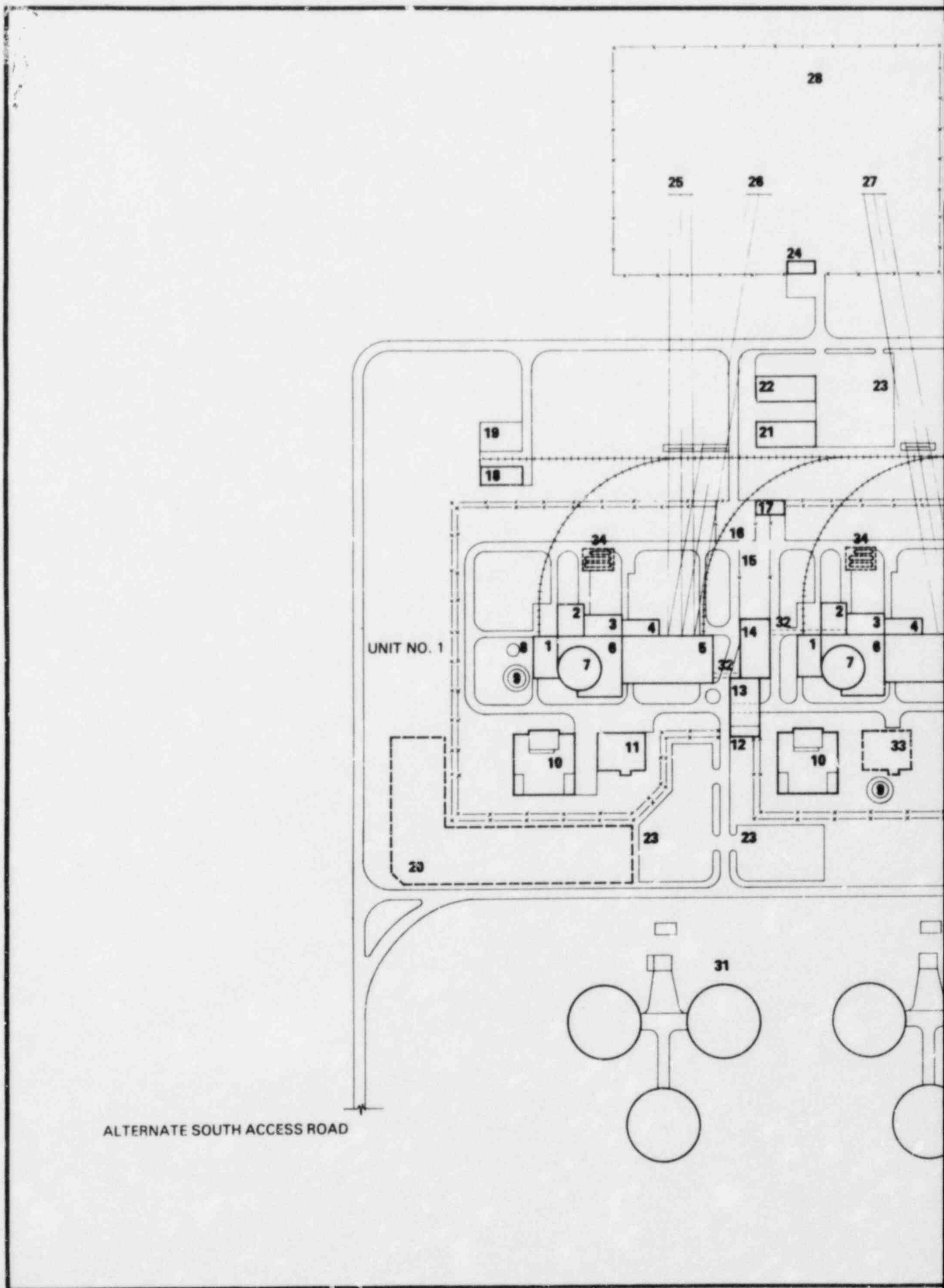
File this instruction sheet in the front of Volume 1 as a record of changes.

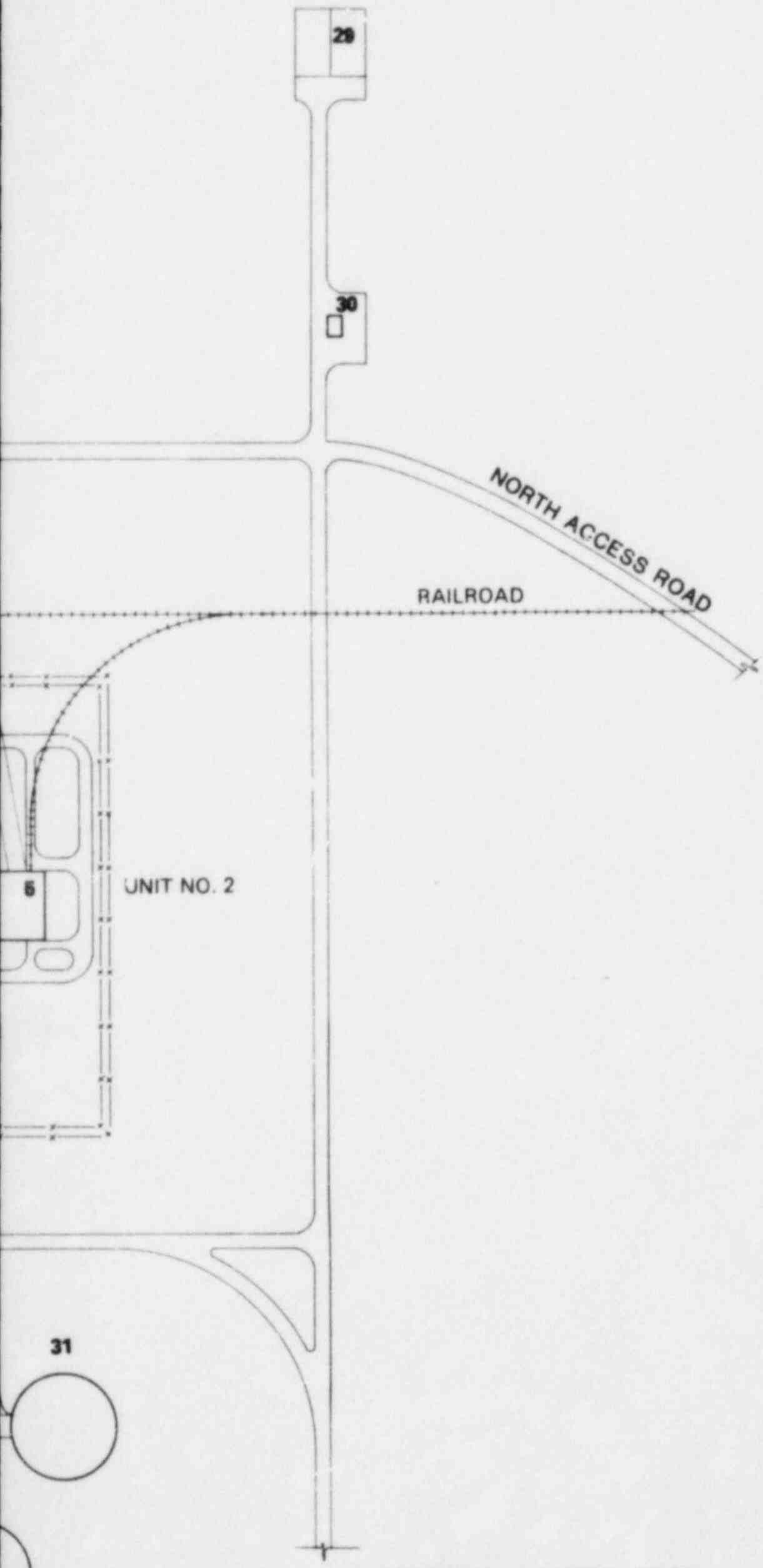
The following information and check list are furnished as a guide for the insertion of new sheets for Amendment 27 into the Preliminary Safety Analysis Report for the Skagit/Hanford Nuclear Project. This material is denoted by use of the amendment date in the upper right-hand corner of the page.

New sheets should be inserted as listed below:

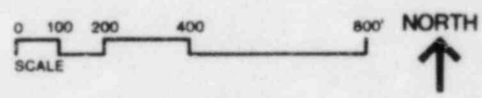
Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)
CHAPTER 1	
Figure 1.2-1	Figure 1.2-1
Table 1.7-1 Sht 7 of 26/ Table 1.7-1 Sht 8 of 26	Table 1.7-1 Sht 7 of 26/ Table 1.7-1 Sht 8 of 26
APPENDIX 1A	
1A-3/1A-4	1A-3/1A-4
Table 15.1.36-2/ Table 3.2.1 (cont'd)	Blank/Table 3.2.1 (cont'd)
15.1.36-8/15.1.36-9	--
CHAPTER 2	
2.1-1/2.1-2	2.1-1/2.1-2
2.1-3/2.1-4	2.1-3/2.1-4
2.1-5/2.1-6	2.1-5/2.1-6
Figure 2.1-2	Figure 2.1-2
Figure 2.1-3 (1 of 2)	Figure 2.1-3 (1 of 2)
Figure 2.1-3 (2 of 2)	Figure 2.1-3 (2 of 2)
2.3-5/2.3-6	2.3-5/2.3-6
Table 2.3-2/Table 2.3-3 Sht 1 of 2	Table 2.3-2/Table 2.3-3 Sht 1 of 2
Figure 2.4-3	Figure 2.4-3
Figure 2.4-17	Figure 2.4-17
Figure 2.4-20	Figure 2.4-20

Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)
Table 15.6-1/Table 15.6-2	Table 15.6-1/Table 15.6-2
Table 15.6-3/Table 15.6-4	Table 15.6-3/Table 15.6-4
Table 15.6-7/Table 15.6-8	Table 15.6-7/Table 15.6-8
Table 15.6-15/Table 15.6-16 through Table 15.6-23/ Table 15.6-24	Table 15.6-15/Table 15.6-16 through Table 15.6-23/ Table 15.6-24
Table 15.6-35/Table 15.6-36	Table 15.6-35/Table 15.6-36
Figure 15.6-1 through Figure 15.6-4	Figure 15.6-1 through Figure 15.6-4
Table 15.7-3/Table 15.7-4	Table 15.7-3/Table 15.7-4
Table 15.7-5/Table 15.7-6	Table 15.7-5/Table 15.7-6
Table 15.7-7/Table 15.7-8	Table 15.7-7/Table 15.7-8
Table 15.7-11/Table 15.7-12	Table 15.7-11/Table 15.7-12
Table 15.7-15/Table 15.7-16	Table 15.7-15/Table 15.7-16
Table 15.7-17/Table 15.7-18	Table 15.7-17/Table 15.7-18
Table 15.7-19/Table 15.7-20	Table 15.7-19/Table 15.7-20


28
25
26
27
24
22
23
21
19
18
17
16
34
15
2
3
4
32
14
UNIT NO. 1
8
1
6
5
7
9
13
12
10
11
33
20
31
ALTERNATE SOUTH ACCESS ROAD



1 FUEL BUILDING
2 DIESEL BUILDING
3 CONTROL BUILDING
4 SWITCHGEAR BUILDING
5 TURBINE BUILDING
6 AUXILIARY BUILDING
7 REACTOR BUILDING
8 REFUELING WATER STORAGE TANK
9 CONDENSATE STORAGE TANK
10 ULTIMATE HEAT SINK
11 RADWASTE BUILDING
12 SOUTH GUARD STATION (below service bldg)
13 SERVICE BUILDING
14 SHOP AND WAREHOUSE
15 WAREHOUSE YARD
16 SALLY PORT
17 NORTH GUARD STATION
18 WATER TREATMENT BUILDING
19 LOW VOLUME WASTE POND
20 TRAINING FACILITY (future)
21 CONSTRUCTION OFFICE
22 CONSTRUCTION WAREHOUSE
23 PARKING
24 CONTROL HOUSE
25 UNIT NO. 1 525 KV LINE OPTION A
26 UNIT NO. 1 525 KV LINE OPTION B
27 UNIT NO. 2 525 KV LINE
28 SUBSTATION
29 PERCOLATION POND
30 SEWAGE TREATMENT PLANT
31 COOLING TOWERS
32 BRIDGE (to power block)
33 RADWASTE BUILDING (future – if required)
34 DIESEL FUEL STORAGE TANKS (underground)



SITE PLAN – UNITS 1 AND 2

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

PROJECT STRUCTURES AND FACILITIES

FIGURE 1.2-1

Term	Definition	Reference
	b. Engineered safety feature system (ESFS) consists of those systems, including essential support systems or components thereof the primary purpose of which during a design basis accident (DBA) will be to: (1) Retain fuel temperatures within design limits by maintaining fuel coolant inventory and temperatures within design limits. (2) Maintain fuel temperatures within design limits by inserting auxiliary negative reactivity. (3) Prevent the escape of radioactive materials to the environment in excess of 10 CFR 100 limits by isolation of the systems or structures. (4) Reduce the quantity of radioactivity available for leakage and its potential for leakage by purification, cleanup, containment heat removal and containment pressure reduction. (5) Control the concentration of combustible gases in the containment systems within established limits.	
Exclusion Area	That area within 1 mile of the line joining the reactor centers as defined by 10 CFR 100.3.	
Failure	The termination of the ability of an item to perform its required function. Failures may be unannounced and not detected until the next test (unannounced failure), or they may be announced and detected by any number of methods at the instant of occurrence (announced failure).	(ii)
Faulted Condition (Limiting Faults)	Those combinations of conditions associated with extremely-low-probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.	(gg)
Forced Shutdown	A forced shutdown is defined as an instance where the Plant is shut down and the reactor cooled to cold shutdown conditions as quickly as possible without violating Technical Specifications requirements or damaging any equipment. A forced shutdown is an unscheduled event.	
Functional Test	The manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (eg, the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).	(mm)
General Design Criteria (GDC)	A set of design criteria for structures, systems, and components important to safety, which are given in Appendix A to 10 CFR 50, and provide reasonable assurance that the Plant can be operated without undue risk to the health and safety of the public.	

27

23

2

Term	Definition	Reference
Heatup	Heatup begins where achieving criticality ends and includes all actions which are normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the turbine generator.	(mm)
High Radiation Area	Any area, accessible to personnel, in which there exists radiation originating in whole or in part within licensed material at such levels that a major portion of the body could receive in any one hour a dose in excess of 100 mrem.	(m)
Hot Functional Testing	This testing is performed prior to loading fuel in reactor. The reactor coolant is raised in temperature to no-load temperature using the heat generated by operation of the recirculation pumps. This condition may be maintained for a considerable period of time (possibly 15 days) while various system controls, instrumentation etc., are checked to ensure their proper operation.	
Hot Safe Shutdown Condition	When reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification 16.3.10 and Tavg is $\geq$212°F.	(a)
Hot Standby Condition	The plant condition in which the coolant temperature is greater than 212°F, system pressure is less than 600 psig, and the mode switch is in startup. The plant condition in which the reactor is sustained at 50-100 percent of rated pressure and a low power level with no electric power being generated. Sufficient control rods are withdrawn to maintain the power level required to hold pressure. If core decay heat is adequate to hold pressure, the reactor may be held below critical but with sufficient control rods withdrawn to minimize the time required to return to power operation.	(hh)
Immediate	Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.	(mm)
Inactive Components	Those components whose operability (eg, valve opening or closing, pump operation or trip) are not relied upon to perform the system function during the transients or events considered in the respective operating condition categories.	(b)
Incident	Any natural or accidental event of infrequent occurrence and its related consequences which affect the Plant operation and require the use of Engineered Safety Feature systems. Such events, which are analyzed independently and are not assumed to occur simultaneously, include the loss-of-coolant accident, steam line ruptures, steam generator tube ruptures, etc. A system blackout may be an isolated occurrence or may be concurrent with any event requiring Engineered Safety Feature systems use.	(n)
Incident Detection Circuitry	Includes those trip systems which are used to sense the occurrence of an incident.	(mm)
Instrument Calibration	An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip.	(mm)

1A.2 Offgas System - Sections

Section | Page

1A.2 Offgas System - Tables

1A.2 Offgas System - Figures

This page intenticnally
left blank 27

TABLE 3.2.1 (Continued)

Principal Component[a]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Assurance Requirement[f]	Seismic Category[g]	Comments
XXIV. Offgas System						
1. Tanks	Other	T	D	N/A	N/A	(v)
2. Heat exchangers	Other	T	D	N/A	N/A	(v)
3. Piping	Other	T	D	N/A	N/A	(v),(m),(q)
4. Pumps	Other	T	D	N/A	N/A	(v),(q)
5. Valves, flow control	Other	T	D	N/A	N/A	(v),(q)
6. Valves, other	Other	T	D	N/A	N/A	(v),(m),(q)
7. Mechanical modules, with safety function	Other	T,A	D	N/A	N/A	(v),(m) (q)
8. Pressure vessels	Other	T,A	D	N/A	N/A	(v)

090374

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION AND DESCRIPTION

2.1.1.1 Location

The Skagit/Hanford Nuclear Project (S/HNP) Site is located in the southeast area of the U.S. Department of Energy's (DOE) Hanford Reservation in Benton County, Washington. The S/HNP Site is approximately 5 miles west of the Washington Public Power Suppl System's (Supply System) Nuclear Project No. 2 (WNP-2, ıit. It is approximately 8 miles west of the Columbia Riv r, 7 miles north of the Yakima River at Horn Rapids Dam, and 12 miles northwest of the City of North Richland. Figures 2.1-1 and 2.1-2 show the S/HNP location with respect to roads, highways, rivers, and population centers within the Site Region and Site Area.

The following table lists the approximate geographical coordinates for the reactor containment structure centroids: 23

Unit	Latitude and Longitude	Universal Transverse Mercator	Lambert Coordinates (State of Washington) (ft)
1	46° 29' 15" 119° 26' 4"	N 5150900 m E 313200 m	N 422710 E 2268390
2	46° 29' 15" 119° 25' 51"	N 5150900 m E 313400 m	N 422710 E 2269290

2.1.1.2 Site Area

Figure 2.1-2 shows the S/HNP Site and its topographic features, and the location and orientation of the principal Plant structures. No public roads or railroads cross the Site.

The S/HNP land requirements consist of the Site and Associated Areas. The major Project facilities will be located on the Site, and other supporting facilities (e.g., transmission lines, intake and discharge pipeline, railroad and access roads) will be located on the Associated Areas.

The Site and Associated Areas are depicted in Figure 2.1-3 and described as follows:

The Site will consist of 1200 acres. Title will be acquired to 640 acres (the owned land) and easements will be obtained for the remaining 560 acres (the easement area). Owned land will be comprised of Section 33 of Township 12 North, Range 27 East of the Willamette Meridian. The easement area will be the south half of Section 28, the west quarter of Section 34 and the west half of the southwest quarter of Section 27 of Township 12 North, Range 27 East of the Willamette Meridian.

The Associated Area will be made up of the following easements and totaling approximately 420 acres on land 23
outside of the Site:

Facility	Easement Width	Estimated Acres Outside Site
1. Intake and discharge pipelines	150 feet (200 feet at pump-house)	134
2. Railroad	100 feet	42
3. Transmission Lines	600 feet	192
4. Access Roads		
a. North	100 feet	19
b. South*	100 feet	17

27

*An alternative access route totaling 33 acres, identified as South Alternative Access Road in Figure 2.1-3, is being considered.

Figure 2.1-3 shows the centerlines for the preliminary corridors (each 1,000 feet wide) in which the final respective easement routes will be selected. A legal 23
description and final area for each easement will be provided after selection of the final routes.

The raw water pumphouse will be located near the west bank of the Columbia River, approximately 75 feet downstream of River Mile 361.5.

Figure 2.1-2 shows the Site Boundary lines and the Plant exclusion area boundary. The Site Boundary, the Plant 23
property lines, and the restricted area boundary are the same. The S/HNP exclusion area boundary encloses an area within 1 mile of the line joining the reactor centers. 27

2.1.1.3 Boundary for Establishing Effluent Release Limits

The boundary for establishing effluent release limits, in conformance with the restricted area as defined by 10 CFR 20, coincides with the Site Boundary (refer to Figure 2.1-2). Table 2.1-1 lists the minimum distances to the Site Boundary from the effluent release points (center of each containment). For purposes of radiation protection and general safety, the area inside the Site Boundary will be under the control of Puget.

The Site Boundary will be fenced. As described in Section 2.1.3, there are no permanent residences or significant numbers of transients within the exclusion area. Vehicles will be able to access the restricted area via two roads that pass through normally open gates at the Site Boundary. If it becomes necessary to prohibit vehicle entry, the gates will be closed and monitored by a guard.

23

2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL

2.1.2.1 Authority

All of the land within the exclusion area is, at present, owned by the United States of America and managed by the Department of Energy as part of the Hanford Reservation. Puget is currently negotiating with the Department of Energy to acquire the legal rights necessary to use the Site for the Project and those necessary to determine all activities within the exclusion area, as required by 10 CFR 100.3(a).

Puget expects to acquire title to 640 acres (the owned land) of the 1200 acre Site and to acquire appropriate easements over the remaining 560 acres (the easement area) of the Site. The owned land, the land being purchased by Puget, is Section 33 of the Township 12 North, Range 27 East of the Willamette Meridian. The easement area is the remainder of the Site described in Section 2.1.1.2.

Puget's use of the owned land will be restricted to the construction and operation of nuclear electric generating facilities. Upon completion of the use of the owned land for these purposes, title to the owned land will revert to the Government. The Government will retain all mineral rights upon or in the owned land, but will agree not to exercise those rights so long as title to the owned land remains vested in Puget.

Except for the Substation, all S/HNP structures to be located on the Site will be located on the owned land. The Substation will be located on the easement area.

The easements to be acquired by Puget over the easement area will include an easement for an access-control perimeter fence, thus permitting Puget to fence the Site boundary and control access to the entire Site, as discussed in Section 2.1.1.3.

In conjunction with purchase of the owned land, Puget expects to acquire from the Government the authority to determine all activities within the exclusion area consistent with the meaning of 10 CFR 100.3(a), including the authority to remove all personnel and property from the area. Puget will agree to exercise this authority in a manner so as not to preclude the Government from undertaking any action or activity within the exclusion area that is permissible under the provisions of 10 CFR 100.3(a). The Government will retain all mineral rights upon or in the exclusion area, but any exercise of these rights will be subject to Puget's above described authority to control all activities within the exclusion area. 23

There are no easements of record within the exclusion area.

2.1.2.2 Control of Activities Unrelated to Plant Operation

There are no activities unrelated to S/HNP operation within the exclusion area. 27

2.1.2.3 Arrangements for Traffic Control

No public roads, railroads, or waterways traverse the exclusion area. The S/HNP access roads and railroad (Figure 2.1-2) will be located on easements to be granted to Puget by the Government. Puget will have the authority to control travel on these facilities within the exclusion 23 area. In the event that evacuation or other control of the exclusion area should become necessary, appropriate notice will be given to the DOE-Richland Operations Office for control of non-Puget related activities. 27

2.1.2.4 Abandonment or Relocations of Roads

There are no public roads traversing the S/HNP Site.

2.1.3 POPULATION DISTRIBUTION

All population estimates and projections were calculated with the centroid of the S/HNP reactors as the geographic reference. For the analysis of the population within 10 miles, a house count was conducted in October, 1981. For the estimate of the population between 10 and 50 miles, data from the 1980 U.S. Census were analyzed for blocks, tracts, and enumeration districts (Ref 1).

Population projections from 1990 through 2030 were based on county forecasts for the states of Washington and Oregon 23
(Refs 2, 3). For the census years 1990 and 2000, existing projections were directly employed. For the years 2010, 2020 and 2030, projections were made following a logic similar to that of the U.S. Census projections through 2030 for the nation as a whole (Ref 4). It was assumed that after the year 2000, stabilization of population growth will gradually occur within a 50-mile radius of the Site. For each of the census years from 2010 through 2030, it was estimated that the rate of population increase in each county would decline by one-half of the rate prevalent in the previous decade. For each county within 50 miles, this procedure results in a stabilized population by the year 2030.

Distribution of population growth was assumed to be equal throughout each county, with the major exceptions of the Benton and Franklin metropolitan counties which are nearest to the Site. Based on interviews with local planners and city officials and review of land use and annexation plans, a number of areas within the Tri-Cities were identified as having high growth potential. These areas include the Horn Rapids Triangle in Richland, the Horn-Willamette area in West Richland and northwest Pasco near the new I-182 bridge. Accordingly, for the period 1980-2000 appropriate enumeration districts and census blocks were projected to grow at approximately twice the rate of the remainder of the metropolitan area. Several areas were projected to sustain growth from 2000-2010 but after 2010 all areas were projected to stabilize and generally parallel the overall metropolitan area growth rates.

2.1.3.1 Population Within Ten Miles

Figure 2.1-4 shows the estimated 1980 population within a 10-mile radius of the Site for each compass sector at distances of 1, 2, 3, 4, 5 and 10 miles. As these data indicate, there are no residences within 5 miles. An October, 1981 house count determined the nearest residence to be approximately 7.5 miles from the Site. Based on average household data for the area, it is estimated that 357 people reside within 10 miles of the Site - all in southerly to easterly directions. These 357 residents represent about .13 percent of the approximately 280,000 residents in the 50-mile radius.

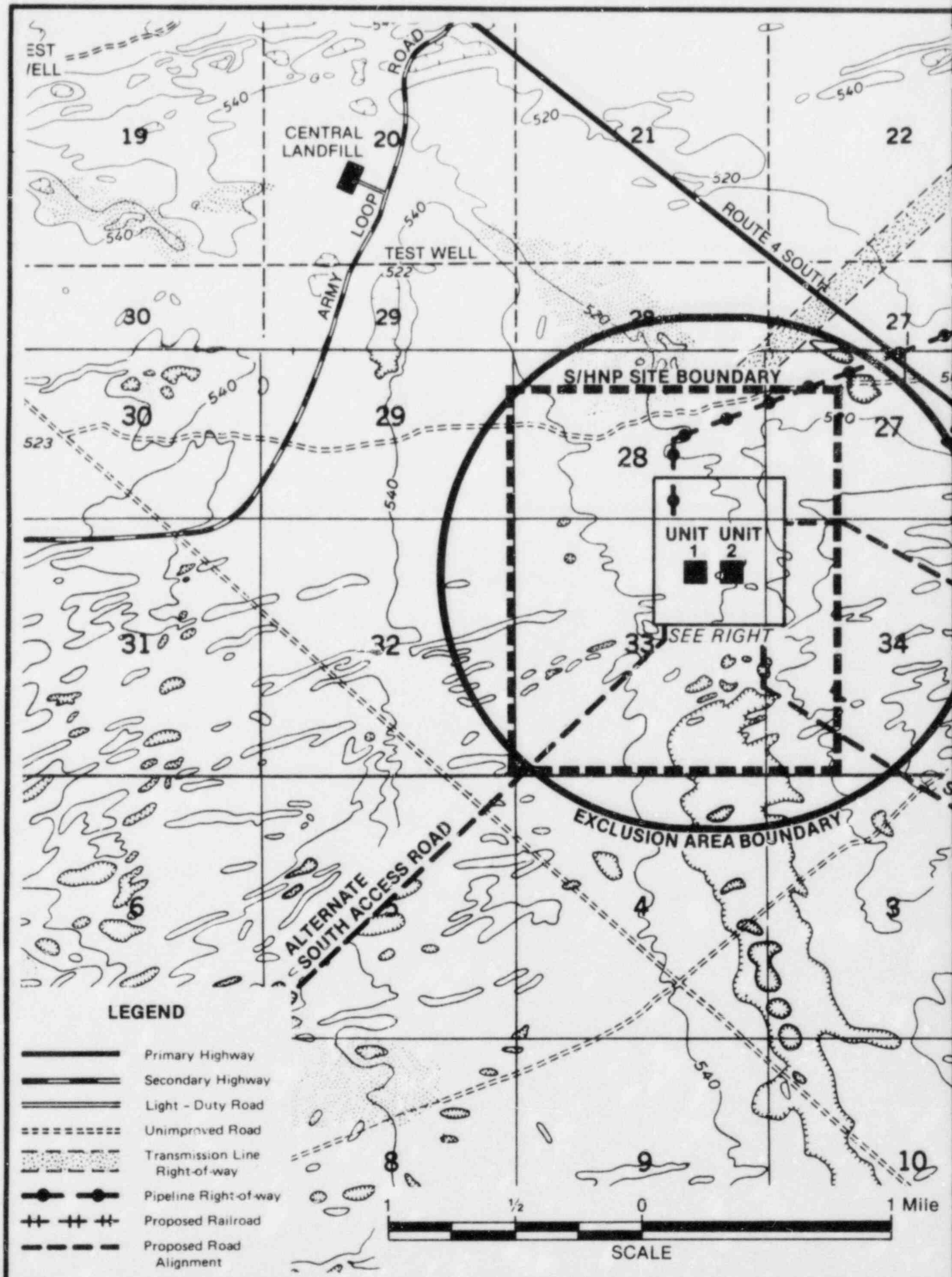
Projected population within 10 miles of the Site for the census years 1990-2030 are shown in Figures 2.1-5 through 2.1-9. As these data indicate, projections are that the population within the 10 mile radius will be 513 in 1990, and 639 in 2000, 683 in 2010, and 691 in 2020. By 2030, the population within ten miles is estimated at 691, which is a 93.5 percent increase over 1980. 23

No major land use changes are projected for the Hanford Reservation and population growth is expected to be concentrated in areas which actually had residents in 1980.

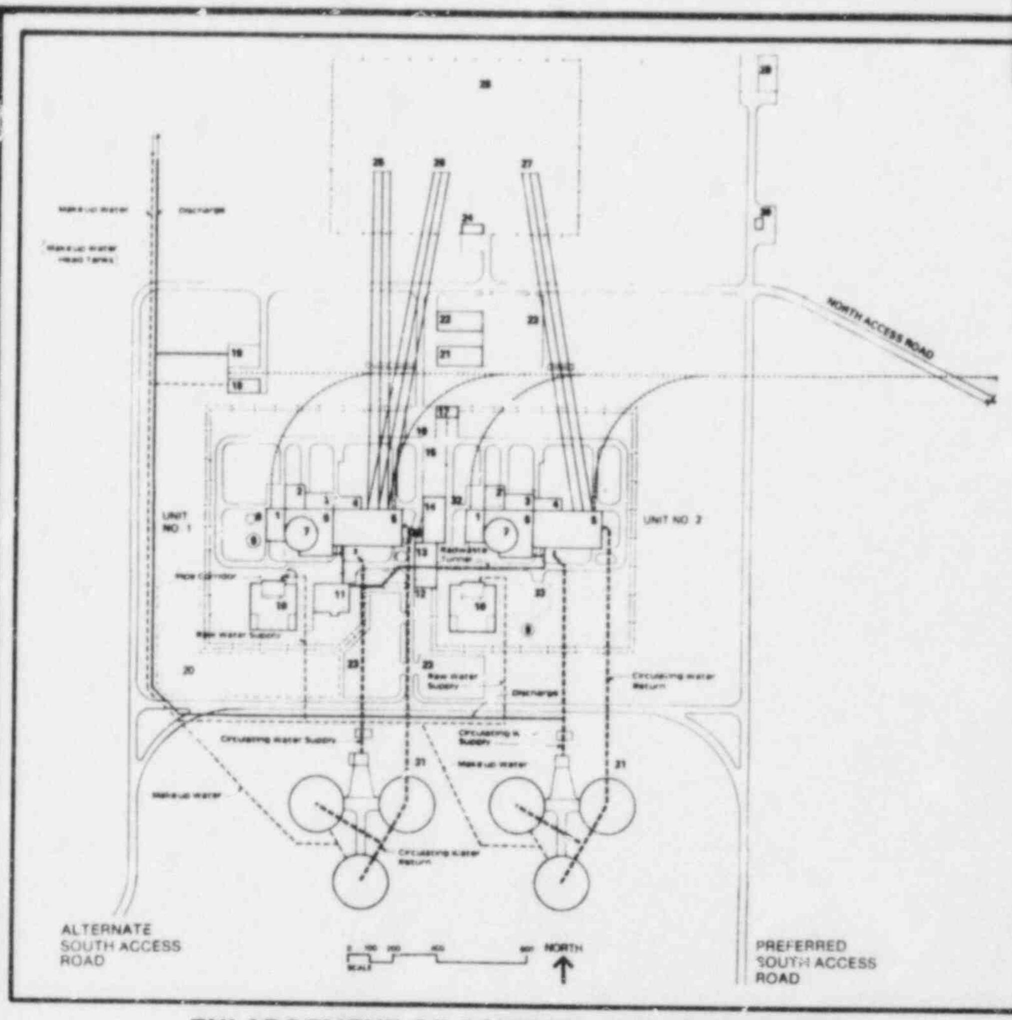
The projected age distribution of the population at the midpoint of S/HNP operating life (2010) is presented in Table 2.1-2. These data are calculated using a cohort survival method which utilized the State of Washington county estimates of age and sex distribution in 2000 as the base.

2.1.3.2 Population Between 10 and 50 Miles

Figure 2.1-10 shows estimates of the number of persons (N = 278,871) residing within the 10-50 mile radius of the Site in 1980. As these data indicate, the bulk of the population within 30 miles is concentrated in the Tri-Cities metropolitan areas in the SE and SSE directions from the S/HNP.



EST WELL
CENTRAL LANDFILL
ARMY LOOP ROAD
TEST WELL
522
ROUTE 4 SOUTH
540
520
523
19
20
21
22
30
29
28
27
31
32
33
34
6
5
4
3
8
9
10
S/HNP SITE BOUNDARY
UNIT 1
UNIT 2
SEE RIGHT
EXCLUSION AREA BOUNDARY
ALTERNATE SOUTH ACCESS ROAD
LEGEND
Primary Highway
Secondary Highway
Light - Duty Road
Unimproved Road
Transmission Line Right-of-way
Pipeline Right-of-way
Proposed Railroad
Proposed Road Alignment
1
½
0
1 Mile
SCALE



ENLARGEMENT OF POWERPLANT STRUCTURES

POWER PLANT STRUCTURE LEGEND

1 FUEL BUILDING
2 DIESEL BUILDING
3 CONTROL BUILDING
4 SWITCHGEAR BUILDING
5 TURBINE BUILDING
6 AUXILIARY BUILDING
7 REACTOR BUILDING
8 REFUELING WATER STORAGE TANK
9 CONDENSATE STORAGE TANK
10 ULTIMATE HEAT SINK
11 RADWASTE BUILDING
12 SOUTH GUARD STATION (below service bldg)
13 SERVICE BUILDING
14 SHOP AND WAREHOUSE
15 WAREHOUSE YARD
16 SALLY PORT
17 NORTH GUARD STATION
18 WATER TREATMENT BUILDING
19 LOW VOLUME WASTE POND
20 TRAINING FACILITY (future)
21 CONSTRUCTION OFFICE
22 CONSTRUCTION WAREHOUSE
23 PARKING (permanent plant)
24 CONTROL HOUSE
25 UNIT NO. 1 525 kV LINE OPTION A
26 UNIT NO. 1 525 kV LINE OPTION B
27 UNIT NO. 2 525 kV LINE
28 SUBSTATION
29 PERCOLATION POND
30 SEWAGE TREATMENT PLANT
31 COOLING TOWERS & PUMP STRUCTURE
32 BRIDGE (to power block)
33 RADWASTE BUILDING (future if required)

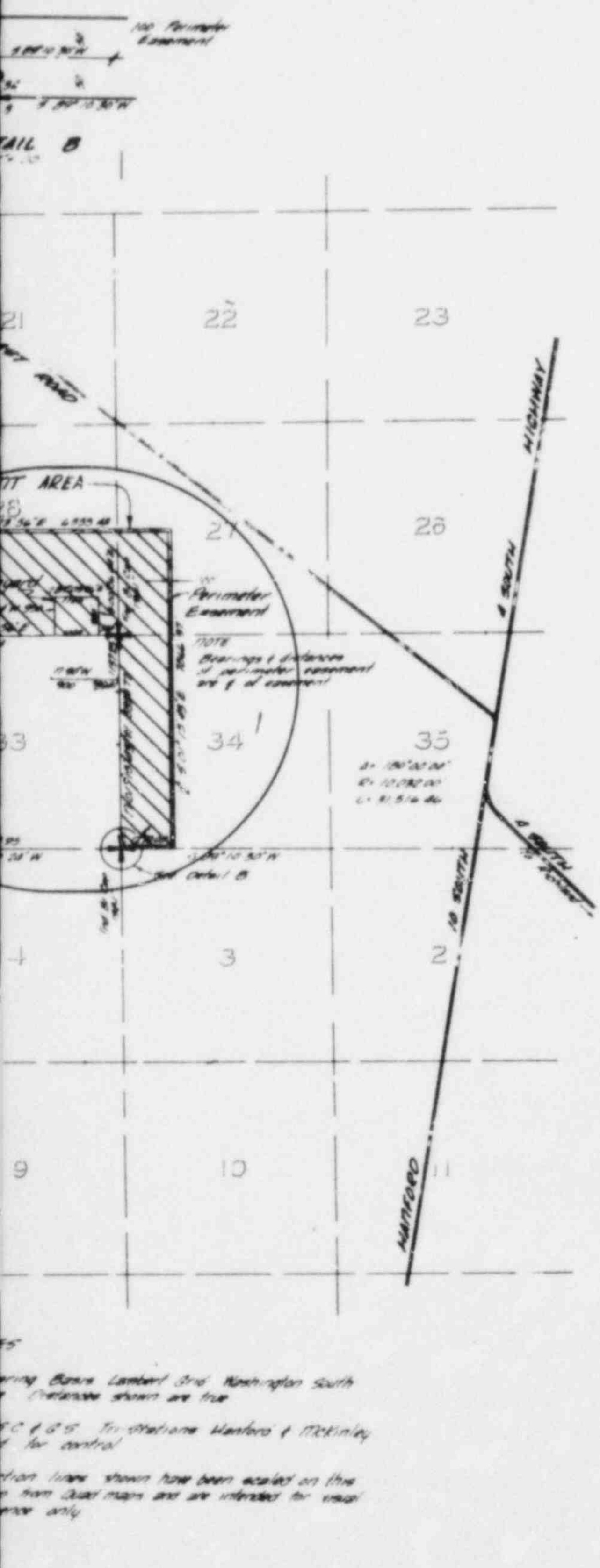
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT

SITE AREA MAP

FIGURE 2.1-2

S

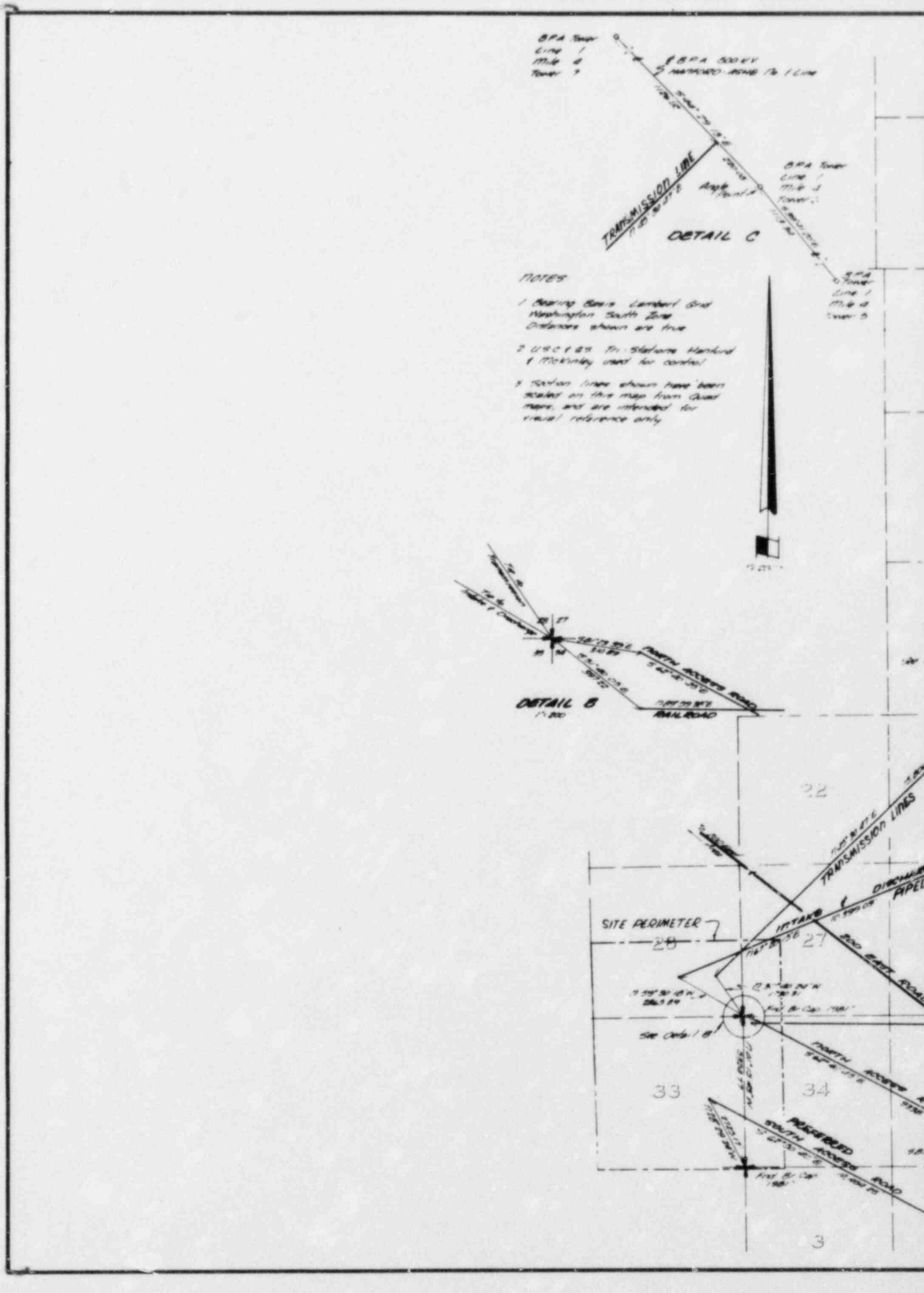
DETAIL A
100' Perimeter Easement
DETAIL C
EASEMENT
Exclusion Area Boundary
OWNED AREA
ALTERNATE SOUTH ACCESS ROAD
SOUTHWEST ACCESS
S.R. 240
1" = 2000'
19
20
30
29
31
32
6
5
12
7
8
13
18
NOTE S.R. 240 stationing obtained from State Highway Commission

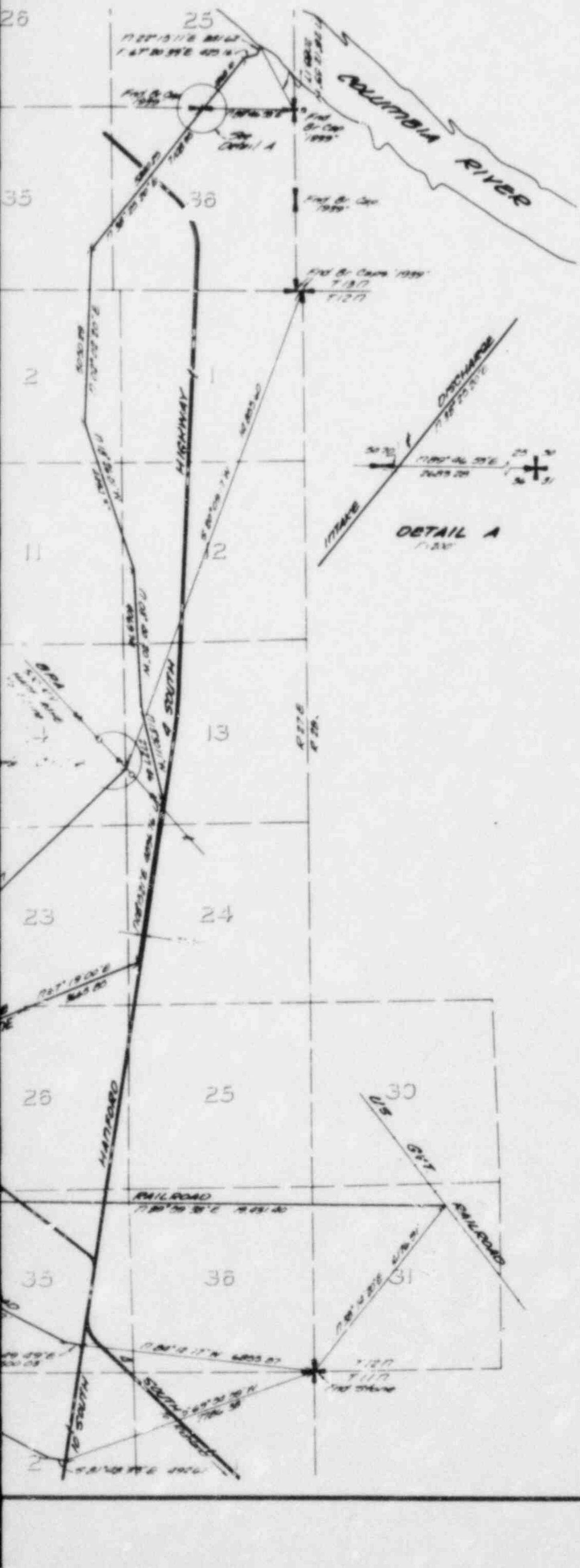


PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

CENTERLINE LOCATIONS OF CORRIDORS FOR RIGHTS-OF-WAY

FIGURE 2.1-3 (1 OF 2)


BPA Tower Line 1 Mile 4 Tower 7
BPA 500 KV
HANFORD-ASHE No. 1 Line
BPA Tower Line 1 Mile 4 Tower 3
Angle Point
TRANSMISSION LINE
DETAIL C
BPA Tower Line 1 Mile 4 Tower 5
NOTES
1 Bearing Basis Lambert Grid Washington South Zone Distances shown are true
2 USC & GS Tri-Stations Hanford & McKinley used for control
3 Section lines shown have been scaled on this map from Quad maps, and are intended for visual reference only
NORTH ACCESS ROAD
DETAIL B
RAILROAD
22
TRANSMISSION LINES
INTAKE
PIPELI
SITE PERIMETER
28
27
200 EAST ROAD
See Detail B
NORTH ACCESS RO
33
34
PREFERRED
SOUTH ACCESS ROAD
3



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

CENTERLINE LOCATIONS OF CORRIDORS FOR RIGHTS-OF-WAY

FIGURE 2.1-3 (2 OF 2)

For all other meteorological conditions (ie, unstable A, B,
or C atmospheric stability and/or 10 meter wind speeds of 6
m/s or greater), plume meander was not considered. The
appropriate χ/Q value was chosen as the higher value
calculated from Equation 2.3-1 or 2.3-2. 23

2.3.4.2 Determination of Conservative χ/Q Values

Cumulative probability distributions of χ/Q values were
determined for each of the 16 wind sectors for the
Exclusion Area Boundary (EAB) (1609m) and Low Population 27
Zone (LPZ) (6437m) distances. The distributions were
structured in terms of probabilities (relative to total
hours in all sectors) of given χ/Q values being exceeded in
a given sector. The conservative estimate was determined by
selecting the χ/Q values which are exceeded not more than
0.5 percent of the time. The χ/Q values thus determined
are applicable for release durations less than or equal to
two hours. The annual average value was calculated for
ground-level release in accordance with methodology
described in Regulatory Guide 1.111, Rev. 1 (Ref 4).
Values for periods of 8 hours, 16 hours, 3 days (72 hours),
and 26 days (624 hours) were obtained by a logarithmic
interpolation between the 2-hour value and the annual
average in the same sector. The maximum-value sector for
each time period becomes the controlling χ/Q value.

However, a direction-independent conservative estimate was
used as an additional constraint on the controlling χ/Q
value for the conservative accident assessment. An overall 23
5th percentile χ/Q was determined from a direction-
independent probability distribution. This overall 5th
percentile value was calculated at the EAB and LPZ
distances and compared to the direction-dependent
conservative estimates. If the overall 5th percentile
value (for a given time period) was greater than the
maximum direction-dependent value, then the direction-
independent value would be used for the accident
assessment.

2.3.4.3 Input Meteorological Data

Input meteorological data consisted of joint frequency
distributions (JFDs) of hourly averages of wind speed and
wind direction by stability class. For computer modeling
purposes, twelve wind speed groups were used to give good
resolution at lower wind speeds (Ref 5). The annual JFD
with the standard 7 wind speed groups is shown in Table

2.3-1. The JFD's were based on two years of data collected nearby at WNP-2. Occurrences of calms and variable wind directions were distributed by direction and stability class to the lowest wind speed group of the JFD's. Calms were assigned a speed one-half of the threshold speed of the wind vane. Winds were based on observations at 33 ft and stability class on observations of delta T (245-33 ft) 23
per Regulatory Guide 1.23 (Ref 2).

2.3.4.4 Short Term Dispersion Estimates

The short-term (χ/Q) values are presented by accident
period in Table 2.3-2 for the EAB distance of 1 mile and 27
the LPZ distance of 4 miles during the course of a 23
hypothetical accident. The 0-2 hour value at the EAB is
1.5×10^{-4} sec/m^3; the sector associated with this value is 27
to the SSE of the Plant. The sector of maximum χ/Qs stays the same for the duration of the accident (30 days).

2.3.5 LONG-TERM ATMOSPHERIC DISPERSION MODEL

2.3.5.1 Dispersion Model

Dispersion factors (χ/Q) were determined using the methodology presented in Regulatory Guide 1.111 (Ref 4) and the NRC computer code XOQDOQ (Ref 6).

The calculations were made for the Site Boundary and at the 23
standard distances discussed in Regulatory Guide 1.70 (Ref 7). All releases were assumed to be at ground level.

χ/Q values were determined by:

$$(\chi/Q)_D = \frac{2.032}{x} \sum_{ij} \frac{n_{ij}}{N \Sigma_{zj} \bar{u}_{ij}} \qquad (2.3.5\text{-}1)$$

where

$(\chi/Q)_D$ = the average effluent concentration, χ, normalized by source strength, Q, at a downwind distance, x, for a given direction, D (sec/m^3)

TABLE 2.3-2

CONSERVATIVE χ/Q VALUES FOR SHORT-TERM (ACCIDENT)
ASSESSMENT AT S/HNP 23

Accident Period	Distance (m)	Maximum Sector χ/Q (sec/m^3)	
2 hours	1609(EAB)	1.5E-4(SSE)	27
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: 23

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

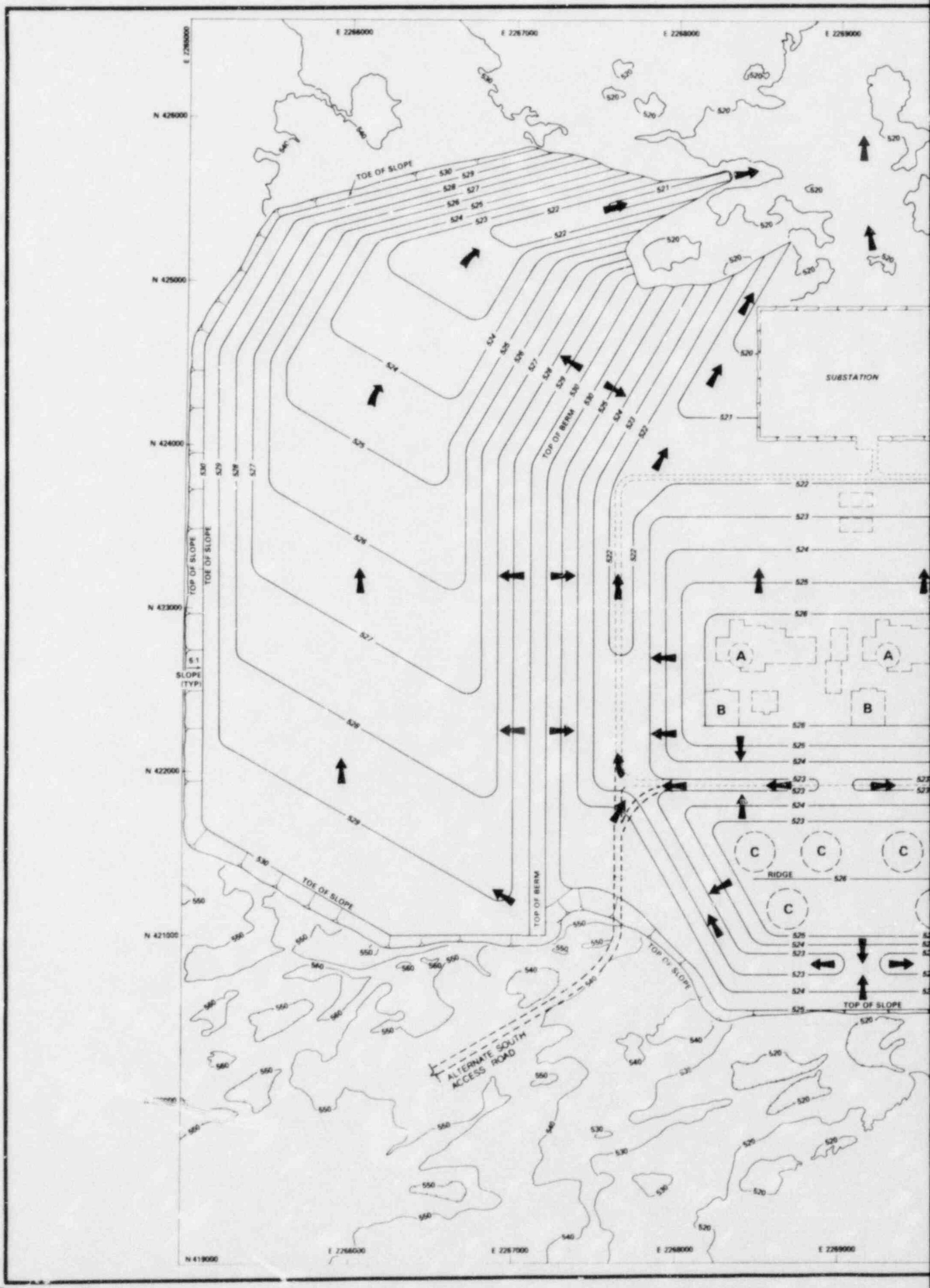
23

NOTES:

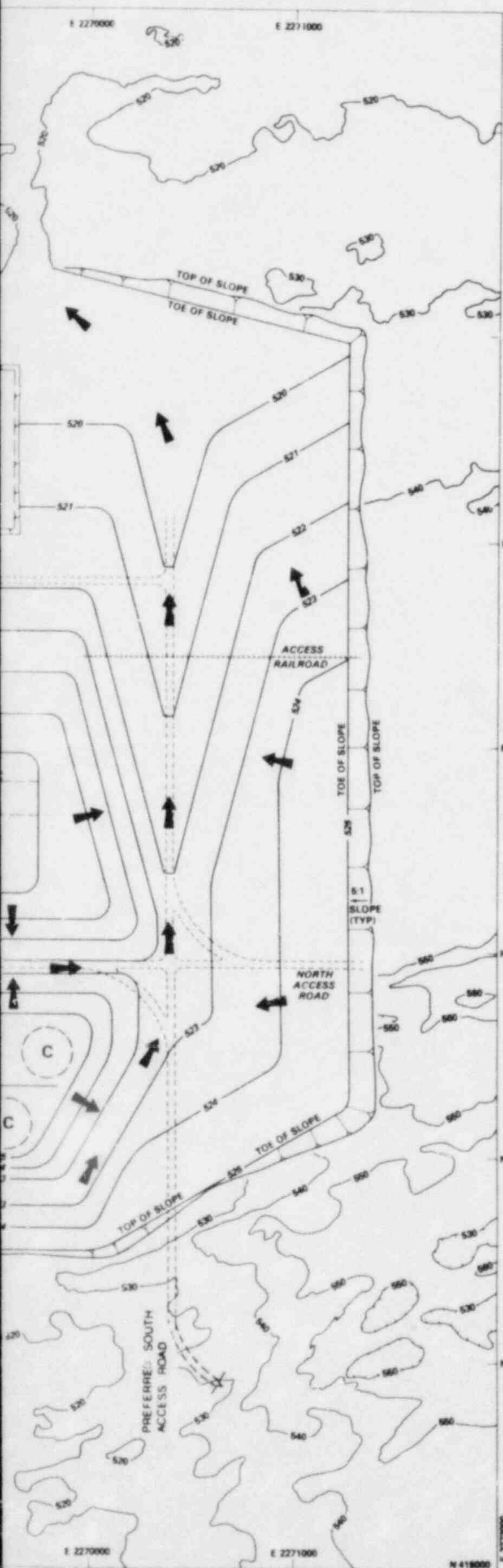
1. Relative concentrations are for a ground-level release to a ground-level receptor, are undepleted and undecayed, and incorporate Pasquill-Gifford dispersion coefficients, building height wake, and open terrain correction factors.

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.



E 2266000
E 2267000
E 2268000
E 2269000
N 426000
N 425000
N 424000
N 423000
N 422000
N 421000
N 419000
TOE OF SLOPE
TOP OF SLOPE
TOP OF BERM
SUBSTATION
5:1
SLOPE
(TYP)
RIDGE
ALTERNATE SOUTH
ACCESS ROAD
A
B
C



LEGEND:

DRAINAGE DIRECTION

A. REACTOR BUILDING
B. ULTIMATE HEAT SINK
C. COOLING TOWERS

Ref: Bechtel Drawing
H-SK-C-10 Rev. B

N

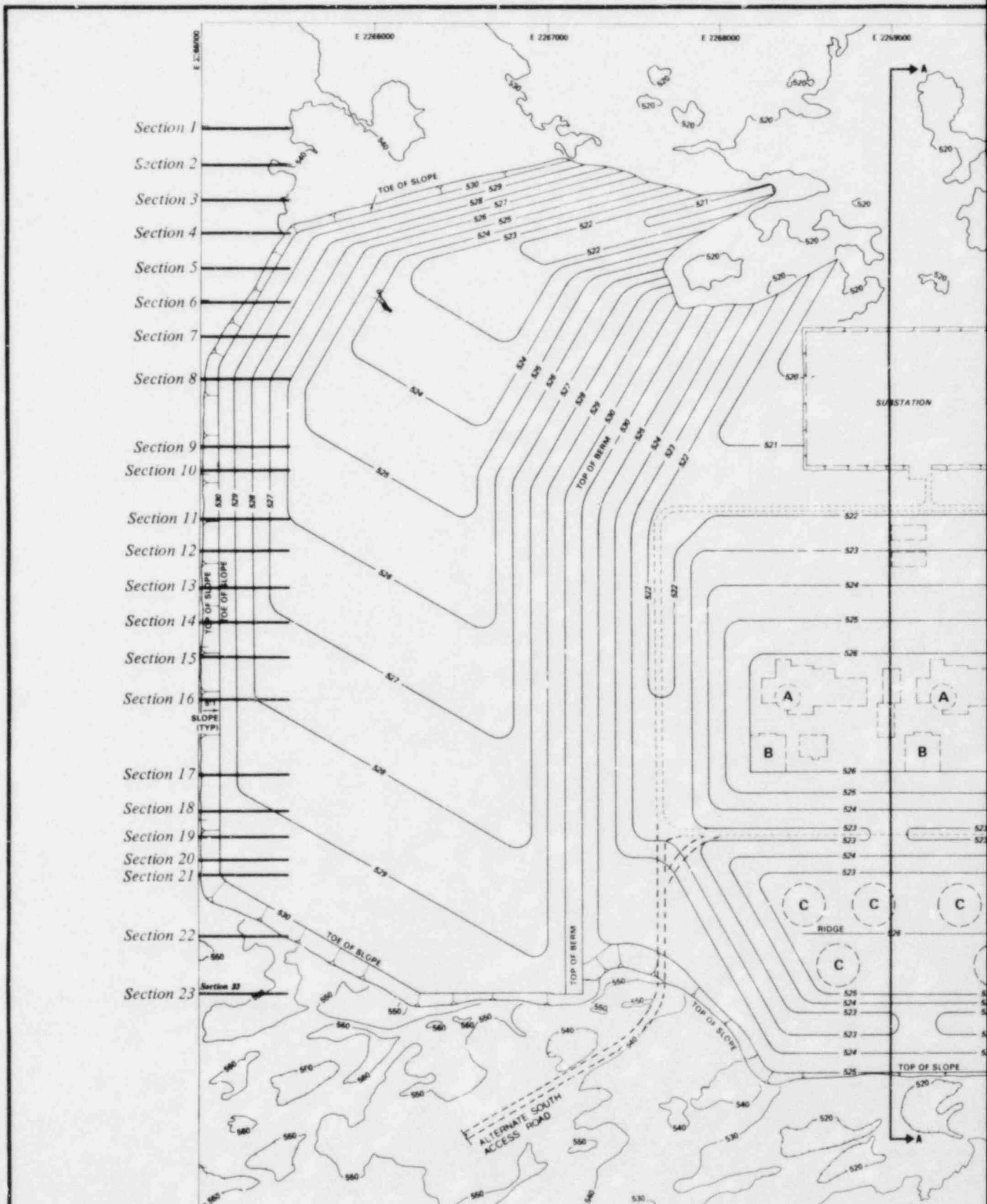
Feet
0 400 800
200 600 1000
Scale

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR PROJECT

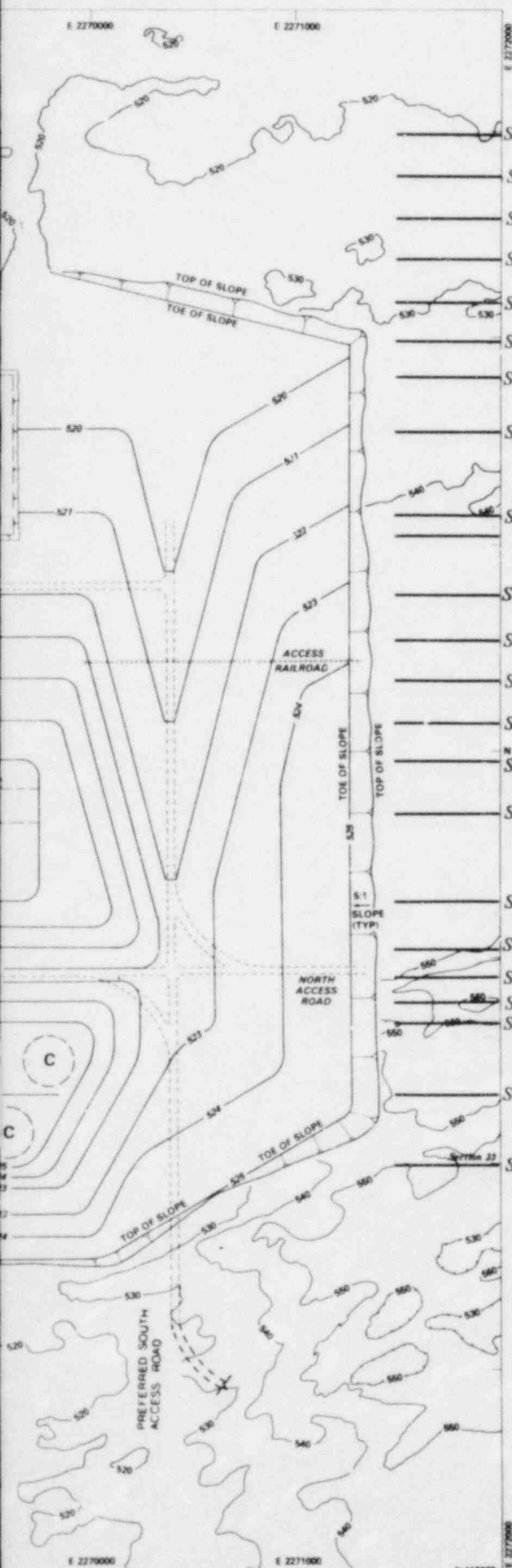
PRELIMINARY SAFETY ANALYSIS REPORT

SITE GRADING, DRAINAGE AND PLANT LAYOUT

FIGURE 2.4-3



Section 1
Section 2
Section 3
Section 4
Section 5
Section 6
Section 7
Section 8
Section 9
Section 10
Section 11
Section 12
Section 13
Section 14
Section 15
Section 16
Section 17
Section 18
Section 19
Section 20
Section 21
Section 22
Section 23
E 2266000
E 2267000
E 2268000
E 2269000
N 419000
TOE OF SLOPE
TOP OF SLOPE
TOP OF BERM
SUBSTATION
RIDGE
SLOPE (TYP)
ALTERNATE SOUTH ACCESS ROAD
A
B
C



LEGEND:

A. REACTOR BUILDING
B. ULTIMATE HEAT SINK
C. COOLING TOWERS

Ref: Bechtel Drawing H-SK-C-10 Rev. B

N

Feet

0 400 800

200 600 1000

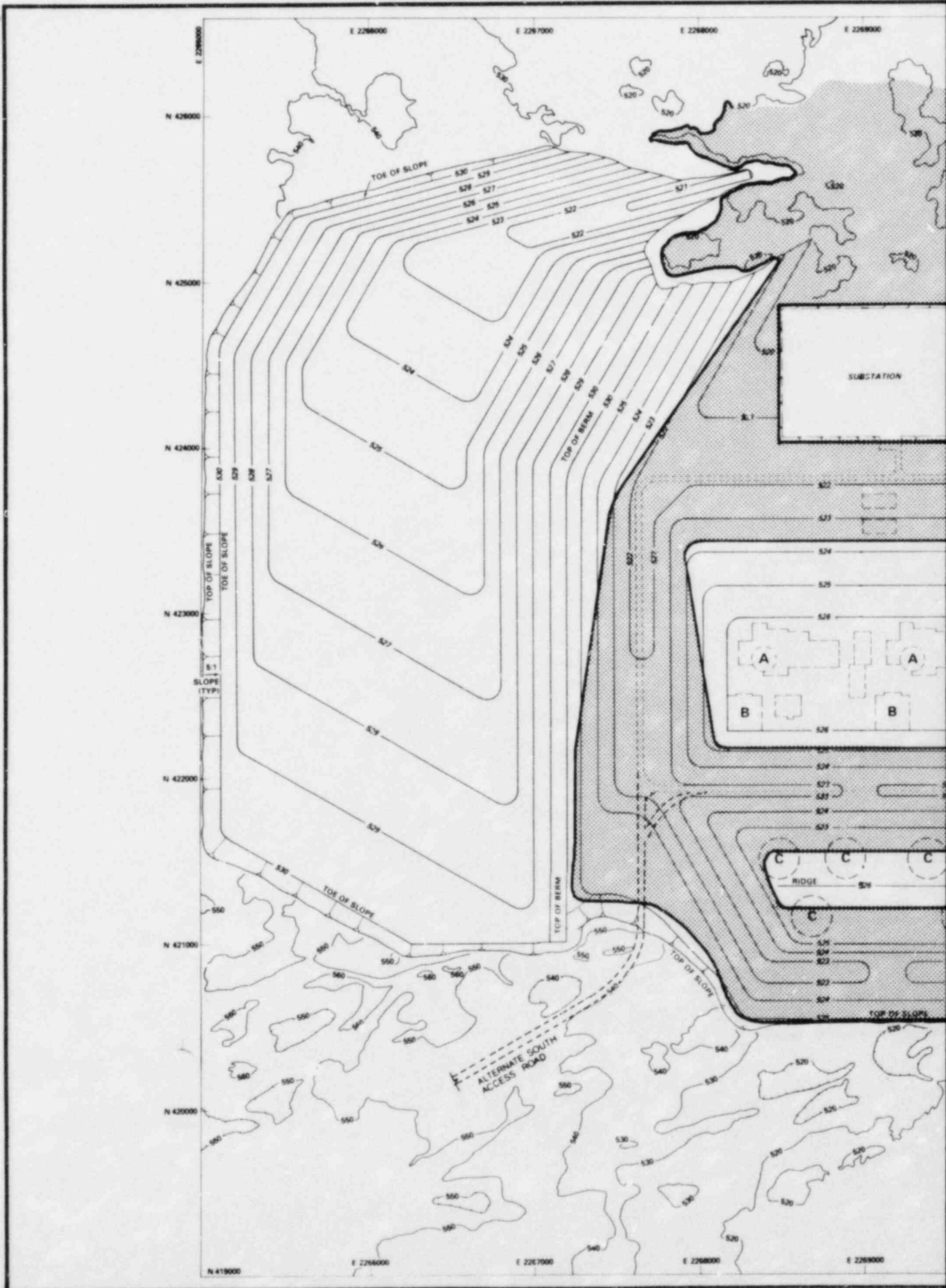
Scale

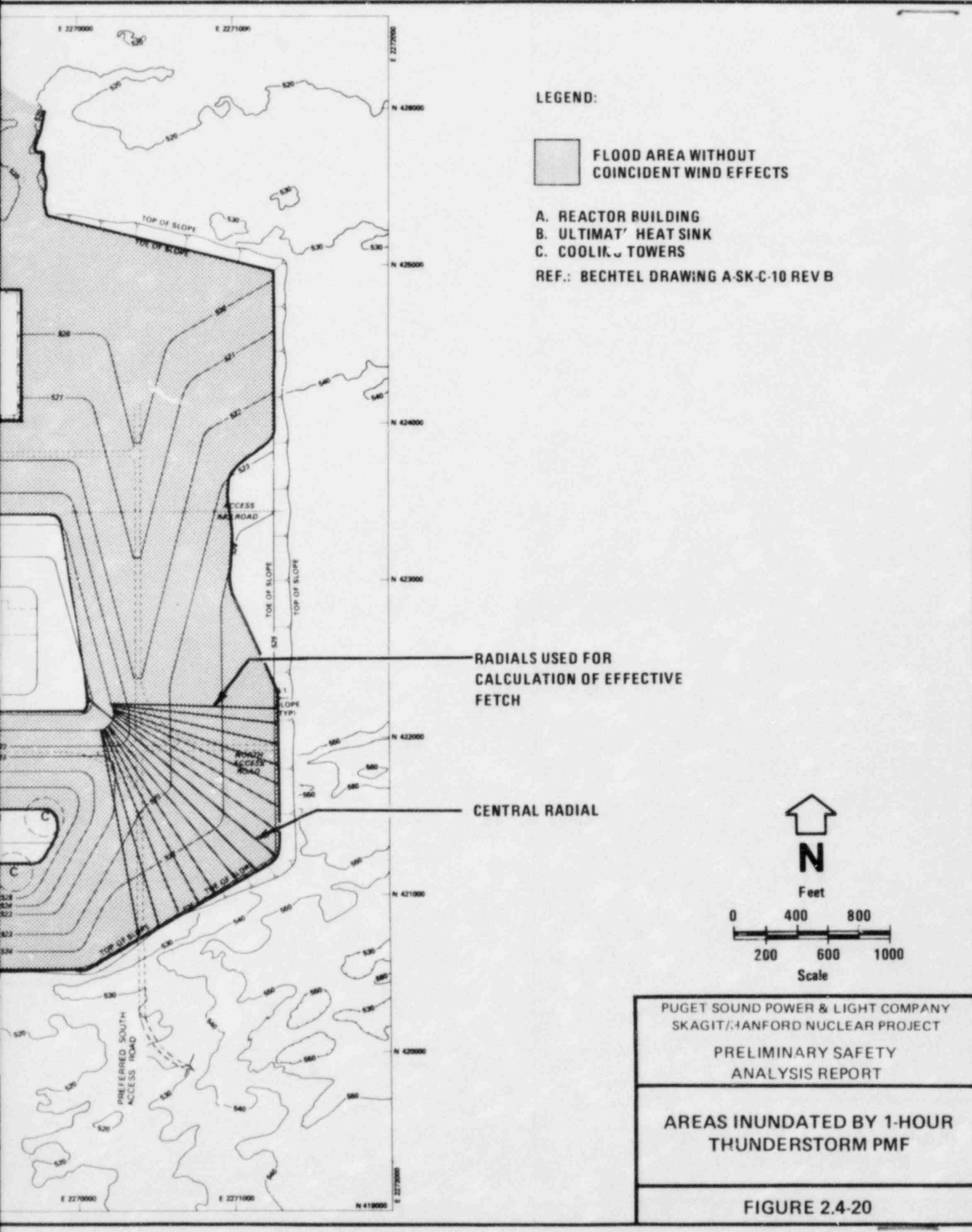
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

LOCATION OF SECTIONS USED FOR BACKWATER CALCULATIONS

FIGURE 2.4-17


E 2266000
E 2267000
E 2268000
E 2269000
N 426000
N 425000
N 424000
N 423000
N 422000
N 421000
N 420000
N 419000
TOE OF SLOPE
TOP OF SLOPE
TOP OF BERM
5:1
SLOPE (TYP)
SUBSTATION
RIDGE
A
B
C
ALTERNATE SOUTH ACCESS ROAD



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

AREAS INUNDATED BY 1-HOUR THUNDERSTORM PMF

FIGURE 2.4-20

L = Live loads including floor occupancy loads, laydown loads due to temporary placement of equipment; nuclear fuel and fuel transfer casks, equipment handling loads, lateral earthfill loads, lateral and vertical surcharge loads due to transport vehicles; 23 pressure differences due to heating, cooling and normal atmospheric changes; roof loads due to snow and impounded rainfall up to 6" deep; hydrostatic loads due to compartment flooding. 27 Loads due to Safety Relief Valve pressures as outlined in Appendix 6C of this PSAR are included.

L_o = Operating live loads likely to occur during normal operation. These are the live loads to be used in seismic analysis and with 23 seismic load combinations. The operating live load (L_o) is a relatively small fraction of the design live load (L); L_o does not include such loads as those due to laydown, maintenance, or temporary cranes or moving equipment. 27

T_o = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

R_o = Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

3.8.6.1.2 Severe Environmental Loads

Severe Environmental loads are those that could infrequently be encountered during the Plant life. Included in this category are:

E_o = Loads generated by the Operating Basis Earthquake (OBE). The earthquake is composed of two horizontal and one vertical components and the effects of the three components are combined, based on the square root of the sum of the squares. Only the dead load (D) and the operating 23 live load (L_o) need be considered in evaluating the seismic response forces.

W = Loads generated by the design wind specified for the Plant.

3.8.6.1.3 Extreme Environmental Loads

Extreme environmental loads are those which are credible but are highly improbable. They include:

E_{ss} = Loads generated by the Safe Shutdown Earthquake (SSE). The earthquake is composed of two horizontal and one vertical components and the effects of the three components are combined, based on the square root of the sum of the squares. Only the dead load (D) and the operating live load (L_o) need to be considered in evaluating the seismic response forces. 23

V = Roof load due to volcanic ashfall. 27

Wt = Effects generated by the design tornado specified for the Plant. They include loads due to the tornado wind pressure and differential pressures, and also the energy resulting from impact of tornado-generated missiles.

P_p = Design-basis winter precipitation resulting from a combination of 11.7 in. of water from the 48-hr PMP coincident with 3.8 in. of water equivalent from the 100-year snowpack. (See Section 2.4.2.3.) 27

3.8.6.1.4 Abnormal Loads

Abnormal loads are those loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a = Design Pressure load within or across a compartment and/or building, generated by the postulated pipe rupture, including the dynamic effects due to the pressure time history and pool-swell phenomena as outlined in Appendix 6C of this PSAR. 12

T_a = Thermal effects due to thermal conditions generated by the postulated break and including T_o.

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:

$$U = D + L_o + T_o + R_o + (E_{ss} \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}9)$$ 27

$$U = D + L + T_a + R_a + 1.5\ P_a \quad (3.8\text{-}10)$$ H220.19

$$U = D + L_o + T_a + R_a + 1.25\ P_a + (Y_r + Y_j + Y_m) + 1.25\ (E_o \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}11)$$ 27

$$U = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + (E_{ss} \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}12)$$ 27

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:

$$S = D + L + R_o + T_o \quad (3.8\text{-}13)$$ H220.19

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

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

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:

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

$$Y = 1.7D + 1.7L_o + 1.7E_o \quad (3.8\text{-}17)$$ H220.19

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

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:

$$Y = 1.3(D + L + T_o + R_o) \quad (3.8\text{-}19)$$

$$Y = 1.3(D + L_o + E_o + T_o + R_o) \quad (3.8\text{-}20)$$ H220.19 25

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

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

$$1.6S = D + L_o + T_o + R_o + (E_{ss} \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}22)$$ 27

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

$$1.6S = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + E_o \quad (3.8\text{-}24)$$ H220.19

$$1.7S = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + (E_{ss} \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}25)$$ 27

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

$$Y = D + L_o + T_o + R_o + (E_{ss} \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}26)$$ 27

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

$$Y = D + L_o + T_a + R_a + 1.25\ P_a + (Y_r + Y_j + Y_m) + 1.25\ E_o \quad (3.8\text{-}28)$$ H220.19

$$Y = D + L_o + T_a + R_a + P_a + (Y_r + Y_j + Y_m) + (E_{ss} \text{ or } W_t \text{ or } V \text{ or } P_p) \quad (3.8\text{-}29)$$ 27

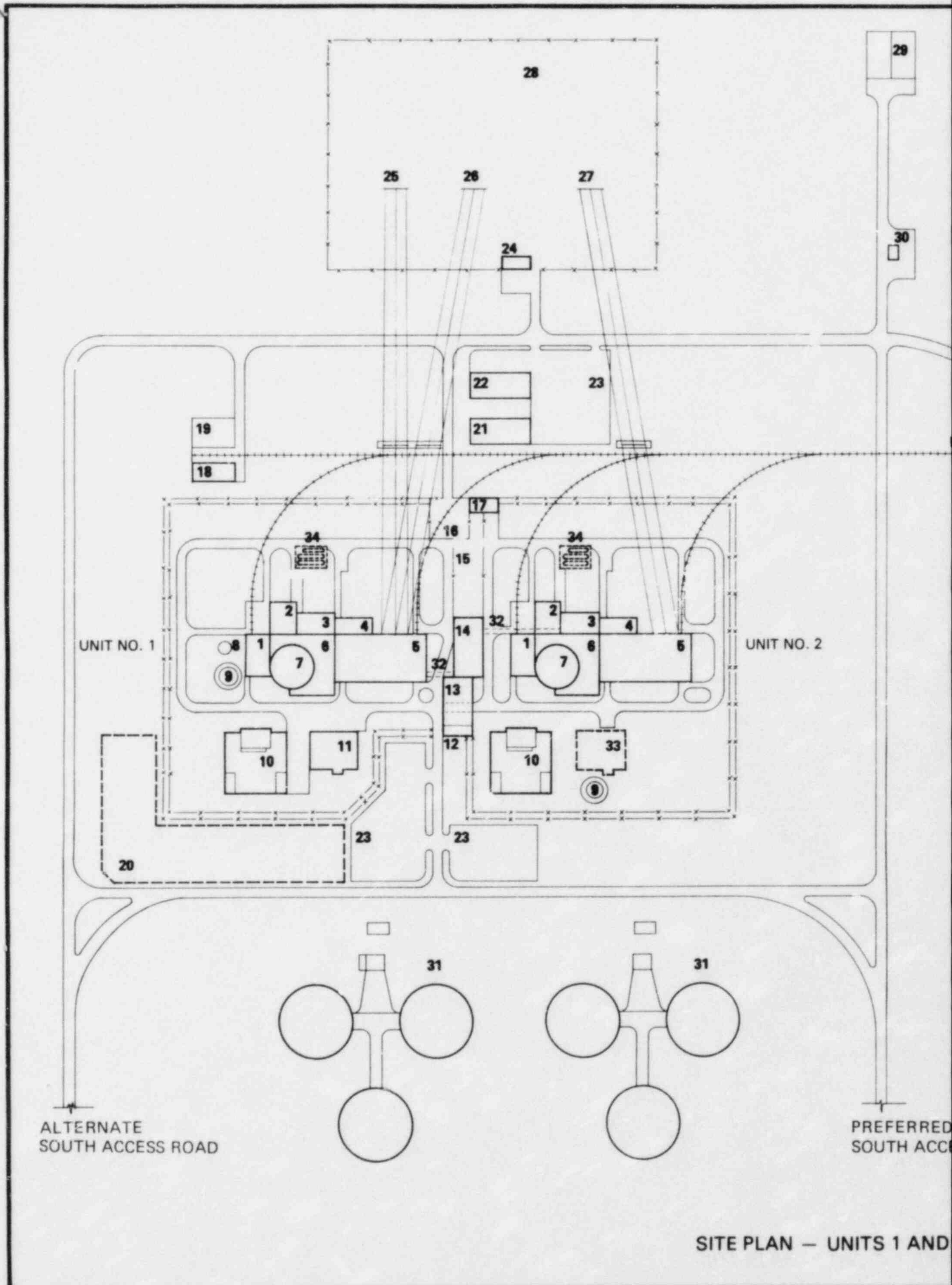
In combinations (3.8-22) to (3.8-29), thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material being designed for is ductile.

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

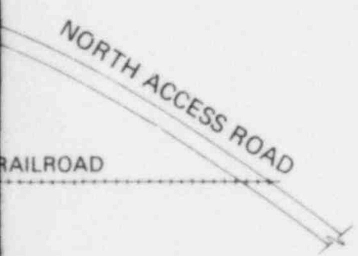
In combinations (3.8-23) through (3.8-25) and (3.8-27) through (3.8-29), the maximum effects of P_a, T_a, R_a, Y_j, Y_r, and Y_m are used unless a time-history analysis is performed to justify otherwise. H220.14

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.



SITE PLAN — UNITS 1 AND



1 FUEL BUILDING
2 DIESEL BUILDING
3 CONTROL BUILDING
4 SWITCHGEAR BUILDING
5 TURBINE BUILDING
6 AUXILIARY BUILDING
7 REACTOR BUILDING
8 REFUELING WATER STORAGE TANK
9 CONDENSATE STORAGE TANK
10 ULTIMATE HEAT SINK
11 RADWASTE BUILDING
12 SOUTH GUARD STATION (below service bldg)
13 SERVICE BUILDING
14 SHOP AND WAREHOUSE
15 WAREHOUSE YARD
16 SALLY PORT
17 NORTH GUARD STATION
18 WATER TREATMENT BUILDING
19 LOW VOLUME WASTE POND
20 TRAINING FACILITY (future)
21 CONSTRUCTION OFFICE
22 CONSTRUCTION WAREHOUSE
23 PARKING
24 CONTROL HOUSE
25 UNIT NO. 1 525 KV LINE OPTION A
26 UNIT NO. 1 525 KV LINE OPTION B
27 UNIT NO. 2 525 KV LINE
28 SUBSTATION
29 PERCOLATION POND
30 SEWAGE TREATMENT PLANT
31 COOLING TOWERS
32 BRIDGE (to power block)
33 RADWASTE BUILDING (future—if required)
34 DIESEL FUEL STORAGE TANKS (underground)

0 100 200 400 800'
SCALE
NORTH ↑

STRUCTURE	SEISMIC CATEGORY I	SEISMIC CATEGORY II	TORNADO RESISTANT
Containment	X		X
Containment Enclosure Building	X		*
Diesel Building	X		X
Control Building	X		X
Radwaste Building		X	
Auxiliary Building	X		X
Fuel Building	X		X
Ultimate Heat Sink	X		X
Condensate Storage Tank Basin		X	
Diesel Fuel Oil Storage Tanks	X		X
Turbine Building		X	
Administration Building		X	
Circulating Water Pump House		X	
Mechanical Draft Cooling Towers		X	
Raw Water Pump House		X	

*NOTE: ALL EXCEPT SIDING

SS ROAD

2

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

LAYOUT OF PLANT STRUCTURES

FIGURE 3.8-10

inlets and exhausts on safety-related buildings will be protected by tornado missile barriers (and if necessary louvers) which will preclude any significant snow, ice or dust from blocking the inlets or exhausts, or any significant snow, water or dust from entering the air systems. 020.24
The Diesel Generator exhaust will be discharged through exhaust stacks which will be designed to preclude any significant amount of rain, ice, snow or dust from entering or blocking them. Section 9.2.5.3.7 discusses ice protection for the Ultimate Heat Sink Complex. 27

permit recirculation of the control room by an A/C unit and a return/exhaust fan and filtration of a portion of the air through the standby filtration unit(s). After the fire has been extinguished, the Control Room HVAC System can be manually changed to the purge mode.

The control room can also be completely isolated 25
by manual operator action.

In the event that the FSAR analysis of the S/HNP offsite hazards identifies the requirement for an
automatic detection and isolation system, this 27
system will be provided in accordance with the criteria of the Standard Review Plan Section 9.4 (NUREG-0800).

9.4.1.1.3 Design Evaluation

The concentration of radioactivity, which will be assumed to surround the control room after the postulated accident, will be evaluated as a function of the fission product decay constants, containment leak rate, and the meteorology for each period of interest. The assessment of the amount of radioactivity within the control room takes into consideration the flow rate through the control room outside air intake duct, and the effectiveness of the standby filtration unit.

Control room shielding design, discussed in Chapter 12, is based on the fission product release to the Containment caused by the design basis LOCA as evaluated in accordance with Regulatory Guide 1.3 in Chapter 15. Shielding is provided to ensure that radiation exposures of the control room personnel for the duration of the accident are within the limits specified by 10 CFR 50, Appendix A, Criterion 19.

Redundant radiation monitors will be provided in the 23
outside air intake duct of the control room central A/C units. Upon detection of a high radiation signal by the monitors, an alarm will be annunciated in the control room, and the control room central A/C unit(s) will be isolated from its source of outside air supply, and the Control Room HVAC System will be automatically transferred to the standby mode of operation. Transfer of the system to the standby mode also may be initiated manually from the control room upon detection of high radiation by an area radiation monitor located within the control room.

The control room standby filtration unit will draw the
incoming air through the high efficiency filters, upstream 23
HEPA filters, carbon adsorbers, and downstream HEPA filters to minimize the exposure of control room personnel to

airborne radioactivity in accordance with 10 CFR 20 requirements. A portion of the control room air can be recirculated continuously through the filter train for further removal of airborne radioactive particulates from the control room atmosphere. Operation of the standby filtration unit reduces the likelihood that outside air will enter the control room via paths other than through the standby filtration train. The resulting calculated doses for control room ingress, egress, and occupancy will not exceed 5 rem to the whole body or its equivalent to any part of the body as specified in the NRC General Design Criterion 19. A detailed discussion of the dose levels in the control room under standby operation is presented in Chapter 15.

Procedures will be provided for proper use of immediately-available breathing apparatus by the emergency crew. A minimum six-hour supply of bottled air for the emergency crew will be readily available on-Site to allow sufficient time for off-Site delivery of bottled air for several hundred hours of consumption.

Noncombustible construction and heat and flame-resistant materials will be used throughout the Plant to minimize the likelihood of fire and consequential fouling of the control room atmosphere with smoke or noxious vapors. Smoke detectors will be provided in each outside air inlet duct and areas of the control room to detect smoke or noxious vapors in the control room. In the event that detectable smoke or noxious vapors exist in the outside air inlet duct, an alarm will be annunciated in the control room and 310.22 the HVAC System will be automatically transferred to the standby mode of operation. If detectable smoke or noxious vapors exist in the control room and clearing of the control room atmosphere should be required, the Control Room HVAC System, operated in the purge mode, will remove smoke or noxious vapor from the control room at the rate of approximately 15 air changes per hour.

The Control Room HVAC equipment, ductwork (except the utility exhaust fans and their associated ductwork), and surrounding structures will be of Seismic Category I design. All components of the system will be operable during a loss of normal power, by connection to the Engineered Safety Features buses. Redundant components are provided wherever necessary, to ensure that any single failure will not preclude adequate control room ventilation, air cleanup, and pressurization. The redundant unit will be automatically started on failure of the operating 23 unit. The Control Room HVAC System failure analysis is presented in Table 9.4-2.

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 N.SS 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 23
energy gammas which it emits as it decays.

Turbine shine doses are calculated using the SKYSHINE 331.3
computer program described in Table 12.1-3. Point sources
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
east and a 31'-0" high wall on the west. The center of the 23
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 Wye Barricade, which
is approximately 2 miles from the turbine building, is
conservatively estimated to be less than 0.5 mrem/yr. This
is the most appropriate point to estimate the dose
potentially incurred by members of the general public as a
result of the operation of S/HNP because the Wye Barricade
is an access control point of the Hanford Reservation and, H471.2
in conjunction with other Hanford Reservation controls,
serves to prohibit residences or long-term transients from
the vicinity of the S/HNP. For this reason occupancy by the
public of any point closer than about 2 miles is expected to
be negligible. Nevertheless, for calculational purposes a
conservatively high occupancy factor of 5% may be assumed
for points closer than 2 miles. Under such circumstances, 27
the highest expected turbine shine dose at the site boundary
(restricted area boundary) is conservatively estimated to be 25
2.5 mrem/yr, based on two unit operation and an availability
of 80%.

Recovery - A period of time beginning when the Plant reaches a safe shutdown condition, and lasting until the Plant is restored as nearly as possible to its pre-emergency condition.

Site - The area controlled by Puget and within the exclusion area boundary as defined in Title 10, Code of Federal Regulations, Part 100.3(a).

Site Area Emergency - An event at the Plant involving actual or potential major failures of key safety-related equipment which might lead to a potential degraded core situation.

Technical Specifications - The limits, operating conditions, and other requirements imposed by the NRC on S/HNP operation. 23

Technical Support Center - On-Site facility which provides a location for Puget technical support of the reactor command and control functions of the control room.

TLD - Thermoluminescent dosimeters. Devices used to measure the level of exposure to radiation.

Unusual Event - An event at the Plant which results in no significant release of radioactive material, but which could lead to a potential degradation in the level of safety of the Plant.

3.0 SITE DESCRIPTION AND EMERGENCY PLANNING ZONES

3.1 SITE DESCRIPTION

The Skagit/Hanford Nuclear Project Site is located in the
southeast area of the U.S. Department of Energy's (DOE)
Hanford Reservation in Benton County, Washington. The Site
is approximately 5 miles west of the Washington Public Power
Supply System's Nuclear Project No. 2 (WNP-2) unit. It is
approximately 8 miles west of the Columbia River, approx-
imately 7.5 miles north of the Yakima River at Horn Rapids
Dam, and approximately 12 miles northwest of North Richland.
Figures 1 and 2 locate the Site within the region and 23
identify the general location of the Plant Site with respect
to roads, highways, rivers, and population centers within
the vicinity.

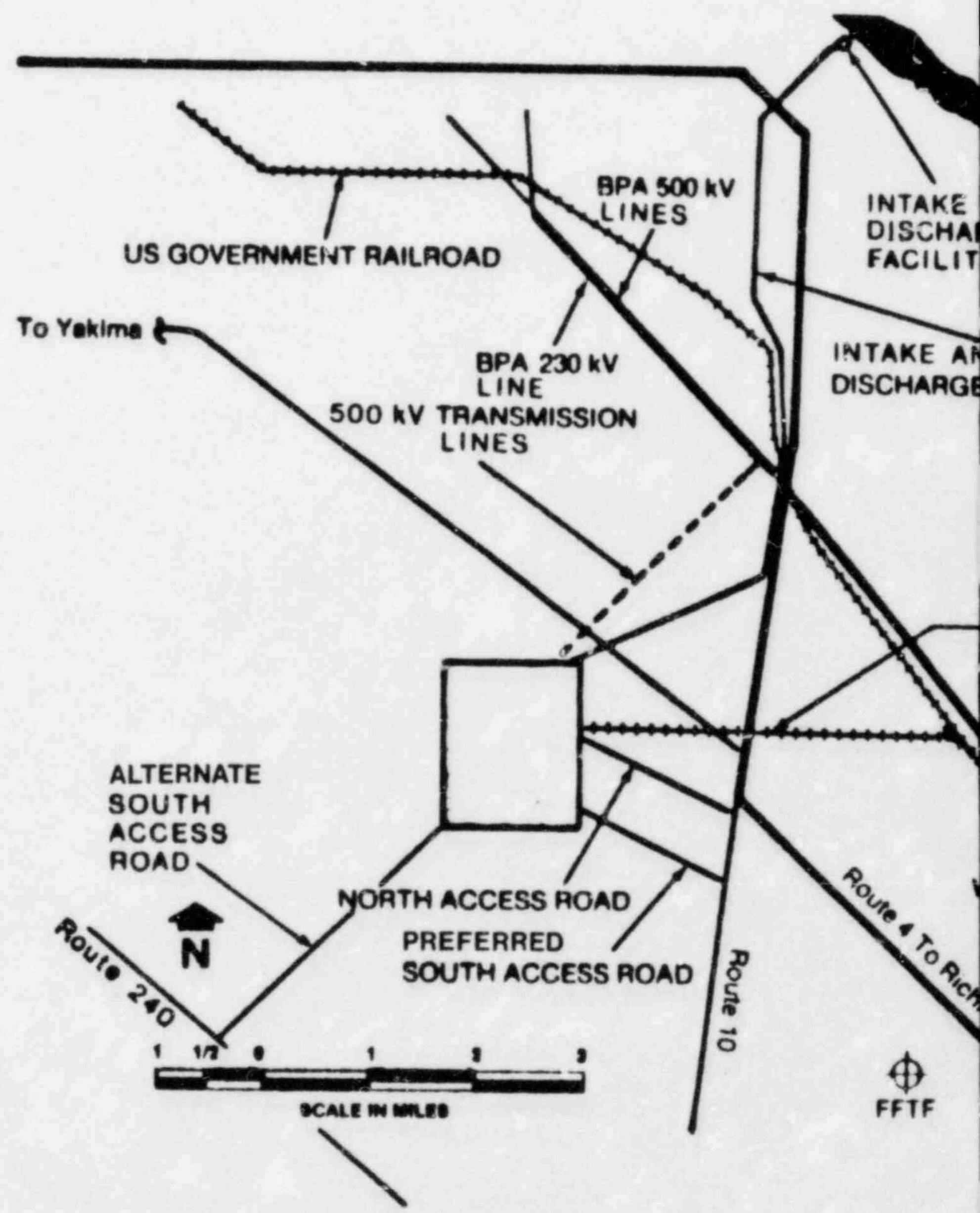
Figure 2 shows the Plant Site, including topographic features, and the location and orientation of principal Plant structures. No public roads or railroads cross the Site.

The Site boundary lines are shown in Figure 2. The Site
area boundary, the station property lines, and the
restricted area boundary are the same. The Plant exclusion
area boundary is shown on Figure 2. The exclusion area is
that area within 1 mile of the line joining the reactor
centers. 27

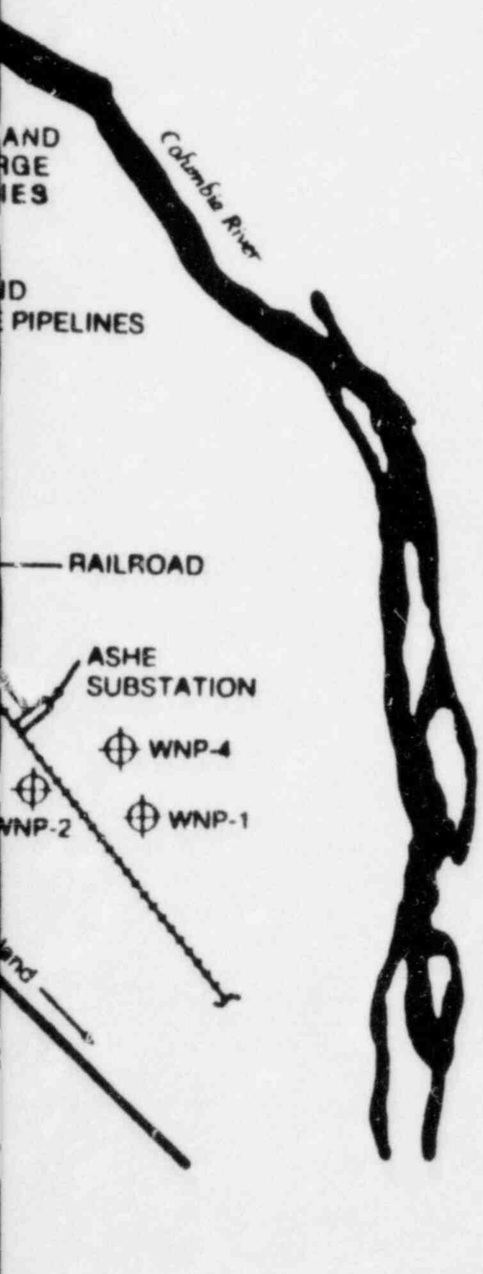
3.2 EMERGENCY PLANNING ZONES

The S/HNP Emergency Program provides for emergency planning
within two Emergency Planning Zones (EPZs). A plume
exposure EPZ, of about a 10-mile radius around the Plant, is
defined for the purpose of planning for public protective
actions based upon exposure or inhalation of a passing
radioactive plume released during an accident. An ingestion 23
exposure EPZ, of about a 50-mile radius around the Plant, is
defined for the purpose of planning for public protective
actions based upon ingestion of contaminated water or foods.

The size of the EPZs have been determined in relation to local emergency response needs and capabilities as they are affected by demography, topography, land characteristics, access routes and jurisdictional boundaries.



INTAKE
DISCHA
FACILIT
BPA 500 kV
LINES
US GOVERNMENT RAILROAD
To Yakima
BPA 230 kV
LINE
INTAKE A
DISCHARGE
500 kV TRANSMISSION
LINES
ALTERNATE
SOUTH
ACCESS
ROAD
NORTH ACCESS ROAD
PREFERRED
SOUTH ACCESS ROAD
N
Route 240
Route 10
Route 4 To Rich
FFTF
1 1/2 0 1 2 3
SCALE IN MILES

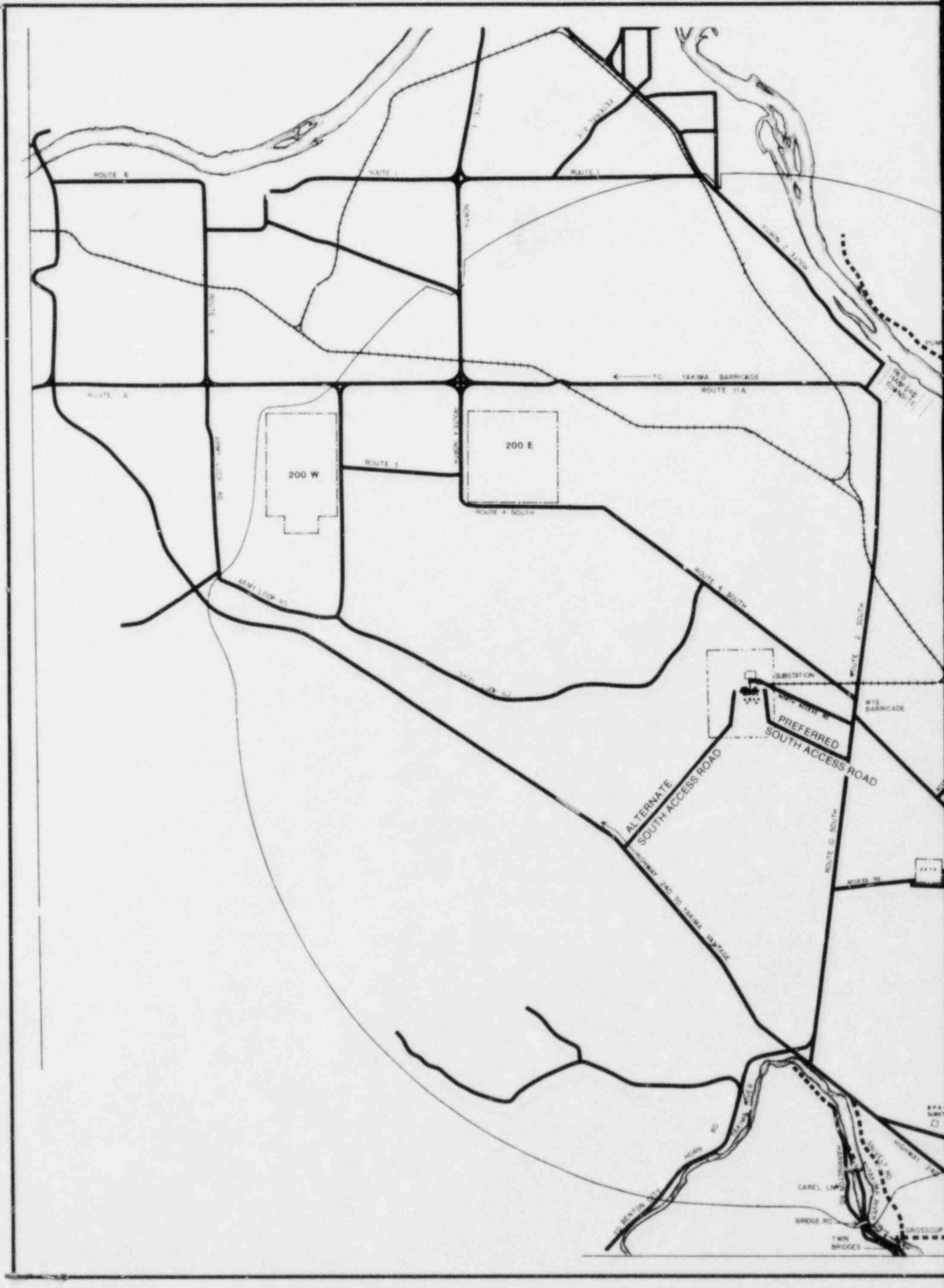


PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

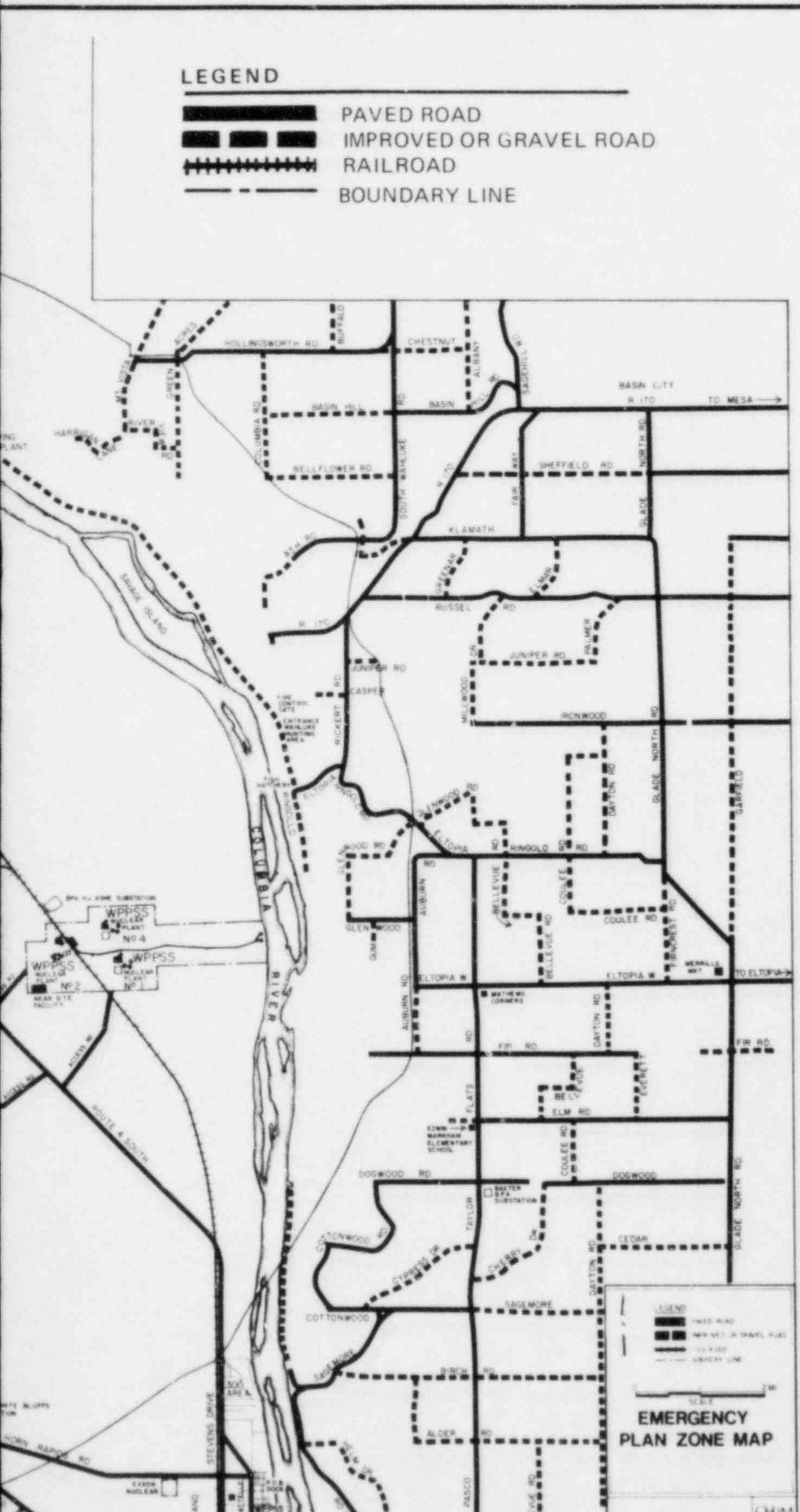
PRELIMINARY EMERGENCY PLAN

SITE VICINITY

FIGURE 2



ROUTE 6
ROUTE 1
ROUTE 1
ROUTE 6
ROUTE 1A
TO YAKIMA BARRICADE
ROUTE 11A
200 W
200 E
ROUTE 3
ROUTE 4 SOUTH
ARMY LOOP RD
ROUTE 4 SOUTH
SUBSTATION
WYE BARRICADE
PREFERRED
SOUTH ACCESS ROAD
ALTERNATE
SOUTH ACCESS ROAD
HIGHWAY 240 TO YAKIMA VANTAGE
ROUTE 2 SOUTH
ROUTE 10 SOUTH
HORN RD
CAREL LN
BRIDGE RD
TWIN BRIDGES
GROSSCUP

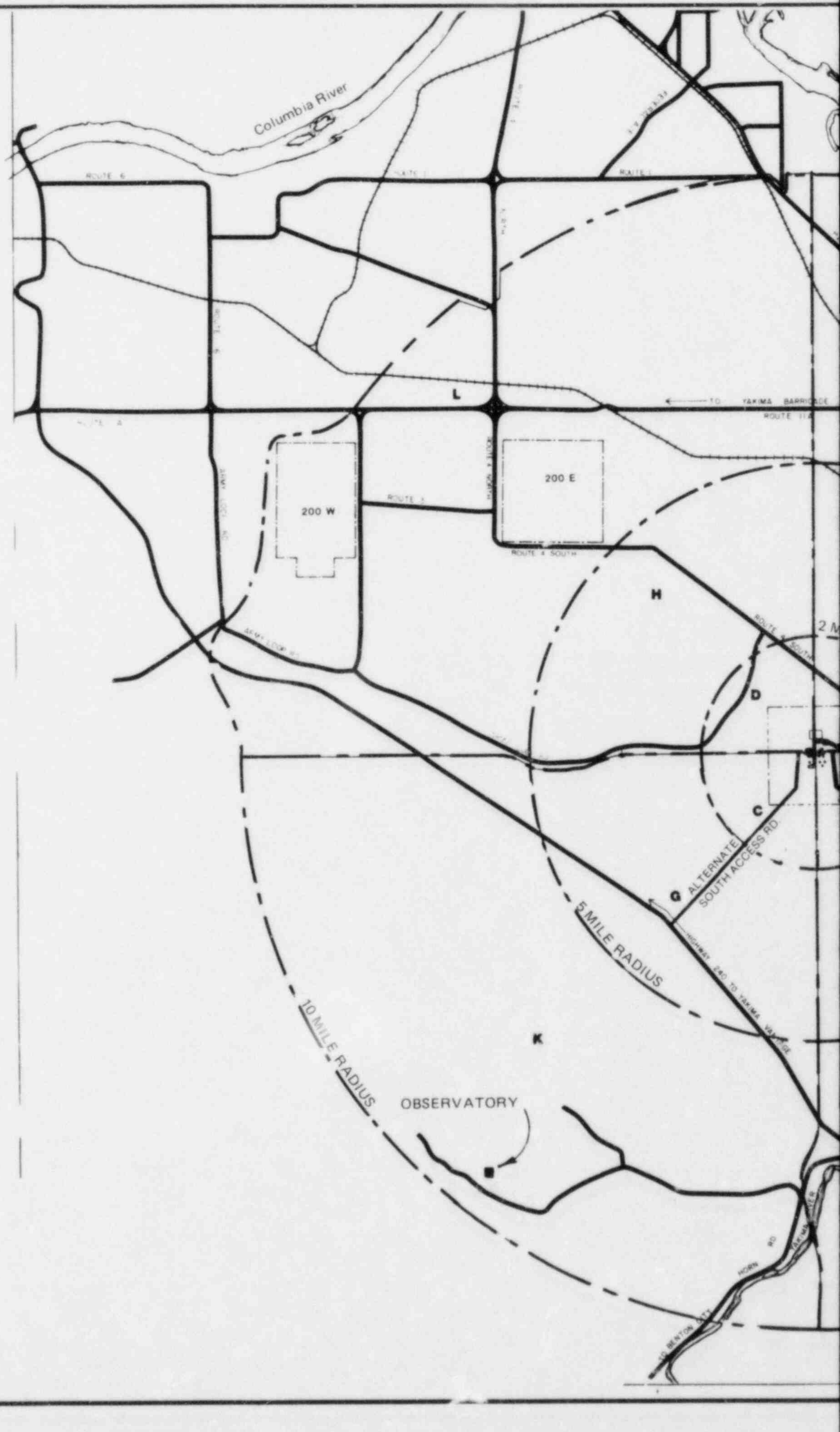


PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

PRELIMINARY EMERGENCY PLAN

PLUME EXPOSURE EPZ

FIGURE 3


Columbia River
ROUTE 6
TO YAKIMA BARRICADE
L
200 W
200 E
ROUTE 4 SOUTH
H
D
C
G
ALTERNATE
SOUTH ACCESS RD
5 MILE RADIUS
10 MILE RADIUS
K
OBSERVATORY

S15398.A0

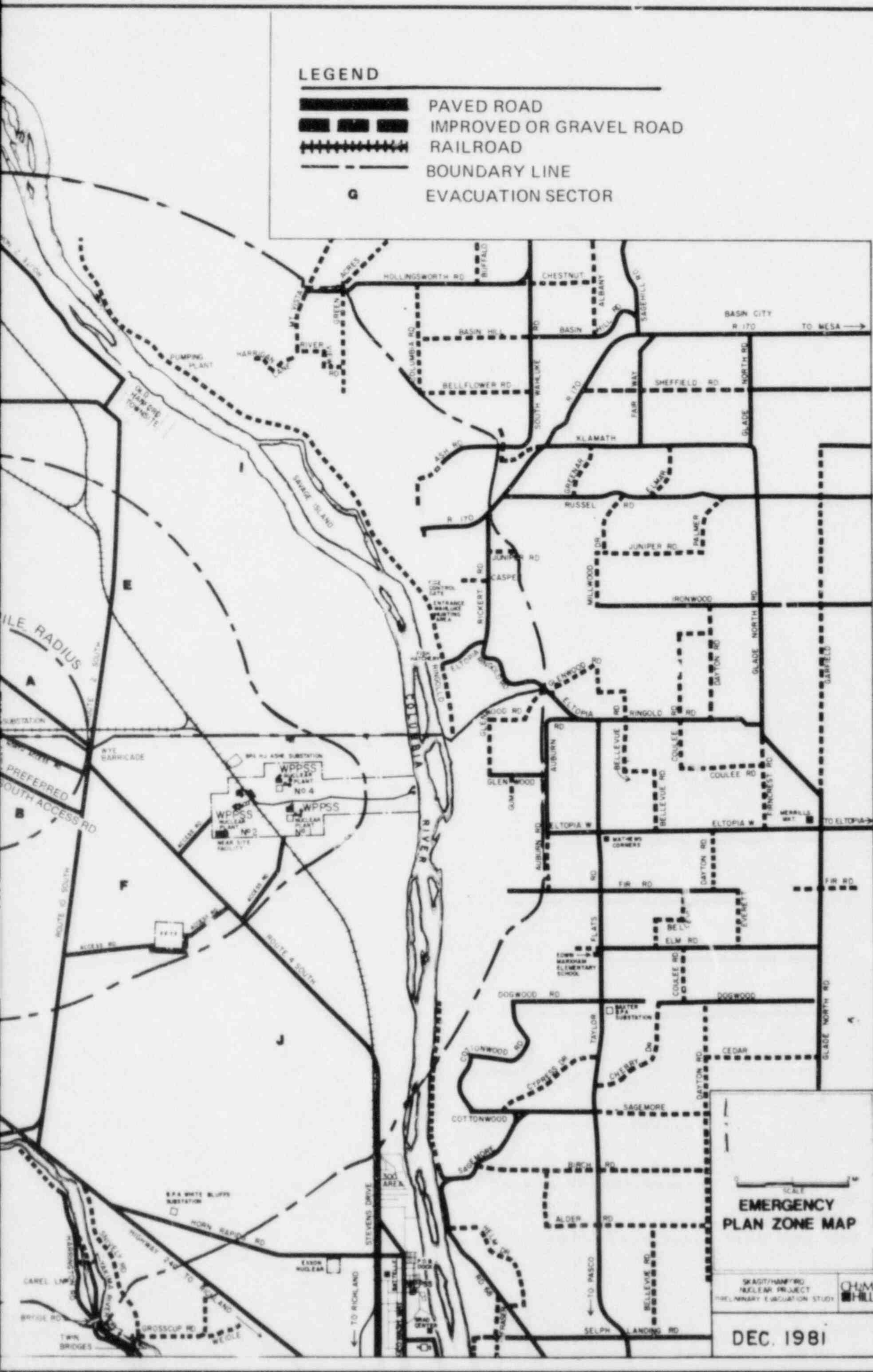
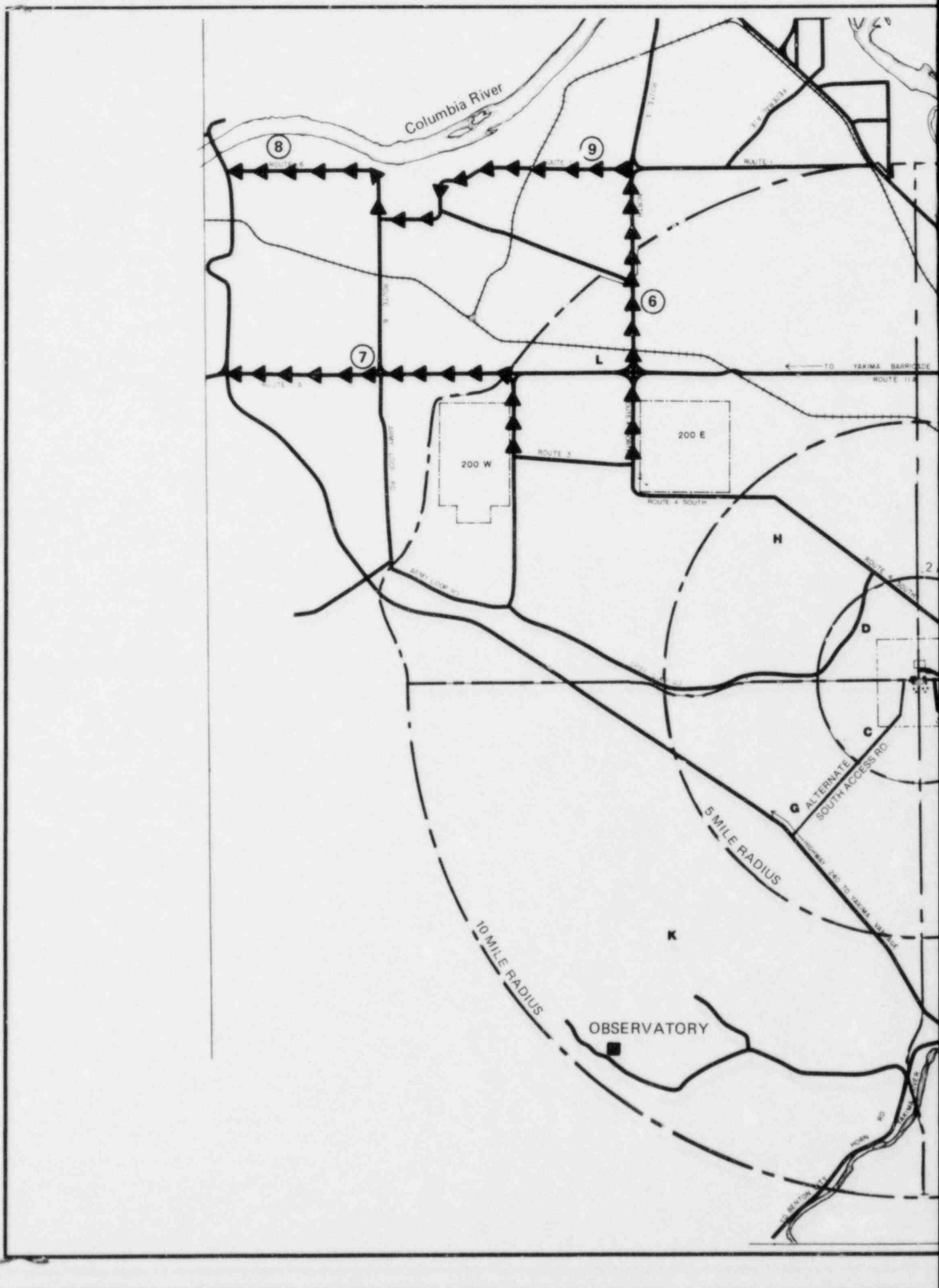


FIGURE 1
PLUME EXPOSURE EPZ


Columbia River
8
9
6
7
L
TO YAKIMA BARRICADE
200 E
200 W
ROUTE 3
ROUTE 4 SOUTH
H
D
C
G
ALTERNATE
SOUTH ACCESS RD
5 MILE RADIUS
K
10 MILE RADIUS
OBSERVATORY

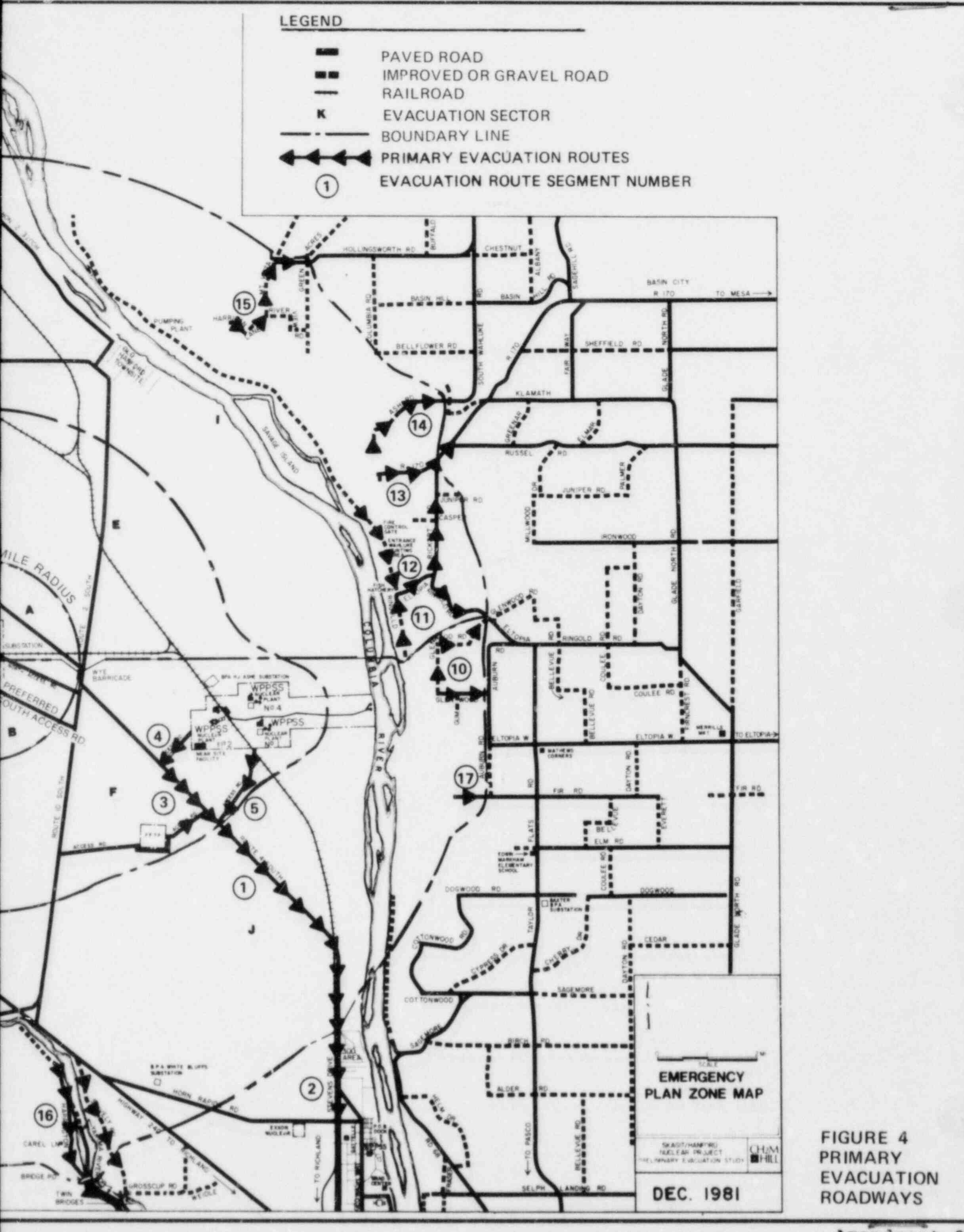


FIGURE 4
PRIMARY
EVACUATION
ROADWAYS

TABLE 15.2-1

TYPE II TRANSIENT OFF-SITE DOSE 23

	Dose Effect (mrem)			
Distance (m)	Whole Body	Skin	Thyroid	
1609 (EAB)[1]	4.90	2.88	2.85×10^{-2}	27
6437 (LPZ)[2]	5.32×10^{-1}	3.13×10^{-1}	3.11×10^{-3}	

[1]EAB Exclusionary Area Boundary 23
[2]LPZ Low Population Zone

TABLE 15.2-2

TYPE II TRANSIENT ON-SITE EXPOSURES

Organ Evaluated	Dose Effect (mRem)
Whole Body	39
Skin	545
Thyroid	0.1
Lung	0.8

23

TABLE 15.2-3

TYPE II S/R VALVE TRANSIENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

		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	23
II.	Data and assumptions used to estimate activity released	NA		
	A. Containment leak rate	NA	Infinite [a]	
	B. Secondary containment leak rate (%/day)	NA	NA	
	C. Valve movement times	NA	15.2.4.2.2	
	D. Adsorption and filtration efficiencies	NA		
	(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	NA		
	(1) Flow rate	NA		
	(2) Mixing efficiency	NA	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	15.2.4.2.2	
III.	Dispersion Data	NA		
	A. EAB and LPZ distances (m)	NA	1609/6437	27
	B. X/Q values in sec/m^3	NA	2.8×10^{-5}/ 3.0×10^{-6}	
IV.	Dose Data			
	A. Method of dose calculation	NA	Sect 15.2.4.2.2	
	B. Dose conversion assumptions	NA	Sect 15.2.4.2.2	
	C. Activity in containment	NA	Sect 12.2.3	
	D. Doses	NA	Tables 15.2-2 and 15.2-1	23

(a) Applicable 8 hours after S/R valve transient commences

TABLE 15.2-4

FEEDWATER LINE BREAK ACCIDENT

Distance (meters)	Thyroid Dose (rem)	
Conservative Analysis		23
1609 (EAB)	2.40×10^{-4}	27
6437 (LPZ)	3.39×10^{-5}	23
Realistic Analysis		
1609 (EAB)	1.25×10^{-5}	27
6437 (LPZ)	1.38×10^{-6}	23

TABLE 15.2-5

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	4100 MWt	4100 MWt
B. Burn-up	NA	NA
C. Fuel damaged	None	None
D. Release of activity by nuclide	15.2.8.2.1.2.2	15.2.8.2.2.2.2
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor coolant activity before the accident	15.2.8.2.1.1.2	15.2.8.2.2.1.2
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Isolation valve closure time (sec)	30	30
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters		
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	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	None	None
III. Dispersion Data		
A. EAB and LPZ distances (m)	1609/6437	1609/6437
B. X/Q values in sec/m^3	1.5×10^{-4}/ 2.1×10^{-5}	2.8×10^{-5}/ 3.0×10^{-6}
IV. Dose Data		
A. Method of dose calculation	Reference 1	Reference 1
B. Dose conversion assumptions	Reference 1	Reference 1
C. Activity in containment	NA	NA
D. Off-Site Doses	Table 15.2-4	Table 15.2-4

23

27

23

TABLE 15.4-7

CONTROL ROD DROP ACCIDENT

OFF-SITE DOSES

Distance (meters)	Dose (rem) Whole-Body	Thyroid	
Conservative Analysis			23
1609 (EAB)	2.95×10^{-2}	1.29×10^{-2}	27
6437 (LPZ)	1.02×10^{-2}	1.99×10^{-2}	23
Realistic Analysis			
1609 (EAB)	8.23×10^{-7}	6.12×10^{-7}	27
6437 (LPZ)	2.83×10^{-6}	7.22×10^{-6}	23

TABLE 15.4-8

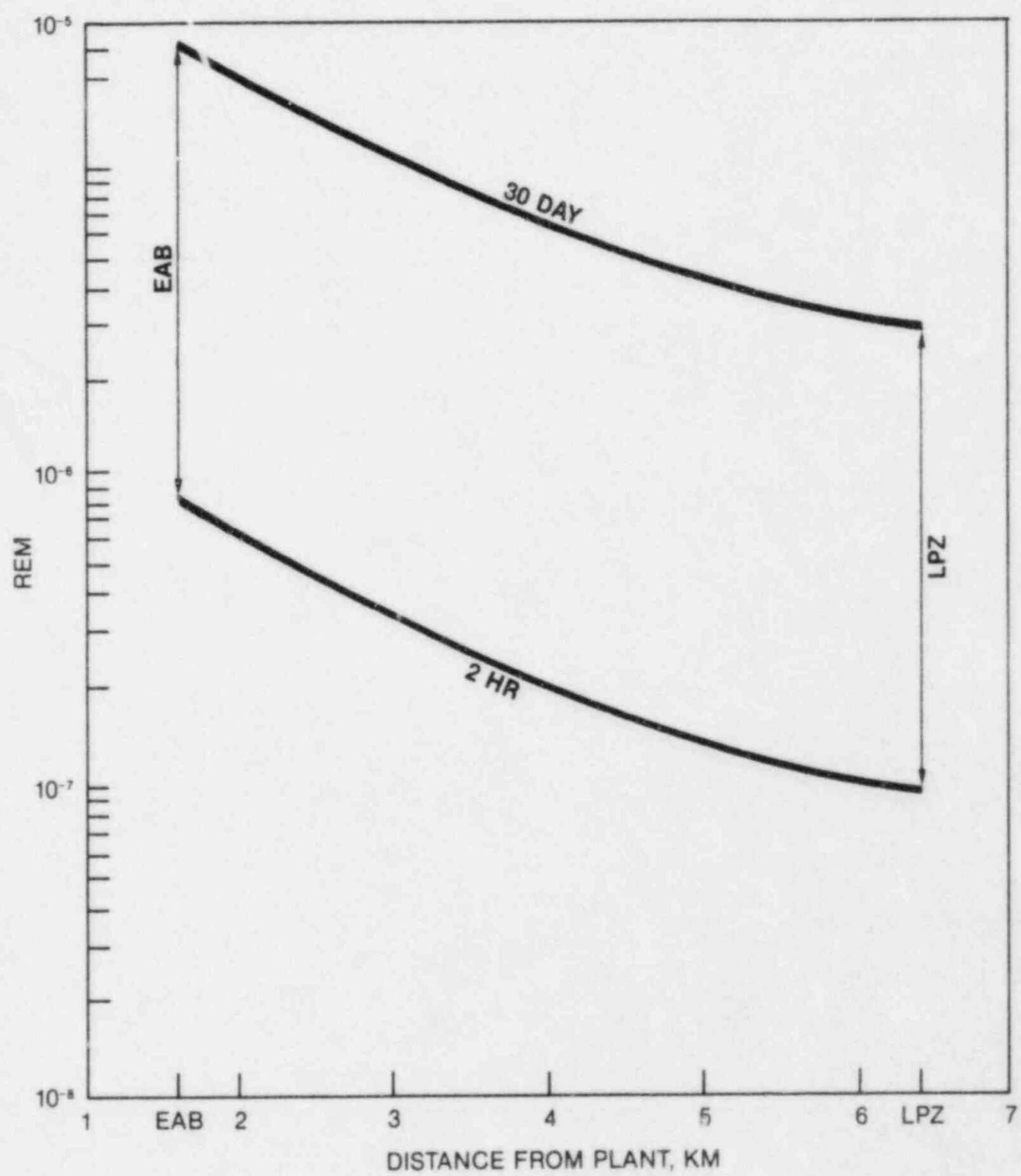
CONTROL ROD DROP ACCIDENT

CRUCIAL VARIABLES

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions
Power, MWt	4100	4100
Fuel Rods Damaged	770	770
Peaking Factor	1.5	1.0
Released from Each Rod, %		
- Halogens	50*	0.32
- Noble Gases	100*	1.8
Retained in Reactor Water, %	90	97
Valve Shut Time, sec		5.5
Halogen Carryover Fraction	1.0	0.02
Partition Factor in Condenser	100	100
Condenser Leak Rate, %/day	1.0	0.5
Turbine Building Leak, %/day	00	700

23

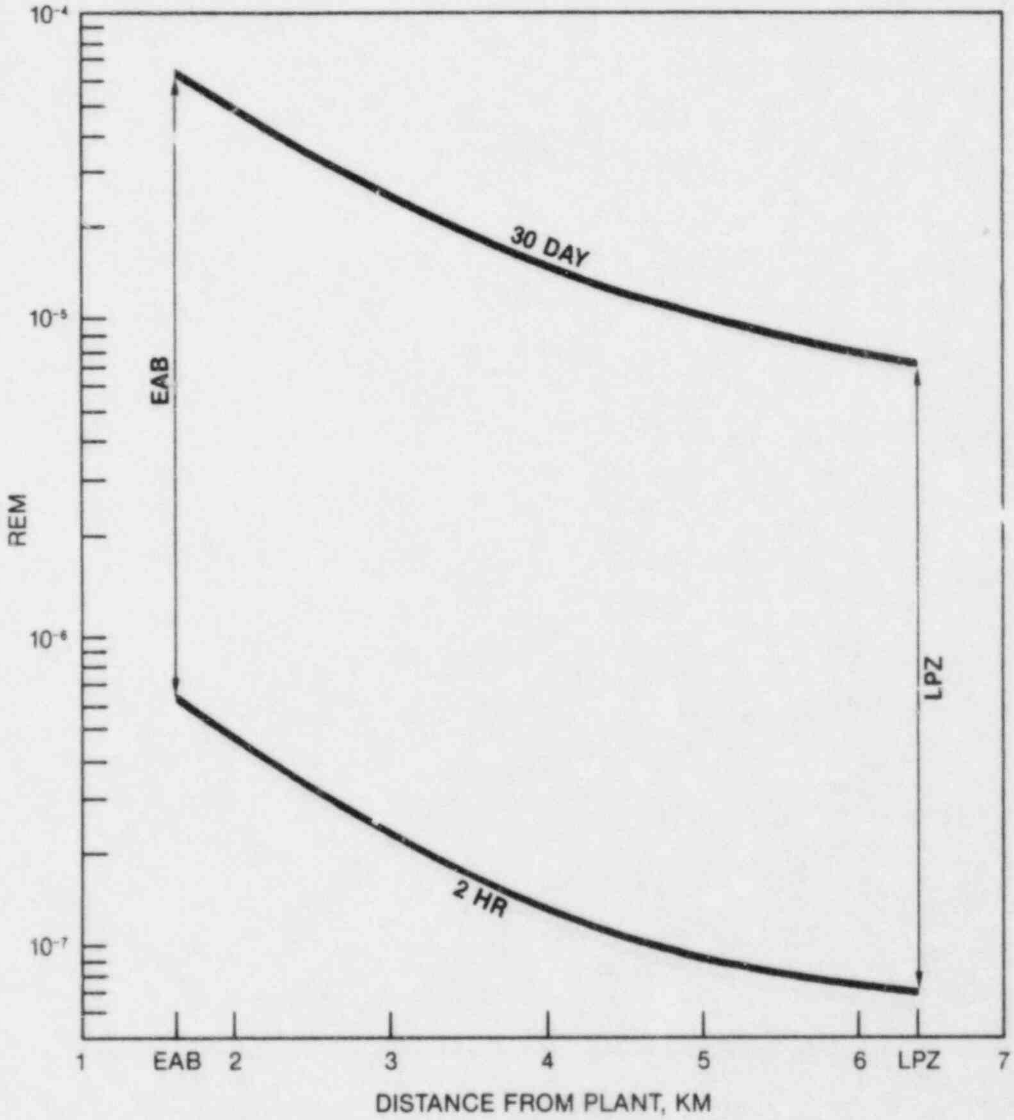
* Gap activity release, gap activity is 10% of the core activity.



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

CONTROL ROD DROP ACCIDENT (REALISTIC CASE) WHOLE-BODY DOSE, REM

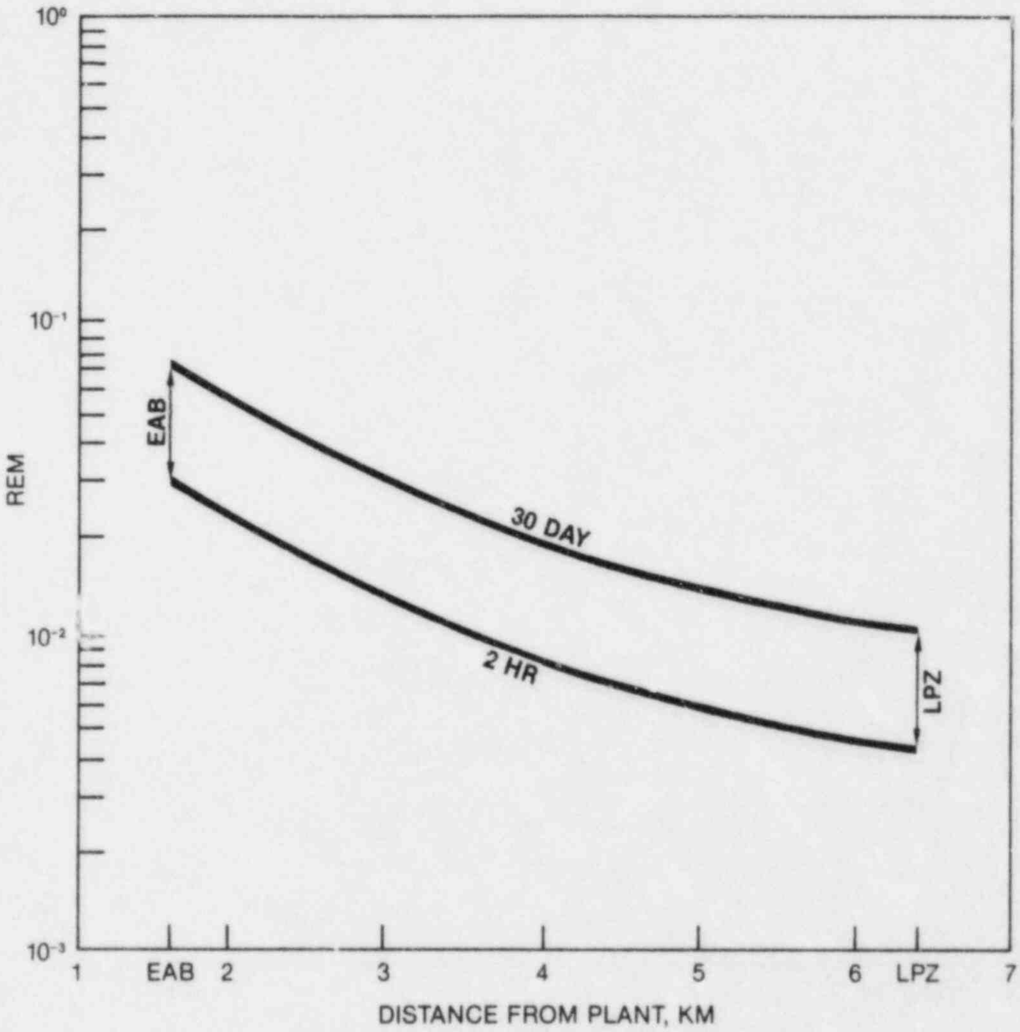
FIGURE 15.4-1



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

CONTROL ROD DROP ACCIDENT (REALISTIC CASE) THYROID DOSE, REM

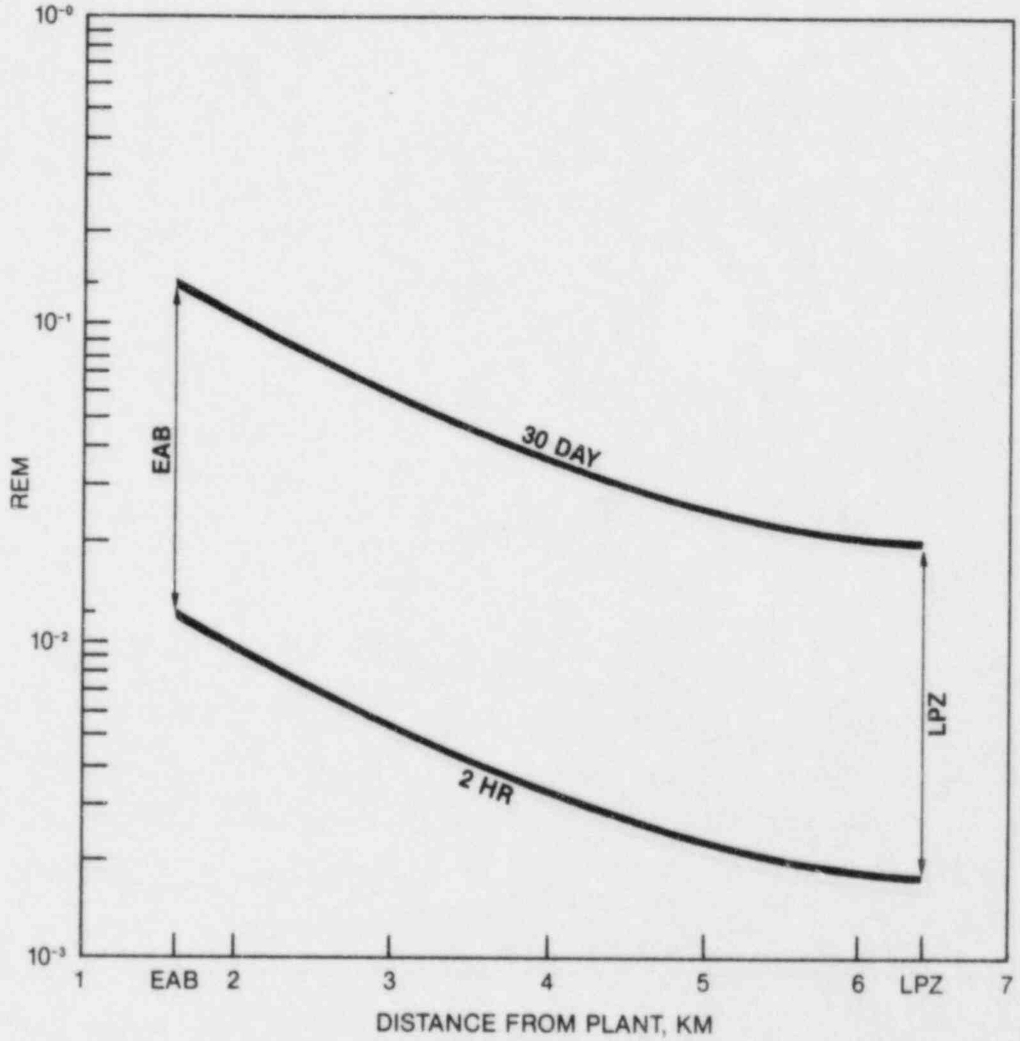
FIGURE 15.4-2



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

CONTROL ROD DROP ACCIDENT (CONSERVATIVE CASE) WHOLE-BODY DOSE, REM

FIGURE 15.4-3



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

CONTROL ROD DROP ACCIDENT (CONSERVATIVE CASE) THYROID DOSE, REM

FIGURE 15.4-4

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	23
(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	24
III. Dispersion Data			
A. EAB and LPZ distances (m)	NA	1609/6437	27
B. X/Q values in sec/m^3	NA	2.8×10^{-5}/ 3.0×10^{-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	23
D. Doses	NA	Tables 15.6-2, 15.6-3, 15.6-4	

(a) Applicable 8 hours after S/R valve transient commences

TABLE 15.6-2

TYPE III TRANSIENT ON-SITE DOSE

Organ Evaluated	Dose Effect (mrem)
Whole Body	43
Skin	119

23

TABLE 15.6-3

TYPE IV TRANSIENT ON-SITE DOSE

Organ Evaluated	Dose Effect (mrem)
Whole Body	39
Skin	545
Thyroid	0.1
Lung	0.8

23

TABLE 15.6-4

TYPE IV TRANSIENT OFF-SITE DOSE

23

Distance (m)	Dose Effect (mrem) Whole Body	Skin	Thyroid	
1609 (EAB)	4.90	2.88	2.85×10^{-2}	27
6437 (LPZ)	5.32×10^{-1}	3.13×10^{-1}	3.11×10^{-3}	

23

TABLE 15.6-7

OFF-SITE EXPOSURE[a]
(CONSERVATIVE BASIS) 23

Exposure Mode (mrem)	Dose Effect (mrem) EAB (1609m)	LPZ (6437m	
Whole Body	9.41×10^{1}	13.00	27
Skin	5.52×10^{1}	7.67	
Thyroid	5.47×10^{-1}	7.62×10^{-2}	

(a) 350,000 μCi/sec off-gas release rate
0 meter effective release height 23

TABLE 15.6-8

INSTRUMENT LINE FAILURE
ACTIVITY AIRBORNE IN THE CONTAINMENT, CURIES
(CONSERVATIVE ANALYSIS)

Isotope	[illegible]	1 Hr	2 Hr	8 Hr	1 Day	4 Days	30 Days
I-131	2.21E-2	1.32	2.63	1.32E+1	1.24E+1	9.53	9.53E-1
132	2.79E-1	1.25E+1	1.87E+1	1.36E+1	1.17E-1	5.82E-11	0
133	1.03E-1	6.0	1.16E+1	4.77E+1	2.80E+1	2.56	2.54E-9
134	5.44E-1	1.54E+1	1.42E+1	4.27E-1	1.34E-6	0	0
135	1.29E-1	6.97	1.26E+1	3.24E+1	6.22	3.71E-3	0
Total	1.08	4.23E+1	5.97E+1	1.07E+2	4.68E+1	1.21E+1	9.53E-1

23

TABLE 15.6-15

INSTRUMENT LINE BREAK
OFF-SITE DOSES, REM 23

Distance, meters	Inhalation Thyroid Dose rem	
Conservative Analysis		
1609 (EAB)	6.44×10^{-4}	27
6437 (LPZ)	9.46×10^{-5}	23
Realistic Analysis		
1609 (EAB)	2.46×10^{-7}	27
6437 (LPZ)	6.56×10^{-7}	23

TABLE 15.6-16

INSTRUMENT LINE FAILURE
CRUCIAL VARIABLES

	Conservative (NRC) Assumptions	Realistic Assumptions	
I First ten minutes			
Containment vent rate cfm	6×10^3	6×10^3	23
Containment air volume ft^3	1.76×10^6	1.76×10^6	
Vent filter efficiency %	0	0	
Iodine plateout factor	NA	2	
II Subsequent five hours			
Containment leak rate %/day	.25	.25	
Enclosure building leak rate %/day	100	100	
Recirculation flow cfm	0	0	
Recirculation filter efficiency %	0	0	
SGTS filter efficiency %	99	99	
III Dispersion Data			
A. EAB and LPZ distance (m)	1609/6437	1609/6437	27
B. X/Q values in sec/m^3	Table 15.6-17	Table 15.6-17	
			23

TABLE 15.6-17

ATMOSPHERIC DISPERSION FACTORS
(X/Q VALUES IN SEC/M^3) 23

Distance, meters	Time Period Hr	Conservative (5% X/Q)	Realistic (50% X/Q)	
1609 (EAB)	0 - 2	1.5×10^{-4}	2.8×10^{-5}	27
6437 (LPZ)	0 - 8	2.1×10^{-5}	3.0×10^{-6}	23
	8 - 24	1.4×10^{-5}	2.4×10^{-6}	
	24 - 96	5.7×10^{-6}	1.5×10^{-6}	
	96 - 720	1.6×10^{-6}	7.8×10^{-7}	

TABLE 15.6-18

STEAM LINE BREAK ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASED FROM THE BREAK (CURIES)

Isotope	Activity
I-131	3.1E-1
I-132	3.5E+0
I-133	2.2E+0
I-134	7.0E+0
I-135	3.5E+0
Kr-83m	2.0E-2
Kr-85m	3.3E-2
Kr-85	1.3E-4
Ke-87	1.0E-1
Kr-88	1.0E-1
Kr-89	4.5E-1
Xe-131m	9.4E-5
Xe-133m	1.6E-3
Xe-133	4.5E-2
Xe-135m	1.3E-1
Xe-135	1.2E-1
Xe-137	5.8E-1
Xe-138	4.5E-1

23

TABLE 15.6-19

STEAM LINE BREAK ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.6E-1
I-132	1.7E+0
I-133	1.1E+0
I-134	3.5E+0
I-135	1.7E+0
KR-83m	2.0E-2
KR-85m	3.3E-2
KR-85	1.3E-4
KR-87	1.0E-1
KR-88	1.0E-1
KR-89	4.5E-1
XE-131m	9.4E-5
XE-133m	1.6E-3
Xe-133	4.5E-2
Xe-135m	1.3E-1
Xe-135	1.2E-1
Xe-137	5.8E-1
Xe-138	4.5E-1

23

TABLE 15.6-20

STEAM LINE BREAK OUTSIDE CONTAINMENT
OFF-SITE DOSES
(REALISTIC ANALYSIS) 23

Distance, meters	Whole Body Dose, rem	Thyroid Dose, rem	
1609 (EAB)	1.19×10^{-4}	1.04×10^{-2}	27
6437 (LPZ)	1.30×10^{-5}	1.13×10^{-3}	23

TABLE 15.6-21

STEAM LINE BREAK ACCIDENT FISSION PRODUCT
RELEASE TO ENVIRONMENT
CONSERVATIVE (NRC) ANALYSIS

Isotope	Activity Released (Ci)
I-131	1.5E+0
I-132	1.7E+1
I-133	1.1E+1
I-134	3.3E+1
I-135	1.7E+1
Kr-83m	5.7E-2
Kr-85m	1.0E-1
Kr-85	3.9E-4
Kr-87	3.1E-1
Kr-88	3.1E-1
Kr-89	1.3E+0
Xe-131m	3.1E-4
Xe-133m	4.8E-3
Xe-133	1.3E-1
Xe-135m	3.9E-1
Xe-135	3.6E-1
Xe-137	1.8E+0
Xe-138	1.3E+0

23

TABLE 15.6-22

STEAM LINE BREAK OUTSIDE CONTAINMENT
OFF-SITE DOSES
(CONSERVATIVE ANALYSIS) 23

Distance, meters	Whole Body Dose, rem	Thyroid Dose, rem	
1609 (EAB)	5.59×10^{-3}	5.38×10^{-1}	27
6437 (LPZ)	7.79×10^{-4}	7.50×10^{-2}	23

TABLE 15.6-23

STEAM LINE BREAK ACCIDENT
(CONSERVATIVE CASE)
CONTROL ROOM PERSONNEL DOSES (1), REM

Sources	Skin (β)	Whole-Body (γ)	Thyroid (2)
Direct Shine	-	Insignificant	-
Immersion Dose	1.9E-5	1.5E-6	1.1E-3
Total Dose	1.9E-5	1.5E-6	1.1E-3

(1) 1000 cfm intake flow, 2000 cfm recirculation flow, filter efficiency of 99% for iodine, 10 cfm unfiltered inleakage for all time periods

(2) Breathing rate of 3.47E-4 m^3/sec for all time periods

23

TABLE 15.6-24

STEAM LINE BREAK ACCIDENT - PARAMETERS
TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions	
I. Data and assumptions used to esimate radioactive source from postulated accidents			
A Power level, MWt	4100	4100 MWt	
B. Burn-up	NA	NA	
C. Fuel damaged	None		
D. Release of activity by nuclide	Table 15.6-21	Table 15.6-19	
E. Iodine fractions			
(1) Organic	0	0	
(2) Elemental	1	1	
(3) Particulate	0	0	
F. Reactor coolant activity before the accident	15.6.5.5.2.2	15.6.5.5.2.2	
II. Data and assumptions used to estimate activity released			23
A. Containment leak rate (%/day)	NA	NA	
B. Secondary containment leak rate (%/day)	NA	NA	
C. Isolation valve closure time (sec)	5	5	
D. Adsorption and filtration efficiencies			
(1) Organic iodine	NA	NA	
(2) Elemenal iodine	NA	NA	
(3) Particulate iodine	NA	NA	
(4) Particulate fission products	NA	NA	
E. Recirculation system parameters	NA	NA	
(1) Flow rate	NA	NA	
(2) Mixing efficiency	NA	NA	
(3) Filter efficiency	NA	NA	
F. Containment spray parameters (flow rate, drop size, ect)	NA	NA	
G. Containnment volumes	NA	NA	
H. All other pertinent data and assumpions	None	None	
III. Dispersion Data			
A. EAB and LPZ distances (m)	1609/6437	1609/6437	
B. X/Q values in Sec/m^3; EAB	1.5×10^{-4}	2.8×10^{-5}	27
LPZ	2.1×10^{-5}	3.0×10^{-6}	
IV. Dose Data			
A. Method of dose calculation	Regulatory Guide 1.5	Reference 2	
B. Dose conversion assumptions	Regulatory Guide 1.5	Reference 2	23
C. Activity in containment	NA	NA	
D. Off-Site Doses	Table 15.6-22	Table 15.6-20	

TABLE 15.6-35

LOSS-OF-COOLANT ACCIDENT
OFF-SITE DOSES, REM

23

Dose Model Assumptions	Whole-Body		Thyroid	
	2-hr EAB	30-Day LPZ	2-hr EAB	30-Day LPZ
Conservative Case (Reg. Guide 1.3)	2.92	1.16	20.6	12.8
Realistic Case	2.3×10^{-7}	1.12×10^{-6}	3.31×10^{-6}	1.94×10^{-6}
Mechanistic Fission Product Distribution	3.02	2.65	25.6	34.2

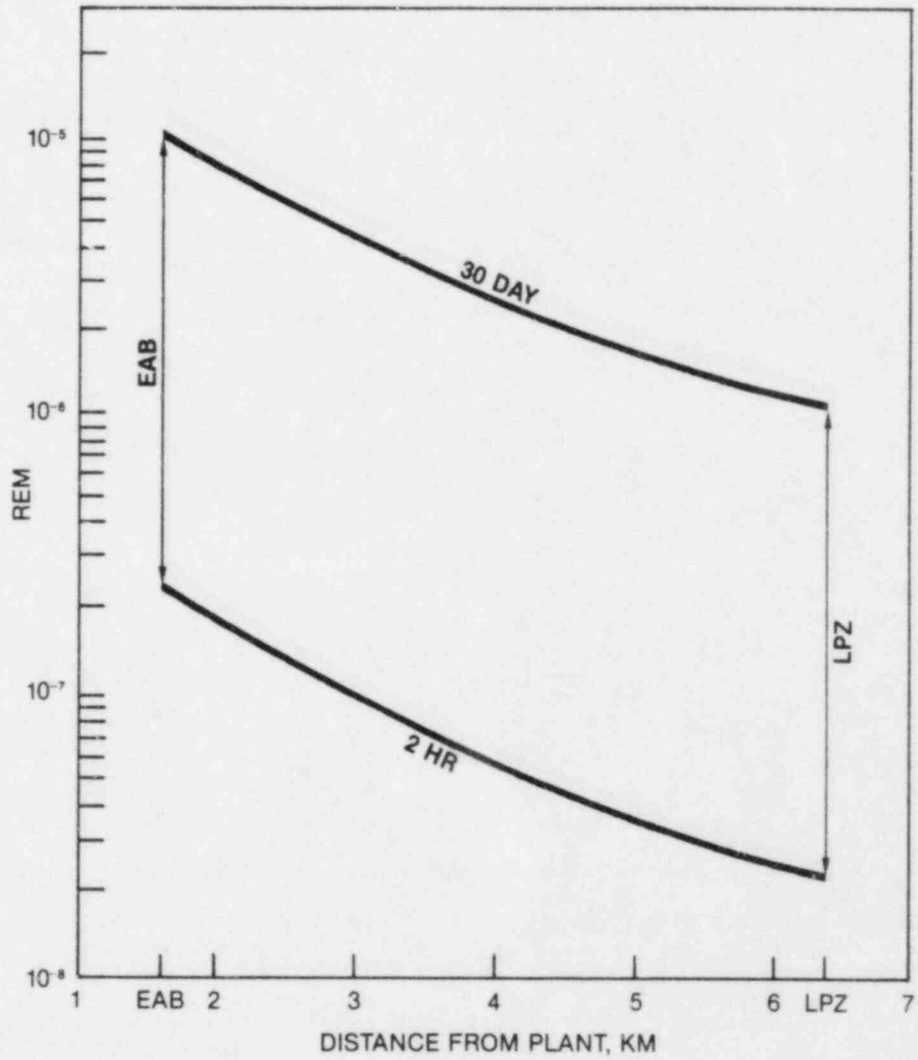
27

TABLE 15.6-36

LOSS-OF-COOLANT ACCIDENT
PARAMETERS TO BE TABULATED* FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions	
I. Data and assumptions used to estimate radioactive source from postulated accidents			
A. Power level	4100 MWt	4100 MWt	
B. Burn-up	NA	NA	
C. Fuel damaged	100%	None	
D. Activity in Containment	Table 15.6-25	Table 15.6-27	
E. Iodine fractions			
(1) Organic	4%	1%	
(2) Elemental	91%	99%	
(3) Particulate	5%	0	
F. Reactor coolant activity before the accident	15.6.5.5.1.2	15.6.5.5.1.2	
II. Data and assumptions used to estimate activity released			
A. Containment leak rate (%/day)	0.25	0.25	23
B. Secondary containment leak rate (%/day)	NA	100	
C. Valve movement times	NA	NA	
D. Adsorption and filtration efficiencies			
(1) Organic iodine	99%	99%	
(2) Elemental iodine	99%	99%	
(3) Particulate iodine	99%	99%	
(4) Particulate fission products	99%	99%	
E. Recirculation system parameters			
(1) Flow rate	NA	NA	
(2) Mixing efficiency	NA	NA	
(3) Filter efficiency	NA	NA	
F. Containment spray parameters (flow rate, drop size, etc.)	Section 6.2.3	Section 6.2.3	
G. Containment volumes	Table 6.2-1	Table 6.2-1	
H. All other pertinent data and assumptions	Table 15.6-32	Table 15.6-32	
III. Dispersion Data			
A. EAB and LPZ distances (m)	1609/6437	1609/6437	27
B. X/Q values in Sec/m^3	Table 15.6-17	Table 15.6-17	
IV. Dose Data			
A. Method of dose calculation	Regulatory Guide 1.3	Reference 3	
B. Dose conversion assumptions	Regulatory Guide 1.3	Reference 3	23
C. Activity in released to the environs	Table 15.6-26	Table 15.6-29	
D. Off-Site Doses	Figures 15.6-3 & 15.6-4 Table 15.6-35	Figures 15.6-1 & 15.6-2 15.6-35	

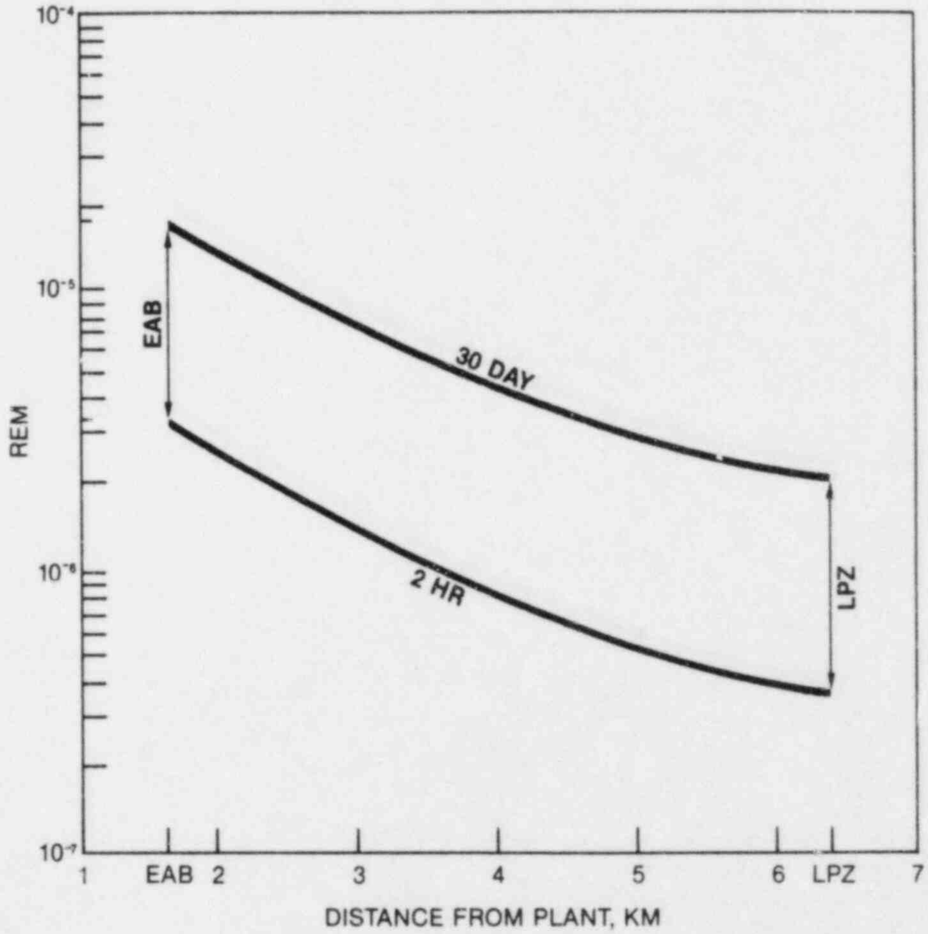
* As applicable to the event being described.



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

LOCA-REALISTIC CASE WHOLE-BODY DOSE, REM

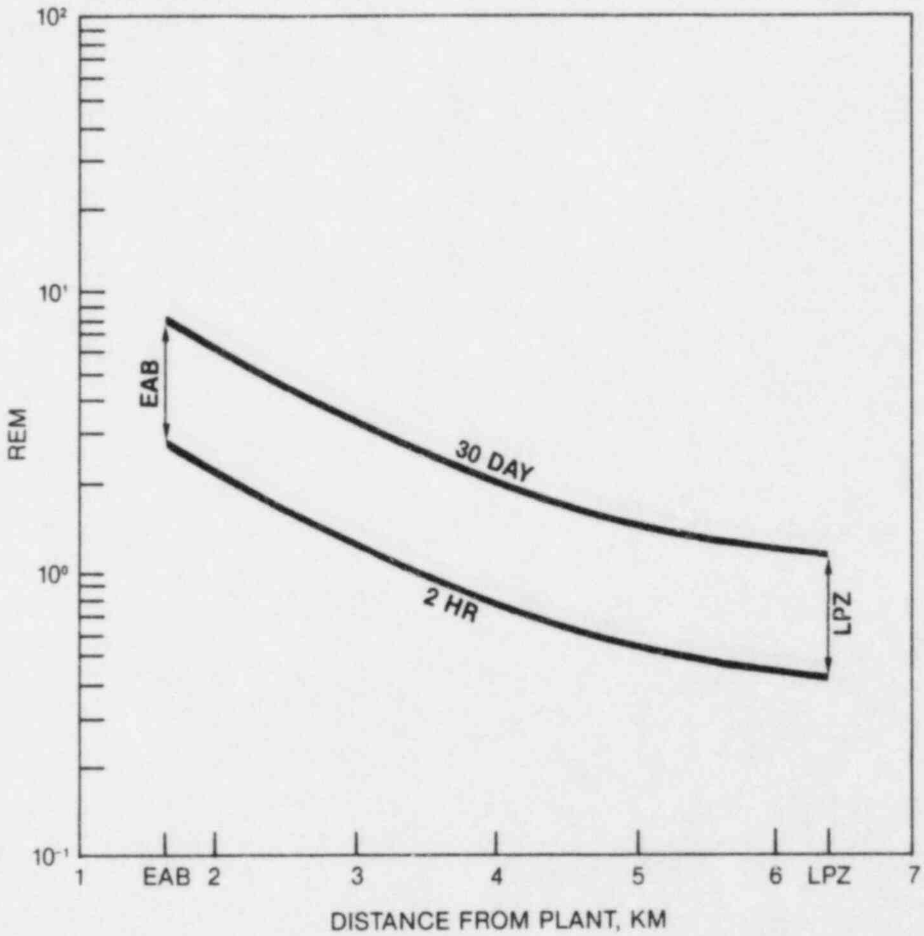
FIGURE 15.6-1



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

LOCA-REALISTIC CASE
INHALATION THYROID DOSE, REM

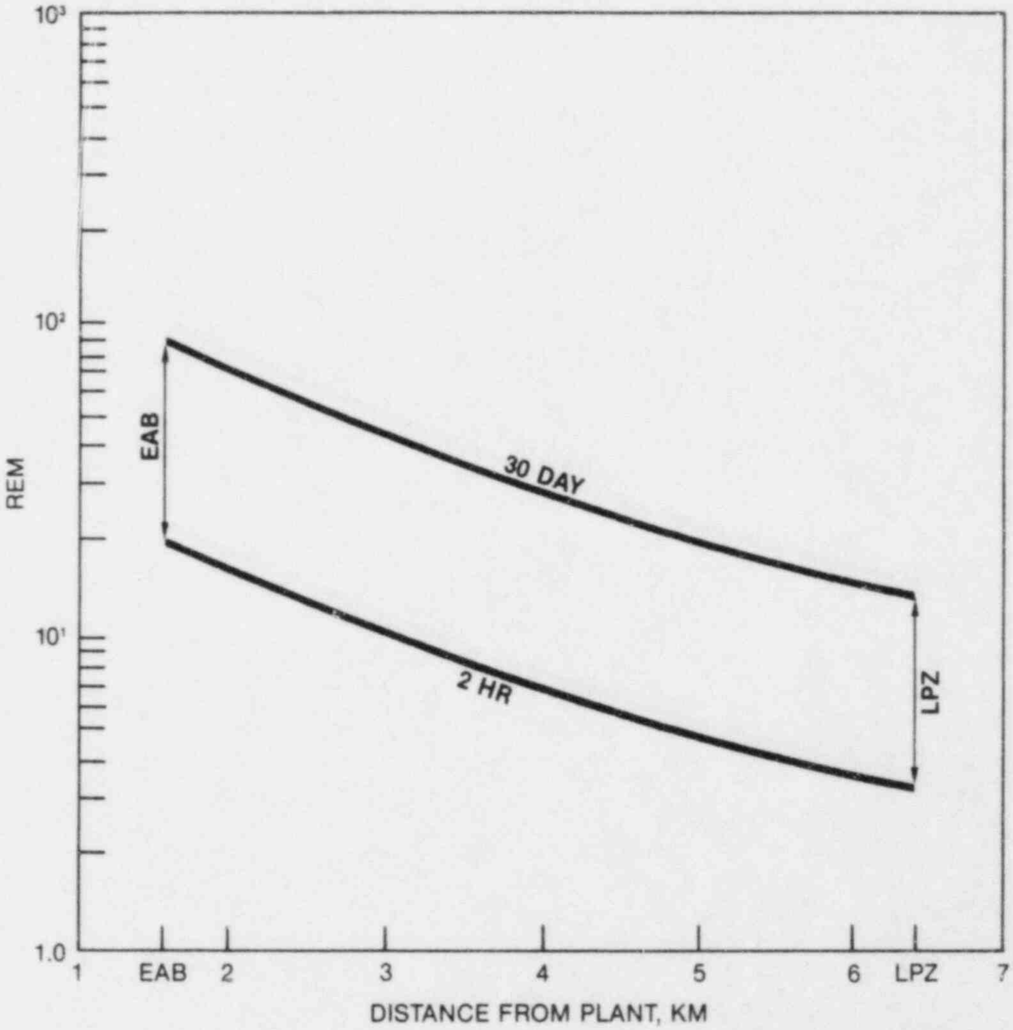
FIGURE 15.6-2



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

LOCA-CONSERVATIVE CASE WHOLE-BODY DOSE, REM

FIGURE 15.6-3



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

LOCA-CONSERVATIVE CASE INHALATION THYROID DOSE, REM

FIGURE 15.6-4

TABLE 15.7-3

OFF-SITE DOSE FROM OFFGAS
SYSTEM FAILURE

(Conservative Analysis) 23

Distance, meters	Whole-Body Dose, rem	
EAB (1609 m)	1.64E-1	27
LPZ (6437 m)	2.27E-2	23

TABLE 15.7-4

OFF-SITE DOSE FROM OFFGAS SYSTEM FAILURE

(Realistic Analysis) 23

	Dose in rem	
	EAB (1609 m)	LPZ (6437 m)
Whole-Body	6.46E-3	7.04E-4
Thyroid	5.41E-5	5.89E-6
Bone	3.24E-4	3.52E-5
Lung	1.30E-3	1.42E-4
G.I.	1.63E-2	1.77E-3

27

TABLE 15.7-5

GASEOUS RADWASTE SYSTEM FAILURE PARAMETERS
TO BE TABULATED* FOR POSTULATED [illegible] ANALYSES

		Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions	
I.	Data and assumptions used to estimate radioactive source from postulated accidents			
	A. Power Level	4100 MWt	4100 MWt	
	B. Burn-up	NA	NA	
	C. Fuel damaged	None	None	
	D. Release of activity by nuclide	Table 15.7-1	251 NSSS GESSAR Table 15.1.36-1 and Table 15.7-2	
	E. Iodine fractions	NA		
	(1) Organic	NA	0	
	(2) Elemental	NA	1	23
	(3) Particulate	NA	0	
	F. Reactor coolant activity before the accident	15.6.5.5.2.2	15.6.5.5.2.2	
II.	Data and assumptions used to estimate activity released			
	A. Containment leak rate (%/day)	NA	NA	
	B. Secondary containment leak rate (%/day)	NA	NA	
	C. Valve movement times	NA	NA	
	D. Adsorption and filtration efficiencies	NA	NA	
	(1) Organic iodine	NA	NA	
	(2) Elemented iodine	NA	NA	
	(3) Particulate iodine	NA	NA	
	(4) Particulate fission products	NA	NA	
	E. Recirculation system parameters	NA	NA	
	(1) Flow Rate	NA	NA	
	(2) Mixing Efficiency	NA	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	None	None	
III.	Dispersion Data			
	A. EAB and LPZ distances (m)	1609/6437	1609/6437	
	B. X/Q values in sec/m^3	$1.5E \times 10^{-4}$/2.1E-5	2.8×10^{-5}/3.0E-6	27
IV.	Dose Data			
	A. Method of dose calculation	Appendix 15A	Appendix 15A	
	B. Dose conversion assumptions	Appendix 15A	Appendix 15A	
	C. Activity in Containment	NA	NA	23
	D. Doses	Table 15.7-3	Table 15.7-4	

* As applicable to the event being described.

TABLE 15.7-6

FAILURE OF AIR EJECTOR LINES
ACTIVITY RELEASED TO ENVIRONMENT

(Realistic Case)

Isotope	Activity Release (Ci)
I-131	3.2E-3
I-132	3.2E-2
I-133	2.1E-2
I-134	6.7E-2
I-135	3.2E-2
Kr-83M	3.1E+0
Kr-85M	5.5E+0
Kr-85	2.1E-2
Kr-87	1.7E+1
Kr-88	1.8E+1
Kr-89	7.4E+1
Kr-90	1.9E+1
Xe-131M	1.4E-2
Xe-133M	2.5E-1
Xe-133	7.3E+0
Xe-135M	2.2E+1
Xe-137	9.7E+1
Xe-138	7.3E+1
Xe-139	3.2E+1

23

TABLE 15.7-7

FAILURE OF AIR EJECTOR LINES
OFF-SITE RADIOLOGICAL DOSES

(Realistic Case) 23

Distance (meters)	Whole-Body (rem)	Thyroid (rem)	
1609 (EAB)	3.15×10^{-3}	1.81×10^{-4}	27
6437 (LPZ)	3.43×10^{-4}	1.97×10^{-5}	23

TABLE 15.7-8

FAILURE OF AIR EJECTOR LINES - PARAMETERS TO BE TABULATED* FOR POSTULATED ACCIDENT ANALYSES

	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	Table 15.7-5	
E. Iodine fractions			
(1) Organic	NA	0	
(2) Elemental	NA	1	
(3) Particulate	NA	0	
F. Reactor coolant activity before the accident	NA	15.6.5.5.2.2	23
II. Data and assumptions used to estimate activity released			
A. Containment leak rate (%/day)	NA	NA	
B. Secondary containment leak rate (%/day)	NA	NA	
C. Valve movement times	NA	NA	
D. Adsorption and filtration efficiencies	NA	NA	
(1) Organic iodine	NA	NA	
(2) Elemental iodine	NA	NA	
(3) Particulate iodine	NA	NA	
(4) Particulate fission products	NA	NA	
E. Recirculation system parameters			
(1) Flow rate	NA	NA	
(2) Mixing efficiency	NA	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	None	
III. Dispersion Data			
A. EAB and LPZ distances (m)	NA	1609/6437	
B. X/Q values in sec/m^3	NA	$2.8 \times 10^{-5}/3.0\times10^{-6}$	27
IV. Dose Data			
A. Method of dose calculation	NA	Reference 1	
B. Dose conversion assumptions	NA	Reference 1	
C. Activity in containment	NA	NA	
D. Off-Site doses	NA	Table 15.7-7	23

* As applicable to the event being described.

TABLE 15.7-11

LIQUID RADWASTE TANK RUPTURE
OFF-SITE DOSES

Distance (meters)	Inhalation Thyroid Dose (rem)	
Conservative Analysis		23
EAB (1609)	1.71×10^{-3}	27
LPZ (6437)	2.38×10^{-4}	23
Realistic Analysis		
EAB (1609)	4.21×10^{-5}	27
LPZ (6437)	4.54×10^{-6}	23

TABLE 15.7-12

LIQUID RADWASTE TANK FAILURE: PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions	
I. Data and assumptions used to estimate radioactive source from postulated accidents			
A. Power level	NA	NA	
B. Burn-up	NA	NA	
C. Fission products released from fuel (fuel damaged)	NA	NA	
D. Release of activity by nuclide	Table 15.7-10	Table 15.7-10	
E. Iodine fractions			
(1) Organic	0.01	0.01	
(2) Elemental	0.01	0.01	
(3) Particulate	0.01	0.01	23
F. Reactor coolant activity before the accident	NA	NA	
II. Data and assumptions used to estimate activity released			
A. Containment leak rate (%/day)	NA	NA	
B. Secondary containment release rate (%/day)	NA	NA	
C. Valve movement times	NA	NA	
D. Adsorption and filtration efficiencies	NA	NA	
(1) Organic iodine	NA	NA	
(2) Elemented iodine	NA	NA	
(3) Particulate iodine	NA	NA	
(4) Particulate fission products	NA	NA	
E. Recirculation system parameters	NA	NA	
(1) Flow rate	NA	NA	
(2) Mixing efficiency	NA	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			
(1) Dilution factor afforded by public waterway	NA	NA	
(2) Dilution of liquid ingestion	NA	NA	
(3) Aquatic life consumed	NA	NA	
III. Dispersion data			
A. EAB and LPZ distances (m)	1609/6437	1609/6437	
B. X/Q values in sec/m^3	1.5×10^{-4}/2.1E-5	2.8×10^{-5}/3.0E-6	27
IV. Dose data			
A. Method of dose calculation	Appendix 15A	Appendix 15A	
B. Dose conversion assumptions	Appendix 15A	Appendix 15A	
C. Peak activity concentrations in containment	NA	NA	23
D. Doses	Table 15.7-11	Tables 15.7-11	

TABLE 15.7-15

FUEL HANDLING ACCIDENT
OFF-SITE RADIOLOGICAL EXPOSURES

(Realistic Analysis) 23

Distance (meters)	Whole-Body (rem)	Thyroid (rem)	
1609 (EAB)	5.41×10^{-4}	2.97×10^{-4}	27
6437 (LPZ)	3.30×10^{-4}	2.05×10^{-4}	23

TABLE 15.7-16

FUEL HANDLING ACCIDENT

(Conservative Analysis)

Isotope	Activity Airborne in Refueling Building, Ci
I-131	2.58E+2
I-132	4.03E-1
I-133	2.92E+2
I-134	-
I-135	4.95E+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 15.7-17

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)	
1609 (EAB)	8.96×10^{-3}	1.34	27
6437 (LPZ)	1.24×10^{-3}	1.86×10^{-1}	23

TABLE 15.7-19

FUEL HANDLING ACCIDENT
CONTROL ROOM PERSONNEL DOSES[1], REM

(Conservative Case)

Sources	Skin (β)	Whole-Body (γ)	Thyroid[2]
Direct Shine	----	1.4E-2	----
Immersion Dose	9.0E-2	3.3E-3	8.3E-4
Total Dose	9.0E-2	1.7E-2	8.3E-4

23

(1) 1000 cfm intake flow, 2000 cfm recirculation flow, filter efficiency of 99% for iodine, 10 cfm unfiltered inleakage for all time periods.

(2) Breathing rate of 3.47E-4 m^3/sec for all time periods.

TABLE 15.7-20

FUEL HANDLING ACCIDENT - PARAMETERS
TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions	Realistic (Conservative Engineering) Assumptions	
I. Data and assumptions used to estimate radioactive source from postulated accidents			
A. Power level	NA	NA	
B. Burn-up factor	1.5	1.0	
C. Fuel damaged	98 rods	98 rods	
D. Release of activity by nuclide	10% noble gas, 10% iodine, 30% Kr-85	15.7.4.5.2.2.2	
E. Iodine fractions			
(1) Organic	0.25%	0	
(2) Elemental	99.75%	1	
(3) Particulate	0	0	23
F. Reactor coolant activity before the accident	NA	NA	
II. Data and assumptions used to estimate activity released			
A. Refueling building release rate	NA	100%/day	
B. Secondary containment release rate (%/day)	NA	NA	
C. Valve movement times	NA	NA	
D. Adsorption and filtration efficiencies			
(1) Organic iodine	99%	99%	
(2) Elemental iodine	99%	99%	
E. Recirculation system parameters			
(1) Flow rate	NA	NA	
(2) Mixing efficiency	NA	NA	
(3) Filter efficiency	NA	NA	
F. Containment spray parameters (flow rate, drop size, etc)			
G. Containment volumes	NA	NA	
H. All other pertinent data and assumptions	None	None	
III. Dispersion data			
A. EAB and LPZ distances (m)	1609/6437	1609/6437	27
B. X/Q values in sec/m^3	Table 15.6-17	Table 15.6-17	
IV. Dose data			
A. Method of dose calculation	Regulatory Guide 1.25	Reference 1	
B. Dose conversion assumptions	Regulatory Guide 1.25	Reference 1	23
C. Activity in Refueling Building	Table 15.7-16	Table 15.7-13	
D. Off-Site doses	Table 15.7-18	Tables 15.7-15	