacy of using the Proctor method (ASTM D-1557) for field density control of the clean sand in view of the testing difficulties encountered during the exploration, as described on page 2QB-4. Propose alternatives for field control of backfill.

RESPONSE:

See revised Sections 2.5.4.5.3 and 2.5.4.5.5.

QUESTION 241.8 (2.5.4.5)

Specify the gradation limits that will be acceptable for structural backfill material. Also describe how the excavated soils that are used for backfilling will be mixed and blended to provide homogeneity; that is, describe how problems with obtaining relative density values in compacted fill will be avoided in view of the problems encountered with determining relative density in the exploratory test trenches (see page 2Q-11, page 2Q-14 and Table 2QB-6).

RESPONSE:

As discussed in Section 8.4.2 of Appendix 2Q, the black clean medium sands (Missoula sand), which occur beneath the upper silty materials and above the very dense Pre-Missoula sediments, will be used for structural backfill. The maximum particle size will be no larger than 3 inches. Prior to construction, a borrow investigation will be made and backfill placement procedures will be established using a test fill program. This program will establish suitable gradation limits for structural backfill, a suitable construction procedure to obtain an average of 85 percent relative density with no values below 75 percent, and appropriate procedures for quality control testing of backfill. The latter will involve development of a correlation between relative density and relative compaction so that the more efficient relative compaction procedure can be used in the site backfill.

QUESTION 271.1

Page 3.10-1 of Section 3.10, and page 3.10-3, state that the seismic qualification of electrical equipment supplied, other than by General Electric, will be in accordance with IEEE 341-1971 (and NRC Staff Technical Position EICSB No. 10). However, both revisions 1 and 2 to the Standard Review Plan (SRP) Section 3.10 state that, for plants for which the construction permit application was docketed after October 27, 1972, the qualification of electrical equipment should be in accordance with IEEE 344-1975 and Regulatory Guide 1.100. Also, the 251 NSSS GESSAR, referenced in the Skagit/Hanford PSAR, references IEEE 344-1975. Therefore, the PSAR should be revised to be consistent with the SRP and the 251 NSSS GESSAR.

RESPONSE:

See revised Sections 3.2, 3.10, 7.1 and 7.3.

QUESTION 403.1 (8.2.1)

Section 8.2.1 of the PSAR states that the proposed Ashe-Hanford No. 2 line will be looped through the S/HNP Substation to serve Unit 2 and that the Ashe-Hanford No. 2 line will be constructed as the generating base in Hanford area increases. Describe the impact on the offsite power system to S/HNP if the Ashe-Hanford No. 2 line is not in place by completion of Unit 2, and the Ashe-Hanford No. 1 line must service both Units. How will the stability analysis and line loading be affected?

RESPONSE:

Puget Sound Power & Light Company (PSPL) plans, for economic reasons, to loop the Ashe-Hanford No. 2 500-kV line into the S/HNP Substation for the startup of Unit 2.

If, for reasons beyond PSPL's control, the Ashe-Hanford No. 2 500-kV line is not looped through the S/HNP Substation at the completion of Unit 2, there will be no reduction in the reliability of the 500-kV buses that serve the Plant Substation Transformers (PSXs) from that of Unit 1. The S/HNP Substation will contain two physically independent 500-kV buses and two (PSXs) for Unit 1 and Unit 2 as shown on PSAR Figure 8.2-6. The 500-kV portion of the Substation will have a breaker-and-a-half configuration for each circuit together with breaker failure backup protection. This system will maximize the reliability of the 500-kV buses that serve the PSXs.

With the assumption that the Ashe-Hanford No. 2 500-kV line would not be completed at the startup of Unit 2, the line loading of 3278 megawatts on the S/HNP - Hanford No. 1 line would be less than the summer rating of 4000 megawatts for this circuit as shown on Figure 403.1-1. The system is dynamically stable when the Ashe-Hanford No. 1 500-kV line is looped into the S/HNP Substation at startup as shown on Figure 403.1-1. The generator rotor swing curves for a 3-phase 4 cycle fault are shown on Figures 403.1-2 and 403.1-3 and are within safe limits. A summary of the dynamic stability cases analyzed is shown in Table 403.1-1.

QUESTION 403.2 (8.2.2)

The stability analysis discussed in Section 8.2.2.2 of the PSAR was based on the grid configuration shown in Figure 8.2-7. Since Washington Public Power Supply System Unit 4 has been cancelled, describe the effects on the stability analysis and power flow without Unit 4.

RESPONSE:

Power flow drawing Figure 403.2-1 shows the transmission grid without Washington Public Power Supply System (WPPSS) Unit 1 and Unit 4. With the assumption that Unit 4 is not built and Unit 1 is out of service, the line loadings are reduced between Ashe-S/HNP and S/HNP-Hanford. The system is dynamically stable for a 3-phase 4 cycle fault as shown on generator rotor swing curves, Figures 403.2-2 and 403.2-3. The rotor swings are about 4 degrees greater with Units 1 and 4 out of service which is within safe limits.

Because the difference in dynamic stability with and without WNP-4 are relatively small, the analysis provided in PSAR Section 8.2.2.2 is sufficiently applicable to cover dynamic stability without WNP-4.

A summary of the dynamic stability cases analyzed is shown in Table 403.1-1.

QUESTION 403.3 (8.2.2)

The cases analyzed in the stability analysis for faulted lines which are identified in Figure 8.2-7 assume the faults to be cleared in four cycles. Will the system still remain stable for longer fault clearing times such as would occur given a circuit breaker failure.

RESPONSE:

Power flow drawing Figure 403.3-1 shows the transmission grid when the Ashe-Hanford No. 1 and No. 2 lines are looped into the S/HNP Substation. With the assumption that a short circuit has occurred on the line to either Ashe or Hanford, the system is dynamically stable for a 3-phase 8 cycle fault. However, in normal operation, the 500 kV transmission system would not experience a 3-phase 8 cycle fault but would generally experience a 3-phase fault for a duration of 4 cycles which would then continue on as a single-phase fault for a duration of 8 cycles. At or near the end of 12 cycles, the backup protective relays on the system would operate and clear the circuit breaker that failed to open. The stress on the system of a 3-phase 8 cycle fault is nearly equal to a three-to-one phase 12 cycle fault. A 3-phase 8 cycle fault as shown on generator rotor swing curves, Figures 403.3-2, 403.3-3 and 403.3-4 are stable. Therefore, the three-to-one phase 12 cycle fault will be stable.

The results of all of the dynamic stability cases analyzed are summarized in Table 403.1-1.

QUESTION 403.4 (8.2)

Inconsistencies which exist in the following amendment 23 figure should be corrected:

Figure 8.3-1 - This drawing does not agree with existing PSAR Figure 8.3-2 in the Class 1E power feed to the non Class 1E load center.

Figure 8.2-7 (Sheets 1 and 2A) -
Figure 8.2-8 sheet 1 is referenced on this drawing for two separate stability cases.

Figure 8.2-7 (Sheet 3) -
The generating unit labeled as 'WPSS 1 and 2" appears mislabeled.

RESPONSE:

PSAR Figure 8.3-2 has been revised to be consistent with Figure 8.3-1.

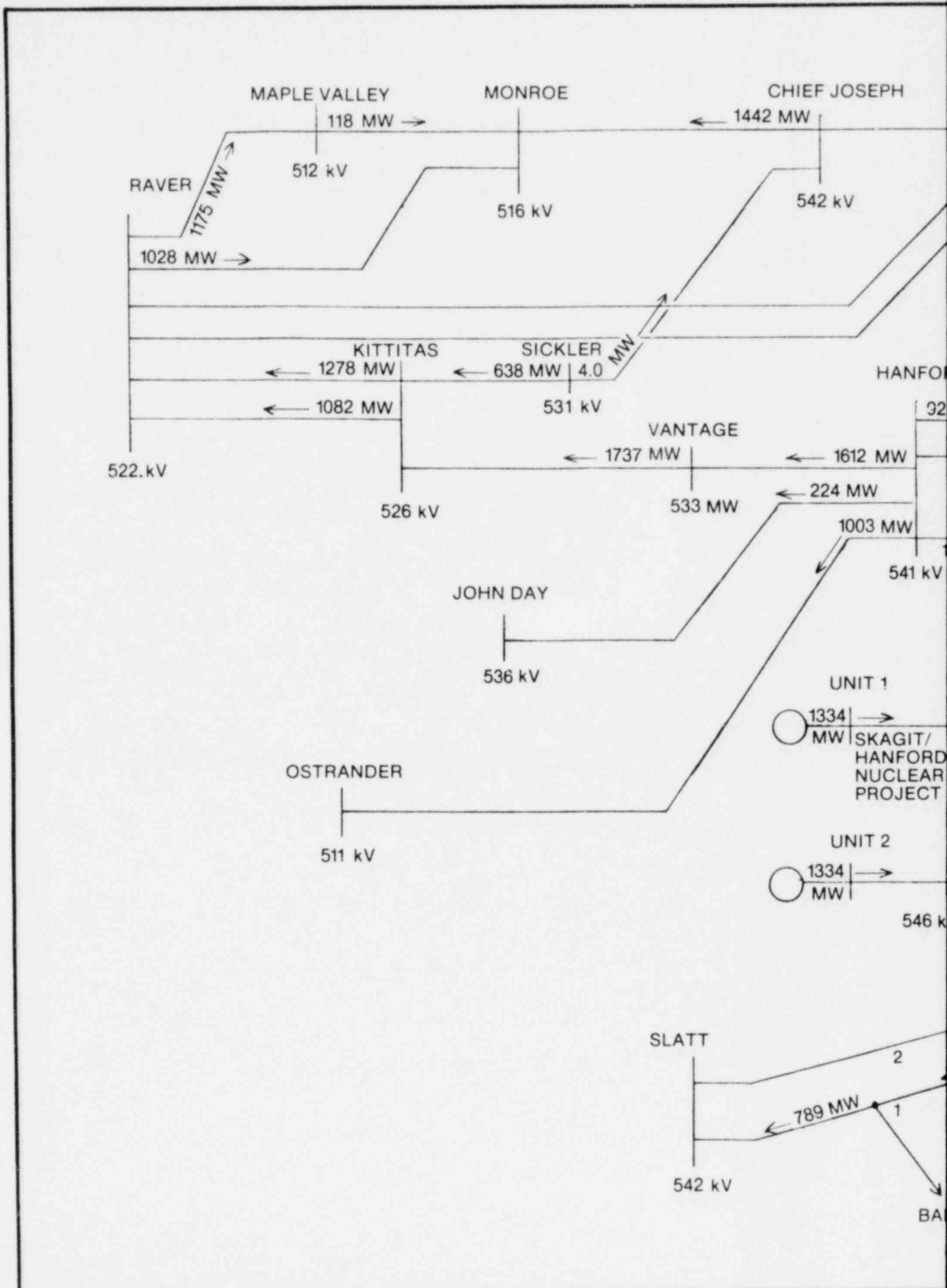
See revised Figure 8.2-7 Sheet (2A of 8). The reference to Figure 8.2-8 Sheet 1 on Figure 8.2-7 Sheet (2A of 8) has been corrected to Figure 8.2-8 Sheet 3.

The generating unit labeled as 'WPPSS 1 & 2' on Figure 8.2-7 Sheet (3 of 8) has been corrected to 'WPPSS 1 & 4'. See revised Figure 8.2-7 (Sheet 3 of 8).

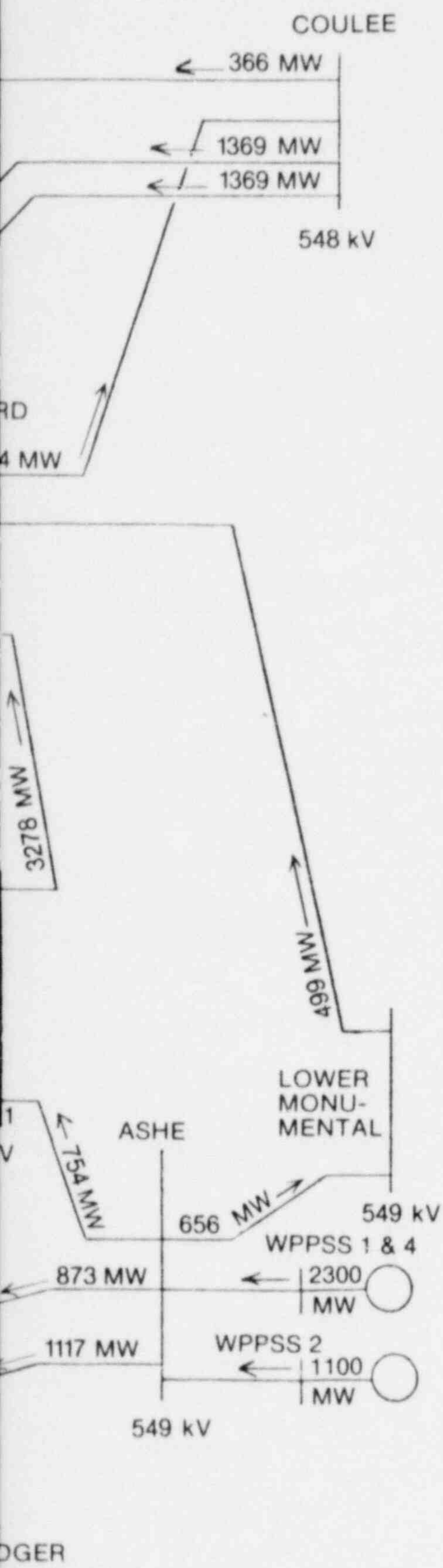
TABLE 403.1-1

DYNAMIC STABILITY SUMMARY TABLE

Powerflow Figure Numbers	Dynamic Stability Swing Curve Figure Numbers	Number of Generators at the Project	Three-Phase Fault on:	Transmission Line Configuration Comment	Result
403.2-1	403.2-2	2	Project Line-End of Project-to-Hanford #1 500 kV Line	WPPSS Units 1 & 4 Disconnected	Stable
403.2-1	403.2-3	2	Project Line-End of Project-to-Ashe #1 500 kV Line	WPPSS Unit 1 & 4 Disconnected	Stable
403.1-1	403.1-2	2	Project Line-End of Project-to-Hanford #1 500 kV Line	Project-to-Hanford #2 and Project-to-Ashe #2 500 kV Line Out of Service	Stable
403.1-1	403.1-3	2	Project Line-End of Project-to-Ashe #1 500 kV Line	Project-to-Hanford #2 and Project-to-Ashe #2 500 kV Line Out of Service	Stable
403.3-1	403.3-2	2	Project Line-End of Project-to-Ashe #2 500 kV Line. Drop Ashe-Project-Hanford #2 500 kV Lines (Simulating Backup Breaker Clearing of A Fault)	None	Stable
403.3-1	403.3-3	2	Project Line-End of Project-to-Ashe #1 500 kV Line. Drop Project-to-Ashe 500 kV Line (Simulating Backup Breaker Clearing of A Fault)	None	Stable
403.3-1	403.3-4	2	Project Line-End of Project-to-Hanford #1 500 kV Line. Drop Project-to-Hanford 500 kV Line and Project Unit #2 (Simulating Backup Breaker Clearing of A Fault)	None	Stable



MAPLE VALLEY
MONROE
CHIEF JOSEPH
118 MW
1442 MW
512 kV
RAVER
1175 MW
516 kV
542 kV
1028 MW
KITTITAS
SICKLER
4.0 MW
1278 MW
638 MW
531 kV
HANFOR
1082 MW
92
VANTAGE
522.kV
1737 MW
1612 MW
526 kV
224 MW
533 MW
1003 MW
541 kV
JOHN DAY
536 kV
UNIT 1
1334 MW
SKAGIT/
HANFORD
NUCLEAR
PROJECT
OSTRANDER
511 kV
UNIT 2
1334 MW
546 k
SLATT
2
789 MW
1
542 kV
BA



DYNAMIC SWING CURVE CASE NUMBER	DESCRIPTION OF DYNAMIC STABILITY TEST	RESULTS
403.1-2	3-Phase fault on Project's 500 kV bus, clear Project-to-Hanford 500 kV line in 4 cycles.	Stable
403.1-3	3-Phase fault on Project's 500 kV bus, clear Project-to-Ashe 500 kV line in 4 cycles.	Stable

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POWER FLOW DIAGRAM AND DYNAMIC STABILITY TEST

FIGURE 403.1-1

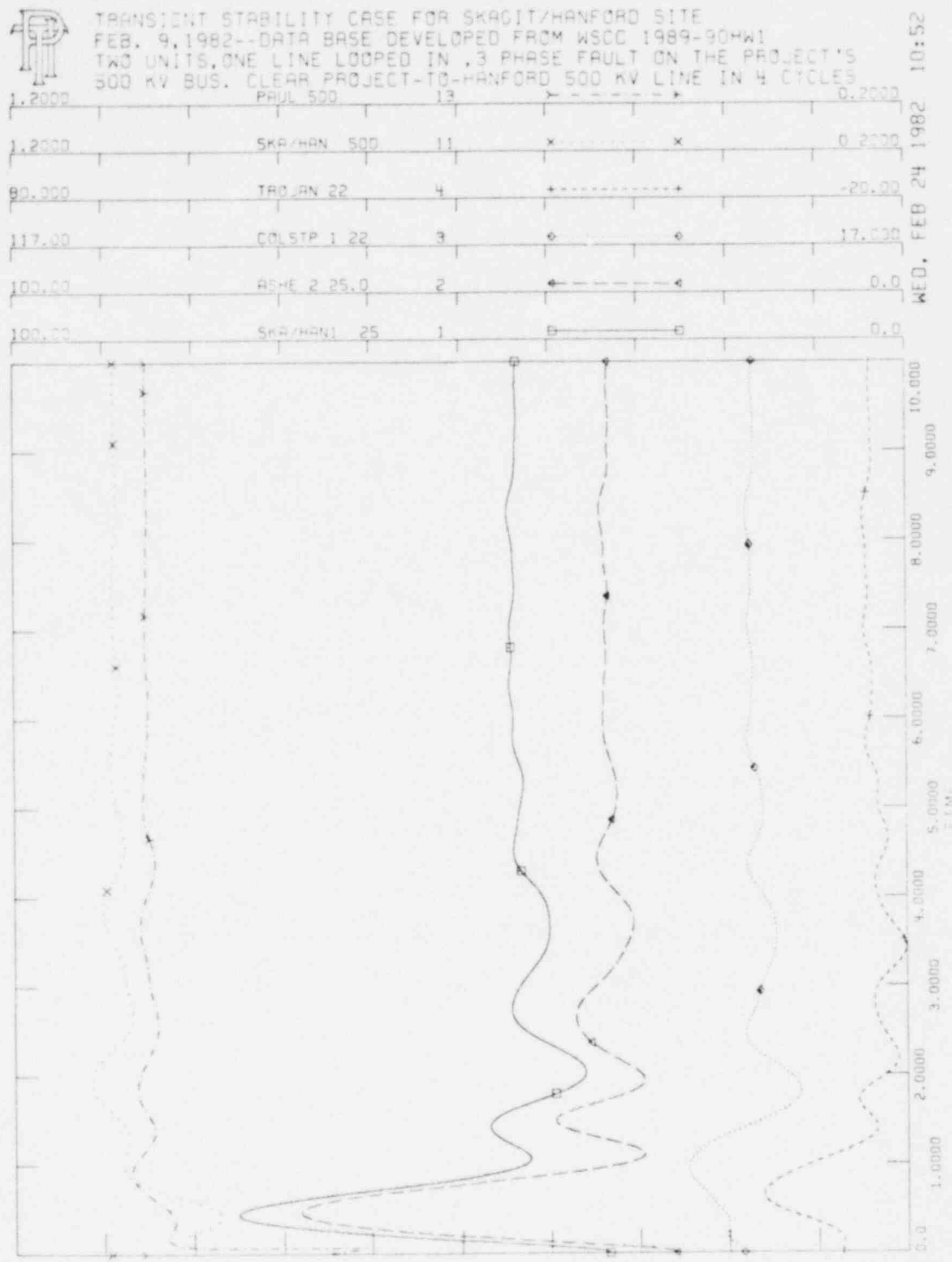

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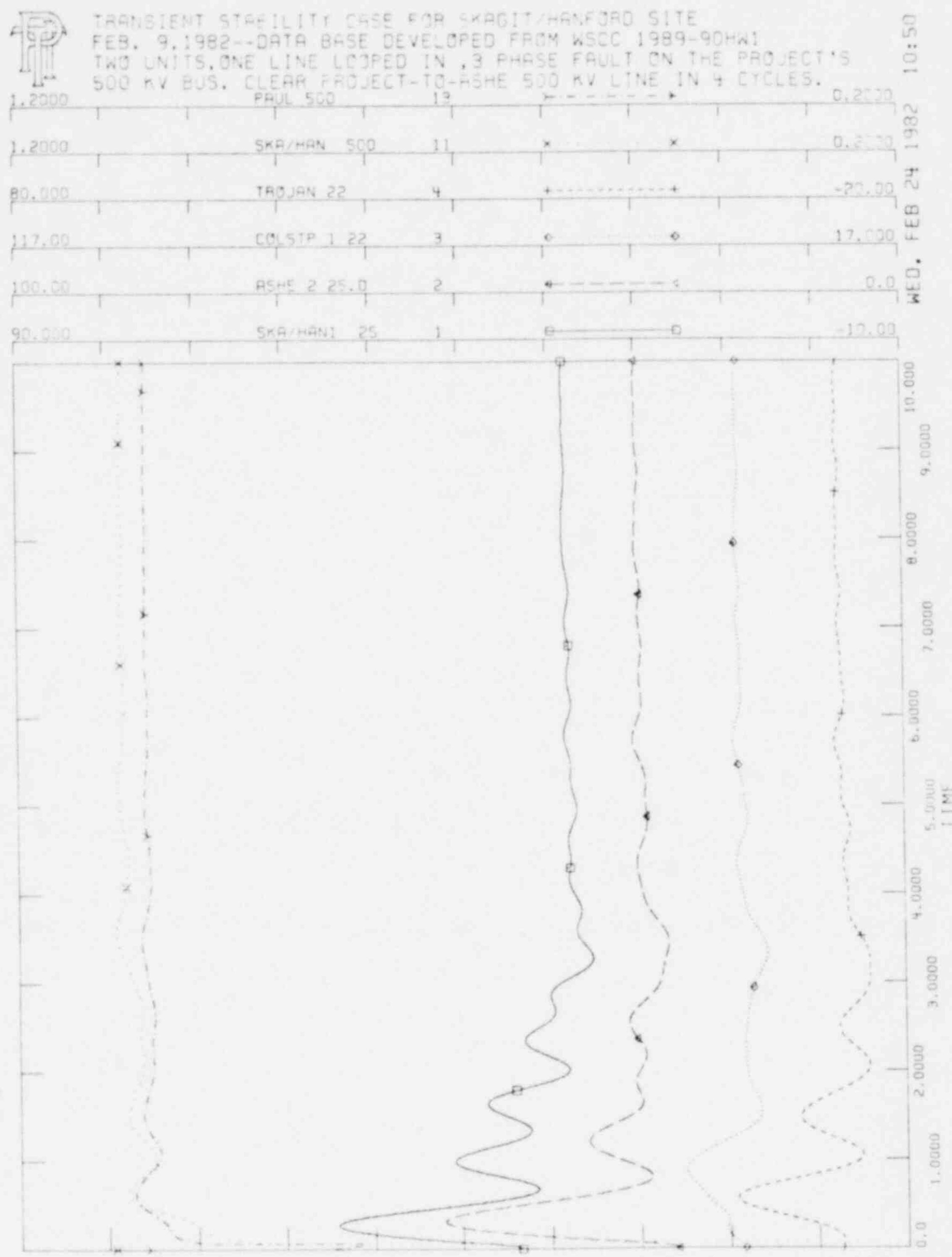
FIGURE 403.1-2



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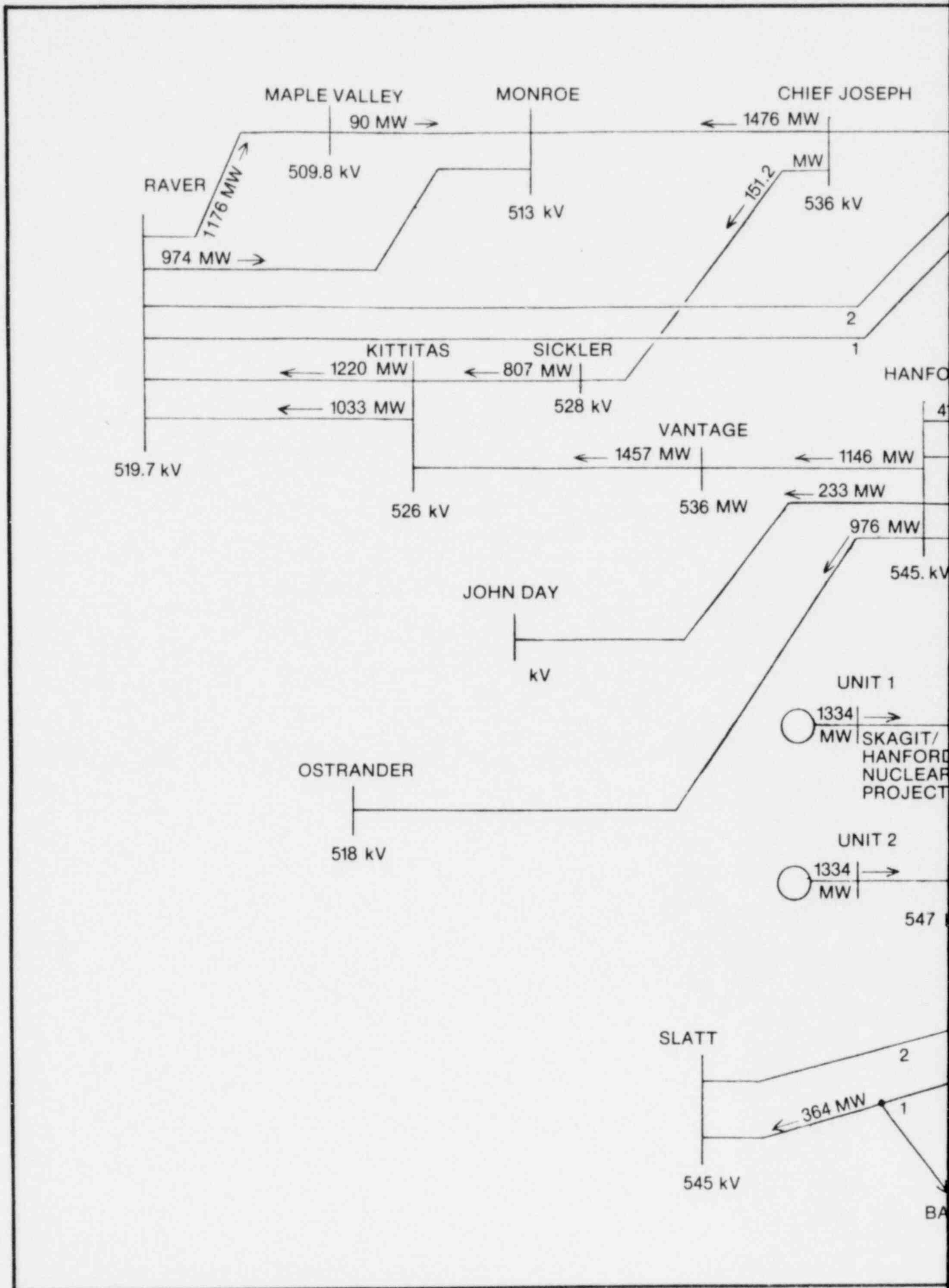
DYNAMIC STABILITY PLOTS

FIGURE 403.1-3


MAPLE VALLEY
MONROE
CHIEF JOSEPH
90 MW
1476 MW
1176 MW
509.8 kV
MW
151.2
RAVER
513 kV
536 kV
974 MW
2
1
KITTITAS
SICKLER
1220 MW
807 MW
HANFO
528 kV
1033 MW
VANTAGE
519.7 kV
1457 MW
1146 MW
526 kV
536 MW
233 MW
976 MW
545. kV
JOHN DAY
kV
UNIT 1
1334 MW
SKAGIT/
HANFORD
NUCLEAR
PROJECT
OSTRANDER
518 kV
UNIT 2
1334 MW
547
SLATT
2
364 MW
1
545 kV
BA

DYNAMIC STABILITY SWING CURVE CASE NUMBER	DESCRIPTION OF DYNAMIC STABILITY TEST	RESULTS
403.2-2	3-Phase fault on Project's 500 kV bus. Clear Project-to-Hanford #1 500 kV line in 4 cycles.	Stable
403.2-3	3-Phase fault on Project's 500 kV bus. Clear Project-to-Ashe #1 500 kV line in 4 cycles.	Stable

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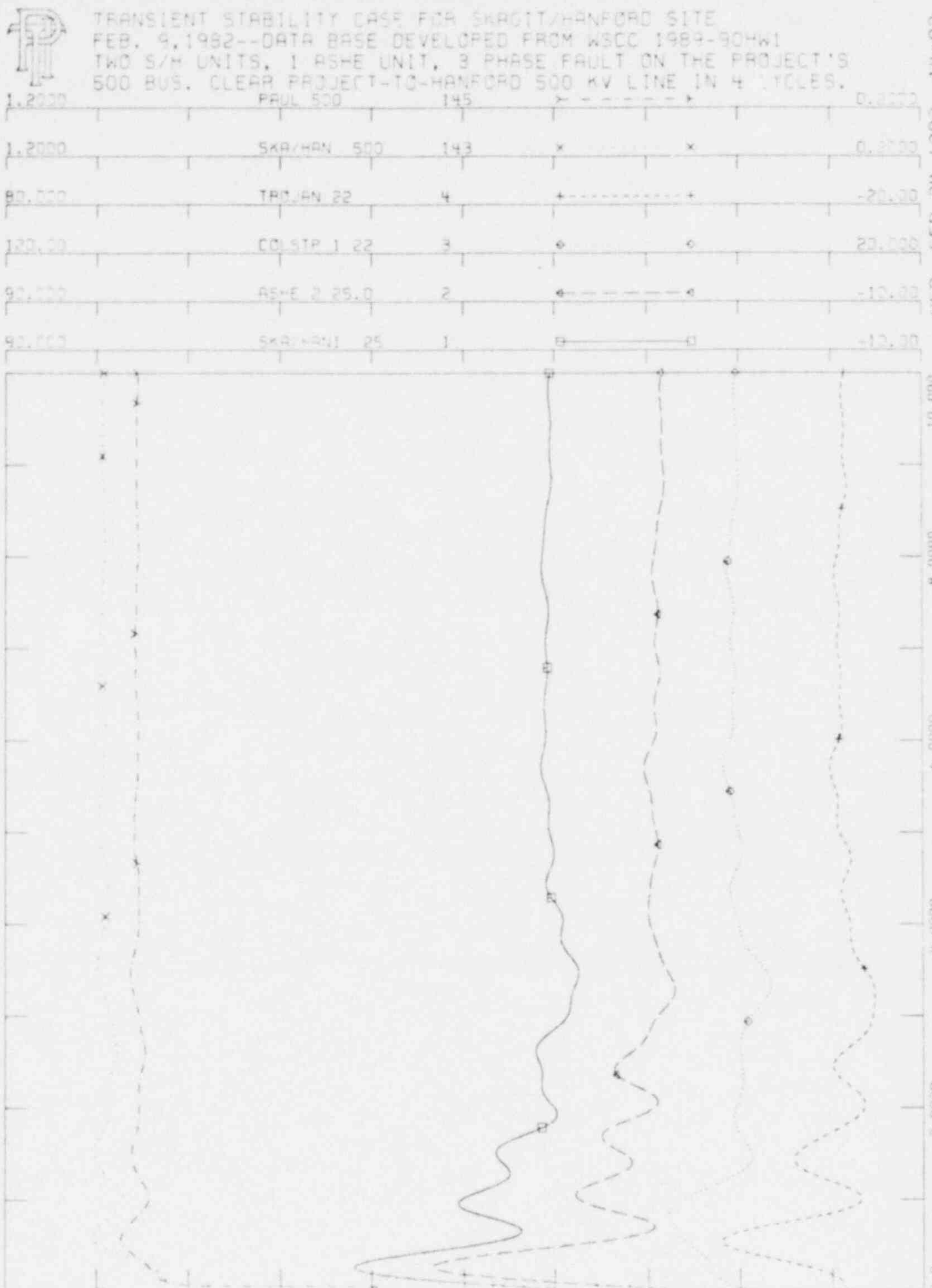
POWER FLOW DIAGRAM
AND
DYNAMIC STABILITY TEST

FIGURE 403.2-1



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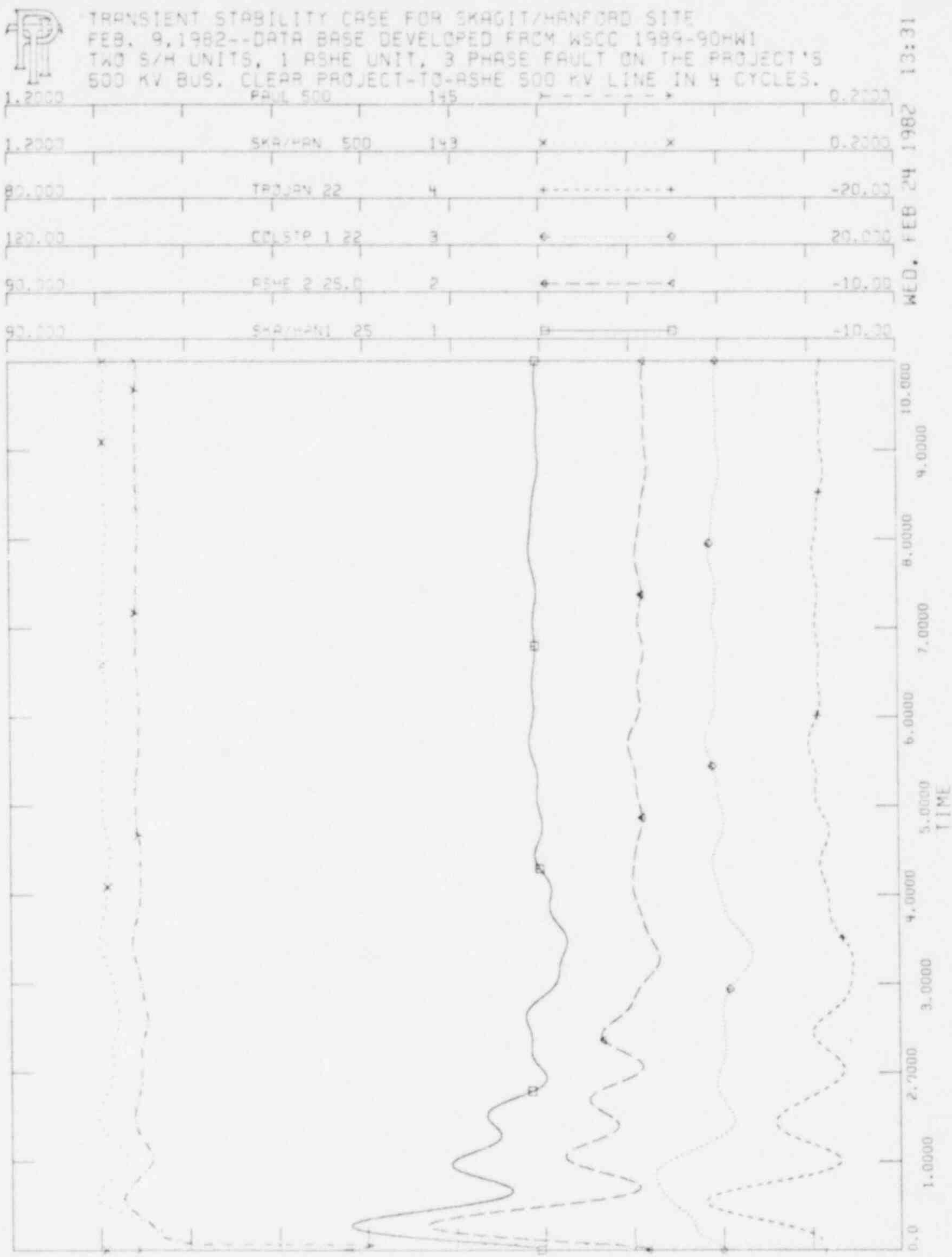
DYNAMIC STABILITY PLOTS

FIGURE 403.2-2



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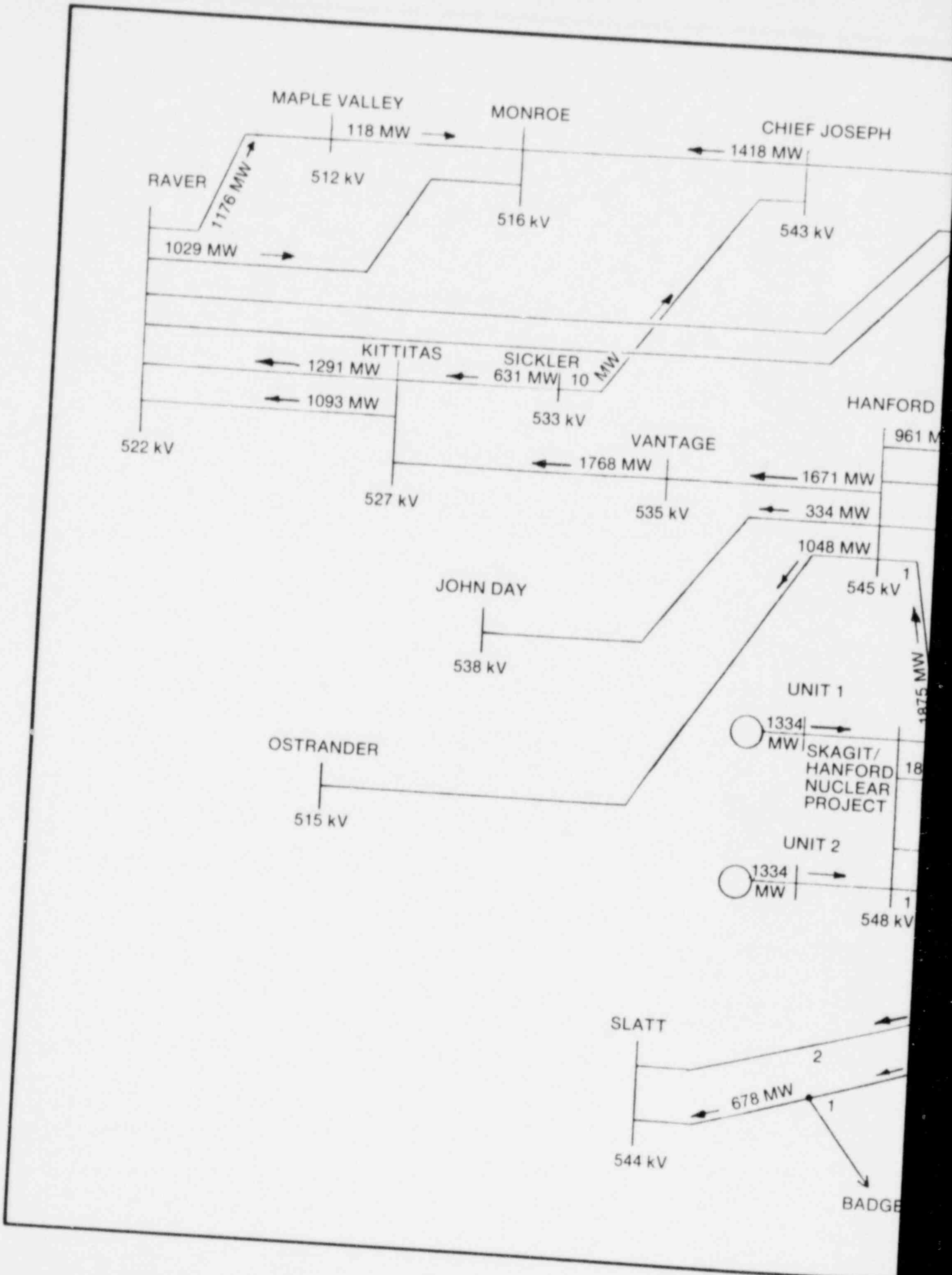
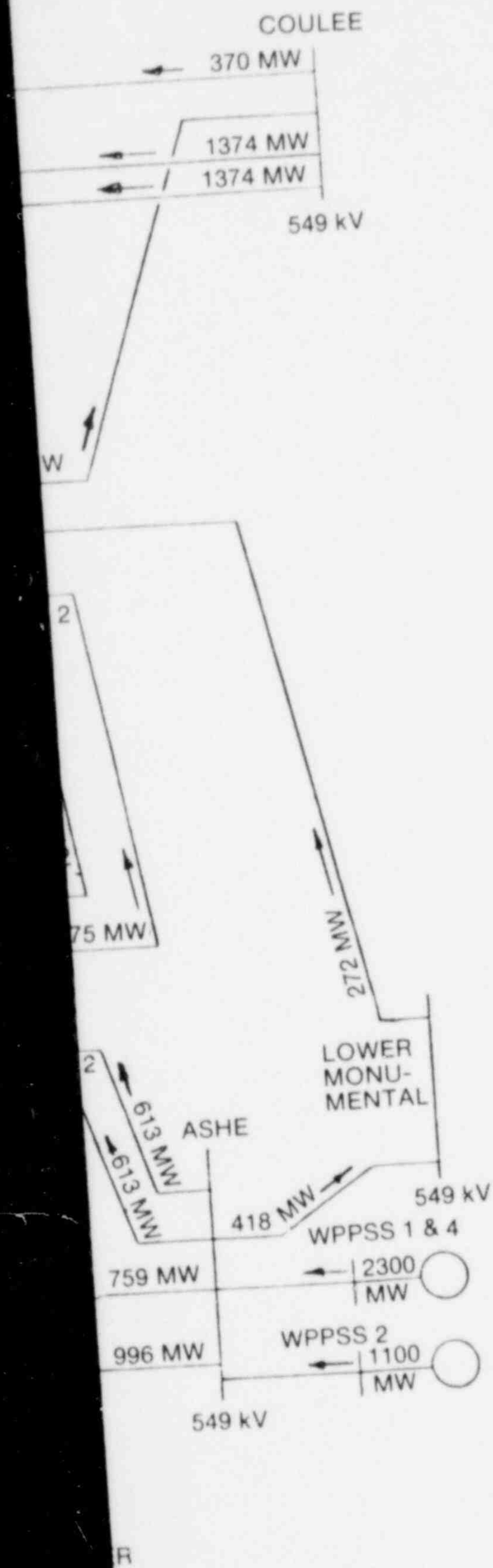
DYNAMIC STABILITY PLOTS

FIGURE 403.2-3



MAPLE VALLEY
118 MW
MONROE
CHIEF JOSEPH
1418 MW
RAVER
1176 MW
512 kV
516 kV
543 kV
1029 MW
KITTITAS
SICKLER
1291 MW
631 MW
10 MW
1093 MW
533 kV
522 kV
HANFORD
961 M
VANTAGE
1768 MW
1671 MW
527 kV
535 kV
334 MW
1048 MW
545 kV
JOHN DAY
538 kV
1875 MW
UNIT 1
1334 MW
SKAGIT/ HANFORD NUCLEAR PROJECT
18
OSTRANDER
515 kV
UNIT 2
1334 MW
548 kV
SLATT
2
678 MW
1
544 kV
BADGE



DYNAMIC STABILITY SWING CURVE CASE NUMBER	DESCRIPTION OF DYNAMIC STABILITY TEST	RESULTS
403.3-2	3-Phase fault on the Project's 500 kV bus, clear Project-to-Hanford #2 and Project-to-Ashe 500 kV lines in 8 cycles, continue run for 10 seconds.	Stable
403.3-3	3-Phase fault on the Project's 500 kV bus, clear Project-to-Ashe #1500 kV line and drop Project's unit #2 in 8 cycles, continue run for 10 seconds.	Stable
403.3-4	3-Phase fault on the Project's 500 kV bus, clear Project-to-Hanford #1500 kV line and drop unit #2 in 8 cycles, continue run for 10 seconds.	Stable

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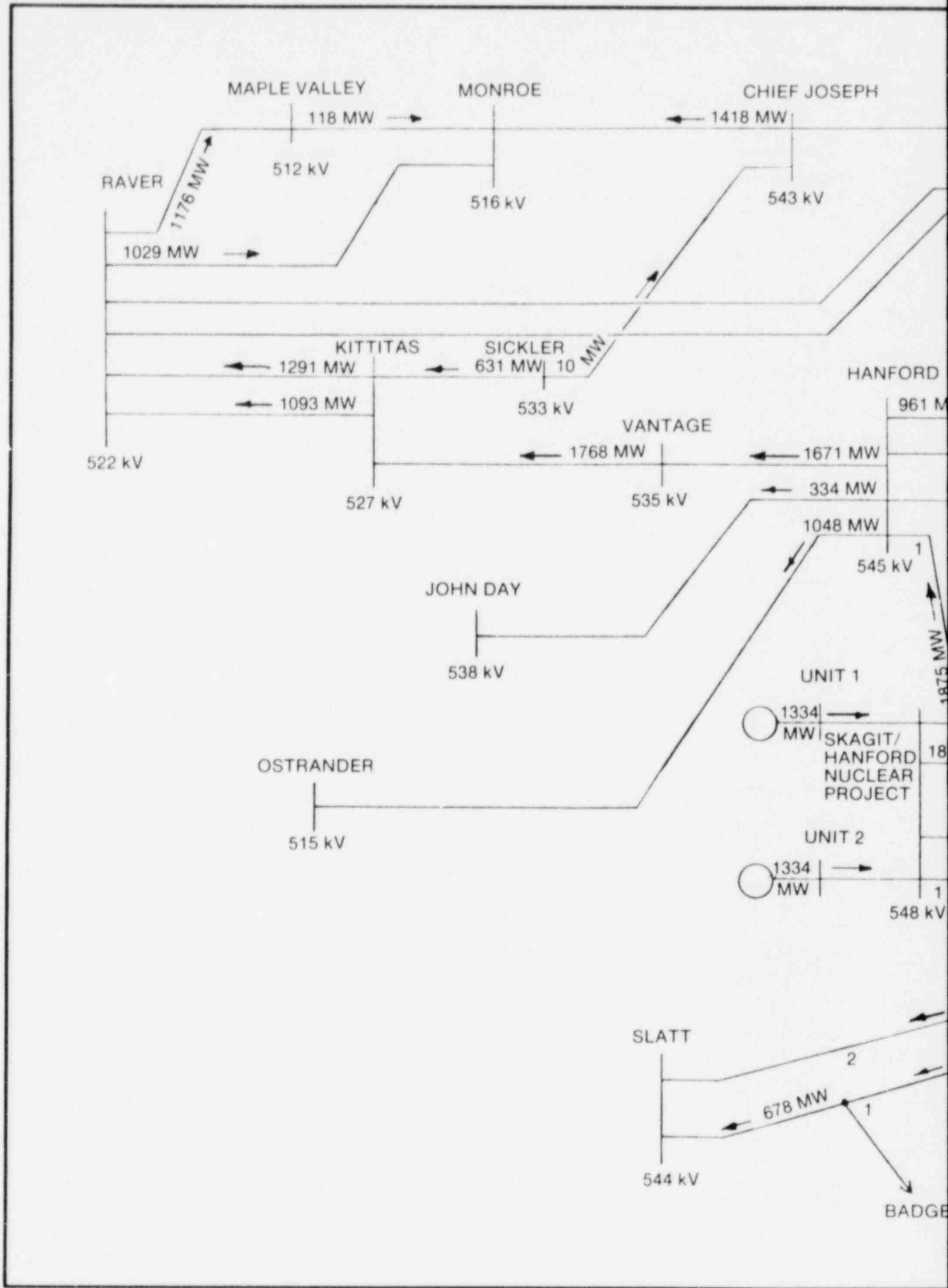
POWER FLOW DIAGRAM AND DYNAMIC STABILITY TEST

FIGURE 403.3-1


MAPLE VALLEY
MONROE
CHIEF JOSEPH
118 MW
1418 MW
512 kV
516 kV
543 kV
RAVER
1176 MW
1029 MW
KITTITAS
SICKLER
1291 MW
631 MW
10 MW
1093 MW
533 kV
HANFORD
961 M
VANTAGE
522 kV
1768 MW
1671 MW
527 kV
535 kV
334 MW
1048 MW
545 kV
JOHN DAY
538 kV
1875 MW
UNIT 1
1334 MW
SKAGIT/ HANFORD NUCLEAR PROJECT
OSTRANDER
515 kV
UNIT 2
1334 MW
548 kV
SLATT
678 MW
544 kV
BADGE

DYNAMIC STABILITY SWING CURVE CASE NUMBER	DESCRIPTION OF DYNAMIC STABILITY TEST	RESULTS
403.3-2	3-Phase fault on the Project's 500 kV bus, clear Project-to-Hanford #2 and Project-to-Ashe 500 kV lines in 8 cycles, continue run for 10 seconds.	Stable
403.3-3	3-Phase fault on the Project's 500 kV bus, clear Project-to-Ashe #1500 kV line and drop Project's unit #2 in 8 cycles, continue run for 10 seconds.	Stable
403.3-4	3-Phase fault on the Project's 500 kV bus, clear Project-to-Hanford #1500 kV line and drop unit #2 in 8 cycles, continue run for 10 seconds.	Stable

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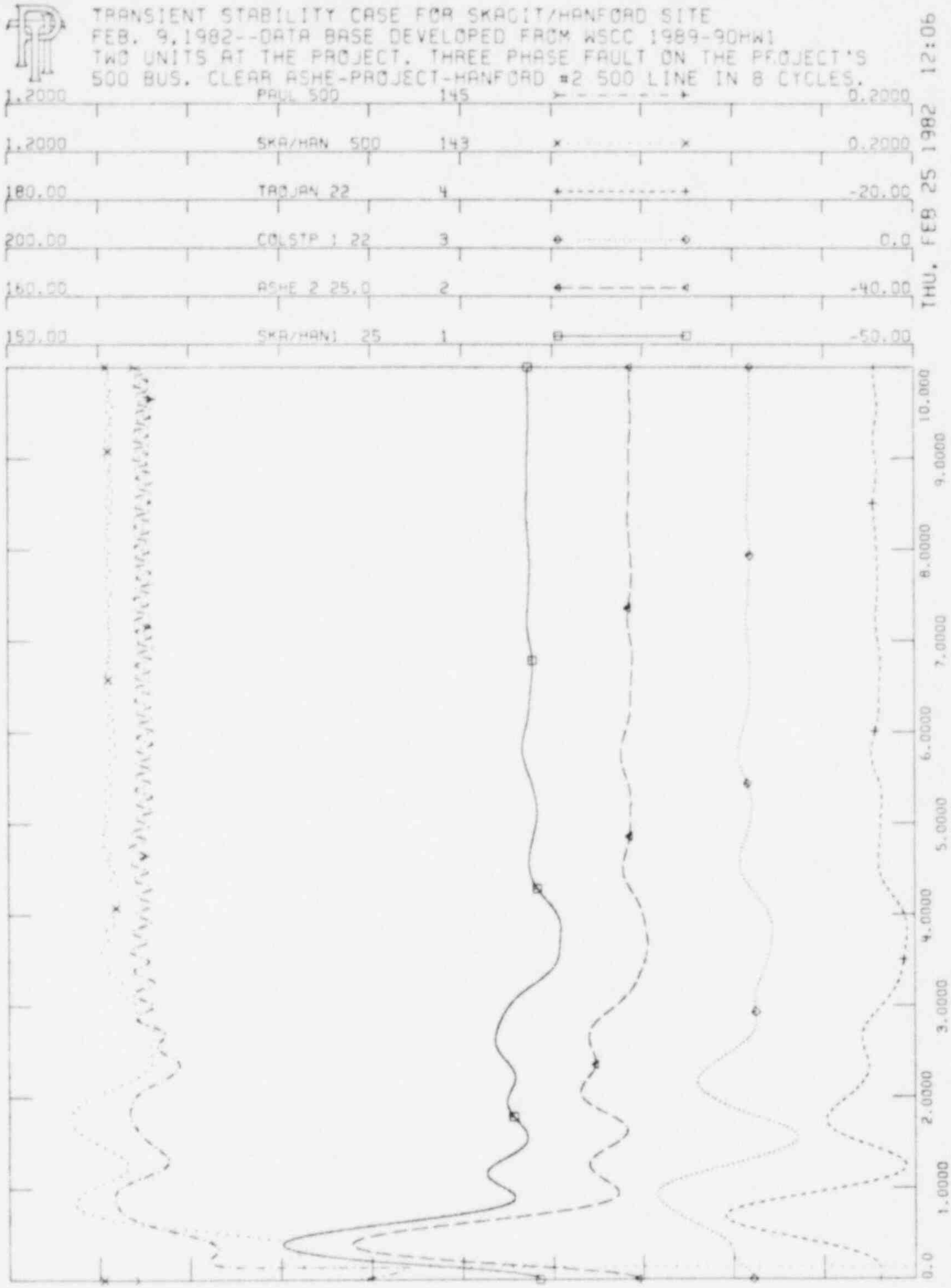
POWER FLOW DIAGRAM AND DYNAMIC STABILITY TEST

FIGURE 403.3-1



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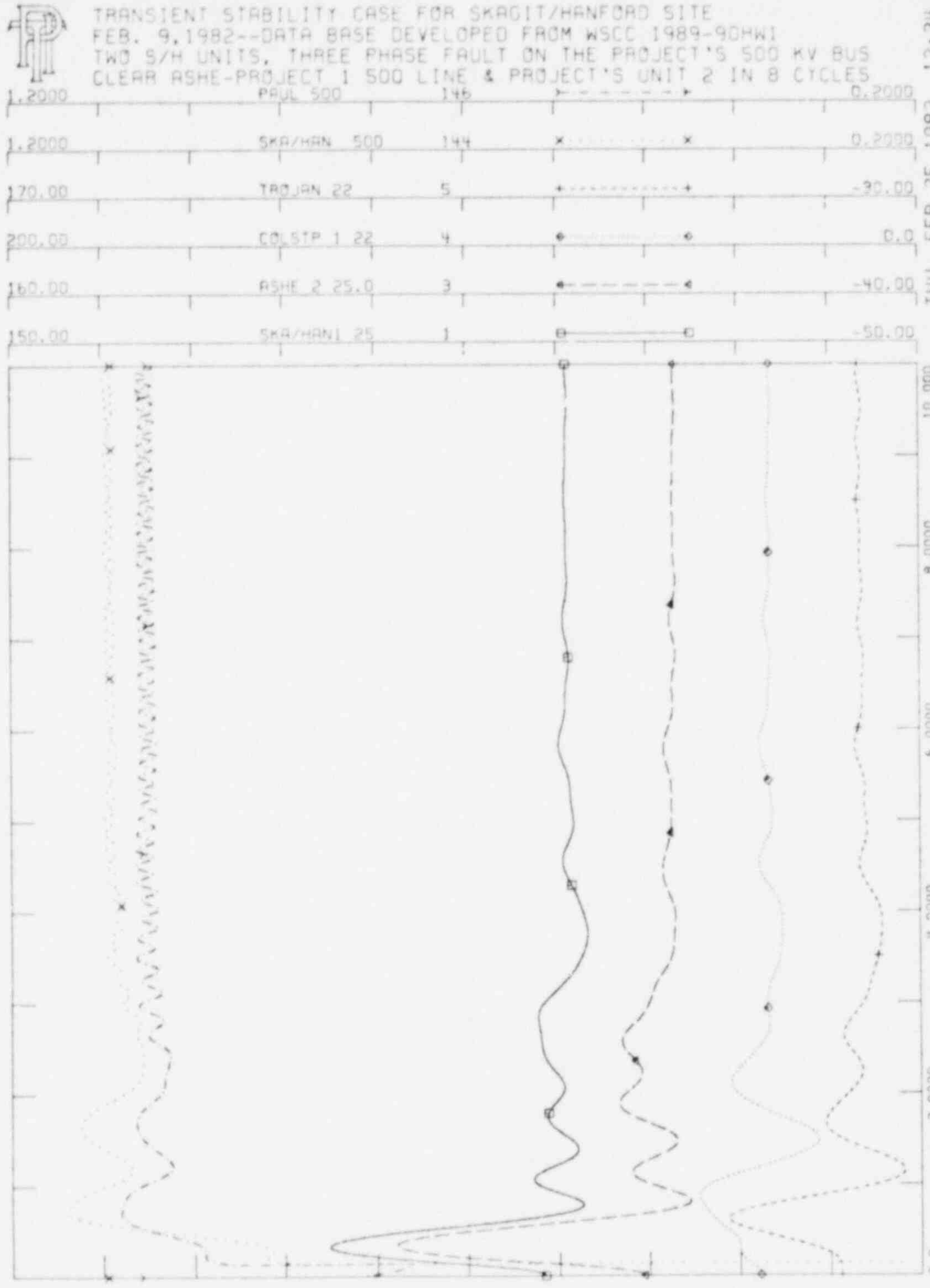
DYNAMIC STABILITY PLOTS

FIGURE 403.3-2



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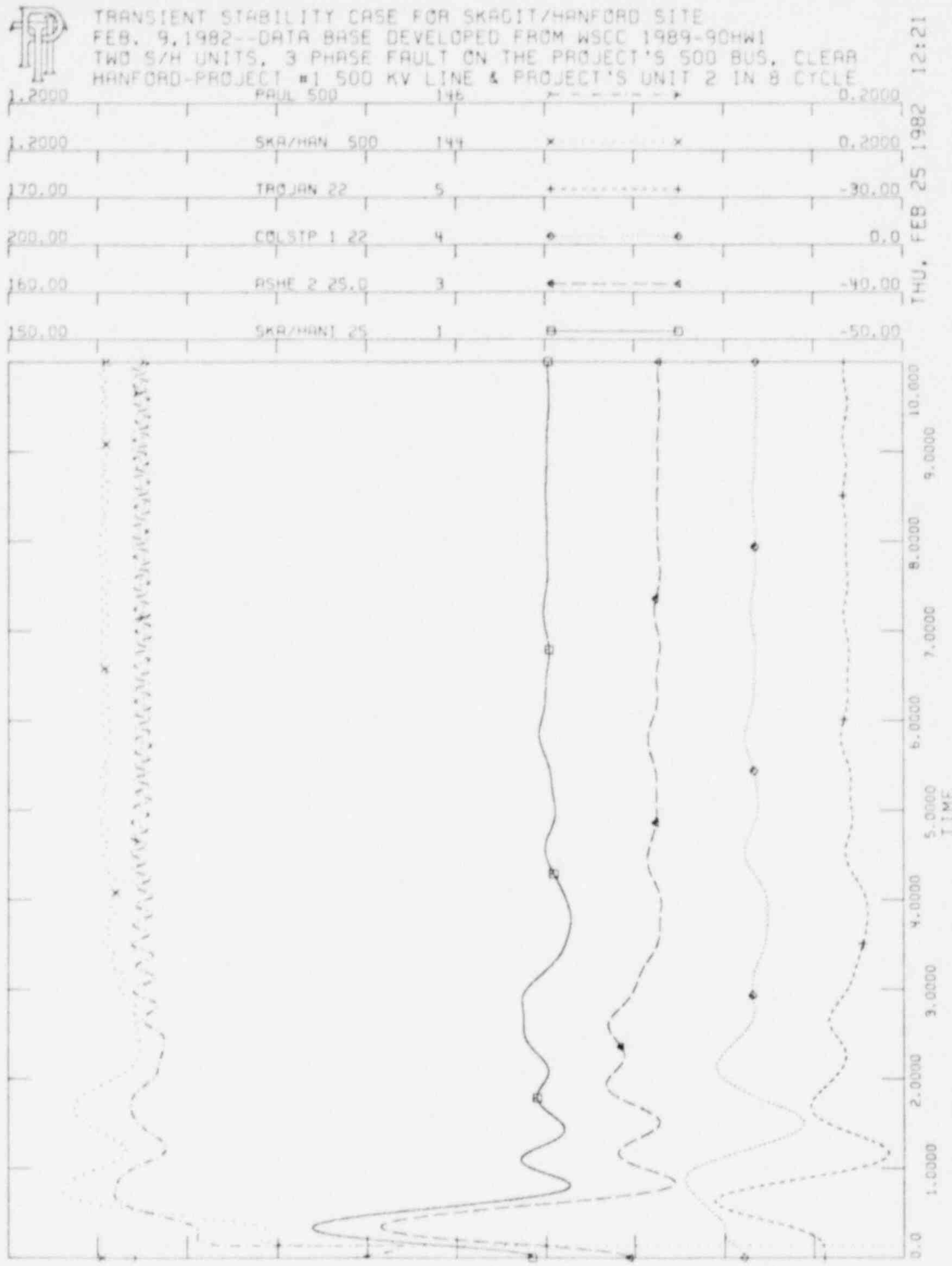
DYNAMIC STABILITY PLOTS

FIGURE 403.3-3



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DYNAMIC STABILITY PLOTS

FIGURE 403.3-4

QUESTION 410.1 (3.4.1 & 3.4.5)

In Amendment 23 to the PSAR you state that flooding of safety-related equipment in the auxiliary building from the turbine building due a rupture of a circulating water system expansion joint will be prevented by operator action following an alarm in the control room which is annunciated by safety-grade detectors located 3 inches above the auxiliary building floor within the airlock between the turbine and auxiliary buildings. In order for us to determine that sufficient time is available for operator action, provide the following information:

a. Provide the elevation of the airlock between the two buildings and revise your plant arrangement drawings to show the airlock.

b. Verify that the airlock is normally closed and discuss any indications and alarms that are provided in the control room.

c. Provide the results of an analysis to show that sufficient time is available for operator action following receipt of a water level detector alarm in the airlock considering the circulating water system flooding rate.

RESPONSE:

(a) The airlock between the Turbine and Auxiliary Buildings is located elevation 527'. Figure 1.2-2 has been updated to show the location of the airlock.

(b) The airlock is a secondary containment boundary and will consist of two normally closed water resistant doors in series which are interlocked such that one door cannot be opened unless the second door is closed. Interlocked operation will be capable of being bypassed under appropriate administrative controls to allow both doors to be open at the same time. An alarm or warning light will be provided to alert control room personnel that the interlock has been bypassed. Loss of integrity of the airlock will be annunciated in the control room.

(c) A time dependent analysis of Turbine Building flooding due to breaks in the Circulating Water (CW) System has been performed. Two separate cases were analyzed:

QUESTION 410.1 (Cont'd)

- o A large CW break resulting in a flood rate equal to runout flow of all operating CW pumps
- o A small CW break.

For the large break case, no operator action is relied upon. The water level in the Turbine Building rises rapidly (approximately 2 to 3 ft/min) until it reaches elevation 531' (4' above the ground floor) at which point the pressure relief panels in the railroad access door open. Approximately 9 minutes into the accident the cooling tower basin is empty and the CW pumps will lose suction. Because of the short duration of this accident and the presence of water resistant doors at the Auxiliary Building airlock and Control Building lobby, water levels in the Auxiliary and Control Buildings will not reach more than a few inches above elevation 527'.

The small break case can be more limiting in that the time for the water to reach elevation 531' in the Turbine Building and open the relief panels can be much longer, thus allowing more time for leakage past the water resistant doors. Because of the broad spectrum of leak rates possible, it was conservatively assumed that the flood reaches elevation 531' undetected, which time is defined as t = 0, and then remains at that level without opening the relief panels.

Because the door sill leading to the Control Building is at elevation 531', no flooding of the Control Building will occur for the small break case. For the Auxiliary and Fuel Buildings the door sills for safety-related rooms are at elevation 528'. The water level is calculated to reach elevation 527' 3" due to leakage past the airlock door at approximately t = 15 min. and elevation 528' at approximately t = 45 min. Thus, 30 minutes will be available for operator action between the time of a flood level alarm (3 inches) and the time when water could enter a safety-related equipment room.

This analysis is considered very conservative because:

QUESTION 410.1 (Cont'd)

- o No credit was taken for floor drain water removal or leakage out of the building through exterior doors.

- o The condensate pump motors, located at elevation 517', would be damaged and go out of service early in the accident causing a loss of feedwater accident leading to reactor scram and a prompt investigation of the cause of the feedwater loss.

- o Flood level was assumed to be at elevation 531' at t = 0 which assumes a long term water level rise from the Turbine Building basement floor elevation 517' to elevation 531' undetected which then suggests that all of the auxiliary equipment located between those elevations in the Turbine Building have gone under water, are out of service and their failure remains unknown to the operators. Such an operating circumstance is, of course, not credible as the turbine generator will have earlier tripped off for multiple reasons.

QUESTION 410.2 (9.2.1)

In Amendment 23 to the PSAR, you revised the design of the Service Water System (SWS) as a result of the site move to Hanford. The proposed new design (Section 9.2.1 of the PSAR) deleted the cross-connection to the Standby Service Water System (SSWS) as reflected on the SWS drawing, PSAR Figure 9.2.1. However, Section 9.2.11 (SSWS) of the PSAR and Figure 9.2.17 still indicate that the cross-connection between the two systems has not changed. From PSAR Section 10.4.5 (circulating water system) it appears that the SSWS will now be supplied water from the circulating water booster pumps during normal plant operations. Revise Section 9.2.11 and Figure 9.2-17 to reflect the cross-connection between the safety-related SSWS and the nonsafety-related circulating water system. Also revise Sections 9.2.2 (reactor component cooling water system), 9.2.9 (essential chilled water system) and 9.1.3 (containment and fuel pools cooling and cleanup system) to indicate the correct normal source of cooling water.

RESPONSE:

See revised Sections 9.2.2, 9.2.9 and 9.2.11, and revised Figures 9.1-1 and 9.2-17. Section 9.1.3 does not require change.

QUESTION 420.1

Summarize changes to the instrumentation and control systems which will be made as a result of the site relocation and confirm that these changes do not involve changes to the safety-related design bases or criteria from that previously submitted by the applicant in the Skagit/Hanford PSAR through Amendment 22. Also confirm that the instrumentation and control system changes resulting from the site relocation do not depend upon advancements in technology beyond the state of the art used for the instrumentation and control systems previously submitted in the PSAR through Amendment 22.

If there are changes to the safety-related design bases or criteria or advancements in technology beyond the state of the art used for the instrumentation and control systems discussed up through Amendment 22 of the PSAR, the changes should be itemized and a discussion provided for each to justify that, with these changes, the General Design Criteria contained in Table 7-1 of the Standard Review Plan (NUREG-0800) and IEEE-279 can be met.

RESPONSE:

PSAR Amendment 23 change to the instrumentation and control systems which will be made as a result of the site relocation is limited to addition of the following: Anhydrous ammonia detection in the control room air intake, and complete and automatic isolation of the control room upon detection of anhydrous ammonia. The ammonia detection and isolation is not safety-related. Failure will not prevent any safety system from performing its safety function with the required minimum redundancy in the safety system. This change in the instrumentation and control systems does not depend upon advancements in technology beyond the state of the art.

QUESTION 421.2 (13.7) (Supplement 19, Response (2)(a))

Confirm that guidance in Regulatory Guide 5.44 will be used in selecting and installing a perimeter intrusion detection system.

RESPONSE:

The guidance contained in Regulatory Guide 5.44 will be incorporated in the design, and used in the installation of the S/HNP perimeter intrusion detection system.

Supplement 19a will be provided under separate cover. This supplement will provide details of the proposed security plan and presents Figures 13.7-1 and 13.7-2. Supplement 19a is to be withheld from public disclosure pursuant to 10 CFR 2.790.

QUESTION 460.1

Figure 11.3-2 of the amended FSAR indicates that the Turbine Building has 3 ventilation exhaust roof vents; Figure 1.2-8 (page 6, 7, and 8 of 9), however, shows eight vents identified as "smoke vents". Are the "smoke vents" in addition to the three ventilation exhaust vents or is this an error?

RESPONSE:

Figure 11.3-2 shows only one vent port over the turbine building roof for HVAC exhaust during normal plant operation. The words "Typ of 3" on Figure 11.3-2 mean one vent port with horizontal air discharge is provided for each of the following non-seismic Category I buildings shown on the drawing:

- o Turbine Building
- o Radwaste Building
- o Service Building

The 8 vent ports shown on Figure 1.2-8 are the smoke and heat vents which will be automatically opened to release the smoke and heat caused by a fire. They are a part of the Fire Protection System and not part of the heating and ventilating system. General arrangement drawings do not normally show H&V intakes, ductwork, or exhaust ports. Since the effluent release points are an extension of the H&V system, they are not shown on Figure 1.2-8.

QUESTION 460.2

Provide the elevation, above grade or mean sea level, of the radioactive gaseous release points shown in Figure 11.3-3. Also indicate the shape and inside dimensions (not area) of each vent.

RESPONSE:

The potential radioactive gaseous effluent release points during normal plant operations are detailed below.

Release Point	Approx. El. Above Grade	Exhaust Duct Size	Release Point Dimen.	Release Vent Config.*
Main Plant Vent	235'	42" Ø	36 " Ø	circular
Fuel Bldg. Roof Vent	95'	30"x30"	4x10"x52"	square annulus
Aux. Bldg. Roof Vent	90'	46"x46"	4x22"x84"	square annulus
Turbine Bldg. Roof Vent	125'	72"x72"	4x48"x108"	rect. louvers
Radwaste Bldg. Roof Vent	50'	56"x56"	4x24"x78"	rect. louvers
Service Bldg. (Admin. Bldg.) Roof Vent	55'	40"x40" 40"x18" 40"x44"	2x28"x106" 2x28"x52"	rect. louvers

*See Figure 11.3-2 for orientation of airflow.

QUESTION 460.3

The response to NUREG-0737, Item II.F.1, Attachment 1, is inadequate in the following respects:

(1) Our examination of the amended PSAR indicates that the Skagit Plant, Unit No. 1, has five gaseous radioactive release vents which could be sources of accident releases of radioactive material. These are the multipurpose vent (vent or reactor building), the fuel building vent, and the rad-waste building vent, the turbine building vent, the auxiliary building vent, the turbine building vents, and the radwaste building vent. Your response on page 1B-94 (Amendment 22) indicates you plan to monitor only the reactor building vent. While this vent would be the release point most likely to contain high concentrations of noble gases after an accident, the other release points cannot be overlooked.

(2) Your response on page 1B-94, Subpart (1), Items A and B, seems to indicate you are providing two accident monitors with ranges differing by a factor of ten but monitoring the same release point. Please clarify why you are using two monitors for this release point when one would apparently suffice.

RESPONSE:

(1) The S/HNP design utilizes two common vents for airborne radioactive materials that may be released from the Plant during and following an accident. These are the Main Plant vent and the Fuel Building vent. The Turbine Building is not connected to the containment and consequently does not contain accident associated airborne radio-active materials subject to release through the Turbine Building vent. The Auxiliary Building vent is isolated under accident conditions and is vented by means of the Standby Gas Treatment System through the Fuel Building vent. Accident level releases, both liquid and gaseous, are isolated and not released to radwaste under accident conditions. Consequently, the radwaste vent is not subject to accident releases of air-borne radioactive materials.

(2) The response on S/HNP PSAR page 1B-94 Subpart (1) indicates the two general categories of noble gas monitors that could be utilized. The Main Plant

QUESTION 460.3 (Cont'd)

vent, which includes the drywell purge, and the Fuel Building vent, which includes the SGTS purge, are both Type B release points.

QUESTION 460.4

The response to NUREG-0737, Item II.F.1, Attachment 2, is inadequate in the following respect:

Your discussion on Page 1B-96 indicates that you plan to sample only the multipurpose vent for radioiodine and particulates from accident releases of radioactive materials. NUREG-0737 requires monitoring for all potential accident release paths, which would include vents from the turbine building, fuel building, auxiliary building, and radwaste building.

RESPONSE:

A post-accident particulate and iodine sampling capability will be installed on the Fuel Building vent and the Main Plant vent. As noted in the response to Question 460.3, the potential accident release paths are only through the Main Plant vent and the Fuel Building vent. The design was found acceptable in S/HNP SER NUREG-0309, Supplement 2, page II-20.

QUESTION 471.1

Section 12.3.1 of Regulatory Guide 1.70 specifies that layout drawings should show shield thickness. In Section 12.1.2.4.7 of the PSAR, the control room shielding is specified to be 2'-0" thick concrete walls, but the roof shield thickness is not specified. The shield thickness above the control room should be specified, either in Subsection 12.1.2.4.7 or in Figure 12.1-16.

RESPONSE:

See revised Section 12.1.6.1.

QUESTION 471.2

Section 12.2.1 of Regulatory Guide 1.70 specifies that applicants should describe the source terms used for shield design for the turbine system. Section 12.1.3.9 states that the source strengths given by Table 12.1-15 are adjusted to reflect the self-absorption in the components and that the equivalent inventory was found to be 117 Ci of N-16, including exposed piping associated with the H.P. Turbine.

471.2(a)

You should specify if the N-16 inventories quoted in Table 12.1-15 are adjusted for component-self absorption, or if they are actual estimated inventories.

RESPONSE:

See revisions to Section 12.1.3.9.

QUESTION 471.2

Section 12.2.1 of Regulatory Guide 1.70 specifies that applicants should describe the source terms used for shield design for the turbine system. Section 12.1.3.9 states that the source strengths given by Table 12.1-15 are adjusted to reflect the self-absorption in the components and that the equivalent inventory was found to be 117 Ci of N-16, including exposed piping associated with the H.P. Turbine.

471.2(b)

In addition, you should explain why only "exposed piping associated with the H.P. Turbine" was used as the N-16 source. It appears that the dose contribution should include a minimum:

H.P. Turbine, (25.37 Ci),
Moisture separators/reheaters (118.25 Ci),
Crossover piping (17.13 Ci), and
L.P. Turbines (19.74 Ci), etc.

Based on Table 12.1-15 the total N-16 source should be at least 180 Ci of N-16. If the above quoted 117 Ci were obtained by adjusting the 180 Ci for the source self-absorption effect, then it should be so stated.

RESPONSE:

See revisions to Section 12.1.3.9.

QUESTION 471.2

Section 12.2.1 of Regulatory Guide 1.70 specifies that applicants should describe the source terms used for shield design for the turbine system. Section 12.1.3.9 states that the source strengths given by Table 12.1-15 are adjusted to reflect the self-absorption in the components and that the equivalent inventory was found to be 117 Ci of N-16, including exposed piping associated with the H.P. Turbine.

471.2(c)

In Table 12.1-15, "N-16 inventories in equipment in the turbine building", reference is made to notes (1), (2), and (3). Note (2) refers to main steam piping. References for Notes (1) and (3) are not specified. You should provide references for Notes (1) and (3).

RESPONSE:

See revisions to Table 12.1-15.

QUESTION 471.2

Section 12.2.1 of Regulatory Guide 1.70 specifies that applicants should describe the source terms used for shield design for the turbine system. Section 12.1.3.9 states that the source strengths given by Table 12.1-15 are adjusted to reflect the self-absorption in the components and that the equivalent inventory was found to be 117 Ci of N-16, including exposed piping associated with the H.P. Turbine.

471.2(d)

In subsection 12.1.3.9, "Turbine Shine Dose," reference is made to the "Exclusion Area Boundry", which is approximately 1.9 miles from the turbine building. At this distance you calculate the dose to be 0.5 mrem/year. Is the exclusion area boundary the closest unrestricted area from radiation control standpoint? If not, you should provide the dose at the closest unrestricted area.

RESPONSE:

See revisions to Section 12.1.3.9.

QUESTION 810.1 (7.5.1)

The description of the on-site first aid and decontamination facilities in Section 7.5.1 of Appendix 13A of the Skagit/Hanford PSAR is stated as being located with the Plant Support Facility for Skagit/Hanford Nuclear Project and the Washington Public Power Supply System approximately 5 miles from the Skagit/Hanford Nuclear Project. Describe in detail the <u>on-site</u> first aid and decontamination facilities for the Skagit/Hanford Nuclear Project.

RESPONSE:

Section 7.5.1 of Appendix 13A will be revised in Amendment 24 to the S/HNP PSAR to also reference Section 12.3.2.7 which provides information about on-Site decontamination facilities. A first aid treatment center, equipped with necessary supplies, is located in the Service Building at Elevation 562, as shown in Figure 12.1-15 and Figure 13 of Appendix 13A. This center is in close proximity to the personnel decontamination facilities and the health physics area. Further description of these facilities will be provided in the FSAR.