ATTACHMENT I

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DCKET 50-155 AND 72-043 – LICENSE DPR-5

REVISION 10 TO THE UPDATED FINAL HAZARDS SUMMARY REPORT (UFHSR)
Submitted September 17, 2002

Revision Instructions
1 Page

and

List of Effective Pages
7 Pages

The following pages have been revised per various QRs, and are now **Revision 10**:
(issued 9/13/02)

INSERT (rev. 10)	REMOVE (various revs.)
Index, pages 1 – 14	Index, pages 1 - 14
List of Effective Pages, pages 1 – 7	List of Effective Pages, pages 1 – 8
Chapter 2 *revised per QR# 384-01* Table of Contents, (1 page) Section 2.1, pages 2.1-1 through 2.1-9 Section 2.2, pages 2.2-1 through 2.2-3 Section 2.3, pages 2.3-1 through 2.3-9 Section 2.5, pages 2.5-1 through 2.5-19 References, (2 pages)	Table of Contents, (1 page) Section 2.1, pages 2.1-1 through 2.1-12 Section 2.2, pages 2.2-1 through 2.2-6 Section 2.3, pages 2.3-1 through 2.3-9 Section 2.5, pages 2.5-1 through 2.5-22 References, (2 pages)
Chapter 3 *revised per QR# 286-02* Table of Contents (2 pages) Section 3.2, pages 3.2-1 through 3.2-14 Section 3.8, pages 3.8-1 through 3.8-19	Table of Contents (2 pages) Section 3.2, pages 3.2-1 through 3.2-18 Section 3.8, pages 3.8-1 through 3.8-21
Chapter 6 *revised per QR# 287-02* Table of Contents (2 pages) Section 6.2, pages 6.2-1 through 6.2-5 Section 6.4, page 6.4-1 Section 6.7, page 6.7-1	Table of Contents (2 pages) Section 6.2, pages 6.2-1 through 6.2-6 Section 6.4, page 6.4-1 Section 6.7, page 6.7-1
Chapter 9 *revised per QR# 288-02* Table of Contents (2 pages) Section 9.1, pages 9.1-1 through 9.1-30 Section 9.2, pages 9.2-1 through 9.2-5 Section 9.3, page 9.3-1 Section 9.4, pages 9.4-1 through 9.4-3 Section 9.5, pages 9.5-1 through 9.5-13 References (1 page)	Table of Contents (2 pages) Section 9.1, pages 9.1-1 through 9.1-38 Section 9.2, pages 9.2-1 through 9.2-5 Section 9.3, page 9.3-1 Section 9.4, pages 9.4-1 through 9.4-7 Section 9.5, pages 9.5-1 through 9.5-13 References (1 page)
Chapter 11 *per QR#s 289-02 and 290-02* Table of Contents, (1 page) Section 11.1, pages 11.1-1 through 11.1-2 Section 11.2, pages 11.2-1 through 11.2-3 Section 11.3, pages 11.3-1 through 11.3-2 Section 11.4, pages 11.4-1 through 11.4-3 Section 11.5, pages 11.5-1 through 11.5-3 Section 11.6, page 11.6-1	Table of Contents, (1 page) Section 11.1, pages 11.1-1 through 11.1-2 Section 11.2, pages 11.2-1 through 11.2-3 Section 11.3, pages 11.3-1 through 11.3-2 Section 11.4, pages 11.4-1 through 11.4-3 Section 11.5, pages 11.5-1 through 11.5-5 Section 11.6, page 11.6-1
Chapter 12, *revised per QR# 291-02* Table of Contents, (1 page) Section 12.1, pages 12.1-1 through 12.1-3 Section 12.2, page 12.2-1 Section 12.3, pages 12.3-1 through 12.3-6 Section 12.4, pages 12.4-1 through 12.4-3 Section 12.5, pages 12.5-1 through 12.5-2	Table of Contents, (1 page) Section 12.1, pages 12.1-1 through 12.1-3 Section 12.2, page 12.2-1 Section 12.3, pages 12.3-1 through 12.3-6 Section 12.4, pages 12.4-1 through 12.4-3 Section 12.5, pages 12.5-1 through 12.5-2
Chapter 13, *revised per QR# 431-02* Table of Contents, (1 page) Section 13.5, pages 13.5-1 through 13.5-5	Table of Contents, (1 page) Section 13.5, pages 13.5-1 through 13.5-6
Chapter 15, *revised per QR# 758-01* Table of Contents, (2 pages) Section 15.10, pages 15.10-1 through 15.10-16	Table of Contents, (2 pages) Section 15.10, pages 15.10-1 through 15.10-15 and unnumbered reference pages for section 15.10 (2 pages)

CHAPTER 1 - INTRODUCTION AND GENERAL PURPOSES

1.1-1	Revision 8
1.2-1	Revision 8
1.2-2	Revision 8
1.2-3	Revision 8
1.2-4	Revision 8
1.3-1	Revision 8
1.4-1	Revision 8
1.5-1	Revision 8
1.6-1	Revision 8
1.7-1	Revision 8
1.7-2	Revision 8

CHAPTER 2 - SITE CHARACTERISTICS

2.1-1	Revision 10
2.1-2	Revision 10
2.1-3	Revision 10
2.1-4	Revision 10
2.1-5	Revision 10
2.1-6	Revision 10
2.1-7	Revision 10
2.1-8	Revision 10
2.1-9	Revision 10
2.2-1	Revision 10
2.2-2	Revision 10
2.2-3	Revision 10
2.3-1	Revision 10
2.3-2	Revision 10
2.3-3	Revision 10
2.3-4	Revision 10
2.3-5	Revision 10
2.3-6	Revision 10
2.3-7	Revision 10
2.3-8	Revision 10
2.3-9	Revision 10
2.4-1	Revision 8
2.4-2	Revision 8
2.4-3	Revision 8
2.4-4	Revision 8
2.4-5	Revision 8
2.4-6	Revision 9
2.5-1	Revision 10
2.5-2	Revision 10
2.5-3	Revision 10
2.5-4	Revision 10
2.5-5	Revision 10
2.5-6	Revision 10
2.5-7	Revision 10

2.5-8	Revision 10
2.5-9	Revision 10
2.5-10	Revision 10
2.5-11	Revision 10
2.5-12	Revision 10
2.5-13	Revision 10
2.5-14	Revision 10
2.5-15	Revision 10
2.5-16	Revision 10
2.5-17	Revision 10
2.5-18	Revision 10
2.5-19	Revision 10

CHAPTER 3 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1-1	Revision 9
3.1-2	Revision 9
3.2-1	Revision 10
3.2-2	Revision 10
3.2-3	Revision 10
3.2-4	Revision 10
3.2-5	Revision 10
3.2-6	Revision 10
3.2-7	Revision 10
3.2-8	Revision 10
3.2-9	Revision 10
3.2-10	Revision 10
3.2-11	Revision 10
3.2-12	Revision 10
3.2-13	Revision 10
3.2-14	Revision 10
3.3-1	Revision 8
3.3-2	Revision 8
3.3-3	Revision 8
3.3-4	Revision 8
3.3-5	Revision 8
3.3-6	Revision 8
3.4-1	Revision 8
3.4-2	Revision 8
3.4-3	Revision 8
3.5-1	Revision 8
3.5-2	Revision 8
3.5-3	Revision 8
3.5-4	Revision 8
3.5-5	Revision 8
3.5-6	Revision 8
3.5-7	Revision 8
3.6-1	Revision 8
3.7-1	Revision 8
3.8-1	Revision 10

3.8-2	Revision 10
3.8-3	Revision 10
3.8-4	Revision 10
3.8-5	Revision 10
3.8-6	Revision 10
3.8-7	Revision 10
3.8-8	Revision 10
3.8-9	Revision 10
3.8-10	Revision 10
3.8-11	Revision 10
3.8-12	Revision 10
3.8-13	Revision 10
3.8-14	Revision 10
3.8-15	Revision 10
3.8-16	Revision 10
3.8-17	Revision 10
3.8-18	Revision 10
3.8-19	Revision 10
3.9-1	Revision 8
3.9-2	Revision 8
3.10-1	Revision 8
3.11-1	Revision 8

CHAPTER 4 - REACTOR

4.0-1	Revision 7

CHAPTER 5 - REACTOR COOLANT AND CONNECTED SYSTEMS

5.1-1	Revision 8
5.2-1	Revision 8
5.2-2	Revision 8
5.2-3	Revision 8
5.2-4	Revision 8
5.3-1	Revision 8
5.4-1	Revision 8
5.4-2	Revision 8

CHAPTER 6 - ENGINEERED SAFETY FEATURES (ESF)

6.1-1	Revision 8
6.2-1	Revision 10
6.2-2	Revision 10
6.2-3	Revision 10
6.2-4	Revision 10
6.2-5	Revision 10
6.3-1	Revision 8
6.4-1	Revision 10
6.5-1	Revision 8
6.6-1	Revision 8

9.1-20	Revision 10
9.1-21	Revision 10
9.1-22	Revision 10
9.1-23	Revision 10
9.1-24	Revision 10
9.1-25	Revision 10
9.1-26	Revision 10
9.1-27	Revision 10
9.1-28	Revision 10
9.1-29	Revision 10
9.1-30	Revision 10
9.2-1	Revision 10
9.2-2	Revision 10
9.2-3	Revision 10
9.2-4	Revision 10
9.2-5	Revision 10
9.3-1	Revision 10
9.4-1	Revision 10
9.4-2	Revision 10
9.4-3	Revision 10
9.5-1	Revision 10
9.5-2	Revision 10
9.5-3	Revision 10
9.5-4	Revision 10
9.5-5	Revision 10
9.5-6	Revision 10
9.5-7	Revision 10
9.5-8	Revision 10
9.5-9	Revision 10
9.5-10	Revision 10
9.5-11	Revision 10
9.5-12	Revision 10
9.5-13	Revision 10
9.6-1	Revision 8

CHAPTER 10 - STEAM POWER CONVERSION SYSTEMS

10.1-1	Revision 8
10.2-1	Revision 8
10.3-1	Revision 8
10.4-1	Revision 9
10.4-2	Revision 9
10.4-3	Revision 9

CHAPTER 11 - RADIOACTIVE WASTE MANAGEMENT

11.1-1	Revision 10
11.1-2	Revision 10
11.2-1	Revision 10
11.2-2	Revision 10

11.2-3	Revision 10
11.3-1	Revision 10
11.3-2	Revision 10
11.4-1	Revision 10
11.4-2	Revision 10
11.4-3	Revision 10
11.5-1	Revision 10
11.5-2	Revision 10
11.5-3	Revision 10
11.6-1	Revision 10

CHAPTER 12 - RADIATION PROTECTION

12.1-1	Revision 10
12.1-2	Revision 10
12.1-3	Revision 10
12.2-1	Revision 10
12.3-1	Revision 10
12.3-2	Revision 10
12.3-3	Revision 10
12.3-4	Revision 10
12.3-5	Revision 10
12.3-6	Revision 10
12.4-1	Revision 10
12.4-2	Revision 10
12.4-3	Revision 10
12.5-1	Revision 10
12.5-2	Revision 10

CHAPTER 13 - CONDUCT OF OPERATIONS

13.1-1	Revision 9
13.1-2	Revision 9
13.1-3	Revision 8
13.1-4	Revision 9
13.2-1	Revision 8
13.2-2	Revision 8
13.2-3	Revision 8
13.2-4	Revision 8
13.3-1	Revision 8
13.4-1	Revision 8
13.5-1	Revision 10
13.5-2	Revision 10
13.5-3	Revision 10
13.5-4	Revision 10
13.5-5	Revision 10
13.6-1	Revision 8

ATTACHMENT II

CONSUMERS ENERGY COMPANY
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DCKET 50-155 AND 72-043 – LICENSE DPR-5

REVISION 10 TO THE UPDATED FINAL HAZARDS SUMMARY REPORT (UFHSR)
Submitted September 17, 2002

Index of Chapters

14 Pages

INDEX

CHAPTER 15: ACCIDENT ANALYSES

ATTACHMENT III

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DCKET 50-155 AND 72-043 – LICENSE DPR-5

REVISION 10 TO THE UPDATED FINAL HAZARDS SUMMARY REPORT (UFHSR)
Submitted September 17, 2002

UFHSR Page Replacements

TABLE OF CONTENTS

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION AND DESCRIPTION

The Big Rock Point Nuclear Plant Site Plan for the facility is shown in Drawing 0740G20003.

The site property consists of gently sloping wooded and cleared land at the western extremity of the southern shore of Little Traverse Bay. The site is 228 miles NNW of Detroit and 262 miles NNE of Chicago.

Figure 2.1 shows the location of the site with respect to the over-all view of the state of Michigan and its surroundings.

Figure 2.2, Site Map, indicates the property owned by Consumers Power Company, in relation to the nearby highway and former railroad. Figure 2.2 also indicates the location of the permanently defueled reactor on the site.

Detailed site location and description is found in BRP Volume 32, Environmental Report for Decommissioning.

2.1.1.1 Plant Features

The principal plant structures include:

A 130 foot diameter spherical containment vessel

- Reactor Building (T-1)

A Turbine Generator Building (B-3)

A structure housing water intake facilities and diesel generator

- Screen, Well and Pump House (B-4)
- Emergency Generator Room (B-5)

A 240 foot stack (chimney) (B-1)

A Security Building (B-16)

Waste Storage Vaults (Liquid) (B-11)
(Solid) (B-10)

Reference BRP Drawing 0740G20003 Site Plan for the Building and Structure locations and to Figure 2.3 for general Plant Facility Identifications.

The containment vessel houses the reactor vessel, steam drum, fuel pool, and equipment for removal of fuel decay heat.

In addition to the structures shown in Figure 2.3 temporary equipment and office structures are being added to support decommissioning.

2.1.1.2 Surrounding Area

Charlevoix County, with a land area of about 400 square miles, has farm earnings (Reference 3) of about $4.2 million per year, with about 17% of its land area in agricultural use. Produce is principally forest, dairy and poultry products, and fruit. Statistics on the economy of the three counties around the site (the approximate thirty-mile radius), are shown in the following table.

TABLE 2.1

STATISTICS OF SURROUNDING AREA
(Reference 3)

County	Antrim	Charlevoix	Emmet
Land Area, sq.mile	477	417	468
Population 2000	23,110	26,090	31,437
Population/sq mile	48.4	62.6	67.2
% of Population Increase 1960-1970	21.6%	23.2%	15.3%
% of Population Increase 1970-1980	28.4%	20.3%	25.4%
% of Population Increase 1980-1990	12.3%	7.8%	8.9%
% of Population Increase 1990-2000	21.3%	17.6%	20.3%
% of Urban Population 1990	~30%	~35%	~30%
Persons/Household 2000	2.51	2.51	2.50
Total Number of Households 2000	6,980	8,243	9,516
Manufacturing Establishments, 1987	26	61	48
% With Over 20 Employees	38.5%	29.5%	33.3%
Average Annual Manufacturing Employment 1987	~600	~200	~1500
Farms, 1987	248	232	211
Average Size Farms, Acres	222	180	213
Value of Farm Products Sold, Average per farm ($)	46,335	18,189	22,061
Including % Farm Crops	48.5%	24.1%	31.7%
% of Livestock and Poultry Products	51.5%	75.9%	68.3%

Typical of most of the northern portion of the southern peninsula of Michigan, and because of comparatively moderate summer climate and abundant lake frontage, the general region of the site is an important summer vacationland. However, this summer occupancy is not a significant factor within about two miles of the plant site.

2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL (Reference 1)

The Big Rock Point Nuclear Power Plant is located on the shore of Lake Michigan in Charlevoix County in the northern part of Michigan's lower peninsula. The plant site is approximately three and one half miles northeast of the city of Charlevoix and eleven miles west of the city of Petoskey, Michigan. The site exclusion area is defined by the site property limits and thus the exclusion area boundary lines are identical to the plant property lines shown on the Site Map. The nearest landside property line is about 2680 feet and the nearest shoreline property line is about 200 feet from the containment sphere.

The approximately 600 acres of property within the exclusion area boundaries including the mineral rights is owned by the Licensee. Parts of the exclusion area are traversed by US Route 31 and the former Chesapeake and Ohio Railroad, portions of which were owned by the Michigan Department of Transportation as shown in Figure 2.2. Arrangements have been made to control traffic on Route 31 in the event of a plant emergency, as documented in the Site Emergency Plan (Reference 2). Similar arrangements, however, have not been made regarding the former railroad line as the access from the west has been rendered impossible by removal of the Pine River Rail trestle and access from the east is currently impossible due to washout of the tracks near Petoskey. Further, sections of track have been removed and portions were abandoned.

The Plant under Michigan law, owns to the water's edge and has the right to control access from the landward side to the lakeshore frontage within the exclusion area. The exclusion area is not defined over the waters of Lake Michigan adjacent to the site. While Big Rock Point has not specifically defined an exclusion area over the water, arrangements have been made with the US Coast Guard, as documented in the Site Emergency Plan (Reference 2), for the control of water traffic offshore of the plant in the event of an emergency.

Evaluation Summary

The topic of Exclusion Area Authority and Control was evaluated by the NRC as part of the Systematic Evaluation Program topic number II-1.A. This review resulted in an assessment and evaluation (Reference 1) which found that the arrangements with the U.S. Coast Guard meet the intent of the criteria in Part 100 and, therefore, the lack of a defined exclusion area over the water does not constitute a significant safety issue for the SEP review.

This evaluation concluded that Big Rock Point has the proper authority, with one exception, to determine all activities within the exclusion area, as required by 10 CFR Part 100. The exception concerned the lack of an arrangement to control traffic on the former Chesapeake and Ohio Railroad line which traversed a part of the exclusion area. This was a departure from current criteria but was not considered a significant safety issue in view of the location of the railroad line in relation to the plant, the then low volume of traffic on the line, and the stated intention of the Licensee to include such an arrangement (* Note 1) in the new Site Emergency Plan. This completed the evaluation of the SEP topic.

*NOTE 1 Since that evaluation was completed, the need to include this arrangement in the Site Emergency Plan has become moot as described in this report. If in the future the railroad line is reopened, arrangements for control of traffic on the line in the event of a plant emergency will be included in the Site Emergency Plan.

2.1.3 POPULATION DISTRIBUTION

The site is remote from any large metropolitan areas, and has generally favorable low surrounding permanent population as shown in Figure 2.4 which was extracted from Reference 4.

Population distribution information is found in BRP Volume 32, Environmental Report for Decommissioning.

2.1.3.1 Population Within Thirty (30) Miles

The region surrounding the Big Rock Point Plant is generally of low population density and rural to suburban in character. The total population within the counties of Charlevoix, Emmet, and Antrim, which covers the majority of the area within 30 miles of the plant, based on 2000 census data, was about 80,600. This region has experienced an overall average increase of 20% in their resident population between 1990 and 2000 (Refer to Table 2.1). The majority of this population increase is attributed to in-migration primarily from other regions of Michigan.

2.1.3.2 Seasonal Population

Seasonal population is an important factor in the area surrounding the plant as this part of Michigan attracts a large number of visitors year round with the peak occurring in the summer season. The seasonal population (ie, seasonal residents, overnight tourists, and daily visitors) in the three county area is established to increase the population by 75% during the height of the season (Reference 6).

2.1.3.3 Low Population Zone and Emergency Planning Zones

The low population zone specified for Big Rock Point site is the area within two and one half (2.5) mile radius of the plant; the primary emergency planning zone is the five (5) mile radius; and the secondary emergency planning zone extends to a thirty (30) mile radius (Reference 2).

2.1.3.4 Population Centers

TABLE 2.2

Principal urban areas within 60 miles are:

Urban Center	Population 1960	Population 1970	Population 1980	Population 1990*	Population 2000**	Distance From Site	Direction From Site
Charlevoix	2,751	3,519	3,296	3,116	2,994	4 Miles	SW
Harbor Springs	1,433	1,662	1,567	1,540	1,567	11 Miles	ENE
Petoskey	6,138	6,342	6,097	6,056	6,080	11 Miles	E
Boyne City	2,797	2,969	3,348	3,478	3,503	14 Miles	SE
East Jordan	1,919	2,041	2,185	2,240	2,507	14 Miles	SSE
Gaylord	2,569	3,012	3,011	3,256	3,681	33 Miles	SE
Cheboygan	5,859	5,553	5,106	4,999	5,295	40 Miles	NE
St Ignace	3,334	2,982	2,632	2,568	2,678	42 Miles	NNE
Traverse City	18,432	18,048	15,516	15,116	14,532	45 Miles	SSW
Grayling	2,015	2,143	1,792	1,944	1,952	52 Miles	SSE

* Population figures are 1990 Census (Reference 5)
** Population figures are 2000 Census (Reference 31)

Charlevoix is the closest urban center and does not currently nor foreseeably fall within the population center definition of 10 CFR Part 100.

2.1.3.5 Population Density

By applying the seasonal population increase to the three county 2000 Census resident population, the cumulative population of the majority of the area within thirty (30) miles of the plant is about 142,000 people for a population density of about one hundred and four (104) persons per square mile. This population density is not expected to approach the 10 CFR Part 100 Guideline Limits during the duration of the plant's NRC license.

2.1.3.6 Evaluation Summary

The topic of Population Distribution was evaluated by the NRC as part of the Systematic Evaluation Program Topic number II-1.B. This review resulted in an assessment and evaluation (Reference 1) which found that based upon an examination of present and projected population data and on observations made during a visit to the site in July 1979, that neither Charlevoix nor any other city within 30 miles of the plant is now, or is likely to become in the foreseeable future, a population center, (more than 25,000 residents), as defined in 10 CFR Part 100. Further, the NRC concluded that the low population zone and population center distances specified for the Big Rock Point site remain valid and the site is in conformance with the distance requirements of 10 CFR Part 100 in that the population center distance is more than one and one-third times the distance from the reactor to the outer boundary of the low population zone.

This completed the evaluation of this SEP Topic. Since the plant site conforms to current licensing criteria, no additional SEP review is required.

Figure 2.1
Location Map Big Rock Plant Site

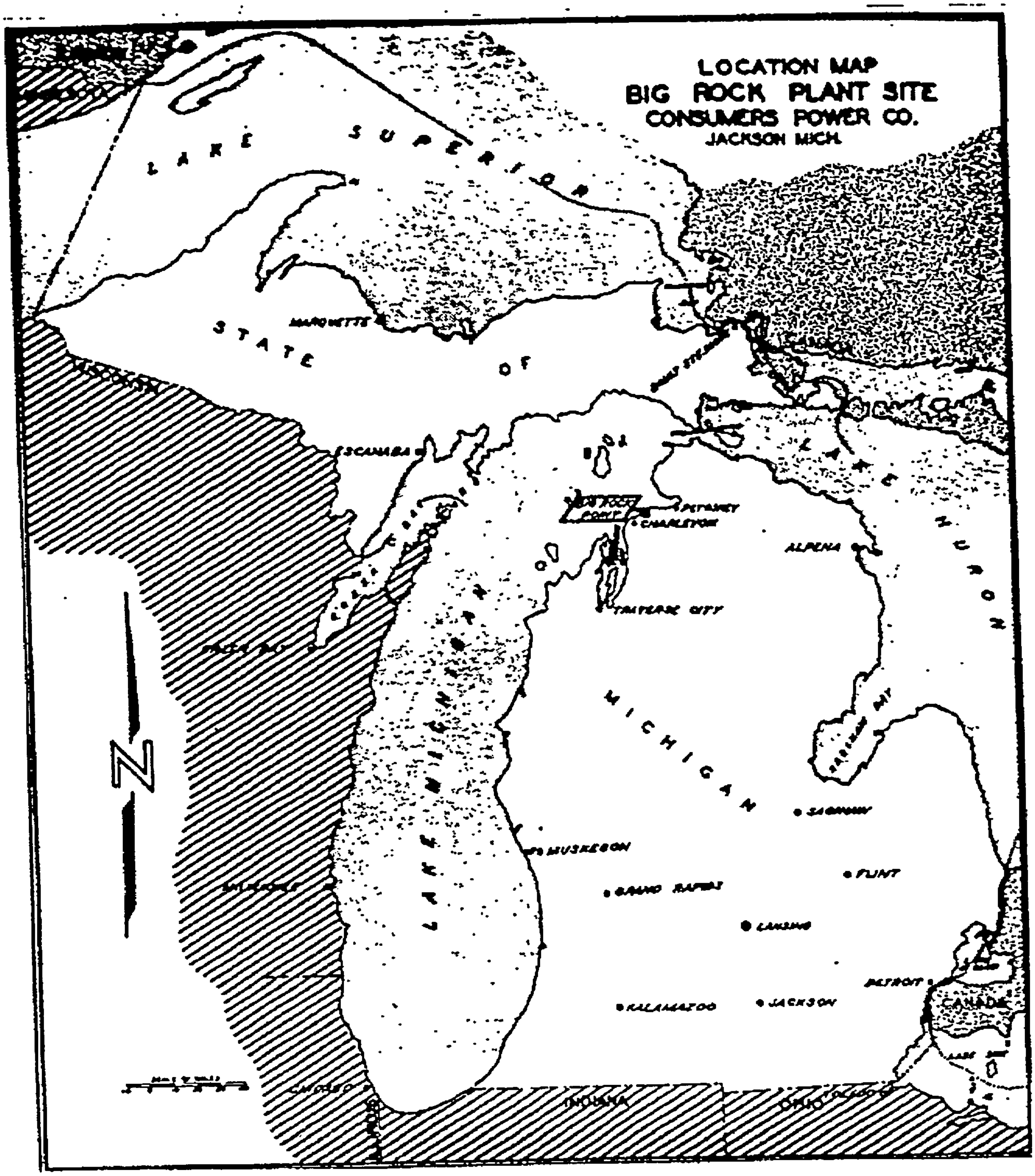


Figure 2.2
Site Map of Big Rock Point Plant

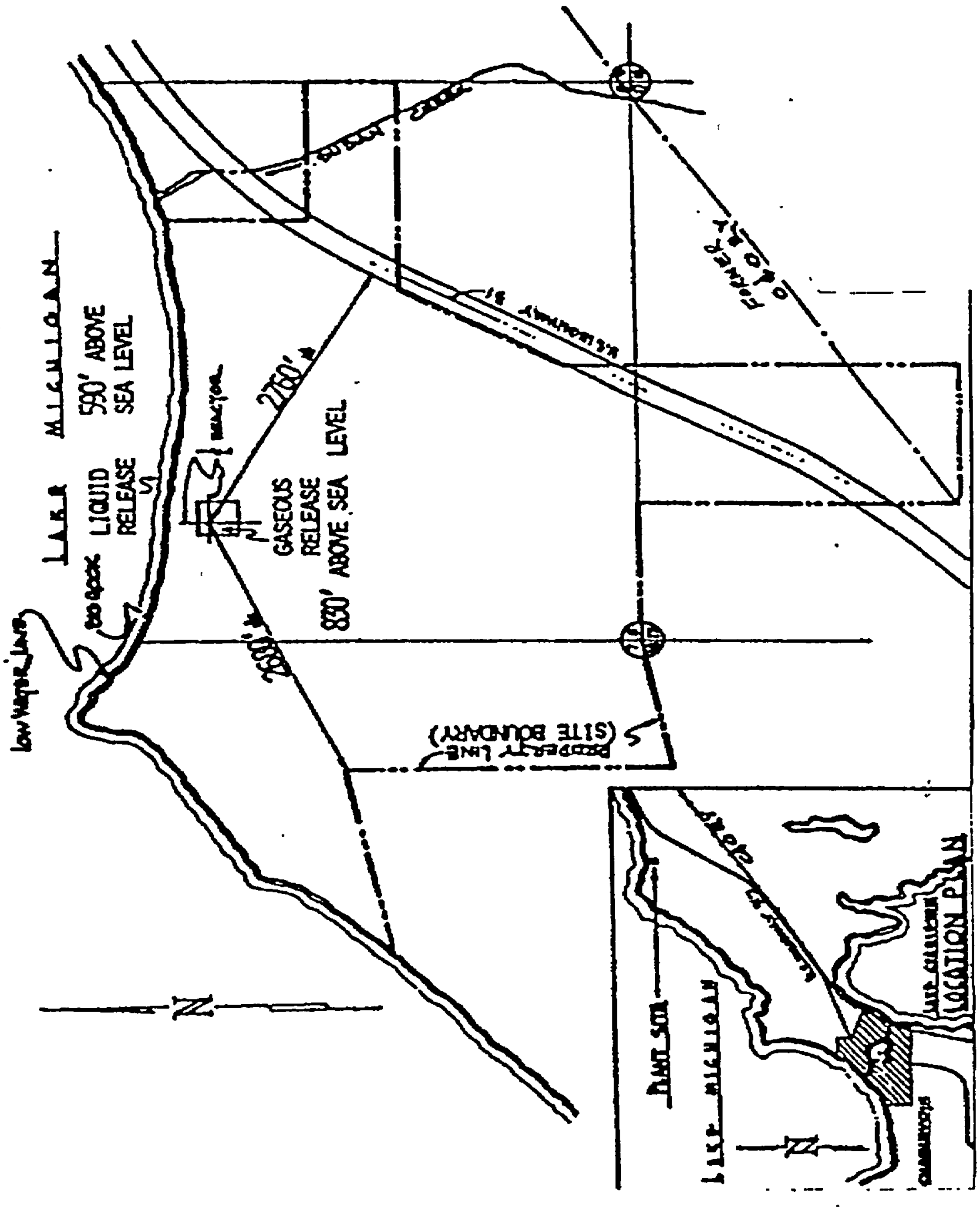
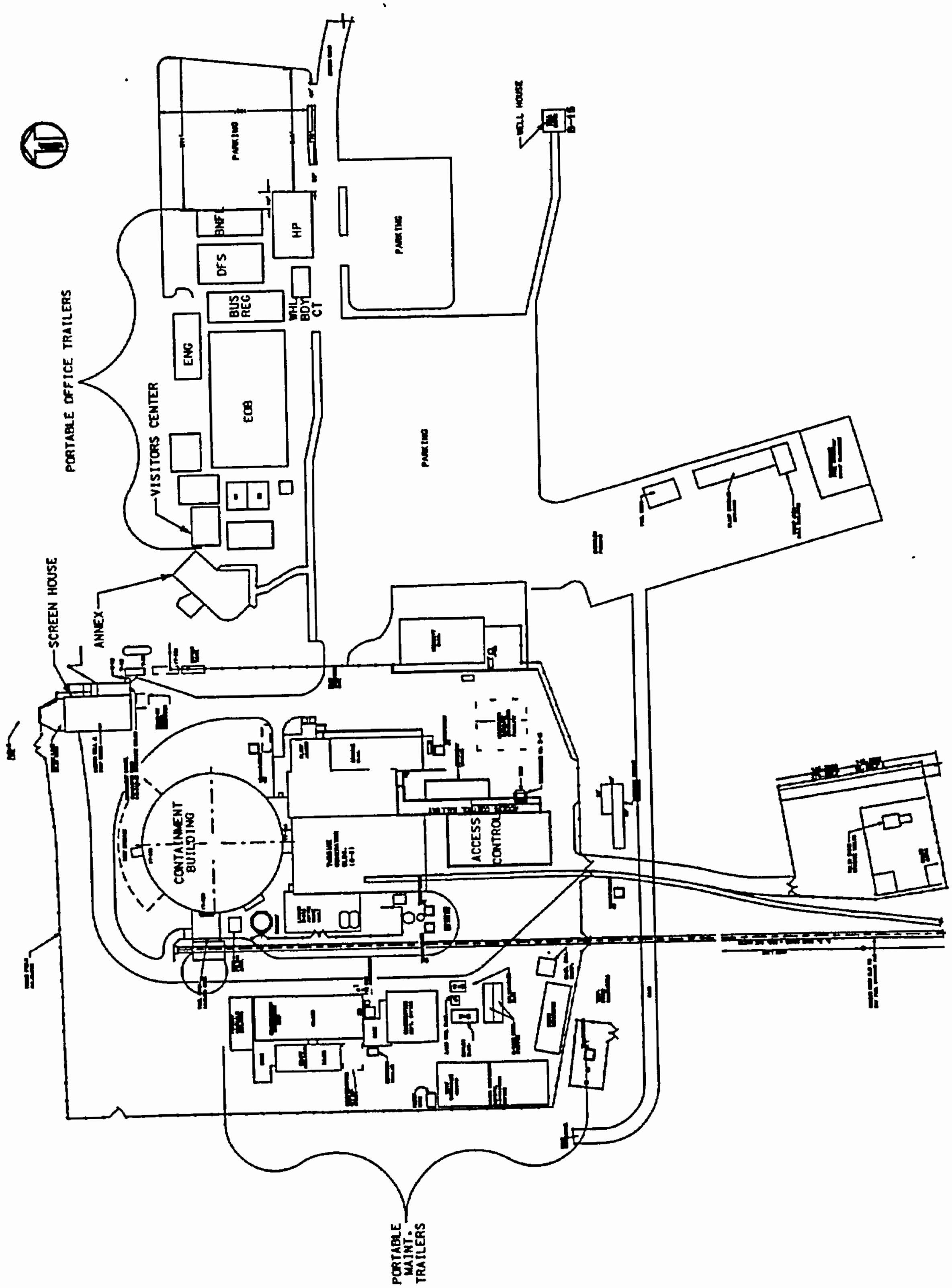


Figure 2.3
Plant Facility Identification



2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 LOCATIONS AND ROUTES

Figure 2.3.1-2 in BRP Volume 32, Environmental Report for Decommissioning, provides a listing of manufacturing plants in the five (5) mile radius of the Big Rock Point Plant.

These figures were extracted from the February 1984 HMM Document No. 83-600, Evacuation Time Estimates for the Big Rock Point Plant. The document was updated in 1993, refer to Reference 4.

Industrial activity in the vicinity of the Big Rock Point Plant consists primarily of small manufacturing companies. There is one cement plant and quarry in the area about six miles to the south-southwest.

Low level military training routes (VR-634 and VR-664) currently pass 10 miles from the Big Rock Point Plant. A former military low level training route (IR 600/601) a simulated radar bomb scoring range over Lake Michigan has been discontinued.

2.2.2 EVALUATION SUMMARY

The topic of Potential Hazards Due to Nearby Industrial, Transportation and Military Facilities was evaluated by the NRC as part of the Systematic Evaluation Program Topic Number II-1.C. This resulted in a safety evaluation (Reference 7) as follows:

2.2.2.1 Industrial Activity

Industrial activity in the vicinity of the Big Rock Point Plant consists primarily of small manufacturing companies. There are also some cement plants and quarries in the area. The closest industrial facility is a manufacturing plant located about one mile east where 105 employees (currently about 200) are engaged in producing custom molded plastic fixtures. An inventory of approximately 100,000 pounds of thermoplastic materials is stored at the facility. These materials are not an explosive hazard but could produce toxic combustion products if a fire should occur. The severity of this event with regard to safe operation of the nuclear plant, in particular, the habitability of the control room, would depend on many factors including source parameters, wind speed and direction, cloud plume rise, and protective actions taken by plant operators. (Control Room Habitability is addressed in Chapter 6 of this updated FHSR.)

An industrial park is located about 2.5 miles southwest of the plant. Several light manufacturing companies employing a total of about 200 persons are located in the park. No hazardous materials in quantities large enough to affect the safe operation of the nuclear plant are known to be processed, stored, or transported at the industrial park. An oil company storage terminal is located on US Route 31 near the industrial park. The maximum storage capacity at the terminal is approximately 46,000 gallons of fuel oil and 40,000 gallons of gasoline. The separation distance between the fuel storage terminal and the nuclear plant (over two miles) is considered adequate to preclude accidents at the terminal affecting the safe operation of the nuclear plant.

2.2.2.2 Transportation Activity

The nearest highway to the plant is US Route 31 (Refer to Figure 2.2) which is located 2,760 feet southeast at its closest point of approach. Shipments of explosives used in local quarry operations travel on Route 31 past the plant. The guidance of Regulatory Guide 1.91, Revision 1 was utilized to evaluate the consequences of a postulated explosive accident on the highway.

We find that the separation distance between the highway and the plant exceeds the minimum distance criteria given in the Regulatory Guide for truck-size shipments of explosive materials and, therefore, there is reasonable assurance that an explosive accident on the highway will not affect the safe operation of the plant.

We have also evaluated the potential consequences of highway accidents involving toxic chemicals. A conservative analysis indicates that certain toxic chemicals which form a gas cloud when released (eg, chlorine, ammonia) could reach the plant in concentrations high enough to be of concern depending on such factors as spill size and atmospheric dispersion conditions. Accident data compiled by the Michigan Department of Highways indicate that the expected frequency of an accident involving hazardous chemicals on the approximately ten-mile stretch of US Route 31 past the plant is about 1.3×10^{-3} per year. The percent of tanker truck accidents which involve a significant loss of material is about 2%. The percent of time on an annual basis that the wind blows from the ten-mile stretch of Route 31 toward the plant is about 51%. Thus, we conservatively estimate that the potential annual exposure rate to the plant due to toxic chemical accidents on Route 31 is about 10^{-5} per year.

The probability of toxic chemical exposure noted above is higher than the acceptance probability level used in current licensing criteria (see SRP 2.2.3). However, the calculated frequency of toxic chemical accidents on Route 31 past the plant is based on the assumption that the toxic chemical traffic on Route 31 is similar to that on other highways in Michigan. Our review of the industrial activity in the region surrounding the plant indicates a lack of industrial or chemical complexes which would generate toxic chemical traffic. Therefore, it is our judgement that the threat to the safe operation of the plant posed by highway accidents involving toxic chemicals is sufficiently remote so that such accidents need not be considered as a design basis event.

A former Chesapeake & Ohio Railroad branch line was approximately 5,600 feet south of the plant at its closest point. As explained in Section 2.1.2 of this updated FHSR, this line is no longer in use.

2.2.2.3 Pipelines

The nearest pipeline to the plant is a six (6) inch diameter natural gas line which is located about 1.5 miles south. At this distance, pipeline accidents will not affect the safe operation of the plant, based on evaluations of pipeline accidents done in previous licensing reviews. There are no gas or oil production fields, underground storage facilities, or refineries in the vicinity of the plant.

2.2.2.4 Waterways

There are no large commercial harbors near the plant but some commercial shipping does take place at Charlevoix Harbor which is approximately four miles southwest of the plant. While the great majority of the cargo consists of non-hazardous commodities such as coal and limestone, some gasoline and fuel oil is shipped from the harbor by barge to Beaver Island which is some 25 miles northwest of Charlevoix. Two barge line companies, each with one barge, are engaged in this trade. Between them, they make about 20 trips per year and the captains estimate that they come no closer than about three to four miles from the plant. Thus, the occurrence of a barge accident with consequences severe enough to affect the safe operation of the plant is extremely unlikely and does not constitute a credible risk to the plant. Similarly, the main shipping route in Lake Michigan which is located about 40 miles northwest of the plant is not a threat to plant operation.

2.2.2.5 Airports

The nearest airport to the plant is Charlevoix Municipal Airport which is located approximately five miles southwest. The airport has one paved runway 3,500 feet in length oriented in an east-west direction and two turf runways. Charlevoix Municipal is a general aviation facility used primarily by light single engine aircraft. There were a total of 16,800 itinerant and local operations at the field in 1976 and this is projected to increase to 71,000 operations in 1997 according to the airport master plan. The master plan recommends that Charlevoix Municipal Airport should be upgraded to a basic transport facility, i.e., one capable of handling turbojet powered aircraft up to 60,000 pounds gross weight. Using the analytical model given in SRP 3.5.1.6, we conservatively calculate the probability of an aircraft from Charlevoix Airport crashing into the Big Rock Point Plant is 8.5×10^{-7} per year. Conservatism in our calculation include the use of the projected 1997 level of operations, the assumption that all aircraft arriving or departing the airport fly over the plant area, and the consideration of the entire plant as a potential "target area". In fact, since the vast majority of aircraft operating at Charlevoix Airport are expected to be light, general aviation aircraft, only a small fraction of postulated aircraft strikes would seriously affect the safety of the plant. The probability of an accident resulting in severe radiological consequences would, therefore, be even lower than the probability value given above. We conclude that the Charlevoix Airport does not represent an undue risk to the safe operation of the nuclear plant.

2.2.2.6 Military Training Routes (Reference 8)

Military low level training routes (VR-634 and VR-664) pass approximately 10 miles from the Big Rock Point Plant.

In the Big Rock Point Spent Fuel Pool Expansion Hearings, the Atomic Safety and Licensing Board (ASLB) concluded "...that the evidence has demonstrated that the risk from aircraft to the Big Rock Point Plant is sufficiently low so that it need not be considered further in the design of the plant,....."

2.2.3 SAFETY EVALUATION CONCLUSIONS (Reference 7)

We conclude that the Big Rock Point Plant is adequately protected and can be operated with an acceptable degree of safety with regard to industrial, transportation, and military activities in the vicinity of the plant.

NOTE: Further support for the NRC Staff's conclusions pertaining to military, general aviation, and Charlevoix Airport cumulative realistic probability of an aircraft crashing into the plant can be found in Reference 8 and was about 2×10^{-8} per year in 1984 and has since been further reduced by the closing of military training routes (IR-600/601 and VR-1634).

2.3 METEOROLOGY

A "Meteorology Study of Natural Ventilation in the Atmosphere, Big Rock Point Nuclear Plant, Charlevoix, Michigan," Final Report was issued in December 1963 by the University of Michigan. This Report is contained in Volume Two of the original FHSR. This study includes collection and analyses of wind data - e.g., speed, direction, and turbulence, variability of these parameters with height, temperature lapse rates, and diffusion studies to determine the local effects of the lakeside location on air passing the site and wasdesigned to furnish that information which would be needed to accurately assess the general air flow and dilution potential of the air passing the plant site.

A 256 foot tower was built on the site to support the study and was instrumented to provide measurements of air temperature at six different levels and wind data at four different levels. In addition, the lake water temperature was measured. The report described the tower installation and summarized the wind data collected from February 1961 through January 1963, and provided typical annual variation of the mean water temperature at a depth of three feet in Little Traverse Bay and the mean daily maximum air temperature at a height of ten feet based on two years of data.

The general meteorological data available from the surrounding areas and the data collected during the two year study indicate that there are no factors which would produce significant limitations on plant operations. Specifically, the high average wind speed coupled with the relatively low percentages of calm conditions at the 256 foot level during most of the year indicate advantageous diffusion conditions would be prevalent a great deal of the time.

To further substantiate that advantageous diffusion conditions would exist much of the time, diffusion studies were initiated during the summer of 1961. These studies utilized the photography of smoke plumes released from the tower in an effort to obtain moderately accurate measurements of diffusion under the most adverse meteorological conditions. The smoke studies were intended to define the lower limits of diffusion capability at the site.

The "Smoke Plume Photography Study, Big Rock Point Nuclear Plant, Charlevoix, Michigan," Progress Report No. 3, was issued in December 1963 by the University of Michigan. This report is contained in Volume Two of the original FHSR.

Based upon the results of the two phase meteorological study made at the site, the annual average licensed stack discharge rate at one curie/second was the highest licensed rate for any reactor operating in 1964.

The 256 foot tower was subsequently removed and present meteorological monitoring is described in Section 2.3.2 below and in the Big Rock Point Site Emergency Plan.

Indications are that the normal meteorology of the site region will produce no significant limitations on plant design and operation. Generally prevailing winds are from the western half of the compass and there are no significant population centers, as defined in 10 CFR Part 100, within the 30 mile radius of the plant.

2.3.1 NORMAL AND SEVERE WEATHER

The topic of Severe Weather Phenomena was evaluated by the NRC as part of the Systematic Evaluation Program; this review resulted in a staff safety evaluation (SE) which assumed a licensing basis (Reference 9) for the following conditions.

Consumers Power Company reviewed the SE and the values selected by the NRC for extreme temperature, lightning strikes, snow and ice loads, and wind and tornado loadings have been verified against climatological data selected to be representative of site conditions. All parameters except the wind and tornado loading were verified against the climatological data recorded for the Pellston FAA weather station. Climatological data recorded for the Muskegon National Weather Service Station were used to verify the wind loading value. Current guidelines for estimating tornado and extreme wind characteristics were used to verify the tornado loading values. The results of the review are documented in Reference 10 and the CPCo conclusions follow each of the conditions from the NRC Safety Evaluation assumption.

2.3.1.1 Temperature (NRC-SE)

Big Rock Point Volume 32, Environmental Report for Decommissioning, contains information on normal and extreme temperatures.

The extreme maximum and minimum temperatures appropriate at the Big Rock Point site for general plant design (i.e., HVAC systems) are 86 degrees Fahrenheit (equaled or exceeded 1% of the time) and -6 degrees Fahrenheit (equaled or exceeded 99% of the time).

(CPCo Verification)
The extreme maximum and minimum temperatures of 86°F and -6°F selected by the NRC are appropriate.

2.3.1.2 Thunderstorms and Lightning Strikes (NRC-SE)

BRP Volume 32, Environmental Report for Decommissioning, contains information on severe weather.

Based on the annual number of thunderstorm days, the calculated annual flash density of ground lightning strikes is four flashes per square kilometer. A structure with the approximate dimensions of the Big Rock Point Reactor Building can be expected to be subjected, on the average, to one strike every seven years.

(CPCo Verification)
The NRC Values of four flashes per square kilometer and one strike every seven years are reasonable.

2.3.1.3 Hail Storms, Freezing Rain, and Ice Loading (NRC-SE)

On the average, hail storms occur about two days annually, and freezing rain occurs approximately twelve days per year. The maximum radial thickness of ice expected in the site region is about 0.75 inch.

(CPCo Verification)
These values are consistent with values determined for our Midland Plant Site and are acceptable.

2.3.1.4 Snowfall and Snow Load (NRC-SE)

BRP Volume 32, Environmental Report for Decommissioning, contains information on snowfall.

Based on the 100 year recurrence accumulated ground snowpack and the probable maximum winter precipitation for the site region, the normal winter precipitation snow load on a flat surface is about 50 pounds per square foot and the extreme winter precipitation snow load on a flat surface is 115 pounds per square foot.

(CPCo Verification)
CPCo agrees with the NRC selected value of 115 lb/ft^2.

2.3.1.5 Design Wind Speed (NRC-SE)

BRP Volume 32, Environmental Report for Decommissioning, contains information on wind.

The design wind speed (defined as the "fastest-mile" wind speed at a height of 30 feet above ground level with a return period of 100 years) acceptable for the site region is 80 miles per hour.

(CPCo Verification)
BRP original design criteria for most buildings was approximately 87 MPH. The value of 80 miles per hour will be considered for future design as practicable within the constraints of existing plant design and considering the improvement in terms of its effect on overall Plant safety.

2.3.1.6 Tornadoes (NRC-SE)

BRP Volume 32, Environmental Report for Decommissioning, contains information on severe weather.

Tornadoes have been reported 25 times during the period 1950-1977 within an approximate 60-mile radius from the Big Rock Point Site, excluding the water area over Lake Michigan. On the average, one tornado can be expected to occur in the vicinity of the Big Rock Point Site every year. Based on the tornado characteristics for the site region and the probability calculations outlined in WASH-1300, the recurrence interval for a tornado at the site is calculated to be about 5150 years.

The assumptions used in Regulatory Guide 1.76 provide an adequate design basis tornado for the site region. These characteristics include a maximum windspeed of 360 miles per hour (a maximum rotational windspeed of 290 miles per hour plus a maximum translational windspeed of 70 miles per hour), a maximum pressure drop of three pounds per square inch, and rate of pressure drop of two pounds per square inch per second.

Based on actual tornado occurrences in the site region area and using the procedures discussed in WASH-1300, a "site-specific" design basis tornado (with a probability of occurrence of 10^{-7} per year) can be calculated. For the Big Rock Point Site, the characteristics of tornadoes occurring within a 60-mile radius are a maximum windspeed of 310 miles per hour (a maximum rotational windspeed of 250 miles per hour plus a maximum translational windspeed of 60 miles per hour), a maximum pressure drop of two pounds per square inch, and rate of pressure drop of one pound per square inch per second. Because of the infrequent occurrences of tornadoes in the site region (19 tornadoes with available data), the site-specific tornado characteristics are based on a very small sample of data which we believe does not provide a reasonable degree of accuracy for calculations of safety-related structure design.

(CPCo Verification)
As previously stated in our letter of January 23, 1981, design basis tornado parameters from Regulatory Guide 1.76 are not consistent with either the recorded tornado frequency and intensity data for the site region or with the current state of knowledge on tornado and extreme wind characteristics. More current guidance for the characteristics of a design basis tornado for the site region suggest the following characteristics:

1) Maximum wind speed of 250 mph (combined rotational and translational).

2) Maximum translational wind speed of 55 mph.

3) Maximum pressure change of 1.35 psi.

These design basis tornado characteristics are more representative of the site and will be used instead of the Regulatory Guide 1.76 design basis tornado characteristics. Since the lake shore environment of the Big Rock Point site exerts an additional moderating influence on severe storm intensity which has not been taken into account, the above parameters are still considered to be conservative.

2.3.1.7 Severe Weather Conclusions

(CPCo Conclusions) (Reference 12)

For the specific case of Big Rock Point, a tornado wind speed value with a probability on the order of approximately 10^{-5} would be appropriate to ensure that the risk from the single event is small compared to other risk contributors. If it were assumed that a tornado wind in excess of this value would result in core damage, a very conservative assumption in itself, then tornadoes would still represent only a small percentage of the total residual core damage probability. This is certainly true now, and will remain true after anyother planned plant modifications are complete. The cost associated with analyses of lower probability conditions more extreme than these are simply not warranted for a plant of small core size like Big Rock Point.

Analyses of Big Rock Point structures have been completed (refer to Section 3.3 of this Updated FHSR). These analyses were performed assuming a tornado wind speed of 250 mph. Although this value is very conservative in view of the discussion, it was selected some time ago because it was the 10^{-7} wind speed value determined by our tornado analysis. This wind speed also corresponds to a probability of approximately 2×10^{-7} in McDonald's work. It is our intent to continue using 250 mph for wind speed. In the event that specific structures are identified which cannot withstand this wind load, then lower values may be selected for further structural evaluations. If the wind-induced failure (below 250 mph) of important structures from gross loading or missiles becomes significant with respect to other risk contributors, then these structures will be evaluated using the PRA during the Integrated Assessment.

2.3.2 METEOROLOGICAL MONITORING

Meteorological data may be obtained from the National Weather Service (NWS) through the automated computer system of Weather Services International (WSI). Access to this system is by telephone or data transfer via the computer in the Emergency Support Center (ESC).

2.3.3 ATMOSPHERIC TRANSPORT AND DIFFUSION ESTIMATES

An evaluation of Systematic Evaluation Program Topic II-2.C, Atmospheric Transport and Diffusion Characteristics for Accident Analysis, (Reference 13), was completed by CPCo April 6, 1982. The objective of this topic was to review atmospheric transport and diffusion characteristics utilized to demonstrate compliance with 10 CFR 100 guidelines with respect to plant design, control room habitability and doses to the public during and following a postulated design basis accident.

Criteria

10 CFR 100 requires that as an aid in evaluating a proposed site, the applicant should hypothesize a fission product release (generally assumed to be a result of a substantial meltdown of the core with subsequent release of appreciable quantities of fission products) from the core, the maximum expected leak rate from the containment and the meteorological conditions pertinent to the site. The total dose to an individual at the boundary of the exclusion area over the first two hours after this hypothesized event must be less than 25 rem to the whole body or 300 rem to the thyroid. Also, the NRC Standard Review Plan (SRP) items of potential hazard from industrial, military and transportation facilities should be evaluated and analysis of the consequences to the plant personnel of accidents involving these facilities should be evaluated. Further, the SRP requests the meteorological data and models used to determine these consequences be presented. Other pertinent guidance is provided in Regulatory Guide 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Boiling Water Reactors and 1.145, Atmospheric

Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

Criteria for Permanently Defueled Plant Analysis

For the permanently defueled plant, accidents involving reactor operation are not feasible. As described in Chapter 15 of this UFHSR the bounding event for the permanently defueled plant is a hypothesized heavy load drop in the spent fuel pool. The assumptions used in performing the analysis of this event include:

- Offsite release occurs over a 2-hr interval, per Regulatory Guide 1.25 [Reference 15.10.1-4].
- No credit for containment ventilation isolation is taken.
- X/Q is 6.48E-04 sec/m^3 for the ground level release, per Regulatory Guide 1.25 [Reference 15.10.1-4] for dose to offsite population (closest site boundary, 805 meters).
- Dose conversion factors are from EPA-400 and EPA-402 [References 5.1-3 and 5.1-5].
- Ground level release results in higher offsite doses, thus has been assumed in calculation of doses to the public.

Big Rock Point implemented the guidelines of the EPA Manual of Protective Action Guides (PAGs) and Protective Actions for Nuclear Accidents, EPA-400 [Reference 15.10.1-3] on January 1, 1994. EPA-400 establishes protective action levels for public protection at one rem total effective dose equivalent (TEDE) for the total body, five rem committed dose equivalent (CDE) for thyroid, and 50 rem skin dose equivalent (SDE) for skin. These doses are small fractions of the limits established in 10 CFR Part 100. Dose calculations reflecting plant decommissioning and dismantlement events as described in Chapter 15 of this UFHSR have been performed in accordance with the guidelines of EPA-400.

For the purposes of comparison with previous calculational methods the following discussion presented previously in this Section is retained.

Summary of Previous Analysis Methods

Transport of airborne radioactivity from the Big Rock Point Site has been calculated by several different means over the past 20 year of plant operation. Briefly, the techniques and reference documentation for each are as follows:

1. Siting criteria calculations - Atmospheric diffusion based on Sutton's method for analyses of onsite preoperational meteorology data. Documented in Sections 13 and 14 of the Big Rock Point Final Hazards Summary Report, November 14, 1961.

2. Current safety analyses, including Emergency Plan and Emergency Implementing Procedure calculations - Atmospheric diffusion parameters from Regulatory Guide 1.3, assuming ground level or elevated release, dependent upon release mode.

3. Environmental dose calculations for 10 CFR 50, Appendix I - Regulatory Guides 1.109 and 1.111 were utilized for computation of doses from elevated releases, based on onsite meteorology data collected February 9, 1961 through February 8, 1962. Bases and results of these calculations are presented in Consumers Power Company (CPCo) letter dated June 4, 1976.

4. Offsite consequences of accidents, Probabilistic Risk Assessment - Preoperational meteorology data was utilized in accordance with the methods of WASH-1400. Results of the consequence analyses were submitted by CPCo letter dated March 31, 1981.

Discussion

An evaluation of X/Q (Note 1) values at the Big Rock Point Plant was presented in Section 14 of the November 14, 1961 Big Rock Point Final Hazards Summary Report (FHSR). As described in Section 14, a meteorological tower was constructed on a point of land at the shore of Lake Michigan about 2,000 feet to the WNW of the stack. Trees in the surrounding area were removed. The area was chosen so that the measured data would be most accurate for winds blowing toward the Harbor Springs - Petoskey and Charlevoix areas. Hourly data was taken from November 1960 to February 1962. Wind direction was obtained from 36 points (0 to 360). Wind direction and speed were obtained from sensors located at 32 feet, 64 feet, 128 feet and 256 feet. Temperature data was obtained at 3 feet below the surface of the water, 10 feet, 50 feet, 100 feet, 150 feet, 200 feet and 250 feet above the surface. The data was analyzed using a computer program and hourly values of X/Q were obtained.

The data has since been used in three ways. First, Section 13 of the November 14, 1961 FHSR (Maximum Credible Accident) used four selected points in the atmospheric diffusion spectrum which encompass the conditions encountered at the site. Atmospheric diffusion methods of Sutton were used for the neutral and unstable cases and Hanford diffusion results (Report HW-54128) were used for inversion cases. These were compared with site data and found to be conservative. Radiation doses at the site boundary and beyond were calculated using the stated diffusion methods. The worst case X/Q at the site boundary for a ground level release was found to be 4E-04 sec/m^3. This compares with Regulatory Guide 1.3 values of 6E-04 sec/m^3 for 0-8 hours, 2.2E-04 for 8-24 hours, 8E-05 for 1-4 days and 1.7E-05 for 4-30 days. Since the radiation doses at the site boundary are very much below the limits given in 10 CFR 100 the actual difference between 4E-04 and 6E-04 is not significant with respect to meeting 10 CFR 100 limits.

The second use of the meteorological data was in the Big Rock Point Probabilistic Risk Assessment (PRA), submitted to NRC by CPCo letter of March 31, 1981. Doses to the public from dominant sequences were calculated using a variety of meteorological conditions with the CRAC code (same methodology as WASH-1400). The conditions were chosen using the sampling technique of WASH-1400. The values for X/Q were not listed in the output of the CRAC code. However, previous analyses of the meteorological tower data show that the worst case X/Q (worst 2-hour interval calculated in accordance with Regulatory Guide 1.145) at the site boundary, 7.5E-04 sec/m^3 is almost the same as that used in the FHSR. Control Room Habitability with regards to external events was also presented in the PRA. Habitability was demonstrated by showing that the operator could isolate the control room ventilation system prior to intake of excessive quantities of toxic gases, smoke, etc. Also, the probability of these events occurring along with the proper meteorological conditions and ventilation failure was small ($<10^{-4}$/yr).

The third use of the meteorological tower data was in the CPCo submittal of June 4, 1976 concerning 10 CFR 50, Appendix I. The meteorological data was used to obtain X/Q and D/Q (Note 2) values, wind roses, monthly and yearly joint frequency distributions, and an annual average X/Q. The methodology used was in accordance with Regulatory Guide 1.111. This data was then input into the GASPAR computer code for radiation dose calculations. The maximum annual average X/Q for an elevated release was found to be 2.5E-07 sec/m^3. This occurred in the East sector at 2414 m from the stack. Additional data may be found in Table 3.1 of the Appendix I submittal dated June 4, 1976.

CPCo Conclusions

Because the radiation doses calculated at the site boundary are small, the demonstration of compliance with 10 CFR 100 limits is not particularly sensitive to the X/Q values used. Consumers Power Company's intent is to continue with the use of onsite preoperational data for realistic analyses performed for PRA and environmental dose purposes. For all other calculations, Regulatory Guide 1.3 values will be used. Assuming a ground level release for all unknown accident conditions, the following values X/Q are applicable at 0.5 miles. Exclusion Area Boundary and Low Population Zone (EAB and LPZ):

0-8 hours	6.0 E-04
8-24 hours	2.2 E-04
1-4 days	7.4 E-05
4-30 days	1.8 E-05

(X/Q Note 1)	X =	the short term average centerline value of the ground level concentration (curie/meter^3)
	Q =	amount of material released (curie/sec)
(D/Q Note 2)	D =	Desposition Constants
	Q =	amount of material released

Evaluation Summary

The topic of Atmospheric Transport and Diffusion Characteristics for Accident Analysis - Big Rock Point Plant was evaluated by the NRC (Reference 14). This revised final evaluation of SEP Topic II-2.C.

On June 23, 1982, the staff issued an SER on Topic II-2.C, which was based on Consumers Power Company evaluation submitted by letter dated April 6, 1982. The Staff SER derived the X/Q values at the outer boundary of the low population zone (LPZ) based on the minimum distance (805 meters) of a variable outer boundary as defined in the Licensee's submittal of April 6, 1982. The actual LPZ boundary is 2.5 miles (4023m) and the LPZ X/Q values have been recalculated based on this distance.

The evaluation was done using the meteorological diffusion parameters described in Regulatory Guide 1.3 since no meteorological observations have been made outside in twenty years. The staff has confirmed that the Regulatory Guide 1.3 parameters are conservative. This confirmation was done with the methods of the guide as described below and resulted in the values given in Table 1 at the 805 meter exclusion area boundary and at the 2.5 mile (4023m) re-defined distance to the outer boundary of the low population zone distance.

Table 1: Relative Concentration at Big Rock Point

Time	Relative Concentration X/Q sec/m^3
0-2 hours EAB	6.7×10^{-4}
0-8 hours LPZ	8.0×10^{-5}
8-24 hours	1.7×10^{-5}
1-4 days	5.5×10^{-6}
4-30 days	1.2×10^{-6}

These values are derived from Figure 3A with the application of the building wake dispersion correction factor in Figure 2 of Regulatory Guide 1.3 for a 500 square meter building surface area. The above values should be used for all contained release accidents.

Conclusion

The Staff concludes that the X/Q values presented in Table 1 are appropriate for estimating exposures from postulated accidents and should be used in all but steam line break accident calculations. This provides a conservative assessment compared to the use of methods in conjunction with Regulatory Guide 1.145, which is the basis of current review for new licenses.

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

2.5.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

The following Geology and Seismology descriptions were extracted from the 1961 Final Hazards Summary Report and are reported in this section. Newer analyses have been completed since that time, and are reported in subsequent sections of this report.

Geology

Professor James H Zumberge of the University of Michigan was retained as a consultant on the geology and hydrology of the reactor site and its environs. His findings are reported in Volume Two of the 1961 FHSR.

Seismology

The seismicity of the site was investigated by Professor James T. Wilson, Professor of Geology, University of Michigan, who was retained as a consultant for this purpose, and his findings are attached in Volume Two of the 1961 FHSR. The probability that earthquakes of significant intensity will occur in the general site area appears to be very low.

The importance of earthquakes to plant design was independently investigated by the Bechtel Corporation. Their summary statement of findings is:

"An investigation of the seismic history indicates that this is a region of low seismic activity. The Coast and Geodetic Survey Publication, Serial 609, Earthquake History of the United States, lists earthquakes in the Michigan area as shown below. All of these are classified as intermediate or minor. The nearest recorded earthquake was the one centered near Menominee, approximately 110 miles from the plant site."

Earthquake history is found in BRP Volume 32, Environmental Report for Decommissioning.

Since no recorded earthquakes have centered near the plant site, and there is no knowledge of earth tremors having been felt near the site, elaborate or special seismic design features were not considered necessary. However, in keeping with good engineering practices, all structures are designed to resist nominal seismic loading. Structural design of the plant complies with the Uniform Building Code (UBC). Horizontal forces based on Zone 1 are used.

The UBC does not clearly cover the reactor containment vessel or the concrete structure and equipment within. In view of their high degree of rigidity, it appeared prudent to use a seismic factor equal to the maximum expected ground acceleration at the site. A study of the brief earthquake history of the region led to the conclusion that an intensity of 7 on the Rossi-Forel scale was a reasonably conservative assumption. This corresponds roughly to a ground acceleration of 0.05 gravity. Therefore, a seismic factor of 0.05 was used for this portion of the plant. This is twice the factor required by the UBC for tanks and similar structures, and appears to be reasonable in view of the high rigidity already mentioned.

For the containment vessel itself, earthquake forces do not govern the design, since the wind force on the vessel at the design velocity of 100 miles per hour is greater than 0.05 times its weight.

2.5.1.1 Regional Geology

Regional Geology in BRP Volume 32, Environmental Report for Decommissioning, was extracted from the NRC assessment of Systematic Evaluation Program Topic II-4 (Reference 20).

2.5.1.2 Site Geology

Site Geology was extracted from (Reference 20) the NRC assessment of Systematic Evaluation Program Topic II-4 and is contained in BRP Volume 32, Environmental Report for Decommissioning.

The water table varies seasonally, but is usually several feet above the normal level of Lake Michigan.

The till and massive bedrock beneath the site are competent foundation materials, however, the Gravel Point limestone is susceptible to solutioning. In northeastern lower peninsula Michigan, karst topography is well developed in the Devonian limestones. This may be due to the relatively thin cover of glacial deposits in that area. In the site area solution features are more subtle and apparently far less common, but several significant features have been found. A more detailed discussion of limestone solutioning is included in Section 2.5.1.3.

Other than the slight possibility of cavernous conditions beneath the site, there are no geologic hazards at this site.

2.5.1.3 The Potential for Subsidence or Collapse Due to Solutioning

During the NRC Review of Systematic Evaluation Program (SEP) Topic II-4.B, Proximity of Capable Tectonic Structures in Plant Vicinity, two concerns were identified (Reference 20):

1. The possible existence of a large cavern under the site that could ultimately cause subsidence or collapse.

2. The possibility of the development and enlargement of a new cavern during the life of the plant.

The bases for the concerns were: 1) the existence of three large sinks and an open cavern in the Penn-Dixie and Medusa quarries, which are located eight miles to the east and several miles to the southwest respectively; 2) the susceptibility to solutioning of the Traverse Group limestones which comprise the site bedrock; 3) the karst-like topography of the rock surface offshore beneath Little Traverse Bay where there is little or no soil cover; and 4) poor rock recovery in the original site exploratory borings and the discovery in three recent borings of a buggy zone between 130 and 190 ft depths.

In their report entitled "Solution Features in the Traverse Group of Northwestern Michigan" (Harding-Lawson Associates, geologist consultants for Consumers Power Company), presented data supporting their conclusion that extensive solutioning is not going on in the site area at the present time, nor has it likely been for the past several thousand years. The evidence cited includes: 1) the sink present in the quarries are filled with undisturbed glacial deposits including sand, gravel and till; thus dating the solution holes as being at least Late Pleistocene age; 2) the open cavern in the Penn-Dixie quarry had been bridged by 60 to 80 feet of rock before excavation and was well below the present level of Lake Michigan, indicating that it probably formed when the level of the Lake was much lower than it is today; 3) movement of groundwater through the rock, related to the wide range of fluctuation of the surface of ancestral Lake Michigan and the local groundwater surface have been relatively stable since the lake reached its present level after the close of the Pleistocene; 4) the site region is covered by a blanket of relatively impermeable soil, causing most precipitation to run off rather than percolate down and move through the rock; 5) extensive karst topography is not apparent at ground surface in the site area.

Based on the evidence available to date, it is not likely that significant solution activity is going on in the rock beneath the site, nor is it likely that there are large caverns beneath the site sufficiently close to the surface to cause subsidence or collapse beneath the plant, as indications of this condition would probably have already been observed during or shortly after construction twenty years ago. However, because of the scarcity of information on the condition of site bedrock it was considered prudent to perform additional studies to confirm its competency.

The additional studies were completed and the results and conclusions on these concerns were addressed in (Reference 21) as follows:

CPCo contracted with Commonwealth Associates, Inc (CAI) of Jackson, Michigan, to investigate the possible existence of solution cavities beneath the plant. CAI reported its conclusions in the report "An Investigation Into the Possible Existence of Solution Cavities Beneath the Big Rock Point Nuclear Power Plant Near Charlevoix, Michigan", February 1983. In that report the consultant concluded that the geological processes that created solution features in the area have not been active since the last episode of glaciation, and there is insufficient information to confirm either the presence or absence of cavities beneath the site.

Evaluation Summary Conclusion

On the basis of the evidence available to date, it is not likely that significant solution activity is going on in the rock beneath the site, nor is it likely that there are large caverns beneath the site sufficiently close to the surface to cause subsidence or collapse beneath the plant, because indications of this condition would probably have been observed during or shortly after construction 20 years ago. The Staff concludes that there is insufficient benefit to be gained from conducting additional onsite investigations; therefore, no further action is required.

One other concern raised during SEP Topic II-4.B review (Reference 20) was the possibility of subsidence and collapse due to the dissolution of salt at depth beneath the site. Wold (1980), based on the examination of the available seismic reflection profiles in Lake Michigan interprets the presence of faults, which he attributes to collapse structures formed by the dissolution of salt within the zone of outcrop of Middle Silurian (445 mybp) through Middle Devonian (360 mybp) strata. The site lies within this zone. Based on NRC review, they don't consider this phenomenon to represent a hazard to the site because:

1. the site is underlain by a relatively thick section (400/500 feet) of Upper Devonian rocks with little or no salt deposits (based on studies by Dr T Buschbach of outcrops, quarries, hydrocarbon exploratory borings, and water well logs); and

2. the section of rocks that are of concern, in addition to being overlain by a thick sequence of Upper Devonian rocks, are also overlain by 40 feet of glacial deposits. There is no apparent evidence of collapse features at depth in the glacial soil at the site.

Evaluation Summary Resolution

Salt deposits lie at depth beneath the site. It has been postulated that inferred faults in Lake Michigan are the results of collapse due to dissolution of salt. We conclude that this phenomenon doesn't present a hazard to the plant because of thick limestones over the salt deposit, and there is no evidence of it having occurred in at least the last 10,000 years in the Pleistocene soils that cover rock in the site area.

2.5.2 VIBRATORY GROUND MOTION

As discussed in Section 2.5.1 above, the probability of earthquakes of significant intensity to provide vibratory ground motions which would cause major damage at Big Rock Point is very low. As a result of the Systematic Evaluation Program (SEP), (Reference 28) the seismicity of the Big Rock Point vicinity has been recently reviewed by experts employed by the NRC, the SEP Owners Group and by Consumers Power Company (see NUREG/CR-1582 and "Eastern United States Tectonic Structures and Provinces Significant to the Selection of a Safe Shutdown Earthquake," Weston Geophysical, August 1979). Based on approximately 200 years of reasonably reliable earthquake history and the known geological and tectonic structure of the area, the experts seem to agree that a design basis earthquake with a return period of one to ten thousand years would be 0.05 to 0.07 g. Earthquakes of this size do not cause major damage to even poor quality construction.

If, in addition to the above, a minimum design earthquake is assigned for the entire eastern United States without regard to structure or location, the design earthquake increases as in Attachment 1 to the August 4, 1980 NRC letter to approximately 0.10 g. Typical industrial construction is not usually damaged by this level of earthquake. Steel and reinforced concrete construction as used at Big Rock might, at worst, suffer minor cracking.

Finally, preliminary calculated results from the Big Rock structural evaluation indicate that major structural elements of all safety-related structures will remain below code allowable stress when subjected to an 0.11 g earthquake of the type shown in Attachment 1 to the August 4, 1980 letter.

In summary, earthquakes are not very probable at Big Rock Point. Even for long return periods, the earthquake is not predicted to be large enough to cause major damage to quality industrial construction. Preliminary calculations for Big Rock structures show no significant damage occurs to the structures from earthquakes of the size proposed in your letter. Independent work being done for the Big Rock Probabilistic Risk Assessment indicates very long return periods for earthquakes of this size. We conclude that continued operation of the Plant while the seismic analysis is completed is entirely acceptable for the above enunciated reasons.

Summary of Seismic Design Considerations

A summary of the Big Rock Point seismic resistance from Systematic Evaluation Program (SEP) Topic III-6 Seismic Design (Reference 27) is provided below:

The initial seismic criteria as applied to Big Rock Point were based on static requirements of the 1958 edition of the Uniform Building Code. The containment design was based on a 0.05g horizontal static coefficient. The turbine building, concrete stack, intake structure, control room and rad waste storage buildings were designed based on a 0.025g horizontal static coefficient. Piping design for seismic resistance was limited to the reactor vessel supports and NSSS major piping. These components incorporated a 0.05g and a 0.025g horizontal static coefficient in the respective designs. The RDS was designed in 1974 in accordance with seismic design requirements as they existed at that time. These compare with more recent requirements which assume a 0.12g (Reg Guide 1.60) safe shutdown earthquake. The Alternate Shutdown Panel Building design and electrical conduit for alternate safe shutdown also utilized the 0.12g (Reg Guide 1.60) safe shutdown earthquake requirement.

A complete review of the seismic design adequacy of the Big Rock Point Plant was initiated by the NRC Staff early in 1979 as a part of Systematic Evaluation Program Topic III-6. Plans were developed by Consumers Power Company and submitted April 25, 1979 with respect to important structures which were to be analyzed. The staff requested that major portions of the primary coolant loop be included in this initial structural analysis in July 1979. Initial structural analyses employed Reg Guide 1.60 Spectra (anchored at .12g) while awaiting staff approval of a site specific seismic response spectra. Preliminary results from analysis of 15 major site structures plus the primary coolant loop were submitted January 9, 1981 with the final report (by D'Appolonia) published August 26, 1981.

In July 1979 (IEB 79-14), the staff required all Licensees to verify that the configuration of safety-related piping systems corresponded to that assumed in the plants existing design analysis. This activity resulted in an inspection of approximately 6000 feet of safety-related piping at Big Rock Point including examination of pipe geometry, support design, embedments, attachments and valve location and orientation. Results associated with this activity were published in October 1979, and were to be used eventually as input to the piping design review associated with SEP Topic III-6.

In January 1980, the staff published a formal request for the immediate identification and evaluation of important electrical equipment and its anchorage. As a part of the request, auxiliary failures which could result in the disabling or failure of safety related equipment (such as gas bottles, dollies, etc) were to be identified and evaluated as well. This Systematic Evaluation Program work resulted in the identification, analysis, and anchorage of over 50 equipment items. Among the major equipment important to safety were motor control centers, distribution panels, batteries and transformers. As requested, auxiliary equipment was also evaluated and included tanks, containers, cabinets and lighting located in the vicinity of important safety equipment. The majority of the electrical equipment anchorage work was completed by March, 1981.

In April, 1981 the staff requested a firm schedule for completion of seismic design review activities. Included in their request were not only the primary coolant loop but verification of fluid and electrical distribution system integrity and analyses of the integrity and functionability of important mechanical and electrical equipment. Also requested was justification for continued operation while the additional work was in progress. At this time the cost of evaluating this single SEP Topic was well in excess of one million dollars and had at least as much evaluation and analysis awaiting completion as had been accomplished to date. In addition, work was ongoing in the development of resolutions to NRC questions raised with respect to work submitted to date. As part of its justification submitted June 19, 1981, Consumers Power Company questioned the benefits of such an extensive, deterministic based reevaluation of the Big Rock Point structural design. Referenced were the results of the Big Rock Point risk assessment published in March, 1981 which suggested that seismic concerns represented only a small contribution to the total risk of operation. Consumers Power Company proposed the detailed analyses completed to date used in conjunction with augmentation of the Probabilistic Risk Assessment (PRA) arguments would demonstrate a basis for concluding that seismic risk at Big Rock Point was small compared to other contributors, and that further deterministic analyses were not necessary.

In a site visit on June 30, 1981, the Staff insisted that the deterministic approach was necessary and that the proposal to use risk assessment as a basis for continued operation had little promise of working. Consumers Power Company submitted a plan for future evaluations with respect to SEP Topic III-6 on July 27, 1981 and on September 29, 1981, the Staff concluded that our plan and justification for operation in the interim were acceptable. Justification was based on analysis of plant structures and systems performed to date, apparent inherent seismic resistance of remaining systems and structures, and thelow seismic hazard associated with the Big Rock Point site.

In April 1982, as a part of its review of Consumers Power Company seismic evaluations that have been completed, the staff raised questions with respect to soil properties assumed in these analyses. This placed into question the adequacy of the Reg Guide and site specific spectra used in the analyses. In August 1982, work explicitly aimed at analysis of piping and equipment was suspended (except for model development) while these uncertainties were resolved.

On October 19, 1982, the Staff issued a draft Safety Evaluation Report (SER) with respect to the status of the seismic reevaluation of Big Rock Point. This report identified several areas of concern that the staff had with respect to the appropriateness and completeness of analyses performed to date. As a result, the staff stated that they were unable to come to a conclusion with respect to the seismic capability of the Big Rock Point Plant. They did conclude, however, that they existed inherent seismic resistance in the design of the plant, that operation was justified in the interim while the Integrated Assessment was performed and that alternate approaches to resolving this topic should be investigated.

A meeting was held with the staff in December, 1982 in which Consumers Power Company was encouraged to respond to the staff comments presented in the October SER. The Staff concluded that because of the significant cost of continuation of the seismic analysis program it was recommended that Consumers Power Company consider and propose alternate approaches. These approaches could include bounding analyses with selected plant upgrading assessments of the consequences of failures, comparison of probabilistic risk and representative cross-sections of current plants, or combinations thereof. The resulting approaches would be considered in the Big Rock Point Integrated Plant Safety Assessment.

In June, 1983, explicit response to the Staff's concern in their draft SER were provided in addition to the alternatives Consumers Power Company was proposing for final resolution of this SEP Topic. The alternatives included a comparison of the risks associated with Big Rock Point Plant consequences on the health and safety of the public in comparison with a newer typical facility, as the staff suggested. Also an approach to identifying, evaluating and upgrading the seismic "weak-links" at Big Rock Point was presented with explicit results. Commitments were made to upgrade the report to more completely identify the perceived weaknesses associated with the plant design, if the staff approved of the approach.

In September and November of 1983 the Staff and Consumers Power Company presented joint testimony before the Advisory Committee on Reactor Safeguards in regard to the alternate "weak-link" approach. The ACRS was requested to comment on the appropriateness of the proposed approach. In their testimony the staff concluded that the "weak-link" approach was prudent and correct for Big Rock Point. They intended to monitor its implementation in the form of analyses and backfits before concluding as to the level of protection afforded by the plant design against seismic events. Preliminary conclusions by ACRS members indicated that it was not necessary to get Big Rock Point up to the level of a new plant and that the "weak-link" approach was appropriate.

In May 1984, the final Integrated Plant Safety Assessment was published by the staff (Reference 21). In that report, both the staff and the ACRS conclude that the proposed "weak-link" approach is appropriate and that they will continue to monitor its implementation.

NRC Evaluation Conclusions (Reference 21)

The following was extracted from NUREG-0828, Final Report May 1984, Section 4.12 and supports the evaluation above.

During its topic evaluation, the Staff concluded that the criteria and analyses supplied by the Licensee for structures, buried piping, and portions of the reactor coolant loop piping were not adequate to resolve questions concerning analytic uncertainty or to quantify the effects of simplifying assumptions. The seismic analyses performed to date are not in accord with either Systematic Evaluation Program (SEP) or Standard Review Plan (SRP) current criteria. The Licensee has indicated that it is not economically feasible to perform the analyses required to demonstrate seismic capability and quantify analytical uncertainty. The Staff agrees that considerable detailed analysis would be required. As an alternative, the Licensee has proposed to evaluate the seismic resistance of equipment important to safety using a combination of probabilistic methods and deterministic analyses...

The Staff concurs with the Licensee's proposed approach to selective seismic upgrading. The original design of Big Rock Point included a static horizontal load for structures. The seismic analyses performed under Topic III-6 have demonstrated that there is inherent seismic resistance in the design; however, to complete the analysis and any modifications necesary to demonstrate a consistent seismic capability for all safety-related equipment and structures would be very time consuming and expensive because of the lack of original seismic design analyses, the complex nature of the "as-built" plant, and (in some cases) lack of original construction details needed to perform seismic analyses. The offsite dose analyses performed in conjunction with SEP Topics and the Licensee's PRA have demonstrated that the relative consequences of accidents, even those involving core melt, are very low because of the small plant size and low population distribution around the plant site.

In view of these considerations, the Staff concludes that the approach proposed by the Licensee (i.e., to selectively upgrade the "weak links" in the system and structures necessary to mitigate accidents that would be expected to result from seismic events) is reasonable and, if properly executed, would provide sufficient seismic resistance so that the health and safety of the public could be ensure.

2.5.2.1 Response Spectra

Various seismic design Response Spectra have been used in the Systematic Evaluation Program to demonstrate the seismic design adequacy of Big Rock Point:

- In the August 4, 1980 NRC letter, the preliminary seismic input ground response spectra recommended for use in the interim until the final NRC Staff decision on Site Specific Spectra at SEP sites was provided at the 50th percentile of 0.102g and 5% damping.

- This Site Specific Response Spectra for SEP Plants Located in the Eastern United States was finalized and issued by NRC letter to all SEP Owners (except San Onofre) June 8, 1981 (reissued June 17, 1981). This Final Site Specific Spectrum recommended ground response spectra (5% damping) was 0.11g.

- In the CPCo April 25, 1979 letter and the July 26, 1979 meeting, we agreed to construct structural models and exercise them using an example spectra. The example spectra is a Reg Guide 1.60 spectra anchored at 0.12g. This seismic input consists of a sample problem earthquake having a zero period horizontal ground acceleration equal to 0.12g.

- In May of 1982, a Site Specific Response Spectrum was prepared for CPCo by Weston Geophysical Corporation and was derived by CPCo independently from the NRC efforts in this area. This report was submitted to the NRC on May 5, 1982. A copy of Attachment 1 from the May 5, 1982 letter is provided at Figure 2.4 of this report and shows a plot of the spectra resulting from the Weston work as well as a 0.12g Reg Guide 1.60 spectrum and the site specific spectrum issued by the NRC (letter of June 8, 1981) of 0.104g.

- In the Spent Fuel Pool Expansion Hearings, an affidavit in support of Motion for Extension of Time (May 3, 1982) was filed noting possible anomalous site conditions which could affect the seismic input ground motion at Big Rock Point.

  The NRC Staff issued an "Assessment of Possible Soil Amplification at Big Rock Point Site," June 30, 1982. This evaluation of the possible need to modify the seismic input ground motion because of shallow soil conditions at the site concluded that the original issued ground response spectra are still appropriate (i.e., 0.11g).

  Extensive studies of amplification at Big Rock Point may only be of marginal safety significance. The seismic hazard at this site is so low such that the chance that there will be amplified ground motion in excess of the previously identified spectrum (Memorandum from R Jackson to W Russell, dated May 20, 1981 attached to the June 17, 1981 NRC letter) is extremely small.

Conclusions

It has been Consumers Power Company's position that safety-related plant improvements or additions should be designed in accordance with current regulatory criteria as practicable within the constraints of the existing plant design and considering the nature of the improvement in terms of its effect on overall plant safety.

In this regard we would intend to use seismic design criteria based either on the Reg Guide 1.60 (0.12g) earthquake or the NRC site specific (0.104g) earthquake as both are acceptable seismic design bases. Big Rock Point is also involved in resolution of Unresolved Safety Issue A-46 for Seismic Qualification of Equipment in Operating Nuclear Power Plants through Generic Letter 87-02 as a member of a Seismic Qualification Utility Group (SQUG).

2.5.2.2 Historical Hazard Analysis

The following historical hazard analysis summary was extracted from (Reference 29) and is included in this report to provide additional seismic hazard analysis which justifies the conclusion by the NRC that further extensive studies of amplification at Big Rock Point may only be of marginal safety significance:

The seismic hazard at Big Rock Point is very low. According to a recent compilation of historical and instrumentally recorded earthquakes (NUREG/CR-1577) the closest earthquake occurred at a distance of more than 100 km from the site and this event was of Modified Mercalli Intensity V or less. In addition, Chen and Bernreuter (1982) performed a historical hazard analysis i.e, using only actual events in the historic record (not moving them) and a ground motion model which estimates ground motion (peak acceleration) at Big Rock Point from these events. They estimated the return periods associated with peak accelerations at the site. Depending on the ground motion model used the peak acceleration associated with 4,000 year return period varied from 0.03g to 0.1g. The high value was determined using a ground motion model that according to Chen and Bernreuter (1982) may over emphasize the distant (over 1,000 km) 1811, 1812 New Madrid Earthquakes. Indeed, using the most recent ground motion model (Nuttli and Hermann, 1981), results in peak accelerations on the order of 0.001g at a distance of 1,000 km. Excluding the New Madrid events (which according to Chen and Bernreuter, 1982, have estimated return periods on the order of 500 to 1,000 years) results in a peak acceleration at Big Rock Point of 0.03g associated with the 4,000 year return period. While no attempt is made to correct for completeness of the data or delineate earthquake zones, these studies indicate that based upon 200 years of earthquake history the ground motion occurring at Big Rock Point has been very low and that simple projections of this history using current ground motion models, to long return periods on the order of thousands of years yield peak accelerations well below that originally recommended (0.1g) for the site. Based on the above, the chance that Big Rock Point will experience earthquake ground motion of any significance is extremely small.

2.5.2.3 Safe Shutdown Earthquake (SSE)

10 CFR Part 100, Appendix A requires that the Safe Shutdown Earthquake (SSE) be defined by response spectra corresponding to the expected maximum ground accelerations. Reg Guide 1.60, Revision 1 describes methods for defining this response spectra as follows:

> Maximum (peak) Ground Acceleration specified for a given site means that value of the acceleration which corresponds to zero period in the design response spectra for that site. At zero period the design response spectra acceleration is identical for all damping values and is equal to the maximum (peak) ground acceleration specified for that site.

For the Big Rock Point Site, this maximum (peak) ground acceleration is graphically depicted in the Design Response Spectrum in Figure 2.4 as the Reg Guide 1.60 at 0.12g. It should be noted that the 0.12g Reg Guide 1.60 Spectrum envelopes both the NRC Site Specific Spectra and CPCo's Big Rock Point Site Specific Spectra as discussed in 2.5.2.1 above.

2.5.2.4 Operating Basis Earthquake (OBE)

Values have not been tabulated or depicted for the Big Rock Point OBE, however these values are normally one half of the Safe Shutdown Earthquake.

2.5.2.5 Site Specific Seismic Floor Response Spectra

Derivation of Site Specific Seismic Floor Response Spectra for the seismic safety margin evaluation of Big Rock Point Plant are contained in D'Appolonia Report dated August, 1983 (Reference 30) and in (Reference 23).

2.5.3 SURFACE FAULTING

The following NRC assessment of the capability of faults in the site region was extracted from Systematic Evaluation Program Topic II-4.B, Proximity of Capable Tectonic Structures in Plant Vicinity (Reference 20):

Major faulting has not been recognized in the subregional area around the site. Although the Michigan Basin has a long history (hundreds of million years) of relative tectonic stability, large-scale structures have been mapped within it, primarily in areas of hydrocarbon exploration and production.

During geological studies in regard to the (proposed) Midland Nuclear Site, a pattern of orthogonal northwest-northeast mild deformation was mapped on several Mississippian and Devonian stratigraphic horizons (US NRC, 1982). Faults were inferred to be associated with that pattern. These investigation showed that the inferred faulting could not be demonstrated to extend upward into overlying Pennsylvanian strata, therefore the faults, if they exist, are at least Late Mississippian in age (more than 300 mybp). Deformation was also identified in Pennsylvanian rocks south and east of the Midland site. It was demonstrated however that these distortions were formed by soft sediment deformation that occurred during or shortly after deposition and were not tectonically derived (US NRC, 1982). All faults in the region around the Midland Site were concluded to have occurred prior to the Pennsylvanian period (more than 300 mybp). That conclusion is consistent with observations on the regional geological history of the Michigan Basin (Haxby et. al., 1976; Cross, 1982; and Fisher, 1979 and 1982).

The intrabasin structure is dominated by a subparallel set of northwest-southeast anticlinal flextures that are asymmetric in cross-section with the strong dip toward the basinward side. They are best defined in the eastern, southeastern, and central portions of the basin. Several prominent features located far to the south of the plant site, namely the Howell antiline, Albion-Scipio syncline, and the Lucas-Monroe monocline, are postulated (but not proven) as having west-flanking in their Paleozoic strata (USNRC 1982).

Several faults are located on the southeast flank of the Michigan Basin that have mid-Paleozoic displacements. These are the Bowling Green Fault, located in northwestern Ohio, with youngest displacement being of upper Silurian age, and faults associated with the Chatham sag, Ontario, Canada. The latter system of faults, which includes the Electric and Osborn faults, indicates that the Chatham sag was inactive after middle Devonian time (more than 350 mybp).

A series of major folds in the Paleozoic rocks characterizes the Michigan Basin (Holst, 1982). A prominent northwest striking joint set may be related to this structural grain. It is likely that faults are associated with these structures, but based on regional associations, these faults are not capable.

During the staff review of the Wisconsin Electric Company's (WEPCO) Haven Site, several sources of seismic reflection data indicated the possible presence of NNE and NW trending faults beneath Lake Michigan. The Staff reviewed these and other data gained during WEBCOs investigation, and studied the seismicity of the Lake Michigan region. Based on that review (memo from R Denise to B Grimes, October 11, 1978) the Staff concluded that 1) faulting within Palezoic strata in the Central Stable Region is widespread in rocks that are Mississippian age and older (320 mybp), therefore, the discovery of faults, or the inference of faulting within Mississippian or older units beneath Lake Michigan is not surprising; 2) no historic earthquake epicenters have been plotted in Lake Michigan, and 3) the faults beneath Lake Michigan are geologically old and pose no potential to increase the earthquake hazard of the region.

There are other structures like those described above within and around the Michigan Basin. All of these structures are considered by the Staff to be post-Devonian to pre-Pleistocene (345 mybp to 1 mybp) with most activity occurring in the late Paleozoic. This conclusion is based on the observation that all Paleozoic rocks are affected by the structures, with Mississippian being the youngest; and there is no evidence that the faults cut Pleistocene sediment.

Several minor faults have been reported in the site area. One small fault mapped by Pohl (1929) was reported as not displacing the Petoskey formation, and is therefore more than 360 million years old. Faulting described in the Penn-Dixie quarry (Walden, 1977) is related to solution slumping because they do not extend below the sinkhole in the north hall (Harding Lawson Associates, 1979).

We assume that there are probably minor faults in bedrock in the site area because faults have been mapped in Paleozoic rocks throughout the Michigan Basin. There is no evidence, however, of fault displacement of Pleistocene soils that cover bedrock in the region. We conclude that there are no faults within the site region that could be expected to localize earthquakes in the site vicinity, or that could cause surface displacements at the site. Based on our review, it is the Staff's conclusion that there are no tectonic faults that represent a hazard to the continued safe operation of the Big Rock Point Plant.

Evaluation Summary Conclusion

Geological investigations that have been carried out in the site area and throughout the Michigan Basin have not found any indication of fault movement in the recent geologic past. Evidence has been found throughout the basin that indicates that the latest movement that occurred along known faults was at least 330 million years ago. No evidence has been found that faults displace Pleistocene deposits. No faults have been identified at the site, however, if they exist, they like all known faults in the Michigan Basin are not capable according to Appendix A, 10 CFR, Part 100.

2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

The following assessment of the foundations and earthworks properties under anticipation loading conditions including earthquakes was extracted from Systematic Evaluation Program Topic II-4.F, Settlement of Foundations and Buried Equipment (Reference 22):

Figure 2.3 shows the general layout of the plant. In addition to the structures shown in Figure 2.3, an Offshore Intake Structure and Offshore Intake Pipe Line are also part of the plant. These supply the cooling water for the operation and also safe shutdown of the plant. The Offshore Intake Structure is a submerged trestle structure located approximately 1,200 feet offshore in Lake Michigan where the depth of water is approximately 30 feet. The Offshore Intake Pipe Line connects the Intake Structure to the Screenwell-Pumphouse/Diesel Generator/Discharge Structure (the total length of the pipeline is about 1,450 feet).

Seismic safety margin evaluation of BRP by D'Appolonia (Reference 23) presents detailed description and functions of these safety-related structures, systems and components.

NOTE: Since issuance of the NRC Safety Evaluation Report (Reference 22), BRP has constructed an Alternate Shutdown Panel Building. This building was analyzed for settlement and for the seismic safe shutdown earthquake ground acceleration of 0.12g by CPCo and the following evaluation data is applicable to this structure.

The foundations of the safety-related structures, systems and components that were considered in the NRC SEP Topic II-4.F settlement evaluations are:

- Reactor Building
- Turbine Building
- Screenwell-Pumphouse/Diesel Generator/Discharge Structure
- Fuel Cask Loading Dock/Core Spray Equipment Room
- Intake Structure (offshore)
- Intake Pipe Line (offshore)
- Buried Fire Main Piping System and Electrical Cables

NOTE: Alternate Shutdown Panel Building evaluated by CPCo.

Foundation Data

Source of Information

Geotechnical data available for this site are:

1. "Soil Report", Big Rock Point Plant, Charlevoix, Michigan by Soil Testing Service, Inc, March 7, 1960.

2. "Big Rock Nuclear Power Plant, Hydrological Survey", Report by Great Lake Research Division, Institute of Science and Technology, University of Michigan for Consumers Power Company, November 1961.

3. "Geophysical Cross-Hole Survey", Big Rock Point Nuclear Power Plant, Charlevoix, Michigan, January 1979, by D'Appolonia, Consulting Engineers.

The first set of data, Soil Reports (1960), presents the geotechnical investigation and analyses performed in connection with the construction of the power plant. The investigation consisted of drilling seven borings and performing laboratory tests on soil samples recovered from the borings.

The second set of data presents a description of the lake bottom as observed by divers during hydrological survey.

The third set of data, Geophysical Cross-Hole Survey Report (1979), presents the geophysical investigations performed to establish the dynamic properties of the materials at the site. This investigation consisted of drilling three borings and performing cross-hole tests to determine the compressional and shear wave velocities as a function of depth.

In addition, data gathered during NRC site visits were also used in the evaluation.

Subsurface Conditions

Plant Site

The Plant Site (ground surface at average elevation 590.0 feet) has approximately 40 feet thick soil overburden overlying limestone bedrock; the overburden is composed of:

> Seven to ten foot thick, medium dense to dense, fine to coarse sand with some gravel and limestone chips, and varying amount of silt. This is a glacial outwash deposit. Standard penetration test (ASTM D1586) blow count ranged from 8 to 33. The ground water table is controlled by the adjoining lake level and is at an approximate depth of 8 feet below ground surface.
>
> Thirty to thirty-five foot thick, fine to coarse sand with some clay, trace of silt and gravel. This is a very stiff cohesive glacial till. The standard penetration test blow count ranged form 19 to 162. Sand lenses were occasionally encountered in this stratum.

The bedrock is limestone. The upper 15 to 17 feet of this is highly fractured and weathered fossiliferous limestone with seams of clay. The core recovery in this zone ranged from 0 to 90 percent and the RQD (Rock Quality Designation) radio ranged from 0 to 26.

The highly fractured limestone zone is underlain by approximately 75 foot thick limestone with occasional seams of clay. The core recovery in this zone ranged from 40 to 100 percent and the RQD radio ranged from 0 to 84.

This limestone is underlain by approximately 50 foot thick, highly fractured limestone with vugs. The core recovery in this zone ranged from 10 to 100 percent and RQD radio was 0.

The fractured vuggy zone is underlain by slightly broken to massive limestone. The core recovery in this zone ranged from 52 to 100 percent and the RQD ration ranged from 55 to 90. The deepest boring at the site (201 feet deep) was terminated in this stratum.

Offshore Intake Structure and Offshore Intake Pipe Line

The surficial material on the lake bed along the intake pipe consists of an initial stretch of beach zone followed by boulder-pavement zone and till-cobble zone. Offshore intake structure is located in the till-cobble zone. The intake pipe line runs from the offshore intake structure to the screen well-pumphouse/Diesel Generator/Discharge Structure. Contours and approximate boundary of the lake bottom material found offshore of the BRP Site are presented in BRP Hydrological Survey contained in Volume II of this Report.

The beach zone, approximately 250 feet wide, consists of cobbles, pebbles and sand, and is continuously subjected to agitation by wave action. This includes zone of water depth shallower than five feet.

The boulder pavement zone, approximately 500 feet wide, is a spread out area of cobbles and small boulders set closely together on the bottom. Wave erosion has removed the clay and sand content of the glacial till (upper two feet zone) leaving the pebbles, cobbles and boulders to form the lake bottom, termed "Boulder Pavement Zone." This boulder pavement is approximately two feet thick and is underlain by glacial till.

In the till-cobble zone, the surficial boulder pavement zone mentioned above is not present and the till is exposed at the lake bottom.

Soil Properties

In addition to the standard penetration test blow counts, the test data available are:

1. insitu moisture content (6 to 10 percent) of till

2. unconsolidated undrained triaxial shear test on till samples recovered from split-spoon sampler (ASTM D1586) indicated an undrained shear strength of 3 TSF cohesion and 30 degrees angle of internal friction

It is concluded that this till is very stiff and highly overconsolidated.

2.5.4.1 Settlement of Structures

Plant Site Structures

All the seismic Category I buildings within the plant site are founded on glacial till stratum which is present at the plant site at a nominal depth of eight feet. Based on the available data (presented in Soil Properties above) it is concluded that the glacial till is very stiff (cohesion 3 TSF) and heavily over-consolidated. The maximum settlement due to the load from the structures was estimated by the applicant during the design stage to be minimal (less than 0.5 inch) and would take place within a short period after load application. (Note - since this evaluation was completed, the Alternate Shutdown Panel Building was constructed, with an analyzed settlement of 0.36 inch.)

The Licensee had not initiated any settlement monitoring program and has no records of any settlement monitoring. The plant has been in operation for nearly 20 years and there is no evidence of any excessive settlement. A few minor cracks were noticed during the site visit, but these minor cracks are judged to be of no significance to the safety-related structures. As the structures have been in place for nearly 20 years, the potential for future settlement is negligible.

Offshore Intake Structure

The offshore intake structure is located approximately 1,200 feet offshore where the depth of the water is approximately 30 feet. The bottom of the intake structure is approximately 12 feet below the lake bottom (till). A two-foot thick sand bedding was provided and the excavation was backfilled with the excavated soil (till) except the upper two feet was backfilled with boulder and cobble. The intake structure is a light structure and is founded on till stratum. There is no data available on either the estimated or measured settlement of this structure. Underwater inspection by the diver did not reveal any signs of tilt due to excessive differential settlement (Reference 24). Based on the information available, it is concluded that the past and future settlement of this structure is minimal with no significance to the safe operation of this safety-related structure.

Liquefaction and Seismic Settlement

The postulated safe shutdown earthquake (SSE) ground acceleration for BRP is 0.12g. The glacial till, material beneath the mat foundation is a very stiff (approximately 20 percent clay content) material which is not susceptible to liquefaction. The granular material (8 feet thick) occurring above the till is in a dense state. The water table is in the vicinity of the top of the till stratum so this granular material is not susceptible to liquefaction because it is not saturated. Seismic induced settlement of the till or dense granular material would be negligible.

The intake structure is founded in the till material which is not susceptible to liquefaction. The two feet thick sand bedding under the intake structure might liquefy and the consequences would be seismically induced settlement of negligible magnitude with no significance to its safe operation.

2.5.4.2 Settlement of Buried Equipment

Buried Fire Main Piping System (BFMPS) and Electrical Cables

Fire main piping system and electrical cables within the plant site are buried at a minimum depth of six feet below ground surface. The construction details and specifications for these are not all available. In the absence of knowledge on the backfill material assuming that the insitu granular material from the excavation was used for backfill, it is judged that this material is amenable to compaction and a modest compactive effort would result in a dense material estimated to be in the 70 percent relative density range. It is the Staff's opinion that, in the plant area, there would be no settlement related loss of support for seismic Category I piping and electrical cables founded on and in this material under static conditions.

Offshore Intake Pipe Line

The Intake Pipe runs from offshore intake structure to the Screenwell-pumphouse/diesel generator/discharge building. This is a 60-inch inside diameter and 6-inch thick wall reinforced concrete pipe buried in the lake bottom to a total length of 1,450 feet, in 16.5 foot sections connected with gasketed joints. The pipe is laid in till material (escalation 12 to 16 feet below bottom of lake bed) on 18-inch thick sand bed. The excavation is backfilled with sand up to one foot above the pipe and with gravel and cobble of six inch size up to the lake bottom. The sand was placed under water by a tremie. There was no compaction control in the specifications. The sand (amenable to compaction) has been subjected to some compaction effort when gravel and cobble stones were dumped on top of the sand. It is the staff's opinion that this material is in the 50 to 60 percent relative density range. The staff is also of the opinion that there would be no settlement related loss of support for this pipe (founded on a 18-inch thick bedding over glacial till) under static conditions.

Liquefaction and Seismic Settlement

The materials beneath and surrounding the buried Fire Main Piping Systems and Electrical Cables are not susceptible to liquefaction (see 2.5.4.1 above). Also, the seismic (SSE) induced settlement of the till or dense granular material would be negligible.

The till beneath the buried offshore intake pipe is not susceptible to liquefaction. The sand bedding under the intake pipe might liquefy. If it did, the pipe would not be affected because:

a) the pore water would escape to the overlying gravel fill

b) a very slight settlement (a few hundredths of an inch) would occur.

Hence liquefaction is not a safety problem and also the seismic (SSE) induced settlement would be negligible.

2.5.4.3 Evaluation Summary Conclusion (For Section 2.5.4 above)

Based on review of the CPCo Safety Analysis Report (Reference 25) and information obtained during the site visit, the NRC Staff concurs with the Licensee's conclusions that:

1. All the seismic Category I structures are founded on competent till material and do not possess any potential for future settlement as the settlement was essentially complete soon after construction. Any future seismic induced settlement should be minimal and will not pose a safety problem.

2. The material beneath and around the seismic Category I structures are not likely to liquefy under postulated SSE with a ground acceleration of 0.12g. The sand bedding under the offshore structures may liquefy and this would result in a seismic induced settlement of negligible magnitude. This would not be a safety concern.

3. Settlement of seismic Category I foundations and buried equipment is not a safety problem at the Big Rock Point Nuclear Power Plant.

2.5.5 STABILITY OF SLOPES

Consumers Power Company and the NRC evaluated Systematic Evaluation Program Topic II-4.D, Stability of Slopes, and determined that there are no significant natural or man made slopes on this site whose failure would affect either the safety of the plant or the attaining of safe shutdown of the plant.

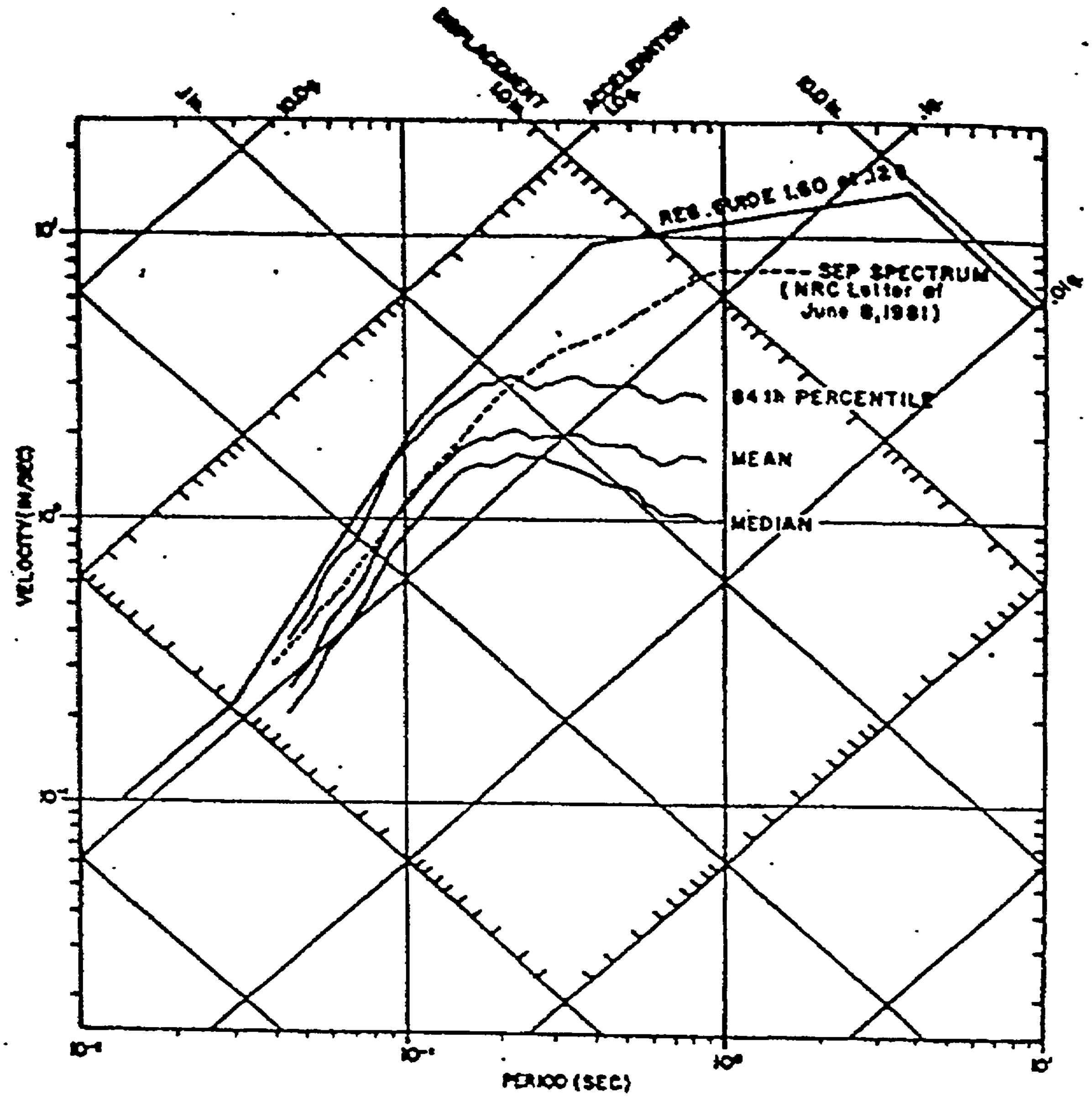
Evaluation Conclusion (Reference 26)

The NRC Staff concludes that slopes stability is not a radiological safety concern at the Big Rock Point site.

2.5.6 EMBANKMENTS AND DAMS

As described in Sections 2.4 and 2.5.4 of this report, there are no significant embankments or slopes and no dams in the site vicinity. The Systematic Evaluation Program Topic II-4.E Dam Integrity, was determined to be "not applicable" to Big Rock Point as documented in the NRC letter dated April 16, 1979 and confirmed by CPCo in the June 22, 1979 response.

Figure 2.4



Response Spectrum for Big Rock Point
Nuclear Power Plant, Compared With
Reg. Guide 1.60 Spectrum at .12 g
and SEP Spectrum (NRC Letter of June 8, 1981)
5% Damping

CHAPTER 2 REFERENCES

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2. Site Emergency Plan, Big Rock Point Plant, Docket No 50-155

3. 1993 City and County Extra, 2nd Edition, Bernan Press, Published 1992

4. HMM Document No. 3829-001, August 1993, Final Report, Evacuation Time Estimates for the Big Rock Point Power Plant Plume Exposure Pathway Emergency Planning Zone

5. US Department of Commerce Bureau of Census Document 1990 CP-1-24 Michigan Census of Population, General Population Characteristics

6. Population Characteristics of Northwest Michigan Counties, Developed by Nancy Haywood, Director, Data Research Center, Incorporated. Traverse City, Michigan, June 1980

7. USNRC letter dated May 13, 1981, SEP Topic II-1.C, Potential Hazards Due to Nearby Industrial, Transportation and Military Facilities (Big Rock Point and Palisades)

8. USNRC Atomic Safety and Licensing Board Initial Decision (on all remaining issues) Docket No 50-155-OLA, (ASLBP No 79-432-11LA), served August 29, 1984, IV O'Neill Contention IID - Risks from Aircraft

9. USNRC letter dated December 17, 1980, SEP Topic II-2.A, Severe Weather Phenomena

10. CPCo letter dated March 9, 1981, SEP Topic II-2.A, Severe Weather Phenomena

11. USNRC letter dated August 3, 1981, SEP Topic II-2.A, Severe Weather Phenomena

12. CPCo letter dated March 1, 1982, SEP Topic II-2.A - Severe Weather Phenomena; III-2 - Wind and Tornado Loading; and III-4.A - Tornado Missiles

13. CPCo letter dated April 6, 1982, SEP Topic II-2.C, Atmospheric Transport and Diffusion Characteristics for Accident Analysis

14. USNRC letter dated October 26, 1982, SEP Topic II-2.C, Atmospheric Transport and Diffusion Characteristics for Accident Analysis

15. USNRC letter dated October 26, 1982, SEP Hydrology Topics II-3.A, II-3.B, II-3.B.1, II-3.C and III-3.B

16. CPCo letter dated June 23, 1983, SEP Topic II-3.A Hydrologic Description; II-3.B Flooding Potential and Protection Requirements; II-3.B.1 Capability of Operating Plant to Cope with Design Basis Floods; II-3.C Safety Related Water Supply (Ultimate Heat Sink); III-3.A Effects of High Water Levels on Structures - Response to Safety Evaluation Reports

17. NRC letter dated December 2, 1982, SEP Topic III-3.A, Effects of High Water on Structures

18. NRC letter dated March 22, 1984, SEP Hydrology Issues

19. CPCo letter dated February 2, 1984, Integrated Assessment of Open Issues and Completion Dates for Issue Resolution

20. NRC letter dated October 12, 1982 SEP Review Topics II-4, Geology and Seismology and II-4.B Proximity of Capable Tectonic Structures in Plant Vicinity

21. Integrated Plant Safety Assessment - Systematic Evaluation Program, NUREG-0828, Final Report, May 1984

22. NRC letter dated July 20, 1982, SEP Safety Topic II-4.F, Settlement of Foundations and Buried Equipment

23. Seismic Safety Margin Evaluation Report D'Appolonia Consulting Engineers, Inc (D'Appolonia), Project 78-435 Dec 80, August 81, Revision 1

24. CPCo letter dated December 21, 1981, SEP Topic III-3.C, Inservice Inspection of Water Control Structures

25. CPCo letter dated October 19, 1981, SEP Topic II-4.F Settlement of Foundations and Buried Equipment

26. NRC letter dated July 6, 1982, SEP Topic II-4.D, Stability of Slopes

27. CPCo letter dated November 21, 1985, Integrated Plan Issue 014 (SEP Topic III-6) Seismic Weak Link Analysis Update

28. CPCo letter dated October 10, 1980, Response to Staff letter dated August 4, 1980, Proposed Seismic Evaluation Program and Basis for Continued Interim Operation

29. NRC letter dated June 30, 1982, Assessment of Possible Soil Amplification at Big Rock Point Site

30. Derivation of Site-Specific Seismic Floor Response Spectra, Seismic Safety Margin Evaluation, D'Appolonia Project No 78-435, August 1983

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TABLE OF CONTENTS

3.8 DESIGN OF CATEGORY I STRUCTURES

Regulatory Guide 1.29, "Seismic Design Classification" was used to provide a design classification system for operating nuclear power plants to identify plant features that should be designed to remain functional and withstand the effects of the Safe Shutdown Earthquake (SSE). These structures, systems, and components were designated Seismic Category I. Table 3-1 of this Updated FHSR identifies the Big Rock Point structures which should be classified Category I per the Reg Guide. Only the Alternate Shutdown Panel Building was designed and constructed to meet SSE requirements for structures (this building is not required for decommissioning, the information was deleted from Table 3-1). Other structures at Big Rock Point have been constructed or evaluated to the Seismic levels identified in Table 3-1. Reference 33 includes the structures previously considered to Seismic category 1 by the NRC Staff. The following Big Rock Point Structures are considered to be Seismic Category I:

Reactor Building Internal Structures

- Support for reactor enclosure plenum
- Fuel Pit

Seismic Design of these structures is addressed in Chapter 2 of this report and the analyses of structural adequacy to the SSE level was submitted to the NRC on January 9, 1981 with the final report by D'Appolonia published August 26, 1981, Revision 1, of the Seismic Safety Margin Evaluation Report.

On September 23, 1997, Consumers Energy submitted Big Rock Point Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certifies that all fuel has been removed from the reactor and that the reactor will not be refueled. Since that date Big Rock Point Plant has been in a decommissioning phase with its spent fuel stored in the spent fuel pool. For decommissioning, the requirements associated with Seismic Category 1 [Safe Shutdown Earthquake (SSE)], changed to Seismic Category 1, for the storage and handling of spent fuel. Structures, Systems, and Components (SSC) associated with the storage and handling of spent fuel should be designed to Seismic Category 1 requirements, with the exception of those SSCs whose failure will <u>NOT</u> cause mechanical damage to the fuel or uncover the fuel.

3.8.1 CONTAINMENT

3.8.1.1 General Description

The reactor containment is provided by a "Hortonsphere - Nuclear Containment Vessel" manufactured by Chicago Bridge and Iron Company. This spherical steel vessel is ASME Unfired Pressure Vessel Code Stamped and is 130 ft. inside diameter. The sphere extends 27 ft. below grade and 103 ft. above grade. Construction requirements are shown in Drawing 0740G20101. Distance from this vessel to the land boundaries of the site is about one-half mile, and to the edge of Lake Michigan is about 200 ft.

The containment vessel's primary purpose is to prevent a harmful spread of radioactive material to the environs as a consequence of dismantling activities or an accident involving the spent fuel.

As a secondary, everyday function, the containment vessel also serves as a weatherproof housing for auxiliary systems and for storage and handling facilities for spent fuel. Figure 3.1 is a cutaway perspective showing the general arrangement inside the sphere.

The plant is designed so that operating personnel may enter the sphere and remain inside as necessary during normal operation and during fuel handling activities.

Insulation

The exposed top half (above the equator) exterior surface of the sphere was originally insulated with a cork mastic coating sprayed to a dry thickness of 3/8 inch, and protected by two coats of acrylic resin base emulsion. The insulation is provided for the following reasons: a) To prevent excessive temperature inside the containment vessel due to solar radiation; b) To reduce heat loss in winter; c) To provide atmospheric corrosion protection; and d) To reduce inside surface condensation. Although temperature control is primarily for operational purposes, it will also tend to maintain ductility of the metal shell.

Approximately 50 to 60 percent of the containment upper surface (above the equator) original insulation has been recoated. The Design Change was accomplished via Specification Field Change (SFC) number 78-012 using a urethane foam with an elastomeric coating. This new coating is much lighter and provides greater insulating qualities.

The original design considered the Insulmastic insulation at ten pounds per square foot Dead Weight Load. The new Urethane Foam #352 Owens Corning is rated at 2.3 pounds per cubic foot or about 0.2 pounds per square foot with a Resistance (R) value of 6.3 at one inch thickness and 9.4 at one and one half inch thickness. The recoating was specified and applied at one inch -0 inch, + 2 inch tolerance.

Maintenance Features (Reference 8)

A manually operated traveling maintenance scaffold is provided and equipped with fixed platforms giving access to the external surface of the sphere. The ladder and platform are capable of withholding a uniform load of 100 pounds per square foot. The scaffold as a whole is designed to support live loads, vertical loads, and uniform load of 100 pounds per square foot on alternate platforms and stair treads existing simultaneously, plus concentrated 2500 pound live load on any one platform in lieu of the 100 pound per square foot uniform load. A painters chair anchor is provided at the top of the sphere exterior surface capable of withstanding 2000 pounds horizontal force.

Support clips are welded on the inside surface of the top hemisphere for attaching scaffolds. Each clip is rated at a safe load of 1000 pounds and are spaced not more than eight ft. apart. In considering the effect of loads on the shell, any two adjacent clips shall be taken as having a live load of 800 pounds each, and all clips as having a dead load of 200 pounds each.

Foundation

The foundation is a reinforced concrete cradle in the shape of an inverted spherical dome segment approximately seven ft. thick. Details of the foundation were submitted to the NRC for evaluation via the D'Appolonia Seismic Safety Margin Evaluation and are shown on Big Rock Point Drawing Number 0740G20152.

3.8.1.2 Penetrations and Access Openings

Penetrations

Prior to decommissioning the spherical shell was penetrated at 100 points. Ninety-five of these penetrations are welded nozzles varying from 3/4 inch to 24 inches in diameter, to permit passage of piping, instrument tubing, and electrical leads. Two other penetrations are manholes at the top and bottom of the sphere, which were used during construction and then welded shut. The remaining three are access locks, which in turn are penetrated by doors, shafts and associated piping, cables, and electrical leads. Location, size and use of the various sphere penetrations are shown in Drawing 0740G20102.

Each pipe passing through a penetration is sealed externally in a manner appropriate to its original service. As shown in Drawing 0740G40217, a pipe which exerts relatively little thermal stress is either welded directly to the ends of the penetration nozzle (Detail C), or in cases where the pipe is smaller than the nozzle, it passes through a hole in the cap which in turn is welded to the nozzle (Details B and D). Detail E shows a variation of this method to avoid contact of dissimilar metals. Where thermal movements prohibit a rigid connection, the pipe passes through a nozzle sufficiently large to allow clearance around the pipe and insulation. This space is closed by a bellows seal as shown in Detail A.

All electrical conductors are sealed where they pass through the containment boundary. For coaxial cables, access lock power and control cables, and instrumented fuel assembly cables, this is accomplished by hermetically sealed bulkhead-type connectors. All other conductors are passed through compound-filled nipples as shown in Drawing 0740G30031. Each penetration assembly was tested by the manufacturer at or above 1.25 times the design pressure of the containment vessel.

During decommissioning containment penetrations that are no longer required are sealed to preclude a direct path from being established. New penetrations will be established as necessary to allow for equipment removal or installation (Decommissioning Power for instance). These penetrations need not be designed to the original specifications for the sphere, rather they may be designed to maintain sphere closure.

Following the declaration of "Certification of Permanent Fuel Removal" and the commencing of decommissioning a 12" opening was added to facilitate powering of essential equipment in containment from the Decommissioning Power System. This penetration was installed and tested to meet containment closure requirements only.

Access Openings

Facility Change FC-702 removed the 12' diameter equipment lock in its entirety and replaced it with a Containment Construction Access opening. The opening is 18'-9" wide and 24'-0" tall and will allow for the movement of large equipment and components into and out of Containment. The Containment was extended out to the existing Fuel Cask Loading Structure.

The other two access locks are cylindrical in shape, but they vary in size. The personnel lock is 7'-7" inside diameter and the escape lock, 5'-6". Each lock has two gasketed doors in series, and the doors are designed and constructed to withstand the design pressure with no leakage detectable by soap bubbles. The doors of the personnel lock are electrically controlled, hydraulically operated, and the two in series gasketed doors are mechanically interlocked to insure that at least one is closed at all times when containment closure is required. These are breech type doors commonly used on steam autoclaves. Each opens away from the lock, so that the inner door opens into the sphere, and the outer door opens outward. The doors of the escape lock are mechanically operated and interlocked, and both doors open toward the center of the sphere. Either door of each lock can be operated from inside the sphere, inside the lock, or outside the sphere.

Personnel Lock (H-2) has a floor capable of supporting a uniform live load of 100 pounds per foot.

Minor Alteration MA-99-0018 installed a support structure used to facilitate the movement of dry fuel storage casks through the sphere out to the transport trailer. The support structure is a bridge though the containment construction access opening and includes associated supports for existing foundation and structural elements. The bridge and associated supports were designed to support 180 tons.

To facilitate dismantlement of the Recirc Pump Room/Steam Drum Area, Facility Change FC-701 cut a new access opening (approximately 7' - 0" wide x 13' - 0" long) in the containment sphere. The opening is located between elevations 593' - 0" and 604' - 1" and horizontally between columns #7 and #8. Cutting this opening in the sphere at the location indicated above removes five of the original containment penetrations (H23, H31, H32, H33, and H35). Containment Closure is achieved as described in Section 3.8.1.3.

3.8.1.3 Isolation Valves

For decommissioning the site boundary doses calculated for the postulated accidents of Chapter 15 do not credit the containment vessel for mitigative purposes.

CONTAINMENT CLOSURE

CONTAINMENT CLOSURE is that condition of containment in which there are no direct paths from containment atmosphere to the outside atmosphere, except for the containment ventilation inlet and exhaust valves, which may be open if at least one line exhaust fan is in operation. Leak tightness is not required for CONTAINMENT CLOSURE to exist.

DIRECT PATHS

A DIRECT PATH is a visually observable opening which permits the free exchange of air between containment and the environs. Equipment configurations or an engineered alternative feature such as a closed valve, check valve, water seal, closed door, membrane layer, or securely fastened plate may be used to preclude direct paths. The enclosure clean and dirty sump lines and the fuel pit and reactor drain line are examples of a water sealed engineered feature. Note that valves can be closed in these lines when they are removed from service (no longer considered water sealed).

The two 24-inch ventilation openings, one for supply and one for exhaust, would present the greatest avenue of escape for contaminants in the event of a fuel pool accident or dismantlement activities. For this reason, these openings are closed on high radiation as monitored by the spent fuel pool area monitor. To maintain consistency with the Generic Environmental Impact Statement, a High Efficiency Particulate Air (HEPA) filter has been installed in the sphere exhaust flow path. This filter will be utilized when decommissioning activities involve the potential to release significant source terms.

Facility Change FC-701 changed the Containment Closure Boundary to include the Turbine Building's Pipe Tunnel, and added a new opening in the Containment Sphere (approximately 7' - 0" wide x 13' - 0" long) between the Pipe Tunnel and the Pipe Way/Recirc Pump Room. Changing the Containment Closure Boundary required the sealing of all openings between the Pipe Tunnel and adjacent areas, and installing a material salvage/scrap removal door. The new door is located at the south end of the Pipe Tunnel. The addition of this opening requires controls to maintain Containment ventilation flow paths and Containment Closure. FC-701 maintained the sphere exhaust flow path through the use of administrative controls (ie, obtaining permission to breach Containment Closure from the Site General Manager and notifying of Operations, Security and Radiation Protection. The new door will be locked, with control maintained by Radiation Protection (High Radiation Area Entry) and Security.

Decommissioning air is supplied to the inlet and exhaust ventilation valves. A minimum of three nitrogen bottles will be maintained in place as backup to the decommissioning air system with two bottles lined up to supply the valves and one bottle lined up to the exhaust valves at one time. Gas bottle pressure will be maintained above 350psig which is sufficient to cycle the valves five times. The requirement to have the capacity to operate the valves 50 times is no longer applicable because the potential for a steam line break no longer exists and containment pressure retention capability is no longer required.

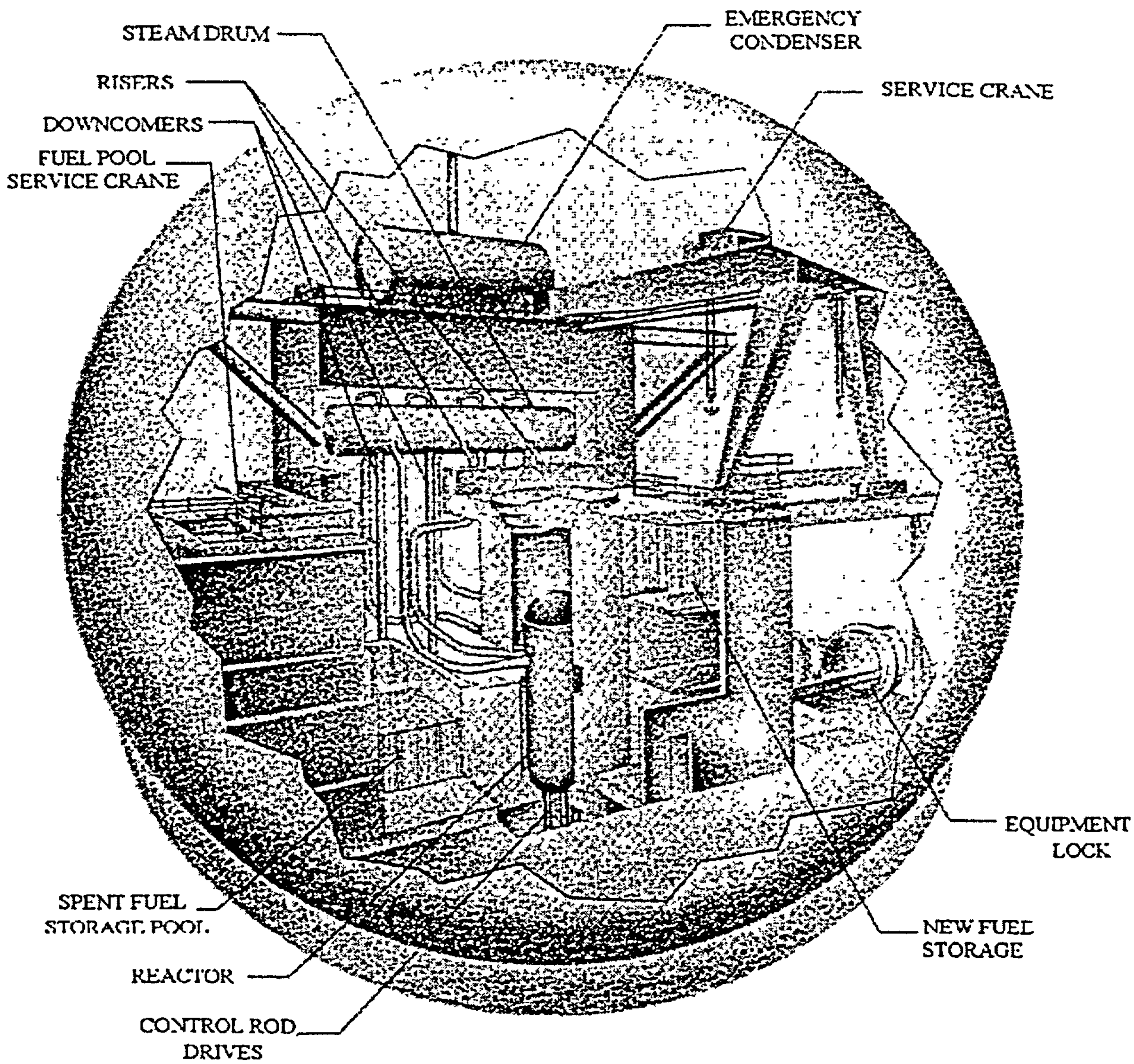
3.8.1.4 Containment Design Criteria and Design Parameters

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since the containment building is still needed. Changes to the facility that affect this section will require evaluation on a case by case basis.

Design Criteria

At an early stage in the design of the plant it was necessary to fix the design pressure of the containment vessel in order to proceed with procurement. A value of 27 psig was conservatively chosen in order to accommodate possible increases in reactor system volume during the course of design. The final calculated peak pressure in the containment was 23 psig, based on the assumption of a nearly instantaneous, complete severance of a recirculating pump discharge line, with the reactor in the hot standby condition at 1500 psia. At this time the reactor system contained its maximum stored energy. The calculation further assumed the release of all pressurized hot water and steam within the reactor, steam drum, recirculation and cleanup loops, and the steam and feedwater piping to the isolation valves. Since the reactor and the supporting power generation related systems are inoperative as a result of defueling, these pressures can no longer be attained nor can the accompanying containment temperature rise.

FIGURE 3-1
CUTAWAY OF CONTAINMENT VESSEL LOOKING SOUTHWEST



CUTAWAY OF CONTAINMENT VESSEL
LOOKING SOUTHWEST

Design Parameters

Design parameters for the containment vessel are as follows:

Table 3-3 Containment Design Parameters	
Design Pressure, Internal	27 psig
Design Pressure at Minimum Temperature, Maximum Internal Pressure	27 psig
Design Pressure, External (Coincident with dead load only) Not limiting, Safe External Pressure is 1.22 psig	0.5 psig*
Design Temperature Rise (Coincident with design internal pressure)	N/A
Design Maximum Temperature	235°F
Design Maximum Ambient Temperature	130°F
Design Minimum Ambient Temperature	45°F
Wind Load Without snow load With snow load	ASA Standard A58.1 - 1955 (Basic wind pressure = 30 psf) 60 mph
Snow Load	ASA Standard A58.1 - 1955 (max = 40 psf at top)
Lateral Seismic Acceleration (Coincident with dead load and snow load only)	5 percent of gravity
Maximum leak rate	N/A
Approximate free volume	9.4×10^5 ft^3
* External pressure does not govern; with shell thickness designed to withstand 27 psig internal pressure, safe external pressure (coincident with dead load only) is 1.22 psig.	

Material of Construction

The principal material was American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code SA-201 Grade B, firebox steel produced at SA-300 American Society for Testing and Material (ASTM) specification. SA-201 Grade B Carbon-Silican steel plates were rated at 60,000 psi minimum tensile with a maximum allowable stress of 15,000 psi at -20 to 650/F. The Charpy impact rating of the parent metal was approximately 15 foot pounds at -50/F. (NOTE: SA-201 has been discontinued as an ASME Material and is replaced by SA-515).

Shell Thickness

The steel plates used in forming the sphere vary in thickness from 0.702 to 0.875 inches. Thus, the containment is commonly referred to as 3/4 inch thick (nominal).

3.8.1.5 Design Codes

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of the information is no longer applicable. This information will be left intact for historical purposes, since the containment building is still needed. Changes to the facility that affect this section will require evaluation on a case by case basis.

The ASME Boiler and Pressure Vessel, Section II, Material Specifications, Section VIII - 1956, Unfired Pressure Vessels, and Section IX, Welding Qualifications - latest edition, including supplements, as modified by code cases 1270 N, 1271 N, and 1272 N were specified for use in the Design Specification 3159-C-1 prepared by Bechtel Corporation, Revision 1, dated December 31, 1959. The design by Chicago Bridge and Iron Company was in accordance with Revision 1 of the Specification as stated in the 5/17/60 Design Report.

3.8.1.6 Containment Construction and Testing

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since the containment building is still needed. Changes to the facility that affect this section will require evaluation on a case by case basis.

Specification 3159-C-1 Revision 4, for Design, Furnishing, Erection and Testing of the Containment Vessel; a copy of the Test Procedure and initial containment pressure test performed in January 1961 were provided in a submittal dated August 8, 1980 in response to a request for additional information for Systematic Evaluation Program Topics, (Reference 8). The following is a summary of the Construction and Testing performed:

- After excavation for the below-ground portion of the sphere steel columns were erected for support of the vessel during construction. The shell was then welded together and all seams were radiographically examined.

- Nozzle penetrations were closed by temporary steel caps, and the sphere was pressurized to 5 psig. All welds and door gaskets were soap-bubble tested for leaks, then the sphere was pneumatically tested at 1.25 times design pressure.

- An integrated leakage rate test was made at just under 27 psig, using the reference vessel method. This test demonstrated a leakage of less than 0.05% per 24 hour day. Test air was then released from the sphere.

- A large opening approximately 24 x 22 ft. was cut in the shell for construction access. Concrete was placed between the sphere and the ground, and concurrently the inside concrete structure was brought up to grade level. A portion of the weight of the sphere was removed from the steel columns by adjusting jacks at their base. The entire weight was removed from the columns upon closing of the construction opening and completion of testing. Thus, the columns remain, but the containment is free standing.

NOTE: As a result of FC-702, Construction Access Opening, column 11 was removed and columns 2, 6,9, 10, 13 and 14 preloaded to take a portion of the sphere weight.

- The interior structure was erected above grade and major pieces of equipment were installed. Piping and electrical leads were run through the nozzles and the permanent seals were made. Near the end of construction the shell plate was re-welded into the construction access opening. All new welds were radiographed.

- Prior to initial loading of fuel, a final test was made of the containment vessel. Welds and seals added or disturbed since the previous test were soaped at 5 psig. A second integrated leakage rate test of the vessel was then made at a pressure not exceeding 10 psig.

  Means are provided for introducing compressed air into each of the three access locks and the space between each pair of ventilation isolation valves, so that these appurtenances may readily be tested for leak-tightness at a suitable pressure and interval during the life of the plant. The first set of these tests was run prior to initial fuel loading.

Containment Structural Integrity Test

By letter dated December 29, 1981 CPCo provided an evaluation of SEP Topic III-7.D, Containment Structural Integrity Test. The submittal concluded that the containment structure will safely perform its intended functions and will withstand the design pressure load of 27 psig.

The NRC staff review of this topic was completed by letter dated March 17, 1982. Results of the NRC final evaluation indicated that the test procedure and results were compared with current NRC criteria for such tests in order to determine if any significant deviations existed. The evaluation described herein is based on the design and test pressure loading of the containment as presented in the CPCo December 29, 1981 evaluation.

NRC Evaluation

The containment was to be subjected to a test pressure of 33.75 psig. However, due to inadequacies in the instrumentation, it was estimated that the actual test pressure was 31.75 psig. Since the design pressure was 27 psig the test pressure was actually 31.75 / 27 = 1.176 times design. This compares favorably with the current ASME-B&PV Code requirement for a test pressure of 1.1 times design.

The test procedure and field log were compared with the current criteria per Article NE-6000 of Subsection NE of the ASME B&PV Code, Section III, Division 1. The following deviations have been identified:

1. The pressure gages used (2 indicating gages and 1 recording gage) were not calibrated. The recording gage seemed to have overestimated by approximately 2 psig at the low range. As noted above, the test pressure actually used is conservative.

2. The inner door pressure equalizing valves on both the personnel and equipment lock were not operating properly during the test. The test was performed with these fully open. This, however, does not invalidate the test. Calculations submitted by the licensee for the design of the doors were examined and found to show conservative stress levels.

Conclusion

Based on the review and the evaluation stated above, we conclude that the test procedure used was adequate and the test results provide assurance that the containment structure will safely perform its intended functions.

3.8.1.7 Containment Seismic and Stress Analyses

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since the containment building is still needed. Changes to the facility that affect this section will require evaluation on a case by case basis.

D'Appolonia Report, Volume II, Appendix A of the Seismic Safety Margin Evaluation, Reactor Building, Project No 78-435, September 80, August 81; Revision 1, Section A1.5.0, Summary and Conclusions states that:

> An analytical model of the containment shell structure has been developed for seismic response analysis. The natural frequencies of this model have been evaluated and response spectrum analyses for all three directions of excitation have been performed. The model stresses are combined using the "square root of the sum of squares" technique for response along each direction of excitation. Modeling techniques, assumptions, and the analytical results have been discussed in detail in this report.
>
> The seismic responses are combined with dead loads to conduct a stress analysis of the structure. Satisfactory compliance with the ASME (1977) code requirements for this design has been demonstrated.
>
> Floor response spectra at locations of major penetrations of the shell have been determined by the method of model superposition.
>
> Based on the investigation described herein, it is felt that the spherical steel containment shell possesses an adequate safety margin under combined dead weight and the postulated sample earthquake input loading conditions.

3.8.1.8 Containment Load Combinations

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since the containment building is still needed. Changes to the facility that affect this section will require evaluation on a case by case basis.

The shell was originally designed to resist the 27 psig internal pressure in combination with the following loads:

1. Dead weight of steel shell and appurtenances

2. Snow load - 30 pounds per square foot on slopes # 45 degrees plus 60 mile per hour wind load or 100 mile per hour wind load without snow. (Subsequently reevaluated for $ 250 mile per hour tornado load, refer to Chapter 3, Section 3.3.2 and Table 3-2).

3. Dead weight of 10 pounds per square foot insulation.

In addition the shell must resist an external pressure of 0.5 psig in combination with the dead load only.

The live load on accessories and the 5 percent earthquake load are not considered as occurring simultaneously with internal pressure. (Note that the Seismic Earthquake load and combined dead weight load were reevaluated for the 0.12 g SSE, refer to 3.8.1.7 above and Table 3-1).

3.8.1.9 Other Seismic Category I Structures Load Combinations (Reference 24)

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since the containment building is still needed. Changes to the facility that affect this section will require evaluation on a case by case basis.

Load combinations considered in the original design of other Big Rock Point structures and the design codes utilized were as follows:

Stack

According to the Bechtel Specification 3159 C-21, the concrete stack was designed to resist stress due to dead load, wind load, seismic load and temperature effects in both the vertical and circumferential directions according to the specification for the design of reinforced concrete chimneys, ACI 505-54. The seismic forces acting on the stack were analyzed as recommended in "Earthquake Design Criteria for Stack-like Structures" Paper 1696 Journal, Structural Division, ASCE, July 1958.

The stack was reanalyzed by D'Appolonia as part of a reevaluation of the Big Rock Point Plant to withstand earthquake loads. From Volume IV, Appendix E of the D'Appolonia August 81, Revision 1 report for the above grade portion of the stack the analysis utilized a combination of dead load, seismic load using the response spectrum method, and thermal loads. For below grade structural elements earth pressure was combined with dead loads and seismic loads. Volume IV, Appendix E, Attachment E1 at the end of the report rationalizes the determination of the allowable compressive strength of concrete and the allowable yield stress of steel.

Summary and conclusions from Section E.6.0 of the D'Appolonia report indicate all stresses calculated in the reinforcing steel and in the concrete have been found to be less than the allowable stresses.

Furthermore, foundation instability due to overturning and sliding has been examined under seismic loading conditions, including consideration of lateral earth pressures. The foundation has been found to have adequate safety margins against overturning and sliding, and to exhibit a no-tension condition at all points in contact with the subgrade. Based on the investigation described herein, the reinforced concrete stack has been found to be stable when subjected to a sample problem seismic input which satisfies the Regulatory Guide 1.60 recommended spectra anchored to 0.12g zero period acceleration, (SSE Level Earthquake).

Screenhouse Discharge Structural Diesel Generator Room

This combined structure has been designed to the fifth edition of the AISC Specification for Design, Fabrication, and Erection of Structural Steel, the ACI Building Code Requirements for Reinforced Concrete (ACI 318-56), and the 1958 Edition of the Uniform Building Code according to the Bechtel Design Criteria for Big Rock.

The loads considered for the screenwell, pumphouse, and discharge structure were snow load, dead load, live load, crane and impact loading, and the earthquake loads. These loads were combined in the following manner to obtain a maximum realistic loading combination:

1. Dead load + live load + snow + crane + impact

2. Dead + live + wind + 1/2 snow

3. Dead + live + seismic

The screenhouse was reanalyzed by D'Appolonia as part of a reevaluation of the Big Rock Point Plant to withstand earthquake loads. From Volume VII, Appendix H of the D'Appolonia report for above grade structures the analysis utilized a combination of deal load and seismic loads using the response spectrum method. For below grade structural elements earth pressures acting on foundation walls was combined with dead loads and seismic loads. The codes utilized for determination of allowable stresses were the AISC 1970 Edition and the ACI 349-76 requirements for nuclear safety related structures.

Summary and conclusions from Section H.6.0 of the D'Appolonia report for selected steel structural element and representative reinforced concrete wall indicates satisfactory compliance with AISC (1970) code and ACI 349-76.

Based on the investigation described herein, the screenhouse/diesel generator room discharge structure has been found to possess an adequate safety margin when subjected to a sample problem seismic input defined by USNRC Regulatory Guide 1.60 spectra anchored at 0.12g zero period ground acceleration, (SSE Level Earthquake).

Other Structures Evaluated

Other structures designated Category I or structures whose failure could affect Category I Structures were evaluated to the 0.12g SSE Level Earthquake by D'Appolonia are identified in Table 3-1 of this Updated FHSR.

3.8.2 CONCRETE AND STEEL STRUCTURES

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since these buildings still exist. Changes to the facility that affect this section will require evaluation on a case by case basis.

3.8.2.1 Codes and Standards (Reference 24)

Governing Codes and Regulations for the original civil, structural, and architectural design of Safety Related structures were:

- Uniform Building Code (UBC) 1958 Edition
- American Institute of Steel Construction (AISC) Specification for the Design, Fabrication and Erection of Structural Steel for Buildings - Fifth Edition
- American Concrete Institute (ACI) Building Code Requirements for Reinforced Concrete (ACI 318-56)
- American Welding Society (AWS) Code for ARC and Gas Welding in Building Construction
- American Standard Building Code (ASA) A 58.1 - 1955 for Wind Design Requirements
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Code, Sections I, II, VIII and IX, including Special Code Cases applicable to Reactor Containment Vessel Requirements
- Regulations of the Michigan Department of Health With Respect to Water Supply and Sewage
- Regulations of the US Army Corps of Engineers with respect to Off-Shore Structures

Table 3-1 provides additional information on the Codes and Standards utilized in the design of specific structures, systems, and components at Big Rock Point and includes information on those that have been added since the original construction.

3.8.2.2 Design Loading Conditions (Reference 34)

The following Loads and Load Combinations were considered in the original design of Safety-Related structures:

(Abbreviations utilized below)

F	=	Static Coefficient Seismic Force
DL	=	Dead Load
LL	=	Live Load
fy	=	Specified Yield Stress (Concrete)
fc	=	Specified Compressive Stress (Concrete)
psf	=	Pounds per Square Foot
ksf	=	Kilo Pounds Per Square Foot
kip	=	1000 Pound Foot

Design Loading Conditions

a. During Construction: DL = Wind or Seismic + Applicable LL

b. During Normal Operation:

1. DL + LL + Snow + Crane + Impact

2. DL + LL + Wind + 1/2 Snow Load

3. DL + LL + Seismic

Allowable stresses, including soil, may be increased 33-1/3% when loadings are combined with wind or seismic. Crane loads need not be combined with wind or seismic.

Missile design was not considered for the containment original design, but has been subsequently evaluated in Section 3.5 of this Updated FHSR.

Pipe Break effects were not considered for the original design, but has been subsequently evaluated in Section 3.6 of this Update FHSR.

Crane loads were reevaluated combined with Seismic SSE Levels. Refer to Table 3-1 for analysis.

Material Properties

Allowable Stresses

a. Structural Steel - As Specified in AISC Code

b. Reinforced Concrete - As Specified in ACI 318-56 (Except for Turbine Generator Pedestal)

Based on:

f_c' = 2500 psi for footings and for walls 12" and thicker.

f_c' = 3000 psi for piers, turbine generator pedestal and structural slabs.

f_c' = 1500 psi for lean concrete under containment vessel.

Reinforcing Steel - Intermediate Grade

Tension	20,000 Psi
Compression	16,000 Psi

c. Turbine Pedestal

Concrete-Compression	350 Psi
Concrete-Bending	400 Psi
Reinforcing Steel-Tension	10,000 Psi
Reinforcing Steel-Compression	4,000 Psi

d. Allowable Earth Pressure

DL - 3.5 ksf " 20% at 4 ft below natural ground line.

DL + LL - 5.0 maximum allowable 4 ft below natural ground line.

<u>Lateral Loads - Design</u>

a. Wind Loads on Flat Vertical Projection

8/13 of value applied as pressure on windward side.

5/13 of value applied as vacuum on leeward side.

Less Than 30 Ft above ground	25 Psf
30 Ft to 49 Ft	30 Psf
50 Ft to 99 Ft	40 Psf
100 Ft to 499 Ft	45 Psf

Cylindrical surfaces - 60% of above values.

Spherical surfaces - 46% of above values.

b. Seismic - UBC Zone 1

For reactor containment vessel only, F = 0.05 x DL.

c. Crane

Lateral load - 20% of lifted load plus trolley, one half to each rail.

Longitudinal load - 10% of maximum wheel load.

Vertical Loads - Design

a. Dead loads (DL) include weight of framing, roof, floors, walls, platforms and all permanent equipment and material.

b. Live loads (LL) - all loads except DL and lateral loads.

General

1.	Roof Snow Load	40 Psf
2.	Offices or Assembly Rooms	50 Psf
3.	Stairways and Walkways	100 Psf
4.	Laydown Areas	600 Psf

Turbine Building

1. Ground Floor and Operating Floor (Except for Designated Laydown Areas) 200 Psf

2. Crane - Maximum Wheel Load + 25% Impact

3. Concentrated 2 Kip Load Applied at Midspan of Beams and Girders, But Not Added to Columns.

Service Wing

1.	Control Room (Including DL of Board and Consoles)	200 Psf
2.	Cable Spreading Area	100 Psf
3.	Computer Area	200 Psf

Reactor Building

1. Ground Floor (Except for Designated Laydown Area) 200 Psf

2. Equipment Lock and Connecting Dolly Tracks - Two 40-Ton Axle Loads on 10 foot heel Base

3. Special Equipment Storage or Handling Areas - To suit

3.8.3 DESIGN CODES, DESIGN CRITERIA, LOAD COMBINATIONS, AND REACTOR CAVITY DESIGN CRITERIA

The information in this section pertains to an operating nuclear plant. With the plant in its decommissioning phase some of this information is no longer applicable. This information will be left intact for historical purposes, since these buildings still exist. Changes to the facility that affect this section will require evaluation on a case by case basis.

Systematic Evaluation Program (SEP) Topic III-7.B required a review of design codes, loads, and load combinations of Category I structures used in the original design be evaluated against current criteria.

This evaluation was performed by the Franklin Research Center (FRC) and a Technical Evaluation Report (TER) was prepared and is attached to the September 30, 1982 NRC Staff draft evaluation (Reference 35). The TER and draft evaluation identified areas of codes where changes occurred which were believed to have decreased safety margins.

By letter dated November 16, 1982 the NRC provided a listing of SEP Topics for which Big Rock Point does not meet current licensing criteria. Within that listing for SEP Topic III-7.B, the NRC recommended that the differences between plant original design and current licensing criteria should be resolved as follows.

1. Review of Seismic Category I Structures at Big Rock Point to determine if any of the structural elements for which a concern exists are a part of the facility design of Big Rock Point. For those that are, assess the impact of the code changes on margins of safety on a plant specific bases.

2. Examine on a sampling basis the margins of safety of Seismic Category I Structures for loads and load combinations not covered by another SEP Topic and denoted by "Ax" in the September 30, 1982 SER. (The load tables should be reviewed to assure their technical accuracy concerning applicability of the loads for each of the structures and their significance. The Category I structures considered should be reviewed to assure completeness.)

By letter dated June 20, 1983 Consumers Power Company requested relief from conducting the level of review requested by the NRC for this topic.

The NRC responded to this request as part of the May, 1984 NUREG-0828 (Reference 1) Section 4.13, as follows:

The licensee has recommended that such detailed studies not be done, but that the safety margins be determined as outlined in the resolution of seismic loads under Topic III-6 (Section 4.12). The licensee has developed similar probabilistic analyses for the loading conditions caused by winds (Section 4.5), tornado missiles (Section 4.8), and pipe breaks (Section 4.10). The staff will require that each of these evaluations explicitly consider the affected structural elements and load combinations described above, on a sampling basis, as part of the determination of the "weak links" for all of these events. Moreover, the staff will require that the licensee consider all of these probabilistic analyses collectively when deciding on selective plant upgrading, so that a relatively equivalent level of protection is achieved for all of the hazards considered (i.e., seismic, winds, tornados, and pipe breaks) and that any necessary corrective actions are integrated to the maximum extent possible. The staff will continue to review the licensee's implementation of this approach and will describe the results in a supplement to this report.

3.8.3.1 CPCo Resolution for Design Codes, Criteria and Load Combination

Evaluation

Subsequent to NUREG-0828, resolution of Topic III-7.B was assigned issue number BN-051 in the Big Rock Point Integrated Plan, with completion to follow the completion of several other related SEP Topics (III-2, III-4.A, III-6). The reviews necessary to complete Topic III-7.B have been completed, and CPCo's February 10, 1989 letter provided the results. The review was conducted in three phases, summarized as follows:

1. The structural elements listed under Section 13, Recommendations, from the FRC report, applicable to each Category 1 Structure at Big Rock were reviewed and evaluated. Attachment 1 of the February 10, 1989 letter presents the results of the review. The review concludes that the structural design was so conservatively done that the code changes do not significantly impact the margin of safety under the loads considered in the original design.

2. The second phase of the review examined those items tabulated with an "Ax" in the FRC report. The results of this examination are included as Attachment 2 of the February 10, 1989 letter. The examination concludes that the plant structures have an adequate safety margin under the combined seismic loads, but are vulnerable to the combined tornado wind load (in particular the Turbine building and the Screenhouse/Diesel Generator building). However, the construction of the Alternate Shutdown Building required to meet Appendix R, and the addition of a portable pump to resolve Wind and Tornado Loading, provides additional safeguards against damage to these two buildings.

3. The third phase of the review resulted in an overall examination of Appendix A to the FRC report. The results of this examination are presented in Attachment 3 of the February 10, 1989 letter. This review concludes that there exist some vulnerabilities from tornado loads to the Control Room, the Screenhouse Diesel Generator building and to the Turbine building. Again, due to the Alternate Shutdown building and the portable pumping capabilities that was provided at Big Rock Point, the vulnerabilities of these structures are less of a weak-link to the safety of the plant than prior to the modifications.

The overall conclusion of this review, as presented below is that no additional plant modifications are required to address the topic of Design Codes, Criteria and Load Combinations. Modifications already completed for other reasons have adequately compensated for potential weaknesses identified in this review. With this submittal Consumers Power Company considers that all actions related to SEP Topic III-7.B and Integrated Plan Issue BN-051 are now complete.

By letter dated June 12, 1991, the NRC Staff documented their resolution of SEP Topic III-7.B. They concluded that the licensee had adequately addressed this SEP Topic.

Evaluation Conclusions

Based upon the above evaluation and the following considerations, it can be concluded that changes in code provisions to not affect the safety margin of plant structures.

A. Control Room

Construction of an independent alternate shutdown building as part of the 10 CFR 50 Appendix R requirement enables the plant to safely shutdown the reactor during tornado wind load up to 250 mph and postulated tornado missile strikes without the Control Room being operable. A small area of vulnerability does exist near the equipment lock on the containment building where power and control cables for the main steam isolation and the emergency condenser are located. Risk analysis has been performed to demonstrate that the likelihood of this small area being struck by tornado missiles is extremely small. Therefore, it can be concluded that the Control Room vulnerability to tornado wind load is no longer a safety issue.

Note: for decommissioning the ability to safely shutdown the reactor is no longer applicable.

B. Diesel Generator Enclosure Screen Well and Pump House

The importance of equipment located in the screenhouse and diesel generator room has been reduced with the permanent defueling of the reactor. Ninety three days post shutdown, the spent fuel pool could experience a complete loss of cooling for 72 hours without exceeding the 150°F temperature criteria. This is considered adequate time to restore or establish spent fuel cooling. This section previously discussed compensatory measures taken during power operations that addressed plant shutdown vulnerabilities to potential tornado or tornado missile strikes. This discussion is no longer applicable and has been deleted.

C. Battery Room

The plant station batteries have been permanently removed from service. Therefore, the discussion of the battery room tornado vulnerability has been deleted.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

For an operating nuclear plant, Seismic and System Quality Group Classifications of Components/ Subsystems were made according to the safety functions to be performed. Table 3-1 contains selected structures, systems and components for the Big Rock Point Plant, the code required for current (Reference 2) licensing criteria, based on NRC Regulatory Guide 1.26, Rev 3, Section 50.55a of the Code of Federal Regulations, and the codes and standards used when the systems and components were originally built. The table also contains information regarding the Seismic Classification of the systems and components. Current NRC design criteria which was not in effect during the design of Big Rock Point requires that structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The earthquake for which these plant features are designed is defined as the Safe Shutdown Earthquake (SSE) in 10 CFR Part 100, Appendix A. The SSE is that earthquake which produces the maximum vibratory ground motion for which safety related structures, systems, and components are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated Seismic Category I in Regulatory Guide, 1.29 Rev 3.

For Big Rock Point, as an operating nuclear power plant, the SSE maximum vibratory ground motion is described in Section 2.5.2.3 of this report as the 0.12g Regulatory Guide 1.60 Rev 1, Response Spectrum.

For an operating nuclear power plant Regulatory Guide 1.29, which identifies structures, systems and components of light-water-cooled reactors on a functional basis, is the principal document used for identifying those plant features important to safety which, as a minimum, should be designed to seismic Category I requirements.

Table 3-1, Classification of Structures, Systems, and Components, was originally a revised and updated version of items evaluated as part of Systematic Evaluation Program (SEP) Topic III-1, Quality Group Classification of Components and Systems, Big Rock Point Plant. The table was originally based on an April 16, 1982 NRC Draft Safety Evaluation Report.

The original table was updated based upon information provided in CPCo letter dated November 23, 1982 and reference cited in Chapter 2 of this report for Section 2.5.2. It should be noted that the table was primarily intended to identify selected piping systems, components, and structures meeting the criteria for Regulatory Guide 1.29, Seismic Category Determinations. The table was not all inclusive, therefore other design information needed to be researched for components is not listed.

For decommissioning, the information associated with systems/structures and components not required to support the safe storage of spent fuel or radiological material control was removed from Table 3-1. Seismic Category 1 classifications, in Table 3-1, were removed from systems, structures and components whose failure could not cause a radioactive release at the site boundary having the potential of exceeding the limits of 10 CFR Part 100.

3.2.1 SEISMIC CLASSIFICATION

Structures, Systems, and Components are identified as "Safety-Related" in the Big Rock Point Plant "Q"-List. For an operating nuclear plant, identifying these items as "Safety-Related" was based upon the guidance provided in Regulatory Guide 1.29. Pertinent 10 CFR 50 Appendix B Quality Assurance Criterion for these identified items were determined in a graded manner using tools such as the plant specific Probabilistic Risk Assessment, the Technical Specifications, and other docketed analyses.

For decommissioning, the same documents are being used in identifying Structures, Systems and Components (SSC) as "Safety-Related". The criteria of Regulatory Guide 1.29, Revision 3, is still being used to identify SSCs that should be seismic Category 1, "Important to Safety". The only difference is the criteria in Regulatory Guide 1.29 is being applied as it pertains to a plant being decommissioned; therefore, only those SSCs with the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100, or those SSCs whose continued function is not required but whose failure could reduce the functioning of any plant SSC having the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. SSCs meeting the above are identified as important to safety and should be classified as Seismic Category 1. The only SSCs that are "Safety-Related" and Seismic Category 1 are the spent fuel pool concrete structure and the spent fuel storage racks.

3.2.2 QUALITY GROUP CLASSIFICATION

Regulatory Guide 1.26 establishes a system for classifying pressure boundary items into four quality groups, which are then correlated with ASME B&PV Code and ANSI Standards requirements. As an operating nuclear plant, Big Rock Point used Regulatory Guide 1.26 as a reference to establish piping system boundaries but not for defining specific quality groups.

As part of the Systematic Evaluation Program (SEP) Topic III-1, Classification of Structures, Systems, and Components (Seismic and Quality), the quality standards used for the design, fabrication, erection, and testing of the Big Rock Point Plant were compared with current codes. The NRC Draft Safety Evaluation Report on this topic (Reference 2) found that where a comparison of original codes against current codes was possible, the changes do not significantly affect the safety of the plant.

The development of the current edition of the American Society of Mechanical Engineers "Boiler and Pressure Vessel Code" (ASME Code) has been a process evolving from earlier ASME Code, American National Standards Institute, and other standards, and manufacturer's requirements. In general, the materials of construction used in earlier designs provide comparable levels of safety.

CPCo provided information on this topic be letters dated December 7, 1981 and December 29, 1981 in response to an NRC letter dated May 19, 1981. The information provided was utilized in the NRC Draft SER (Reference 2). The information was utilized by the Franklin Research Center Technical Evaluation Report (TER) for this topic in developing a table which was then used to compare the current code requirements against the original codes used during Big Rock Point design and construction. An updated version of this table is provided in this report in Table 3-1.

CPCo evaluated the Draft Safety Evaluation Report (Reference 2) and provided additional information and analyses by letter dated November 23, 1982 (Reference 3).

Based upon the information provided by Franklin and CPCo, the NRC completed a final Safety Evaluation Report (SER) September 19, 1983 (Reference 4).

For decommissioning, Regulatory Guide 1.26 continues to be used as a reference to establish piping system boundaries, but not for defining specific quality groups. Quality Groups A and B no longer exist in the decommissioning phase of Big Rock Point Plant. Quality Group C remains as it pertains to cooling water systems for residual heat removal from the spent fuel storage pool (including primary and secondary cooling systems), cooling of support systems determined important to safety, and systems that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite dose that exceeds 0.5 rem to the whole body or its equivalent to any part of the body.

NRC SER Evaluation

The basic input for the Safety Evaluation is Table 4.1 in Section 4 of the Franklin Report in (Reference 2) of this Updated FHSR Section. This table has been updated and revised and is provided in this FHSR as Table 3-1 which among other things presents a compilation of selected systems and components which are required to be classified by Regulatory Guide 1.26 and the original codes and standards used in the plant design. After comparing the original codes with those currently used for licensing new facilities, the following areas were identified where the requirements have changed:

1. Fracture Toughness
2. Quality Group Classification
3. Code Stress Limits
4. Radiography Requirements
5. Fatigue Analysis of Piping Systems

An evaluation of each of these areas is presented in Section 5 of the Franklin Report with a detailed discussion included in the Appendix.

We have determined that changes in the following areas have not significantly affected the safety functions of the systems and components reviewed in that report:

1. Quality Group
2. Code Stress Limits
3. Fatigue Analysis of Piping Systems

As part of the NRC SER evaluation on this topic, a review of information provided in CPCo letter of November 23, 1982 (Reference 3) resulted in the conclusion that for Fracture Toughness and Radiography, open issues identified in the Franklin TER, that these issues were fully resolved. Thus the only remaining open issue for this topic was Piping and Vessel Fatigue Analysis.

NRC SER Conclusions

The staff has reviewed the information in CPCo letter dated November 23, 1982 and considers the information adequate to fully resolve the open issues in this SEP topic, except for the required piping and vessel fatigue analyses. The staff recommends that these analysis be conducted by the licensee after the relevant loads have been defined in SEP Topic III-6, "Seismic Design Considerations."

Subsequent to the above, CPCo provided sample Piping Fatigue analysis by letter dated February 10, 1986 (Reference 5), and sample Vessel Fatigue Analysis by letter dated August 29, 1986 (Reference 6), as part of the Big Rock Point Integrated Assessment in response to Section 4.4.1 and 4.4.2 of the Integrated Plant Safety Assessment-Systematic Evaluation Program, NUREG-0828, Final Report, May 1984 (Reference 1).

Quality Group Classification for a Defueled Nuclear Power Plant

Table 3-1 originally reflected the Quality Group Classifications of an operating nuclear power generating plant. On September 23, 1997, Consumers Energy submitted Big Rock Point Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certifies that all fuel has been removed from the reactor and that the reactor will not be refueled. Since this date, Big Rock Point Plant has been in a decommissioning phase. As a result of initial decommissioning, Table 3-1 was revised, removing SSCs not required to support the storage of spent fuel. The requirements for the remaining SSC remain unchanged.

For decommissioning, the Quality Group Classifications for the remaining Structures, Systems, and Components (SSC) will change to reflect requirements associated with the storage and handling of spent fuel. SSCs associated with the storage and handling of spent fuel should be designed to Seismic Category I requirements, with the exception of those SSCs whose failure will NOT cause mechanical damage to the fuel or uncover the fuel.

TABLE 3-1 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS BIG ROCK POINT NUCLEAR POWER PLANT					
	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
REACTOR COOLANT SYSTEM Information on the reactor pressure vessel, vessel supports and vessel internals was deleted for decommissioning.					
RECIRCULATION SYSTEM Information on the steam drum, recirculation pumps, valves and primary system piping to the steam drum was deleted for decommissioning.					
EMERGENCY SYSTEMS Liquid Poison System (LPS) Information on the liquid poison system (storage tank, nitrogen bottles, piping and valves beyond the isolation valves) was deleted for decommissioning.					
Emergency Core Cooling System (ECCS) Information on the core spray system and enclosure spray system (core spray pumps, valves, spray nozzles, pipe and fittings, suction strainers and heat exchangers) was deleted for decommissioning.					
Emergency Condenser System (ECS) Information on the emergency condenser system (emergency condenser shell and tubes, pipe and valves) was deleted for decommissioning.					
Reactor Depressurization System (RDS) Information on the reactor depressurization system (depressurization valves, isolation valves and pipe) was deleted for decommissioning.					
Fire Protection System (FPS)					

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
Pumps		Manufacturer's Standards Bechtel Spec M-14		UBC Zone 1 (1958)	Diesel, electric and jockey pumps
Pipe and Fittings		ASA B31.1 (1955)		UBC Zone 1 (1958)	
Underground buried pipe		ASA B31.1 (1955) NFPA		UBC Zone 1 (1958)	
SAFETY RELIEF VALVE Information on the steam drum relief valves was deleted for decommissioning.					
REACTOR COOLANT PRESSURE BOUNDARY (RCPB) Information on the reactor coolant pressure boundary (which includes piping from the reactor vessel up to and including the first isolation valve) was deleted for decommissioning.					
ISOLATION VALVES Information on isolation valves other than those identified under RCB was deleted for decommissioning.					
CONTAINMENT PENETRATION VALVES AND PIPING					
	ASME III Class MC	ASA B31.1 (1955)		0.05g	

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
CONTROL ROD DRIVE COMPONENTS (CRD) Information on the control rod drive components (including the CRD housing and assembly and the CRD hydraulic system) was deleted for decommissioning.					
SPENT FUEL POOL COOLING SYSTEM (SFP)					
	ASME III Class 3	Manufacturer's Standards (6)		0.05g	
Heat Exchanger	ASME III Class 3	ASME VIII (1959) TEMA Class A		0.05g	
CONDENSATE AND FEEDWATER SYSTEMS (CDS/FWS) Information on these systems was deleted for decommissioning.					
MAIN STEAM SYSTEM (MSS) Information of the main steam system (not including the containment isolation valves) was deleted for decommissioning.					
REACTOR CLEANUP SYSTEM (RCS) Information on the cleanup system (including the cleanup demineralizer, regenerative and non-regenerative heat exchangers) was deleted for decommissioning.					
REACTOR SHUTDOWN COOLING SYSTEM (SDC) Information on the shutdown cooling system (including the pumps, valves, heat exchangers and pipe) was deleted for decommissioning.					

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
REACTOR COOLING WATER SYSTEM (RCW) Information on the reactor cooling water system (including the pumps, valves, heat exchangers, tanks and pipe) was deleted for decommissioning.					
MAKEUP AND CONDENSATE DEMINERALIZER SYSTEM (CDS) Information on the condensate system (condensate pumps, pipe and valves from hotwell to the condensate storage tank and the storage tank) was deleted for decommissioning.					
SERVICE AND INSTRUMENT AIR SYSTEM (CAS)					
		ASA B31.1 (1955)		UBC Zone 1 (1958)	Containment Closure Function
Piping, Air Receiver Tanks and Air Dryer Tank		ASME VIII (1959)		UBC Zone 1 (1958)	
MAIN DIESEL GENERATOR SYSTEM (MDG)					
	ASME III Class 3	(7)		UBC Zone 1 (1958)	
Diesel Generator Control Panel C-18	----	----		UBC Zone 1 (1958)	
MAIN Diesel Transformer	----	----		----	
SERVICE WATER SYSTEM (SWS)					
	ASME III Class 3	ASA B31.1 (1955)		UBC Zone 1 (1958)	

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
REACTOR RECIRCULATING WATER PUMP SEAL WATER SYSTEM The information on the recirculating water pump seal water components (heat exchangers, pipe and valves) was deleted for decommissioning.					
STRUCTURES					
Spherical Containment	ASME III Subsection MC	ASME II, VIII and IX as modified by applicable code cases		0.05g	(6)
Reactor Building Reinforced Concrete Internal Structure	----	----	Category I	0.05g	(6)
Support for Reactor Enclosure Plenum	----	----	Category I	0.05g	(6)
Stack	----	----		UBC Zone 1 (1958)	(6)
Support for Exhaust Stack Plenum	----	----		UBC Zone 1 (1958)	(6)
Water Intake Structure	----	----		UBC Zone 1 (1958)	
Turbine Building	----	----		UBC Zone 1 (1958)	
Service Building	----	----		UBC Zone 1 (1958)	
Office Building	----	----		UBC Zone 1 (1958)	

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
Screenhouse, Diesel Generator, Discharge Structure	----	----		UBC Zone 1 (1958)	
Alternate shutdown building Information on the alternate shutdown building was deleted for decommissioning,					
Control Room Information on the control room was deleted for decommissioning	----	----			
Access Control Building	----	----	----	----	
	----	----			
Core Spray Equipment Room	----	----		UBC Zone 1 (1958)	
Fuel Cask Loading Dock	----	----		UBC Zone 1 (1958)	
Structures Housing Liquid Radwaste	----	----		UBC Zone 1 (1958)	(6)
Waste Storage Vault	----	----		UBC Zone 1 (1958)	(6)
Spent Fuel Storage Racks	----	----	Category I	0.05g	(11)
Fuel Pit	ASME III Class 3	ACI 318-56 AISC, 5th Ed UBC (1958)	Category I	0.05g	(6)
Fuel Pool Makeup Line	----	ASA B31.1 (1977)		0.12g	

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
Concrete Block Walls	----	----	IEB 80-11	UBC Zone 1 (1958)	(8)(16)
Containment Building Crane 125-ton (DRL)/105-ton (MCL)	----	ASME B30.2 CMAA spec. #70	Category I	0.104g	(10)
60" Buried Intake Line	----	----		(7)	
Buried Electrical Cable	----	----		(7)	
Buried MAIN Diesel Generator Tank	----	----		(7)	
Buried Diesel Fire Pump Tank	----	----		(7)	
Heater, over C-18	----	----	----	----	
MDG Room Emergency Light, above MDG Batteries	----	----	----	----	
MDG Batteries (Evaluated by Consumers Energy)	----	----		----	
MDG Battery Charger	----	----		----	
MAIN Diesel Generator	----	----		----	
MDG Muffler	----	----	----	----	
MDG Cooling Water Head Tank (disabled)	----	----	----	----	
Diesel Fire Pump Batteries	----	----		----	
2 Ton Screenhouse Overhead Crane	----	----	----	----	(17)
MCC-1C and MCC-2C	----	----		----	
Diesel Fire Pump Control Panel	----	----		----	

TABLE 3-1
CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
BIG ROCK POINT NUCLEAR POWER PLANT

	Quality Classification		Seismic Classification		
Structures, Systems and Components	Codes and Standards RG 1.26(1)	Codes and Standards Used in Plant Design (2)	RG 1.29	Used In Plant Design (3)(4)	Remarks
Electric Fire Pump Control Panel C-17	----	----		----	
Hypochlorite Tank in Screenhouse	----	----	----	----	(17)
Junction Boxes, JB-19, 20,21 and 108	----	----		----	
Metal Shipping Container adjacent to JB-19, 20, 21 and 108	----	----	----	----	(17)
Junction Boxes JB-42, 43 and 44	----	----		----	
Panel C-30	----	----		----	
MCC-2D	----	----	----	----	(17)
Panel C-20	----	----	----	----	(17)
Panel 2P	----	----	----	----	(17)
Personnel Lock Control Panel Inside Containment	----	----	----	----	
Panel C-26	----	----		----	
Stack Gas Sampling Monitoring Panel	----	----		----	
Junction Box JB-97	----	----		----	

TABLE 3-1 NOTES

1. ASME III stands for the Boiler and Pressure Vessel Code, Section III Division I, Published by the American Society of Mechanical Engineers, 1977 Edition with Addenda through Summer 1978.

2. 1959 Edition is assumed when plant design is in accordance with Sections I and VIII of ASME Boiler and Pressure Vessel Code.

3. The 1961 Final Hazards Summary Report for the Big Rock Point Nuclear Power Plant stated that the reactor enclosure and equipment within are designed to withstand a ground acceleration equivalent to 0.05 g; equipment and structures outside are designed to withstand a ground acceleration of 0.025 g. It must be noted, however, that an extensive seismic reevaluation program is in effect under SEP Topic III-6, and is ongoing as part of the Integrated Plant Safety Assessment.

4. Structures or Substructures, UBC = Uniform Building Code.

5. Specific code cases given where known.

6. Seismic reevaluation to Reg Guide 1.60 (0.12 g) zero period horizontal ground acceleration. Refer to D'Appolonia Report "Seismic Safety Margin Evaluation BRP, Project 78-435, August 81, Revision 1.

7. Code edition and class not specified.

8. Seismic reevaluation for IE Bulletin 80-11 for (0.104 g) Peak Ground Acceleration Site Specific Ground Response Spectrum for selected walls. Refer to Structural Mechanics Associates (SMA) Report attached to CPCo November 24, 1992 letter to NRR.

9. Facility Change (FC-506) used 2% Damped floor response spectra from Equipment Lock Location resulting from a Reg Guide 1.60 (0.12 g) Spectrum. Refer to NRC letter dated November 17, 1983.

10. 125 Ton design rated load (105 Ton Maximum Critical Load) crane installed via FC-706. Selection of response spectra for design of the 125 Ton crane made by Sargent and Lundy (Calculation No. S-10900-000-001, Revision 1.) Containment building crane upgrade dynamic analysis performed by Bigge Power Constructors, Bigge File No. 2005-023.

11. Facility Change (FC-465) evaluated to a Reg Guide 1.60 (0.12 g) Spectrum. Refer to CPCo September 14, 1983 Supplement to Consolidated Application from Note 10 above for High Density Racks.

12. Facility Change (FC-462J-2) used Reg Guide 1.60 (0.12 g) Spectrum.

13. Structural Mechanics Associates (SMA) Report, "Seismic Fragility of BRP Core Assembly and Reactor Vessel Supports," October, 1985. Unit load responses were factored to correspond to the spectral acceleration resulting from a (0.12 g) Reg Guide Spectrum from D'Appolonia Report 78-435 "Seismic Safety Margin Evaluation, Reactor Building Primary Coolant Loop, Volume II] Appendixes "A" and "B", September 1980.

14. Facility Change (FC-444) Seismic Loading to B31.1 (1973) for (0.05 g).

15. Facility Change (FC-464) Static Load Equivalent method of stress analysis performed assuming 3.0 g Horizontal and 1.0 g Vertical acceleration. Refer to CPCo to NRC letter dated 1/16/79.

16. Structural Mechanics Associates (SMA) Report, "Seismic Capacities of Selected BRP Structures and Components," April, 1983. Capacities to withstand seismic excitation were determined using the results of existing analyses as supplemented by limited ultimate load analyses. Two levels of capacities (fragilities) were determined: a median (of best estimate) and a-1 logarithmic standard deviation reported in terms of peak effective ground accelerations.

17. URS/John A Blume & Associates, Engineers July 1982 Report "Seismic Evaluation of Safety-Related Electrical Equipment at BRP Nuclear Plant," in response to USNRC Information Notice (IEIN) 80-21 "Anchorage and Support of Safety Related Electrical Equipment," as submitted by CPCo on December 16, 1982. Equipment anchorage was evaluated using 0.12 g) Safe Shutdown Earthquake loading. Also reference BRP Plant Specification Field Change (SC 80-022 A through X) and FC-683.

18. Attachment "C", of CPCo June 29, 1973 Request for Technical Specification Change Request (Proposed Change #39), "Evaluation of the Effects of Jet Thrust and Pipe Whip Due to Pipe System Break Outside Containment" provided an analysis of Main Steam and Feedwater Piping using Seismic Dynamic Analysis at 0.05 g.

19. Facility Change (FC-515) Seismic Stress Calculation to B31.1 (1977) using (1.0 g) in each horizontal and vertical direction.

20. Facility Change (FC-607) Diesel Fire Pump Driver Replacement analysis at 0.12 g.

TABLE OF CONTENTS

6.2 CONTAINMENT SYSTEMS

The containment structure and design parameters are described Chapter 3, Section 3.8.1 of this UFHSR. With the plant in the permanently defueled condition the containment functional requirements are less demanding than during reactor operation.

6.2.1 CONTAINMENT FUNCTIONAL DESIGN DESCRIPTION

During decommissioning with the reactor permanently defueled and irradiated fuel stored in the spent fuel pool the containment vessel provides:

1. the capability to control air flow from the containment vessel to the environment through monitored and filtered pathways;

2. a weather enclosure for atmospheric control (e.g. temperature); and

3. physical protection of the systems, structures and components (e.g. spent fuel pool) housed in the containment vessel.

The containment ventilation system consists of two redundant supply fans and an inlet plenum. The fans draw outside air into the containment while air is exhausted through the exhaust plenum. The containment ventilation system is designed to maintain the gage pressure in the containment structure at a slight negative pressure during normal operation. This is done through pressure controllers which increase or decrease flow through supply fans in response to decreasing or increasing containment pressure.

Ventilation valves operate to provide containment ventilation and containment closure.

Previously, this section included discussions of the containment design requirements required to withstand operating events such as primary coolant and steam line breaks. Because the primary coolant and steam systems are not required in the permanently defueled condition these discussions have been deleted from this UFHSR.

6.2.2 CONTAINMENT ISOLATION SYSTEM (CIS)

6.2.2.1 CIS General Description

Containment isolation as implemented for the operating condition is not required for the permanently defueled condition. However, containment closure is required under certain situations in the permanently defueled condition as defined and discussed in the Defueled Technical Specifications. In order to be consistent with previous revisions of the UFHSR "isolation" is retained in Section titles of this UFHSR.

The containment structure design description is provided in Chapter 3, Section 3.8.1 of this Updated FHSR and includes a general overview of the components involved in the Containment Isolation System. BRP Drawing 0740G20102 provides a listing and location of the containment penetrations. BRP Drawing 0740B40539, Sheets 1 through 4, provides the Containment Isolation Valve Summary.

During decommissioning the containment is normally ventilated through one 24-inch inlet opening and one 24-inch exhaust opening. Thus, strictly speaking, the containment is a "confinement" rather than a "containment." Continuous ventilation provides contamination and temperature control to provide a habitable environment. Containment entry is required on a routine basis to perform decommissioning activities.

During reactor operation, which is no longer feasible, automatic containment isolation would occur upon reactor vessel low water level or containment building high pressure. Low reactor water level is not applicable because the reactor is not functioning in the permanently defueled condition. Containment building high pressure is not feasible because the energy sources necessary to create the high pressure were the primary coolant and steam systems, which are not functioning in the permanently defueled condition.

Although not required to mitigate the design basis accidents described in Chapter 15 of this UFHSR, automatic containment closure on high radiation has been retained as described in Section 6.2.4.1.8.

6.2.2.2 CIS Design Description

This section previously described the design requirements related to isolation of the containment due to operating reactor events such as low reactor water level or high containment pressure. Because these events are not applicable to the permanently defueled condition, the descriptions have been deleted.

6.2.2.3 CIS Compliance With NRC Design Criteria

This section previously described the design requirements related to isolation of the containment related to operating reactor events. Because these events are not applicable to the permanently defueled condition, the descriptions have been deleted.

6.2.3 CONTAINMENT CONFORMANCE TO 10 CFR 50 APPENDIX J - LEAKAGE TESTING

Because reactor building closure capability is required rather than containment integrity conformance to 10 CFR 50 Appendix J is not applicable to the permanently defueled plant.

6.2.4 CIS VENTILATION VALVES ISOLATION

The 24-inch valves in the containment ventilation system close automatically as described in Section 6.2.4.1.8 of this UFSHR.

6.2.4.1 Ventilation Isolation

The two 24 inch containment ventilation air inlet and outlet penetrations are each provided with a pair of pneumatically operated valves. Each penetration contains a swing type valve and a butterfly type valve with elastomeric seats. Each pair of valves is connected in series and are actuated by AC powered solenoid valves. The valve operators are arranged for "air to open" and "spring to close." The operator springs in the closed position exert sufficient torque on the valve disc shafts to positively seat the valves.

The redundancy provided by use of in series valves provides assurance of containment closure in the event of damage to a valve. The valves will automatically close on a high radiation signal from either area monitor on the refueling deck.

6.2.4.1.1 Butterfly Valves

The 24-inch valves have been certified to withstand the dynamic stresses imposed on them during the containment pressure transient, (which is not applicable to the permanently defueled condition), without incurring any disabling damage. In the case of the butterfly valves, a slight tilt of the disc in the direction of closure is required to assure closure in the correct direction during a LOCA (which is not applicable to the permanently defueled condition). Based upon data provided by the vendor, an opening angle between 80° and 85° for the butterfly valves was recommended due to shaft torque limitations. Both butterfly valves have been mechanically restricted to an opening of 75°, (full open for these valves is 90°). Throttling of these valves was accomplished via Specification Field Change SFC-80-007 and 80-008.

6.2.4.1.2 Swing-Check Valves

An analysis of dynamic stresses involved during the Containment Pressure Transient, (which is not applicable to the permanently defueled condition), was accomplished to develop the maximum disc closing velocity. The analysis resulted in identification of an over pressure condition on the air operator resulting from rapid closure of the valve. This condition was identified and submitted to the NRC by letter dated May 23, 1980. The air operator cylinders were modified (via Specification Field Change SFC-80-015 and 80-016) to be three inches longer to increase cylinder volume which decreased final pressure on the cylinders to less than two times normal operator pressure which is within the vendor recommended value. Based upon these modifications, the swing-check valves are not throttled.

6.2.4.1.3 Containment Purge and Vent Valve Evaluation

The potential for loose debris resulting in valve inoperability was reduced by installation of debris screens inboard of the supply and exhaust butterfly valves (via Specification Change SC-84-002), in response to the NRC Generic Item B-24, Containment Purging/Venting, and the NUREG 0828 Section 5.3.6.2. These screens function to prevent debris from being lodged in the valve seats during high containment pressure conditions (which is not applicable for the permanently defueled condition). Screens and filters in the air shed and on the inlet piping outboard of the air inlet valves perform a similar function for supply air.

The remainder of this section previously described additional requirements related to operation of the containment during normal reactor operation and operating reactor events. Because these situations are not applicable to the permanently defueled condition, the descriptions have been deleted.

6.2.4.1.4 Containment Ventilation Valve Operability Requirements

Closure of the ventilation valves is discussed in the following sections. Operability requirements for the ventilation valves for closure are included in the Defueled Technical Specifications.

6.2.4.1.5 Low Reactor Water Level and High Enclosure Pressure Isolation

This section previously described requirements related to isolation of the containment related to operating reactor events. Because these events are not applicable to the permanently defueled condition, the descriptions have been deleted.

6.2.4.1.6 Loss of Auxiliary Power Supply (Voltage Relay)

Loss of the auxiliary power supply voltage closes the ventilation valves.

6.2.4.1.7 Mode Selector Switch in the "Shutdown" Position

This section previously described requirements related to reactor and containment conditions not applicable to the permanently defueled condition. Therefore, the descriptions have been deleted.

6.2.4.1.8 Containment Ventilation Isolation on High Radiation

Even though the design basis accidents described in Chapter 15 do not take credit for containment closure automatic ventilation valve closure on a high radiation signal from the spent fuel storage area monitor has been retained for the permanently defueled condition.

6.2.4.1.9 Containment Vacuum Relief

This section described requirements related to containment conditions prior to FC-0702 that are no longer applicable to the permanently defueled condition. Therefore, the description has been deleted.

6.2.5 CONTAINMENT SPHERE INTEGRITY REQUIREMENTS

Containment integrity is not required for the permanently defueled condition. However, the capability to control air flow from the containment to the environment through monitored pathways is required when decommissioning activities have the potential to create airborne radioactivity release. Pressure retention capability is not required. Temporary containment penetrations shall be capable of being closed in a timely manner or a net positive inflow of air shall be demonstrated. As discussed in the Defueled Technical Specifications, containment closure or the ability to initiate containment closure is required during Fuel Handling (as defined in chapter 1 of the Defueled Technical Specifications) and under certain off-normal conditions identified in the Defueled Technical Specifications.

6.2.6 CONTAINMENT VISUAL EXAMINATION REQUIREMENTS

Controls are established to ensure that closure can be achieved in a timely manner or alternatively, a positive inflow of air can be demonstrated. Controls that may be utilized include administrative records or visual examination of penetrations.

6.2.7 CONTAINMENT LEAKAGE TESTING

Leak tightness is not required for containment closure to exist. Therefore, containment leakage testing is not required.

6.2.8 CONTAINMENT ISOLATION SYSTEM DEPENDABILITY

Because containment isolation is not required, this section is not applicable to the permanently defueled plant.

6.2.9 SAFETY CIRCUIT OVERRIDES ANNUNCIATION

Previously this section discussed annunciation of overrides associated with equipment not required in the permanently defueled condition. Therefore, this discussion has been deleted.

6.2.10 ENGINEERED SAFETY FEATURES (ESF) - RESET CONTROLS (IEB 80-06)

Engineered Safety Features (ESF) are those systems which are required to function to mitigate the consequences of a postulated design basis accident (Reference 1). For the permanently defueled plant no ESF systems exist because no systems are required to mitigate the heavy load event discussed in Chapter 15. Previously, this section discussed certain safety-related functions of ESF systems. Because there are no ESF systems for the permanently defueled plant, the descriptions in this section have been deleted.

6.2.11 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Previously this section discussed the potential for hydrogen generation during a loss of coolant accident (LOCA). In the permanently defueled condition a LOCA is not feasible. Therefore, the previous description has been deleted.

6.2.12 CONTAINMENT VENTILATION

Ventilation is provided through two full-capacity fans, each rated at 30,000 cfm, located in the ventilation stack. A filtration system consisting of HEPA filters, booster fan, ducting and dampers is included in the containment exhaust path. This system can be bypassed for normal operation or can be operated in series with the exhaust fans during decommissioning activities that could potentially provide a significant radioactive particulate production. Ventilation air to and from the containment sphere is via equipment located within the ventilating room. This room is located outside the containment sphere and contains the ventilation isolation valves, air heating equipment, supply air filters and necessary controls.

6.2.13 CONTAINMENT HEAT-UP

Previously, this section discussed containment heat up due to an operating reactor event. With the reactor permanently defueled this section is not applicable.

6.4 HABITABILITY SYSTEMS

The safety of plant personnel after an accident depends, in part, on the location of suitable shielding and habitability systems.

6.4.1 PLANT SHIELDING FOR SERIOUS CORE DAMAGE ACCIDENTS

Previously, this section discussed the possible exposure of plant personnel to radiation fields resulting from serious core damage accidents. These accidents are not applicable to the permanently defueled plant. Therefore, these discussions have been deleted from this section. Chapter 15 of this UFHSR discusses onsite doses due to the design basis event (mechanical damage to 500 assemblies in the spent fuel pool) for the permanently defueled plant.

6.4.2 CONTROL ROOM HABITABILITY

The control room as been replaced with the Plant Monitoring Station. Previously this section provided discussions on the habitability of the control room relative to reactor operation, which is not applicable in the permanently defueled condition. Response to the design basis event (mechanical damage to 500 assemblies in the spent fuel pool) does not require actions which would require habitability of the plant monitoring station. Therefore the habitability discussions have been deleted.

6.4.3 CONTROL ROOM AIR CONDITIONING

Previously, this section discussed control room issues associated with reactor operational events. These events are not applicable to the permanently defueled plant. Therefore these discussions have been deleted from this section.

6.4.4 CONTROL ROOM HEAT-UP TEST

Previously, this section discussed control room issues associated with reactor operational events. These events are not applicable to the permanently defueled plant. Therefore these discussions have been deleted from this section.

6.7 MAIN STEAM ISOLATION VALVE SEAL LEAKAGE CONTROL SYSTEM

During plant operation, the Main Steam Isolation Valve (MSIV) did not employ a Seal Leakage Control System. The Systematic Evaluation Program (SEP) Topic VI-9-A, "MSIV Seal System," was determined to be "Not Applicable" as discussed in the NRC November 16, 1979 letter which deleted this SEP Topic from consideration for Big Rock Point.

For the permanently defueled plant containment isolation is not required. For the permanently defueled plant the main steam system is not required.

TABLE OF CONTENTS

CHAPTER 9: AUXILIARY SYSTEMS

9.6 ALTERNATE SHUTDOWN (ASD) SYSTEM

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

New fuel assemblies that were stored inside containment prior to the plant shutdown of August 29, 1997 were returned to the fuel vendor. During decommissioning new fuel will not be stored on the plant site. The discussion of the new fuel storage facilities has therefore been deleted in its entirety.

9.1.2 SPENT FUEL POOL SYSTEM (SFP)

The (26 foot long by 20 foot wide by 30 feet 5 inch deep - nominal) spent fuel storage pool is located inside containment as shown on Consumers Energy Drawing 0740G40103.

The spent fuel storage pool is utilized for the following purposes:

a) Storing used fuel elements in racks until transfer into an approved storage/shipping container.

b) Storing highly radioactive equipment until disposal.

c) Underwater inspection, disassembly, testing, examination, sipping, reconstitution, etc. of certain irradiated fuel elements, bundles, or radioactive components.

The fuel pool is traversed by a moving platform with an electric winch to allow moving single fuel elements within the pool.

The (28 feet - 5 inch - nominal) water depth in the spent fuel storage pool is sufficient to provide adequate shielding over irradiated fuel while being moved within the pool or loaded into an approved transfer cask and for performing fuel inspection in the elevator.

Irradiated fuel will remain in the storage pool for a suitable decay period. When the fuel activity has decayed sufficiently it is transferred to an approved storage/shipping container. When fully loaded the container is then sealed and moved out of containment to a cask storage area until shipment offsite is available.

The safety of the above operation is assured by:

a) Handling and storing the irradiated fuel under water or with a single-failure-proof crane and dry fuel storage system.

b) Continuously monitoring operator action for any unusual radiation levels.

c) The handling and storage facilities which have been designed to preclude any critical arrangement.

Provisions for fuel inspection have been included in the fuel pool arrangement. There are locations allocated were fuel assemblies can be disassembled and reassembled. Irradiated fuel rods from these assemblies are stored in shipping liners and storage cans, the number of cans being determined by the number of rods to be stored, so that in addition to the fuel bundles there may also be a group of irradiated fuel rods stored in the pool (refer to Section 9.1.2.1.1 for further details).

Fuel Pool Floor Loading (Reference Bechtel Letter February 14, 1964)

Bechtel Corporation calculations show that as much as 5,000 psf could be loaded onto the entire bottom of the pool, in addition to the water, without overstressing the steel or concrete. Because this amount of load could introduce eccentricity in the foundation, permissible loading of the entire pool floor was arbitrarily established at 1500 psf in addition to the water. Any portion of the pool floor is capable of carrying the load of a 75 ton cask occupying 42 square feet of floor area.

Fuel Pool Walls

The fuel pool walls vary in thickness from three feet six inches to six feet nine inches.

9.1.2.1 Spent Fuel Pool Design

The spent fuel pool is a concrete structure which was modified via Facility Change FC-244 in 1974 to resolve blistering problems with the original phenolic coating. The modification consisted of lining the walls and floor with 3/16 inch stainless steel plate-type 304 and adding a leak chase system to detect fuel pool leakage. In order to install the liner, one inch of lead and six inches of concrete were added to the fuel pool floor below the liner. The original four inch drain line is now used for the routing of the eight zone leak chase tubing. The drain is also covered with a section of 3/16 inch stainless steel plate and six inches of concrete grout and thus offers no means of escape for pool water. The liner plate enclosure added an inner boundary to contain the water which did not exist in the original design. Postulating a liner plate rupture still allows for water to be contained by the concrete walls and floor. This change was reported in the BRP Twentieth Semi-Annual Report August 29, 1974.

The spent fuel pool is considered essentially leaktight. No significant leakage from the spent fuel pool has been encountered since initial operation of the plant in 1962. Some moist areas were identified from time to time, but no collectible amounts of water resulted. Since installation of a stainless steel liner no liquid attributable to the spent fuel pool has been observed.

The leakage detection system consisting of stainless steel channels imbedded between the concrete pool structure and the stainless steel liner was installed along with the liner. Any leakage from the liner flows through the channels into collection lines (with normally open manual valves) terminating at an open basin. Periodic inspection is made at the collection lines by observation for any flow. Any such flow from the collection basin drains to the reactor building enclosure "dirty" sump and then to the liquid radwaste system.

The moisture observed during periodic inspections is believed to be due to condensation rather than pool leakage. Any pool leakage, should it occur, would be transferred to the liquid radwaste system via the reactor building "dirty" sump.

Based upon the FC-244 changes, the fuel pool liner floor elevation is now 602 feet, one inch nominal. Normal pool water level is approximately 630 feet, 6 inches for a water depth of about 28 feet, 5 inches.

Spent Fuel Pool Structure Seismic Design

The original 0.05g seismic design was reevaluated to the 0.12g Regulatory Guide 1.70 Safe Shutdown Earthquake level as discussed in Chapter 3, Table 3-1 of this Updated FHSR.

9.1.2.1.1 Spent Fuel Storage Capacity and Restrictions

By letter dated October 11, 1984, the Commission issued Amendment 70 to License DPR-6 for BRP, this amendment authorized the storage capacity of the spent fuel pool to be increased to 441 assemblies.

The basis for 441 assemblies is described in Section 2.1 of the April 1982 Consolidated Environmental Impact Evaluation and Description and Safety Analysis as follows:

> The increased capacity will allow storage of spent fuel discharged from refuelings until 1990, while retaining full-core off-load capability. It is expected that temporary or permanent offsite spent fuel storage facilities will be available by 1990. If need be, full core off-load capability can be eliminated and four more refuelings can be accommodated. However, this is considered to be a last resort.

The maximum number of fuel assemblies to be placed in the spent fuel pool remains at the Technical Specification limit of 441 assemblies (References 2, 3 and 4). Prior to final plant shutdown on August 29, 1997, the remaining rods of a partial fuel assembly (scavenged during previous fuel reconstitutions) were redistributed into open rod locations within other host assemblies. This reconstitution effort provided sufficient fuel pool storage locations to allow a complete core off-load after final shutdown. Following core off-load, individual fuel rods, removed and stored from assemblies during previous reconstitution efforts, were redistributed within available host assemblies.

The addition of the fuel rods into the various host assemblies maintained fuel assembly uranium loading within the limits specified in the Technical Specifications. Following the fuel rod movements the number of fuel assemblies stored in the fuel pool remained at 441, with no individual fuel rods stored outside of a fuel assembly.

The spent fuel pool, storage racks, and other components within the pool are depicted in Figure 9-1. Rack types, center-to-center fuel spacing, and certain Administrative Controls are provided in Table 9-1.

FIGURE 9-1
SPENT FUEL POOL

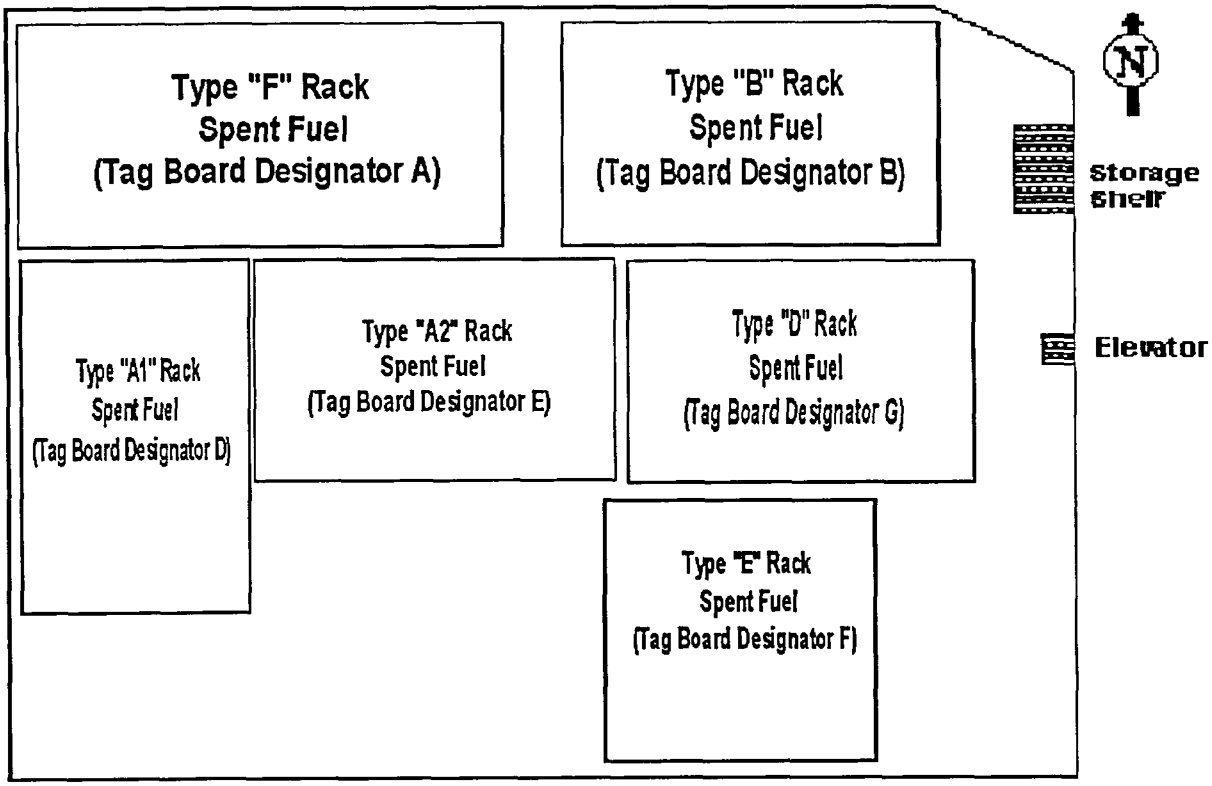


Table 9-1
SPENT FUEL POOL STORAGE RACKS

BRP Tag Board Designator	Type of Rack	Rack Cell Array	Center-to-Center Fuel Spacing	Actual Spaces	Type of Storage	Notes
A	"F"	8 x 13	9"	104	Fuel	(1)
B	"B"	6 x 12	12"	72	Fuel	(1)(3)
D	"A_1"	6 x 8	12"	48	Fuel	(1)
E	"A_2"	6 x 8	12"	48	Fuel	(1)
F	"E"	9 x 9	9"	81	Fuel	(1)(2)
G	"D"	8 x 11	9"	88	Fuel	(1)

NOTES:

1. Administrative controls have been established to ensure that no cask is moved over stored spent fuel. These controls will preclude the dropping of a cask onto a fuel rack with stored fuel.

2. Radiation levels at the south wall of the spent fuel pool, elevation 600' 6", shall be maintained at less than 50 mrem/hr above the background level during FUEL HANDLING operations. If this requirement is not met, initiate action to restore radiation levels to less than 50 mrem/hr above background levels and conduct a prompt investigation to determine the cause of increased radiation levels.

3. The storage of materials in the area between the Type "B" rack and the east wall of the spent fuel pool is prohibited. This applies to the area from the pool floor to the top of the fuel rack, to assure that the makeup line flow patterns are not blocked.

The 441 fuel assemblies stored in the pool are of several different assembly designs. The number of each design type is shown below:

Fuel Type	Number
DA	4
D	4
E	5
F	69
G	93
H	80
I	186
Total	441

Storage of spent fuel is restricted as follows:

Spent fuel shall be stored in fuel storage racks located in the spent fuel storage pool. Fuel stored in these racks shall be limited to those types identified above. The spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} less than or equal to 0.95 when flooded with unborated water. Also, to ensure a k_{eff} less than or equal to 0.95, the fuel loading in any assembly placed in the spent fuel pool shall be limited to a maximum of 28.3 grams of uranium - 235 per axial centimeter of fuel assembly or equivalent. Fuel assemblies may also be placed in inspection stations located in the spent fuel storage pool for inspection, detection of failed fuel, exchange of fuel pins or similar purposes. These inspection stations shall also be maintained with sufficient center-to-center distance between fuel assemblies placed in the inspection stations and between such assemblies and other assemblies placed in the storage racks to ensure a k_{eff} less than or equal to 0.95 when flooded with unborated water.

The spent fuel storage pool shall be maintained with a storage capacity limited to no more than 441 fuel assemblies.

Movement of spent fuel into and out of the storage racks or inspection stations shall be restricted to one assembly at a time.

For spent fuel criticality considerations, the limiting fuel design for storage in the 9 inch center- to-center spaced fuel storage racks is based upon fuel with 3.80 w/o maximum uniformly distributed U-235 enrichment (Reference 9.1.2.1.2).

Spent Fuel Pool Water Level and Temperature Indication

Spent Fuel Pool Water Level Indication is provided by:

a. A sight glass for the surge tank located on the south wall of the fuel pool surge tank, and percent level indicators located on the east wall of steam drum cavity and in the radwaste pump room area.

b. Direct visual observation of the fuel pool from the reactor operating deck.

c. Spent Fuel Pool Water Level Monitor was installed via Facility Change FC-502. The level transmitter output is transmitted to the Monitoring Station, where SFP water level, in feet elevation, indication and alarms are available (reference FC-695). The monitor can be powered from the diesel generator in the event off-site power is lost.

Spent Fuel Pool Water Temperature Indication is provided locally at the discharge of the two spent fuel pool cooling system pumps and at the outlet of the two spent fuel pool cooling system heat exchangers. A resistance temperature detector (RTD) is installed in the northeast corner of the pool. The RTD output is transmitted to the monitoring station, where SFP water temperature indication and alarms are available (reference FC-695). Operational procedures require visual verification of surge tank level and pool water level and logging of the temperature indication at the pump discharge. Visual verification of spent fuel pool cooling system pump operating status can also be made during the temperature logging function.

9.1.2.1.2 Spent Fuel Criticality Considerations

Spent fuel criticality considerations are addressed in Section 2.4 of the April 1982 Consolidated Environmental Impact Evaluation and Description and Safety Analysis and are summarized below:

A detailed nuclear analysis was performed to demonstrate that for all anticipated normal and abnormal configurations of fuel assemblies within the fuel storage racks, the neutron multiplication factor (k_{eff}) of the system does not exceed the allowable maximum value of 0.95. The normal rack configuration considered in the analysis is an array of square stainless steel cans (boxes) with a 7.00-inch inside dimension and 0.250-inch wall thickness, and spaced 9.00 inches center-to-center with centrally positioned fuel. The Big Rock Point fuel types considered for storage were G-1 (UO_2), G-1 (MOX), G-3 (UO_2), and other previously irradiated fuels. It was determined that the system containing (UO_2), is the most reactive of the three systems from a criticality standpoint. In addition, the racks were analyzed to determine the maximum allowable enrichment of 11 x 11 fuel that could be stored. The results presented here are based on this limiting fuel design with 3.80 w/o maximum uniformly distributed U-235 enrichment:

- The center-to-center spacing of 9.00 inches by 9.00 inches between fuel assemblies results in a $k\infty$ of 0.886 under nominal conditions at 68°F.

- The worst-case situations, considering maximum variations in the position of fuel assemblies within the storage rack, variations in can spacing and dimensions, variations in fuel enrichment and stainless steel composition, the most reactive temperature, calculational uncertainties, and accidents, result in a $k\infty$ less than 0.950 with a confidence level of 95 percent.

The as-fabricated limiting design fuel bundle is expected to have a non-uniform enrichment distribution. Since the non-uniform distribution system was found to be appreciably less reactive than the uniform distribution system, the above $k\infty$ (max) of 0.950 calculated based on uniform distribution of enrichment in all fuel rods will be substantially reduced for the actual limiting design fuel stored in the racks.

Although the normal expected maximum pool water temperature is 114°F, which is below the system normal design temperature of 119°F, pump failures may cause temporary surface boiling of pool water. The expected maximum average steam void near the top of the fuel assembly under this accident condition is calculated to be 20.6% (as shown in Section 2.6.3 of the consolidated application) when the pool cooling system is lost. The rise in $\Delta k\infty$ due to the formation of this maximum void at the saturated pool water temperature is 0.0044. Thus, under the pessimistic assumption of the formation of 20.6% void fraction throughout the pool, the maximum $k\infty$ for the rack configuration including the uncertainties and worst-case tolerances is expected to be 0.95 or below.

The analysis above is for the most limiting (center-to-center spacing of 9 inches) fuel storage racks added via Facility Change FC-465 as part of the expansion from 193 to 441 assemblies.

Other fuel storage racks which remained in the pool are aluminum with a minimum center-to center spacing of 12 inches. The design of these racks is such that maximum k_{eff} is approximately 0.80, as analyzed in the same manner for the new fuel storage racks discussed in Section 9.1.1.2 above, and is less than the current Standard Review Plan allowable k_{eff} value of 0.95.

9.1.2.1.3 Spent Fuel Handling Accident Analysis

The Accident Analysis of fuel handling in relation to the fuel pool expansion is provided in Section 2.4 of the April 1982 Consolidated Environmental Impact Evaluation and Description and Safety Analysis as follows:

Two types of fuel handling incidents are considered credible: (1) a fuel assembly dropped during spent fuel handling that lands horizontally on top of the storage racks, or (2) a fuel assembly inadvertently positioned vertically in a water gap between the pool wall and the rack assembly.

For case (1), since the rack structure separates the fuel in the dropped assembly from other fuel in storage by a distance of more than 12 inches and the maximum rack cell k_∞ given in the preceding section is based on the vertically infinite dimension, the rack cell k_∞ will not be affected by the case (1) accident. The case (2) accident is not feasible either because a barrier is provided on the periphery of the storage locations that precludes the insertion of a fuel assembly in the water channel, or because the rack arrangement is such that it precludes the insertion of a fuel assembly in the water channel.

Administrative controls have been established to ensure that no cask is moved over stored spent fuel. These controls will preclude the dropping of a cask onto a fuel rack with stored fuel.

9.1.2.1.4 Spent Fuel Pool Piping Systems and Failure Analyses

Piping systems that connect with the spent fuel pool are:

a. Spent Fuel Pool Cooling System (SFPCS) piping,

b. Demineralized Water System piping,

c. Treated Waste piping, and

d. Fire System piping. (Provides emergency make-up and discharges above the pool, with no direct connection to water in the Spent Fuel Pool.)

A spent fuel pool drain line was originally provided in the plant design. However, during the spent fuel pool liner modification, this drain was plugged and no penetration or provision for draining the pool was provided through the liner.

The redundant SFPCS pumps draw suction from the surge tank through the fuel pool "bag" filter. No direct piping connection to the pool exists on the suction end of the SFPCS. Surge tank level is maintained by water from the spent fuel pool flowing across a weir and into the surge tank. A suction pipe failure in the SFPCS will result in draining of the surge tank, cooling/filter equipment and connected piping. Fuel pool level will not fall below the concrete weir as a result of any suction pipe failure in the SFPCS. A water level at the weir provides approximately 20 feet of water over the top of the fuel assemblies. A siphon breaker in the SFPCS discharge to the pool prevents siphoning pool water out of the pool as a result of pipe failures in the SFPCS or any other interconnected piping systems. The spent fuel pool makeup line discharges into the pool above the highest capable water level and cannot create a siphon. Therefore, no pipe failures can allow the fuel pool water to be drained, pumped, or siphoned out.

The siphon breaker added to the inlet piping of the spent fuel pool cooling line was reported to the NRC by CPCo letter dated March 3, 1972 and consists of three 7/16 inch holes drilled in a horizontal pattern on the vertical section of inlet piping about 2 inches below normal fuel pool level. This siphon breaker was tested February 2, 1972 and was verified to eliminate any potential for losing spent fuel pool water level via the cooling line, as the siphoning action ceased when the pool level reached the three holes.

9.1.2.1.5 Spent Fuel Pool Surge Tank

The 4,750 gallon nominal capacity aluminum welded surge tank is located east of the pool and accepts the pool overflow. Surge tank level is maintained by water flowing across a weir and into the tank. An outlet pipe leaves near the bottom of the tank and goes to the fuel pool filter or directly to the fuel pit pumps suction via a filter bypass. Local level instrumentation is provided by means of a sight glass on the tank. The sight glass level gage covers the range from the outlet pipe near the tank bottom to the overflow line to the enclosure clean sump. The overflow line provides for a level of approximately 2,630 gallons. The surge tank collects overflow surges from the fuel pool with design inflow from 250 to 500 gpm. The tank material is 5052-H32 by 3/16 inch plate with a design pressure of atmospheric and design temperature of 150°F. The tank was braced for a 0.05 g lateral earthquake load.

9.1.3 SPENT FUEL POOL COOLING, CLEANUP, AND MAKEUP SYSTEMS

The following information and analyses were extracted from the April 1982 Spent Fuel Rack Consolidated Environmental Impact Evaluation Description and Safety Analysis. When pertinent the discussion has been updated for decommissioning.

9.1.3.1 Spent Fuel Pool Heat Dissipation Effects

Heat loads were calculated using the methods and data given in Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," which is attached to Section 9.2.5, Ultimate Heat Sink, of the NRC Standard Review Plan. For decommissioning the decay heat calculations were revised from those included in the April 1982 report.

Reference 5 contains the decay heat load calculations based on the spent fuel inventory given the final core was operated until August 30, 1997 (final shutdown occurred August 29). The decay heat load data provided by Reference 5 was further refined by calculations performed in Reference 6. TABLE 9-2 presents the spent fuel heat generation rate as a function of time for the 441 assemblies in the pool.

TABLE 9-2
SPENT FUEL POOL DECAY HEAT FORECAST

Days Since Shutdown	Date	Heat Up BTU/HR	Days Since Shutdown	Date	Heat Up BTU/HR	Days Since Shutdown	Date	Heat Up BTU/HR
3	9/1/97	2,644,075	1037	7/1/00	141,184	2071	5/1/03	114,765
33	10/1/97	1,186,218	1068	8/1/00	139,413	2102	6/1/03	114,428
64	11/1/97	853,101	1099	9/1/00	137,762	2132	7/1/03	114,109
94	12/1/97	705,639	1129	10/1/00	136,268	2163	8/1/03	113,788
125	1/1/98	528,066	1160	11/1/00	134,824	2194	9/1/03	113,475
156	2/1/98	463,661	1190	12/1/00	133,514	2224	10/1/03	113,178
184	3/1/98	417,627	1221	1/1/01	132,246	2255	11/1/03	112,877
215	4/1/98	376,340	1252	2/1/01	131,056	2285	12/1/03	112,592
245	5/1/98	343,808	1280	3/1/01	130,044	2316	1/1/04	112,303
276	6/1/98	316,142	1311	4/1/01	128,989	2347	2/1/04	112,020
306	7/1/98	293,896	1341	5/1/01	128,027	2376	3/1/04	111,758
337	8/1/98	274,603	1372	6/1/01	127,091	2407	4/1/04	111,484
368	9/1/98	258,296	1402	7/1/01	126,235	2437	5/1/04	111,222
398	10/1/98	244,794	1433	8/1/01	125,400	2468	6/1/04	110,955
429	11/1/98	232,742	1464	9/1/01	124,611	2498	7/1/04	110,700
459	12/1/98	222,579	1494	10/1/01	123,888	2529	8/1/04	110,441
490	1/1/99	213,352	1525	11/1/01	123,179	2560	9/1/04	110,184
521	2/1/99	205,200	1555	12/1/01	122,528	2590	10/1/04	109,939
549	3/1/99	198,613	1586	1/1/02	121,888	2621	11/1/04	109,688
580	4/1/99	192,048	1617	2/1/02	121,279	2651	12/1/04	109,447
610	5/1/99	186,321	1645	3/1/02	120,754	2682	1/1/05	109,202
641	6/1/99	180,959	1676	4/1/02	120,198	2713	2/1/05	108,958
671	7/1/99	176,238	1706	5/1/02	119,684	2741	3/1/05	108,740
702	8/1/99	171,783	1737	6/1/02	119,175	2772	4/1/05	108,500
733	9/1/99	167,707	1767	7/1/02	118,703	2802	5/1/05	108,270
763	10/1/99	164,083	1798	8/1/02	118,235	2833	6/1/05	108,035
794	11/1/99	160,634	1829	9/1/02	117,785	2863	7/1/05	107,808
824	12/1/99	157,554	1859	10/1/02	117,365	2894	8/1/05	107,575
855	1/1/00	154,611	1890	11/1/02	116,947	2925	9/1/05	107,344
886	2/1/00	151,888	1920	12/1/02	116,557	2955	10/1/05	107,122
915	3/1/00	149,523	1951	1/1/03	116,166	2986	11/1/05	106,894
946	4/1/00	147,171	1982	2/1/03	115,788	3016	12/1/05	106,674
976	5/1/00	145,055	2010	3/1/03	115,457	3047	1/1/06	106,448
1007	6/1/00	143,020	2041	4/1/03	115,101			

9.1.3.2 Spent Fuel Pool Cooling

An alternate spent fuel pool cooling system was installed in 1999 as part of the site decommissioning efforts. Installation of this system subsequently disabled the original spent fuel pool cooling equipment. The information related to the original cooling equipment has been retained for historical and reference purposes.

The original fuel pool cooling system was a closed-loop system consisting of two half-capacity pumps, two half-capacity heat exchangers in parallel, a strainer, piping, valves, and instrumentation. Cooling system piping was fabricated of aluminum to avoid scaling. The following design information was utilized in the analyses performed in the April 1982 report.

The original spent fuel pool cooling system was conservatively designed to maintain pool average temperature at less than 95°F with a one-quarter core of fuel with full cycle exposure in the pool, 48 hours after reactor shutdown. The entire original fuel pool cooling system is protected against tornado-induced damage and is located in a Seismic Category I structure, although the cooling system itself is non-seismic.

The original spent fuel pool cooling system pumps are horizontal-centrifugal with a design pressure of 140 psia, 250 gpm capacity and a differential head of 110 feet each.

The original spent fuel pool cooling system heat exchangers have a design duty of $3x10^6$ Btu/hr. Shell design flow of 125,000 lb/hr, 90 psia and 150°F with a temperature of 70°F in and 94°F out each. Tube design flow, pressure and temperature are the same as the shell with a temperature of 119°F in and 95°F out each.

The original spent fuel pool cooling system heat exchangers had an alternate design duty of $1.5x10^6$ Btu/hr at reduced shell side temperature of 70°F in and 82°F out with tube side temperature of 95°F in and 83°F out at the same 125,000 lb/hr.

Prior to the final plant shutdown the heat exchanger tube banks were examined for tube leaks. Several tubes were plugged in both heat exchangers. Specification Change package SC-97-0011 contains the details of the tube plugging. For heat exchanger HX-6A, 7 tubes of the 246 total were plugged, removing approximately 3% of the total heat exchange area. For heat exchanger HX-6B, 13 tubes were plugged removing approximately 5% of the total area. The reduction in heat transfer area was not expected to affect the heat transfer characteristics of the units, no cooling problems were encountered following core offload.

Alternate Spent Fuel Pool Cooling

An alternate skid mounted spent fuel pool cooling system has been installed to support decommissioning efforts (Ref. FC-699). The alternate system was designed to remove the decay heat calculated at the time of installation (Ref. EA-BRPDP-CH5-3 and EA-FC-699-06). An installed bypass line serves to allow isolation of the original SFP cooling equipment. The alternate equipment is located in the area adjacent to the inner cable penetration room. The alternate equipment utilizes the original sock filter piping connection for the supply and return of spent fuel water. It includes a bag filter and two full-capacity pump and heat exchanger trains. Scaling and corrosion are prevented by utilizing stainless steel piping and components. The alternate system uses Service Water as a cooling medium. Each heat exchanger is sized for a 100% capacity design duty of a minimum of 205,200 BTU/hr. Spent fuel pool side design flow is 150 gpm, 150 psig and 210°F with 95°F water in and 91.4°F water out. Service Water side design flow is 100 gpm, 150 psig and 210°F with 80°F water in and 85.4°F water out. Each pump is a 100% capacity centrifugal pump producing approximately 100 to 150 gpm, with a minimum of 100 feet of head. Cooling capability of the alternate system will increase with time as a function of the decaying spent fuel heat load. The cooling equipment includes several local alarms. Conditions resulting in a local alarm will induce a trouble alarm in the Monitoring Station.

UFHSR Section 3.5, Missile Protection, provided evaluation of the original spent fuel pool cooling system susceptibility to missiles. The installation and location of the alternate cooling system is bounded by that analysis. The new equipment is located in an area previously occupied by critical SFP cooling equipment and piping. The continued use of this location for SFP cooling equipment does not create nor increase the likelihood or consequences of a loss of SFP cooling accident.

Fuel Element Heat Transfer

In the new fuel rack design the base is elevated above the floor to ensure adequate flow under the rack to each fuel assembly. Analyses performed for this rack design showed that natural convection flow is sufficient to preclude local boiling at the hottest storage location.

Note: in the original fuel rack design the fuel assembly is not elevated and the assembly is not contained in a channel. Rather each fuel cell is constructed with an open cell arrangement which allows for free exchange of water along the full length of the fuel assembly.

With a single active failure in the spent fuel pool cooling system, the bulk temperature will not exceed 132°F for a full core off-load. Under these conditions, the maximum fuel rod surface temperature is less than 198°F, providing a 39°F margin to local boiling. The margin to bulk boiling is 77° in the assemblies and 58°F in the undercool region. This represents the limiting thermal condition in the pool.

A natural convection analysis was also performed assuming loss of all pool cooling systems. As in the previous analysis, subcooled liquid enters the bottom of the fuel assembly at the mixed hot temperature of the pool, which in this case is 212°F. Calculations show that the saturation temperature is reached within one-half inch of the fuel assembly outlet, where the maximum void fraction of 0.206.

Since the undercool region is opened to the water above the rack at the corners of each assembly, air or steam cannot be trapped between the assemblies.

In summary, with a single active failure or in normal operation, the hottest location is below the local saturation temperature and thus local boiling will not occur. Even if all pool cooling systems are lost, less than one-half inch of the assembly height will be in bulk boiling. The maximum void fraction at the outlet is 0.206. Design of the new rack is such that spaces between assemblies will always have water in them.

Fuel Pool Cooling System Performance Analyses

For the pool expansion the adequacy of the cooling system was analyzed in view of the added fuel storage capacity. For the April 1982 analyses a projection was made on the continual addition of spent fuel to the pool. These projections allowed an assessment of the pool cooling system.

On September 20, 1997 the last fuel assembly was removed from the reactor vessel and placed in the spent fuel pool. The adequacy of the cooling system has been demonstrated for the maximum heat load. On December 5, 1997 the fuel had decayed sufficiently to allow pool cooling to be removed for 72 hours without the pool water temperature exceeding 150°F (from an initial temperature of 80°F).

On June 3, 1999, an analysis was performed (Reference 7), which demonstrated:

1) The SFP temperature will gain 0.30°F per hour.
2) With a starting temperature of 100°F, the pool can experience a loss of cooling for 72 hours without exceeding the 150°F temperature limit (164 hours to reach 150°F)

The remaining discussion included projections from the April 1982 report on fuel bundle loading to the pool. This discussion has been removed for decommissioning based on the actual number of assemblies stored in the pool.

Spent Fuel Pool Cooling Failure Analyses

As discussed above, analyses were performed for the April 1982 report on the existing spent fuel cooling and fuel pool cleanup systems. It was concluded that the presently installed systems provide sufficient capacity and redundancy to accommodate the decay heat from the 441 assemblies at the maximum heat load. On December 5, 1997 the fuel had decayed sufficiently to allow pool cooling to be removed for 72 hours without the pool water temperature exceeding 150°F (from an initial temperature of 80°F).

9.1.3.3 Spent Fuel Pool Cleanup

9.1.3.3.1 Spent Fuel Pool Sock Filter

The fuel pool water was normally cleaned by continuously passing a portion of the cooling system flow through a sock filter pre-coated with diatomaceous earth. During periods of high pool water activity (e.g., during and subsequent to fuel or other component moves in the pool), the radwaste system demineralizer was used to clean the pool water.

The original spent fuel pool bypass filters were replaced during pre-operational testing when the testing revealed that these filters were inadequate. The filter was replaced by a larger vacuum type sock filter installed on the suction side of the fuel pool pumps. Installation of these full flow filters was reported in the October 9, 1963 First Annual Report.

The vacuum type sock filter has been removed in support of installation of the alternate cooling system (Ref. FC-699). In its place, a bag type filter is installed on the suction side of the cooling system pumps. The filter is intended to maintain water cleanliness during times of inactivity in the spent fuel pool. The Alternate Radwaste System and temporary in-pool filter(s) will be used during periods of high pool water activity (e.g., during and subsequent to fuel or other component moves in the pool). Supplemental demineralizer connections are provided by the alternate cooling system.

9.1.3.3.2 Portable Filtering Dolly

A portable filtration dolly is available for use in the spent fuel pool. This unit is available to provide local filtration and vacuuming within the spent fuel pool.

9.1.3.3.3 Spent Fuel Pool Water Chemistry

Spent fuel pool water is periodically sampled, and pool water chemistry is maintained within the following limits:

a. pH5.0 - 9.0

b. Conductivity - <10 micromho/cm @ 77°F

9.1.3.4 Spent Fuel Pool Makeup Water

Normal fuel pool makeup water is supplied from the treated radwaste or demineralized water system through the fuel pool cooling system.

A secondary backup supply of water is available from a fire protection system fire hose station as a damage control measure. This would be utilized to replenish the fuel pool water in the event of loss of pool water up to 200 gpm and the containment is accessible.

A spent fuel pool makeup line was installed via Facility Change FC-506. This line was modified by MA-00-0026. The primary purpose of this line was to provide cooling to the pool in the event normal fuel pool cooling was lost and the containment was inaccessible.

The makeup line allows fire protection system water to be pumped directly into the spent fuel pool. To supply water through this pathway, a manually operated valve inside the containment building must be opened.

The makeup line was designed to supply sufficient flow of water to remove the decay heat from the spent fuel maintaining the water temperature below 150°F. Seismic design of the piping was to a Regulatory Guide 1.60 earthquake anchored at 0.12g response spectra; the design criteria used for the pipe and its supports were to ANSI B31.1-1976; and the American Institute of Steel Construction (AISC) Manual for supports. These design parameters were utilized due to the interface this line had (during power operations) with the core and enclosure spray systems.

The makeup line flow goes to the top of the fuel pool where it is mixed by natural circulation.

The makeup line provides approximately 30 gpm which has been determined to be sufficient capacity to limit the pool wall average surface temperatures to about 150°F during spent fuel pool maximum heat load conditions (under worst case conditions, flow rate is 28 gpm minimum with maximum localized water temperature of less than 153°F, Reference August 29, 1984 ASLB Initial Decision).

Over-pressurization of the containment as a result of pool boiling is considered unlikely due to the fact that the boil-off rate is so low (approximately 11 gpm in the worst case).

The NRC Staff by letter dated September 30, 1983 provided a supplement to the Safety Evaluation Report (SER) related to the expansion of the spent fuel storage pool which concluded the design of the SFP make-up line is acceptable and there is reasonable assurance that the line will deliver a minimum flow rate of approximately 30 gpm. There is also reasonable assurance that the line is properly designed for seismic loads although the staff's seismic review of Big Rock Point as part of SEP is not complete. The pool water will be well mixed by natural circulation and the temperature will be maintained at or below 150°F if normal pool cooling is lost.

The licensee has calculated appropriate temperature gradients in the pool structure for use in pool thermal/structural analysis. Finally, the structural integrity of the pool will be maintained for temperatures up to 150°F.

9.1.3.4.1 Spent Fuel Pool Thermal Hydraulic Analysis

Appendix II, Part B of the "BRP Spent Fuel Rack Addition Consolidated Environmental Impact Evaluation and Description and Safety Evaluation," April 1982, as amended by Amendment 2, dated January 1983 provided the Spent Fuel Pool Thermal-Hydraulic Analysis. A summary of the analysis is provided below:

> The objective of the thermal-hydraulic analysis was to provide a conservative prediction of maximum wall temperatures in the Big Rock Point spent fuel pool during a period of time when the regular fuel pool cooling system pumps are not operating.
>
> Pool cooling was assumed to be due solely to the natural circulation of 100°F water entering at 30 gallons per minute from the makeup pipe located at the top northeast corner of the fuel storage pool. Heating was due to 441 assemblies generating a total of 216,783 watts with 62% of the total coming from 21 assemblies located in the bottom northwest corner of the pool.
>
> The calculated results show that the largest temperature occurs just above the hot assembly in the northwest corner and is less than 3°F warmer than the average pool temperature. This result was also confirmed by a simpler, one-dimensional thermal-hydraulic analysis. The calculated flow paths of the cooling water were analyzed to assist in visualizing the results. For detailed information, refer to the consolidated application.

In performing the analysis, it was conservatively assumed that heat was not lost through the walls, floor, or pool surface.

The analysis was based upon a "Computational Pool Geometry" with the fuel storage racks located approximately as shown in Figure 9-1. This computational pool geometry divided the spent fuel pool into 1430 computational volumes, in each of which the equations governing buoyant flow were solved. The shape of each computational volume was chosen to capture various geometric features of the pool, including the space between the spent fuel racks and the floor, the large gaps between the racks and the east and south walls, and the shapes of the racks themselves. Certain small gaps between the racks and between the racks and the north and west walls were not explicitly modeled, but their contributions to the total flow area were taken into account.

There are additional items in the spent fuel pool besides spent fuel and spent fuel racks. These items include channels and a small amount of research and development equipment. Radioactive maintenance materials are stored temporarily in buckets in the pool for biological shielding purposes. During decommissioning these items will be packaged and sent off site for disposal. Large casks are periodically stored in a designated area in the southwest corner of the pool.

All equipment of significant size permanently stored in the spent fuel pool was taken into account in the computer model. Moreover, as a conservatism the analyses included in its calculations more racks than were presently in the pool.

The placement of additional objects on the floor of the spent fuel pool could, under certain circumstances, block or divert flow patterns and influence local temperatures. However, local temperatures will not be affected as long as important flow patterns are not blocked. In this case, the important flow pattern is through the space between "B" type rack and the east wall of the pool. This space, if not blocked, will provide the necessary cooling path, and local temperatures will remain consistent with the analysis.

The design basis for the makeup water system was initially conceived to be 30 gpm as the maximum amount of flow that could be diverted from the core spray system to feed the makeup line under the worst case conditions. The makeup system, as designed and tested, delivers a minimum flow rate of 28 gpm flow rate under worst case conditions. Using a 28 gpm flow rate with a heat rate of 205,000 watts, it was determined that the general circulation patterns predicted at the higher flow and heat rates remained unchanged. The only difference was a drop of .1°F in the temperature at the warmest spot in the pool (2.7°F to 2.6°F).

Note: On January 1, 1998 the recent core off-load had decayed for 125 days. The total spent fuel heat load on the pool at this time was 6.13×10^5 BTU/hr (179,700 watts) (Reference 5).

9.1.3.4.2 Spent Fuel Pool Makeup Operational Requirements

The spent fuel pool makeup system was required to be operable whenever spent fuel was stored in the spent fuel pool and the plant was in power operation. For decommissioning the ability to provide a 28 gpm water source to the northeast corner of the spent fuel pool will be retained while fuel remains in the pool. Surveillance requirements are described in the defueled technical specifications.

Procedural controls specify that the pool level, pool cooling system operation, and pool circulating water temperature be periodically verified.

The following additional commitments were extracted from the August 29, 1984 "Spent Fuel Pool Expansion Hearing Initial Decision on All Remaining Issues" as they relate to the Makeup Line, (Note, the Atomic Safety and Licensing Board finding number is included in parentheses):

- The makeup line itself is 190 feet long and consists of 115 feet of two inch diameter piping and 75 feet of one inch diameter piping. Given the pipe diameters, there is no credible possibility of pipe blockage by crud, scale, rust or other foreign objects. Nevertheless, as an additional precaution, the pipe will be flushed each year with rust inhibiting chromated water. (Finding A-28) CPCo clarification - chromates are not added during the flush. The water supply contains some chromate carryover to this line which is subsequently drained from the line. This discussion was applicable during power operations. For decommissioning the ability to provide a remote actuated cooling source for the pool is not applicable since the conditions required (containment in-accessability and loss of cooling) are no longer credible. Therefore the added precaution of using chromated water is not applicable.

- The Board also has considered the possible adverse consequences of one surveillance test which is performed while the plant is at power and temporarily removes the core spray heat exchanger from service. This discussion was applicable for power operations and has been deleted for decommissioning. (Finding A-29)

- We also note that administrative controls require than hand operated valves routinely remain in positions necessary for the makeup water system to function. This discussion of the locked post incident valves was applicable for power operations and has been deleted for decommissioning. (Finding A-30)

- The discussion of the post incident system redundancy and the availability to provide remote makeup was applicable for power operations and has been deleted for decommissioning. (Finding A-31)

- This discussion of ensuring that the plant would not return to power without assuring the adequacy of the makeup line to remove the added decay heat due to refueling was applicable for power operations and has been deleted for decommissioning (Findings A-32 and A-33).

- This discussion of flow testing requirements of the makeup line during refueling was applicable for power operations and has been deleted for decommissioning. (Finding A-34) The testing requirements for the ability to provide the makeup water to the pool is specified in the Technical Specifications.

- CPCo has administrative controls to prevent fuel elements from falling on or near the makeup line. (Finding A-40)

- CPCo commitment to maintain the 150°F bulk pool temperature because that is the pool's design basis (Finding A-15). 150°F is the temperature at which loss of concrete strength is not significant (Finding A-16). The American Concrete Institute (ACI) building code indicates that strength properties of concrete are not degraded at a temperature of 150°F, and it allows temperatures up to 200°F in local areas (Finding B-13).

9.1.4 FUEL HANDLING SYSTEM (FHS)

9.1.4.1 General Description and Servicing Equipment

Most fuel pool servicing functions are carried out with the use of a remote-controlled electric overhead semi-gantry crane. This crane is a single-failure-proof crane that meets the requirements of NUREG-0544, "Single-Failure-Proof Cranes for Nuclear Power Plants." The main trolley contains a 125 ton hook for handling the reactor head and dry fuel storage system components and a 5-ton hook is provided for handling or services auxiliary equipment.

In general, fuel handling will be accomplished by manual guidance and visual observation of all fuel handling operations. Water is used as the basic shielding material except for the transfer of irradiated components between the reactor and storage pool. In addition, the SFP monorail crane may be used for moving fuel.

9.1.4.1.1 Refueling General Description

With the reactor permanently defueled (last bundle removed on September 20, 1997) the general discussion of refueling is no longer applicable and has been deleted.

9.1.4.1.2 Miscellaneous Reactor Tools

Because the reactor vessel will no longer be used, no special tools for movement or operation of reactor vessel internals are required. Therefore, the discussion related to these tools has been deleted.

9.1.4.1.3 Refueling Platform

The refueling platform and reactor vessel jib crane have been removed.

9.1.4.2 Transfer Cask

The fuel transfer cask has been removed from service, and all discussion related to the fuel transfer cask has been deleted.

9.1.4.2.1 Fuel Transfer Cask Safety Slings

The previous information on the safety slings has been deleted, as their usage is not required for the single-failure-proof crane.

9.1.4.2.2 Fuel Transfer Cask Winch

This section has been deleted. The fuel transfer cask has been removed.

9.1.4.2.3 Fuel Transfer Cask Drop Analyses (Reference CPCo January 22, 1976 letter.)

This section has been deleted. The fuel transfer cask has been removed.

Information and analysis for accidents considered during decommissioning are located in Chapter 15, Section 15.0 of this updated FHSR.

9.1.4.2.4 Fuel Transfer Cask Operability Requirements

This section has been deleted. The fuel transfer cask has been removed.

9.1.4.3 Refueling Basic Principles and Requirements

The reactor was permanently defueled on September 20, 1997, therefore the basic principles and requirements that were applicable for refueling are not applicable during decommissioning. Therefore this section has been deleted in its entirety.

9.1.4.4 Fuel Handling

A summary of the procedural sequence for fuel handling is given below. Detailed procedures will be established when transfer of the spent fuel from the pool into approved storage/transport containers becomes available. Prior to moving fuel from the spent fuel pool, a safety analysis of the process will be required.

To move a fuel assembly a fuel grapple will be attached to the selected bundle. The fuel pool winch on the traveling platform may be used to lift an assembly.

The fuel bundle will then be withdrawn vertically from the spent fuel rack. When clear, the fuel assembly is then moved within the pool.

Once the assembly has reached its destination and fully lowered, removing the assembly weight from the grapple, the fuel bundle is removed from the grapple.

9.1.4.5 Fuel Handling Control of Radiation Exposure (Reference EA-BRP-LEB-00-01)

Radiation exposure will be controlled and minimized by these considerations:

a. The grappling operation will be done through about 20 feet of water.

b. During transfer, irradiated fuel will normally be under approximately 10 feet of water, with an expected minimum depth of 4 feet of water.

c. The ungrappling operation will be done at a minimum depth of approximately 11 feet of water.

To assure the protection of the environs against any possible accident involving the fuel, containment closure provisions will be in effect during Fuel Handling operations, as defined in the Defueled Technical Specifications.

Storage of irradiated fuel and irradiated channels in the spent fuel storage pool provides a sufficient water depth to shield personnel from the irradiated material during operations over the pool.

9.1.4.5.1 Controls For Loading/Low Power Testing

The discussion included in this section was applicable only to core loading. With the reactor permanently defueled these restrictions are no longer applicable. This section has therefore been deleted for decommissioning.

9.1.4.6 Relocation and Orificing of Stainless Steel Channels

The discussion included in this section was applicable only to core loading. With the reactor permanently defueled these restrictions are no longer applicable. This section has therefore been deleted for decommissioning.

9.1.4.7 Fuel Handling or Loading Accident Analyses

The discussion included in this section was applicable only to core loading. With the reactor permanently defueled these restrictions are no longer applicable. This section has therefore been deleted for decommissioning.

9.1.4.7.1 Original FHSR Fuel Handling or Loading Accident Analyses

The discussion included in this section was applicable only to core loading. With the reactor permanently defueled these restrictions are no longer applicable. This section hastherefore been deleted for decommissioning.

9.1.4.7.2 Fuel Bundle Drop Analyses

A discussion of fuel handling accidents during decommissioning is included in Section 15.10 of the Updated FHSR.

9.1.5 OVERHEAD LOAD HANDLING/HEAVY LOAD SUMMARY

Heavy Loads Summary

During decommissioning the control of heavy loads will be primarily applicable to loads carried near the spent fuel pool and in the screenhouse. The previous discussion detailed the Big Rock Point Plant heavy load program as applied prior to permanent fuel removal from the reactor vessel. Where applicable the discussion has been modified to reflect the changes in the heavy load requirements due to decommissioning.

In response to Generic Letter 81-07, Control of Heavy Loads, issued by letters dated December 22, 1980 and supplemented February 3, 1981, CPCo reviewed the plant controls associated with overhead load handling versus the Guidelines of NUREG-0612, "Control of Heavy Loads."

CPCo letters dated June 10, July 1, and September 23, 1981 responded to the generic letter. The NRC staff forwarded a draft Technical Evaluation Report (TER) on this issue by letter dated July 2, 1982. CPCo letter dated January 28, 1983 responded to the TER and the response was reviewed within Section 5.3.20 of the May 1984 NUREG 0828 Integrated Plant Safety Assessment Report (IPSAR). CPCo letters dated January 18, 1984 and April 13, 1984 addressed NRC concerns related to the final implementation of Phase I Administrative Controls; use of the TN6/3 and Treat II fuel shipping casks; and use of the screen house trolley and equipment lock crane which were items the NRC incorporated into the May 9, 1984 letter, "Phase I of Control of Heavy Loads - NUREG- 612." That letter included the NRC Safety Evaluation Report (SER) and revised Technical Evaluation Report which described the review and conclusion that implementation of Phase I was completed in an acceptable manner for Big Rock Point.

Phase I included the guidance dealing with administrative controls such as safe load paths and procedures.

Phase II included the guidance on hardware modifications to systems such as interlocks or single failure proof cranes. The information provided by CPCo in the letters above and in subsequent Integrated Assessment - Living Schedule Updates determined no additional Phase II modifications would be required (reference Integrated Plan Update No. 4 dated February 28, 1986). This decision was made based upon the review of Generic Letter 85-11, completion of Phase II of "Control of Heavy Loads," NUREG- 612, in which the NRC determined that based upon the improvements in heavy load handling obtained from Phase I implementation, that Phase II is considered complete. The following sections describe the evaluation and analysis conclusions resulting from the Phase I comparisons to the NUREG-0612 guidelines.

9.1.5.1 Overhead Load Handling Systems - Cranes, Hoists, Lifting Devices (CLP) System

NOTE: FC-702 removed the equipment lock crane so any discussion pertinent to it is for information only.

CPCo letter dated July 1, 1981 provided a description of the cranes, hoists, and lifting devices in use at BRP. The following overhead handling systems were determined to be subject to the general guidelines of NUREG-0612 for Control of Heavy Loads:

- reactor crane - 75T
- reactor auxiliary hoist - 5T
- reactor depressurization system hoist - 2T
- cleanup demineralizer hoist - 3T
- SRV hoist - 1/2T
- emergency condenser beam - 1T
- turbine crane - 25T

In addition, portable gantries capable of lifting several tons are located in the reactor building laydown area and the emergency condenser level. A 500 lb winch is located on the bridge over the fuel pool.

CPCo excluded the following load handling systems from the guidelines of NUREG-0612 on the basis provided below:

- decontamination room hoist - variable
- equipment lock crane 75T
- screen house trolley - 2T
- machine shop trollies - variable
- CRD hoist and trolley - variable

Decontamination Room Number 121 Hoist

This electric hoist is located in the turbine building and was excluded because failure of this device will not result in damage to fuel or safe shutdown equipment. Station power cables exist around the upper perimeter of the room but at an elevation over which the hoist cannot carry loads due to welded mechanical blocks. For decommissioning this hoist cannot affect equipment associated with the safe storage of spent fuel in the fuel pool.

Equipment Lock Crane (Partial Exclusion)

During decommissioning, this crane was removed permanently from the site.

Screen House Trolley

This section previously discussed the equipment located within the screenhouse that could affect plant operation or safe shutdown should damage occur due to a heavy load drop. The discussion has been modified to remove reference to systems and system functions that were required to support power operations or safe shutdown.

A load drop in the screen house could result in a loss of the condenser circulating water pumps, the fire protection pumps and the service water pumps. Containment boundaries are unaffected by a load drop in this area. The diesel generator is located within the screen house but in a separate room, not in the vicinity of the overhead trolley. As stated previously, pool cooling can be lost for 72 hours without the pool water temperature exceeding 150°F. Adequate time is available to restore cooling or to provide cooling water to the pool.

Exclusion of the screen house trolley from the guidelines of NUREG-0612 was considered by CPCo in the April 13, 1984 submittal and in the May 9, 1984 NRC SER.

As discussed in the April 13, 1984 submittal, safe load paths are not defined for the screen house trolley due to the layout of equipment in the screen house. That is, a safe bad path which would minimize the potential for heavy loads, if dropped, to impact plant equipment located in the screen house, cannot be defined. Instead, the procedure which controls the use of the screen house trolley for lifting heavy loads requires the trolley operator to notify the Shift Supervisor should a heavy object be dropped. In this eventuality, the heavy loads procedures requires that an inspection be performed.

Consumers Energy considers that the screenhouse trolley satisfies the intent of NUREG-0612, guidelines 5.1.1(1).

Based on a review of the April 13, 1984 submittal, the NRC staff concluded that the trolley meets the intent of the applicable guidelines of Section 5.1.1 of NUREG-0612 subject to the Safe Load Path limitation identified above. In view of these considerations, the staff concluded that the operation of the screenhouse trolley for handling heavy loads was acceptable.

Machine Shop Trolleys

No equipment required for decommissioning lies beneath the load paths of the machine shop trolleys with the exception of the service water and circulating water supply and discharge piping buried nine feet below the machine shop floor. Service water, demineralized water and fire system piping can be found around the shop perimeter but none of this equipment exists under the load paths of the trolleys.

CRD Hoist and Trolley

This hoist is located in the control rod drive access room. Equipment utilized to support cooling the spent fuel during decommissioning is not located in this room.

9.1.5.2 Overhead Load Handling Safe Load Paths

The discussion in this section included consideration for heavy loads carried over the reactor vessel and safe shutdown equipment. For decommissioning these considerations are no longer applicable and have been deleted from the discussion.

(Reference NRC SER dated September 28, 2001) Procedure T7-38, Containment Building Crane (CBC) Inspection and Heavy Load Control, describes heavy load handling processes, parameters and limitations to ensure continued safety of the spent fuel pool. If a load is not bounded by T7-38, a load drop analysis must be done in accordance with NUREG-0612.

In addition to the safe load path information addressed in 9.1.5.1 above, safe load paths and restricted areas are designed to minimize the potential for damage to the spent fuel should a heavy load be dropped. Lifts within restricted areas are procedurally controlled.

Volume 3 Operating Procedure, "Control of Heavy Loads," reflects the requirements for lifting heavy loads.

Load Path Marking

Specified areas which must be avoided during reactor crane operation are identified and the procedure requires the crane operator to always have someone in the immediate vicinity of the load to direct the load away from those areas which must be avoided.

The restricted area applied to the turbine crane during power operations is not applicable during decommissioning.

All other cranes and hoists which lift heavy loads have a very limited area of coverage. The swinging jib cranes cover a small arc and the monorails are restricted to a line path. Therefore, it is impractical to mark the floor under them.

The use of a designated individual to direct the loads in containment provides a suitable visual aid to the crane operator in lieu of marking load paths on the floor. The procedure for Control of Heavy Loads ensures this individual is sufficiently knowledgeable of the safe load paths in order to direct the load handling evolution and that his duties and functions are formally delineated. CPCo's commitment to provide such an individual satisfies this requirement.

The load path for the W100 Transfer Cask, where it traverses the reactor deck, will be marked (reference NRC Submittal letter dated July 13, 2001.)

The implementation of safe load paths for cranes identified in Section 9.1.5.1 above is consistent with NUREG-0612 Section 5.1.1(1).

9.1.5.3 Overhead Load Handling Procedures

A Volume 3 Operating Procedure for "Control of Heavy Loads" provides a controlled method to cover handling operations for heavy loads that are or could be handled in proximity to irradiated fuel and equipment needed for safe storage of the fuel. Heavy loads at Big Rock are defined as any load weighing more than 500 pounds. (Per NUREG-0612 any load weighing more than a fuel assembly and its handling device.)

NOTE: In proximity to irradiated fuel means in areas where load drops have the potential to adversely affect spent fuel or structures which contain spent fuel.

Other load handling procedures are in place to assure the required maintenance, rigging, inspection, and test activities are performed.

Procedures involved in the handling of heavy loads with the potential to affect the spent fuel will include the identification of required equipment, inspection and acceptance criteria required before movement of load, steps and proper sequence to be followed in handling the load, safe load path, and other special precautions.

Deviations from the routing and load restrictions stated within the procedure for control of heavy loads requires analysis and the use of an approved procedure. Prior to decommissioning a requirement for Reactor Engineer and Shift Supervisor approval of safe load path deviations had been established. With the installation of the single-failure-proof crane (FC-706), safe handling of heavy loads will be ensure via approved plant procedures.

The operating procedures have been revised to restrict lifting any load greater than 105 tons over the spent fuel pool based upon the structural analysis of the containment building crane for seismic events. Use of this crane for loads greater than 105 tons over the fuel pool is prohibited when spent fuel is stored in the pool.

All load handling procedures including the heavy load operating procedure were reviewed relative to NUREG-0612 Guidelines. CPCo considers that these procedures satisfy the Guideline 5.1.1(2) (reference CPCo April 13, 1984 letter).

CPCo May 8, 1981 letter committed to not move the Fuel Transfer Cask (24 ton) during reactor power operation, this commitment is not applicable during decommissioning as the reactor will not return to power operation and the fuel transfer cask was removed.

9.1.5.4 Crane Operator Training

Crane operator training, qualification, and conduct were reviewed and found to be in compliance with ANSI B30.2-1976, "Overhead and Gantry Cranes," except for Crane Operator Visual Examination. (Reference CPCo letter dated June 10, 1981 and July 1, 1981.) Subsequent to these reference letters, CPCo updated the crane operator visual examinations to be in accordance with ANSI B30.2-1976. Based upon this update, BRP complies with NUREG-0612 Guideline 5.1.1(3) and meets ANSI B30.2-1976 Chapter 2-3 "Qualifications for Operators." The BRP single-failure-proof crane is controlled only from one operator controller.

9.1.5.5 Special Lifting Devices

The special lifting devices for the FuelSolutions™ dry fuel storage transfer cask and components are designed and fabricated to the requirements of Section 6 of ANSI N14.6-1993. The lifting yoke is designed, fabricated and tested as a non-redundant special lifting device for critical lifts, in accordance with NUREG-0612 and ANSI N14.6. The trunnions are designed with factors of six and ten on yield and ultimate strength, respectively. The design is based on dead load plus handling load (15% of the dead load.) Reference Consumers Energy letter to the NRC dated August 2, 2001.

Special lifting devices as defined in ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More For Nuclear Materials," was reviewed.

By letter to the NRC dated January 28, 1983, CPCo made a commitment to require contractors to use special lifting devices that comply with ANSI N14.6-1976 formovement of heavy load casks in areas defined by NUREG-0612.

9.1.5.6 Overhead Load Handling Slings

Slings used with the single-failure-proof 105 ton crane hook are selected based on the following criteria: (Reference Consumers Energy letter to the NRC dated August 2, 2001)

1. If the evaluation criteria of NUREG-0612, Section 5.1 can be met for the drop of the heavy load, the slings must comply with ANSI B30.9-1996.

2. If the evaluation criteria of NUREG-0612, Section 5.1 cannot be met for the drop of the heavy load, one of the following two options must be complied with: Redundant slings or the slings are selected using twice the sum of the static and maximum dynamic load (based on a maximum hoist speed of 3 feet per minute.) For either option, the slings must meet the requirements of ANSI B30.9-1996.

Slings used in the handling of heavy loads are inspected periodically and prior to use to comply with the ANSI-B30.9-1971 requirements. These slings have been numbered and rated for the sum of static and dynamic load if the dynamic load amounts to more than 10% of the static load. When a given sling is used with more than one crane, the crane with the fastest hoist speed will be used to determine dynamic load. Slings used only on the big hook of the reactor crane will be marked for static load only. The 3-fpm (foot per minute) speed on this hoist results in a minimal dynamic load.

Those slings restricted in use to certain cranes are clearly marked and comply with ANSI B30.9-1971. Slings are inspected prior to use and in the daily inspection checklist. Personnel performing rigging activities are trained with respect to inspection and acceptance criteria and proper rigging practices.

Thus, slings are used in accordance with ANSI B30.9-1971 which meets the intent of NUREG-0612 Guideline 5.1.1(5). (Reference CPCo January 28, 1983 submittal.)

9.1.5.7 Crane Inspection, Testing, and Maintenance

Crane inspection, testing, and maintenance were established in accordance with Michigan Occupational Safety and Health Administration (MIOSHA) standards which are comparable to the ANSI B30.2-1976, Chapter 2-2 standards with the exception of inspection intervals.

By letter dated July 1, 1981, CPCo revised the reactor semi-gantry crane inspection intervals to meet the monthly and yearly requirements; the turbine building crane is inspected on quarterly and yearly intervals due to their infrequent service; and inspections prior to use are conducted on all cranes in accordance with the ANSI standard. Crane testing per the requirements of the ANSI standard have not been required as they only apply to new, reinstalled, altered, extensively repaired or modified cranes. BRPs cranes did not fall within these categories. Crane maintenance as required by ANSI B30.2 has been included as part of the inspection program. Although not specifically covered under the ANSI standard, the remaining hoists and lifting devices used in the handling of heavy loads were reviewed and the applicable inspection, testing and maintenance requirements were invoked on them as well.

The 125 ton (design rated load), 105 ton (maximum critical load) crane inspection, testing, and maintenance will be performed in accordance with the requirements of ANSI B30.2, MIOSHA standards, and NUREG-0544 before handling loads over the spent fuel pool

(Reference NRC SER dated September 28, 2001) In the event that the single-failure-proof containment building crane is used for fuel handling for a period of greater than four years, non-destructive examination (NDE) will be performed in accordance with the requirements of NUREG-0544.

Rated load test documentation requirements of ANSI B30.2-1976, Subsection 2-2.2.2, for all cranes which lift heavy loads is only available for the Reactor Crane and RDS Jib Crane as follows (reference CPCo January 28, 1983 submittal and September 28, 2001, Defueled Technical Specification Amendment 122)

- Reactor Crane

  The single-failure-proof containment building crane initial load test met the requirements of ANSI B30.2-1996. This lift was documented in Facility change FC-706, Containment Building Crane.

- RDS Jib Crane Telescoping Beam

  The RDS Jib Crane was modified via Facility Change FC-634. The modification involved the addition of a one ton capacity telescoping beam assembly to enable the capability to extend to the farthest RDS valve. Upon completion of the telescoping beam installation the beam was fully extended and load tested to 2500 pounds as documented in the FC design package. This work was completed to ANSI B30.11-1980, Section 11-1.3.5 Guidelines for Monorails and Underslung Cranes."

  When the hoist assembly is mounted on the bottom boom, this crane is rated at 4000 pound capacity. Documentation of the RDS Jib Crane load test to 5000 pounds is in the RDS General Work Order Package.

All other cranes which lift heavy loads have repeatedly lifted the loads identified as heavy loads and, as verified by subsequent inspection, have not been adversely effected. No loads weighing more than the rated load of these other cranes will be lifted without preparing an initial load test as recommended by ANSI B30.1-1976.

Based upon the above considerations, CPCo crane inspection, testing, and maintenance is considered to be consistent with NUREG-0612 Guideline 5.1.1(6). (Reference NRC May 9, 1984 SER.)

9.1.5.8 Reactor, Loading Dock, and Turbine Building Crane Design

NOTE: FC-702 removed the equipment lock crane so any discussion pertinent to it is for information only.

CPCo, "Report on the Analysis and Evaluation of the Consequences of Postulated Fuel Cask Drop Accidents," July 1, 1974 provided information on the reactor and turbine cranes. These cranes were specified and designed to comply with Specification #49 of the Electric Overhead Crane Institute (EOCI-49). A detailed comparison was made between the requirements of that standard and those of the current standards specified in NUREG-0612 Guideline 7. Additional information was provided concerning the equivalency of actual design features with the requirements of these later standards, in areas associated with structural and mechanical reliability, where specific compliance was not apparent. (Reference NRC May 9, 1984 SER.)

The standards used in designing and constructing these cranes were compared to AWS D14.1 for welding, AGMA standards for gear design, and CMAA-70 for gantry leg structural design. In all cases, except for some written procedure and personnel qualification requirements of AWS D14.1, the criteria used meet or exceed the requirements of the above standards. The 125 ton (design rated load), 105 ton (maximum critical load) crane meets or exceeds the requirements of the above standards.

The original reactor building and turbine cranes were designed and procured prior to the publication of the standards identified in Guideline 7. Since these standards were not invoked in the original design specification, it is not feasible, in many cases, to determine unequivocally that a specific requirement of these latter standards has been satisfied. Consequently, design features associated with load handling safety have been reviewed and compared with applicable current requirements employing engineering judgement, where appropriate, to determine if the intent of the current standard has been satisfied.

CPCo considers that these cranes, although not in verbatim compliance, possess a degree of load handling reliability consistent with the intent of Guideline 7.

NRC May 9, 1984 SER Conclusion

Design of the Big Rock Point reactor building and turbine cranes is consistent with the criteria of Guideline 7, NUREG-0612, Section 5.1.1(7), with respect to load handling reliability on the basis of a detailed comparison of the requirements of then-current design standards (EOCI-49 and other industry standards) and actual design features with requirements of CMAA-70.

FC-706, Containment Building Crane, removed the original 75 ton crane and gantry. The replacement 125 ton single-failure-proof crane was designed to meet the applicable criteria and guidelines of CMAA Specification 70 and ASME B30.2 – 1976 and NUREG-0544.

Facility Change FC-706 – Containment Building Crane

In 2001, the 75 ton crane was removed and replaced with a single-failure-proof crane having a design rated load of 125 ton and a maximum critical load rated at 105 ton. This crane meets the requirements of NUREG-0544, "Single-Failure-Proof Cranes for Nuclear Power Plants."

NRC September 28, 2001 SER Conclusion

Load drop analyses for heavy loads, other than casks, will be completed in accordance with NUREG-0612. No casks will be moved over fuel.

Replacement of the existing reactor crane at Big Rock Point with a single-failure-proof containment building crane was in accordance with General Design Criteria 61 and meets the applicable portions of NUREG-0800, Regulatory Guide 1.13, NUREG-0612, NUREG-0544, Generic Letter 85-11 and NRC Bulletin 96-02.

CPCo Addition for the Turbine Crane

Additional design information for the 25 ton Turbine Crane was submitted as an attachment to our January 28, 1983 Response to the NRC Draft Technical Evaluation. The specification provided supports the NRC SER conclusions.

9.1.5.9 Interim Protection Measures (IPM) for Heavy Load Handling

NUREG-0612, Control of Heavy Loads, required certain Interim Protection Measures (IPM) to be assessed for implementation at operating reactors.

The NRC May 9, 1984 Safety Evaluation Report (SER) and attached consultant Technical Evaluation Report (TER) provided the results and conclusions to support completion of the Phase I of Control of Heavy Loads for BRP. The TER indicated that the plant complies with the measures for interim protection with one exception, which was the IPM-1 requirement to provide Technical Specifications (T.S.) to prohibit handling of heavy loads over the fuel in the fuel storage pool unless the crane is single-failure-proof.

With the installation of the single-failure-proof crane (reference FC-706) during decommissioning, the discussion on the acceptability of using the 75 ton crane with the 24-ton fuel transfer cask over fuel in the spent fuel pool has been deleted.

9.1.5.10 Radwaste (Demin) System Trolley

During Plant decommissioning, a 4-ton chain hoist and trolley were installed to facilitate the handling of the cask and components for the replacement demineralizer, filters and piping for liquid radwaste processing (Reference MA-98-0033). This hoist is located on the ground floor of the Turbine Building (area formerly referred to as the Condensate Pump room).

Failure of this device will not result in damage to spent fuel, thus it is excluded from the guidelines of NUREG-0612.

9.1.6 HEAVY OBJECT MOVEMENT

CPCo submittals dated July 1 and September 23, 1981 provided a listing of typical loads lifted at BRP.

9.1.6.1 Heavy Object Movement Analysis

As part of the Spent Fuel Rack Addition Consolidated Environmental Impact Evaluation and Description and Safety Analysis, April 1982 as amended, a Heavy Object Movement Analysis was presented. The issues this analysis addressed have been deleted due to the replacement of the 75 ton crane with a single-failure-proof crane during decommissioning.

9.1.7 CASK MOVEMENT/DROP ANALYSES

Refer to Chapter 15, Section 15.10 for a discussion of the fuel handling accidents applicable during decommissioning.

With the installation of a single-failure-proof crane, cask movements on or near the Spent Fuel Pool will be administratively controlled to ensurestructural damage will not result from a single failure. Although a single failure will not result in a loss of load, a slight (less than 18 inches) downward movement will occur prior to the emergency hoist drum brake stopping the load movement. With the maximum critical load (dry fuel transfer cask) attached, the SFP floor, Room 444 and Elevation 599'5" can withstand this downward movement; however, the elevation 632'6" (reactor deck) floor cannot. Thus, procedures will require having the dry fuel transfer cask at least 18 inches above this floor during handling movements. When handling the dry fuel transfer cask, the load path must preclude going over fuel. The handling of other heavy loads using the main hoist should use a load path that avoids going over fuel. In those cases where a heavy load must be handled above the fuel, compensatory measures and justification must be provided to ensure compliance with the requirements of UFHSR Section 9.1.5. Because the 5 ton auxiliary hoist is not single-failure-proof, the handling of heavy loads is allowed provided the load is analyzed and shown to satisfy the evaluation criteria of NUREG-0612, Section 5.5.

9.1.7.1 Fuel Transfer Cask

The discussion in this section was applicable to restrictions on moving the fuel transfer cask while in power operations. Since the plant is permanently defueled, the restriction is no longer applicable and the discussion has been deleted. The fuel transfer cask has been removed.

9.1.7.2 Cobalt Cask

The discussion in this section was applicable to restrictions on moving the cobalt cask while in power operations. Since the plant is permanently defueled, the restriction is no longer applicable and the discussion has been deleted.

9.1.7.3 Treat II Cask and TN6/3 Cask ~7½ Ton

The discussion in this section was applicable to restrictions on moving either of these casks while in power operations. Since the plant is permanently defueled, the restriction is no longer applicable and the discussion has been deleted.

9.1.7.4 CNSI 1 - 13G Cask ~12 Ton

The discussion in this section was applicable to restrictions on moving the CNSI 1 cask while in power operations. Since the plant is permanently defueled, the restriction is no longer applicable and the discussion has been deleted.

9.2 WATER SYSTEMS

Auxiliary cooling water and other water systems are shown on Drawings D740G40111 and 0740G40118. This section will provide information on the following systems or features:

Station Service Water System (SWS)

Demineralizer Water Makeup System (DMW)

Well Water System (WWS)

Domestic Water System (DWS)

Ultimate Heat Sink

Condensate Storage Facilities

Sanitary Water Disposal

Sewage and Chlorination System (SEC)

Laundry Water Services

9.2.1 SERVICE WATER SYSTEM (SWS)

9.2.1.1 Service Water System Description (Reference CPCo December 10, 1981 and July 28, 1982 letters)

The service water system is an open system which takes its supply from Lake Michigan. This water is supplied by two full capacity vertical turbine type centrifugal pumps located in the screenhouse, Reference Drawings D740G40111 and 0740G40141. The pumps each have a rated capacity of 2100 gallons per minute with a differential head of 88 feet. Pump suction is from the center bay of the intake structure (refer to Section 10.4.4 of this Updated FHSR for a description of the intake structure bays).

One pump supplies the normal needs of the system. The pumps discharge into a common header which contains a dual basket strainer which further strains the raw water. The pumps are normally powered from 480 Volt Motor Control Centers. If header pressure drops, the remote monitoring station will receive an alarm/indication and operations will be required to start the alternate pump.

The system provides the water to various coolers, heat exchangers, and other components as follows:

- Radwaste Batching (Dilution) to Discharge Canal
- Turbine Building Air Coolers
- Reactor Enclosure Air Coolers (Pipeway Coolers) (2)
- Alternate Spent Fuel Pool Cooling Hx (2)

- Chlorination System in the Screenhouse
- Condenser Circulating Water Pump Shaft Seals (2)
- Heating Boiler Blowdown Tank
- Priming Water to Main Diesel Generator Cooling Water Pump
- Washdown Connections, Emergency Showers, and miscellaneous sample coolers.

A manually operated tie valve between the fire protection system and the SWS, located in containment, is provided to assure the availability of cooling water to the spent fuel pool cooling heat exchangers in the event that the service water system fails. The manual valve is normally locked closed. A check valve in the service water supply piping to the spent fuel pool cooling heat exchangers will prevent fire system flow back through the service water header to any of the other equipment in the turbine building serviced by the SWS.

The service water return from containment is to the lake via the discharge canal where the water mixes with the discharge from the circulating water pump, if the circulating water pump is running. SWS bypass lines in the Turbine Building have been installed by minor alterations. For further information on the monitors, refer to Chapter 11 of this Updated FHSR.

9.2.1.2 Service Water System Design

Service water piping material is seamless carbon steel, ASTM A-53, Grade A. The pipeway air coolers are Admiralty metal. Stainless steel tubing is used in the reactor building heating and cooling heat exchangers, reference Specification Field Changes SFC-83-003 and SFC-83-051 for tube replacement.

All coolers within containment supplied by the service water system are provided with manual isolation valves. Single failure criteria was not considered in system design.

9.2.1.4 Service Water System Evaluations

The service water system has been evaluated in response to NRC IE Bulletin 80-24, November 21, 1980, "Prevention of Damage Due To Water Leakage Inside Containment;" as part of the Systematic Evaluation Program (SEP) Topic IX-3, "Station Service and Cooling Water Systems," reference NRC Safety Evaluation Report (SER) dated July 20, 1982; and as a result of Three Mile Island (TMI) NUREG-0737 Section II.K.3.20, "Loss of Service Water," reference NRC Safety Evaluation Report dated June 16, 1981. The following provides results and conclusions from these evaluations.

9.2.1.4.1 Prevention of Damage Due To Water Leakage Inside Containment

CPCo letters dated December 23, 1980 as corrected January 29, 1981 and revised July 28, 1982 determined that the only "open" cooling water system within the scope of concern was the service water system. An "open" system utilizes an indefinite volume, such as a lake, so that leakage from the system could not be detected by inventory decrease.

CPCo concluded that existing procedures and detection systems are sufficient to preclude undetected leakage in containment (refer to Section 5.2.5 for leak detection systems).

9.2.1.4.2 Systematic Evaluation Program (SEP) Topic IX-3 - Station Service Water

The discussion included in this section addressed the fact that the SWS was not credited for mitigative purposes for loss-of coolant accidents. It was also stated that SWS was safety related with respect to containment integrity. The discussion has been deleted.

For decommissioning the loss of coolant accidents are no longer applicable and the service water system interface with containment is no longer considered safety related.

9.2.1.4.3 Loss of Service Water Evaluation

This evaluation verified the acceptability of the loss of service water on safe plant shutdown. As of December 5, 1997 it has been determined that the spent fuel pool water temperature will not exceed 150°F for the first 72 hours following loss of spent fuel pool cooling. This is considered adequate time in which to recover spent fuel pool cooling.

The remaining discussion included in this section dealt with the effects of a loss of service water of safe plant shutdown and has been deleted.

9.2.2 COOLING SYSTEM FOR REACTOR AUXILIARIES

Cooling of reactor auxiliaries is no longer required; therefore, this section has been deleted. Cooling of spent fuel is described in Section 9.1.3 of this Updated FHSR.

9.2.3 DEMINERALIZED WATER SYSTEM

Information on the Condensate and Make-Up Water Demineralizers is provided in Section 10.4.5 of this Updated FHSR.

The system is shown on Drawings 0740G40110 and 0740G44011. The demineralized water pump is driven by a 480 volt, 3 hp motor. The pump takes its suction from the 6700 gallon (nominal) demineralized water storage tank.

9.2.4 WELL WATER SYSTEM (WWS) AND DOMESTIC WATER SYSTEM (DWS)

The well water system provides potable water for the domestic water system, the make-up demineralizer system, and for priming the main Diesel Generator cooling water pump. Refer to Drawings 0740G40118 and 0740G44022 for details.

Domestic water is supplied by a 300 foot deep well pump located approximately 800 feet east of the plant. The well water pump supplies water to the 5000 gallon well water storage tank. A domestic water transfer pump draws water from the storage tank and delivers it to the domestic water accumulator located in the screenhouse. The accumulator maintains domestic water system pressure of approximately 80 psig. The accumulator is pressurized by an air compressor located in the screenhouse, and provides an inventory of water for normal system demands.

9.2.5 SANITARY WATER SERVICES

The sanitary water services are those related to the Sewage and Chlorination (SEC) System and water for Laundry Services.

9.2.5.1 Sanitary Waste Disposal

The sanitary waste system collects wastes from plant buildings and conveys them by gravity into septic tanks or chambers. From the tanks or chambers, pumps deliver the effluent to leaching fields west of the plant. Refer to Drawing 0740G82015.

The original plant sanitary waste disposal system was modified via Facility Change FC-432 Water Quality Control Project to meet 1977 disposal permit requirements.

9.2.5.2 Laundry Water Service

A laundry facility is provided for washing and drying articles of clothing that are used in radiological controlled access areas of the plant.

The laundry facility was relocated from the access control area to an area in the turbine building above the decontamination room. This change was made to reduce the background radiation levels in access control. The modification was accomplished via Facility Change FC-382 in which the original laundry waste tank was disconnected, filled with six inches of concrete, and capped.

Water for the laundry is from the well water via the domestic water system. Water from the laundry is collected in the Chemical Waste Receiver tank and monitored. Depending on the activity level or composition, the laundry wastes will be either processed or discharged to the circulating water discharge canal. Refer to Drawings 0740G40108 and 0740G40132.

An off-site laundry service cares for the majority of protective clothing. Contaminated personal clothing is laundered on-site.

9.2.5.3 Chlorination System

The Chlorination System was deleted with FC-633.

9.2.6 ULTIMATE HEAT SINK

The ultimate heat sink (UHS) for the plant is Lake Michigan. Evaluations for the loss of the UHS are discussed in Section 2.4.5 and 2.4.6. Evaluations for the structures relating to the availability and protection of the UHS are discussed in Section 3.4.3.

9.2.7 CONDENSATE STORAGE FACILITIES

A 25,000 gallon capacity condensate storage tank is shown on Drawings 0740G40110 and 0740G44011. The aluminum tank is located outside of the turbine building. Tank level is normally maintained by processing water from radwaste via the waste hold tanks. Tank overflow is piped back to radwaste to the clean waste receiver tanks. Demineralized water can be added from the 6700 gallon (nominal) capacity demineralized water storage tank.

9.2.8 BIOLOGICAL CONTROL SYSTEM

This system involves injecting a solution of sodium hypochlorite into the service water bay at the screenhouse structure to control zebra mussel infestation within the service water system. Chlorine concentrations are desired for effective control. The system does not require dechlorination, depending on the larger untreated circulating water flows for dilution at the discharge.

The main mechanical components are a 1600 gallon storage tank, a chemical metering pump and associated piping which were installed via Facility Change FC-633.

9.3 PROCESS AUXILIARIES

9.3.1 COMPRESSED AIR SYSTEM

Decommissioning air (CAS) provides compressed gases for service use and moisture free compressed air for control air demands. Compressed nitrogen gas or air cylinders are used for air operated devices that have low demand or are remote from the decommissioning air supply piping. The compressed air is supplied by motor driven, non-lubricated air compressors rated at 100 psig. Each compressor has its own receiver tank. A dryer is provided in the decommissioning air supply line to prevent freezing. The screen structure air supply is separated from the rest of decommissioning air. A compressor installed in the screen structure supplies air requirements for equipment located therein and is not dried since the area is heated. The system is shown in Drawing 0740G40133.

CPCo letter dated February 20, 1989 responded to Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment." The Generic Letter requested CPCo review NUREG-1275, Volume 2 and perform certain actions to verify design and operation of the Big Rock Point instrument air system. These actions are no longer required for the decommissioning air system.

During decommissioning there are no safety related components requiring instrument air (decommissioning air). The discussion related to the operation of the instrument air system and its effect on safety related components is no longer applicable and has been segregated for decommissioning.

9.3.2 PROCESS SAMPLING SYSTEM

The Liquid Process Monitoring System is shown on Drawing 0740G44021. Details on this system are provided in Chapter 11 of this Updated FHSR. Details on sampling system operability, testing, and limiting conditions for operations are addressed in the Offsite Dose Calculation Manual.

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Floor drains that drain directly to the discharge canal were identified as a result of a CPCo January 31, 1975 uncontrolled release of low activity liquid to the canal.

Plant administrative controls have been established to control normal status (plugged or unplugged) of drains outside of containment. Caution signs are placed near selected unplugged drains to prevent inadvertent releases. Any time a drain is changed from its normal (plugged or unplugged) condition, the system status is changed accordingly and controlled by Plant Operations personnel to prevent flooding or uncontrolled release.

9.3.4 STANDBY LIQUID CONTROL SYSTEM

The Liquid Poison System (LPS) is the Standby Liquid Control System at Big Rock Point. Refer to Chapter 4, Section 4.8 for information on the LPS. The discussion in Section 4.8 has been deleted for decommissioning since the LPS does not support safe storage of fuel in the spent fuel pool.

9.4 HEATING AND VENTILATING SYSTEM (VAS)

This original section described the heating and ventilation system (VAS) as constituted during power operation. Discussions relating to power operations and not appropriate to decommissioning have been deleted. The installed VAS will undergo change as the decommissioning progresses. Functions will be maintained either by portions of the current installed system, or by temporary equipment/systems, as follows:

- Airborne gaseous effluent release control and monitoring when activities produce the potential to create airborne radioactivity release.

  The capability to control airflow from the reactor building, the turbine building, and the liquid radioactive waste vault to the environment through monitoring pathways is provided when activities in these areas have the potential to create airborne radioactivity release. Pressure retention capability is not required. This does not preclude removal of existing penetrations or making temporary penetrations provided that he opening can be closed in a timely manner, or a net positive inflow of air can be demonstrated.

  Gaseous effluent paths are identified in Figure 1-1 of the Offsite Dose Calculation manual (ODCM), Volume II. Monitoring is provided at the stack (or other release point) for the reactor building, liquid radwaste vault, and other miscellaneous areas of the Service and Turbine Buildings which may generate airborne radioactivity.

  The radioactive waste storage building contains its own heating and roof ventilation. No radioactive waste processing is performed in this facility. Consequently, no effluent monitoring is required. If this or any other facility (including work enclosures) process waste with the potential to create airborne radioactivity release, and is not serviced by the plant VAS, effluents to the environment will be independently monitored and controlled in accordance with ODCM requirements.

- Worker protection from airborne radioactive and hazardous non-radioactive materials during the course of decommissioning.

  Engineering controls, especially ventilation controls, are the first line of defense for worker protection against airborne inhalation hazards. However, flow patterns from the installed VAS applicable to power operation are not necessarily beneficial for all decommissioning activities. Consequently, temporary confinement tents and other enclosures are used where such use is beneficial to worker protection. Such enclosures are ventilated and filtered as appropriate for protection of workers within the enclosure, as well as those who may be exposed to exhaust emissions. Exhaust from such enclosures may be routed directly to the environment if potential for radioactivity release and other environmental hazards are not present, or if monitoring and control is provided in accordance with the ODCM. However, exhaust is normally collected by the VAS.

- Heating sufficient to assure a safe (non-freezing) environment for the spent fuel pool, until such time as all fuel is removed to dry fuel storage.

  Heating system is supplied to various parts of the plant by 15 psig oil fired package boiler. Remote area heating is provided for certain areas by electrical unit heaters, LP gas infrared heaters, or furnaces. Upon removal of spent nuclear fuel, heating will continue primarily for worker comfort, water supply availability, and other standard industrial considerations.

9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM

The control room has been dismantled. This section has been deleted.

9.4.2 SPENT FUEL POOL VENTILATION SYSTEM

The spent fuel pool ventilation system is part of the reactor containment ventilation system. The function of the spent fuel ventilation system is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access and control airborne radioactivity in this area.

The spent fuel pool ventilation system has an exhaust air vent located at the top edge of the fuel pool. The fuel pool area vents to the stack through the exhaust system, which is a draft induced system. Reference Drawing D740G40125 for schematic representation of spent fuel pool area ventilation system.

9.4.3 RADWASTE AREA VENTILATION SYSTEM

The radwaste area ventilation system is supplied from the condensate pump room ventilation system, which is part of the turbine and service building ventilation system. Exhaust is induced by the stack exhaust fans and is then vented through the stack. The condensate pump room has been dismantled. Detailed description of the original system has been deleted. Radwaste area ventilation to a monitored pathway currently is provided by a vent path through the plant stack. During radwaste area dismantlement, ventilation and monitoring may be supplemented or replaced by temporary systems.

9.4.4 TURBINE AND SERVICE BUILDING VENTILATION SYSTEM

The turbine and service building ventilation system is composed of three air supply systems. These include the condensate pump room heating and ventilation system, the shop heating and ventilation system and the equipment room cooling system (Note: The cooling coil for this unit was abandoned in place).

The section included a discussion of turbine and service building ventilation following a design basis accident (for power operations). The systems were not required to function following an accident. The discussion is not applicable during decommissioning and has therefore been deleted.

9.4.5 ENGINEERED SAFETY FEATURES VENTILATION SYSTEM

The reactor was permanently defueled on September 20, 1997 when the last fuel assembly was removed from the reactor vessel. With permanent defueling of the reactor came the elimination of equipment designated as an Engineered Safety Feature (ESF). The ventilation of the equipment previously classified as an ESF is therefore not applicable during decommissioning. The discussion included in this section has been deleted in its entirety.

9.4.6 CONTAINMENT SPHERE VENTILATION

The following information has been extracted from information which was provided in proposed Technical Specification Change 32 submitted June 30, 1972 in response to a March 28, 1972 NRC request. The design information was provided in support of leak detection limits for primary coolant recirculation system. Information previously present in this section which is not directly applicable to decommissioning has been deleted, and discussion of the stack HEPA filter, installed specifically to support decommissioning activities, has been added.

Ventilation air to the reactor building is supplied at design rates varying from 0 cfm to 14,500 cfm and is so controlled as to maintain a slight negative pressure within the containment sphere.

The ventilation system is a forced-induced system, the stack exhaust fans acting as the induced fans to draw containment sphere exhaust air through the 24" exhaust duct. Two full-capacity ventilation supply air fans are provided as forced draft fans and discharge ventilation air directly into the general areas of the containment sphere.

A ventilation building, or air shed, attached to the containment sphere in a line between the containment sphere and stack, contains the outdoor air louvers, filters, and air heating coils for tempering incoming air. Each supply air fan suction has an open-shut damper and inlet vanes controlled by the differential pressure existing between the inside and outside of the containment sphere. Ventilation airflow through the reactor building rooms and passages is equal to the induced draft exhaust flow to the building exhaust plenum created by the plant exhaust fans in the stack.

A High Efficiency Particulate Activity (HEPA) filter has been added for use when decommissioning activities have the potential to release significant quantities of radioactive material to the environment (Reference Section 11.3). Examples which were identified as having potential for such releases were chemical decon, reactor vessel segmentation via plasma arc cutting, demolition of the reactor vessel concrete shielding, and plasma arc cutting of the steam drum. Subsequently, steam drum cutting was eliminated from this list due to the reduction of activity for steam drum due to the chemical decontamination process. The stack HEPA also is required by the ODCM to be inservice if gaseous effluents are forecast to exceed 2% of the annual limit in a 30-day period. Operationally, HEPA filter use has been found beneficial in reducing particulate loading on the stack monitor particulate filter (thus improving operability of the monitor), and suggests significant reduction in dust emissions to the environment.

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION SYSTEM (FPS) GENERAL

The fire protection system furnishes water to all points throughout the plant area, buildings, etc., where water for fighting fires may be required. Water is provided by an electric driven fire pump rated for 1000 gpm at 110 psig, or during an outage of the electric pump, a full capacity standby diesel driven pump. Normally, pressure is maintained in the fire system by a small fire system jockey pump and accumulator tank. Each of the pumps on the fire system takes suction from the circulating water intake structure. The fire pumps start automatically on pressure drop in the system.

Hose houses, hose-racks, automatic sprinkler-heads, and manual fire extinguishers are located throughout the plant.
Refer to Drawing 0740G40123, 0740G40141 and 0740G44019 for the Fire Protection System.

Comprehensive Fire Protection Summary Document

By letter dated February 27, 1987 CPCo submitted a "Summary of Fire Protection Provisions for Big Rock Point Nuclear Plant." The document summarized efforts at BRP to assure safe operation of the facility through compliance with the applicable provisions of fire protection requirements described in 10 CFR 50 Appendix R, Applicable General Design Criteria, and other Requirements/Commitments.

The remainder of Section 9.5 of this Updated FHSR provides details of certain fire protection features at BRP. Further details and analyses may be obtained by review of the "Summary" document which is currently being updated, revised, and controlled as an "Administrative Volume" at the plant.

Certain fire protection requirements in this section of the Updated FHSR were extracted from the Technical Specifications in preparation for their removal from that document utilizing the NRC guidance provided in Generic Letter 88-12 dated August 2, 1988. These requirements were added to the Technical Specifications with Amendment Number 17, issued March 3, 1978 and modified with Amendment Number 25, issued April 4, 1978.

9.5.1.1 Fire Detection Instrumentation

The plant has protective signaling systems that transmit fire alarm signals to the Monitoring Station, where audible and visual alarms are provided. Fire pump running and trouble, and low fire water pressure signals are also provided in the Monitoring Station.

The protective signaling system is supplied backup power from battery packs installed for the control panels, transmitters and receiver, on loss of normal AC, and comply with those provisions of the NFPA Standards which are considered essential for a facility of this type.

Decommissioning Fire Detection equipment (reference Minor Alteration 98-077) allows for detection of fires in the following areas; Area #1: Screenhouse and Diesel Generator Room; Area #2: Radwaste Processing Building: and Area #3: Cable Penetration Room, Cable spreading area in the Electrical Equipment Room, and the Cable spreading area in Containment (including the Alternate Spent Fuel Pool Cooling Skid, FC-699). Detectors are connected to a common alarm panel for each of the above areas. Alarm signals are radio transmitted to a common receiver in the Monitoring Station.

Volume 3, Operating Procedure for the Fire Protection System, will specify the minimum number of detectors required for each area. Should the number of detectors be less than the minimum required, a fire watch patrol to inspect the area will be established within one hour, with hourly inspections thereafter.

The reactor was permanently defueled on September 20, 1997 and will not return to power. Elimination of equipment designated as an Engineered Safety Feature (ESF) has occurred as a result of the permanent defueling. Fire detection devices may be removed as equipment is made available for decommissioning.

Surveillance Required

Each of the above fire detection instruments shall be demonstrated to be operable:

a. Once per six months by a channel functional test and

b. Once per 31 days by verifying proper alignment of power sources to the circuits.

9.5.1.2 Fire Suppression System

The fire suppression system consists of the water system, spray and/or sprinklers and fire hose stations.

9.5.1.2.1 Fire Suppression Water System

Fire protection water is supplied from Lake Michigan through the intake line to the intake bay. From the inlet bay water passes through redundant intake screens to water canals below the screen, well and pumphouse (screenhouse). The clogging of one traveling screen will not interfere with water delivery if the other is clear. The water inlet canals feed the two circulating water pump bays under the screenhouse. Water from each circulating water bay feeds to a central bay from which water for the fire pumps is drawn. Reference Drawing 0740G40141. The diesel and electric fire pumps each draw water from this central bay.

Fire Suppression Water System Operability

The fire suppression water system is required to be operable at all times with both 1000 gpm capacity pumps (electric and diesel) aligned to the fire suppression header and supplying the sprinkler and hose stations described in 9.5.1.2.4 and 9.5.1.2.5 below.

In order to be considered operable, the level of the intake bay must be above 570 feet elevation.

The pumps must be aligned for automatic starting on decaying fire system pressure.

Actions Required for Inoperable Fire Suppression Water System and Inoperable Fire Pumps

a. With the Fire Suppression Water System inoperable, restore the inoperable equipment to operable status within 7 days or, prepare and submit a Special Report to the Commission within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

   If both fire pumps (electric and diesel) or the piping systems are inoperable:

   1. Initiate procedures to provide a backup Fire Suppression Water System within 24 hours by notifying the Charlevoix Fire Department to standby, and

   2. Restore the inoperable fire pump or piping system to operable status within 14 days or, prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the pump or piping system to operable status.

b. This action was applicable to power operations and the tie between the FPS and core spray systems. During decommissioning the actions are no longer applicable and have been deleted.

Fire Suppression Water System Surveillance Requirements

The fire suppression water system will be demonstrated to be operable:

a. Once per 7 days verifying the Intake Bay water level is above 570' elevation.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM-D975-74 with respect to viscosity, water content and sediment, (reference Section 9.5.4.4 of this Updated FHSR).

c. Once per 18 months:

   1. By a system flush and by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   2. Subjecting the diesel driver to an inspection in accordance with procedures prepared in connection with its manufacturer's recommendations for the class of service.

d. Once per 3 years by performing flow tests to meet or exceed the requirements of Section 11, Chapter 5 of the Fire Protection Handbook, 14th Edition published by National Fire Protection Association.

Fire Suppression System Basis

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, and fire hose stations. The collective capability of the fire suppression system is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression system are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt adequate fire suppression capability for the continued protection of the nuclear plant.

9.5.1.2.2 Fire Pumps

Both the electric and diesel vertical centrifugal fire pumps have a rated capacity of 1000 gpm at 110 psig (254 foot head). Appendix A, Item E.2(e) of Branch Technical Position APESB 9.5-1 "Fire Protection Water Supply Systems", requires that the flow rate of the fire system be calculated on the basis of the longest expected flow rate for a period of two hours, but not less than 300,000 gallons (2500 gpm). The operating plant open head Switchyard deluge system which required 1160 gpm at 52 psig combined with 1000 gpm for manual hose streams totals approximately 2160 gpm, therefore the 2500 gpm evaluation criteria was assumed.

Each of the two fire water pumps is capable of delivering approximately 1350 gpm at 72 psig. Therefore a combined pump flow rate of approximately 2700 gpm at 72 psig, available from Lake Michigan is considered adequate to meet this criteria.

The pumps are separated by about 15 feet at their suction lines in the screenhouse water bay and are separated by a sheet metal radiant energy shield.

An electric jockey pump and an accumulator are provided to maintain pressure on the fire water system. The fire pumps are arranged to start automatically when the fire loop pressure drops due to a large water demand.

Fire Pumps Single Active Failure Analysis

There are two redundant fire pumps, one electric driven pump and one diesel driven pump. A single failure in either pump, driver, power supply, discharge check or isolation valve will not affect the redundant pump. A failure of a discharge check valve in the open position will bypass flow from the other pump and may require manual closure of the associated isolation valve.

Certain fire operability, surveillance, and basis requirements for operation are addressed under the Fire Suppression System in 9.5.1.2.1 above.

IE Bulletin 79-15: Deep Draft Pump Deficiencies

In letter dated October 17, 1990, the NRC provided a safety evaluation which concluded that safety concerns regarding the two Worthington fire pumps installed at Big Rock Point were resolved. A review of test data collected from the past fire (5) years showed no signs of performance degradation in either pump thus providing the basis that the Bulletin 79-15 deficiencies did not adversely impact these pumps.

Diesel Fire Pump Surveillance Requirements

The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each battery is above the plates, and

   2. The overall battery voltage is $\geq$ 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

   1. The batteries and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2. The battery-to-battery and terminal connection are clean, tight, free of corrosion and coated with anti-corrosion material.

9.5.1.2.3 Fire Water Piping System

Both electric and diesel fire pumps supply the underground fire main loop by a common 8 inch supply line from the single header. This header was provided with additional supports via Facility Change FC-535 to improve seismic response capability. The seismic capability of certain fire protection equipment and piping are addressed in Table 3-1 found in Chapter 3 of this Updated FHSR. The fire loop supplies the fixed water suppression systems, fire hose stations and exterior fire hydrants.

Sectionalizing valves are provided to allow isolation of various sections of the fire loop. Piping and valving is arranged so that automatic suppression systems and manual fire hose stations can be taken out of service independently for maintenance or repair. A single break in the internal header supplying sprinkler and hose stations could affect both automatic and manual suppression; however, the small size of the plant would permit effective use of hose from exterior hydrants in such an unlikely event.

Fire hydrants are strategically placed around the exterior of the plant. Hydrants are not equipped with auxiliary gate valves but the arrangement of the fire loop and sectionalizing valves is such that any hydrant can be taken out of service for repair or maintenance without shutting off water supply to interior plant suppression systems. Note: Hydrant repair could require isolation of the West Warehouse (stockroom) sprinklers.

A hose house containing 200 feet minimum of 1½ inch coupled fire hose on a reel is provided at each yard hydrant. In addition, a hose cart with 250 feet minimum of 2½ inch coupled fire hose is located in the screen well and pump house.

Single Active Failure Analyses for Fire Pump Supply Lines, Main Distribution Loop and Fire Hydrants

Fire Pump Supply Lines to Underground Fire Main Loop

Both electric and diesel fire pumps feed the underground main loop by a common 8 inch supply line. A single failure (pipe break) of this supply line to the main loop would result in the loss of immediate supply of fire water to the main. Fire hose connections have been provided on the discharge of the diesel fire pump for supplying water to the main loop should a break occur in the common supply line.

In addition, CPCo has verified by test that a local (offsite) fire department pumper can draft water from the intake bay to provide fire protection water if needed.

Main Distribution Loop

A pipe break in the main distribution loop can be isolated by the sectionalizing valves. Fire water supply is then available around the main distribution loop from both directions up to the closed sectionalizing valves.

Fire Hydrant

A pipe break attributed to fire hydrant failure can be isolated with the sectionalizing valves which can be used to isolate the failed fire hydrant. The adjacent fire hydrants and extra hose can then serve the needs of the out-of-service fire hydrant.

9.5.1.2.4 Fire Spray and/or Sprinkler Systems

The sprinkler systems and hose stations are supplied by a common 6 inch distribution header which is connected at both ends to the underground main distribution loop.

Automatic wet pipe sprinklers are provided in part of the electrical equipment room (cable spreading area under the control room), auxiliary boiler room, turbine lube oil tank rooms, tool crib, RDS/UPS battery rooms, instrument and electrical shop - storage area, generator lube oil line, and west warehouse area.

Single Active Failure Analysis of Sprinkler Systems and Hose Stations

The sprinkler systems and hose stations are supplied by a common 6 inch distribution header which is connected at both ends to the underground fire main distribution loop. A single failure of this distribution header between the isolation valves at either end could result in the loss of fire water supply to the sprinkler systems and hose stations in the turbine generator building only. The small size of the plant would permit effective use of hose from exterior hydrants in such an unlikely event.

Fire Spray and/or Sprinkler Surveillance Requirements

The spray and/or sprinkler systems will be demonstrated to be operable:

a. Once per 18 months:

   1. By visual inspection of spray headers to verify their integrity.

   2. By visual inspection of each nozzle to verify no external blockage.

9.5.1.2.5 Interior Fire Hose Stations

Interior hose stations with 1½ inch fire hose have been provided throughout the plant including containment. All interior hoses are equipped with electrically safe nozzles where electrical equipment or cabling is located. The hose stations are supplied by the same 6 inch distribution header as the sprinkler systems. Single active failure of hose stations is addressed in 9.5.1.2.4 above.

Fire Hose Station Operability Requirements

The fire hose stations in the following locations will be operable whenever equipment in the areas protected by the fire hose stations is required to be operable:

1. Electrical equipment room.
2. Emergency condenser level.
3. Equipment lock laydown area.
4. Third floor corridor.
5. Screenhouse.
6. Machine shop.
7. Condensate pump area.
8. Core spray pump room (hydrant and hose house west of pump room).
9. Reactor cooling water pump area.
10. Interior Cable penetration room.

Action Required When a Hose Station is Inoperable

With a hose station inoperable, provide an additional hose for the unprotected area at an operable hose station within one hour.

Fire Hose Station Surveillance Requirements

Each fire hose station will be verified to be operable:

a. Once per 31 days by visual inspection of the station to assure all equipment is available at the station.

b. Once per 18 months by removing the hose for inspection and reracking and replacing (as required) all gaskets in the couplings that are degraded.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve operability and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

Emergency Core Cooling System Hose Surveillance Requirements

The ECCS hose is not required for decommissioning activities, therefore the surveillance requirements are no longer applicable and have been deleted.

9.5.1.3 Fire Barriers and Penetration Seals

Various walls throughout the plant are designated as fire barriers. Doors in fire barriers are rated fire doors or of similar construction, with ratings equivalent to the surrounding walls. Several fire doors are equipped with automatic closure devices and fusible links. Ventilation duct work through fire barriers is equipped with rated fire dampers where required.

Electrical and piping penetrations through fire barriers are sealed with various products approved for such use, and having a fire resistance equivalent to the fire barrier. Penetration seal construction is based on designs tested using ASTM E-119 exposure fires. Fire barriers for each plant location are discussed in the February 27, 1987 Fire Protection Summary Document.

Penetration Fire Barrier Functional Requirements

All penetration fire barriers including fire doors and fire dampers protecting safety-related areas will be functional at all times.

Actions Required for Non-Functional Penetration Fire Barriers

a. With one or more of the above required penetration fire barriers non-functional and with the area of the affected barrier(s) monitored by operable (reference Section 9.5.1.1 above) fire detection instrumentation, within 1 hour establish a fire watch patrol and inspect the affected area(s) at least one per hour.

b. With one or more of the above required penetration fire barriers non-functional and with the area of the affected barrier(s) not monitored by operable (reference Section 9.5.1.1 above) fire detection instrumentation, establish a continuous fire watch on at least one side of the affected penetration within 1 hours.

Penetration Fire Barrier Surveillance Requirements

Each of the above required penetration fire barriers shall be verified to be functional by a visual inspection;

a. At least one per 18 months, and

b. Prior to declaring a penetration fire barrier functional following repairs or maintenance.

Penetration Fire Barrier Basis

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers include active and passive elements in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, routine fire watch patrols in conjunction with operable fire detection instrumentation or a continuous fire watch are required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

The reactor was permanently defueled on September 20, 1997 and will not return to power operation. Elimination of equipment designated as an Engineered Safety Feature (ESF) has occurred as a result of permanent defueling.

Fire barriers are only required for the protection of safety related equipment and may be utilized for the minimization of property loss exposures. However, fire barriers limit the spread of fire and smoke, are beneficial to providing safe personnel egress routes and will be retained where practicable during decommissioning. Degradation of fire barriers is anticipated during the dismantlement phase.

9.5.1.4 Fire Brigade

A fire brigade of at least three members will be maintained on site at all times. This excludes one member of the minimum shift crew of the Plant and any other personnel required for other essential functions during a fire emergency.

NOTE: Fire brigade composition may be less than the minimum requirements for a period of time not to exceed two hours to accommodate unexpected absence of fire brigade members provided immediate action is taken to restore the fire brigade to the minimum requirements.

A fire brigade training program will be maintained and implemented and will, as practicable, meet or exceed the requirements of Section 27 of the NFPA Code-1975. Fire Brigade training drills will be held at least quarterly.

Fire brigade organization, training, and drills follow Nuclear Mutual Limited (NML) Loss Prevention Standards, which are based on NFPA Standards. Fire brigade drills are held monthly so all members have the opportunity to train as a team. The local fire department is invited to participate in training and drills each year. In accordance with the Big Rock Point Fire Plan, members of each operations shift and security shift are members of the fire brigade and are trained in fire protection. Training between the plant and the local fire department has been established and will be continued if possible. The responsibilities and duties of the fire brigades, local fire departments, and each individual are established. The Charlevoix Fire Department is made aware of the need for radioactive protection of personnel and the special hazards associated with the Big Rock Point Nuclear Plant.

9.5.2 COMMUNICATIONS (COM) AND WARNING SYSTEMS

Communications systems include intro-plant and plant-to-offsite capability. Normal and voice powered communications are supplemented by portable radios and bullhorns which are provided for Fire Brigade, Security or Operator use.

Information on communications capabilities, onsite public address and plant siren use, and the offsite public warning system are provided in the BRP Site Emergency Plan.

A prompt Notification System with public warning capability was installed offsite in response to Three Mile Island NUREG-0737 Item III.A.2 in order to "Improve Emergency Preparedness-Long Term." The system design is in compliance with NUREG-0654 as amended and the Federal Emergency Management Administration FEMA-REP-1. Details on the system were provided in CPCo letter dated February 4, 1982. The system control point is located in the Charlevoix County Sheriffs Office. Only a single control point is provided, (reference CPCo letter dated January 27, 1983 and Facility Change FC-541). One additional public warning siren speaker assembly was installed to improve coverage, as described in the NRC Inspection and Enforcement Report dated October 27, 1982. This increased the total speaker assemblies to twelve, as currently reflected in the Site Emergency Plan.

Onsite emergency notification is annunciated by means of twelve sirens which are inter-tied with the plant public address system via a microphone pick-up. This inter-tie was added as a plant feature by Facility Change FC-336. Addition of the siren pick-up for the PA System improved the siren signal levels in weak signal areas.

Communications and Warning System Evaluations

CPCo evaluated the effectiveness of the plant public address (PA) system in response to NRC Inspection and Enforcement Bulletin IEB 79-18, "Audibility Problems Encountered on Evacuation of Personnel From High Noise Areas." The results of the test of the PA/Siren system found audibility in certain areas to be marginal, as described in CPCo December 12, 1979 Bulletin response. These areas were corrected via Facility Change FC-491.

The plant siren controls were modified so that the various siren functions could be manually selected and automatically timed. The modification was performed via Facility Change FC-627 in response to Control Room Design Review (CRDR) Human Engineering Deficiencies.

Alternate Shutdown Communications

The Alternate Shutdown Communications discussion is not applicable for decommissioning and was deleted.

9.5.3 EMERGENCY LIGHTING SYSTEMS

Self-contained battery powered emergency lighting units which actuate automatically when normal lighting circuits are no longer available were installed via Facility Change FC-462-G. These lighting units were installed in various locations of the plant and were intended to be utilized for access to or egress from areas on loss of the normal lighting circuits. These lighting units are tested periodically to verify an administrative limit of three hours minimum illumination.

If the self contained emergency lighting units described above do not have power restored within about 8 hours, the emergency lighting battery packs will run down and switch off.

9.5.3.1 Appendix R - Emergency Lighting

The Appendix R emergency lighting requirements were designed to facilitate safe plant shutdown following an on-site fire. During decommissioning, safe shut down considerations are no longer applicable, therefore the discussion contained in this section has been deleted.

9.5.4 DIESEL FUEL OIL STORAGE

This section contained a discussion of plant modifications made to minimize post accident worker dose by increasing the size of the diesel fuel storage tanks for the diesel fire pump and the MDG. During decommissioning the post accident conditions are no longer applicable. The detailed discussion has therefore been deleted.

The diesel fuel tank changes for the MDG and fire pump were accomplished via Facility Change FC-511A.

Drawing 0740G40123 provides details of the 5000 gallon MDG and 1000 gallon diesel fire pump fuel oil storage tanks.

9.5.4.1 Main Diesel Generator (MDG) Fuel Storage Level

The MDG consumes fuel at approximately 16.5 gallons per hour (reference Special Site Test SST-17), when loaded at 190 $\pm$ 10 KW. The 5000 gallon storage tank is assumed to have 8 inches of unusable fuel level (reference Facility Change FC-511A), in the lower portion of the tank for a deduction of 295 gallons. Thus, with a full tank, 4705 gallons are available which provides an approximate 11-1/2 day supply.

9.5.4.2 Diesel Fire Pump Fuel Storage Level

The diesel fire pump consumes fuel at approximately 6.4 gallons per hour (reference Facility Change FC-607), based on manufacturers similarity testing report. The 1,000 gallon storage tank is assumed to have 6 inches of unusable fuel in the lower portion of the tank for a deduction of 74 gallons. Thus, with a full tank, 926 gallons are available which provides an approximate six day supply.

For fire protection, only a two hour supply is required (reference Appendix R to 10 CFR 50).

9.5.4.3 Standby Diesel Generator Fuel Storage Level

A Standby Diesel Generator was available onsite during the power operation phase of Big Rock Point. NRC Memorandum and Order dated May 26, 1976 required BRP ....to assure a second emergency diesel will be obtained and operational within 24 hours after a LOCA....

During decommissioning the requirement to maintain available a standby diesel generator is no longer applicable. Therefore the discussion of fuel storage level is no longer applicable and has been deleted.

9.5.4.4 Diesel Generators and Fire Pump Diesel Fuel Oil Requirements

CPCo letter dated April 21, 1980 in response to NRC letter dated January 7, 1980 concerning "Quality Assurance Requirements Regarding Diesel Generator Fuel Oil," provided the results of our review of diesel generator fuel oil.

As a result of the review, diesel fuel oil is included in the BRP QA Program by adding fuel oil to the Plant Q-List as a consumable. This approach adds diesel fuel as a Commercial Grade Q-Listed item and imposes adequate QA Program Control.

The NRC January 7, 1980 letter included a reference to Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October, 1979.

CPCo review indicated that, based on Part D, (Implementation) of Regulatory Guide 1.137, BRP is exempt from the testing requirements in the Guide.

However, testing requirements as outlined in Section 9.5.1.2.1 require that a sample of diesel fuel from the fuel storage tank be tested in accordance with American Society for Testing and Materials ASTM-D270-65 and be within the acceptable limits specified in Table 1 of ASTM-D975-74 with respect to viscosity, water content and sediment. This test is conducted at least once per 92 days.

It should be noted that, although the above fuel oil testing is applicable only to the diesel fire pump fuel storage tank, the actual testing being performed is applied to the MDG storage tank to meet the intent of our April 21, 1980 submittal.

9.5.5 MAIN DIESEL GENERATOR AND DIESEL FIRE PUMP PROTECTIVE TRIPS

The discussion of historical perspective of bypassing the protective trip signals associated with the MDG has been deleted from the UFHSR. The Appendix K (ECCS) accident conditions for which this discussion addressed are not applicable during decommissioning. The diesel generator and the diesel fire pump are not considered safety related components for the plant dismantlement.

9.5.6 MAIN DIESEL GENERATOR ALARM AND CONTROL CIRCUITRY

NRC letters dated April 7, 1977 and April 12, 1978 requested information on EDG alarm and control circuitry. The information was provided by letters dated May 24, 1977 and May 11, 1978.

The alarm and control circuitry was designed to provide the operator with an indication of emergency power indication inside the control room. The emergency designation for the diesel generator is not applicable during decommissioning. The previous loads automatically picked-up by the MDG (electric fire pump , redundant core spray and backup enclosure spray for example) are not required due to permanent fuel removal from the reactor.

The remaining portion of this section has therefore been deleted.

9.5.7 MAIN DIESEL GENERATOR COOLING WATER

The MDG cooling water system is shown on Drawing 0740G40123. The cooling water is from the circulating water discharge bay by a self-priming engine driven centrifugal cooling water pump.

A trickle flow of water is being supplied to keep the cooling water pump case filled. The cooling water pump case must be filled in order for the pump to self prime. SC-96-016 installed a priming flow indicator to indicate the pump is filled and in a state of readiness. SC-96-016 was declared operable April 4, 1996. A backup supply of priming water also exists from the fire water and domestic water systems, thus assuring an adequate supply of priming water. The pump discharges cool water through the diesel engine lube oil cooler and excess priming water is discharged via this same route. Details on the system are contained in a letter to NRC dated May 18, 1973.

On May 8, 1978 the cooling water pump packing and lantern ring were replaced with a mechanical seal, thus eliminating the need for sealing water (Reference SFC-78-006). Cooling water to the mechanical seal is provided via the shaft sealing water line.

FC-688 installed a flowmeter to measure the cooling water flow rate from the MDG heat exchanger. FC-688 also retired the cooling water head tank in place and re-routed the mechanical seal water line to the cooling water pump discharge. FC-688 was declared operable January 29, 1996.

The water pump suction inlet is cleaned periodically as a preventative maintenance item.

The cooling water suction line contains an electric heating element, used when freezing weather is a possibility, which is checked for circuit reliability periodically.

Chapter 9 References

1. Amendment 2 to the Consolidated Application for the Big Rock Point Spent Fuel Pool Rack Addition. CPCo to NRC January 10, 1983.

2. NRC to CPCo (D VandeWalle), November 17, 1983 Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to the Modification of the Spent Fuel Storage Pool.

3. NRC to CPCo (D VandeWalle), October 11, 1984 Spent Fuel Pool Modification, Amendment 70 to Facility Operating License Change to Technical Specification.

4. NRC to CPCo (D Hoffman), May 15, 1981 Safety Evaluation Report and Environmental Impact Appraisal Relating to the Modification of the Spent Fuel Storage Pool.

5. EA-BRPDP-CCH5-3, Revision 2, July 15, 1997, Spent Fuel Pool Decay Heat Analysis.

6. EA-FC-699-06, Revision 0, January 1, 1999, Spent Fuel Pool Heat Forecast.

7. EA-DOP4-01, Revision 0, June 3, 1999, Spent Fuel Pool Heatup Spreadsheet Results.

8. NRC Safety Evaluation Report (SER) to CECo (Kurt M. Haas), September 28, 2001, Defueled Technical Specification Amendment 122, Fuel Handling and Control of Heavy Loads.

9. EA-BRP-LEB-00-01, SFP Surface Dose Rate During DFS Transfer Cask Loading, February 24, 2000.