INSERTION INSTRUCTIONS FOR AMENDMENT 2

Remove old pages and insert Amendment 2 pages as instructed below.

Amendment 2 text, tables, and figures have the words "Amendment 2 August 1983" at the bottom of the page. A few unrevised pages have been reprinted because they fall within a run of closely spaced revised pages.

Transmittal letters along with these insertion instructions should either be filed or entered in Volume 1 in front of any existing letters, instructions, distribution lists, etc.

LEGEND

Remove/Insert Columns

Entries beginning with "T" or "F" designate table or figure numbers, respectively. All other entries are page numbers:

T2.3-14 = Table 2.3-14 F5.4-3 = Figure 5.4-3

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Pages printed back to back are indicated by a "/":

1.2-5/1.2-6 = Page 1.2-5 backed by Page 1.2-6

T2.3-14(5 of 5)/T2.3-15(1 of 3) = Table 2.3-14, sheet 5 of 5, backed by Table 2.3-15, sheet 1 of 3

Location Column

Ch = Chapter, S = Section, Ap = Appendix

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	VOLUME 1	
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EP2-1/EP2-2 EP2-3/blank	EP2-1/EP2-2 EP2-3/blank	After Ch 2 tab
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EP12-1/blank	EP12-1/blank	After Ch 12 tab
T12.1-1 (1 of 5)/(2 of 5) thru (5 of 5)/blank	T12.1-1 (1 of 5)/(2 of 5) thru (5 of 5)/blank	S 12.1
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LIST OF EFFECTIVE PAGES (Cont)

2.2 ECOLOGY

Intensive terrestrial ecological studies were conducted on the site from April 1974 to June 1975. Results are detailed in the NUS Corporation (1976a) report on the studies. The material reported in Section 2.2.1 and the accompanying tables and figures are primarily new information, collected after the publication of the Environmental Report - Construction Permit Stage (ER-CPS). Baseline aquatic ecological studies were conducted at the site (Beaver Valley Power Station - Unit 1 (BVPS-1) preoperational studies) and were discussed in the Beaver Valley Power Station - Unit 2 (BVPS-2) ER-CPS. Aquatic ecological information presented in Section 2.2.2 is based on studies conducted from 1976 to the present.

2.2.1 Terrestrial Ecology

The Beaver Valley Power Station (BVPS) site encompasses approximately 501 acres of the Ohio River floodplain and adjacent uplands of Beaver County, Pennsylvania. The county receives about 35 inches or more of rainfall annually, and the frost-free season averages 241 days (FSAR Section 2.3.1.1). Deciduous forest covers more than half of the site while existing power plants and associated facilities occupy less than 40 percent. Approximately 7 percent of the site consists largely of scrubland, old fields, and paved roads. (Table 2.2-1, Figure 2.2-1).

2.2.1.1 Soils

Most soils on the site have been formed from residual material weathered from sandstone, siltstone, and shale. Notable exceptions are the soils formed from materials transported and deposited by stream action (Table 2.2-2). Depth to bedrock on the ridge to the south of the facility varies from 20 inches to greater than 60 inches. Most soils are fine textured loams, silt loams, or silty clay loams, are well drained, and are moderately permeable (Table 2.2-2). Much of the site is on steep slopes on which soils are liable to erode easily when denuded of vegetation. The soils underlaying the facility proper are discussed in FSAR Section 2.5.

With few exceptions, the soils are marginal for crop production because of the steep slopes. Woodland productivity and wildlife potential, however, are fair to good on most soils (Table 2.2-3).

The distribution of various soils on the BVPS site is shown on Figure 2.2-2. The acreages and potential land use of the soils on the site and in the county are presented in Table 2.2-3.

2.2.1.2 Vegetation

The site has a proportion of wooded land similar to that of Beaver County. Much of the deciduous forest on the site is in early successional or subclimax stages due to man-induced and natural

perturbations. The structure and composition of the site vegetation (Tables 2.2-4 and 2.2-5) have been greatly influenced by spoil banks from coal mining, abandoned road beds, past farming on the more level uplands, maintenance of transmission corridors and pipelines, and selective logging. Natural perturbations which also affect the floral community on the site include the locust leaf miner (prevalent during 1977 and 1978), Dutch elm disease, ice, and wind storms.

The ridge tops are dominated by black locust and black cherry, east- to south-facing slopes by mixed oak and sugar maple, south- to southwest-facing slopes by mixed oak and mountain laurel, lower slopes by beech, and the floodplain by silver maple and sycamore. The north-facing slopes are covered by second growth mixed mesophytic forests (Table 2.2-1, Figure 2.2-1). The amount of vegetative cover (Table 2.2-4) varies with slope, aspect, and land use. The tree canopy of slope forests generally shades 60 to 80 percent of the ground. Shrub cover is low (10 to 20 percent) except in more disturbed forests where it increases to 60 percent under a sparse tree canopy. Herbaceous cover is substantially higher in mixed mesophytic communities than in other upland forest communities. More disturbed areas such as the floodplain and transmission corridors also support a heavy herbaceous cover. No endangered or threatened plants (U.S. Department of Interior 1980) occur on the site.

2.2.1.3 Mammals

About 47 species of mammals have geographic ranges that include the site (Table 2.2-6). Of these mammals, 27 were seen, captured, or evidence of their presence was noted. No endangered mammals (U.S. Department of Interior 1980) occur on the site. Game species and furbearers are discussed in Section 2.2.1.5.

Most of the common mammals on the site are characteristic of wooded and shrubby areas in southwestern Pennsylvania. The site provides few open habitats for mammals, except along transmission corridors or adjacent to construction areas.

Tables 2.2-7 through 2.2-9 provide information on count and distribution of cows, doe goats, and beef cattle (Porter Consultants 1981). These are the major groups of domestic fauna that may be involved in the radiological exposure of man via the iodine-milk route. These tables show animal counts by compass direction in one-mile increments from the midpoint of the BVPS-1 and BVPS-2 reactors.

2.2.1.4 Birds

About 220 species of birds may be expected in southwestern Pennsylvania in habitats similar to those on, or immediately adjacent to, the site (Table 2.2-10). During terrestrial ecological studies on the site (NUS Corporation 1976a), 112 bird species were identified on the site or in the site region (Tables 2.2-10 through 2.2-13). Of these, 20 were year-round residents, 41 were summer visitors, 13 were

TABLE 2.2-1

CLASSIFICATION OF SITE ACREAGE BY VEGETATION TYPE AND LAND USE

Type	Subtype	Acreage*		Percent of Site		Vegetation Study Areas**
Deciduous forest	Black locust - Black cherry	93.2		18.6		4,16,18
	Mixed mesophytic	58.9		11.7		3
	Mixed oak - Sugar maple	87.9		17.5		7,10,11,12,13,14,17
	Beech - Maple	11.8		2.4		8,15
	Black locust - Tree-of-heaven	1.9		0.4		2
	Mountain laurel - Mixed oak	11.3		2.2		6,9
	Silver maple - Sycamore	8.3		1.7		1
			(273.3)		(54.5)	
Scrubland	Mountain laurel - Hawthorn	2.4		0.5		5
	Old field	3.5		0.7		
			(5.9)		(1.2)	
Power plant and associated facilities	Transmission corridors	53.8		10.7		
	Unpaved roads (not on transmission corridors)	3.1		0.6		
	Pipeline	1.9		0.4		
	Power plant	106.6		21.3		
	Spoil areas	29.0		5.8		
			(194.4)		(38.8)	
Other disturbed areas	Rights-of-way (paved roads)	25.9		5.2		
	Oil tank	0.2		<0.1		
	Abandoned single-family dwelling (removed)	1.5		0.3		
			(27.6)		(5.5)	
Total		501.2		100.0		

NOTES:

*Subtotals (in parentheses) may not add due to rounding.
**Indicated on Figure 2.2-1.

TABLE 2.2-2

CHARACTERISTICS OF SOILS ONSITE THAT INFLUENCE THE DISTRIBUTION OF PLANTS AND PLANT COMMUNITIES [1,2,3]

Map Symbols [4] and Soil Series	Parent Material [5]	Topographic Setting	Texture of Surface Horizon	Depth to Seasonal High Water Table (feet)	Depth to Bedrock (feet) [6]	Range of Permeability (inches per hour) [7]	Range of Available Moisture Capacity (inches per inch of depth) [8]	Reaction Range (pH)	Erodibility Class [9]
Pope 02A1,2A1	Stream deposits (alluvium washed from uplands underlain by shale and sandstone)	Floodplains; <4% slopes	Silt loam	3+	5+	0.6-2.0	0.14-0.18	5.5-6.5	High
Gilpin 13EF2, 14CZ,14D2, 34DE6	Residual material weathered from acid shale and fine grained sandstone	Gently sloping steep convex dissected uplands; 3-100% (mostly 3-80%) slopes	Shaly silt loam	6+	1.5-3.5	0.6-6.0	0.18-0.24	5.1-5.5	Medium
Vandergrift 34DE6	Residual material weathered from clay shale	Uplands; 15-35% slopes	Silty clay loam	0.5-3	5+	0.6-2.0	0.16-0.20	5.0-6.0	Medium
Wellston 45B2,45CZ	Residual material weathered from shale, thin bedded sandstone or siltstone; or wind-blown (loess) deposits	Gently sloping to steep uplands; 2-30% (mostly 4-18%) slopes	Silt loam	6+	3.5-5	0.6-2.0	0.14-0.18	[illegible].5-5.5	High

TABLE 2.2-2 (Cont)

Map Symbols [4] and Soil Series	Parent Material [5]	Topographic Setting	Texture of Surface Horizon	Depth to Seasonal High Water Table (feet)	Depth to Bedrock (feet) [6]	Range of Permeability (inches per hour) [7]	Range of Available Moisture Capacity (inches per inch of depth) [8]	Reaction Range (pH)	Erodibility Class [9]
Wharton 57C2,57D2	Residual material weathered from clay shale and some sandstone	Nearly level to sloping broad ridges, benches and concave hillside slopes 2-20% slopes	Silt loam	1.5-3	4+	0.6-2.0	0.18-0.24	5.1-6.0	Medium
Hazelton 60D2	Residual material weathered from sandstone and shale	Uplands; 3-25% slopes	Channery loam [10]	6+	3.5-6	2.0-6.0	0.12-0.16	4.5-5.0	Low
Conotton 73B2,73D2	Water deposited sand and gravel [5]	Stream terraces; less than 25% slopes	Gravelly loam	5+	5+	2.0-6.0	0.13-0.15	5.1-6.5	Medium
Weikert [11] 99,13EF2	Residual material weathered from shale and sandstone	Gently sloping to very steep convex dissected uplands and rock outcrops; 5-150% (mostly 5-80%) slopes	Shaly silt loam	6+	1-1.5	2.0-6.0	0.08-0.14	4.5-5.5	Medium
Allegheny 340B2	Deposits of stratified sand, silt and clay	Stream terraces; 3-15% slopes	Silt loam	6+	5+	2.0-6.0	0.18-0.24	4.5-5.5	Medium

TABLE 2.2-2 (Cont)

Map Symbols [4] and Soil Series	Parent Material [5]	Topographic Setting	Texture of Surface Horizon	Depth to Seasonal High Water Table (feet)	Depth to Bedrock (feet) [6]	Range of Permeability (inches per hour) [7]	Range of Available Moisture Capacity (inches per inch of depth) [8]	Reaction Range (pH)	Erodibility Class [9]
Urban land MAAB, MSCD, MSAEF, MTAB, MTCD	Variable, depending on native parent material and nature of disturbance from construction and earth-moving activities [5]	Variable	Variable	Variable	Variable	Variable	Variable	Variable	Variable

NOTES:

1. U.S. Department of Agriculture 1969-1972.
2. U.S. Department of Agriculture, Soil Conservation Service 1973.
3. U.S. Department of Agriculture, Soil Conservation Service 1974.
4. Distribution of soils on the BVPS site is indicated on Figure 2.2-2.
5. Refer to FSAR Section 2.5.
6. Surficial soils only (information on deeper soils is found in FSAR Section 2.5).
7. Rate of water movement through a saturated soil; surface horizon only.
8. Amount of water in the soil which can be extracted and used by plants; surface horizon only.
9. Ease with which a soil erodes.
10. Thin flat pieces of sandstone 2 to 15 mm long.
11. Descriptions not applicable to rock outcrop phase (Map Symbol 99, Figure 2.2-2).

TABLE 2.2-3

POTENTIAL LAND USE OF SOILS FOUND ON THE SITE[1]

Map Symbol[2]	Soil Name	Site Acreage[3]	Beaver County Acreage[4]	Woodland Site Quality[5]	Cropland Capability Class and Subclass[6]	Productivity Potential Rating[7]		Potential as Wildlife Habitat[8] for		
						Corn	Alfalfa	Openland Wildlife	Woodland Wildlife	Wetland Wildlife
O2A1	Pope silt loam, high bottom (0-3% slopes)	1.5	446	Excellent	I	Excellent	Excellent	Good	Good	Good
2A1	Pope silt loam (0-3% slopes)	4.1	901	Excellent	I	Excellent	Excellent	Good	Good	Very poor
13EF2	Gilpin-Weikert shaly silt loams (25-100% slopes)	175.5	49,336	Fair	VIIe	Un-suitable	Un-suitable	Poor	Poor	Very poor
14C2	Gilpin shaly silt loam (8-15% slopes)	35.7	14,741	Very good	IIIe	Fair	Fair	Good	Good	Very poor
14D2	Gilpin shaly silt loam (15-25% slopes)	10.6	11,615	Very good	IVe	Fair	Fair	Fair	Good	Very poor
34DE6	Vandergrift-Gil-pin complex (15-35% slopes)	5.1	1,8[illegible]2	Good	VIIe	Un-suitable	Un-suitable	Poor	Good	Very poor
45B2	Wellston silt loam (3-8% slopes)	71.2	1,099	Very good	IIe	Excellent	Good	Good	Good	Very poor
45C2	Wellston silt loam (8-15% slopes)	10.2	228	Very good	IIIe	Good	Good	Good	Good	Very poor
57C2	Wharton silt loam (8-15% slopes)	7.1	9,121	Very good	IIIe	Fair	Fair	Good	Good	Very poor

TABLE 2.2-3 (Cont)

Map Symbol[2]	Soil Name	Site Acreage[3]	Beaver County Acreage[4]	Woodland Site Quality[5]	Cropland Capability Class and Subclass[6]	Productivity Potential Rating[7] Corn	Productivity Potential Rating[7] Alfalfa	Potential as Wildlife Habitat[8] for Openland Wildlife	Potential as Wildlife Habitat[8] for Woodland Wildlife	Potential as Wildlife Habitat[8] for Wetland Wildlife
57D2	Wharton silt loam (15-25% slopes)	16.8	3,635	Very good	IVe	Fair	Fair	Fair	Good	Very poor
60D2	Hazelton channery loam (15-25% slopes)	3.0	2,580	Good	IVe	Good	Good	Fair	Good	Very poor
73B2	Conotton gravelly loam (2-8% slopes)	9.7	967	Good	IIs	Good	Good	Good	Good	Very poor
73D2	Conotton gravelly loam (15-25% slopes)	5.5	409	Good	IVe	Fair	Good	Fair	Good	Very poor
99	Weikert-Rock outcrop complex (25-150% slopes)	1.5	4,900	Unsuitable	VIIs	Unsuitable	Unsuitable	Very poor	Poor	Very poor
340B2	Allegheny silt loam (2-8% slopes)	0.2	1,010	Very good	IIe	Excellent	Good	Good	Good	Very poor
342B2	Monongahela silt loam (3-8% slopes)	4.9	2,549	Good	IIe	Fair	Fair	Good	Good	Very poor
342C2	Monongahela silt loam (8-15% slopes)	2.2	969	Good	IIIe	Fair	Fair	Good	Good	Very poor
MAAB	Urban and fill land (0-3% slopes)	56.9	1,645	Variable	Variable	Variable	Variable	Variable	Variable	Variable
MSCD	Urban land - Gilpin complex (8-25% slopes)	1.0	1,354	Variable	Variable	Variable	Variable	Variable	Variable	Variable

TABLE 2.2-3 (Cont)

Map Symbol [2]	Soil Name	Site Acreage [3]	Beaver County Acreage [4]	Woodland Site Quality [5]	Cropland Capability Class and Subclass [6]	Productivity Potential Rating [7]		Potential as Wildlife Habitat [8] for		
						Corn	Alfalfa	Openland Wildlife	Woodland Wildlife	Wetland Wildlife
MTAB	Urban land - Conotton complex (0-8% slopes) [9]	64.7	6,575	Variable	Variable	Variable	Variable	Variable	Variable	Variable
MTCD	Urban land - Conotton complex (8-25% slopes) [9]	13.8	947	Variable	Variable	Variable	Variable	Variable	Variable	Variable

NOTES:

1. U.S. Department of Agriculture 1969-1972.

2. Distribution of soils is indicated on Figure 2.2-2.

3. Total BVPS site acreage is 501.

4. Total Beaver County acreage (all soil types) is 282,240 (U.S. Department of Agriculture, Soil Conservation Service 1974).

5. Indicates general ability of soils to produce timber; ratings are based on the average height of dominant and co-dominant trees (yellow poplar and upland oaks) at the age of 50 years.

6. Soil Conservation Service capability classes indicate general suitability of soils for most kinds of field crops. The classes range from I to VIII, with I being the best for crop production and VIII considered not usable for general agricultural purposes. Lower case letters are capability subclasses based on limitations to crop production: e designates risk of erosion unless close-growing plant cover is maintained; s designates shallow, droughty, or stony soils.

7. Potential productivity rating expressed as excellent, good, fair, or unsuitable based on the relative productivity of the soil for the specified crop in relation to a standard index of 100. The standard index represents the average acre yield of the crop obtained on the most productive soils in the county under normal management.

8. Suitability of soils for wildlife habitat is expressed as good, fair, poor, or very poor based on the general amount and distribution of food, shelter, and water relative to the different soils for three classes of wildlife.

9. Also indicated in FSAR Section 2.5.

TABLE 2.2-4

SUMMARY OF THE STRUCTURE OF SITE VEGETATION BY STUDY AREAS

Vegetation Study Area*	Type	Percent Cover**					Average Height (feet) of Dominants in Tree Stratum	Average DBH (inches) of Dominants in Tree Stratum***	Average Age (years) of Dominants in Tree Stratum***
		Tree Stratum	Shrub Stratum	Ground Stratum Spring	Ground Stratum Summer	Ground Stratum Fall			
1	Silver maple - Sycamore	45	35	80	85	70	55	19 (1)	-
2	Black locust - Tree of heaven	75	50	80	80	75	52	12 (1)	-
3	Mixed mesophytic	75	15	60	70	10	58	13.9 (9)	58 (9)
4	Black locust - Black cherry	80	60	20	15	10	72	21.1 (5)	45 (1)
5	Mountain laurel - Hawthorn	50	40	30	25	25	30	9.8 (6)	-
6	Mountain laurel - Mixed oak	65	20	25	40	10	60	13.0 (2)	69 (2)
8	Beech - Maple	65	15	30	-	-	60	12.5 (2)	-
9	Mountain Laurel - Mixed oak	70	16	25	40	15	47	10.7 (8)	52 (8)
12	Mixed oak - Sugar maple	80	10	12	10	10	58	12.1 (7)	59 (6)
17	Mixed oak - Sugar maple	75	15	5	-	-	65	14.1 (2)	56 (2)
-	Transmission lines	2	15	55	65	60	-	-	-

NOTES:

*The location of these areas is indicated on Figure 2.2-1. Only representative areas were sampled.
**Percent cover (crown closure) is for all species.
***DBH = Diameter at Breast Height. Numbers in parentheses indicate number of stems sampled.

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CHAPTER 3

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TABLE 3.5-1

EXPECTED OPERATING CONDITIONS FOR THE BVPS-2 PWR WITH U-TUBE STEAM GENERATORS (VOLATILE CHEMISTRY)

Parameter	
Thermal power (MWt)	2,766
Steam flow rate (lb/hr)	1.16×10^{7}
Weight of water in reactor coolant system (lb)	4.19×10^{5}
Weight of water in all steam generators (lb)	2.98×10^{5}
Reactor coolant letdown flow (purification)(lb/hr)	2.80×10^{4}
Reactor coolant letdown flow (nominal value for boron control) (lb/hr)	5.00×10^{2}
Steam generator blowdown flow (total) (lb/hr)	2.24×10^{4}
Fraction of radioactivity in blowdown stream which is treated by condensate demineralizers	
Halogens	9.50×10^{-1}
Cs, Rb	9.99×10^{-1}
Other	9.99×10^{-1}
Flow through the purification system cation demineralizers (lb/hr)	2.80×10^{3}
Ratio of condensate demineralizer flow rate to the total steam flow rate	7.33×10^{-1}
Ratio of the total amount of noble gases routed to gaseous radwaste system from the purification system, to the total amount of noble gases routed from the primary coolant system to the purification system (not including the boron recovery system)	0.0
Holdup time (days) in the charcoal delay beds for noble gases being returned to the primary coolant system	
Kr	2.6
Xe	46
Primary to secondary leakage rate (lb/day)	1.00×10^{2}

TABLE 3.5-2

EXPECTED PRIMARY AND SECONDARY EQUILIBRIUM CONCENTRATIONS*,**

Isotope	Primary Coolant*** (μCi/g)	Secondary Liquid**** (μCi/g)	Secondary Steam***** (μCi/g)
		Noble Gases	
Kr-83m	$2.0x10^{-2}$	0.0	$7.3x10^{-9}$
Kr-85m	$8.9x10^{-2}$	0.0	$3.2x10^{-8}$
Kr-85	$1.2x10^{-1}$	0.0	$4.3x10^{-8}$
Kr-87	$6.2x10^{-2}$	0.0	$2.1x10^{-8}$
Kr-88	$1.8x10^{-1}$	0.0	$6.5x10^{-8}$
Kr-89	$5.8x10^{-3}$	0.0	$2.1x10^{-9}$
Xe-131m	$6.0x10^{-3}$	0.0	$2.3x10^{-9}$
Xe-133m	$4.2x10^{-2}$	0.0	$1.5x10^{-8}$
Xe-133	1.7	0.0	$6.2x10^{-7}$
Xe-135m	$1.5x10^{-2}$	0.0	$5.3x10^{-9}$
Xe-135	$2.1x10^{-1}$	0.0	$7.8x10^{-8}$
Xe-137	$9.9x10^{-3}$	0.0	$3.8x10^{-9}$
Xe-138	$5.0x10^{-2}$	0.0	$1.8x10^{-8}$
		Halogens	
Br-83	$5.5x10^{-3}$	$1.3x10^{-7}$	$1.3x10^{-9}$
Br-84	$3.0x10^{-3}$	$2.7x10^{-8}$	$2.7x10^{-10}$
Br-85	$3.5x10^{-4}$	$3.5x10^{-10}$	$3.5x10^{-12}$
I-130	$2.3x10^{-3}$	$8.5x10^{-8}$	$8.5x10^{-10}$
I-131	$2.9x10^{-1}$	$1.2x10^{-5}$	$1.2x10^{-7}$
I-132	$1.1x10^{-1}$	$3.5x10^{-6}$	$3.5x10^{-8}$
I-133	$4.2x10^{-1}$	$1.6x10^{-5}$	$1.6x10^{-7}$
I-134	$5.4x10^{-2}$	$6.9x10^{-7}$	$6.9x10^{-9}$
I-135	$2.1x10^{-1}$	$7.1x10^{-6}$	$7.1x10^{-8}$
		Cs, Rb	
Rb-86	$9.1x10^{-5}$	$2.4x10^{-8}$	$2.4x10^{-11}$
Rb-88	$2.3x10^{-1}$	$1.4x10^{-6}$	$1.4x10^{-9}$
Cs-134	$2.7x10^{-2}$	$7.3x10^{-6}$	$7.3x10^{-9}$
Cs-136	$1.4x10^{-2}$	$3.7x10^{-6}$	$3.7x10^{-9}$
Cs-137	$1.9x10^{-2}$	$5.3x10^{-6}$	$5.3x10^{-9}$
		Water Activated Products	
N-16	$4.0x10^{-1}$	$1.5x10^{-6}$	$1.5x10^{-6}$

TABLE 3.5-10

BVPS-1 AND BVPS-2 GASEOUS WASTE MANAGEMENT SYSTEMS, EFFLUENT RELEASE ANALYSIS ARAMETERS

Characteristic	Parameter BVPS-1	Parameter BVPS-2
Plant capacity factor	0.8	0.8
Containment building		
Noble gas release to containment building (fraction/day of primary coolant activity)	0.01	0.01
Iodine release to containment building (fraction/day of primary coolant activity)	10^{-5}	10^{-5}
Purge exhaust ventilation rate (cfm)	$3x10^{4}$	$3x10^{4}$
Purge exhaust ventilation time (hours)	8	8
Recirculation rate during purge (cfm)	0	0
Iodine exhaust filter efficiency (percent)	90	0*
Particulate exhaust filter efficiency (percent)	99	0*
Number of hot purges/year	2	0
Number of cold purges/year	2	4
Continuous ventilation exhaust rate (cfm)	0	0
Free containment volume (ft^3)	$1.8x10^{6}$	$1.8x10^{6}$
Containment internal cleanup system		
Containment internal cleanup system operates prior to purging (hours)	16	16

TABLE 3.5-10 (Cont)

Characteristic	Parameter BVPS-1	BVPS-2
Containment internal cleanup system mixing efficiency	0.7	0.7
Recirculation rate prior to purge (cfm)	$2x10^3$	$2x10^4$
Iodine filter efficiency (percent)	90	90
Particulate filter efficiency (percent)	0*	99
Auxiliary building		
Iodine exhaust filter efficiency (percent)	0*	90
Particulate exhaust filter efficiency (percent)	0*	95
Primary coolant leakage rate into building (lb/day)	160	160
Iodine partition factor	$7.5x10^{-3}$	$7.5x10^{-3}$
Turbine building		
No special design to collect valve leakage below 8 inches		
Iodine exhaust filter efficiency (percent)	0*	0*
Particulate exhaust filter efficiency (percent)	0*	0*
Steam leakage (lb/hr)	1,700	1,700
Main condenser/air ejector (MC/AE)		
Volatile iodine/total iodine in primary system	0.05	0.05
Volatile iodine is treated as noble gas in steam generator		

TABLE 3.5-10 (Cont)

Characteristic	Parameter BVPS-1	Parameter BVPS-2
Expected primary to secondary leak rate (lb/day)	100**	100**
MC/AE volatile iodine partition factor	0.15	0.15
Volatile iodine condenser bypass fraction	0.44	0.27
Steam generator blowdown flash tank overheads are vented to the second point feedwater heaters which are vented and drained to the main condenser (included in the above listed leak rate)		
Radioactive gaseous waste system (process gas system)		
Letdown flow to degasifier (lb/hr)	$3x10^4$	$3x10^4$
Process gas released to the atmosphere (lb/hr)	500	500
Holdup time prior to charcoal beds (minutes)	0	0
Krypton dynamic adsorption coefficient (cm^3/g)	16.0	18.5
Xenon dynamic adsorption coefficient (cm^3/g)	292	330
System flow rate (cfm)	0.33	0.31
Total mass of charcoal in beds ($x10^3$ lb) 4	4	4
Charcoal depth (feet, each of 2 in series)	14	14
No iodine or particulates are released from system		
Krypton holdup time in delay bed (days)	2	2.6

TABLE 3.5-10 (Cont)

Characteristic	Parameter BVPS-1	BVPS-2
Xenon holdup time in delay bed (days)	39	46
Number of complete primary system degasifications/year (complete degasification is handled by the same equipment as normal operation)	2	2

NOTES:

*Efficiency of zero indic' es the actual parameter is not used in release analysis.

**Design leak rate is 1,188 lb/day.

TABLE 3.5-11

BVPS-1 GASEOUS WASTE MANAGEMENT SYSTEM AND VENTILATION SYSTEM, GASEOUS RELEASES (Ci/yr)

Nuclide	Containment Building	Auxiliary Building	Turbine Building	Main Condenser/ Air Ejector	Blowdown Flash Tank	Radioactive Gaseous Waste System	Total
Kr-83m	2.2×10^{-2}	4.2×10^{-1}	3.9×10^{-5}	2.7×10^{-1}	0.0	0.0	7.1×10^{-1}
Kr-85m	1.5×10^{-1}	1.9	1.7×10^{-4}	1.2	0.0	7.3×10^{-2}	3.3
Kr-85	6.1×10^{1}	2.5	2.3×10^{-4}	1.6	0.0	2.3×10^{2}	3.0×10^{2}
Kr-87	5.4×10^{-2}	1.3	1.1×10^{-4}	8.2×10^{-1}	0.0	0.0	2.2
Kr-88	2.4×10^{-1}	3.8	3.5×10^{-4}	2.4	0.0	0.0	6.4
Kr-89	4.7×10^{-4}	1.2×10^{-1}	1.1×10^{-5}	7.7×10^{-2}	0.0	0.0	2.0×10^{-1}
Xe-131m	7.4×10^{-1}	1.3×10^{-1}	1.2×10^{-5}	8.0×10^{-2}	0.0	1.3	2.2
Xe-133m	8.9×10^{-1}	8.9×10^{-1}	8.1×10^{-5}	5.6×10^{-1}	0.0	0.0	2.3
Xe-133	8.9×10^{1}	3.6×10^{1}	3.4×10^{-3}	2.3×10^{1}	0.0	2.3×10^{1}	1.7×10^{2}
Xe-135m	4.5×10^{-3}	3.2×10^{-1}	2.9×10^{-5}	2.0×10^{-1}	0.0	0.0	5.2×10^{-1}
Xe-135	7.0×10^{-1}	4.5	4.2×10^{-4}	2.8	0.0	0.0	7.9
Xe-137	1.0×10^{-3}	2.1×10^{-1}	2.1×10^{-5}	1.3×10^{-1}	0.0	0.0	3.5×10^{-1}
Xe-138	1.5×10^{-2}	1.1	9.7×10^{-5}	6.6×10^{-1}	0.0	0.0	1.7
I-131	1.2×10^{-3}	4.6×10^{-2}	1.4×10^{-3}	1.6×10^{-2}	0.0	0.0	6.5×10^{-2}
I-133	2.0×10^{-4}	6.7×10^{-2}	1.7×10^{-3}	2.3×10^{-2}	0.0	0.0	9.2×10^{-2}
Co-58	7.5×10^{-4}	6.0×10^{-2}	0.0	0.0	0.0	0.0	6.1×10^{-2}
Co-60	3.4×10^{-4}	2.7×10^{-2}	0.0	0.0	0.0	0.0	2.7×10^{-2}
Mn-54	2.2×10^{-4}	1.8×10^{-2}	0.0	0.0	0.0	0.0	1.8×10^{-2}
Fe-59	7.5×10^{-5}	6.0×10^{-3}	0.0	0.0	0.0	0.0	6.1×10^{-3}
Sr-89	1.7×10^{-5}	1.3×10^{-3}	0.0	0.0	0.0	0.0	1.3×10^{-3}
Sr-90	3.0×10^{-6}	2.0×10^{-4}	0.0	0.0	0.0	0.0	2.0×10^{-4}
Cs-134	2.2×10^{-4}	1.8×10^{-2}	0.0	0.0	0.0	0.0	1.8×10^{-2}
Cs-137	3.8×10^{-4}	3.0×10^{-2}	0.0	0.0	0.0	0.0	3.0×10^{-2}
C-14	1.0	0.0	0.0	0.0	0.0	7.0	8.0
Ar-41	2.5×10^{1}	0.0	0.0	0.0	0.0	0.0	2.5×10^{1}
H-3	1.0×10^{2}	4.5×10^{2}	0.0	0.0	0.0	0.0	5.5×10^{2}

TABLE 3.5-12

BVPS-2 GASEOUS WASTE MANAGEMENT SYSTEM AND VENTILATION SYSTEM, GASEOUS RELEASES (Ci/yr)

Nuclide	Containment Building	Auxiliary Building	Turbine Building	Main Condenser/ Air Ejector	Blowdown Flash Tank	Radioactive Gaseous Waste System	Total
Kr-83m	4.0×10^{-5}	4.2×10^{-1}	3.9×10^{-5}	2.7×10^{-1}	0.0	0.0	6.9×10^{-1}
Kr-85m	1.4×10^{-2}	1.9	1.7×10^{-4}	1.2	0.0	1.0×10^{-2}	3.1
Kr-85	6.1×10^{1}	2.5	2.3×10^{-4}	1.6	0.0	2.3×10^{2}	3.0×10^{2}
Kr-87	5.3×10^{-6}	1.3	1.1×10^{-4}	8.2×10^{-1}	0.0	0.0	2.1
Kr-88	4.1×10^{-3}	3.8	3.5×10^{-4}	2.4	0.0	0.0	6.2
Kr-89	0.0	1.2×10^{-1}	1.1×10^{-5}	7.7×10^{-2}	0.0	0.0	2.0×10^{-1}
Xe-131m	7.2×10^{-1}	1.3×10^{-1}	1.2×10^{-5}	8.0×10^{-2}	0.0	8.3×10^{-1}	1.8
Xe-133m	7.6×10^{-1}	8.9×10^{-1}	8.1×10^{-5}	5.6×10^{-1}	0.0	0.0	2.2
Xe-133	8.4×10^{1}	3.6×10^{1}	3.4×10^{-3}	2.3×10^{1}	0.0	8.2	1.5×10^{2}
Xe-135m	0.0	3.2×10^{-1}	2.9×10^{-5}	2.0×10^{-1}	0.0	0.0	5.2×10^{-1}
Xe-135	2.4×10^{-1}	4.5	4.2×10^{-4}	2.8	0.0	0.0	7.5
Xe-137	0.0	2.1×10^{-1}	2.1×10^{-5}	1.3×10^{-1}	0.0	0.0	3.4×10^{-1}
Xe-138	0.0	1.1	9.7×10^{-5}	6.6×10^{-1}	0.0	0.0	1.7
I-131	2.7×10^{-5}	4.6×10^{-3}	6.5×10^{-4}	2.1×10^{-2}	0.0	0.0	2.6×10^{-2}
I-133	2.6×10^{-6}	6.7×10^{-3}	8.7×10^{-4}	3.0×10^{-2}	0.0	0.0	3.8×10^{-2}
Co-58	4.2×10^{-5}	6.0×10^{-4}	0.0	0.0	0.0	0.0	6.4×10^{-4}
Co-60	1.9×10^{-5}	2.7×10^{-4}	0.0	0.0	0.0	0.0	2.9×10^{-4}
Mn-54	1.2×10^{-5}	1.8×10^{-4}	0.0	0.0	0.0	0.0	1.9×10^{-4}
Fe-59	4.2×10^{-6}	6.0×10^{-5}	0.0	0.0	0.0	0.0	6.4×10^{-5}
Sr-89	9.5×10^{-7}	1.3×10^{-5}	0.0	0.0	0.0	0.0	1.4×10^{-5}
Sr-90	1.7×10^{-7}	2.0×10^{-6}	0.0	0.0	0.0	0.0	2.2×10^{-6}
Cs-134	1.2×10^{-5}	1.8×10^{-4}	0.0	0.0	0.0	0.0	1.9×10^{-4}
Cs-137	2.1×10^{-5}	3.0×10^{-4}	0.0	0.0	0.0	0.0	3.2×10^{-4}
C-14	1.0	0.0	0.0	0.0	0.0	7.0	8.0
Ar-41	2.5×10^{1}	0.0	0.0	0.0	0.0	0.0	2.5×10^{1}
H-3	1.0×10^{2}	4.5×10^{2}	0.0	0.0	0.0	0.0	5.5×10^{2}

3.7 SANITARY AND OTHER WASTE SYSTEMS

3.7.1 Sanitary Waste Treatment

The original sewage treatment system at BVPS-1, as described in Section 3.8 of the BVPS-2 Environmental Report - Construction Permit Stage (ER-CPS), utilized the extended aeration modification of the activated sludge process and was designed for a peak flow of 10,000 gpd from both BVPS-1 and BVPS-2. Given the actual operating data and an increase in the number of personnel onsite, the original treatment system capacity was insufficient to treat the combined volume of sewage from BVPS-1 and BVPS-2. Therefore, the BVPS-1 sanitary waste treatment system was modified to treat only sanitary wastes from the BVPS-1 and BVPS-2 permanent plant buildings. A rotating biological contactor (RBC) unit, clarifier, and sludge holding tank were added, and piping was modified to use the prior aeration tank as a flow equalization basin. These modifications increased the BVPS-1 system design capacity to 23,000 gpd.

The BVPS-2 sewage treatment system is designed to provide secondary treatment of the sanitary wastes from the BVPS-1 and BVPS-2 support buildings prior to discharge. The BVPS-2 sewage treatment system is an RBC unit designed to treat a maximum of 42,400 gallons of sewage per day. The estimated sewage flow during normal operation is 22,325 gpd. In addition to the RBC, the treatment system includes screening, pre-aeration (equalization), primary settling, and chlorination.

The RBC is a secondary biological process consisting of closely-spaced discs on a horizontal shaft. The discs are partially submerged in wastewater and are slowly rotated such that any point on the discs is alternately submerged and aerated. Biological growths develop on the discs which oxidixe organic material in the wastewater. Excess biomass is sloughed into the wastewater by the shearing force resulting from disc rotation. Effluent from the biological contactor is treated in a clarifier to remove sloughed biomass and other suspended matter. Sloughed biomass is pumped from the clarifier to an aerated sludge holding tank and removed periodically for disposal by a licensed contractor.

Following disinfection by chlorination, the treated effluent from the BVPS-1 treatment system is discharged to the Ohio River. Average daily concentrations of biochemical oxygen demand and suspended solids in the treated effluent will each be less than 30 mg/l, maximum daily concentrations will be less than 45 mg/l, and instantaneous maximum concentrations will be less than 60 mg/l, in accordance with the U.S. Environmental Protection Agency (USEPA) secondary treatment standards (40 CFR 133). The sewage treatment system is in compliance with the guidelines of Pennsylvania Department of Environmental Resources (DER) Permit No. 0479403 and

the effluent specifications of Pennsylvania DER Permit No. PA 0025615.

Following disinfection by chlorination, the treated effluent from the BVPS-2 treatment system is discharged to the Peggs Run Creek. Average monthly concentrations of biochemical oxygen demand and suspended solids in the treated effluent will each be less than 30 mg/1, and instantaneous maximum concentrations will be less than 60 mg/1, in accordance with the USEPA secondary treatment standards (40 CFR 133). The sewage treatment system is in compliance with the guidelines of Pennsylvania DER Permit No. 0482404 and the effluent specifications of Pennsylvania DER Permit No. PA 0025615.

3.7.2 Fossil-Fueled Equipment

The information presented in this section supplements that provided in ER-CPS Section 3.10 and in the response to U.S. Atomic Energy Commission Question D.21, Amendment 1 of the ER-CPS. New equipment has been added, and operational frequencies and the sulfur-in-fuel content have changed since the publication of the ER-CPS. Additionally, gaseous and particulate emissions are provided and federal, state, and local standards are discussed herein.

Two 150,000-lb/hr oil-fired auxiliary boilers are installed in BVPS-2. These supplement the two 43,000-lb/hr oil-fired auxiliary boilers installed in BVPS-1. Additional fossil-fueled auxiliary equipment includes a diesel driven fire pump and a standby diesel generator, both common to BVPS-1 and BVPS-2, and four emergency diesel generators. Two 2,600-kW emergency diesel generators are provided for BVPS-1, and two 4,238-kW emergency diesel generators are provided for BVPS-2.

The auxiliary boilers are used to supply auxiliary steam when the main steam system is secure. The standby diesel, or "black diesel," is used strictly on an emergency basis to supply electric power to the emergency response facility and the electric motor-driven fire pump, which is also common to both units. The standby diesel generator has only recently been added to the onsite auxiliary equipment and is not described in the ER-CPS. The emergency diesel generators are used to supply emergency onsite ac electrical power. The fossil-fueled auxiliary equipment burns No. 2 fuel oil containing 0.3 to 0.7 percent sulfur and negligible amounts (0.08 percent) of ash.

Each emergency diesel generator is tested at least once per month for approximately 1 hour. However, to ensure conservatism and flexibility of the testing mode, emission estimates in Table 3.7-1 were based on twice-per-month testing. The diesel-driven fire pump is tested once per week for approximately 1/2 hour, and the standby diesel once per month for approximately 1 hour. Cooling water for

the BVPS-2 diesel generators is discharged to the Ohio River at approximately 1,100 gallons per minute (gpm) during these test periods.

It is expected that the auxiliary boilers will be operated several days a year for testing purposes and approximately 6 to 8 weeks per year during shutdown and refueling of BVPS-2. This modification to the ER-CPS operational estimate of 48 hours per year for each boiler (ER-CPS, Amendment 1, Table D.21-1) more realistically approximates the expected annual usage of these units. Table 3.7-1 summarizes the total annual fuel consumption of auxiliary equipment due to BVPS-1 and BVPS-2 operation. It also shows that the estimated annual air pollutant emissions from these sources are minimal.

Auxiliary boiler blowdown, discharged at a maximum rate of 15 gpm (one boiler in operation), flows to the BVPS-2 service water system where it is either re-used as makeup to the cooling tower or discharged through the emergency outfall structure. Discharge concentrations of total suspended solids and oil and grease in the auxiliary boiler blowdown are within the limits contained in 40 CFR 423.

Due to the relatively small amounts of increased emissions from BVPS-2 and the infrequency of equipment operation, no federal, state, or local emission standards are applicable. However, for purposes of tracking minor source emissions, the state of Pennsylvania requires an operational permit for the auxiliary equipment.

3.7.3 References for Section 3.7

U.S. Environmental Protection Agency (USEPA) 1977. Compilation of Air Pollutant Emission Factors. Second Edition, AP-42, Supplement No. 7, April 1977.

TABLE 3.7-1

FUEL CONSUMPTION AND EMISSIONS OF FOSSIL-FUELED EQUIPMENT*

	Auxiliary Boilers		Emergency Diesel Generators		Diesel Fire Pump	Standby Diesel Generator
	BVPS-1	BVPS-2	BVPS-1	BVPS-2		
Quantity, each	2	2	2	2	1	1
Rating, each	43,000 lb/hr	150,000 lb/hr	2,600 kW	4,238 kW	-	-
Total maximum hours of operation per year	1,440	1,440	24 (each)	24 (each)	26	12
Total annual equipment fuel consumption (gal)	1.18×10^{6}	1.86×10^{6}	-------6,480 Total--------		975	810
Emissions (lb/year)						
Particulates	3,400**	3,800**	22***	36***	18	6
Sulfur dioxide	85,200	134,200****	344	600	30	86
Carbon monoxide	5,800	9,200	1,116	1,200	99	279
Hydrocarbons	1,200	1,800	412	720	37	103
Nitrogen oxides	26,600	41,600	5,168	5,414	457	1,292

NOTES:

*The two BVPS-1 auxiliary boilers may operate simultaneously; the two BVPS-2 auxiliary boilers operate alternately. The diesel fire pump is installed in BVPS-1 and shared by BVPS-2. The standby or "black" diesel generator, utilized for the emergency response facility, is common to both units.

**Based on a USEPA AP-42 emission factor of 2 pounds of particulates per 1,000 gallons of oil (USEPA 1977).

***Based on smoke emissions provided by vendor.

****Based on an assumed sulfur-in-fuel content of 0.5 percent.

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APPENDIX 3A

DATA NEEDED FOR RADIOACTIVE SOURCE-TERM CALCULATIONS FOR PRESSURIZED WATER REACTORS

The following data are presented in response to the information requested in Regulatory Guide 1.112 and satisfies the intent of Appendix E to Regulatory Guide 4.2, Revision 2.

I. General (Reference FSAR Chapter 1 and Section 11.1)

1. The maximum core thermal power (MWt) evaluated for safety considerations in the SAR. (Note: All of the following responses are adjusted to this power level.)

   The maximum core thermal power is 2,766 MWt. This is the licensed power level plus 4 percent, based on a 5-percent increase in steam flow (Section 3.2).

2. Core properties:

   a. The total mass (lb) of uranium and plutonium in an equilibrium core (metal weight).

      The total mass of uranium and plutonium in an equilibrium core is not readily available. The fuel weight in the 157 fuel assemblies is 181,190 lb of UO_2 and the fuel burnout at 3.2 weight percent uranium-235 is 29,000 MWD/MTU (Section 3.2).

   b. The percent enrichment of uranium in reload fuel.

      Typical enrichments vary from 1.6 percent to 3.3 percent (Section 3.2).

   c. The percent of fissile plutonium in reload fuel.

      There is no plutonium in reload fuel.

3. If methods and parameters used in estimating the source terms in the primary coolant are different from those given in Regulatory Guide 1.112, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors, describe in detail the methods and parameters used.

   The methods and parameters used are consistent with those presented in Regulatory Guide 1.112.

4. The quantity of tritium released in liquid and gaseous effluents (Ci/yr per reactor).

The total tritium released is 1,106 Ci/year/reactor according to the U.S. Nuclear Regulatory Commission's (USNRC) NUREG-0017 (1976). It is assumed that half of this quantity is released via the ventilation and gaseous waste systems, and the remainder is discharged from the liquid waste system.

II. Primary System (Reference FSAR Section 11.1 and Table 11.1-3)

1. The total mass (lb) of coolant in the primary system, excluding the pressurizer and primary coolant purification system at full power.

   420,000 pounds.

2. The average primary system letdown rate (gpm) to the primary coolant purification system.

   60 gallons per minute.

3. The average flow rate (gpm) through the primary coolant purification system cation demineralizers. (Note: The letdown rate includes the fraction of time the cation demineralizers are in service.)

   6.0 gallons per minute. The letdown is assumed to flow through the cation demineralizers at 60 gpm, one-tenth of the time.

4. The average shim bleed flow (gpm).

   1 gallon per minute (approximately 500 pounds per hour).

III. Secondary System (Reference FSAR Chapter 10, Section 11.1, and Table 11.1-3)

1. The number and type of steam generators and the carryover factor used in the applicant's evaluation for iodine and nonvolatiles.

   There are three vertical U-tube recirculating-type steam generators. All volatile treatment chemistry is used. The carryover fractions used are 0.01 for iodine and 0.001 for nonvolatiles.

2. The total steam flow (lb/hr) in the secondary system.

   11,600,000 pounds per hour.

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CHAPTER 5

ENVIRONMENTAL EFFECTS OF STATION OPERATION

5.1 EFFECTS OF OPERATION OF HEAT DISSIPATION SYSTEM

5.1.1 Effluent Limitations and Water Quality Standards

Liquid waste discharges during operation of Beaver Valley Power Station - Unit 2 (BVPS-2) will be in compliance with the following:

1. U.S. Environmental Protection Agency (USEPA) effluent guidelines and standards for steam electric power plants (40 CFR 423), and

2. Discharge limitations established and/or certified by the Pennsylvania Department of Environmental Resources (DER) in the National Pollution Discharge Elimination System (NPDES) discharge permit and in the DER industrial waste permits.

The USEPA granted an NPDES permit for Beaver Valley Power Station - Unit 1 (BVPS-1) in 1975 with amendments through 1977. The Pennsylvania DER then gained NPDES permitting authority and amended the BVPS-1 permit in 1979. The DER has indicated that it will further amend the existing BVPS-1 NPDES permit to include discharges from BVPS-2. Discharges from BVPS-2 which will be included in the amended permit are:

1. Cooling tower blowdown from BVPS-2,

2. Service water (Section 3.6.3),

3. Floor and equipment drainage, and

4. Low level radwaste from BVPS-2.

The application for an amended NPDES permit was submitted to the DER on March 15, 1983. When issued, the amended permit will be contained in Appendix 5A. Table 5.1-1 lists the existing discharge permits for BVPS-1 as well as the anticipated permits for BVPS-2.

Effluent limitations contained in the existing BVPS-1 NPDES permit are presented in Table 5.1-2. Table 5.1-3 presents effluent limitations anticipated for BVPS-2 discharges, based upon the existing BVPS-1 limitations, and current USEPA effluent guidelines for the steam electric industry (40 CFR 423). However, the USEPA is in the process of revising its 40 CFR 423 effluent guidelines for the steam electric industry. Among the revisions being contemplated are more stringent limitations for chlorine, a prohibition on the

discharge of cooling tower maintenance chemicals, and limitations on toxic and hazardous substances not presently regulated. It is anticipated that the new BVPS-1 and BVPS-2 NPDES permit will be issued by the Pennsylvania DER after revision by the USEPA of 40 CFR 423, and the permit will include the revised USEPA effluent limitations.

In addition, the Pennsylvania DER has been charged with establishing water quality standards for the Ohio River and to ensure these standards are maintained. Therefore, it is possible that the Pennsylvania DER could impose discharge limitations more stringent than those outlined in the USEPA's effluent guidelines for the steam electric industry (40 CFR 423). The effects of wastewater discharges to the Ohio River are discussed in Section 5.3.

Water quality standards applicable to the Ohio River are presented in Table 5.1-4. These standards are taken from the Pennsylvania Code, Title 25, Part I, Chapter 93, Water Quality Standards. General water quality criteria for all waters of the state, as outlined in Section 93.6 of this Code, are as follows:

a) Water shall not contain substances attributable to point or nonpoint source waste discharges in concentration or amounts sufficient to be inimical or harmful to the water uses to be protected or to human, animal, plant, or aquatic life, and

b) Specific substances to be controlled shall include, but shall not be limited to, floating debris, oil, grease, scum and other floating materials, toxic substances, pesticides, chlorinated hydrocarbons, carcinogenic, mutagenic and teratogenic materials, and substances which produce color, tastes, odors or settle to form sludge deposits.

In addition to these water quality standards, the Ohio River Valley Water Sanitation Commission (ORSANCO) has developed a set of recommended water quality criteria for the Ohio River in order to provide uniform standards for the main stem of the river. These standards were adopted by ORSANCO on September 9, 1976 and amended on May 12, 1977 and September 8, 1977. The Commonwealth of Pennsylvania is a signatory member of ORSANCO. Table 5.1-5 presents a summary of the ORSANCO water quality criteria for selected parameters.

As discussed in Section 5.3, waste discharges from BVPS-1 and BVPS-2 will have no detectable impact on the Ohio River water quality in the vicinity of the site and, therefore, will have no impact on the river water quality in the state of Ohio. The state boundary is approximately 5 miles downstream of the BVPS site.

5.2.4.4.2 Contiguous U.S. Population Doses

In addition to the 50-mile radius population doses, population doses associated with the export of food crops produced within the 50-mile region and the atmospheric and hydrospheric transport of the more mobile effluent species, such as noble gases, tritium, and carbon-14, were calculated. Table 5.2-29 presents the calculated annual total body and thyroid doses to the contiguous U.S. population.

5.2.5 Summary of Annual Radiation Doses

The calculated annual radiation doses to the maximum individual from liquid and gaseous pathways are presented in Tables 5.2-7 through 5.2-10 and Tables 5.2-12 through 5.2-27. These tables and Table 5.2-11 show that the calculated doses are below the RM-50-2 design objectives (USNRC 1975). For the site, the maximum calculated organ dose to a child's thyroid is 7.7 mRem per year.

For the liquid releases, it was assumed that the maximum individual obtains drinking water from the downstream public water supply having the highest potential concentration of station effluents. It was assumed the maximum individual would consume fish caught at the edge of the initial mixing zone. This location was also used in calculating the dose to maximum individuals from swimming. Boating was assumed to occur in the outfall area. Shoreline recreation was analyzed at the shore of Little Beaver Creek.

The calculated organ dose to a maximum individual from liquid pathways is 4.7 mRem per year to a teen's liver. This dose is primarily the result of fish consumption. It was assumed that a teen consumes 16 kg of fish per year which were caught at the edge of the initial mixing zone.

For gaseous releases, a separate analysis was performed for each location of the maximum residence, milk cow, milk goat, and beef animal. Each of these locations was analyzed for submersion, inhalation, ground deposition, ingestion of vegetation, as well as consumption of deer, rabbit, grouse, and pheasant. In addition to these pathways, the cow location was also analyzed for consumption of cow milk, the goat location was analyzed for consumption of goat milk, and the beef animal location was analyzed for consumption of beef meat.

The calculated dose to the maximum individual from gaseous pathways is 7.7 mRem per year to a child's thyroid. The calculated dose represents the dose to a child who lives at the maximum residence location 1,432 meters (0.89 mile) northwest of the site. A majority of this dose is due to the consumption of 494 kg per year of stored vegetables and from inhalation of radionuclides.

The calculated annual doses to the population residing within a 50-mile radius of the site are presented in Table 5.2-28. For the

liquid effluents, the calculated whole body and thyroid doses are 3.3×10^{-1} and 1.5 manRem per year, respectively. The calculated doses from gaseous pathways are 1.5×10^{1} manRem per year whole body and 2.2×10^{1} manRem per year thyroid. These doses were calculated for a projected population in the year 2010 of 3,949,000 people within 50 miles of the site. The milk, meat, and vegetation 50-mile radius crop yield, as well as the 50-mile radius sport fish harvest, are presented in Appendix 5C.

The calculated doses to the contiguous U.S. population are presented in Table 5.2-29. The total annual doses are calculated to be 5.5×10^{1} manRem to the whole body and 6.4×10^{1} manRem to the thyroid.

5.2.6 References for Section 5.2

Environmental Analysts, Inc. 1975. Standard Methodology for Calculating Radiation Dose to Lower Form of Biota. Prepared for the Atomic Industrial Forum and the National Environmental Studies Project AIF/NESP-006, United States Nuclear Regulatory Commission.

U.S. Nuclear Regulatory Commission (USNRC) 1975. 10 CFR Part 50, Appendix I, Annex, concluding statement of position of the Regulatory Staff, (Docket-RM-50-2), Guides on Design Objective for Light-Water-Cooled Nuclear Power Reactors.

U.S. Nuclear Regulatory Commission 1976. Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors (PWR-GALE code). NUREG-0017, Washington, D.C.

TABLE 5.2-28

CALCULATED ANNUAL DOSES FOR
POPULATION WITHIN A 50-MILE RADIUS OF BVPS*

Effluents	Population (manRem) Whole Body	Thyroid
Liquid		
Ingestion of potable water	3.2×10^{-1}	1.5
Ingestion of fish	9.0×10^{-4}	2.7×10^{-4}
Shoreline recreation	5.7×10^{-3}	5.7×10^{-3}
Swimming	1.2×10^{-5}	1.2×10^{-5}
Boating	7.2×10^{-6}	7.2×10^{-6}
Total	3.3×10^{-1}	1.5
Gaseous		
Submersion	9.4×10^{-1}	9.4×10^{-1}
Inhalation	6.1	1.1×10^{1}
Standing on contaminated ground	2.9	2.9
Ingestion of fruits, grains, and vegetation	8.2×10^{-1}	1.3
Ingestion of cow milk	2.8	5.1
Ingestion of meat	1.3	1.4
Total	1.5×10^{1}	2.2×10^{1}

NOTE:

*Based upon a projected 50-mile population of 3,949,000 for the year 2010.

TABLE 5.2-29

CALCULATED DOSE COMMITMENT
TO THE CONTIGUOUS U.S. POPULATION

	Annual Dose per Site	
	Total Body (manRem)	Thyroid (manRem)
Liquid effluents	3.4×10^{-1}	1.5
Noble gas effluents	9.9×10^{-1}	1.7
Radioiodines and particulates*	5.4×10^{1}	6.1×10^{1}
Total	5.5×10^{1}	6.4×10^{1}

NOTE:

*Carbon-14 and tritium have been added to this category.

Cooling tower blowdown concentration is calculated assuming that the maximum river concentration and the maximum cooling tower concentration of 2.4 occur simultaneously. It should be noted that this is a conservative assumption, since it requires that the BVPS-1 cooling system is not operating (maximum concentration factor with both units operating is 2.0).

Distance downstream at which water quality standard is met is the calculated distance along the stream axis from the point of discharge to the point at which sufficient mixing has taken place to dilute the discharge concentration to the applicable water quality standard. The Ohio River average low monthly flow of 10,900 cfs and the 7-day, once-in-10-years low flow of 5,200 cfs were used for this analysis.

Area within which water quality standard is exceeded is the area enclosed by the concentration isopleth within which the applicable water quality standard is exceeded. Table 5.3-4 lists 30 constituents for which Pennsylvania has established water quality standards. The BVPS discharge concentration will never exceed the applicable standard for 12 of these constituents. Fifteen constituents occasionally exceed the water quality standard in the ambient river, for reasons unrelated to BVPS. For 11 of the constituents, the maximum ambient Ohio River concentration equals or exceeds the water quality criteria. The remaining four constituents occasionally reach a high enough concentration in the ambient river that, should the BVPS-2 cooling tower be operating at its maximum concentration factor, the cooling tower discharge concentration will exceed the applicable water quality standard. These constituents are diluted in the receiving water as the discharge mixes with the ambient river water. The distance downstream from the point of discharge at which the river is restored to compliance with the applicable water quality standard is a function of the ambient water concentration, the discharge concentration, and the water quality standard. Table 5.3-4a shows that for all four constituents, the required degree of mixing takes place within 790 feet of the point of discharge for the average low monthly flow and 2,150 feet for the 7-day, once-in-10-years low flow. No ambient data are available for three of the 30 constituents listed in Table 5.3-4.

The 15 constituents in the cooling tower blowdown which occasionally exceed Pennsylvania water quality standards as a result of ambient river conditions or concentration in the cooling tower are aluminum, bacteria, copper, total iron, lead, manganese, phenolics, maximum total dissolved solids, zinc, ammonia, cadmium, total cyanide, nitrite nitrogen, selenium, and mercury. The following is a discussion of the impacts of each of these constituents; the impact of biocide additions to the cooling system is also discussed.

The stream standard for aluminum is 0.013 mg/l. This is 0.1 times the 96-hour LC_{50} for a sensitive indigenous species. Driscoll (et al 1980), in the only applicable aluminum toxicity study, has determined the 96-hour LC_{50} of white sucker fry to be 0.13 mg/l. This was the

concentration of total aluminum in waters with a pH of 5.0. The toxicity of aluminum is pH dependent (Driscoll et al 1980), and a pH of 5.0 is not typical of Ohio River waters. Only data using total aluminum concentrations and waters of different pH are available for evaluating the LC_{50} (and thus the water quality standard). The actual stream standard for aluminum would probably be greater than the 0.013 mg/l listed in Table 5.3-4 if more complete data on aluminum toxicity were available. Total elimination of aluminum in the blowdown waters would not prevent Ohio River water from exceeding water quality standards since the plant provides no net addition of aluminum.

The Pennsylvania water quality standard for fecal coliform bacteria of 2,000/100 ml for October through April has historically been exceeded in ambient Ohio River waters. Fecal coliform bacteria standards are related to the presence of human pathogens, and are not set as a safe level for populations of aquatic organisms. The U.S. Environmental Protection Agency (USEPA) water quality criteria (USEPA 1976) do not include criteria or rationale for limiting coliform bacteria for maintenance of aquatic organisms. Because of biological activity in the cooling system, the concentration of bacteria in the cooling tower blowdown is not a function of the concentration factor as is the case for other constituents. Therefore, the concentration of fecal bacteria in the cooling tower blowdown cannot be calculated, but the operation of the cooling water system is not likely to result in a significant change in the coliform count of Ohio River water. Any concentration of fecal coliform bacteria that could occur in the cooling system is not likely to impact aquatic organisms in the Ohio River.

The ambient Ohio River concentration of copper exceeds the Pennsylvania stream standard. As shown in Table 5.3-3, the maximum ambient Ohio River water concentration of copper was 0.48 mg/l. This ambient concentration can result in a cooling tower blowdown concentration of 1.15 mg/l under maximum evaporation conditions. This concentration exceeds reported 96-hour TL_{50} concentrations for brown bullhead juveniles and some adult fish species found in the area of BVPS (USEPA 1976). The concentrations reported to be toxic to fish species did not consistently occur even in replicate water samples taken at the same station at the same time. Most of the ambient water samples taken were well below levels that would result in toxic concentrations of copper in the cooling tower blowdown. The higher concentrations observed for copper are probably due to contamination of samples with sediment. Therefore, it is more likely that concentrations of copper in the cooling tower blowdown will be much less than those presented in Table 5.3-3.

The stream standard for total iron is 1.5 mg/l. Average and maximum ambient concentrations (Table 5.3-3) of 1.5 and 3.8 mg/l, respectively, equal or exceed this standard. Total iron concentrations in the combined cooling tower blowdown from both units range from an average of 2.6 mg/l to a maximum of 7.6 mg/l. The

discharge concentrations of soluble iron (<0.1 mg/l) are within the allowable limit of 7.0 mg/l dissolved iron contained in the

A cooling tower chlorine blowdown study conducted on the Clinch River showed that chlorine concentrations dropped rapidly after discharge, probably due to the presence of ammonia. Ambient ammonia concentrations at BVPS are higher than those at Clinch River, and a similiar chlorine concentration drop may be expected at BVPS.

Chlorine levels similar to 0.2 mg/l have produced lethal effects during bioassay determinations (USEPA 1976) on fish species similar to those found at the BVPS site. However, most bioassay studies are conducted with continuous exposure of test organisms to constant chemical concentrations. Chlorine releases from cooling tower blowdown are not very similar to these constant concentration exposure tests. A bioassay study conducted by Brooks and Seegert (1978) using carp, bluegills, and common, spotfin, and emerald shiners using daily quadruple 40-minute exposures to monochloramine provides more realistic toxicity data for organisms subjected to the BVPS chlorine release. In addition, these species, with the exception of common shiner, are the most abundant species found at BVPS. Brooks and Seegert (1978) show that even under the worst thermal conditions, the most sensitive of the five species shows no mortality at total residual chlorine concentrations of 0.21 mg/l. Therefore, it is expected that there is little potential for impacts due to toxic effects of chlorine discharges.

5.3.3 Salt and Water Drift Impacts

Cooling tower operation results in the release of water droplets containing low concentrations of sodium, calcium, chloride, sulfate ions, and other products found in ambient river water. The emission of these droplets (salt drift) represents a potential source of impact to terrestrial systems.

From the standpoint of soil salinization (the effects of the accumulation of salts in the soil), no appreciable impact resulting from operation of the natural draft cooling towers is anticipated for vegetation either on or off the BVPS site, including sensitive agricultural crops. This assessment is based on the following:

1. The two natural draft cooling towers will deposit salts at a maximum annual average rate of 9.9 lb/acre/year. The maximum deposition rate occurs 4,750 feet east of a center point between the two towers (Section 3.6.9, Figure 3B-4).

2. The potential for accumulation of this salt in the soil is greatly reduced by the average rate of precipitation of 36.2 inches annually (FSAR Table 2.3-1).

3. Ongoing monitoring to determine the effects of operation of the BVPS-1 cooling tower on vegetation has found no evidence of stress.

The greatest concentration of cropland in the BVPS area is southwest of the site (Figure 2.1-4). Of the crops grown in the Beaver County area (Pennsylvania Crop Reporting Service 1978), the field crops, barley, wheat, oats, and corn (for grain silage), appear to be the most tolerant to soil salinization (Table 5.3-5). Of the forage crops, alfalfa is also moderately tolerant, and hay (timothy and clover mixtures) has been found to be moderately sensitive to soil salinization. Of the vegetable crops, tomatoes and spinach are moderately salt-tolerant while potatoes and sweet corn are moderately salt sensitive. Onions, carrots, and beans all have low salt tolerances.

An estimate of the potential increase in dissolved solids in the water passing through the soil in the areas of maximum deposition was made. The estimate is extremely conservative because it assumes that all of the salt from operation of the towers deposited on the soil remains in the soil and is not leached. The average increase in dissolved solids in water passing through the soil, based on the maximum annual salt drift and the average annual rainfall, would be 1.23 parts per million (ppm) annually. Therefore, given the tolerances of field and forage crops to soil salinization and the potential incremental increase in soil salinities, it is unlikely that even salt-sensitive species would be measurably affected by operation of the cooling towers.

In addition, Curtis (et al 1977) used cooling tower basin water from the Chalk Point natural draft cooling tower to treat native perennials during the summer of 1976. They concluded from their studies that, in the presence of a corresponding increase in leaf chloride levels, the minimum deposition of salt drift which would cause injury to sensitive species of native perennials (such as flowering dogwood) was about 10 kilograms per hectare per month (8.9 lb/acre/month). Since the maximum monthly salt deposition rates from both natural draft cooling towers combined is predicted to be 3.22 lb/acre/month, no adverse impact on sensitive woody species at the BVPS site is anticipated.

Relative to foliar salinization (the deposition of salts either as particulates or in solution on the foliage of plants), no appreciable impact is anticipated for vegetation onsite or for vegetation and agricultural crops growing offsite. The natural draft cooling towers will produce maximum annual average airborne salt concentrations of 0.07 microgram per cubic meter ($\mu g/m^3$), occurring 7,000 feet east of the towers, and maximum hourly airborne concentrations of 21.9 $\mu g/m^3$, occurring 3,250 feet west-southwest of the towers. Studies conducted on the effects of foliar salinization have indicated that annual average concentrations of airborne salts above 10 $\mu g/m^3$ may affect the long-term distribution of natural vegetation in eastern coastal areas (Roffman et al 1973). Similarly, short-term or maximum airborne concentrations exceeding 60 to 100 $\mu g/m^3$ for a few hours may cause acute injury to salt-sensitive deciduous vegetation during the growing season (Roffman et al 1973). The concentrations and

composition of salts produced by operation of the cooling towers are substantially below these levels; thus, no long- or short-term

5.4 EFFECTS OF SANITARY WASTE DISCHARGES

The Beaver Valley Power Station - Unit 2 (BVPS-2) sewage treatment facility differs from that described in Section 3.8 of the Environmental Report - Construction Permit Stage (ER-CPS). Section 5.3 of the ER-CPS discusses the impact of a maximum 6-gpm discharge from a treatment plant shared by Beaver Valley Power Station - Unit 1 (BVPS-1) and BVPS-2.

The BVPS-1 treatment facility was modified, and its design flow rate was increased to 23,000 gallons per day (gpd). In addition, a BVPS-2 sanitary waste treatment facility was constructed to treat wastes from the BVPS-1 and BVPS-2 support buildings. The design maximum flow of the BVPS-2 sewage treatment plant is 42,400 gpd, based on a potential flow occurrence during refueling. During normal operation, estimated sewage flow to the BVPS-2 treatment facility is 22,325 gpd.

Both sewage treatment plants are designed to remove a minimum of 90 percent of the biochemical oxygen demand and 90 percent of the suspended solids in the raw sewage. Effluent from the sewage treatment plants is regulated by discharge permits from the Pennsylvania Department of Environmental Resources (DER) NPDES Permit No. PA 0025615, and Sewerage Permit Nos. 0479403 and 0482404. The sewage treatment systems are described in Section 3.7.

With both sewage treatment plants operating at their design flows and the Ohio River at its minimum monthly mean flow, treated sewage discharge is approximately 0.0009 percent of the river flow. Due to the high degree of treatment and the relatively low flow, this discharge is not expected to have any significant impact on Ohio River water quality or aquatic life. Table 5.4-1 lists the effluent limitations contained in the Pennsylvania DER sewerage permits and in the NPDES permit.

Section 5.3 addresses the impact of plant operation on the concentration of constituents in the Ohio River for which Pennsylvania has established water quality standards. Constituents discharged with treated sanitary wastes are included with those addressed in Section 5.3.

TABLE 5.4-1

SEWAGE TREATMENT SYSTEM
EFFLUENT LIMITATIONS*

BVPS-1 Treatment System:

Parameter	Daily Average	Daily Maximum	Instantaneous Maximum
Biochemical oxygen demand	30 mg/l 2.61 kg 5.75 lb	45 mg/l 3.92 kg 8.63 lb	60 mg/l
Suspended solids	30 mg/l 2.61 kg 5.75 lb	45 mg/l 3.92 kg 8.63 lb	60 mg/l
Fecal coliform bacteria	200/100 ml	400/100 ml	1,000/100 ml
pH	Not less than 6.0 nor greater than 9.0		

BVPS-2 Treatment System:

Parameter	Average Monthly	Daily Maximum	Instantaneous Maximum
Biochemical oxygen demand	30 mg/l 4.91 kg/day 10.8 lb/day	- 9.77 kg 21.5 lb	60 mg/l
Suspended solids	30 mg/l 4.91 kg/day 10.8 lb/day	- 9.77 kg 21.5 lb	60 mg/l
Fecal coliform			
May 1 - Sept 30	200/100 ml	-	1,000/100 ml
Oct 1 - Apr 30	2,000/100 ml	-	
pH	Not less than 6.0 nor greater than 9.0		

NOTE:

*Based on limitations contained in Pennsylvania DER NPDES Permit No. PA 0025615 and in Pennsylvania DER Permit Nos. 0479403 and 0482404.

5.7 RESOURCES COMMITTED

5.7.1 Resources Utilized in the Operation of BVPS-2

This section describes the permanent commitment of resources due to the operation of Beaver Valley Power Station - Unit 2 (BVPS-2). Section 4.3 of the Environmental Report - Construction Permit Stage (ER-CPS) describes a site acreage of 449 acres, with 58 of those acres affected by construction. The site acreage has increased to 501 acres due to the need for additional land for construction and plant expansion, with construction of BVPS-2 having affected 101 acres. Chapter 10 of the ER-CPS discusses the use of uranium fuel and the irretrievable loss of land. Because BVPS-2 is not reusing fuel as stated in the ER-CPS, lifetime uranium consumption currently differs from that stated therein. Also, because the exact method of decommissioning BVPS-2 is not yet known, the ER-CPS commitment of 1 acre of lost land cannot be verified.

The BVPS-2 utilizes the uranium-235 fuel cycle wherein the fission process creates heat. The fissionable uranium isotopes within the reactor fuel are irretrievably lost as they become depleted. Based on an annual refueling rate of 52 assemblies per year and the 3.2 percent enrichment factor, a total uranium usage of 6,660 metric tons of U_3O_8 is expected over the 40-year life of the station.

The BVPS-2 also uses various chemicals for river water treatment, demineralization, biofouling, and condensate treatment. These chemicals and the quantities used are presented in Table 3.6-3.

The BVPS-2 site is similar to Beaver County in the proportion of land that is wooded. Of the 501-acre site, the permanent plant area for Shippingport Atomic Power Station (SAPS), Beaver Valley Power Station - Unit 1 (BVPS-1), and BVPS-2 occupies 125 acres (Table 5.7-1). Of this, the permanent facilities of the three units occupy approximately 53 acres which remain unvegetated during the lives of the power stations. Within the permanent station area, grass has been or will be planted in the remaining 72 acres of open area not occupied by station facilities; however, the grass will not be available to most wildlife due to the presence of fences. The combination of field land and industry is not unique in the BVPS area (Section 2.1.3), and BVPS-2 operation does not represent any significant land use change.

After the station is operational and the terrain is revegetated, animals entering the site will be those capable of penetrating fenced-in areas. Such animals include birds and burrowing mammals.

The approximately 101 acres of cleared land constitute the major natural resource committed during construction of BVPS-2. Normal succession plus deliberate planting on the approximately 30 acres of land unaffected by facilities, paving or fences will restore some wildlife habitat during station operation. Approximately 346 acres

of land within the site are undisturbed and left as forest and old field habitat.

Vegetation, timber resources, and animal communities are affected to varying degrees as a consequence of the land requirements. Mammals and birds displaced during land clearing constitute a minor fraction of the regional populations and are expected to return after construction activity decreases. The reproductive potential of these regional populations is not significantly impaired, and normal populations are expected to be maintained during the operation of BVPS-2.

When the disturbed land is restored during plant operation, the populations of species carried solely on grassland (for example, Eastern Meadowlark) are expected to increase.

The process of natural succession will allow partial self-restoration of some of the lost habitat following seeding or placement of other ground cover. As existing old fields and shrubby areas advance successionally towards forest and existing acreages of timberland mature, available forest habitat and food sources will increase, thus maintaining an increasing carrying capacity for many species as described in Section 2.2.

Following decommissioning, many areas of the site may be cleared of structures and permitted to revert to their natural state, as described in Section 5.8. Should the plant be totally dismantled, no land would be placed on permanent restriction. Thus, no land would be irretrievably lost. Other alternatives, which include inplace entombment, mothballing, or conversion to a new steam supply system, require various amounts of secured land (Section 5.8). Fewer than 16 acres of land would be required for BVPS-1 and BVPS-2 after decommissioning if the stations are not dismantled. The 138-kV and 345-kV switchyards will remain part of the system loop and will require an additional 12.3 acres (Section 5.8).

About 3.5 billion gallons of water are lost each year through drift and evaporation from the BVPS-2 natural draft cooling tower (Section 3.3). This average water loss totals about 0.05 percent of the annual flow of the Ohio River.

The operation of the proposed units will affect nearshore aquatic habitats to a limited degree. The presence of slope protection, barge slips, intake and discharge facilities, and sediment from the movement of equipment will result in a loss of river bottom and a consequent reduction in benthic biomass. A few individual fish may be lost due to this reduction in benthic biomass, but no permanent adverse impact on the aquatic populations is expected. Similarly, the covering of Peggs Run (Section 2.4) will result in a loss of productivity in about 1,850 feet of stream.

TABLE 5.7-1

LAND USE COMMITMENTS FOR THE BVPS SITE

Land Use		Acres
BVPS-2 acreage disturbed prior to operation		
Within permanent station area*	56	
Support facilities**	15	
To be restored outside fences	30	
Total		101
Permanent station acreage for existing units*		
SAPS***	29	
BVPS-1	25	
Total		54
Undisturbed land area		346
Total BVPS site area		501

NOTES:

*Plant facilities and lawns unavailable to major flora and fauna.
**Includes training center and emergency response facility.
***Shippingport Atomic Power Station.

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APPENDIX 5A

NATIONAL POLLUTION DISCHARGE ELIMINATION SYSTEM PERMIT

APPENDIX 5A

NATIONAL POLLUTION DISCHARGE ELIMINATION SYSTEM PERMIT

The National Pollution Discharge Elimination System permit will be provided in an amendment upon issuance by the Pennsylvania Department of Environmental Resources.

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TABLE 5C-2 (Cont)

Parameter	Values Assigned: Deer	Rabbit	Grouse	Pheasant
D/Q ($1/m^2$)				
Release point 1	2.86×10^{-8}	2.27×10^{-7}	2.27×10^{-7}	2.27×10^{-7}
Release point 2	3.56×10^{-9}	7.06×10^{-9}	7.06×10^{-9}	7.06×10^{-9}
Release point 3	6.39×10^{-9}	4.47×10^{-10}	4.47×10^{-10}	4.47×10^{-10}
Release point 4	3.56×10^{-9}	7.06×10^{-9}	7.06×10^{-9}	7.06×10^{-9}
C-14 fractional equilibrium				
Release point 1	0.0073	0.0073	0.0073	0.0073
Release point 2	1.0	1.0	1.0	1.0
Release point 3	1.0	1.0	1.0	1.0
Release point 4	1.0	1.0	1.0	1.0

NOTES:

1. BVPS-1 elevated release and BVPS-2 ventilation vent.
2. BVPS-1 ventilation vent.
3. BVPS-1 and BVPS-2 process vent.
4. BVPS-2 elevated release.
5. Deer is assumed to graze at the location of the maximum beef animal, 1,577 meters east-southeast.
6. Location of analysis is the site boundary, 567 meters northwest (sector with highest χ/Q and D/Q values).

TABLE 5C-3

DILUTION FACTORS, POPULATION SERVED, AND TRAVEL TIMES FROM SITE

Public Water Supply Systems[1]	Distance from Site to Point of Intake (river mile)	Dilution Factor	Population Served (people/yr)	Transit Time, Release to Intake (hour)
Midland, Pa.	1.3	623	9,600	1.4
East Liverpool, Ohio	5.2	623	26,000	5.7
Chester, W. Va.[2]	7.1	545	3,800	7.7
Toronto, Ohio	24.1	623	8,000	26.2
Wierton, W. Va.[3]	27.0	623	30,000	29.3
Steubenville, Ohio	30.2	623	35,000	32.8
Mingo Junction, Ohio	36.0	623	15,000	39.1
Wheeling, W. Va.	51.8	623	65,000	56.3
Martins Ferry, Ohio[3]	53.6	623	19,000	58.2
Bellaire, Ohio[3]	59.0	623	9,500	64.1

Incremental Regions[4] (river mile)	Distance from Site to Point of Analysis (river mile)	Dilution Factor	Population Usage[5] (annual attendance)	Transit Time, Release to Point of Analysis (hour)
0-11	5.5	489	65,812	6.0
11-22	16.5	621	65,812	17.9
22-33	27.5	623	65,812	29.9
33-44	38.5	623	65,812	41.8
44-55	49.5	623	65,812	53.8
55-66	60.5	623	65,812	65.7

Other Locations				
Discharge outfall[6]	0.0	1	-	0.0
Edge of initial mixing zone[7]	0.1	3	-	0.3
Junction of Little Beaver Creek[8]	4.0	623	-	4.4
Nearest residential river-bank wells Georgetown Borough Glasgow Borough Green Township	4.0	826	21(total)	4.4

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CHAPTER 6

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BVPS-2 which are pertinent to aquatic ecology are similar to those of BVPS-1, these data provide some measurement of the potential incremental impact of BVPS-2.

In February 1980, the technical specifications for aquatic ecology monitoring of BVPS-1 were amended after review and approval by the USNRC. The previous 5 years of preoperational data and 3 years of operational data indicated that operation of BVPS-1 caused no significant impact to biota as measured in the benthos, plankton, fish, ichthyoplankton, and plankton entrainment studies. The amendment concluded that these studies could be deleted from the aquatic ecology monitoring program.

6.1.2 Ground Water

Domestic water for BVPS-2 will be obtained from onsite wells as described in Section 3.3.2. Station operation will not affect regional or local ground-water supplies.

Offsite wells are isolated from ground water beneath the station as described in FSAR Section 2.4.13. Any future development of local or regional ground-water resources near BVPS-2 would likewise be isolated from ground water beneath the station. Aquifers on the north side of the Ohio River are isolated from the plant aquifer by the river, which forms a ground-water boundary. Site aquifer characteristics are discussed further in Section 2.4.

6.1.2.1 Physical and Chemical Parameters

Samples of ground water were obtained for chemical analysis. The results are presented in Table 2.4-2.

6.1.2.2 Models

Releases of radioactive liquid waste from BVPS-2 storage tanks were postulated in order to evaluate the dilution and dispersion of an accidental spill to the ground-water system. The results of this study are described in FSAR Section 2.4.13.3.

6.1.3 Air

The preoperational monitoring program described in the Environmental Report - Operating License Stage (ER-OLS) comprises a completely new system from that described in the ER-CPS, Section 5.5.2. The change in the monitoring program was primarily in response to the requirements of Regulatory Guide 1.23 (Safety Guide 23) issued in February 1972 regarding meteorological tower siting, instrument levels, parameters measured, instrument accuracy, data reduction, and maintenance schedules. A new 500-foot meteorological tower was located approximately 3,600 feet northeast of the previous tower location near the BVPS-2 containment building. New instrumentation which meets the accuracy specifications of Regulatory Guide 1.23 was

also installed on the new tower, along with a mini-computer for data recording purposes. Also, the recording levels on the tower were changed from 50 and 150 feet to 35, 150, and 500 feet.

6.1.3.1 Meteorology

The onsite meteorological program for the collection of data used in the Section 2.3 analyses began on January 1, 1976, and was an upgrading of the meteorological program for BVPS-1. A new 500-foot guyed meteorological tower was relocated approximately 3,600 feet northeast of BVPS-1. The base of the tower is approximately 730 feet above mean sea level. The meteorological monitoring system consisted of three levels of instrumentation on the tower. Wind speed and direction measurements were made at elevations of 35, 150, and 500 feet. Ambient temperature and dew point measurements were made at the 35-foot level. Temperature differential measurements were made between 35 feet and 150 feet (ΔT_{150}feet-$_{35}$ feet) and between 35 feet and 500 feet (ΔT_{500}feet-$_{35}$ feet). Precipitation data were obtained a few feet above the surface from a rain gauge near the base of the tower.

The tower is situated on a relatively flat plot of land in the Ohio River Valley and is enclosed by a fence. The ground surface in the immediate area was composed of slag and dirt.

Meteorological instrumentation on the tower included:

1. Wind instrumentation

   Climet wind direction and speed sensors at the 35-, 150-, and 500-foot levels,

2. Temperature instrumentation

   a. One Rosemont "RTB" at each of the 35-, 150-, and 500-foot levels,

   b. Endevco signal conditioners,

   c. Geotech aspirated solar radiation shields to house the RTBs at the 35-, 150-, and 500-foot levels,

3. Dew point instrumentation

   One Cambridge System dew point measuring unit at the 35-foot level,

6.2 APPLICANT'S PROPOSED OPERATIONAL MONITORING PROGRAMS

It is anticipated that much of the environmental monitoring program for the Beaver Valley Power Station - Unit 2 (BVPS-2) site will be based on the ongoing Beaver Valley Power Station - Unit 1 (BVPS-1) operational monitoring programs. These programs are governed by the current Environmental Technical Specifications for BVPS-1. Brief summaries of these programs and some of the proposed modifications are included in the following sections. The Environmental Report - Construction Permit Stage did not discuss the specifics of proposed operational monitoring programs for BVPS-2.

6.2.1 Surface Waters

6.2.1.1 Physical and Chemical Parameters

6.2.1.1.1 Physical Parameters

A description of the thermal effluent and its physical effects is presented in Section 5.1.2. Due to the small thermal effect on the Ohio River, no major operational monitoring program is planned. The temperature of the plant blowdown will be measured at the Beaver Valley Power Station discharge location.

6.2.1.1.2 Chemical Parameters

Liquid waste effluent monitoring will be conducted in accordance with the requirements of the final National Pollution Discharge Elimination System (NPDES) discharge permit.

6.2.1.2 Ecological Parameters

Aquatic ecological monitoring will be conducted in accordance with requirements of the NPDES discharge permit. The impingement monitoring conducted for the BVPS-1 operational program is expected to continue for 1 year after BVPS-2 operation begins. Based on findings at BVPS-1 and the low potential for ecological impacts associated with BVPS-2, this monitoring should be sufficient to provide an indication of the ecological effects of BVPS-2 operation.

6.2.2 Ground Water

No operational ground-water monitoring program is planned (Section 6.1.2) other than that being conducted as part of the environmental radiological monitoring program (Section 6.1.5).

6.2.3 Air

The Applicant's current preoperational meteorological monitoring program is described in detail in Section 6.1.3. This program will continue as the operational monitoring program. The operational

program will meet the intent of Regulatory Guide 1.23, Revision 1 requirements.

6.2.4 Land

6.2.4.1 Geology and Soils

At present, no soils monitoring program is planned other than that being conducted as part of the environmental radiological monitoring program (Section 6.2.5).

6.2.4.2 Land Use and Demographic Surveys

No ongoing land use or demographic surveys are planned at this time.

6.2.4.3 Ecological Parameters

Infrared aerial photographs will be taken every other year. The photographs will be compared with preoperational photographs of the BVPS-2 area (Section 6.1.4.3). Any sign of stress due to salt drift or other sources could then be field checked.

6.2.5 Radiological Monitoring

The operational phase environmental radiological monitoring program is a continuation of the preoperational monitoring program discussed in Section 6.1.5. The program monitors selected environmental parameters identified during the preoperational phase both in areas potentially affected by the plant and in control areas beyond the measurable influence of the plant. The program determines whether BVPS-2 radioactive releases result in radioactivity concentrations or radiation doses in excess of the limits specified in 10 CFR 20 and 10 CFR 50, Appendix I.

The BVPS-1 environmental radiological monitoring program outlined in the BVPS-1 Environmental Technical Specifications serves as the operational program for BVPS-2. Results of sample analyses, summaries, and interpretation of data will be provided in accordance with technical specifications.

6.2.6 Noise

Anticipated noise impact on the adjacent community from the operation of BVPS-2 is described in Section 5.6.1. The noise impact is minimal due to the existing noise levels from other industrial noise sources; therefore, no operational noise monitoring program is planned.

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CHAPTER 7

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CHAPTER 7

ENVIRONMENTAL EFFECTS OF ACCIDENTS

7.1 STATION ACCIDENTS INVOLVING RADIOACTIVITY

7.1.1 Introduction

During normal operation of the Beaver Valley Power Station - Unit 2 (BVPS-2), small quantities of radioactive nuclides including fission products and activation products are released to the environment. The design and operation of BVPS-2 is such that releases of these radionuclides are within the limits specified by U.S. Nuclear Regulatory Commission (USNRC) regulations 10 CFR 20, 10 CFR 50 Appendix I, 10 CFR 100, and the U.S. Environmental Protection Agency (USEPA) regulation 40 CFR 190.

Any unintentional event not considered as part of normal plant operation which leads to the release of radioactive nuclides into the environment is defined as an accident. Chapter 15 of the FSAR discusses station accidents using conservative analyses. This section re-examines the less probable but more severe accidents, including steamline breaks, steam generator tube ruptures, the design basis loss-of-coolant accident (LOCA), and, in accordance with the USNRC's statement of interim policy (45 Federal Register 40101), the most severe, or Class 9, accidents. The analyses of these accidents use less conservative parameters to provide a realistic assessment of the health effects to the public and the economic costs.

7.1.2 Mitigation of Accident Consequences

7.1.2.1 Design Features

The BVPS-2 is designed to prevent accidental release of radioactive fission products from the fuel and to lessen the consequences should such a release occur. Many of the design and operating specifications of BVPS-2 systems are derived from the analysis of postulated events known as design basis accidents (DBA). These accident preventive and mitigative systems are collectively referred to as engineered safety features (ESF). The possibilities or probabilities of failure of these systems are incorporated in the assessments discussed in Section 7.1.3.2.

The ESFs consist of a number of plant systems. The emergency core cooling system (ECCS) consists of high head safety injection/charging pumps, the refueling water storage tank, low head safety injection pumps, recirculation spray pumps, and the safety injection accumulators with associated valves, instrumentation, and piping. The containment is of the sub-atmospheric type and consists of a reinforced reactor building with a steel plate liner designed to minimize accidental release of radioactivity to the environment. The containment depressurization system consists of the quench spray and

recirculation spray systems. Other ESF systems at BVPS-2 include the post-DBA hydrogen control system, the supplemental leak collection and release system, and the containment isolation system. Detailed descriptions of these ESFs are provided in FSAR Chapter 6. Each of the ESF systems mentioned previously is supplied with emergency power from onsite diesel generators in the event that normal offsite station power is interrupted.

The fuel handling building for BVPS-2 also has accident-mitigating systems. The safety grade ventilation system contains both charcoal and high efficiency particulate filters. This ventilation system is designed to keep the area around the spent fuel pool below the prevailing barometric pressure during fuel handling operations so that outleakage will not occur through building openings. If radioactivity were to be released into the building, it would be drawn through the ventilation system, and most radioactive iodine and particulate fission products would be removed from the flow stream before exhausting to the outdoor atmosphere.

Some plant systems that are necessary for power generation can also play a role in mitigating certain accident consequences. For example, the main condenser, although not classified as an ESF, can act to mitigate the consequences of accidents involving leakage from the primary to the secondary side of the steam generators (such as steam generator tube ruptures). If normal offsite power is maintained and the turbine bypass system is operable, the ability of the plant to send contaminated steam to the condenser instead of releasing it through the safety valves or atmospheric dump valves can significantly reduce the amount of radioactivity released to the environment.

7.1.2.2 Site Features

The USNRC's reactor site criteria, found in 10 CFR 100, require that the site for a power reactor have certain characteristics that tend to reduce the risk and potential impact of accidents. The discussion that follows briefly describes the BVPS-2 site characteristics and how they meet these requirements.

The site has an exclusion area as required by 10 CFR 100: a radius of 1,500 feet from the BVPS-2 reactor, extending in part to the north shore of the Ohio River. The BVPS site contains approximately 501 acres, of which 479.5 acres are owned by the Duquesne Light Company (DLC) and 21.5 acres are owned jointly by the Central Area Power Coordinating group (CAPCO). Phillis Island, located in the Ohio River approximately 400 feet off the shoreline from BVPS, is owned by the Dravo Corporation. An agreement is in place, binding on Dravo Corporation and any future lessee or purchaser, which restricts the uses that can be made of the island. Immediately to the west of the BVPS-2 reactor location, and also onsite, are Beaver Valley Power Station - Unit 1 (BVPS-1) and the Shippingport Atomic Power Station (SAPS). The latter is managed by DLC for the Division of Naval

Reactors, U.S. Department of Energy (USDOE), formerly ERDA. The SAPS terminated operation in 1982 and is scheduled for decommissioning by the USDOE. The SAPS area is leased by DLC to the USDOE.

The low population zone (LPZ) surrounding BVPS-2 encompasses an area within an approximate 3.6-mile radius of the BVPS-2 reactor containment centerline. The distance for the LPZ is based on the requirements of 10 CFR 100. The 3.6-mile radius meets the requirement that the LPZ be an area in which sufficient protective measures can be taken to assure that the resident population does not receive a dose in excess of a specified level resulting from a postulated accident condition. No population centers with populations equal to or greater than 25,000 exist within the LPZ, or within an area of radius 1 1/3 times the LPZ radius, approximately 4.8 miles.

The population center nearest to BVPS-2, as defined by 10 CFR 100, is the township of McCandless, Pennsylvania, which supported approximately 26,250 people in 1980 at a density of 1,608 people per square mile. The township's closest corporate boundary to the station is approximately 17 miles east of BVPS-2. Existing population centers with over 25,000 people are presented in FSAR Table 2.1-24. Cities and towns projected to become population centers with over 25,000 people during the period 1980 to 2030 are listed in FSAR Table 2.1-25. The population distribution for 1980, based on 1980 census data, is found in FSAR Tables 2.1-3, 2.1-10, and 2.1-17.

The safety analysis of the BVPS-2 site has also included a review of potential external hazards generated by man (offsite activities that might cause an accident). This review encompassed nearby industrial, transportation, and military facilities that might create explosive, missile, toxic gas, or similar hazards. The risk to BVPS-2 from such hazards has been found to be negligible. A more detailed discussion of the compliance with siting criteria and the consideration of external hazards is given in FSAR Section 2.2. Also discussed in the FSAR are design provisions required for severe natural phenomena such as earthquakes, floods, and tornadoes.

7.1.2.3 Emergency Preparedness

An Emergency Preparedness Plan for BVPS is in place and is exercised on an annual basis. The plan applies to the operation of BVPS-1 during the construction of BVPS-2 and will be modified to address both units upon completion and operation of BVPS-2.

The BVPS Emergency Preparedness Plan provides guidance for coping with both onsite and offsite emergency situations. It ranges in scope from relatively minor unusual events and occurrences involving small releases of radioactive material to a major nuclear accident having significant offsite radiological consequences. This plan, together with the interrelated state and county emergency plans,

provides detailed arrangements for extending emergency measures to a radius of approximately 10 miles from BVPS. The elements of response to offsite emergency conditions are continued in the emergency plans and emergency operating procedures of the responsible offsite emergency organization. The 10-mile emergency planning zone surrounding BVPS encompasses three states: Pennsylvania, Ohio, and West Virginia; and three counties: Beaver County, Pa; Columbiana County, Ohio; and Hancock County, West Virginia. Continuing liaison with the emergency organizations of these jurisdictions ensures compatability with the BVPS plan. The coordination and liaison with offsite emergency organizations include formal agreements that individual organizations will perform their respective emergency functions in response to requests from BVPS.

7.1.3 Accident Risk and Impact Assessment

7.1.3.1 Design Basis Accidents

This section evaluates the environmental impact of postulated accidents and occurrences involving radioactivity. Assumptions used in these evaluations are based on normal operating conditions. This is in contrast to the highly conservative assumptions employed in the FSAR, where worst-case conditions are postulated.

Radiation doses are calculated using the core radioactivity given in FSAR Table 12.2-3, a χ/Q value of 9.6×10^{-4} sec/m^3, and the calculational methods described in Regulatory Guides 1.4, Revision 2, and 1.145, Revision 1, except that the average breathing rate is used. The χ/Q value of 9.6×10^{-4} sec/m^3 is the 0- to 2-hour value for the northwest sector of the exclusion area boundary. This is the 50-percent direction dependent (worst sector) value and is based on onsite meteorological data from January 1, 1976 through December 31, 1980. Section 2.3.4 describes the methodology used to develop meteorological values. Table 7.1-1 gives the integrated gamma, beta, and thyroid doses at the exclusion area boundary (EAB) from these accident analyses.

As is evidenced in Table 7.1-1, the resulting doses from the accidents described in the following sections are well below the values specified in 10 CFR 100. The environmental impact from such accidents would be minimal.

The design conditions are categorized as follows:

1. Incidents of moderate frequency

   These are events that can reasonably be expected to occur during any year of operation.

2. Infrequent accidents

   The following are incidents which might occur during the life of the plant:

   a. Process gas charcoal bed adsorber system release,

   b. Steam generator tube rupture,

   c. Refueling accidents in the containment structure,

   d. Fuel assembly drop in the fuel storage pool,

   e. Heavy object drop onto the fuel storage rack,

   f. Fuel cask drop in the fuel building,

   g. Loss of coolant through a small pipe break (6-inch diameter or less) or from a crack in a large pipe, and

   h. Small steam line break outside the containment structure.

3. Limiting Faults

   These faults are not expected to occur during the life of the plant, but are postulated because they would incur the potential for release of significant amounts of radioactive material. They are the most drastic, and therefore represent the limiting design cases.

   a. Major rupture of a pipe containing reactor coolant, the rupture larger than that defined as an infrequent accident and up to and including double-ended rupture of the largest pipe in the reactor coolant pressure boundary,

   b. Ejection of any single control rod, and

   c. Major steam or feedwater system pipe rupture outside containment up to and including double-ended rupture of the largest steam line pipe.

7.1.3.1.1 Incidents of Moderate Frequency

Examples of these incidents include releases through steam line relief valves and small spills and leaks of radioactive material outside the containment structure. Releases from these minor incidents are included in expected normal operational releases given in Section 3.5. The resulting radiological impact is given in Section 5.2.

7.1.3.1.2 Infrequent Accidents

7.1.3.1.2.1 Gaseous Releases from the Process Gas Charcoal Bed Adsorber in the Radioactive Gaseous Waste System

Description of Accident

The radioactive gaseous waste system contains four process gas charcoal bed adsorbers with approximately 1,000 pounds of charcoal adsorbent in each vessel. It is postulated, for the purpose of this accident analysis, that a leak occurs in the piping leading into the adsorbers during normal full power operation of the plant.

Assumptions

The following assumptions are used for this accident analysis:

1. The entire contents of the piping from the degasifier to the charcoal beds, plus the amount of gas that leaks back from the process gas charcoal bed adsorbers, are released. The radioactive gaseous waste system includes a surge tank for temporary storage of processed gas and seven gaseous waste storage tanks which are used for storage of processed gas during cold shutdown conditions. A rupture with a subsequent release from one of these tanks is not as limiting as the accident analyzed. The vessels containing waste gas with the largest volume and highest radioactivity that might be accidentally released are the process gas charcoal bed adsorbers and associated piping.

2. Normal operation letdown purification flow rate is 60 gpm.

3. Pipe volume is 6 ft^3.

4. Radioactivity of noble gases and halogens is based on operation with 0.5-percent fuel defects.

5. Maximum expected charcoal delay bed holdup is 2.6 days for krypton and 45.7 days for xenon.

Measures to Prevent Occurrence and Mitigate Consequences

This system is designed in accordance with Regulatory Guide 1.143 requirements. Periodic maintenance and installed radiation monitors in the auxiliary building permit location and repair of any small leaks which might develop. It is not considered that an accident of this type could occur during the life of the plant.

7.1.3.1.2.2 Steam Generator Tube Rupture

Description of Accident

For this accident analysis, it is postulated that a significant leak develops from the reactor coolant system to the secondary system as the result of a steam generator tube rupture.

Assumptions

The following assumptions are used for this accident analysis:

1. All noble gases and 0.1 percent of the halogens in the steam reaching the condenser are released by the condenser air ejector.

2. Secondary side steam and liquid equilibrium radioactivity concentrations prior to the accident are based on a normal expected rate of 100 pounds per day primary to secondary steam generator leakage in accordance with NUREG-0017 (USNRC 1976), with normal operation steam generator blowdown rate of 15 gpm (at 120°F) per steam generator to the flash tank where 14.5 percent of the iodines and 100 percent of the noble gases are returned to the secondary side.

3. Fifteen percent of the average activity of noble gases and halogens in the reactor coolant is released into the secondary side of the steam generator. The releases are based on an iodine partition factor of 10 in the steam generators, as given in NUREG-0017. Average radioactivity of noble gases and halogens in the reactor coolant system prior to the accident is based on operation with 0.5-percent fuel defects.

4. The condensate polishing system is normally operating at 8.5×10^6 lb/hr with a condensate demineralizer decontamination factor of 10 for iodines, in accordance with NUREG-0017.

Measures to Prevent Occurrence and Mitigate Consequences

The steam generator tube side and shell side are designed to ASME III, Code Classes 1 and 2, respectively. Both sides are designed to Seismic Category I requirements. The tubes are seamless Inconel 600, a highly ductile material. A complete tube severance is highly unlikely. The more probable mode of tube failure is several minor leaks.

7.1.3.1.2.3 Refueling Accidents

Refueling accidents occur inside the containment structure and result in fuel clad rupture. The accidents postulated are a fuel assembly

drop and a heavy object drop onto fuel in the core. For both cases, radioactivity is released to the refueling cavity. The refueling cavity water retains most of the halogens but does not hold up the release of noble gases. However, release from the containment is precluded by a design which automatically isolates the containment following the detection of radioactivity by the redundant containment purge monitors. The time required for any release activity to travel to the first containment isolation valve is greater than the detector response time plus the closure time of the containment isolation valves. The activity released to the containment atmosphere is reduced by radioactive decay and the containment air filtration system. The atmosphere is subsequently purged under controlled conditions.

7.1.3.1.2.4 Fuel Assembly Drop in the Fuel Pool

Description of Accident

For this accident analysis, it is postulated that a single fuel assembly is dropped in the fuel pool as it is being transferred to the spent fuel storage racks.

Assumptions

The following assumptions are used for this accident analysis:

1. The iodine decontamination factor in the water-filled fuel pool is 500.

2. One week decay time before the accident occurs.

3. The gap radioactivity of noble gases and halogens in one row of fuel pins is released into the water-filled fuel pool. The gap radioactivity is 1 percent of the total radioactivity in a fuel pin.

4. Carbon adsorbers in the fuel building ventilation system have an efficiency of 99 percent for the iodines released by the accident. This is conservative based upon experimental data.

Measures to Prevent Occurrence and Mitigate Consequences

All fuel-handling operations are conducted in accordance with prescribed procedures and under the supervision of senior licensed personnel trained in nuclear safety. Special precautions are taken in all fuel-handling operations to minimize the possibility of damage to fuel assemblies. The water covering the fuel would dissolve a large portion of any released iodine.

Prior to beginning the fuel-handling operations, one of the emergency filtration trains of the fuel building ventilation system is placed

in operation. This system maintains a reduced pressure in the fuel building to prevent building out-leakage and filters the ventilation exhaust to remove radioactivity.

7.1.3.1.2.5 Heavy Object Drop Onto Fuel Storage Rack

Description of Accident

For this accident analysis, it is postulated that a heavy object is dropped onto the spent fuel storage rack, causing a fuel assembly to rupture.

Assumptions

The following assumptions are used for this accident analysis:

1. The iodine decontamination factor in the water-filled spent fuel pool is 500.

2. A 30-day decay time before the accident occurs.

3. The gap radioactivity of noble gases and halogens in one average fuel assembly is released into the water-filled spent fuel pool. The gap radioactivity is 1 percent of the total radioactivity in a fuel pin.

4. Carbon adsorbers in the fuel building ventilation system have a removal efficiency of 99 percent for the iodines released by the accident. This is conservative based upon experimental data.

Measures to Prevent Occurrence and Mitigate Consequences

All fuel-handling operations are conducted in accordance with prescribed procedures and under the supervision of senior licensed personnel trained in nuclear safety. The only objects normally suspended in or over the spent fuel storage rack are fuel assemblies, handling tools, or other devices necessary for fuel-handling inspection. To ensure that a fuel assembly being transferred to the spent fuel storage rack always has sufficient water shielding over it, the fuel assembly is raised a minimum distance over the spent fuel storage rack. Handling tools could cause minor damage if dropped on a fuel assembly, but not to the extent of a heavy object drop. The fuel handling crane is a heavy object which is routinely positioned over the spent fuel storage rack during fuel handling and inspection. The fuel handling crane is designed to Seismic Category I requirements and is designed to prevent derailment during a seismic event.

The possibility of a heavy object being dropped onto the spent fuel storage rack during the life of the plant is remote.

7.1.3.1.2.6 Fuel Cask Drop

A fuel cask drop accident is defined as the dropping of the spent fuel cask from a height of 9.1 meters (30 feet) onto a hard unyielding surface. All refueling operations are conducted in accordance with approved procedures. The spent fuel shipping cask will be designed to withstand a 9.1-meter (30-foot) drop onto an unyielding surface without damage to the cask, in accordance with 10 CFR 71 requirements. During the entire transfer operation, the cask will not be lifted 9.1 meters (30 feet) or more above any surface.

Since potential cask drop distances are less than 9.1 meters (30 feet), a spent fuel cask drop accident is not limiting and does not result in radioactivity release.

7.1.3.1.2.7 Loss of Coolant Through a Small Pipe Break

Description of Accident

For this accident analysis, it is postulated that a small pipe breaks and results in the loss of all the reactor coolant inventory into the containment. The pipe size considered is 6 inches in diameter or less.

Assumptions

The following assumptions are used for this accident analysis:

1. The effects of plateout, chemical additive sprays, and decontamination factor of water collected on the containment floor (from spilled reactor coolant and containment spray) provide a halogen reduction factor of 0.05.

2. The containment structure leakage rate is 0.1 volume percent per day for the first hour, and zero thereafter. This value is demonstrated to be a maximum by the Type A testing program per Appendix J to 10 CFR 50. The ESF systems are designed to reduce containment pressure below atmospheric within 1 hour after the LOCA, terminating leakage from the containment to the environment.

3. The analysis is based on a reactor coolant system liquid mass of 420,000 pounds as given in Section 3.5.

4. Average radioactivity of noble gases and halogens in the reactor coolant system prior to the accident is based on operation with 0.5-percent fuel defects.

Measures to Prevent Occurrence and Mitigate Consequences

The piping and vessels in the reactor coolant system are fabricated of corrosion-resistant materials with all components having welded fittings. The system is designed and tested according to ASME Boiler and Pressure Vessel Code Section III, Code Class 1 specifications and meet Seismic Category I requirements. Inservice inspection is performed periodically according to the guidelines of ASME Boiler and Pressure Vessel Code Section XI. Because of these precautions and high standards of design, it is unlikely that the piping will rupture in the manner postulated during the life of the plant.

7.1.3.1.2.8 Small Steam Line Break Outside Containment Structure

Description of Accident

For this accident analysis, a small break is postulated to occur in a main steam line outside the containment structure.

Assumptions

The following assumptions are used for this accident analysis:

1. During the course of the accident, a halogen reduction factor of 0.1 is applied to the reactor coolant radioactivity.

2. The volume of one steam generator is released to the atmosphere with an iodine partition factor of 10.

3. Secondary side steam and liquid equilibrium radioactivity concentrations prior to the accident are based on a normal expected rate of 100 pounds per day primary to secondary steam generator leakage in accordance with NUREG-0017, with a normal operation steam generator blowdown rate of 15 gpm (at 120°F) per steam generator to the flash tank where 14.5 percent of the iodines and 100 percent of the noble gases are returned to the secondary side.

4. Average radioactivity of noble gases and halogens in the reactor coolant system prior to the accident is based on operation with 0.5-percent fuel defects.

Measures to Prevent Occurrence and Mitigate Consequences

The main steam lines are designed according to ASME Section III, Code Class 2 specifications and Seismic Category I requirements up to and including the main steam isolation valve outside the containment structure. A steam line break would initiate a safety injection (SIS) signal and steam line isolation (SLI) signal. The SIS signal closes the main steam isolation valves in the main steam lines to limit steam loss. Upon receipt of the SIS signal, the auxiliary feed

pumps start and supply water to the steam generators. The main steam isolation valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. It is considered unlikely that an accident of this type could occur.

7.1.3.1.3 Limiting Faults

Limiting faults are accidents that are not expected to occur during the life of the plant, but are considered because they have the potential to incur significant releases of radioactivity. These accidents, as well as the infrequent accidents addressed in Section 7.1.3.1.2, are designated design basis because specific design and operating features, as described in Section 7.1.2, are provided to limit their potential radiological consequences.

7.1.3.1.3.1 Loss of Coolant Through a Large Pipe Break

Description of Accident

For this accident analysis, it is postulated that a large pipe breaks and results in the loss of all the reactor coolant inventory into the containment. The pipe considered here is a 31-inch inside diameter reactor coolant pump suction pipe.

Assumptions

The following assumptions are used for this accident analysis:

1. The effects of plateout, chemical additive sprays, and decontamination factor of water collected on the containment floor (from spilled reactor coolant and containment spray) provide a halogen reduction factor of 0.05.

2. The containment structure leakage rate is 0.1 volume percent per day for the first hour, and zero thereafter. This value is demonstrated to be a maximum by the Type A testing program per Appendix J to 10 CFR 50. The ESF systems are designed to reduce containment pressure below atmospheric within 1 hour after the LOCA, terminating leakage from the containment to the environment.

3. The analysis is based on a reactor coolant system liquid mass of 420,000 pounds as given in Section 3.5.

4. Average radioactivity of noble gases and halogens in the reactor coolant system prior to the accident is based on operation with 0.5-percent fuel defects. Additionally, 2 percent of the core inventory of halogens and noble gases is assumed to be released to the primary coolant upon initiation of the accident.

Measures to Prevent Occurrence and Mitigate Consequences

The piping and vessels in the reactor coolant system are fabricated of corrosion-resistant materials with all components having welded fittings. The system is designed and tested according to ASME Boiler and Pressure Vessel Code Section III, Code Class 1 specifications and meet Seismic Category I requirements. Inservice inspection is performed periodically according to the guidelines of ASME Boiler and Pressure Vessel Code Section XI. Because of the precautions and high standards of design, it is unlikely that the piping will rupture in the manner postulated during the life of the plant.

7.1.3.1.3.2 Rod Ejection Accident

Description of Accident

For this accident analysis, it is postulated that a mechanical failure of a control rod drive mechanism pressure housing results in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution possibly leading to localized fuel rod damage.

Assumptions

The following assumptions are used in this accident analysis:

1. The effects of plateout, chemical additive sprays, and decontamination factor of water collected on the containment floor (from spilled reactor coolant and containment spray) provide a halogen reduction factor of 0.05. Manual or automatic initiation of the containment spray system is assumed.

2. The containment structure leakage rate is assumed to be 0.1 volume percent per day for the first hour, and zero thereafter. This value is demonstrated to be a maximum by the Type A testing program per Appendix J to 10 CFR 50. The ESF systems are designed to reduce containment pressure below atmospheric within 1 hour after the LOCA, terminating leakage from the containment to the environment.

3. The analysis is based on a reactor coolant system liquid mass of 420,000 pounds as given in Section 3.5.

4. Average radioactivity of noble gases and halogens in the reactor coolant system prior to the accident is based on operation with 0.5-percent fuel defects. Additionally, 0.2 percent of the core inventory of halogens and noble gases is assumed to be released to the primary coolant upon initiation of the accident.

Measures to Prevent Occurrence and Mitigate Consequences

The latch mechanism housing and rod travel housing are each a single length of forged type 304 stainless steel. The entire rod mechanism is designed and tested according to ASME Section III, Code Class 1 specifications and meets Seismic Category I requirements. Plant operation using chemical shim for reactivity control inherently limits the severity of rod ejection. Procedures for full-power operation limit the insertion of control rod banks to limit the reactivity change due to a rod ejection incident. The reactor shutdown instruments are designed to prevent damage to the core by initiating rapid and secure shutdown in the event of severe reactivity changes. It is considered unlikely that such an accident could occur.

7.1.3.1.3.3 Large Main Steam Line Break Outside Containment Structure

Description of Accident

For this accident analysis, it is postulated that a large break occurs in a main steam line outside the containment structure.

Assumptions

The following assumptions are used for this accident analysis:

1. During the course of the accident, a halogen reduction factor of 0.5 is applied to the reactor coolant radioactivity.

2. The volume of one steam generator is released to the atmosphere with an iodine partition factor of 10.

3. Secondary side steam and liquid equilibrium radioactivity concentrations prior to the accident are based on a normal expected rate of 100 pounds per day primary to secondary steam generator leakage, in accordance with NUREG-0017, with a normal operation steam generator blowdown rate of 15 gpm (at 120°F) per steam generator to the flash tank where 14.5 percent of the iodines and 100 percent of the noble gases are returned to the secondary side.

4. Average radioactivity of noble gases and halogens in the reactor coolant system prior to the accident is based on operation with 0.5 percent fuel defects.

Measures to Prevent Occurrence and Mitigate Consequences

The main steam lines are designed according to ASME Section III, Code Class 2 specifications and Seismic Category I requirements up to and including the main steam isolation valve, so that a steam line break

upstream of this valve is considered unlikely to occur. A steam line break would initiate an SIS and an SLI signal. The SLI signal closes the main steam isolation valves in the main steam lines to limit steam loss. Upon receipt of the SIS, the auxiliary feed pumps start and supply water to the steam generators. It is considered unlikely that an accident of this type would occur.

7.1.3.2 Probabilistic Assessment of Severe Accidents

The probabilities and consequences of accidents of greater severity than design basis accidents are discussed in this section. These severe accidents are distinguished from design basis accidents in two main respects. The accidents result from unlikely multiple system failures leading to inadequate cooling of the core. Continuance of this state leads to severe overheating and subsequent melting of the core. Additionally, severe accident scenarios include a loss of containment integrity, where the containment does not perform its intended function of limiting the release of radioactive material to the environment.

The probabilistic assessment technique employed herein is that described in the Reactor Safety Study (RSS) which was published by the USNRC in 1975 as WASH-1400 (USNRC 1975). In another study subsequent to the RSS, published as NUREG/CR-0400, consideration was given to those accident sequences that were dominant contributors to risk (USNRC 1978b). This "rebaselining" of accident sequences and their associated probabilities of occurrence was an improvement on the WASH-1400 study, but was also limited by its generic applicability. The original WASH-1400 study and the NUREG/CR-0400 rebaselining used the Surry plant as a model. The probability of occurrence of various accidents, or plant damage states, is a function of the actual equipment, equipment location, and containment type particular to each plant. Within the past two years, several major probabilistic risk assessment (PRA) studies (Zion, Indian Point, Sizewell, Seabrook, Millstone) have been completed or are in progress. In order to better represent BVPS-2, the configurations and equipment layouts of these nuclear stations were reviewed and compared with plant features specific to BVPS-2.

The BVPS-2 differs in several respects from the Surry plant used as a model in the WASH-1400 study. For example, BVPS-2 has a core rated at 2,652 MWt, as compared to the 2,441 MWt rating for the Surry core. The BVPS-2 containment net free volume is 1.73×10^6 ft^3, while the Surry containment volume is 1.86×10^6 ft^3. An important consideration not addressed by the WASH-1400 study is the unavailability of water to the cavity directly below the reactor vessel during a core melt accident. This is significant in terms of noncondensible gas generation and basemat penetration during accident progression. The BVPS-2 residual heat removal (RHR) system heat exchanger is located within the containment building whereas the Surry RHR heat exchanger is located in the auxiliary building.

Such basic differences in plant design indicated that the WASH-1400 release categories and NUREG/CR-0400 accident sequences were not directly applicable to BVPS-2. In place of these categories or sequences, a mini-PRA study was performed; highlights of the methodology employed follow.

7.1.3.2.1 Methodology

The frequency of various initiating events particular to BVPS-2 was established by using the event tree methodology described in WASH-1400. Analyses were made of the core and containment response to these initiating events to determine accident classes. The accident classes were arranged into release categories. The values of fractional release, together with time, duration, and enthalpy of release, were used as input to the CRAC2 code to determine consequences. The final step in the assessment was to determine risk curves by a matrix multiplication process. Details of the various analytical procedures are given in the following sections.

7.1.3.2.1.1 Initiating Event Frequencies

Initiating events that were considered for the BVPS-2 study included various loss-of-coolant accidents and transients. The loss-of-coolant accidents considered were large LOCA, intermediate LOCA, small LOCA, steam generator tube rupture, incore instrument tube rupture, reactor vessel rupture, and interfacing systems LOCA. A comprehensive transient initiator list was established based upon the Zion Probabilistic Safety Study (Commonwealth Edison Company 1981), WASH-1400 (USNRC 1975), Electric Power Research Institute 1982 (EPRI) report NP-2230 (1982), and the Sizewell B Safety Study (Westinghouse Electric Corporation 1982). The transient initiators were grouped into categories based on plant response, signal actuation, systems required for mitigation, and subsequent plant-related effects. Only internal initiating events were considered. These are events which result from failures within the plant systems or within the electrical grid associated with the plant.

7.1.3.2.1.2 Accident Sequence Analysis

For each of the initiating event categories considered, an event tree analysis was carried out to define accident sequences and to quantify the frequency of plant states. The results of this accident sequence analysis were summarized as a "plant matrix" whose entries are the conditional probabilities of reaching a plant state given the occurrence of an initiating event.

To include the effects of failure of major support systems (such as offsite power, onsite emergency power, and service water), six plant support states were defined, and the failure of engineered safety systems was quantified for each of these states. Failure probabilities of plant-specific engineered safety systems were

quantified whenever needed. Common cause failure rates were included in the calculation of the system failure rates.

7.1.3.2.1.3 Core and Containment Severe Accident Response

An evaluation of primary and containment system response to postulated severe accident scenarios for BVPS-2 was made using design and functional information. Information such as containment volumes, containment design pressure, reactor cavity dimensions, basemat thickness, refueling water storage tank capacities, and quench and recirculation spray pump flow rates were compared with the same parameters of various plants of similar design.

The BVPS-2 reactor coolant system is similar in design to that of Zion. In the Zion Probabilistic Safety Study, detailed phenomenological studies of reactor core damage and related containment effects are discussed. These phenomena include core overheating, core melting, core slump, in-vessel steam explosions and spikes, steam production, steam/hydrogen blowdown, reactor core melt debris dispersal, and long-term debris bed cooling. Unlike the Zion study, thermal hydraulic computer codes were not used to determine containment pressures and temperatures resulting from these phenomena for BVPS-2. Instead, detailed containment response evaluations performed recently for a similarly designed containment were used as a baseline. The similarities and differences between the two plants were examined with respect to the effect they would have on the response of the plant to severe accident scenarios. Supplemental calculations were made to quantify the containment pressure response for BVPS-2.

7.1.3.2.1.4 Release Category Assignment

The characteristics of the fission product releases associated with containment failure depend upon the mode of failure (such as overpressure or bypass), on the timing of failure (early or late), and on containment phenomena which affect source terms. The status of the spray system is also important, because significant removal of certain radionuclides from the containment atmosphere would occur prior to containment failure. In prior Zion, Sizewell, and Millstone studies, relevant data from the thermo-hydraulic analyses described in the previous section were used to estimate the timing of fission product release. These data included time of core melt, time of containment failure (for accident sequences where the containment failure is indicated), duration of release, heat content of the release, containment pressure, temperature, temperature difference between the containment atmosphere and walls, and the mole fraction of various gases generated by the accident. Starting with the WASH-1400 source term from the core, this information was used to calculate various removal processes (such as sprays or deposition), which attenuated the source term while residing in the containment. Some 38 accident sequences were analyzed and grouped into release

categories based upon nodes in the containment event tree, timing, and magnitude of fission product release.

The release categories applicable to BVPS-2 are as follows:

BV-1 This release category is restricted to core melt sequences where a containment bypass directly to the environment could exist through a steam generator tube rupture. Prior bypass sequences in other plant studies considered release through the RHR system and auxiliary building. Since the RHR system at BVPS-2 is located within the containment building rather than the auxiliary building, this pathway was eliminated from the present analysis.

BV-2 and BV-3 These release categories are used for those accident sequences which could lead to an early overpressure failure of the containment with no sprays operational. Release category BV-2 accounts for early core melt sequences with a short warning time for evacuation, while BV-3 accounts for late core melt sequences with a slightly longer warning time for evacuation.

BV-4 and BV-5 These release categories are used for core melt accidents which could lead to intermediate containment failure times without containment sprays operational. Release category BV-4 accounts for late melt sequences while BV-5 accounts for early melt sequences.

BV-6 This release category is used for core melt accident sequences which could lead to late containment failure times without containment sprays in operation.

BV-7 This release category is used for core melt sequences which could lead to intermediate containment failure times with functional containment sprays.

BV-8 This release category is used for core melt sequences which could lead to late containment failure times with functional containment sprays.

BV-9 This release category is used for core melt sequences which could lead to basemat melt-through.

7.1.3.2.1.5 Consequence Analysis

For each of the nine release categories, the magnitudes of the release are calculated by multiplying the core inventory by the release fractions shown in Table 7.1-2. The core inventory used in the analyses was based on ORIGEN (Bell 1973; Rose and Burrows 1976) calculations for a three-region, end-of-life core, operated at 2,652 MWt with average burnup of 2.30×10^4 MWD/MTU. Additional

release category parameters used for the analysis are presented in Table 7.1-3.

Potential radiological consequences have been calculated for the nine release categories using the CRAC2 computer code. This code incorporates the models developed in the RSS. The analyses are site specific and employ environmental parameters associated with the BVPS-2 location. Meteorological data used in the analyses represent one full year of hourly averages of wind speed and direction, atmospheric lapse rate, and precipitation. Population data input is based on 1980 census data for the area within a 350-mile radius of the plant. Habitable land fraction out to 350 miles is also a site-specific input. In addition, the analyses require land use statistics as a function of state, including land value, farm product values, and growing season information for the area out to 350 miles. A diagram of the consequence analysis model is given on Figure 7.1-1.

In order to calculate consequences and their associated probabilities, weather data for 1 year are sampled. It would be prohibitively expensive to model the radioactivity release starting at each of the 8,760 possible hourly start times. Instead, a provision known as "bin sampling" is used. In this method, all of the hourly meteorological data are first classified into groups or bins (there are 29 bins defined in CRAC2) on the basis of similarity of weather sequences. Samples are selected from each bin. This ensures against exclusion or inappropriate weighting of some weather sequences.

Protective actions modeled which would reduce radiation doses include evacuation of nearby population, sheltering of non-evacuees, long-term relocation of people, and interdiction and decontamination of contaminated land. The evacuation plan used (refer to Appendix 7A) is site specific and includes two possible evacuation schemes. The evacuation parameters used in the model were extracted from a recent evacuation and mass notification study performed for BVPS (Alan M. Voorhees and Associates 1980).

7.1.3.2.1.6 Calculation of Risk Curves

This section describes the analytical techniques used to compile and assemble the results into risk diagrams. The assembly process is based on a matrix multiplication process. Matrices were constructed for the following separate analyses: initiating events, plant states, release categories, and consequences. Multiplication of the first three of these matrices provides the probability of occurrence associated with each release category, as shown in Table 7.1-2. These probabilities were then multiplied by the consequence matrix provided by the CRAC2 runs done on a conditional basis (release category probability equal to 1). This operation provides the additional probability associated with the weather conditions and evacuation scenarios to yield the total probability and consequences that make up the risk curves.

7.1.3.2.2 Dose and Health Impacts of Atmospheric Releases

The results of the calculations for dose and health impacts are presented in the form of probability distributions functions, sometimes referred to as CCDFs (complementary cumulative distribution functions), or final risk curves. All of the release categories shown in Table 7.1-2 are included in each risk curve.

Figures 7.1-2, 7.1-3, and 7.1-4 represent the probability distribution functions for the damage indices of early fatalities, total latent effects, and population whole body man-Rem, respectively. Early fatalities refer to those deaths which might result within 1 year of a postulated accident. The total latent effects include fatalities from leukemia, cancers of the lung, gastrointestinal tract, breast, and bone, but do not include thyroid cancers. The population whole body man-Rem is the sum of the number of people exposed times the whole body dose received for each dose level.

The final risk curves represent the product of the analysis of initiating events, plant response, containment response, and site consequences for severe accident sequences at BVPS-2. Incorporated into each of these distributions is the probability of a plant state resulting in a release to the environment described by any of the nine release categories, for each year of reactor operation.

The left side of Figure 7.1-2 shows that there is approximately one chance in 10 million per reactor year that one or more fatalities might occur as a result of a severe accident at BVPS-2. This frequency of exceeding a given number of fatalities decreases as the damage index (number of fatalities) increases. Additionally, the risk curve shows that the probability of exceeding one early fatality and the probability of exceeding approximately 200 early fatalities are about the same.

Figure 7.1-3 shows the probability distribution for fatal cancers (excluding data groups "thyroid" and "whole body"). Calculation of these effects is based upon the dose response model presented in the BEIR report (National Academy of Sciences 1972) and is consistent with models developed in BEIR III (National Academy of Sciences 1980). There is typically a latent period after exposure of from 2 to 30 years before a cancer might be manifested. In addition, individuals are considered to be at risk of leukemia induction for 30 years after the latent period following exposure, while for all other cancers they are assumed to be at risk for the remainder of their lives. As a result, the predicted number of fatal cancers may be distributed over many years. The dose used in predicting the number of latent effects is a result of both acute and chronic exposure.

Figure 7.1-4 provides the probability distribution for total population whole body dose in man-Rem. Acute and chronic whole body

doses to the population within 350 miles of the plant were used to generate this risk curve.

As an indication of the spatial distribution of individual latent cancer fatality risk per year of reactor operation, contours of constant risk (isorisk curves) are plotted on a map of the site out to 10 miles (Figure 7.1-5).

7.1.3.2.3 Economic Impacts

Placing monetary values on the cost of human radiation exposure is at best a difficult and controversial task. However, it is possible to assess the economic impacts of avoidance of adverse health effects. Such a calculation has been performed for BVPS-2. The results are shown on Figure 7.1-6, which is the probability distribution function for total cost of offsite mitigating actions including decontamination efforts. Factors which contribute to these estimated costs include the following:

- Evacuation costs,
- Value of milk contaminated and condemned,
- Costs of decontamination of property where practical, and
- Indirect costs due to loss of property use and incomes derived therefrom.

The loss of land use and associated incomes derives from the possible need to interdict the land and property until it is free of contamination or can be economically decontaminated.

Several comments can be made concerning economic risks based upon Figure 7.1-6. The initial plateau portion of the curve indicates that the probability of total costs exceeding 1 million dollars and the probability of total costs exceeding 20 million dollars are the same. This is a direct result of calculating evacuation costs based upon evacuation of the entire 10-radial-mile area of the plant. The right side of the risk curve indicates that, although very unlikely (less than one chance in one hundred million), costs could exceed several billion dollars (1980 dollars). These costs do not include the cost of decontamination and repair or replacement of the facility, nor the cost of replacement power.

7.1.3.2.4 Releases to Ground Water

7.1.3.2.4.1 Introduction

Releases of radioactivity to the ground-water system could occur following a postulated core meltdown with eventual penetration of the containment basemat. Core debris which exits the melt hole would then enter the water table, which is normally at an elevation of

approximately 20 feet below the containment basemat, and radionuclides in the debris would be leached in the ground-water system. It is also possible for containment sump water, which could be rich in dissolved fission products, to be released through a breach in the containment.

An analysis of the potential consequences of such an event is presented by the USNRC staff in NUREG-0440, "Liquid Pathway Generic Study" or LPGS (USNRC 1978a). This generic report provides the basis for the comparative evaluation of BVPS-2.

The LPGS presents analyses for a four-loop Westinghouse pressurized water reactor (PWR) located at a number of land sites, one of which is on a river system which is very similar to the one at BVPS-2. The LPGS river site is on the Clinch River approximately 21 miles upstream of its juncture with the Tennessee River. The balance of this river system consists of the Tennessee River (567 miles), the Ohio River (46 miles) and the Mississippi River (954 miles). This system contains a number of dams and reservoirs, principally along the Tennessee River segment.

The BVPS-2 river system is similar, consisting of 946 miles of the Ohio River and the same 954 miles of the lower Mississippi River. Like the Tennessee, there are numerous dams along the Ohio. However, unlike the Tennessee, which is an important recreational resource, the Ohio is heavily industrialized.

In the LPGS, parameters for each generic site (including the Clinch River site) were chosen to be representative of the full spectrum of similar sites, in this case river sites. Parameters used for analysis in the LPGS, although typical, do not represent any actual plant site. In the LPGS it was concluded that individual and population doses for the liquid pathways would be small fractions of the airborne pathways dose which could result from a core meltdown accident.

Individual and population doses are reported in the LPGS for the principal liquid pathways: drinking water, aquatic food, and direct exposure from swimming and shoreline usage. Exposure resulting from crop irrigation was also considered but was found to contribute insignificantly to dose (USNRC 1978a).

Doses to individuals and populations were calculated in the LPGS without taking credit for possible interdiction methods such as isolation of contaminated ground water, the temporary restriction of fishing, or the provision of alternate sources of drinking water (or additional purification equipment). Such interdiction methods would be highly successful in preventing exposure to radioactivity, and the liquid pathways consequences would therefore be economic and societal rather than radiological.

7.1.3.2.4.2 Method of Comparison

The estimate of the liquid pathways consequences resulting from a radionuclide release at BVPS-2 is developed by using ratios to compare the principal parameters applicable to the BVPS site to the parameter values used for the generic river site calculations in the LPGS.

Ratio comparisons are developed using the following:

- The radionuclide source released to the river,
- The population along the river system which obtains drinking water from the river,
- The annual fish harvest on the river system, and
- The annual recreational usage of the river system.

In a very general way, the consequences of a major radionuclide release to the ground-water system at BVPS-2 can be expressed as follows:

$$\text{BVPS Dose} = \frac{\text{BVPS Source}}{\text{LPGS Source}} \sum \begin{matrix}\text{LPGS Dose for} \\ \text{the ith pathway}\end{matrix} \times \begin{matrix}\text{Usage ratio for} \\ \text{the ith pathway}\end{matrix}$$

Pathways "usage" ratios are the following:

1. $\dfrac{\text{Drinking water population for BVPS river system}}{\text{Drinking water population for LPGS river system}}$

2. $\dfrac{\text{Annual fish harvest for BVPS river system}}{\text{Annual fish harvest for LPGS river system}}$

3. $\dfrac{\text{Man-hours direct exposure for BVPS river system}}{\text{Man-hours direct exposure for LPGS river system}}$

To be strictly correct, this summation should be carried out for each radionuclide. However, it has been found that the liquid pathway doses tend to be dominated by a very few radionuclides. As will be shown in a subsequent section, the characteristics of the BVPS-2 site are such that most of the important radionuclides will undergo substantial decay during the process of ground-water transport to the Ohio River. Therefore, the preceding general equation provides an adequate approach to developing a comparative liquid pathways dose evaluation.

7.1.3.2.4.3 Beaver Valley Power Station Site Description

Topography

Beaver Valley Power Station - Unit 2 is located on the left descending bank (the south bank) of the Ohio River at river mile 34.7. At this location, the normal river level is 664.5 feet (msl).

In the general site region, the south bank of the river is characterized by bedrock bluffs which rise abruptly from the water's edge to elevations as much as several hundred feet above the river. Beginning approximately 12,000 feet upstream of BVPS-2, the location of the bedrock bluff begins to deviate inland from the river bank. At its maximum point, the bluff is about 2,500 feet inland (south) of the river bank. At a point approximately 2,000 feet downstream of BVPS-2, the bedrock bluff again joins the river's edge. Thus a saddle-like bedrock formation exists at the BVPS site. The bottom of the saddle is bedrock at an approximate elevation of 620 feet msl. The back and sides of the saddle are formed by upswings in the bedrock surface which reach elevations in excess of 800 feet msl. The bedrock saddle is covered with alluvial deposits in the form of three terraces. Typically, the upper terrace elevation is approximately 750-760 feet msl. These alluvial soils consist of glacial outwashes overlain by thin deposits of silt, sand, and clay. These topological site features are illustrated on FSAR Figures 2.1-2 and 2.5.4-1 and are described in FSAR Section 2.4. The saddle is the location of the Bruce Mansfield fossil plant, BVPS-1, BVPS-2, and the Shippingport Atomic Power Station.

Local Ground-Water System

The alluvial soils make up the only significant aquifer in the site vicinity. Ground-water flow within this local system is normally directed toward the Ohio River. Both upstream and downstream of the site, the soils are pinched out against the steep bedrock valley wall. To the northeast of the BVPS-2 location, ground-water flow is further impeded by a buried structure which is thought to be a bedrock bench. This buried structure runs perpendicular to the river bank at a location approximately 2,500 feet upstream of BVPS-2.

Some ground-water is present in the upland surfaces which lie behind the plant (to the south) and at substantially higher elevations. Ground-water migration in the underlying bedrock takes place along connected joint systems and occasional sandstone leases. Because of the low permeability of the bedrock, recharge from the rock to the alluvial soils at the site is negligible. Precipitation is therefore the principal source of ground-water recharge and amounts to approximately 900 gallons per day per acre. At this level of recharge, the ground-water level under the plant is approximately 1 foot above the adjacent river level.

Onsite wells consist of two temporary construction wells and two wells which are located adjacent to the BVPS emergency response facility. There are less than 40 domestic and farm wells within the region formed by the bedrock saddle described earlier. However, almost all of these wells are either isolated from the site ground-water system by the buried bedrock bench (to the north of the site), or the wells terminate at elevations significantly above that of the plant. Thus the principal consequence of a release of radionuclides to the site ground-water system would be their transport to the Ohio River. The local ground-water system is shown in FSAR Figure 2.4-17 and is described in FSAR Section 2.4.13.

Ground-Water Transport Time

Radionuclides which enter the ground-water system would be entrained in the natural ground-water flow to the Ohio River. The calculation of the ground-water seepage velocity is discussed in FSAR Section 2.4.13.3.1. The ground-water transport time from the containment location to the river's edge is the travel distance (770 feet) divided by the seepage velocity (30.3 ft/year), or 25 years.

7.1.3.2.4.4 Source Comparison

The radionuclide source which is ultimately transmitted through a ground-water system to an adjacent surface water is determined by the following three factors:

- The core radionuclide inventory,
- The fraction of the core radionuclide inventory released to ground-water via such mechanisms as sump water release and leaching from the core debris, and
- The attenuation which takes place during transport through the ground-water system, principally from radioactive decay and adsorption.

The LPGS analyses are based on the core inventory for a four-loop Westinghouse PWR. Since BVPS-2 is a three-loop Westinghouse PWR, the core inventory would be somewhat less. The difference was not quantified for the liquid pathway evaluation, and it is therefore assumed for convenience that the core source is the same as in the LPGS.

The fraction of the core inventory which could be released to the ground water depends on numerous factors, such as the specific accident sequence and containment failure mode, the containment sump structure, and the nature of the soils which separate the containment basemat from the underlying ground-water system. Again, for convenience, it is assumed that the LPGS assumptions apply to BVPS-2. A number of release cases are considered in the LPGS; however, the

worst cases considered (instantaneous release of all sump water and of all activity available for leaching) are clearly bounding for any plant-site combination.

The fraction of each radionuclide source released into the ground water which eventually enters the Ohio River depends on the velocity of ground-water movement as well as retardation of radionuclide travel caused by adsorption on the aquifer soils.

The relationship between ground-water velocity (equivalent ground-water transport time for a given distance of travel), radionuclide adsorption, and the radionuclide fraction which is ultimately transmitted without decay is given (Sandia National Laboratories 1981; USNRC 1978a) by the expression:

$$\text{Log}_{10}\ (\text{Transmitted Fraction}) = \frac{-0.693(\text{GWTT})(a)}{2.3\ T}$$

where:

GWTT = ground-water transport time

a = adsorption retention factor

T = radionuclide half life

The adsorption retention factor is equal to $(1 + \frac{\rho}{\eta}\text{Kd})$, where:

ρ = bulk density of the aquifer media

η = porosity of the aquifer

Kd = distribution coefficient which is defined as the mass of radionuclide adsorbed per gram of soil divided by the mass of radionuclide dissolved per milliliter of ground water.

A typical value of the ratio $\frac{\rho}{\eta}$ is 5; however, for consistency, the value of 4.1 used in the LPGS is adopted here as well (Sandia National Laboratories 1981; USNRC 1978a).

The equation for the transmitted fraction can therefore be written as either:

$$\text{Log}_{10}\ \text{T.F.} = \frac{-0.693(\text{GWTT})(a)}{2.3T},\ \text{or}\ \text{Log}_{10}\ \text{T.F.} = \frac{-0.693(\text{GWTT})(1+4.1\text{Kd})}{2.3T}$$

The use of these two forms is convenient because in some cases values of "a" are tabulated, while in other cases values for "Kd" are given.

Table 6.2.1 of the LPGS lists the transmitted fraction for a number of radionuclides, the more important of which are reproduced as follows:

Nuclide	$T_{1/2}$ (yrs)	T.F.
H-3	12.1	0.97
Sr-90	28	0.87
Ru-106	1	0.33
Cs-137	30	0.31

These values are based on the following data assumed in the LPGS for the generic river site (USNRC 1978a):

$$GWTT_{LPGS} = \frac{1500\ \text{ft}}{6.7\ \text{ft/day}} = 224\ \text{days} = 0.61\ \text{year}$$

Nuclide	Absorption Retention Factor	Equivalent Kd
H-3	1	0
Sr-90	9.2	2
Ru-106	1	0
Cs-137	83	20

The equivalent values of Kd used in the LPGS are low in comparison to other estimates for Sr-90 and Cs-137. In the Sandia liquid pathway study, Kd values of 20 and 200 were used for Sr-90 and Cs-137, respectively (Sandia National Laboratories 1981). Duke Power Company estimated Kd values of 6 (Sr-90) and 560 (Cs-137) for the fractured bedrock underlying its Catawba Nuclear Station (USNRC 1983). Kd values of 5 (Sr-90) and 50 (Cs-137) were estimated to represent the complex ground-water hydrology at the Seabrook Station site. At Seabrook, ground water exists both in bedrock and in surface soils (USNRC 1982). Values of Kd for the granular materials underlying the San Onofre Nuclear Station were estimated at 31 (Sr-90) and 2204 (Cs-137) (USNRC 1981a). Regardless of this evidence for larger values and because no specific Kd estimates were available for the BVPS-2 site, the values used in the LPGS are adopted.

The ground-water transport time at BVPS-2 is estimated to be 25 years. On the basis of this and the Kd (or "a") values used in the LPGS, the transmitted fractions for the principal radionuclides are as follows:

Nuclide	$T_{1/2}$(yrs)	Log_{10}(T.F.)	T.F.	T.F.(BVPS)/T.F.(LPGS)
H-3	12.1	-0.62	0.24	0.25
Sr-90	28	-2.5	0.003	0.003
Ru-106	1	-7.5	0	0
Cs-137	30	-20.8	0	0

The effect of the much longer ground-water transport time at BVPS (25 years compared to 0.61 years in the LPGS), even with the relatively small assumed values of Kd, is very significant. Virtually no Cs-137 or Ru-106 would be expected to reach the Ohio River. Approximately 3/1000 of the released Sr-90 would reach the river (compared to a transmitted fraction of 0.87 in the LPGS). The tritium value is closer to LPGS results with a transmitted fraction of 0.24 for BVPS compared with 0.97.

The source effect on liquid pathway consequences can be summarized as follows:

1. Pathways doses which are dominated by Cs-137 and/or Ru-106 will be negligible in comparison with doses calculated in the LPGS.

2. Pathways doses which are dominated by Sr-90 will be about 3 orders of magnitude lower than those calculated in the LPGS, assuming equal pathways exposure.

3. Pathways doses from H-3 will be lower but within the same order of magnitude, assuming equal pathways exposure. At the levels of population dose calculated in both NUREG-0440 (USNRC 1978a) and in the Sandia study (Sandia National Laboratories 1981), tritium is not a significant contributor. This is due in part to the smaller core inventory of tritium (two to three orders of magnitude less curie content than Sr-90, Cs-137, or Ru-106) (USNRC 1978a) and also in part to the relatively low total body dose factor for tritium of 1×10^2 man-Rem/curie compared with 1.9×10^6 man-Rem/curie for Sr-90 and 8×10^4 man-Rem/curie for Cs-137 (USNRC 1978a).

The BVPS site was estimated to have about five times more downstream drinking water population than that utilized in the LPGS. Since the drinking water pathway dose is dominated by Sr-90 and Cs-137 (Sandia National Laboratories 1981) and since the fractions of these isotopes transmitted to the Ohio River is much smaller than the fractions estimated in the LPGS, the effect of the larger drinking water population is more than offset by the reduced quantity of radioactivity which ultimately enters the river. The drinking water pathway dose for BVPS would therefore be expected to be substantially

(perhaps orders of magnitude) smaller than that calculated in the LPGS.

The fish harvest downstream of the BVPS site is estimated to be more than a factor of 3.8 larger than that estimated in the LPGS. Since this pathway is also dominated by Sr-90 and Cs-137 (Sandia National Laboratories 1981), population exposure via this pathway will also be substantially less than that calculated in the LPGS.

The shoreline and immersion pathway includes such activities as swimming, wading, and sunbathing. These are external exposure pathways, and dosage is dominated by Ru-106, Cs-137, and Co-60. For the BVPS site having a ground-water travel time of 25 years and assuming that Kd for Co-60 is 75, the transmitted fraction is vanishingly small, as has already been shown to be the case for Cs-137 and Ru-106. It is therefore concluded that the direct exposure dose would be negligible in comparison with those calculated in the LPGS.

7.1.3.2.4.5 Liquid Pathway Comparison Results

On the basis of BVPS site features and the specific comparisons of both radionuclide source and pathway populations, it is apparent that the spectrum of liquid pathways doses following a severe accident would be much lower for BVPS-2 than the doses calculated in the LPGS for a river-sited plant.

This is mainly due to the much smaller source released to the Ohio River which in turn results mainly from a much longer ground-water transport time. If shorter times are postulated, the adverse effect would be small and would probably be offset through the assumption of more realistic distribution coefficient (Kd) values.

In the extreme, if the same radionuclide source as in the LPGS were postulated, the pathways doses would still be within the same order of magnitude, since the pathways population ratios are not large.

7.1.3.3 Risk Consideration

The previous material has dealt with the likelihood of occurrence (frequency) of accidents as well as their impacts (consequences). Because it is difficult to compare such results with other types of risk in our environment, it is useful to combine these results into average risk values.

A common way in which this combination of factors is used to estimate risk is to multiply the probabilities by the consequences. The resultant risk is then expressed as a number of consequences expected per unit of time. Such a quantification of risk does not at all mean that there is universal agreement that peoples' attitudes about risks, or what constitutes an acceptable risk, can or should be

covered solely by such a measure. At best, it can be a contributing factor to a risk judgment, but not necessarily a decisive factor.

To accomplish these comparisons, the mean values for a damage index (for example, early fatalities) as calculated by CRAC2 for each release category were weighted by the corresponding release category frequency, and then summed. The resulting value is representative of the risk for the particular damage index per year of reactor operation. The results of these calculations are presented in Table 7.1-4 for early fatalities, latent cancer fatalities, population whole body man-Rem, and the total cost with decontamination.

The mean risk of early fatality per year of BVPS-2 operation for all release categories is approximately 0.0011, for evacuation to 10 miles. The population at risk for early fatalities is within a 20-mile radius of the plant. The 1980 population within 20 miles of the plant, shown in FSAR Table 2.1-10, is 461,393. Accidental fatalities per year for a population of this size, based upon overall averages for the United States, are approximately 102 from motor vehicle accidents, 36 from falls, 14 from drowning, 13 from burns, and 6 from firearms (National Research Council 1979). The risk of early fatalities due to the operation of BVPS-2 is therefore very small in relation to other fatal risks to which the population is exposed.

For comparison with the population whole body dose mean value of 256 man-Rem, also shown in Table 7.1-4, the whole body dose to the population within 350 miles of BVPS-2 due to natural radiation from the environment can be calculated. In the northeastern United States, at the altitude of BVPS-2, the average annual dose to an individual due to natural radiation is approximately 103 mRem (National Academy of Sciences 1980). If the population at risk around BVPS-2 is considered to be those people within 350 miles, or a population of 79,118,950 as shown in FSAR Table 2.1-17, the population dose from natural radiation is 8,150,000 man-Rem. Based upon population whole body dose, the risk due to severe accidents resulting from operation of BVPS-2 is very small compared to the risk due to natural radiation.

The average number of latent cancer fatalities for the lifetime of the population within 350 miles of the reactor, due to one year of reactor operation, is approximately 1.7×10^{-2}. In comparison, assuming a representative United States population, the expected number of deaths per year due to cancer would be approximately 162,700 (U.S. Bureau of Census 1982). Although this is comparing a lifetime risk to a yearly risk, it is evident that the risk of cancer fatality due to severe accidents is very small in relation to existing cancer fatality risks.

7.1.3.4 Uncertainties

The methodology employed to provide this present risk assessment for BVPS-2 is essentially that of the Reactor Safety Study. There are substantial uncertainties associated with the numerical estimates of the likelihood, as well as the consequences, of reactor accidents that are evaluated using this methodology.

One of the chief areas of uncertainty is the magnitude of the source term, or fission product release to the atmosphere. This uncertainty arises from neglect of attenuation of fission products during transit through the primary coolant system following a postulated accident. Present analyses assume that fission products released from the core exit directly to the containment. Another uncertainty is the chemical form of the fission products released. WASH-1400 considered elemental iodine as the species released while recent investigations (USNRC 1981b) indicate iodine may be released as a less volatile cesium iodine. A thorough discussion of the uncertainties in the source term is provided in a British study, SRD 256 (United Kingdom Atomic Energy Authority 1982). Reduction of the source term as postulated by these documents would result in substantially lower estimates of detrimental health effects.

Additional uncertainties arise in the dose response model, evacuation model, and meteorological model employed by the CRAC2 code. In this present analysis, conservative reductions in evacuation speed based upon estimates of the relative probability of bad weather have been made. Although the attempt has been made in probabilistic risk assessment studies to provide realistic state-of-the-art analysis of consequences, there remain many areas of conservatism.

7.1.4 References for Section 7.1

Alan M. Voorhees and Associates 1980. Study Report for the Beaver Valley Power Station, Evacuation and Mass Notification.

Bell, M.J. 1973. ORIGEN - The ORNL Isotope Generation and Depletion Code.

Commonwealth Edison Company 1981. Zion Probabilistic Safety Study.

Electric Power Research Institute (EPRI) 1982. ATWS: A Reappraisal, Part III. Frequency of Anticipated Transients. EPRI-NP-2230.

National Academy of Sciences 1972. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation.

National Academy of Sciences 1980. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation.

National Research Council 1979. Energy in Transition 1985-2010, Final Report of the Committee on Nuclear and Alternative Energy Systems (CONAES), Chapter 9.

Rose, P.S. and Burrows, T.W. 1976. ENDF/B-IV Fission Product Decay Data.

Sandia National Laboratories 1981. The Consequences from Liquid Pathways After a Reactor Meltdown Accident. NUREG/CR-1596.

United Kingdom Atomic Energy Authority 1982. The Technical Basis of Spectral Source Terms for Assessing Uncertainties in Fission Product Release During Accidents in PWR's with Special Reference to Sizewall B. SRD 256.

U.S. Bureau of Census 1982. Statistical Abstract of the United States: 1982-83, 103d edition. Washington, D.C.

U.S. Nuclear Regulatory Commission (USNRC) 1975. Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plant. NUREG-75/014 (WASH-1400).

USNRC 1976. Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code), NUREG-0017, Office of Standards Development.

USNRC 1978a. Liquid Pathway Generic Study. NUREG-0440.

USNRC 1978b. Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission. NUREG/CR-0400.

USNRC 1980. Clarification of TMI Action Plan Requirements. NUREG-0737.

USNRC 1981a. Final Environmental Statement Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3. NUREG-0490.

USNRC 1981b. Technical Basis for Estimating Fission Product Behavior During LWR Accidents. NUREG-0772.

USNRC 1982. Draft Environmental Statement Related to the Operation of Seabrook Station, Units 1 and 2. NUREG-0895.

USNRC 1983. Final Environmental Statement Related to the Operation of Catawba Nuclear Station, Units 1 and 2. NUREG-0921.

Westinghouse Electric Corporation 1982. Sizewell B Probabilistic Safety Study. WCAP-9991, Revision 1.

TABLE 7.1-1

INTEGRATED RADIATION DOSES TO INDIVIDUALS FROM VARIOUS POSTULATED ACCIDENTS

Postulated Accidents	0-2 hr Exclusion Area Boundary Dose (Rem) Thyroid	Gamma	Beta
Infrequent Accidents			
1. Process gas charcoal bed adsorber system release	-	2.3×10^{-2}	1.7×10^{-2}
2. Steam generator tube rupture	6.9×10^{-3}	1.9×10^{-2}	1.3×10^{-2}
3. Fuel assembly drop in the fuel storage pool	1.0×10^{-3}	5.5×10^{-3}	9.1×10^{-3}
4. Heavy object drop onto fuel storage rack	2.2×10^{-3}	4.2×10^{-3}	9.2×10^{-3}
5. Loss of coolant through a small pipe break	2.3×10^{-4}	4.9×10^{-6}	3.1×10^{-6}
6. Small steam line break outside containment structure	1.3×10^{-4}	2.7×10^{-6}	1.8×10^{-6}
Limiting Faults			
1. Loss of coolant through a large pipe break	1.8	6.5×10^{-2}	3.1×10^{-2}
2. Rod ejection accident	1.8×10^{-1}	6.5×10^{-3}	3.1×10^{-3}
3. Large steam line break outside containment structure	2.2×10^{-4}	2.9×10^{-6}	1.8×10^{-6}

TABLE 7.1-2

RELEASE CATEGORY SUMMARY

Category	Fraction of Core Inventory Released								Probability Per Reactor Year
	Xe-Kr	OI*	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La	
BV-1	0.9	7×10^{-3}	7×10^{-2}	5×10^{-2}	3×10^{-2}	6×10^{-3}	2×10^{-3}	4×10^{-4}	4.02×10^{-7}
BV-2	0.7	5×10^{-3}	0.5	0.6	0.2	7×10^{-2}	2×10^{-2}	3×10^{-3}	1.96×10^{-6}
BV-3	0.8	5×10^{-3}	0.5	0.6	0.2	8×10^{-2}	3×10^{-2}	3×10^{-3}	2.72×10^{-7}
BV-4	0.9	6×10^{-3}	1×10^{-2}	0.5	0.5	5×10^{-2}	4×10^{-2}	6×10^{-3}	1.13×10^{-6}
BV-5	0.9	6×10^{-3}	1×10^{-2}	0.5	0.5	5×10^{-2}	4×10^{-2}	7×10^{-3}	1.47×10^{-7}
BV-6	0.9	6×10^{-3}	9×10^{-3}	0.3	0.3	3×10^{-2}	2×10^{-2}	4×10^{-2}	8.51×10^{-6}
BV-7	0.9	7×10^{-3}	8×10^{-3}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	1.61×10^{-5}
BV-8	0.9	6×10^{-3}	2×10^{-3}	2×10^{-6}	1×10^{-6}	2×10^{-7}	9×10^{-8}	1×10^{-8}	1.77×10^{-5}
BV-9	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	2.30×10^{-5}

NOTE:

*Organic Iodine.

TABLE 7.1-3

RELEASE CATEGORY PARAMETERS

Release Category	Time to Release* (hr)	Release Duration (hr)	Warning Time** (hr)	Sensible Heat Rate*** (cal/sec)
BV-1	2.5	1.0	1.0	1.5×10^7
BV-2	0.72	2.0	0.2	1.1×10^9
BV-3	6.0	2.0	0.5	1.4×10^8
BV-4	8.3	0.5	4.1	3.4×10^8
BV-5	4.3	0.5	4.1	3.3×10^8
BV-6	20.1	0.5	16.0	4.1×10^8
BV-7	4.5	0.5	4.0	1.5×10^7
BV-8	21.0	0.5	2.0	1.5×10^7
BV-9	95.0	10.0	80.0	0

NOTES:

*Time between reactor shutdown and release to atmosphere.

**Time from beginning of official warning to beginning of atmospheric release.

***Due to thermal heat content of the released gases.

TABLE 7.1-4

MEAN VALUES FOR DAMAGE INDICES

Release Category	Early Fatalities	Latent Cancer Fatalities	Population Whole Body Man-Rem	Total Cost with Decontamination (1980 dollars)
BV-1	0	2.12×10^{-4}	3.37	94.1
BV-2	6.8×10^{-7}	5.7×10^{-5}	0.89	41.8
BV-3	7.7×10^{-6}	7.9×10^{-4}	12.4	563.0
BV-4	4.49×10^{-5}	2.35×10^{-3}	36.2	1,830
BV-5	3.77×10^{-4}	3.09×10^{-4}	4.70	238
BV-6	7.01×10^{-4}	1.28×10^{-2}	197.4	7,710
BV-7	0	7.33×10^{-5}	1.2	388
BV-8	0	3.33×10^{-5}	0.5	412
BV-9	0	4.1×10^{-6}	8.0×10^{-2}	524
Total	1.13×10^{-3}	1.66×10^{-2}	256.0	11,800

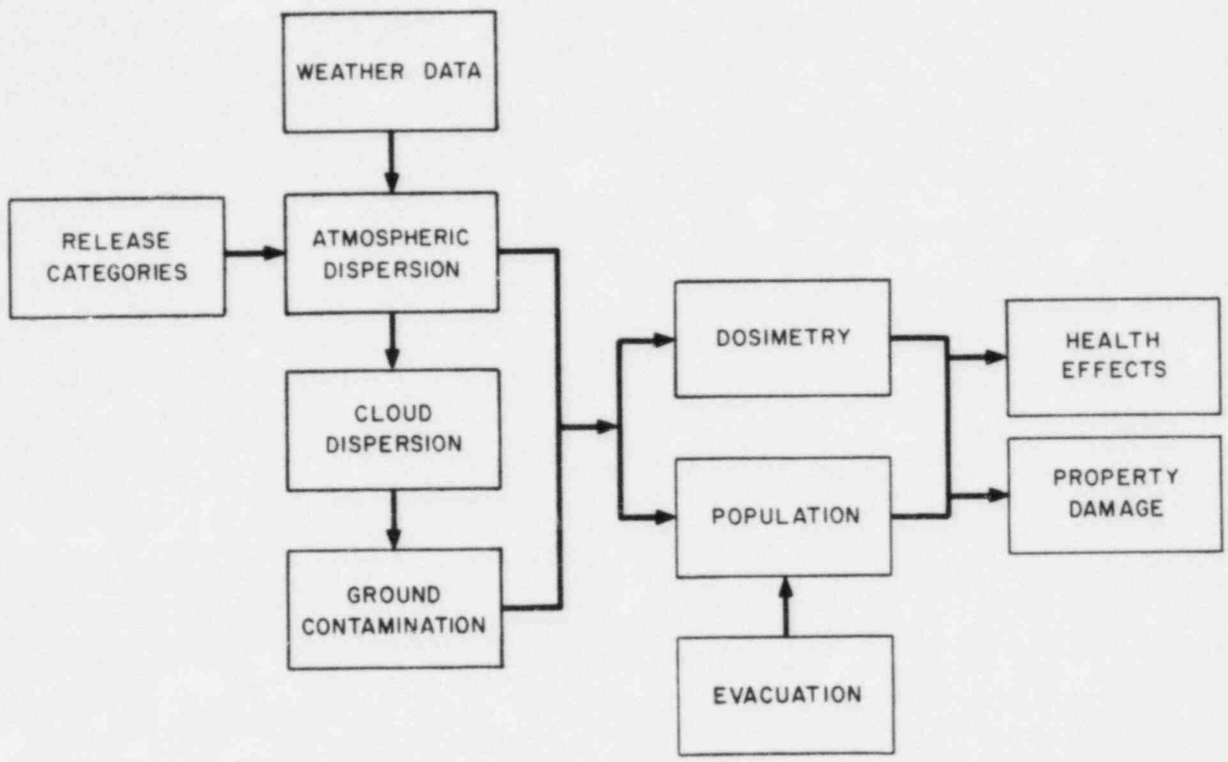


FIGURE 7.1-1

SCHEMATIC OUTLINE OF ATMOSPHERIC PATHWAY CONSEQUENCE MODEL

BEAVER VALLEY POWER STATION - UNIT 2
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

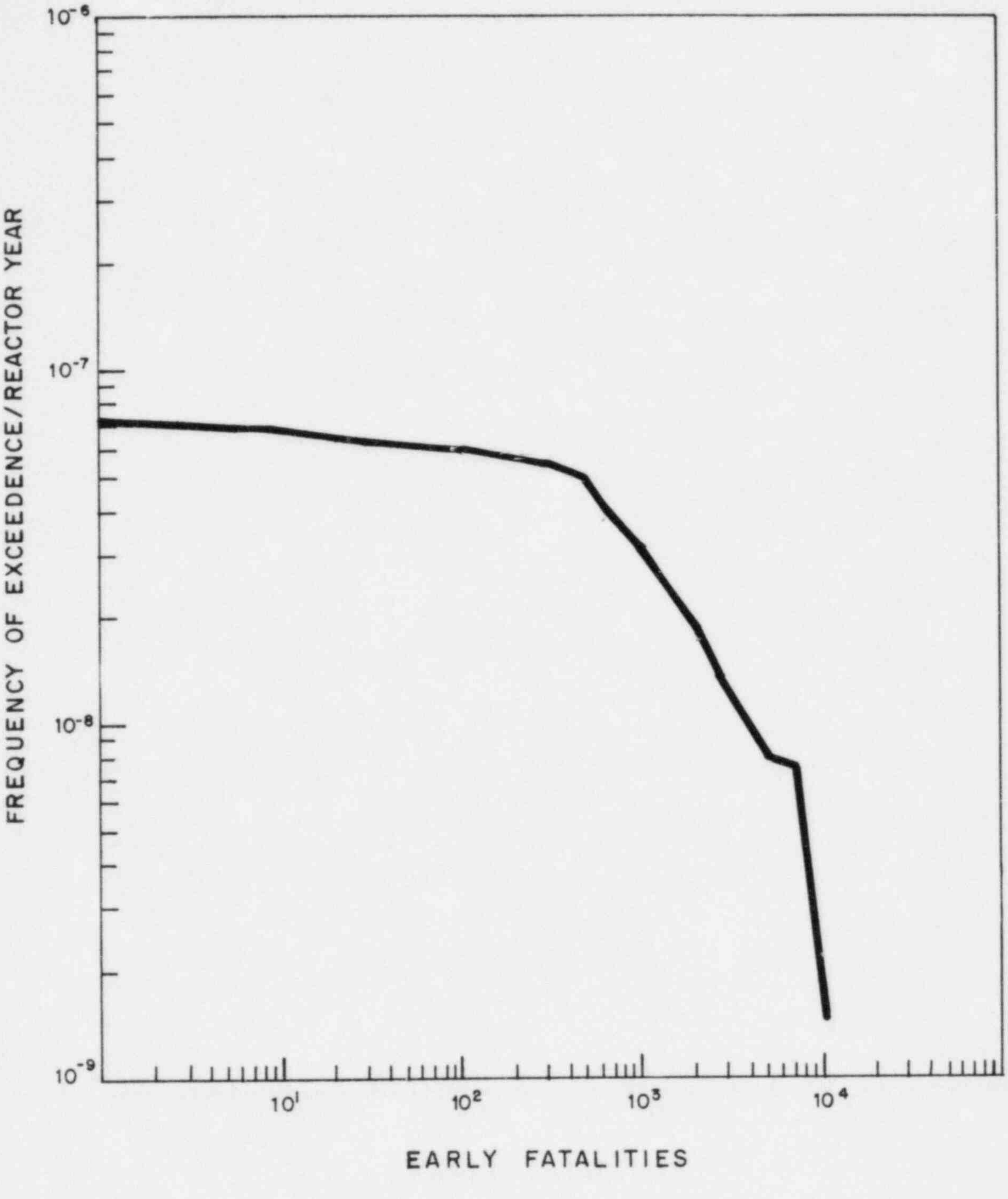


FIGURE 7.1-2

RISK CURVE FOR EARLY FATALITIES

BEAVER VALLEY POWER STATION-UNIT 2
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

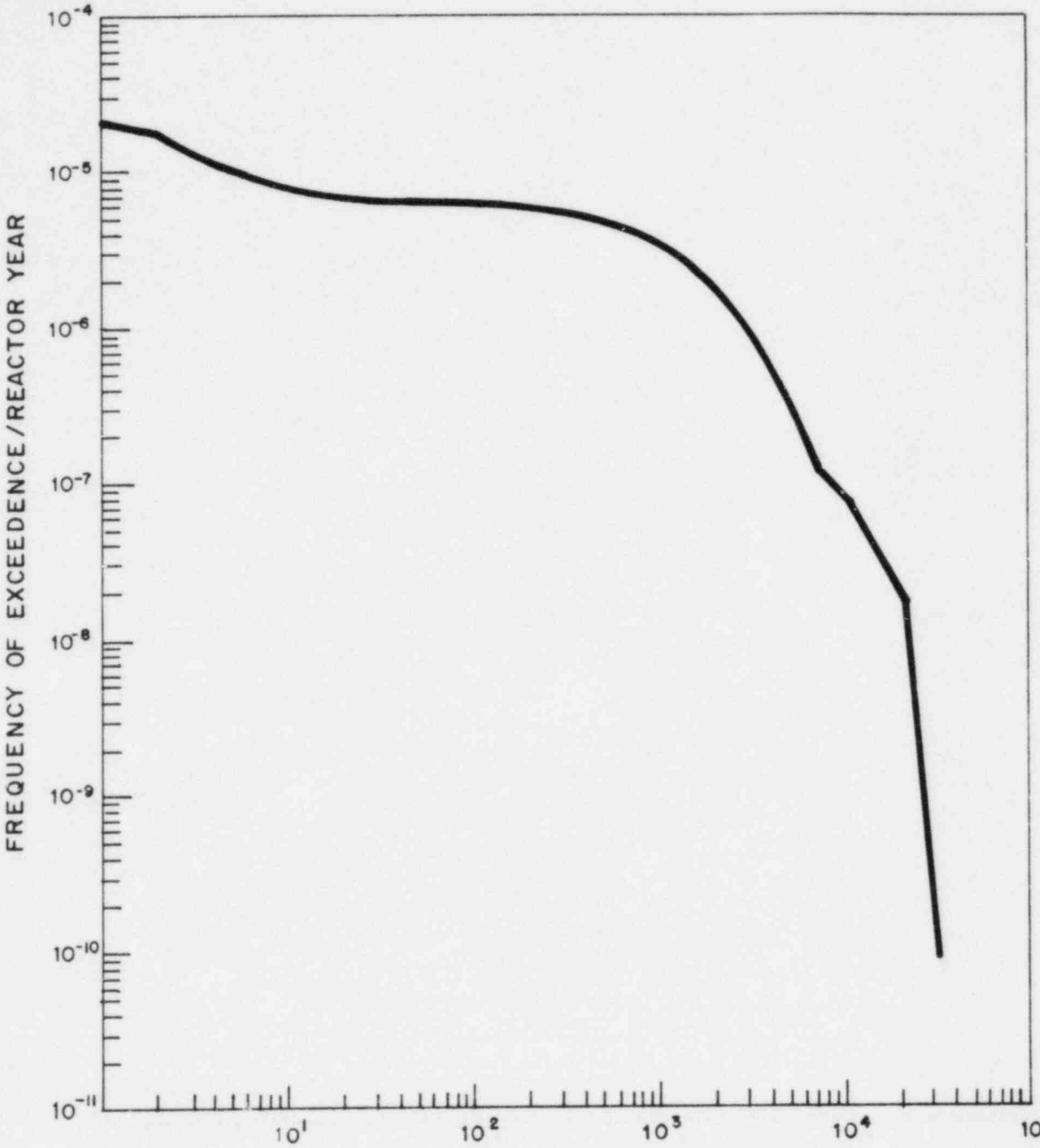


FIGURE 7.1-3
RISK CURVE FOR TOTAL
LATENT CANCER FATALITIES
BEAVER VALLEY POWER STATION-UNIT 2
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

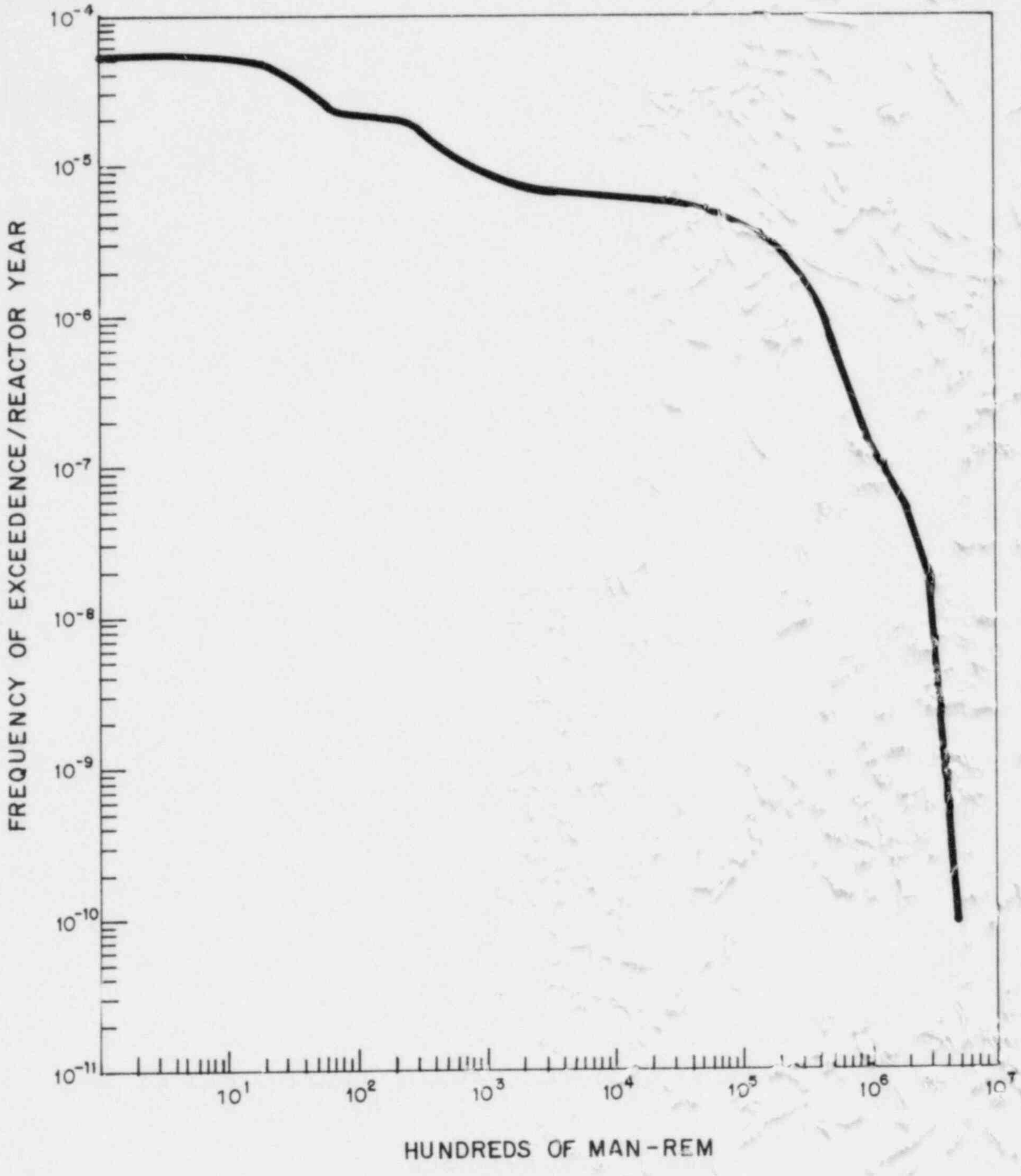
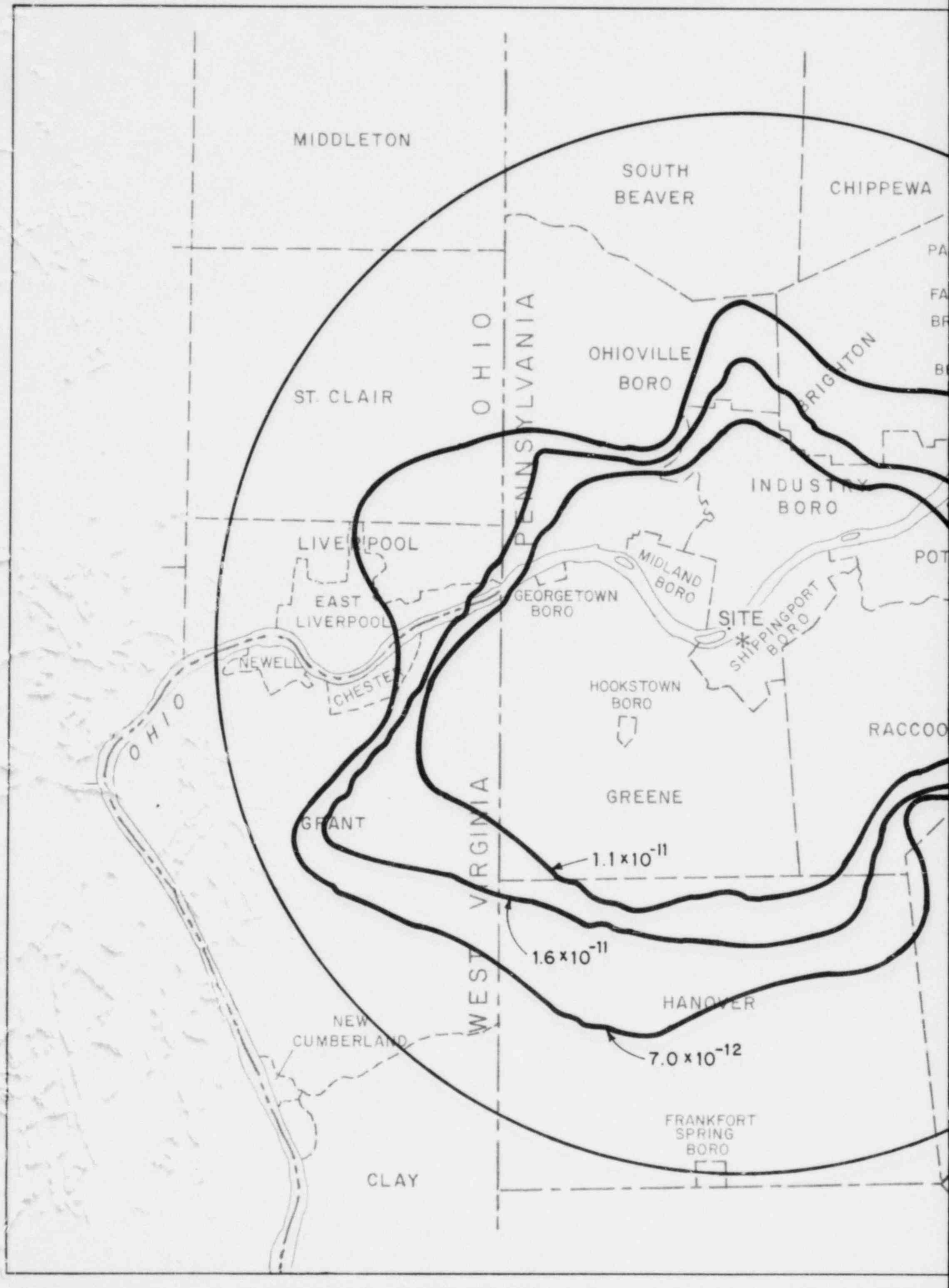


FIGURE 7.1-4
RISK CURVE FOR TOTAL
POPULATION WHOLE BODY MAN-REM
BEAVER VALLEY POWER STATION-UNIT 2
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE



MIDDLETON
SOUTH BEAVER
CHIPPEWA
OHIO
PENNSYLVANIA
OHIOVILLE BORO
BRIGHTON
ST. CLAIR
INDUSTRY BORO
LIVERPOOL
MIDLAND BORO
EAST LIVERPOOL
GEORGETOWN BORO
SITE
SHIPPINGPORT BORO
NEWELL
CHESTER
OHIO
HOOKSTOWN BORO
RACCOO
GREENE
GRANT
WEST VIRGINIA
1.1×10^-11
1.6×10^-11
HANOVER
NEW CUMBERLAND
7.0×10^-12
FRANKFORT SPRING BORO
CLAY

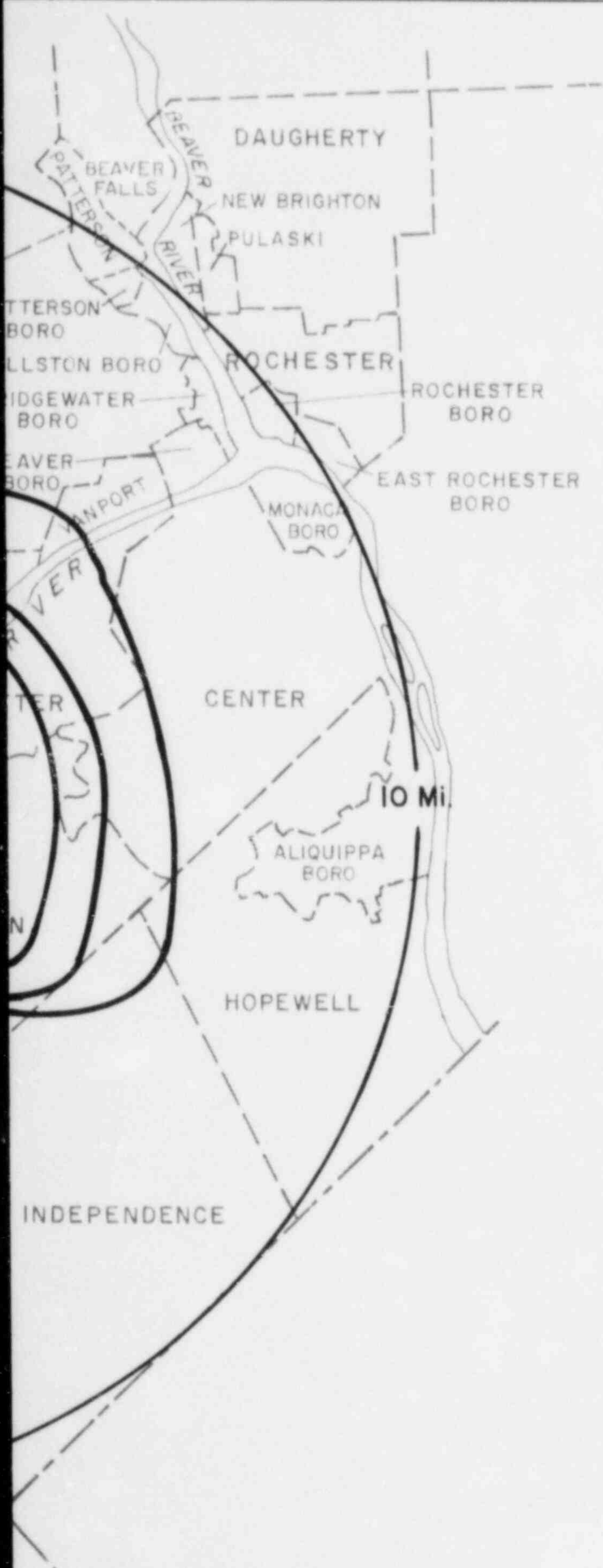


FIGURE 7.1-5
ISORISK CONTOURS OF LATENT CANCER FATALITY PER REACTOR-YEAR TO AN INDIVIDUAL
BEAVER VALLEY POWER STATION - UNIT 2
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

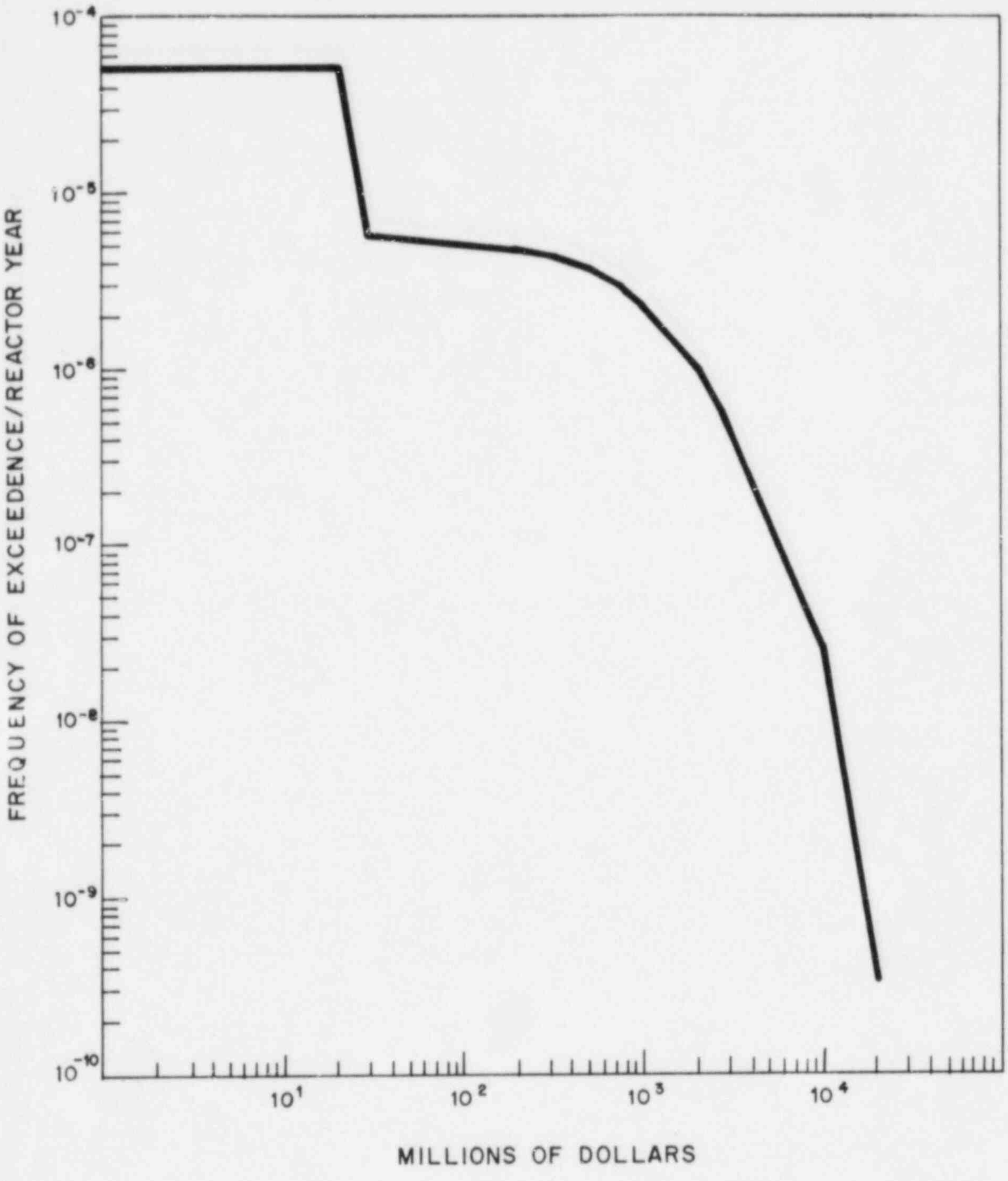


FIGURE 7.1-6
RISK CURVE FOR TOTAL COST WITH DECONTAMINATION
BEAVER VALLEY POWER STATION-UNIT 2
ENVIRONMENTAL REPORT
OPERATING LICENSE STAGE

7.3 OTHER ACCIDENTS

In addition to accidents that may release radioactivity to the surroundings, other accidents may occur on the Beaver Valley Power Station - Unit 2 (BVPS-2) site that, although they do not involve radioactive materials, could have consequences that affect the environment. Accidents such as chemical explosions, fires, and leakage or rupture of vessels containing oil or toxic materials could have significant environmental impacts. A discussion of chemical toxicity appears in FSAR Section 2.2.3.1.3.

Onsite liquid storage is shown in Table 7.3-1. Equipment, systems, and structural design minimize the probability of occurrence of any of these accidents, and confine any effects to the immediate BVPS-2 site, thus precluding the possibility of any significant impact to the surrounding environment. In addition, major liquid deliveries and spills will be handled in accordance with the Preparedness, Prevention, and Contingency (PPC) Plan as required by the Pennsylvania Department of Environmental Resources (DER).

An analysis of these accidents, with respect to onsite safety systems, is included in FSAR Section 2.2.3. The BVPS-2 protection systems, operating precautions, and procedures limit the effects of these postulated events. These features are discussed in the sections that follow. Nonradiological accidents were not discussed in the Environmental Report - Construction Permit Stage.

7.3.1 Fires

Three types of fire protection systems (FPSs), as described in FSAR Section 9.5.1, are located throughout the facility as appropriate: carbon dioxide, Halon 1301, and water. Each FPS conforms to the applicable requirements of the Occupational Safety and Health Administration (OSHA), the recommendations of the National Fire Protection Association, the American Nuclear Insurers, and the U.S. Nuclear Regulatory Commission. Since any BVPS-2 fire will be extinguished within the confines of the plant, the offsite effects of fire will be negligible.

7.3.2 Liquid Storage Tanks

7.3.2.1 Sodium Hydroxide

Sodium hydroxide serves a dual purpose at BVPS-2 and is stored at two locations convenient to the systems which use it.

Sodium hydroxide is added to the quench spray system to remove airborne radioactive iodine from the containment atmosphere. In order to prevent re-evolution of iodine from solution in the containment sump water, sufficient sodium hydroxide is added by the quench spray to maintain pH greater than 8.5 in the sump during the time when the recirculation sprays are activated.

The sodium hydroxide (23 to 25 percent by weight) supplied for this function is stored in a 10,000-gallon tank located outside the safeguards building, next to the refueling water storage tank. Major spills from this tank will be handled in accordance with the PPC plan as required by the Pennsylvania DER.

Sodium hydroxide is also used to maintain a pH level of 9 to 11 in the solid waste system storage tanks. This sodium hydroxide (50 percent by weight) is stored in a 1,500-gallon tank located inside the waste handling building. The area around the tank is diked to contain any leakage. Leakage is collected in the building sump, where it is transferred to and treated in the liquid waste system.

7.3.2.2 Hydrazine and Ammonia

During normal power operation, a 1-percent hydrazine solution and a 7-percent ammonia solution are continuously injected into the main condensate flow downstream of the condensate polishers. During wet layup, a 35-percent hydrazine solution and a 29.4-percent ammonia solution are injected immediately upstream of the steam generators. Hydrazine is used to control dissolved oxygen concentration during normal power operation and during wet layup. The chemical feed system maintains condensate pH and dissolved oxygen concentrations within the limits of the steam generator manufacturer's specifications.

There are two measuring tanks located in the turbine building: one for hydrazine and one for ammonia. Each has a capacity of 11 gallons.

There are five feed tanks, each with a 415-gallon capacity, located on the ground floor of the turbine building. These consist of two dilute (1-percent solution) hydrazine feed tanks (used during normal operation), one concentrated (35-percent solution) hydrazine feed tank (used during wet layup), one 7-percent solution ammonia feed tank (used during normal operation), and one 29.4-percent solution ammonia feed tank (used during wet layup). Any overflow and drainage from these tanks is collected in the chemical waste sump and processed in the liquid waste systems. In addition, the area around these tanks is diked with drains piped to the chemical waste sump. Fresh air ducts provide adequate ventilation to this storage area.

7.3.2.3 Boric Acid

Boric acid is used as a supplement to the control rod system for reactivity control of the reactor. The chemical and volume control system, primary grade water system, and boron recovery system provide a means of changing the primary system boric acid concentration. Boric acid (4-weight percent solution) is stored in two 13,600-gallon boric acid tanks (designed and built to ASME III) located within the auxiliary building. Any leakage from these tanks is collected in the building sump and is treated by the liquid waste system.

LIST OF EFFECTIVE PAGES

APPENDIX 7A

CONSEQUENCE MODELING CONSIDERATIONS

APPENDIX 7A

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APPENDIX 7A

CONSEQUENCE MODELING CONSIDERATIONS

7A.1 EVACUATION MODEL

"Evacuation," used in the context of offsite emergency response in the event of substantial amount of radioactivity release to the atmosphere in a reactor accident, denotes an early and expeditious movement of people to avoid exposure to the passing radioactive cloud and/or to acute ground contamination in the wake of the cloud passage. It should be distinguished from "relocation" which denotes a post-accident response to reduce exposure from long-term ground contamination. The Reactor Safety Study (RSS) (USNRC 1975) consequence model contains provision for incorporating radiological consequence reduction benefits of public evacuation. The benefits of a properly planned and expeditiously carried-out public evacuation would result in a reduction of early health effects associated with early exposure; namely, in the number of cases of early fatality and acute radiation sickness which would require hospitalization. The evacuation model originally used in the RSS consequence model is described in WASH-1400 as well as in NUREG-0340 (USNRC 1977). However, the evacuation model which has been used herein is a modified version of the RSS model (Sandia Laboratories 1978) and is, to a certain extent, oriented toward site emergency planning. The modified version is briefly outlined here.

The model uses a circular area with a specified radius (such as a 10-mile plume) exposure pathway emergency planning zone (EPZ), with the reactor at the center. It is assumed that people living within portions of this area would evacuate if an accident occurred involving imminent or actual release of significant quantities of radioactive material to the atmosphere.

Significant atmospheric releases of radioactive material would in general be preceded by 1 or more hours of warning time (postulated as the time interval between the awareness of impending core melt and the beginning of the radioactive release from the containment building). For the purpose of calculation of radiological exposure, the model assumes that all people who live in a fan-shaped area (fanning out from the reactor) within the circular zone with the downwind direction as its median--those people who would potentially be under the radioactive cloud that would develop following the release--would leave their residences after a specified delay* and then evacuate. The delay time is reckoned from the beginning of the warning time and is recognized as the sum of the time required by the reactor operators to notify the responsible authorities, time required by the authorities to interpret the data, decide to

*Assumed to be of a constant value which would be the same for all evacuees.

evacuate, and direct the people to evacuate, and time required for the people to mobilize and get underway.

The model assumes that each evacuee would move radially out in the downwind direction** with an average effective speed* (obtained by dividing the zone radius by the average time taken to clear the zone after the delay time) over a fixed distance* from the evacuee's starting point.

This distance is selected to be 14 miles (which is 4 miles more than the 10-mile plume exposure pathway EPZ radius), based upon approximate distances to mass care centers (evacuation endpoints) discussed in the evacuation plan (Alan M. Voorhees and Associates 1980). After reaching the end of the travel distance, the evacuee is assumed to receive no additional early radiation exposure. However, these individuals can accumulate additional dose when allowed to repopulate land areas contaminated at levels less than interdiction set points. For example, in CRAC2, one interdiction level is a projected external radiation dose of 25 Rem over a period of 30 years.

The model incorporates a finite length of the radioactive cloud in the downwind direction which would be determined by the product of the duration over which the atmospheric release would take place and the average wind speed during the release. It is assumed that the front and the back of the cloud formed would move at equal speed which would be the same as the prevailing wind speed; therefore, its length would remain constant at its initial value. At any time after the release, the concentration of radioactivity is assumed to be uniform over the length of the cloud. If the delay time were less than the warning time, then all evacuees would have a head start; that is, the cloud would be trailing behind the evacuees initially. On the other hand, if the delay time were more than the warning time, then depending on initial locations of the evacuees, the following possibilities exist:

1. An evacuee will still have a head start,
2. The cloud would be overhead when an evacuee starts to leave, or
3. An evacuee initially would be trailing behind the cloud.

However, this initial picture of cloud/people disposition would change as the evacuees travel depending on the relative speed and

*Assumed to be of a constant value which would be the same for all evacuees.

**In the RSS consequence model, the radioactive cloud is assumed to travel radially outward only, and the evacuees travel with the wind, thereby tending to result in higher calculated doses.

positions between the cloud and the people. The cloud and an evacuee might overtake one another one or more times before the evacuee would reach his or her destination. In the model, the radial position of an evacuating person, either stationary or in transit, is compared to the front and the back of the cloud as a function of time to determine a realistic period of exposure to airborne radionuclides. The model calculates the time periods during which people are exposed to radionuclides on the ground while they are stationary and while they are evacuating. Because radionuclides would be deposited continually from the cloud as it passed a given location, a person who is under the cloud would be exposed to ground contamination less concentrated than if the cloud had completely passed. To account for this at least in part, the revised model assumes that persons are:

1. Exposed to the total ground contamination concentration which is calculated to exist after complete passage of the cloud, after they have been completely passed by the cloud;

2. Exposed to one-half the calculated concentration anywhere under the cloud; and

3. Not exposed when they are in front of the cloud. Different values of the shielding protection factors for exposures from airborne radioactivity and ground contamination have been used.

Results are discussed in Sections 7.1.3.2.2, 7.1.3.2.3, and 7.1.3.3 for accidents involving significant release of radioactivity to the atmosphere based on the assumption that all people within the 10-mile plume exposure pathway EPZ would evacuate in accordance with the evacuation scenario described above. It is not expected that detailed inclusion of any special facility near a specific plant site, where not all persons would be quickly evacuated, would significantly alter the conclusions. A delay time before evacuation of 1.75 hours was used, based on an evacuation and mass notification study performed for DLC by Alan M. Voorhees and Associates (1980). The estimated evacuation speed, based upon this study, is 3.2 mph (1.45 m/sec) for typical weather conditions. Due to the hilly terrain in the vicinity of BVPS and the potential for severe weather, a second evacuation scheme was modeled using an evacuation speed of 1.9 mph (0.9 m/sec). Based upon weather data for the BVPS area presented in Table 2.3-1 and in Section 2.3.1.2.6, the mean number of days with either greater than 1 inch of snow, thunderstorms, low visibility, or freezing rain is 76. Therefore, the probability for the slower speed evacuation scheme is 76/365 = 0.208, and 0.792 for evacuation during typical weather conditions.

The evacuation model as applied to the BVPS-2 analysis assumed evacuation and temporary relocation of all people within a 10-mile radius of the reactor, irrespective of release duration. The cost associated with this evacuation and relocation was taken to be $165

per person which includes the cost of food and temporary sheltering for a period of one week (USNRC 1983).

7A.2 EARLY HEALTH EFFECTS MODELS

The early health effects model used in this evaluation assumes the availability of supportive medical treatment for individuals whose absorbed dose is greater than 200 Rem. The effect of the availability of such treatment is to increase the LD 50/60 (dose which is lethal to 50 percent of an exposed population within 60 days) from approximately 340 Rem to approximately 510 Rem (whole body dose). A detailed description of this model can be found in NUREG-0340 (USNRC 1977).

7A.3 CONSEQUENCE CODE INPUT DATA

The sources of input data for the consequence analysis code are summarized in Table 7A-1.

7A.4 REFERENCES FOR APPENDIX 7A

Alan M. Voorhees and Associates 1980. Study Report for the Beaver Valley Power Station, Evacuation and Mass Notification.

Bell, M.J. 1973. ORIGEN - The ORNL Isotope Generation and Depletion Code.

Rose, P. S. and Burrow, T. W. 1976. ENDF/B-IV Fission Product Decay Data.

Sandia Laboratories 1978. A Model of Public Evacuation for Atmospheric Radiological Release.

U.S. Nuclear Regulatory Commission (USNRC) 1975. Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. WASH-1400 (NUREG-75/014).

USNRC 1977. Overview of the Reactor Safety Study Consequences Model. NUREG-0340.

USNRC 1983. Calculations of Reactor Accident Consequences Version 2: CRAC2 Computer Code. NUREG/CR-2326.

TABLE 7A-1

CONSEQUENCE ANALYSIS INPUT DATA

	CRAC2 Input Subgroup	Source
1.	Spatial	USNRC 1983
2.	Site	USNRC 1983
3.	Economic	USNRC 1983
4.	Population (1980)	FSAR Chapter 2
5.	Topography	Calculation
6.	Isotope	USNRC 1983; Bell 1973; Rose and Burrows 1976
7.	Leakage	Section 7.1.3.2.1.4
8.	Dispersion	USNRC 1983; FSAR Chapter 2
9.	Evacuate	Alan M. Voorhees and Associates 1980; FSAR Section 2.3
10.	Acute	USNRC 1983
11.	Latent	USNRC 1983; Central Estimate Model
12.	Chronic	USNRC 1983
13.	Scale	USNRC 1983
14.	Result	USNRC 1983
15.	Options	USNRC 1983
16.	Meteorological Data	DLC site meteorological tower, 1979 data*

NOTE:

*Selected for the high percentage of data recovery.

LIST OF EFFECTIVE PAGES

TABLE 12.1-1

ENVIRONMENTAL APPROVALS AND CONSULTATION

Agency	Permit or Approval	Statutory Authority	Purpose	Status
I. FEDERAL AGENCIES				
U.S. Nuclear Regulatory Commission	Construction Permit	Atomic Energy Act of 1954, as amended; 10 CFR 50	Construct BVPS-2	Permit #CPPR-105 issued 5/3/74 renewed 1/30/80
	Operating License	Atomic Energy Act of 1954, as amended; 10 CFR 50	Operate BVPS-2	Application filed 12/21/82 FSAR, ER tendered 1/26/83 Application docketed 5/18/83
	Special Nuclear Materials License	Atomic Energy Act of 1954, as amended; 10 CFR 70	Reactor neutron sources, RPV surveillance capsules, U_{235} enriched fuel	Application to be filed
	Source and By-Product Materials License	Atomic Energy Act of 1954, as amended; 10 CFR 30	Instrumentation calibration	Application to be filed
U.S. Department of Energy	Contract with the Secretary of Energy	Nuclear Waste Policy Act of 1982 P.L. 97-425	High-level radioactive waste disposal	Contract to be negotiated
U.S. Environmental Protection Agency	NPDES Permit*	FWPCA Section 402, P.L. 92-500; 33 USC 1251	Discharge from site construction sedimentation pond	Permit #PA 0027707 issued 6/17/76
			Discharge from construction sedimentation pond for auxiliary intake structure	Permit #PA 0025615 (Amendment 1) issued 12/16/75
			Test and operate auxiliary intake	Permit #PA 0025615 (Amendment 3) issued 6/7/76

TABLE 12.1-1 (Cont)

Agency	Permit or Approval	Statutory Authority	Purpose	Status
U.S. Environmental Protection Agency (Cont)	RCRA Hazardous Waste Facility Permanent Identification Number	Resource Conservation and Recovery Act, P.L. 74-580; 40 CFR 122-125, 260-266	Facility tracking under RCRA hazardous waste program	Identification Number PAD 98 071 4661 issued 7/23/82
Federal Aviation Administration	Air Navigation Clearance for Structures Approval	Federal Aviation Act of 1958; 14 CFR 77	Construct cooling tower	Permit #80 AEA-155-OE issued 4/1/80
Federal Communications Commission	Radio Frequency Approval	47 USC Section 151	Operate overhead monorail crane	Application to be filed
Army Corps of Engineers	Work in Navigable Waters, Dredge and Fill Permit	Rivers and Harbors Act, Section 10, 33 USC 403; Federal Water Pollution Control Act, Section 404, 86 Stat. 816, P.L. 92-500	Construct auxiliary intake	Permit #75089 issued 10/6/75
			Construct barge slip and haul road	Permit #77029 issued 7/12/77
			Construct parking lot	Permit #77030 issued 8/4/77
			Construct emergency outfall structure	Application to be filed
Department of Transportation - U.S. Coast Guard	Navigation Lighting Approval	33 CFR 66.01-35	Navigation aid for barge slip	Issued 8/17/77
II. STATE AGENCIES				
Pennsylvania Department of Environmental Resources	NPDES Permit*	Federal Water Pollution Control Act, Section 402, P.L. 92-500	Discharge liquids from BVPS-2	Application filed 3/15/83

TABLE 12.1-1 (Cont)

Agency	Permit or Approval	Statutory Authority	Purpose	Status
Pennsylvania Department of Environmental Resources (Cont)	Water Quality Management Permit*	Clean Streams Law, P.L. 1987, as amended; 35 P.S. 691.1 et seq.; Water Obstruction Act, P.L. 555, as amended; 32 P.S. 681 et seq.	Discharge industrial waste	Permit #0473211 issued 4/11/74 amended 3/18/76 for auxiliary intake structure
			Construct sewage treatment facilities	Permit #0479403 issued 4/1/80
			Construct BVPS-2 sewage treatment facilities	Permit #0482404 issued 11/10/82
	State Water Quality Certification	Federal Water Pollution Control Act, Section 401, P.L. 92-500	Construct BVPS-2	Certification granted 1/23/74
			Construct auxiliary intake structure	Certification for Permit #0475711 issued 9/19/75
			Construct barge slip and haul road	Certification for Permit #0477705 issued 4/29/77
			Construct parking lot	Issued 7/12/77
	Encroachment Permit	P.L. 555, as amended, 71 P.S. 51 et seq.	Construct Peggs Run sheet piling retaining wall	Permit #0473734 issued 2/26/74
			Construct auxiliary intake structure	Permit #0475711 issued 8/29/75

TABLE 12.1-1 (Cont)

Agency	Permit or Approval	Statutory Authority	Purpose	Status
Pennsylvania Department of Environmental Resources (Cont)	Encroachment Permit (Cont)	P.L. 55, as amended, 71 P.S. 51 et seq. (Cont)	Construct barge slip and haul road	Permit #0477705 issued 4/21/77
			Construct parking lot	Issued 5/21/77
			Construct Peggs Run culvert extension	Permit #0477723 Issued 12/8/77
	Erosion and Sedimentation Control Plan Approval	Clean Streams Law, P.L. 1987; 35 P.S. Section 691 et seq.	Construct BVPS-2	Permit #0473802 Approved 1/16/74
			Construct auxiliar. intake structure	Approved 8/4/75
	Erosion and Sedimentation Control Plan Approval (Cont)		Construct barge slip and haul road	Approved 3/8/77
			Construct parking lot	Approved 3/8/77
			Construct emergency outfall structure	Application to be filed
	Plan Approval	Air Pollution Control Act, P.L. 2119, as amended	Construct air contamination source - radiological	Approval #04-306-002 issued 1/6/76 extension not required 5/26/82
			Operate air contamination source - radiological	Approval not required 5/26/82
			Construct air contamination source - auxiliary boilers	Application to be filed

TABLE 12.1-1 (Cont)

Agency	Permit or Approval	Statutory Authority	Purpose	Status
Pennsylvania Department of Environmental Resources (Cont)			Construct air contamination source - diesel generators	Application to be filed
	PPC Plan	25 Pa. Code, Chapters 75 and 101; 40 CFR 125, Subpart K and 40 CFR 151	Prevention and management of accidental pollution releases	Application to be filed
Pennsylvania Department of Labor and Industry	Plan Approval	Fire and Panic Act, P.L. 465, as amended	Building permit - construct various buildings	Issued 10/24/73 amended as required
			Install diesel fuel oil tanks	Issued 10/23/78
	ASME Stamp	35 P.S. Section 1301 et seq.	Install reactor containment liner	Issued 10/1/79
Pennsylvania Department of Transportation	Highway Occupancy Permit	Pennsylvania Act No. 287	Place fill within highway right-of-way	Permit #P-328799 issued 10/19/77 Supplement #67114 issued 7/13/82
Pennsylvania State Police	Installation Approval	P.L. 450, Section 1, as amended; 35 P.S. 1181	Install storage for combustible liquids	Application to be filed
III. LOCAL AGENCIES				
Borough of Shippingport	Planning Module for Land Development	Pennsylvania Sewage Facilities Act, Act 537	Construct sewage treatment system	Issued 6/8/78

NOTE:

*NPDES Permit is now under the jurisdiction of the Pennsylvania Department of Environmental Resources (DER). Pennsylvania DER permit combines the establishment of effluent limitations with a permit to construct waste treatment facilities.

NRC QUESTIONS AND RESPONSES

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES (Cont)

NRC Letter: May 4, 1983

Question E291.5 (ER Section 2.2.2)

Discuss the potential for occurrence of Federal- or State-listed endangered or threatened species of fish and shellfish in the vicinity of the BVPS site.

Response:

There are no State-listed endangered or threatened or Federal-listed threatened fish or shellfish in the vicinity of the BVPS site.

However, three Federal-listed endangered species of mollusks could occur in the Ohio River. The three species are Plethobasus striatus, Pleurobema plenum, and Lampsilis abrupta. The baseline aquatic studies (1973-1974) and the subsequent eight annual reports of the biological monitoring program, through 1982, reveal that these species were not collected. The U.S. Fish and Wildlife Service State List for Pennsylvania indicates these species have not been collected recently in the Ohio River. Therefore, their potential occurrence in the vicinity of the BVPS site is slight.

NRC Letter: May 4, 1983

Question E291.6 (ER Section 2.2.2)

Have any conclusions been drawn (or hypotheses proposed) with regard to the apparent disappearance of the Asiatic clam from the New Cumberland Pool; if so, provide additional details.

Response:

The absence of Asiatic clam, Corbicula, from benthos samples taken at the site in 1979 and 1980 does not indicate that Corbicula has disappeared from New Cumberland Pool. Corbicula was collected at all benthos sampling stations in 1981 (Duquesne Light Company 1982). The absence of Corbicula in 1979 and 1980 samples was probably due to reduced abundance in the sampling areas. Temporal and spatial variation in abundance of Corbicula can be expected due to natural and other variations in physical conditions. Corbicula are found in a variety of habitats similar to that of New Cumberland Pool. Unless major long-term changes in physical conditions occur, it is unlikely that Corbicula will disappear from New Cumberland Pool.

Reference:

Duquesne Light Company 1982. 1981 Annual Environmental Report. Nonradiological, Vol. 1. Pittsburgh, PA.

NRC Letter: May 4, 1983

Question E291.15 (ER Section 5.3.2)

Indicate why the more recent ORSANCO data, recorded from October 1976 through March 1981, were not used for all parameters rather than using the older NUS Corporation data of 1974.

Response:

The decision to use the NUS Corporation data of 1974 rather than the more recent ORSANCO data recorded from October 1976 through March 1981 was based upon the following factors:

1. A subjective evaluation of the two sets of data summarized in Table 2.4-10 did not reveal meaningful changes in river quality during the period of 1976 through 1981.

2. The NUS Corporation data were collected at five locations in the vicinity of the plant site. There are two ORSANCO stations in the vicinity of the site: one located approximately 20 miles upstream, and one located 5 miles downstream. Of the two, the downstream station (East Liverpool, Ohio) is the most relevant since it is located in the same New Cumberland pool. Yet this station may not be totally indicative of the characteristics at BVPS-2 because of pollution sources and discharges located between BVPS and the ORSANCO sampling station.

3. The NUS samples consisted of composites of the transect of the river and are probably more representative of the full river width than are the ORSANCO samples, which were collected at single points.

4. The use of one data set in place of the other does not produce meaningful differences in assessing the impacts on the aquatic biota. Table 5.3-4, Amendment 1, presents a summary of the impacts of the plant cooling water blowdown on the Ohio River. The maximum ambient river concentrations in Table 5.3-4 are based upon both NUS and ORSANCO data. For constituents included in the NUS data, the maximum recorded NUS value was used. For constituents not included in the NUS data, the maximum ORSANCO value was used. To determine the validity of the NUS data, a comparison of ORSANCO data and NUS data was performed and is included in Table E291.15-1.

In comparing the single highest value recorded by ORSANCO for each constituent during the period October 1975 through December 1982 to the NUS maximum values, it was found that seven constituents in the ORSANCO data were higher than the corresponding values in

Table 5.3-4. The seven constituents are alkalinity, total iron, nickel, nitrate/nitrite, phenolics, zinc, and manganese. Each of these is discussed below.

Alkalinity - The effect of a somewhat higher alkalinity is generally an increase in the buffering capacity of the water and a reduction in the toxicity of heavy metals to fish. Therefore, use of the NUS data represents a more conservative value upon which to base the impact assessments. Both the maximum ORSANCO and NUS values are within the Pennsylvania Water Quality Criteria (PWQC) since the criterion is a minimum value (20 mg/l).

Total Iron - The maximum value observed by ORSANCO was 17 mg/l while the maximum value observed by NUS was 3.8 mg/l. The mean concentrations for ORSANCO and NUS data were 3.14 mg/l and 1.5 mg/l, respectively.

Iron will primarily be in the precipitate form, and potential impacts will therefore be of the same nature as discussed in ER Section 5.3. Since the addition of iron to the Ohio River from BVPS is very small relative to river conditions, impacts on the aquatic habitat are expected to be localized and primarily limited to effects on benthic organisms. These potential effects are due more to the existing high ambient river concentrations of iron than to concentration in the cooling tower.

Nickel - The maximum recorded nickel concentrations in the NUS and ORSANCO data are 0.02 mg/l and 0.07 mg/l, respectively. These values translate to a maximum cooling tower blowdown concentration of 0.05 mg/l using NUS data and 0.168 mg/l using ORSANCO data. Both values are within the range of the PWQC which, based upon hardness values typically seen at the site, is 0.052 mg/l to 0.176 mg/l. The cooling tower blowdown concentrations are about 0.01 of the LC_{50} for a sensitive congener of a resident species. Therefore, the discharge of nickel is not expected to have adverse impacts on the biota of the Ohio River.

Nitrate/Nitrite - The maximum concentrations of nitrate/nitrite recorded in the NUS and ORSANCO data are 2.47 mg/l and 2.6 mg/l, respectively. The maximum concentration of this constituent in the cooling tower blowdown is 5.93 mg/l using NUS data and 6.24 mg/l using ORSANCO data. Both of these values are within the PWQC of 10 mg/l. Therefore, the blowdown concentrations of nitrate/nitrite are not expected to impact the aquatic biota of the Ohio River.

Phenolics - The maximum concentrations of phenolics recorded in the NUS and ORSANCO data are 0.017 mg/l and 0.07 mg/l, respectively. Both of these values exceed the PWQC of 0.005 mg/l. However, the maximum cooling tower blowdown concentration of 0.041 mg/l using NUS data and 0.168 mg/l using ORSANCO data are below the level considered harmful to fish and aquatic life (0.20 mg/l), according to McKee and Wolf in a USEPA (1976) report.

NRC Letter: May 4, 1983

Question E291.29 (ER Section 3.6)

Provide the details of the intermittent chlorination program planned for the circulating water system (i.e., frequency of application, duration of application, estimated applied dosage range, estimated total residual chlorine concentration in the Unit 2 blowdown). Provide similar information for the service water system.

Response:

Refer to Section 3.6.1, Amendment 1, for the circulating water system chlorination program and Section 3.6.3, Amendment 1, for the service water system chlorination program.

NRC Letter: May 4, 1983

Question E291.30 (ER Section 3.6)

Provide aquatic organism toxicity data and biodegradability information, if available, for Calgon Cl-95.

Response:

Calgon Chemical Cl-4000 has been substituted for Cl-95 (Table 3.6-3, Amendment 1). Information received from Calgon indicates that the 96-hour LC_{50} for bluegill sunfish is greater than 1,000 ppm, and the 48-hour LC_{50} for daphnia magna is 590 ppm. Calgon states that Cl-4000 is readily biodegradable, with BOD and TOC values of 30 mg/g and 150 mg/g, respectively. Cl-4000 will be added to the cooling tower so that the average concentration will be approximately 2.2 ppm.