Consumers Energy

A CMS Energy Company

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10 CFR 50.59(d)(3)
10 CFR 50.71
10 CFR 50.4

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DOCKETS 50-155 AND 72-043 - LICENSE DPR-6 - BIG ROCK POINT PLANT - 10 CFR 50.59 REPORT OF CHANGES, TESTS AND EXPERIMENTS

Pursuant to 10 CFR 50.59(d)(3), attached to this letter is Consumers Energy Company's report of completed changes, tests, and experiments for the Big Rock Point Restoration Project as described in the Updated Final Hazards Summary Report (UFHSR).

Attachment 1 describes the modifications performed to support decommissioning and site restoration. This report provides a summary of changes to the facility performed from March 31, 2005 through August 1, 2006. The report includes a brief description of each change and a summary of the 10 CFR 50.59 evaluations.

Attachment 2 provides changes to procedures, programs, and Defueled Technical Specifications, where appropriate. This report provides a summary of changes to the procedures, programs, and Defueled Technical Specifications from March 31, 2005 through August 15, 2006. The report includes a brief description of each change and a summary of the 10 CFR 50.59 evaluations.

Attachment 3 is a complete, replacement copy of the Updated Final Hazards Summary Report (UFHSR). Revision 14 to the UFHSR (safety analysis report) replaced previous revisions of the UFHSR in its entirety. The required evaluations concluded that these changes did not require NRC approval pursuant to 10 CFR 50.59.

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Site General Manager

cc: Administrator, Region III, USNRC
NRC Decommissioning Inspector – Big Rock Point
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ATTACHMENT 1

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKETS 50-155 AND 72-043 – LICENSE DPR-6

REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS

September 15, 2006

Modifications

1 page

BIG ROCK POINT PLANT
Annual Report of Facility
Changes, Tests and Experiments

Attachment 1
Modifications

In 2004, the remaining Big Rock Point process for modifications was the Work Order process. On December 6, 2006, the work process became specific to the Independent Spent Fuel Storage Installation (ISFSI), as all Important-to-Safety (ITS) Structures, Systems, and Components (SSCs) remaining onsite were only at the ISFSI facility. However, the following two activities were determined to be within scope of the procedure due to their interaction with a "safety-related" program, namely the Radiation Protection Program. In 2005, these activities were "screened" under the 10 CFR 50.59 criteria.

1. **WO 12520057 – Containment Concrete Softening**[1]
   No changes to the UFHSR or PSDAR resulted from the screening of this activity. This activity originally "screened-in" as affected the completion date of a PSDAR activity. The PSDAR slated completion of containment interior concrete demolition for the second quarter of 2006. The activity was completed on April 19, 2006, in the second quarter, as originally projected.

2. **WO 2005004 – Shipment and Transfer of the Plutonium-Beryllium Source to the Department of Energy**[2]
   Minor revisions to the UFHSR (Chapters 1 and 9) resulted when the Big Rock Point Plutonium-Beryllium source was transferred to the Department of Energy on January 20, 2006.

   The shipment and transfer had no affect on dry fuel storage accidents in Chapter 15 of the UFHSR. Work Order precautions and use of approved procedure (34A-13, Administration of the Nuclear Material Control Plan) minimized risk from the activity. The source was a passive component – no malfunction, other than fission product barrier breach is credible. The source was transferred in a container.

[1] Quality Review Form Log # 324-05, cartridge/frame 4973/1820

[2] Quality Review Form Log # 233-05, cartridge/frame 4968/1471

ATTACHMENT 2

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKETS 50-155 AND 72-043 – LICENSE DPR-6

REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS

September 15, 2006

Procedures and Programs

6 pages

U.S. NRC Safety Evaluation Report (SER) dated January 19, 2005
Approval of Proposed Disposal Procedures in Accordance with 10 CFR 20.2002[3]

Debris disposal is a radiation protection program. On September 15, 2004, U.S. NRC approval was sought for an amendment to the previously approved 10 CFR Part 20.2002 procedures for disposal of debris. The amendment was requested to include evaluations for disposal of potentially slightly contaminated PCB materials in a landfill a greater distance from that addressed in the original approval. The revised distance and editorial corrections to the volume of debris (reconciliation of volume verses density of materials) slightly altered the conclusions of the analysis used to support the original 10 CFR 20.2002 program approval (U.S. NRC SER dated December 3, 2001.)

The NRC SER of January 19, 2005 approved the requested amendment to the disposal procedures.

<u>10 CFR 50.59 Evaluation Summary</u>

Approval of the disposal procedures resulted in changes to Chapter 11 of the UFHSR. The analysis of record was amended, resulting in addition of reference and editorial revisions to the Chapter. The distance in the analysis of record were increased to accommodate disposal of PCB contaminated waste in a licensed landfill. In addition, the density and quantity of material was revised, along with increasing the number of trips used to transport the material.

The same method of evaluation (RESRAD) was used to perform the evaluations with the revised inputs for dose evaluations for Transportation Workers, Land Fill workers, and Resident Farmer (living on the landfill property.) The following summarizes dose revision as a result of the amended inputs:

- Transportation Worker to Licensed Type II landfill dose decreased from 0.366 mrem to 0.320 mrem. An additional dose to a Transportation Worker to the Licensed PCB landfill was added (0.0182 mrem).

- Licensed Type II Landfill Worker annual dose increased slightly from 0.290 mrem to 0.291 mrem. PCB Landfill Worker annual dose of 0.178 mrem was added.

- Landfill Resident Farmer annual dose at the Type II landfill increased from 0.009 mrem to 0.0178 mrem. PCB landfill Resident Farmer annual dose of 0.001 mrem was added.

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[3] Quality Review Form Log # 073-05, cartridge/frame 4954/1689

Attachment 2
Changes to Procedures, Programs, Defueled Technical Specifications, and the UFHSR

Amendment 126 to the Defueled Technical Specifications
U.S. NRC Safety Evaluation Report (SER) dated March 24, 2005
License Termination Plan (LTP)[4]

Reference to the License Termination Plan (LTP) was added to the Updated Final Hazards Summary Report (UFHSR) and new License Condition C.2.C.(4) was added to DPR-6 to reflect conditions required by NRC acceptance of Big Rock Point's LTP.

10 CFR 50.59 Evaluation Summary

Updated Final Hazards Summary Report (UFHSR) Chapter 15 accident analyses were reviewed. Any accidents resulting from LTP activities are bounded by the fuel accidents in the UFHSR. Accidents analyses related to decommissioning activities are also addressed in the UFHSR, and they bound accidents created by LTP activities.

The LTP is consistent with the activities described by the Big Rock Point UFHSR and Post Shutdown Decommissioning Activities Report (PSDAR). The changes to the UFHSR were administrative.

No new failure mechanisms were created by the UFHSR administrative update (addition of references to the License Termination Plan). All spent nuclear fuel is stored in dry fuel storage canisters under a 10 CFR Part 72 general license. LTP remediation and survey activities are consistent with Radiation Protection Program and previously U.S. NRC approved 10 CFR 20.2002 procedures.

The purpose of the LTP is to ensure residual radioactivity from reactor operation is below the unrestricted release criteria of 10 CFR 20.1402. Since the activities are bounded by previously analyzed accidents in the UFHSR and demonstrate compliance with Part 20 unrestricted release criteria, no new or different type of accident is created. No fission product barriers (dry fuel storage canisters and over packs) are affected by this activity. Release of the site will be in accordance with the LTP [phased approach – 1. Release of area other than those associated with the operation of the Independent Spent Fuel Storage Installation (ISFSI) and 2. Termination of the 10 CFR Part 50 Operating License and 10 CFR Part 72 General License once all spent nuclear fuel and reactor-related Greater Than Class C (GTCC) waste is shipped from the site.]

[4] Quality Review Form Log # 091-05, Cartridge/Frame 4951/0552

U.S. NRC Safety Evaluation Report (SER) dated October 13, 2005
Independent Spent Fuel Storage Installation (ISFSI) Emergency Plan[5]

The reduction in commitment to the Emergency Plan included the following elements that resulted in Administrative revisions to the Updated Final Hazards Summary Report (UFHSR):

- A 300-meter Independent Spent Fuel Storage Installation (ISFSI) Owner Controlled Area was defined – which resulted in Chapter 2 revisions to UFHSR distances to the site boundary and made the Emergency Plan specific to the ISFSI and the 300-meter area surrounding it;
- The title of the Emergency Plan was revised from "Decommissioning Emergency Plan" to "ISFSI Emergency Plan" to reflect the decreased scope of the plan;
- Addition of a new analysis (EA-BRP-RAE-0404, ISFSI Dose Calculation) was added to Chapter 15.

Chapter 2 site plan was revised and the300-meter distance replaced former distance from the centerline of the reactor to the site boundary.

Chapter 15 was updated to add the ISFSI Dose Calculation evaluation and to reflect that no Protective Actions are anticipated at a distance of 100 meters form the edge of the ISFSI pad for the emergencies outlined in the ISFSI Emergency Plan.

<u>10 CFR 50.59 Evaluation Summary</u>

The UFHSR revisions have no affect on analyzed, design basis accidents. Revisions are a result of an NRC SER on revised Emergency Action Levels (EALs). Addition of the new analysis used in justification of the decreased scope of the Emergency Plan from one which applied to both the former operating plant area and the ISFSI are to only the ISFSI area. The reviewed analysis was added to provide documentation of Emergency Plan assumptions.

The revisions have no affect on analyzed, design basis accidents. EAL scheme was revised to reflect site-specific, analyzed accidents referenced in BNFL/BNG Safety Analysis Reports and in UFHSR Accident analysis in Chapter 15. Since the ISFSI is passive and the proposed activity is a program (Emergency Plan) no increases in malfunctions of structures, systems, or components result from this activity.

The method of analysis used in the dry fuel storage dose analysis is essentially the same as that used in the analysis for decommissioning spent fuel storage events. Available

[5] Quality Review Form Log # 264-04, Cartridge/Frame 4967/1590 and
QRF Log # 297-05, Cartridge/Frame 4969/2054

radionuclides and quantities from a spent fuel containment breach (in this case from a dry fuel storage cask as opposed to the former spent fuel pool structure). This revision reflects transfer of all spent nuclear fuel and Greater Than Class C (GTCC) waste to the ISFSI. The type of accident (fission product release) remains the same. Design basis accident for fission product release was analyzed in the modification to install the ISFSI and relocate the fuel and GTCC waste and documented in the site-specific 10 CFR 72.212 report (Big Rock Point Volume 33.)

Volume 34
Quality Program Description for Big Rock Point [formerly titled: Quality Program Description for Nuclear Power Plants Part 1 – Big Rock Point Nuclear Power Plant (CPC-2A)] [6]

The Updated Final Hazards Summary Report (UFHSR) revisions as a result of this Volume revision were editorial – reflecting the Title change.

<u>10 CFR 50.59 Evaluation Summary</u>

The title change of this volume had no affect on the accident analyses in Chapter 15 of the UFHSR.

Editorial changes to the UFHSR were not a reduction in effectiveness of the Quality Program Description (QPD, formerly CPC-2A). A detailed QPD section-by-section analysis, justifying no reduction in commitment, is attached to the quality review form. Criteria and requirements for determining activities affecting safety-related programs and important-to-safety structures, systems, or components have not been revised. Revisions also included addition of personnel and responsibilities to Chapter 2 of the QPD. The revisions were reviewed to ensure the specific commitments to ANSI and Regulatory Guide requirements were maintained.

The editorial and administrative revisions to this program do not increase accidents, affect components, or accident analyses, as the changes were not a reduction in commitment in the quality program.

[6] Quality Review Form Log # 327-05, Cartridge/Frame 4968/1532

UFHSR Revisions due to Radiation Protection Program Administrative Procedure D5.1, Radiation Protection and Environmental Services Program Description [7]

The Updated Final Hazards Summary Report (UFHSR) Chapter 12, Radiation Protection, revisions as a result of Procedure D5.1 revisions included cross-references to commitments in the Quality Program Description (QPD), Big Rock Point Volume 34. Revisions continue to provide a level of compliance to 10 CFR Part 20 appropriate for the licensed activities currently being performed at Big Rock. A general review of Chapter 12 was performed along with the procedure revision. This review ensured the descriptions in the UFHSR reflect the site configuration. It also reduced the scope of Radiation Protection Program commensurate to the scope needed for a stand-alone ISFSI site, with all spent nuclear fuel and reactor-related Greater-Than Class C (GTCC) waste stored on the ISFSI. This reduction in scope is consistent with the return of the former operating plant areas to "Greenfield" and is in accordance with the Big Rock Point License Termination Plan.

<u>10 CFR 50.59 Evaluation Summary</u>

Revisions to the procedure had no affect on any accident analyses discussed in Chapter 15 of the UFHSR and Radiation Protection Program continues to be consistent with the requirements in the Administrative Sections of the Defueled Technical Specifications, the UFHSR (Chapter 13), and the Quality Program Description.

The Radiation Protection Program contains specific information on activities to meet the requirements of 10 CFR Part 20. The revision reflected reduced scope of the program, commensurate with current site condition, and there is no affect to any important-to-safety structures, systems, or components.

The Radiation protection Program and ALARA elements of the program continue to provide compliance with the provisions of 10 CFR Part 20.

No analytical method or references are affected by administrative revisions.

[7] Quality Review Form Log # 039-06 and 040-06, as of 8/09/2006 – QRF not filmed.

ATTACHMENT 3

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKETS 50-155 AND 72-043 – LICENSE DPR-6

REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS
September 15, 2006

Revision 14, Updated Final Hazards Summary Report (UFHSR)

138 pages

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CHAPTER 18: HUMAN FACTORS ENGINEERING

18.1 CONTROL ROOM DESIGN REVIEW (CRDR)

18.2 SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

18.3 EMERGENCY RESPONSE CAPABILITY

1.0 INTRODUCTION AND GENERAL DESCRIPTION

1.1 INTRODUCTION AND PURPOSE OF THIS REPORT

This report presents information to give reasonable assurance that the Big Rock Point Plant (BRP) as described does not pose an undue risk to the health and safety of the public. The updated information is presented to support Consumers Energy's license Number DPR-6 in accordance with requirements established by 10 CFR 50.71(e) and 10 CFR 50.82. The report presents the design bases and limits on facility operation, describes the facility, and presents safety analyses of selected structures, systems and components.

1.2 BACKGROUND AND PLANT DESCRIPTION

1.2.1 BACKGROUND

The general features of the BRP Nuclear Power Plant including the pertinent details of the site are described in this report and were described in the following previous reports:

1) Big Rock Point Nuclear Power Station, Application to US Atomic Energy Commission for Reactor Construction Permit and Operating License, Part B, Preliminary Hazards Summary Report, January 14, 1960

2) Amendment Number 2 to Application for Reactor Construction Permit and Operating License, Revised Hazards Summary Report, October 14, 1960

3) Amendment Number 3 to Application for Provisional Operating License, Revised Final Hazards Summary Report, November 14, 1961, as revised 3/12/62, 3/19/62, and 3/23/62, Revision 1

4) Big Rock Point Plant, Revision 6 to the Updated Final Hazards Summary Report (FHSR), October 8, 1996. This is the final submittal of the UFHSR discussing power operations; all subsequent revisions incorporated plant changes during various stages of dismantlement.

On the basis of submittal 1), the US Atomic Energy Commission issued a construction permit (Number CPPR-9) on May 31, 1960. On the basis of submittals 2) and 3) as supplemented and amended, Operating License Number DPR-6 was issued.

Events of significance are:

a. Provisional Operating License issued August 30, 1962;

b. Initial Criticality was achieved September 27, 1962;

c. Initial Power Operations was achieved December 8, 1962;

d. The Date of Commercial Operation was March 29, 1963;

e. The Full Term Operating License was issued on May 1, 1964;

f. Power level was increased from 157 MWt to 240 MWt in May, 1964;

g. The plant permanently ceased operation on August 29, 1997;

h. The fuel was permanently removed from the reactor vessel on September 20, 1997;

i. Certificate of Compliance (CoC) Number 72-1026 was issued to BNFL FuelSolutions™ on February 15, 2001 for the FuelSolutions™ Storage System;

j. December 18, 2001, Notification of Storage of Spent Nuclear Fuel, as a general licensee under the provisions of 10 CFR 72;

k. All fuel was permanently moved from wet fuel storage to an on-site Independent Spent Fuel Storage Installation (ISFSI) on March 26, 2003;

l. Post Shutdown Decommissioning Activities Report (PSDAR), Revision 3, submitted on May 28, 2003; and

m. License Termination Plan was submitted on April 1, 2003 and incorporated in License DPR-6 by Amendment 126 dated March 24, 2005.

In accordance with 10 CFR 50.82, Consumers Energy is decommissioning the direct cycle, forced circulation boiling water reactor at the BRP site, which is located in Charlevoix County, between the towns of Charlevoix and Petoskey, on the northern shore of Michigan's lower peninsula.

1.2.2 BIG ROCK POINT PLANT DESCRIPTION

Big Rock Point has been permanently defueled and is being dismantled. Previously, this section provided a brief description of the general features of the plant with emphasis on the differences between BRP and more recent boiling water reactors.

For a detailed description of the BRP Plant, refer to item 4 of Section 1.2.1, referencing Revision 6 of the UFHSR.

As of March 26, 2003 all of the spent nuclear fuel was transferred from the spent fuel pool into dry storage canisters for storage at the Independent Spent Fuel Storage Installation (ISFSI). Chapter 9 of this UFHSR includes a description of the ISFSI. With all fuel stored at the ISFSI facility, BRP Plant presents a significantly reduced risk to public health and safety. Big Rock Point Structures, Systems and Components (SSCs) that were required for reactor operation are not required for the permanently defueled plant.

Programs that are important to the site are those that satisfy the requirements of 10 CFR 20 (Standards for Protection Against Radiation), 10 CFR 71 (Packaging and Transportation of Radioactive Material), 10 CFR 72 (Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste) and 10 CFR 73 (Physical Protection of Plants and Materials). Dismantlement activities are monitored and controlled in a manner sufficient to provide reasonable assurance that they do not result in radiological hazards.

1.3 SCOPE, CHARACTER, AND CONCLUSIONS OF THIS REPORT

1.3.1 SCOPE

This report constitutes an update to the November 14, 1961 Final Hazards Summary Report, as revised 3/12/62, 3/19/62, and 3/23/62, Revision 1, and is now titled Updated Final Hazards Summary Report (UFHSR). The information presented is based upon the plant in its present condition and reflects the current Safety Analysis Design Bases and applicable operating requirements.

This report is a unique document which stands alone as the BRP UFHSR, which can serve as the baseline for future periodic changes.

The UFHSR is an integrated document containing or referencing the latest information developed in response to US Nuclear Regulatory Commission (NRC) Requirements. The information presented is not based upon the descriptions or degree of detail required to meet Standard Review Plan content for modern plant Safety Analysis Reports (FSAR), however, this update is intended to be similar or comparable to an FSAR for those items, areas, or activities evaluated or referenced within this report.

1.3.2 CHARACTER

This report presents descriptions and safety evaluations or provides specific reference to direct the users of this report to appropriate information to explain the design bases, operating requirements or other analyses, activities plans, manuals, programs, or reports pertinent to the safe storage of spent fuel, radiological controls, and related activities.

1.3.3 CONCLUSIONS

This report presents or references information demonstrating that the BRP Nuclear Plant and ISFSI have been designed with adequate protection against credible accidents and events. In addition, this report presents or references information that shows that there is reasonable assurance that the BRP Nuclear Power Plant is being dismantled by Consumers Energy without undue risk to the health and safety of the public.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

This section previously discussed the organizations involved in the original design and construction of the plant which is not applicable to the permanently defueled plant. Organizational structure is discussed in Section 13.1.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

Previously this section provided information on Integrated Assessment of Open Issues and the use of the Probabilistic Risk Assessment (PRA) to address these issues. Because the plant is permanently defueled, these issues are no longer relevant and the discussions previously contained in Section 1.5 and its subsections have been deleted.

1.6 MATERIAL INCORPORATED BY REFERENCE

The following provides a tabulation of topical reports, plans, programs, manuals, etc. which are incorporated by reference as part of License DPR-6, Docket 50-155, or this UFHSR for the BRP Plant. These documents are updated and revised on schedules separate from this UFHSR.

a. Docket 50-155 Big Rock Point Plant Facility Operating License DPR-6, Appendix A, Defueled Technical Specifications

b. Quality Program Description for Big Rock Point

c. Big Rock Point Plant:

   1. ISFSI Emergency Plan
   2. ISFSI Security Plan
   3. Fire Protection Plan
   4. Environmental Report for Decommissioning
   5. ISFSI Docket Number 72-043
   6. BNFL Certificate of Compliance (CoC) 72-1026
   7. BNFL FuelSolutions™ W74 Canister Storage Final Safety Analysis Report, Document Number WSNF-223
   8. BNFL FuelSolutions™ Storage System Final Safety Analysis Report, Document Number WSNF-220
   9. License Termination Plan
   10. Off-site Dose Calculation Manual (ODCM)

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

Drawings such as Piping and Instrument Diagrams, and Electrical, Instrumentation, and Control Drawings were provided to the NRC as "Information Copies" as part of the distribution of BRP Manual, Volume 22 and as such are updated separate from this report and are not considered part of this report. Any unique drawings referenced in this report that are not included in BRP Manual, Volume 22 will be included in this UFHSR and these drawings will be updated with revisions to this report.

2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION AND DESCRIPTION

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 Energy Company, in relation to the nearby highway.

Figure 2.3, Facility Identification, illustrates the relationship of the existing facility [the Independent Spent Fuel Storage Installation (ISFSI)] to the location of the former operating plant area, which has been decommissioned in accordance with the provisions of the License Termination Plan.

Detailed site location and description is found in BRP Volume 32, Environmental Report for Decommissioning (Reference 2.6.1).

2.1.1.1 Former Operating Plant Features (Former Protected Area)

The information in this section is retained for historical information. During reactor operation, the former protected area structures included:

a. A 130-foot diameter spherical containment vessel

   Reactor Building (T-1)

b. A Turbine Generator Building (B-3)

c. A structure housing water intake facilities and diesel generator

   1. Screen, Well and Pump House (B-4)

   2. Emergency Generator Room (B-5)

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

e. A Security Building (B-16)

f. Waste Storage Vaults (Liquid) (B-11)

The containment vessel housed 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 were added to support dismantlement.

2.1.1.2 Independent Spent Fuel Storage Installation (ISFSI)

The principle ISFSI structure includes a pad installed in Big Rock Point (BRP) Nuclear Plant Facility Change (FC)-0704, ISFSI Storage Facilities and Equipment. The design of the ISFSI is discussed in Chapter 9 of this UFHSR.

2.1.1.3 Surrounding Area

Charlevoix County, with a land area of about 400 square miles, has farm earnings (Reference 2.6.2) 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
(References 2.6.2 and 2.6.3)

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,	26	61	48

County	Antrim	Charlevoix	Emmet
1987			
% 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 2.6.4)

The BRP site is located on the shore of Lake Michigan in Charlevoix County in the northern part of Michigan's lower peninsula. The 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, Figure 2.2. The nearest landside property and the nearest shoreline property line (high water mark of Lake Michigan) is more than 300 meters from the Independent Spent Fuel Storage Installation (ISFSI).

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 bordered by US Route 31 and the former Chesapeake and Ohio Railroad, portions of which were owned by the Michigan Department of Transportation. 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 2.6.5). 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.

Evaluation Summary

The topic of Exclusion Area Authority and Control was evaluated by the US Nuclear Regulatory Commission (NRC) as part of the Systematic Evaluation Program (SEP) Topic Number II-1.A.

This evaluation concluded that BRP has the proper authority, with one exception, to determine all activities within the exclusion area, as required by 10 CFR 100. The exception concerned the lack of an arrangement to control traffic on the former Chesapeake and Ohio Railroad line that 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 [1] in the new ISFSI Emergency Plan. This completed the evaluation of the SEP topic.

2.1.3 POPULATION DISTRIBUTION

The site is remote from any large metropolitan areas, and has a generally favorable low surrounding permanent population.

Population distribution information is found in BRP Environmental Report for Decommissioning (Reference 2.6.1).

[1] Since that evaluation was completed, the need to include this arrangement in the ISFSI 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 site emergency will be included in the ISFSI Emergency Plan.

2.1.3.1 Population within Thirty (30) Miles

The region surrounding the BRP 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 (i.e., 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 2.6.7).

2.1.3.3 Low Population Zone and Emergency Planning Zones

U.S. Nuclear Regulatory Commission Safety Evaluation Report (SER) (Reference 2.6-38) granted exemption from certain requirements to 10 CFR 50.54(q). Big Rock Point was allowed to discontinue offsite emergency planning activities. The Big Rock Point Owner Controlled Area became the Emergency Planning Zone.

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 2.6.8)
** Population figures are 2000 Census (Reference 2.6.3)

Charlevoix is the closest urban center and does not currently nor foreseeably fall within the population center definition of 10 CFR 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 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 SEP Topic number II-1.B. This review resulted in an assessment and evaluation (Reference 2.6.4) 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 100. Further, the NRC concluded that the low population zone and population center distances specified for the BRP site remain valid and the site is in conformance with the distance requirements of 10 CFR 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

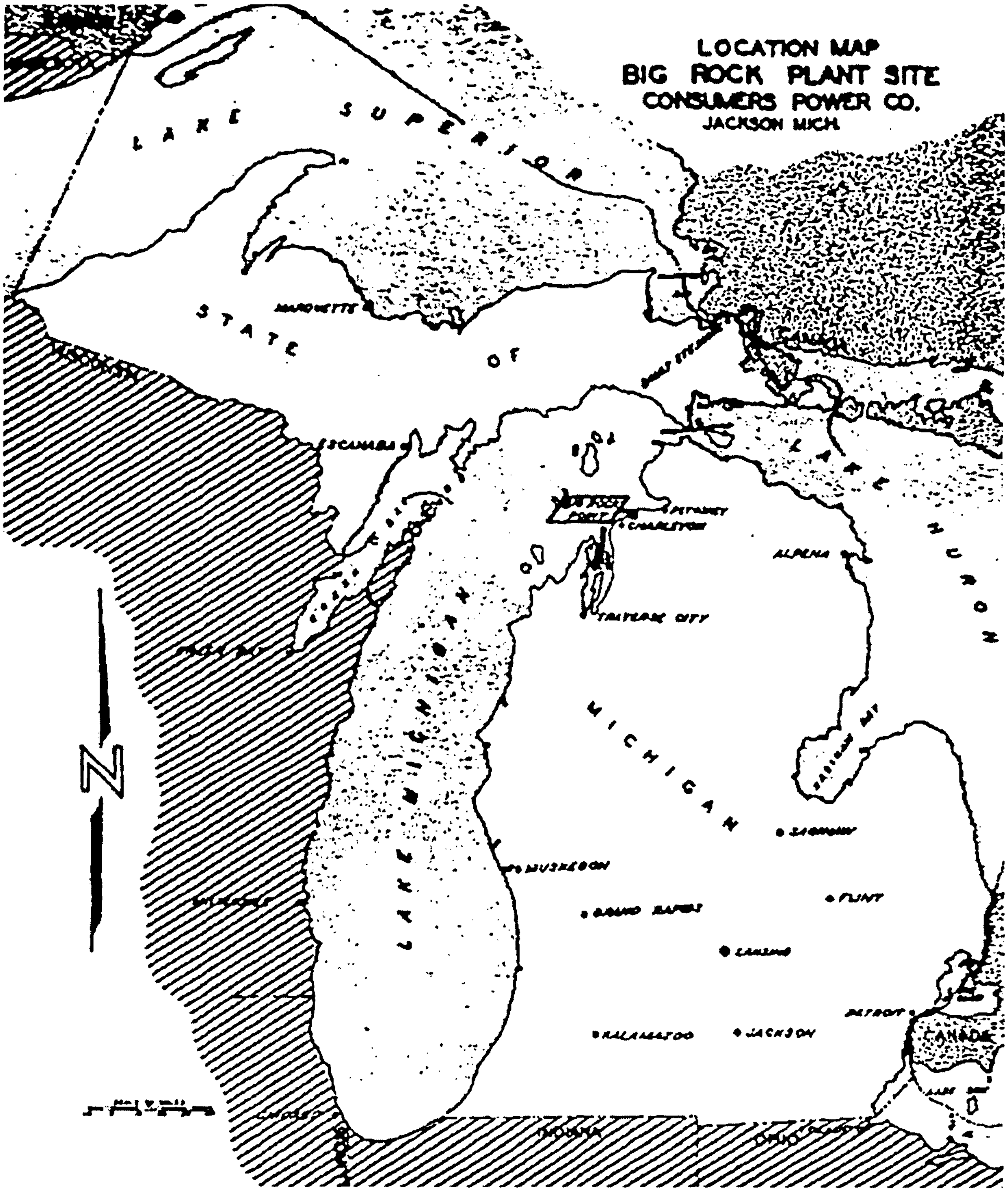


Figure 2.2
Site Map of Big Rock Point Plant

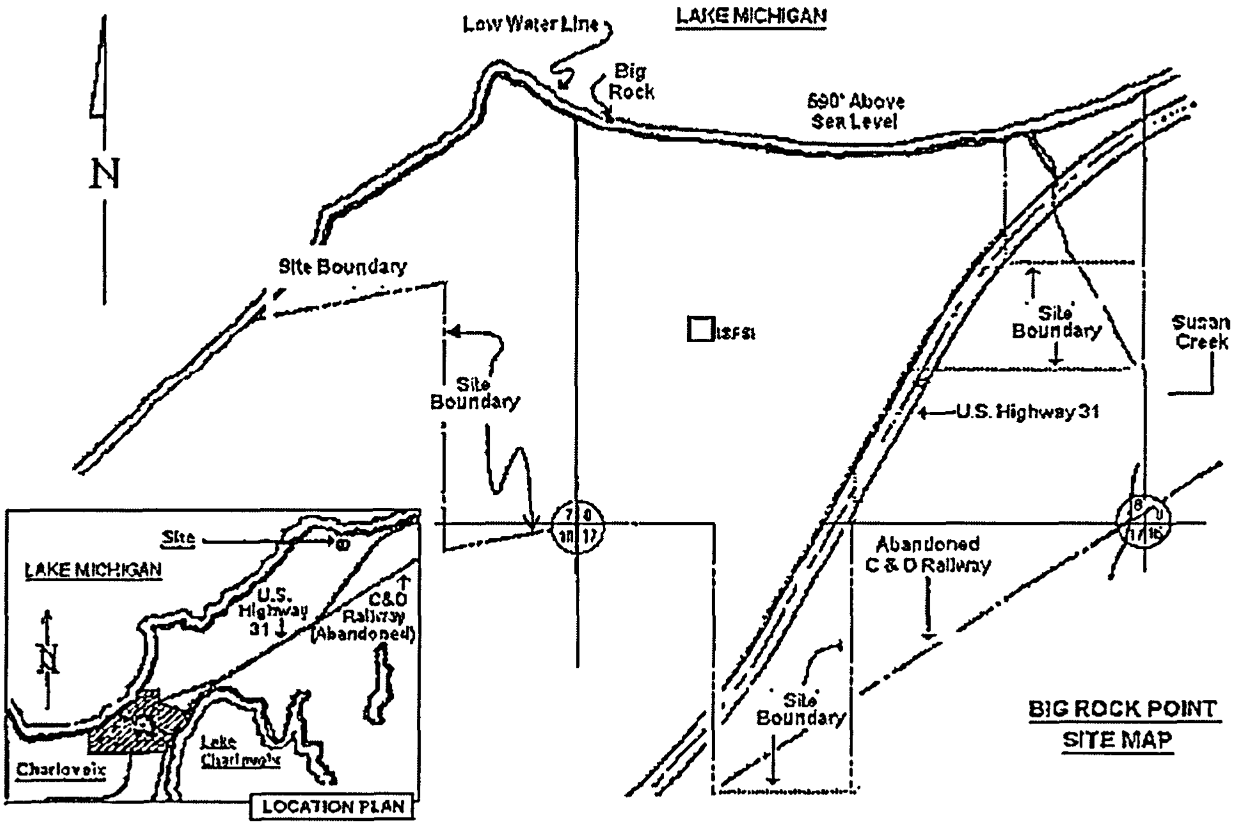
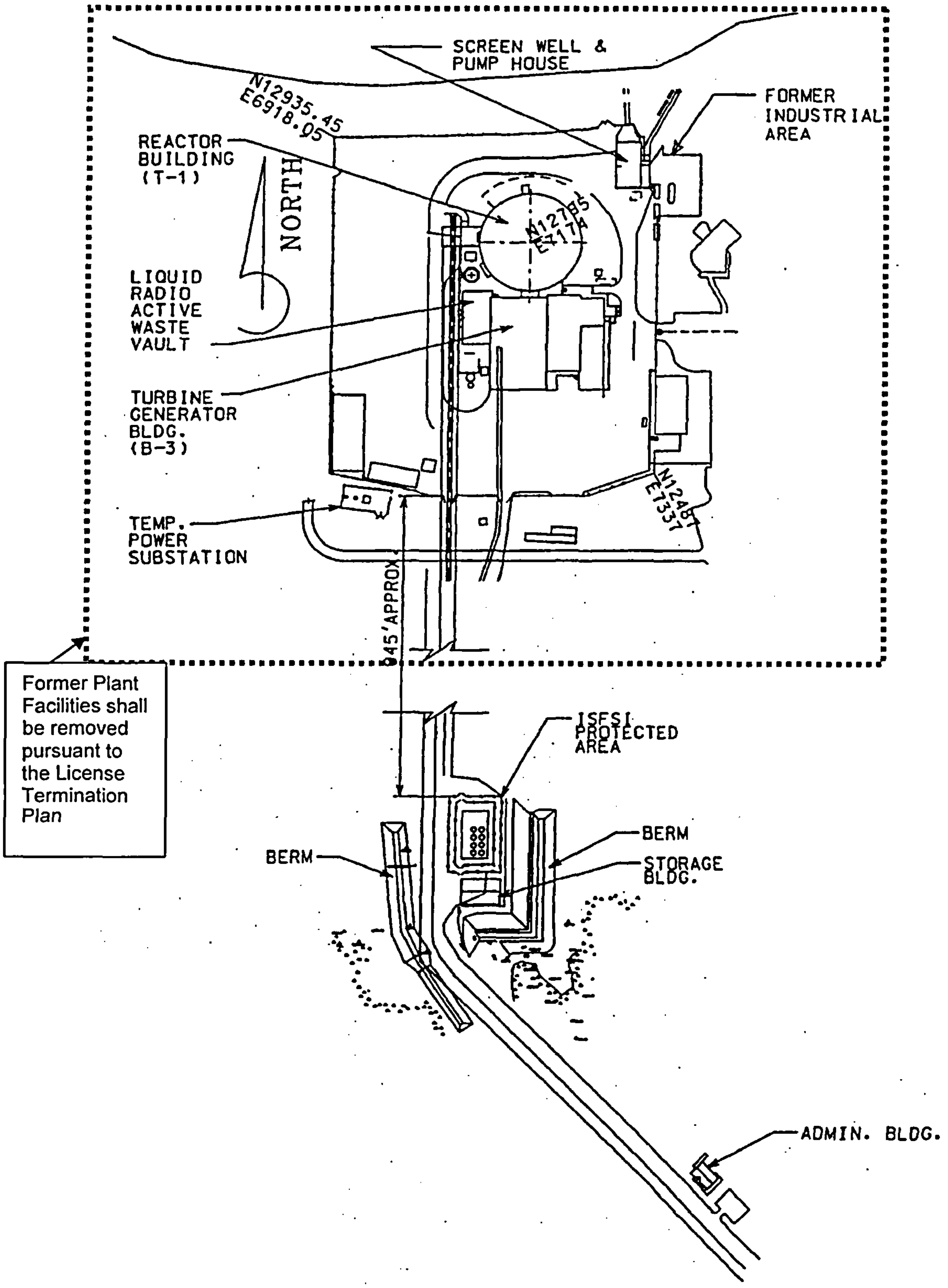


Figure 2.3
Facility Identification



2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 LOCATIONS AND ROUTES

BRP Environmental Report for Decommissioning (Reference 2.6.1), provides a listing of manufacturing plants in the five (5)-mile radius of t BRP.

Manufacturing plant data was extracted from the February 1984 HMM Document Number 83-600, Evacuation Time Estimates for BRP. The document was updated in 1993 (Reference 2.6.6).

Industrial activity in the vicinity of BRP 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 currently pass 10 miles from the BRP Plant. A former military low level training route 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 SEP Topic Number II-1.C. This evaluation is extended to the ISFSI pad due to the pad's location, in proximity to the plant site. This resulted in a safety evaluation (Reference 2.6.9) as follows:

2.2.2.1 Industrial Activity

Industrial activity in the vicinity of BRP 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 approximately 100 employees 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 ISFSI would depend on many factors including source parameters, wind speed and direction, cloud plume rise, and protective actions taken by site security.

An industrial park is located about 2.5 miles southwest of the site. 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 ISFSI (over two miles) is considered adequate to preclude accidents at the terminal affecting the safe operation of the ISFSI.

2.2.2.2 Transportation Activity

The nearest highway to the site is US Route 31 (refer to Figure 2.2), which is located outside of the ISFSI Owner Controlled Area (greater than 300 meters from the ISFSI pad.) 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.

The separation distance between the highway and the ISFSI 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 ISFSI (Reference 2.6.10).

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 (i.e., 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 that involve a significant loss of material is about two percent. 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 that would generate toxic chemical traffic. Therefore, it is our judgment that the threat to the safe operation of the ISFSI 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 and 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 UFHSR, 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 that is located about 1.5 miles south. At this distance, pipeline accidents will not affect the safe operation of the ISFSI, 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 site. 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 site. Thus, the occurrence of a barge accident with consequences severe enough to affect the ISFSI is extremely unlikely and does not constitute a credible risk. Similarly, the main shipping route in Lake Michigan that is located about 40 miles northwest of the plant is not a threat to ISFSI operation.

2.2.2.5 Airports (Reference 2.6.38)

The nearest airport to the plant is Charlevoix Municipal Airport, which is located approximately five miles to the southwest. Charlevoix Municipal is a general aviation facility used primarily by light single-engine aircraft. In Reference 2.6.38, the Nuclear Regulatory Commission concluded that operations associated with the Charlevoix Municipal Airport did not represent an undue risk to the safe operation of the nuclear plant. Because the ISFSI is located within the same owner-controlled area as was the former operating plant, the conclusion stated in Reference 2.6.38 also applies to the operation of the ISFSI. Additional discussions related to aircraft hazards are provided in Chapter 15 of this UFHSR.

2.2.2.6 Military Training Routes (Reference 2.6.11)

Military low level training routes pass approximately 10 miles from the BRP Plant.

In the BRP 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,..." This conclusion also applies to the design of the ISFSI.

2.2.3 SAFETY EVALUATION CONCLUSIONS (Reference 2.6.9)

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 2.6.11 and was about 2×10^{-8} per year in 1984 and has since been further reduced by the closing of military training routes.

We conclude that the BRP site 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 ISFSI.

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 – i.e., 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 was designed 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 that 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.

The 256-foot tower was subsequently removed and present meteorological monitoring is described in Section 2.3.2.

Indications are that the normal meteorology of the site region will produce no significant limitations on ISFSI 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 100, within the 30-mile radius of the site.

2.3.1 NORMAL AND SEVERE WEATHER

The topic of Severe Weather Phenomena was evaluated by the NRC as part of the SEP; this review resulted in a staff safety evaluation (SE) which assumed a licensing basis (Reference 2.6.12) for the following conditions.

Consumers Energy (formerly 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 2.6.13 and the conclusions follow each of the conditions from the NRC Safety Evaluation assumption.

2.3.1.1 Temperature (NRC-SE)

Big Rock Point Environmental Report for Decommissioning (Reference 2.6.1), contains information on normal and extreme temperatures.

The extreme maximum and minimum temperatures appropriate at the BRP site are 86 degrees Fahrenheit (equaled or exceeded one percent of the time) and minus six degrees Fahrenheit (equaled or exceeded 99% of the time).

The FuelSolutions™ Storage System was designed to more severe weather conditions. A temperature of 77°F was used as the long-term annual average design temperature, without solar incidence. Variations in the normal condition ambient temperature in the range of 0°F to 100°F were also considered in the design of the Storage Cask and Transfer Cask. The storage system design steady state temperature extreme was 125°F (average daily temperature), with incident solar radiation and -40°F, with no solar incidence, for off-normal conditions.

2.3.1.2 Thunderstorms and Lightning Strikes (NRC-SE)

BRP Environmental Report for Decommissioning (Reference 2.6.1), 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 BRP Reactor Building can be expected to be subjected, on the average, to one strike every seven years.

Grounded masts acting as lightning receptors provide lightning protection for the Storage Casks on the ISFSI. The height, spacing and configuration of the masts are selected to provide protection for the storage configurations expected on the ISFSI pad. The design provides protection in the event of lightning strikes in accordance with the Lightning Protection Code. ISFSI lightning protection is discussed in Chapter 9 of this UFHSR.

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 Environmental Report for Decommissioning (Reference 1) contains information on snowfall.

Based on the 100-year recurrence accumulated ground snow pack 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.

A bounding value of 100 psf ground snow load is conservatively used for snow and ice loading on the Storage Cask and is applied as a live load.

The Storage Cask loading analysis was performed by FuelSolutions™, using a value of 115 psf instead of the 100 psf used in the analysis contained in the FuelSolutions™ SAR (Reference 2.6.14). The results of the analysis are as follows:

> As a result of using the site-specific snow loading at BRP, the conclusions of the Storage Cask structural analysis reported in the FuelSolutions™ FSAR for the Storage Cask are not affected. The change in ground snow loading from 100 psf to 115 psf leads to a small increase in stresses of the top cover and top liner, but nevertheless all allowable stresses are met for the BRP snow loading. The BRP snow loading did not result in any significant increase in stresses, forces or displacements for all other components of the Storage Cask. In fact, it was shown that the assumed Cask weight used in the original structural analysis bounds the actual maximum weight of the Cask plus the additional snow loading at BRP, and hence this structural analysis remains valid for all the Cask components other than the top cover, top liner and shear lug.

2.3.1.5 Design Wind Speed (NRC-SE)

BRP Environmental Report for Decommissioning (Reference 1), 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 structure design and considering the improvement in terms of its effect on overall structural safety.

2.3.1.6 Tornadoes (NRC-SE)

BRP Environmental Report for Decommissioning (Reference 2.6.1), contains information on severe weather.

Tornadoes have been reported 25 times during the period 1950-1977 within an approximate 60-mile radius from the BRP Site, excluding the water area over Lake Michigan. On the average, one tornado can be expected to occur in the vicinity of the BRP 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 wind speed of 360 miles per hour (a maximum rotational wind speed of 290 miles per hour plus a maximum translational wind speed 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 BRP Site, the characteristics of tornadoes occurring within a 60-mile radius are a maximum wind speed of 310 miles per hour (a maximum rotational wind speed of 250 miles per hour plus a maximum translational wind speed 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 that we believe does not provide a reasonable degree of accuracy for calculations of design of structures that are important to safety.

As previously stated in our letter of January 23, 1981 (Reference 2.6.14), 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:

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

b. Maximum translational wind speed of 55 mph; and

c. 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 BRP site exerts an additional moderating influence on severe storm intensity that has not been taken into account, the above parameters are still considered to be conservative.

2.3.1.6.1 Tornado/Wind Loading for the FuelSolutions™ Storage System

Design criteria are:

Tornado and Wind Loadings		
Rotational Wind Speed	290 mph	
Maximum Translational Wind Speed	70 mph	
Maximum Wind Speed	360 mph	
Radius of Maximum Wind Load	150 feet	
Surface Pressure	356 psf	
Tornado Missiles:		
Automobile	4000 pounds	126 mph
Armor-Piercing Shell	275 pounds	126 mph
Hardened Steel Shell	0.15 pounds	126 mph
High Wind:		
Design Basis Wind (Off-normal Condition)	0.15 pounds	126 mph

2.3.1.7 Severe Weather Conclusions

Analyses of BRP ISFSI important-to-safety structures were completed (refer to Section 9 of this UFHSR). These analyses were performed assuming a tornado wind speed of 290 mph.

ISFSI Severe Weather Conclusions

Explosive overpressure loadings were considered to be the same as the tornado wind pressure load. Based on the evaluation of potential hazards due to nearby industrial, transportation and military facilities addressed above, explosive overpressure loadings are not postulated to exceed tornado wind pressure loads. The pressure loadings generated by an explosion of equipment or combustible fuels in the proximity of the Storage Casks are shown to be bounded by the pressure loadings caused by tornado wind speeds (References 2.6.16 and 2.6.17).

FuelSolutions™ developed calculations and evaluations of the BRP-specific missiles and their impact on the Storage Cask and Transfer Cask to determine if the components would continue to perform their intended functions (References 2.6.16 and 2.6.17). The same methodology was used for these calculations as was used for the original FuelSolutions™ tornado missile analysis. The results for the Storage Cask showed:

a. The Storage Cask was found to maintain its stability when subjected to massive high kinetic energy missile (i.e., a utility pole) during a tornado event;

b. The Storage Cask was found to be stable when subjected to combined effects of tornado wind and missile action;

c. Local damage analysis showed that none of the BRP-defined missiles will penetrate the Storage Cask or the Canister within the Cask; and

d. The force on the Storage Cask as a result of the specific tornado missile impact at BRP was calculated to be 136.6 kips, which is less than the 198 kips calculated previously for the FuelSolutions™ bounding tornado missiles. Since all allowables for the load combinations involving tornado missile loads were met with considerable margins for the Storage Cask components, it can be concluded that all allowables will be met for the specific tornado missile loads at BRP.

The results for the Transfer Cask showed:

a. The Transfer Cask was found to maintain its stability when subjected to massive high kinetic energy missile (i.e., a utility pole) during a tornado event;

b. The Transfer Cask was found to be stable when subjected to combined effects of tornado wind and missile action;

c. Local damage analysis showed that a missile will penetrate the Cask to a depth of 0.615 inch for the one-inch diameter steel rod and 0.865 inch for the 13.5-inch diameter utility pole. In either case, considering the thickness of all components of the Cask, only the Neutron Shield will be entirely penetrated (through wall) as a result of missile penetration. Since the Neutron Shield does not perform any structural function, the structural integrity of the Cask will be maintained; and

d. Stress analysis was performed for all the components of the Transfer Cask when subjected to a combination of wind and missile generated loads during a tornado event. All the stresses are below the corresponding allowable by considerable margins. Furthermore, the stress levels in the Cask due to the combined effects of tornado wind and tornado missiles are below the material yield strength. Therefore, no significant permanent deformation of the Cask structure will result which would prevent retrieval of the Cask contents.

Thus, during a tornado event, the integrity of the Storage Cask and Transfer Cask will be maintained and, as such, they will be able to perform their important-to-safety functions.

2.3.2 METEOROLOGICAL MONITORING

Meteorological data may be obtained from the National Weather Service (NWS).

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 2.6.18), was completed 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 doses to the public during and following a postulated design basis accident.

2.3.3.1 General 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.

2.3.3.1.1 Criteria for ISFSI Accident Analyses

For the permanently defueled plant with all spent fuel stored at the ISFSI, accidents involving reactor operation and wet storage of spent fuel in a spent fuel pool are not feasible. Chapter 11 of the BNFL FuelSolutions™ Storage System FSAR, Document Number WSNF-220 and BNFL FuelSolutions™ W74 Canister Storage FSAR, Document Number WSNF-223 (SARs), provide accident analyses applicable to dry storage of spent nuclear fuel at BRP. Design basis off-normal and postulated accident events include those resulting from mechanistic, non-mechanistic, and natural phenomena.

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

2.3.3.2 Summary of Previous Analysis Methods

2.3.3.2.1 Criteria for Permanently Defueled Plant Analysis

For the permanently defueled plant, accidents involving reactor operation were not feasible. The bounding event for the permanently defueled plant was a heavy load drop in the spent fuel pool. The assumptions used in performing the analysis of this event included:

a. Offsite release occurs over a two-hour interval, per Regulatory Guide 1.25 (Safety Guide 25), Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident, US NRC, 1972;

b. No credit for containment ventilation isolation is taken;

c. X/Q is 6.48E-04 sec/m^3 for the ground level release, per Regulatory Guide 1.25 for dose to offsite population (closest site boundary, 805 meters);

d. Dose conversion factors are from EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, May 1992, and EPA-402-R-93-081, External Exposure to Radionuclides in Air, Water, and Soil, September 1993; and

e. Ground level release results in higher offsite doses, thus has been assumed in calculation of doses.

Big Rock Point has implemented the guidelines of the EPA Manual of Protective Action Guides (PAGs) and Protective Actions for Nuclear Accidents, EPA-400 (Reference 15.11.5) 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 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.

2.3.3.2.2 Criteria for ISFSI Analysis Methods

Transport of airborne radioactivity from the BRP Site was calculated by several different means during the lifetime of the plant. Briefly, the techniques and reference documentation for each are as follows:

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

b. Safety analyses, including ISFSI Emergency Plan and ISFSI Emergency Implementing Procedure calculations - Atmospheric diffusion parameters from Regulatory Guide 1.3, assuming ground level release.

2.3.3.2.3 Discussion

An evaluation of X/Q (Note 1) values at the BRP Plant was presented in Section 14 of the November 14, 1961 BRP 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 three 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 sec/m^3 for 8-24 hours, 8E-05 sec/m^3 for 1-4 days and 1.7E-05 sec/m^3 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 sec/m^3 and 6E-04 sec/m^3 is not significant with respect to meeting 10 CFR 100 limits.

(X/Q Note 1) X = the short term average centerline value of the ground level concentration (curie/meter^3)

2.3.3.2.4 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 environmental dose purposes. For all other calculations, Regulatory Guide 1.3 values will be used. Assuming a ground level release for all accident conditions, the following values X/Q are applicable at 0.19 miles (300 meters), which is the Exclusion Area Boundary (EAB):

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

Q = amount of material released (curie/sec)

(D/Q Note 2) D = deposition constant

Q = amount of material released

2.4 HYDROLOGY

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 November 14, 1961 FHSR.

The surface drainage of the immediate area of the reactor plant building is from the building locations directly to Lake Michigan rather than inland towards inhabited areas and local wells. There appears to be a high probability that any accidental release of material at the immediate location of the plant buildings which penetrated to ground water would also be drained directly into Lake Michigan.

The surface soils are of types that generally have low permeability and might be expected to have fair to good ion exchange capacity.

The principal currents in Lake Michigan important to the site for considerations of liquid waste disposal are generally favorable. Most of the time it is indicated that the current along the plant shore will be from east to west, with significant rapid diffusion into the main body of upper Lake Michigan.

A Hydrological Survey to determine currents and dilution of Lake Michigan and Little Traverse Bay in the region near the site was completed during the summer of 1960 under the direction of Professor John C Ayers, University of Michigan, the Report issued November 1961 is contained in Volume Two of the November 14, 1961 FHSR.

The NRC completed an evaluation of Hydrology topics as part of the Systematic Evaluation Program (SEP). The results of this evaluation (Reference 2.6.21) and the evaluation conclusions (Reference 2.6.22) along with the revised NRC Safety Evaluation Reports (References 2.6.23 and 2.6.24) were utilized to provide the following summary and conclusions for Hydrology issues.

2.4.1 HYDROLOGICAL DESCRIPTION

Refer to Figures 2.1 and 2.2 for Location and Site Plan Maps.

Big Rock Point Nuclear Plant is located on the eastern shore of Lake Michigan in the northwest section of Michigan's lower peninsula, on the south side of Little Traverse Bay. It is three miles north of Lake Charlevoix, an inland extension of Lake Michigan. To the east of the site is Susan Creek, which flows from Susan Lake north into Lake Michigan. Plant grade varies from 592.5 to 594 feet MSL at the containment, to 596.6 feet MSL at the stack. Nominal finished grade at the containment structure is 592.5 feet MSL.

2.4.1.1 Lake Water Level

The water level of Lake Michigan has varied between 576 and 584 feet MSL. Lake Michigan water level experiences long term, seasonal, and short-term variations. Long-term variations are caused by periods of higher or lower-than-usual precipitation lasting several years and extending over a large part of the Great Lakes watershed. The highest recorded (1905-1986) mean monthly water level on northern Lake Michigan near the site was 582.6 ft MSL (September 1986). Seasonal variations average one foot between high water in July and low water in February. In some years, the range may be as high as two feet. Short-term water level fluctuations have a period of a few hours and have produced changes in water level of up to three feet. The minimum monthly level of Lake Michigan was elevation 576.4 MSL (USGS-March, 1964).

2.4.1.2 Watershed

The BRP structures are located in an area where surface runoff flows directly into Lake Michigan. There are no perennial streams or rivers in this watershed, which has an area of between three and four square miles.

2.4.1.3 Drainage

Site drainage from building areas is generally away from the structures toward Lake Michigan. Some runoff from high ground was diverted around the former plant to the lake by a ditch on the south and west sides of the site. Drainage areas are well vegetated and relatively flat.

The drainage system as necessary to prevent ponding on the site is described in the License Termination Plan.

2.4.1.4 Groundwater

Groundwater at the BRP site moves north into Lake Michigan from the groundwater divide between Lakes Charlevoix and Michigan. At the site of the former operating plant and the ISFSI, the soil is well drained. Before construction, the water table elevation was approximately 580 feet MSL. A thick sequence of Traverse limestone is overlaid by 50 feet of compact clay till, interbedded with artesian sand zones. The top 10 feet of limestone are badly fractured, and groundwater conditions are artesian. The fractured bedrock is directly connected with Lake Michigan and the groundwater gradient responds to short-term lake water level variations.

In 1994, to support the site radiological scoping survey for decommissioning, nine (9) ground water monitoring wells were installed on the plant site. Periodic groundwater monitoring is described in the License Termination Plan.

2.4.1.5 Hydrologic Design Basis

For the permanently defueled plant with all fuel stored at the ISFSI facility, safe shutdown capability is not required and no structures are safety-related. Therefore, the discussions on hydrologic design basis were historical and were deleted in revision 14 of the UFHSR.

2.4.2 FLOODS

The potential for a flooding event that could exceed the 594.0 feet MSL at the Turbine Building is very low.

2.4.2.1 Flood History

There is no record of any flooding at BRP.

2.4.3 PROBABLE MAXIMUM FLOODING (PMF)

2.4.3.1 PMF from Probable Maximum Precipitation (PMP) Event

The possibility of plant flooding from local intense PMP was evaluated for the following drainage areas:

1. Susan Creek;
2. Unnamed drainage basin south of the plant site; and
3. On-site area.

2.4.3.1.1 Susan Creek

The Susan Creek drainage basin lies east of the plant and drains to Lake Michigan. The drainage basin has an area of 5.7 square miles. For Susan Creek near the plant site a PMF peak discharge of 20,000 cfs was determined which resulted in a peak water surface elevation of 590 feet MSL. There is a ridge line above this elevation that separates the Susan Creek drainage basin from the ISFSI, hence, a PMF on Susan Creek will not flood the site.

2.4.3.1.2 Unnamed Drainage Basin

The unnamed drainage basin adjacent to the plant site has a drainage area of approximately 0.75 square miles. Using a one-hour Probable Maximum Precipitation (PMP) of 16 inches for this location (current licensing criteria) resulted in a revised analysis as follows:

The watershed is divided into the two sub-areas (a west and east area) by a railroad bed that runs into the former plant area. The west sub-area was estimated by the NRC Staff to have a PMF peak flow of 2,640 cfs. The capacity of the drainage ditch and road cut to the west side of the former plant is such that flood elevations would not exceed 594 feet MSL along the opposite side of the railroad bed that comes in beside the location of the former turbine building. Also, this maximum elevation of 594 feet MSL is at least two feet below the railroad bed and thus this flood flow would be blocked from flowing onto former plant. The peak PMF flow from the east sub-area was estimated at 1280 cfs and by the nature of the topography and situation of plant site would drain right onto former plant grade. The flow would spread out through the former parking lot and then drain through trees and a cleared space into Lake Michigan. From an evaluation of the flow depths, the maximum depth at the south end of the former turbine building area was calculated to be 593.0 feet MSL.

2.4.3.1.3 Former Operating Plant Area

The former operating plant area analyzed was a small semi-closed basin adjacent to the turbine generator and reactor buildings. This area is bounded by an access road having an elevation of 595.4 feet MSL and is drained by a culvert system. If the culvert system is plugged, water from a local intense PMP will flow and drain onto plant grade as outlined for the "Unnamed Drainage Basin" above. The possibility of complete blockage of the yard drain is highly unlikely. Yard drains were removed in accordance with the License Termination Plan.

2.4.3.1.2 ISFSI Area

The maximum flooding height (surge height) from lake flooding is elevation 587.4 feet. The nominal finished grade at the ISFSI pad is elevation 618.5 feet. Thus, the Storage Casks on the pad are not subjected to the maximum predicted flood height from lake flooding.

2.4.3.2 PMF from Lake Flooding

Surge heights resulting from a moving squall line storm and wind storm were determined in the "High Water Level Study" BRP Plant, performed by RM Noble & Associates, Job No. 13-02, June 14, 1982. These surge heights were combined with maximum mean monthly lake levels and the results submitted to the NRC (Reference 2.6.22).

The results of these analyses gave a high water elevation for the moving squall line storm (surge plus maximum mean monthly lake level) of 584.1 feet MSL. From the consultant's report wave run-up would add one to two feet to the high water level for the squall line storm; thus the total height would not be greater than 586.1 feet MSL. Static water level from this event would rise in the intake structure to 584.1 feet MSL, about seven inches above the pump floor level of 583.5 feet MSL.

For the wind storm, the high water level was determined from a combination of maximum-mean monthly lake level, plus set-up, plus wave run-up resulting in a high water level of 587.4 feet MSL. That is also below the north (former) plant grade elevation of 590 feet MSL.

An analysis (Reference 2.6.25) was performed to determine if the drainage designed for the ISFSI pad would also prevent flooding of the pad from upstream drainage basin, considering a 50-year and 100-year storm event. The analysis verified that the drainage ditch and storm water piping was adequate to keep the pad elevation for being reached by flood waters during the Probable Maximum Flood (PMF).

2.4.4 PROBABLE MAXIMUM PRECIPITATION (PMP)

The PMP for the BRP site is based on a drainage area of 10 square miles, which is applicable to smaller areas also. It is found to be 22.5 inches for the most severe six-hour period of the assumed probable maximum storm. This value is taken from Hydro-meteorological Report Number 33, National Weather Service.

The water level from PMP would not have a depth greater than flooding depths calculated for the Probable Maximum Flood (PMF) from the adjacent watershed (593.6 feet MSL).

2.4.5 LOSS OF ULTIMATE HEAT SINK (UHS)

With the spent nuclear fuel stored in passive Casks on the ISFSI, no heat sink is required. The discussion in this section was deleted.

2.4.6 FLOOD EMERGENCY OPERATIONAL REQUIREMENTS

No SSCs are required for response to flooding with spent nuclear fuel stored on the ISFSI.

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.

2.5.1.1 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.

2.5.1.2 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 Environmental Report for Decommissioning (Reference 2.6.1).

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 did 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 seven 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 former 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.

2.5.1.3 Regional Geology

Regional Geology in BRP Environmental Report for Decommissioning (Reference 2.6.1), was extracted from the NRC assessment of Systematic Evaluation Program Topic II-4 (Reference 2.6.26).

2.5.1.4 Site Geology

Site Geology was extracted from the NRC assessment of Systematic Evaluation Program Topic II-4 (Reference 2.6.26) and is contained in BRP Environmental Report for Decommissioning (Reference 2.6.1).

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 2.5.1.5.

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

2.5.1.5 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 2.6.26):

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 vuggy zone between 130 and 190 feet 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. 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 2.6.27) as follows:

Big Rock Point 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.

2.5.1.6 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. 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 2.6.26) 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 their review, the NRC didn'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.

2.5.1.7 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 that would cause major damage at BRP is very low. As a result of the SEP, (Reference 2.6.28) the seismicity of the BRP vicinity has been reviewed by experts employed by the NRC, the SEP Owners Group and by BRP (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 BRP might, at worst, suffer minor cracking.

Finally, preliminary calculated results from the BRP structural evaluations indicated 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. Note that with spent fuel permanently stored at the ISFSI, no safety-related structures exist at BRP.

In summary, earthquakes are not very probable at BRP. 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 BRP structures showed no significant damage occurs to the structures from earthquakes of the size proposed in BRP letter dated October 10, 1980, Response to Staff Letter dated August 4, 1980, Proposed Seismic Evaluation Program and Basis for Continued Interim Operation (Reference 2.6.29). Independent work completed for the BRP Probabilistic Risk Assessment indicated very long return periods for earthquakes of this size.

<u>Summary of Seismic Design Considerations</u>

A summary of the BRP seismic resistance from SEP Topic III-6 Seismic Design (Reference 2.6.29) is provided below:

The initial seismic criteria as applied to BRP were based on static requirements of the 1958 edition of the Uniform Building Code. The containment design was based on a 0.05 g horizontal static coefficient. The turbine building, concrete stack, intake structure, control room and rad waste storage buildings were designed based on a 0.025 g horizontal static coefficient. These compare with more recent requirements that assume a 0.12 g (Reg Guide 1.60) safe shutdown earthquake. The ISFSI facility utilized the 0.12 g horizontal and vertical static coefficients for site-specific safe shutdown earthquake requirement.

2.5.2.1 Response Spectra

Various seismic design Response Spectra have been used in the SEP to demonstrate the seismic design adequacy of BRP:

a. 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 0.102 g and 5% damping.

b. 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 0.11 g.

c. 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 0.12 g. This seismic input consists of a sample problem earthquake having a zero period horizontal ground acceleration equal to 0.12 g.

d. 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. The May 5, 1982 letter provided 0.12 g Reg Guide 1.60 spectrum and the site-specific spectrum issued by the NRC (letter of June 8, 1981) was 0.104 g.

e. 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 BRP.

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., 0.11 g). Extensive studies of amplification at BRP 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 (0.12 g) earthquake or the NRC site specific (0.104 g) earthquake as both are acceptable seismic design bases.

2.5.2.2 Historical Hazard Analysis

The following historical hazard analysis summary was extracted from (Reference 2.6.37) 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 BRP may only be of marginal safety significance:

The seismic hazard at BRP 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 BRP 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 0.03 g to 0.1 g. 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 0.001 g 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 BRP of 0.03 g 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 BRP 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 (0.1 g) for the site. Based on the above, the chance that BRP will experience earthquake ground motion of any significance is extremely small.

2.5.2.3 Safe Shutdown Earthquake (SSE)

10 CFR 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 that 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 BRP Site, this maximum (peak) ground acceleration is 0.12 g. It should be noted that the 0.12 g Reg. Guide 1.60 Spectrum envelopes both the NRC Site Specific Spectra and CPCo's BRP Site Specific Spectra as discussed in Section 2.5.2.1.

2.5.2.4 Operating Basis Earthquake (OBE)

Values have not been tabulated or depicted for the BRP 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 BRP Plant are contained in D'Appolonia Report dated August, 1983 (Reference 2.6.30) and in (Reference 2.6.31).

2.5.2.6 FuelSolutions™ W150 Storage Cask Site-specific Sliding Analysis

The FuelSolutions™ W150 Storage Cask has been evaluated for sliding in the unlikely event of a seismic event. This analysis is contained in the FuelSolutions™ Storage System SAR, Document WSNF-220, Section 2.6.3. The results demonstrated that the factor of safety against tip-over due to an earthquake with a peak ground acceleration of 0.5 g is at least 1.4. A conservative sliding coefficient of friction of 0.8 was used in this analysis. A site-specific evaluation determined that the extent of Cask sliding during the postulated seismic event is acceptable. The coefficient of friction of 0.3 was bounded by that used in the SAR evaluation (Reference 2.6.32).

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 2.6.26):

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 BRP Plant.

2.5.3.1 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 100.

2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

The following assessment of the foundations and earthworks properties under anticipated loading conditions including earthquakes was extracted from Systematic Evaluation Program Topic II-4.F, Settlement of Foundations and Buried Equipment (Reference 2.6.22). It remains as historical information with regard to the former operating plant structures and as a general discussion that could be applied to the ISFSI site.

Figure 2.3 shows the general layout of the site structures. In addition to the structures shown in Figure 2.3, an Offshore Intake Structure and Offshore Intake Pipe Line was also part of the plant. These supplied 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 connected 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 2.6.31) presented detailed description and functions of former operating plant safety-related subsurface piping and foundations.

The foundations of the former safety-related SSCs that were considered in the NRC SEP Topic II-4.F settlement evaluations were:

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

2.5.4.1 Foundation Data

Source of Information

Geotechnical data available for this site are:

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

b. "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; and

c. "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.

2.5.4.2 Subsurface Conditions

2.5.4.2.1 Former Operating Plant Site

The former operating 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 feet 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 eight to 33. The ground water table is controlled by the adjoining lake level and is at an approximate depth of eight feet below ground surface.

Thirty to 35 feet 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 from 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) ratio ranged from zero 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 ratio ranged from zero 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 ratio 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 ratio ranged from 55 to 90. The deepest boring at the site (201 feet deep) was terminated in this stratum.

2.5.4.2.2 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 ran from the offshore intake structure to the screen well pumphouse/Diesel Generator/Discharge Structure. Its final disposition and configuration is described in the License Termination Plan.

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.

2.5.4.2.3 Soil Properties

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

1. insitu moisture content (six 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 three TSF cohesion and 30 degrees angle of internal friction

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

2.5.4.3 Settlement of Structures

2.5.4.3.1 Site Structures

All the buildings within the 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 three 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.

The Licensee had not initiated any settlement-monitoring program and has no records of any settlement monitoring. The plant was 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 since plant construction, the potential for future settlement is negligible.

2.5.4.3.2 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 2.6.36). Based on the information available, it is concluded that the past and future settlement of this structure is minimal.

2.5.4.3.3 Liquefaction and Seismic Settlement

The postulated safe shutdown earthquake (SSE) ground acceleration for BRP is 0.12 g. The glacial till, material beneath the mat foundation is a very stiff (approximately 20 percent clay content) material that is not susceptible to liquefaction. The granular material (eight 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-foot thick sand bedding under the intake structure might liquefy and the consequences would be seismically induced settlement of negligible magnitude.

2.5.4.4 Settlement of Buried Equipment

2.5.4.4.1 Buried Fire Main Piping System (BFMPS) and Electrical Cables

Fire main piping system and electrical cables within the plant site were buried at a minimum depth of six feet below ground surface. Fire and yard piping were removed in accordance with the License Termination Plan. Discussion on settlement of these components is no longer relevant, since they no longer support Q-Listed functions. Information on settlement was deleted in revision 14 of this UFHSR.

2.5.4.4.2 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 six-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 an 18-inch thick bedding over glacial till) under static conditions.

2.5.4.4.3 Liquefaction and Seismic Settlement

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; and

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.

2.5.4.4.4 Evaluation Summary Conclusion

Based on review of the CPCo Safety Analysis Report (Reference 2.6.34) and information obtained during the site visit, the NRC Staff concurs with the Licensee's conclusions settlement of seismic Category I foundations and buried equipment was not a safety problem at the BRP Nuclear Power Plant.

2.5.5 STABILITY OF SLOPES

Consumers Power Company and the NRC evaluated SEP 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 2.6.35)

The NRC Staff concludes that slopes stability is not a radiological safety concern at the BRP 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 BRP as documented in the NRC letter dated April 16, 1979 and confirmed by BRP in the June 22, 1979 response.

2.6 REFERENCES

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2.6.4 US NRC Letter dated June 6, 1980, SEP Topics II-1.A, II-1.B, and II-1.C (BRP)

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

2.6.6 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

2.6.7 Population Characteristics of northwest Michigan Counties, Developed by Nancy Haywood, Director, Data Research Center, Incorporated, Traverse City, Michigan, June 1980

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

2.6.9 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)

2.6.10 EA-FC-704-61, Regulatory Guide 1.91 (Evaluation of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plants) Evaluation of ISFSI Pad Location

2.6.11 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

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2.6.13 CPCo Letter dated March 9, 1981, SEP Topic II-2.A, Severe Weather Phenomena (BRP Microfilm Cartridge/Frame 4830/1812)

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2.6.17 Big Rock Point Nuclear Plant, DFS Explosion Hazards Analysis, S&L Calculation Number S-11102-10, Revision 0 (BRP Microfilm Cartridge/Frame 4830/2240)

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

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

2.6.20 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

2.6.21 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

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

2.6.23 NRC Letter dated March 22, 1984, SEP Hydrology Issues

2.6.24 Calculation S-10881-03, Revision A, Big Rock Point ISFSI (Analysis of PMF for ISFSI)

2.6.25 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

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

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2.6.29 Derivation of Site-specific Seismic Floor Response Spectra, Seismic Safety Margin Evaluation, D'Appolonia Project Number 78-435, August 1983

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

2.6.31 EA-FC-704-56, ISFSI Slab – Results of Field Test for Friction after Concrete Scabbling

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

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

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

2.6.36 NRC Letter dated June 30, 1982, Assessment of Possible Soil Amplification at Big Rock Point Site

2.6.37 Letter from US Nuclear Regulatory Commission to Consumers Power Company, dated August 12, 1982, SEP Topic III-4.D, Site Proximity Missiles (including Aircraft) Big Rock Point Nuclear Plant

2.6.38 Letter from the US Nuclear Regulatory Commission to Big Rock Point Nuclear Power Plant, dated September 30, 1998, Exemption from Certain Requirements of 10 CFR 50.54(q) Regarding Offsite Emergency Planning Activities at Big Rock Point Nuclear Power Plant and Approval of Defueled Emergency Plan

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

On September 23, 1997, Consumers Energy submitted Big Rock Point (BRP) Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certified that all fuel had been removed from the reactor and that the reactor would not be refueled. Since this date, Consumers Energy has been dismantling the facility. On March 26, 2003, all spent fuel was transferred from the wet storage (Spent Fuel Pool) in containment to a dry fuel storage system located on an Independent Spent Fuel Storage Installation (ISFSI).

With all fuel stored on the ISFSI, the former plant industrial area contains no structures, systems, or components that are safety-related. Formerly, this chapter addressed safety-related Structures, Systems, and Components (SSCs) and their design criteria. Quality requirements for SSCs and activities are outlined in BRP Volume 34, Quality Program Description for Big Rock Point. Independent Spent Fuel Storage Installation components are quality-classified as important-to-safety, consistent with 10 CFR 72. Quality classifications and design criteria for ISFSI components are contained in the BNFL FuelSolutions™ Safety Analysis Reports (SARs) document numbers WSNF-220 and WSNF-223.

With all spent fuel permanently stored on the ISFSI, 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 the UFHSR. No site systems, structures and components (SSCs) failure could not cause a radioactive release at the site boundary having the potential of exceeding the limits of 10 CFR 100.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section previously included a discussion of the Systematic Evaluation Program (SEP) and historical information regarding design review of the former operating plant. Reference 3.12.1 provides this discussion. The discussion has been removed from the UFHSR as the systems and structures have been dismantled.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

For BRP as an operating nuclear plant, Seismic and System Quality Group Classifications of Components/Subsystems were made according to the safety functions to be performed. This section formerly contained selected structures, systems, and components (SSCs) for the BRP Operating Plant and the code required for licensing criteria (based on NRC Regulatory Guide 1.26, Revision 3, Section 50.55a of the Code of Federal Regulations, and the codes and standards used when the systems and components were originally built). This section also contained information regarding the Seismic Classification of the systems and components. Current NRC design criteria which were not in effect during the design of BRP require 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 former operating plant features were designed is defined as the Safe Shutdown Earthquake (SSE) in 10CFR 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 were designed to remain functional if an SSE occurs are designated Seismic Category I in Regulatory Guide 1.29, Revision 3.

For BRP, as an operating nuclear power plant, the SSE maximum vibratory ground motion was 0.12g, Regulatory Guide 1.60, Revision 1, Response Spectrum.

For an operating nuclear power plant, Regulatory Guide 1.29, which identifies safety-related 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. With all spent nuclear fuel permanently stored on the Independent Spent Fuel Storage Installation (ISFSI), no "plant" structures, systems, or components (SSCs) fall within the scope of Regulatory Guide 1.26, Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants, or Regulatory Guide 1.29, Seismic Design Classification.

On September 23, 1997; Consumers Energy submitted BRP Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certified that all fuel had been removed from the reactor and that the reactor would not be refueled. In March of 2003, all spent fuel had been permanently removed from containment and stored at the ISFSI.

With all spent nuclear fuel permanently stored on the ISFSI, no "plant" SSCs fall within the scope of Regulatory Guide 1.26, Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants, or Regulatory Guide 1.29, Seismic Design Classification.

No SSCs exist whose failure could cause a radioactive release at the site boundary having the potential to exceed the limits of 10 CFR 100.

3.2.1 SEISMIC CLASSIFICATION

No SSCs are "Safety-Related" and Seismic Category 1 with all spent nuclear fuel permanently in dry storage on the ISFSI.

3.2.2 QUALITY GROUP CLASSIFICATION

Regulatory Guide 1.26, which was used to establish piping system boundaries during wet fuel storage in a Spent Fuel Pool, is no longer applicable. No important-to-safety SSCs or 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, exist. Therefore, the discussion on the former Systematic Evaluation Program (SEP) on Quality Group Classification was deleted.

Big Rock Point Plant is a general licensee under Part 72. The components described in the BNFL SARs as important-to-safety are controlled under BRP's Quality Assurance program at a level equivalent to the safety-related components.

3.3 <u>WIND AND TORNADO LOADINGS</u>

3.3.1 WIND LOADINGS

This section formerly contained information on the design wind loading for various structures in the Former operating plant Area. The structures addressed are no longer needed, with all spent fuel on the ISFSI. Discussions on reactor building, stack, and radwaste building wind loading were deleted.

3.3.2 TORNADO LOADINGS

The original design criteria for the BRP Nuclear Plant did not consider tornado wind loadings as tornadoes rarely occur in the plant area.

As discussed in Chapter 2 of this report under Severe Weather Conclusions, the design basis tornado characteristics used by Big Rock Point (BRP) during the evaluation of SEP Topic III-2, Wind and Tornado Loading, was based upon an analysis performed using ANSI/ANS 2.3, Draft 2, Revision 4, dated April 1980, "Guidelines for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Plant Sites." Using this standard, a maximum tornado wind speed of 250 mph (annual probability 10^{-7}) was indicated (refer to BRP January 23, 1981 letter). Thus, the following design basis tornado intensity characteristics were considered to be representative of extreme weather winds at BRP (Reference 3.12.2).

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

2. Maximum translation wind speed of 55 mph

3. Maximum pressure change of 1.35 psi

These characteristics were used by BRP during the SEP evaluation of structures and structural elements.

Refer to Section 4.5.2 of Reference 3.12.1 for details of the NRC SER for this topic.

UFHSR Chapter 9 discussion of ISFSI design considerations documents the seismic input in the design of ISFSI SSCs.

3.4 WATER LEVEL (FLOOD) DESIGN

"Flooding" and "Flood Operational Requirements" are addressed in Section 2.4 of this UFHSR.

3.4.1 FLOOD PROTECTION

UFHSR Chapter 9 discussion of ISFSI design considerations documents the probable maximum flood on the input in the design of ISFSI SSCs. The ISFSI pad will not be undercut by floodwaters from the Probable Maximum Flood event, thereby maintaining the design requirements

3.4.1.1 Flood Protection Measures for Structures

The "Effects of High Water Level on Structures" was evaluated by BRP and the NRC under Systematic Evaluation Program (SEP) Topic III-3.A. The following information, extracted from the BRP Evaluation (Reference 3.12.3), provides the water level that was considered in the design of structures and discusses the design provisions required to account for uplifting due to buoyancy.

Lake levels used in the original design of the plant structures were:

Extreme High:	100 Yr	Elevation 583.6 feet
Summer		Elevation 578.5 feet
Winter		Elevation 577.5 feet
Extreme Low:	100 Yr	Elevation 577.4 feet

A search through available documentation did not reveal any special design provisions to account for uplifting due to buoyancy; however, the possibility of adverse effects resulting from such uplifting forces was reviewed.

3.4.2 ANALYTICAL AND TEST PROCEDURES

Methods by which effects of design basis flood or groundwater conditions are applied to structures are addressed in Chapters 2 and 3 of this UFHSR.

3.4.3 INSERVICE INSPECTION OF WATER CONTROL STRUCTURES

With all fuel in dry storage on the ISFSI, no water control structures are safety-related or perform safety-related or quality functions. This section formerly discussed surveillance and preventive maintenance of the screenhouse and associated components. Since these SSCs have been dismantled, the information contained in this section was deleted.

3.5 MISSILE PROTECTION

3.5.1 MISSILE EFFECTS

With the reactor permanently defueled and all spent fuel stored on the ISFSI and the primary system piping removed, possibility for pressurization of the primary system has been eliminated. In addition, no SSCs important to safety remain in the former plant industrial area.

Tornado missiles were evaluated in response to Systematic Evaluation Program (SEP) Topic III-4.A on March 16, 1982 (Reference 3.12.4). Additional information regarding the calculations and methodologies used was submitted by letter dated June 16, 1982 (Reference 3.12.5). The purpose of the evaluation was to provide an assessment of the adequacy of BRP to withstand the impact of tornado missiles.

Missiles "C" and "F" as described in the NRC Standard Review Plan are appropriate for performance of the evaluation:

Steel rod: One inch in diameter by three feet long; weight eight pounds; horizontal velocity of 0.6 times total tornado velocity.

Utility Pole: 13.5 inches in diameter by 35 feet long; weight 1,490 pounds; horizontal velocity of 0.4 times total tornado velocity.

The following design basis tornado intensity characteristics were used in the BRP Analysis:

a. Maximum wind speed of 250 mph (combined rotational and translational)
b. Maximum translational wind speed of 55 mph
c. Maximum pressure change of 1.35 psi

In accordance with the above design basis tornado velocity, the total horizontal velocity for the two postulated missiles are:

a. Steel rod: 220 feet/second
b. Utility pole: 147 feet/second

These missiles are considered to be capable of striking in all directions with vertical speeds equal to 80% of the horizontal speeds listed above.

<u>BRP Evaluation Results</u>

The evaluation results for each of the three categories required by this SEP Topic are summarized below and have been amended to include site-specific evaluation results for the ISFSI:

a. Reactor Coolant Pressure Boundary

   The primary system pressure boundary piping has been removed; the conclusions included in this section are not applicable.

b. Safe Shutdown Systems

   There are no safe shutdown systems remaining; therefore the conclusions of this section have been deleted.

c. Potential Accidents that Could Result in Unacceptable Offsite Exposures

   The ISFSI components were evaluated in BNFL FuelSolutions™ SARs. No unacceptable offsite exposures will result from a tornado missile strike.

<u>Site-specific Missile Evaluations and Their Impact on the FuelSolutions™ Storage and Transfer Casks</u>

The same methodology was used for these calculations as was used for the original FuelSolutions™ tornado missile analysis.

The results for the Storage Cask showed:

a. The Storage Cask was found to maintain its stability when subjected to massive high kinetic energy missile (utility pole) during a tornado event;

b. The Storage Cask was found to be stable when subjected to combined effects of tornado wind and missile action;

c. Local damage analysis showed that none of the BRP defined missiles will penetrate the Storage Cask or the Canister within the Cask; and

d. The force on the Storage Cask as a result of the specific tornado missile impact at BRP was calculated to be 136.6 kips which is less than the 198 kips calculated previously for the FuelSolutions™ bounding tornado missiles. Since all allowables for the load combinations involving tornado missile loads were met with considerable margins for the Storage Cask components, it can be concluded that all allowables will be met for the specific tornado missile loads at BRP.

The results for the Transfer Cask showed:

a. The Transfer Cask was found to maintain its stability when subjected to massive high kinetic energy missile (utility pole) during a tornado event;

b. The Transfer Cask was found to be stable when subjected to combined effects of tornado wind and missile action;

c. Local damage analysis showed that a missile will penetrate the Cask to a depth of 0.615 inch for the one-inch diameter steel rod and 0.865 inch for the 13.5-inch diameter utility pole. In either case, considering the thickness of all components of the Cask, only the Neutron Shield will be entirely penetrated (throughwall) as a result of missile penetration. Since the Neutron Shield does not perform any structural function, the structural integrity of the Cask will be maintained; and

d. Stress analysis was performed for all the components of the Transfer Cask when subjected to a combination of wind and missile generated loads during a tornado event. All the stresses are below the corresponding allowable by considerable margins. Furthermore, the stress levels in the Cask due to the combined effects of tornado wind and tornado missiles are below the material yield strength. Therefore, no significant permanent deformation of the Cask structure will result which would prevent retrieval of the Cask contents.

Thus, during a tornado event, the integrity of the Storage Cask and Transfer Cask will be maintained and as such, they will be able to perform their important-to-safety functions.

3.5.1.4 Site Proximity Missiles Including Aircraft

Section .2 of this UFHSR, "Nearby Industrial, Transportation, and Military Facilities," provides a description of the potential missile hazards and aircraft hazards. BRP evaluated SEP Topic III-4.D, Site Proximity Missiles Including Aircraft and submitted this to the NRC December 14, 1981 (Reference 3.12.7). The evaluation of nearby highways, railways, waterways, commercial airports, military maneuvers, underground lines, industrial complexes, and manufacturing facilities concluded there are no industrial hazards in the vicinity of the BRP that could affect safe operation.

Evaluation Conclusions

Externally generated missiles resulting from the nearness of airports, transportation routes, and industrial military facilities were not postulated for the BRP Nuclear Plant, therefore, no specific protection was required for proximity missiles. Based upon BRP evaluation of this topic, the NRC provided a final Safety Evaluation and performed an independent review of the risks associated with all aircraft activities near the BRP site in response to interveners contentions and Atomic Safety and Licensing Board questions regarding aircraft hazards (Reference 3.12.8). The staff concluded that potential site proximity missiles did not pose a significant hazard to the safe operation of BRP.

NRC Safety Evaluation Conclusion (Reference 3.12.8)

Based on our review, operation of the BRP Plant did not present an undue risk to the health and safety of the public as a result of potential aircraft and site proximity missiles.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

The results of this analysis are no longer applicable. The ability to shut down the reactor is no longer important as the fuel has been permanently moved to the ISFSI and piping failure is no longer credible. The ability to shutdown the reactor is no longer important as the fuel has permanently been moved to the ISFSI and piping failure is no longer credible.

3.7 SEISMIC DESIGN (Reference 3.12.1)

This section and associated subsections previously addressed and/or referenced for the analyses provided in Chapters 2 and 3 of this UFHSR. Specific Seismic Analyses performed for BRP SSCs are identified in Section 3.2 of this UFHSR.

3.8 DESIGN OF CATEGORY I STRUCTURES

On September 23, 1997, Consumers Energy submitted BRP Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certified that all fuel had been removed from the reactor and that the reactor would not be refueled. All spent fuel has been moved to dry storage at the ISFSI. The discussion of the Seismic Category I structures has been deleted.

3.8.1 CONTAINMENT

The information in this section has been deleted.

3.8.2 CONCRETE AND STEEL STRUCTURES

The information in this section pertained to an operating nuclear plant. With the plant being dismantled and all spent nuclear fuel stored on the ISFSI, this information is no longer applicable. A discussion of the design codes and standards can be found in References 3.12.7 and 3.12.10.

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

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

The information in this section pertained to an operating nuclear plant. With all spent nuclear fuel permanently stored on the ISFSI, the information is no longer applicable and has been deleted.

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

3.9 MECHANICAL SYSTEMS AND COMPONENTS

Special Topics (ASME Class; ISI; IST; IGSCC; Reactor Material Surveillance; Reactor Internals; and Loose Parts Monitoring)

As of March 26, 2003, all spent nuclear fuel is permanently stored on the Independent Spent Fuel Storage Installation (ISFSI). ASME code classifications, inservice inspection, inservice testing, reactor material surveillance, reactor internals, and loose parts monitoring programs formerly discussed in this section are no longer applicable and have been deleted.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

On September 23, 1997, Consumers Energy submitted BRP Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certified that all fuel had been removed from the reactor and that the reactor would not be refueled. All spent fuel has been moved to dry storage at the ISFSI. The discussion of Seismic Category I equipment has been deleted.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

On September 23, 1997, Consumers Energy submitted BRP Plant's "Certification of Permanent Fuel Removal" to the NRC. This submittal certified that all fuel had been removed from the reactor and that the reactor would not be refueled. All spent fuel has been moved to dry storage at the ISFSI. The discussion of the environmental design of equipment has been deleted.

3.12 REFERENCES

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

3.12.2 CPCo Letter dated August 3, 1982, SEP Topic III-2, Wind and Tornado Loading (CPCo Evaluation)

3.12.3 CPCo Letter dated August 8, 1980, BRP Plant Response to Request for Additional Information - SEP Structural Topics (III-2, III-3A, III-7.B and III-7.D)

3.12.4 CPCo Letter dated March 16, 1982, SEP Topic III-4.A, Tornado Missiles (Response to Request for Information)

3.12.5 CPCo Letter dated June 16, 1982, SEP Topic III-4.A, Tornado Missiles (Response to Request for Information)

3.12.6 NRC Letter dated November 29, 1982, SEP Topic III-4.A, Tornado Missiles (NRC Safety Evaluation Report)

3.12.7 CPCo Letter dated December 14, 1981, SEP Topic III-4.D, Site Proximity Missiles (Including Aircraft), (CPCo Evaluation)

3.12.8 NRC Letter dated August 12, 1982, SEP Topic III-4.D, Site Proximity Missiles (including Aircraft) (NRC Safety Evaluation Report)

3.12.9 NUREG-0586, Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities, August 1988

3.12.10 NRC Letter dated September 30, 1982, SEP Topic III-7.B, Design Codes, Design Criteria and Load Combinations (Draft Safety Evaluation Report)

3.12.11 NRC Letter dated March 5, 1980, NUREG-0569, Evaluation of the Integrity of SEP Reactor Vessels, SEP Topic V-6

4.0 REACTOR

Previously, this chapter described the design requirements of the reactor, the fuel as utilized in the reactor and the systems used to control nuclear reactivity when fuel was in the reactor.
Paragraph 2.C.1 of the Big Rock Point License (DPR-6) prohibits fuel from being placed in the reactor vessel. The reactor vessel and all associated support systems have been dismantled and removed.
The fuel system design (as required for reactor operation), nuclear design, thermal hydraulic design, operation with less than all loops, reactivity control systems, control rod drive systems and liquid poison systems are no longer applicable. Therefore this chapter has been deleted. Chapter 9 of this UFHSR discusses the fuel relative to storage on the Independent Spent Fuel Storage Installation.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Previously, this chapter identified the main components of the reactor coolant system and connected support systems. These systems have been dismantled and the fuel has been transferred to dry fuel storage at the Independent Spent Fuel Storage Installation. Therefore, this chapter has been deleted.

6.0 ENGINEERED SAFETY FEATURES (ESF)

6.1 ENGINEERED SAFETY FEATURES (ESF) SYSTEMS DEFINED

Engineered Safety Features (ESF) are those systems which are required to function to mitigate the consequences of a postulated design basis accident (previously Reference 6.2-1). For the permanently defueled plant with all spent nuclear fuel stored on the Independent Spent Fuel Storage Installation (ISFSI), the only remaining credible accident is the non-fuel accident discussed in Chapter 15 of this Updated Final Hazards Summary Report (UFHSR). Releases from this accident are well within 10 CFR 100 release criteria. Therefore, the discussion on ESFs is no longer applicable and this chapter has been deleted in its entirety.

7.0 INSTRUMENT AND CONTROLS (I&C)

This chapter formerly discussed plant safety and monitoring systems that included the reactor safety system, control rod drive system, reactor and steam drum temperature and pressure monitoring, reactor circulation monitoring, reactor clean-up system monitoring, stack gas monitoring, process liquid monitor systems, and area monitoring system. For discussion of Process Radiation Monitoring Systems, refer to Chapter 11 of the UFHSR.

With all spent fuel removed from the Spent Fuel pool, no instrumentation and control systems that provide safety-related or important-to-safety functions remain. Therefore, this chapter has been deleted.

8.0 ELECTRIC POWER

8.1 INTRODUCTION

8.1.1 OFF-SITE POWER SYSTEMS

For Independent Spent Fuel Storage Installation (ISFSI), a 7.2/12.47kV transmission line from the Consumers Energy System is utilized as a power source. The 7.2/12.47kV line is fed from McNally Substation (MA-02-0020).

With all spent fuel permanently stored on the ISFSI, there are no safety-related direct current powered circuits remaining.

8.1.2 FUNCTIONAL DESIGN DESCRIPTION

This section formerly described the safety-related function of the off-site power system. With all spent fuel permanently stored at the ISFSI, this section is for historical information only and remains to describe power sources during dismantlement of the former plant industrial area.

8.1.2.1 ISFSI Power

The ISFSI site uses a separate 7.2/12.47kV source of off-site power from the McNally substation. This off-site power source was installed in October 2002 by MA-02-0020, and extended the 7.2/12.47kV transmission line from the McNally substation paralleling US 31 and under building the existing 46kV transmission line to the plant. A pole-mounted 150kVA transformer bank attached to the 7.2/12.47kV line supplies the necessary 480/277V power to the ISFSI.

8.1.3 OFF-SITE POWER FREQUENCY DECAY

The evaluation of frequency decay addressed the effect of recirculating water pump operation on core performance and is no longer applicable.

8.1.4 DISTRIBUTION SYSTEM VOLTAGES AND DEGRADED GRID PROTECTION

This section previously described the adequacy of the off-site and on-site electrical distribution system to provide sufficient quality electrical power to allow operation of safety-related electrical loads. With all spent nuclear fuel permanently in dry storage on the ISFSI, there are no safety related electrical loads. Therefore the discussion of the electrical distribution system analyses performed is no longer applicable and has been deleted.

8.2 ON-SITE AC POWER SYSTEM

8.2.1 FUNCTIONAL DESIGN DESCRIPTION

The Decommissioning Power System (DPS) has been removed and is no longer part of the plant design basis.

8.2.2 MAIN DIESEL GENERATOR

Minor Alteration MA-03-003 removed the Main Diesel Generator.

8.2.3 STANDBY DIESEL GENERATOR

Minor Alteration MA-03-0003 removed the standby diesel generator.

8.3 ON-SITE DC POWER SYSTEMS

This section formerly discussed the station battery system, the alternate shutdown battery system, reactor depressurization system uninterruptible power supplies, and diesel starting systems. With the spent nuclear fuel permanently stored at the ISFSI, these systems have been removed.

8.4 ELECTRICAL PENETRATIONS

With all spent nuclear fuel permanently stored at the ISFSI, containment closure is not credited for design basis accidents. The discussion on design and function of electrical penetrations in this section is no longer applicable and has been deleted.

9.0 AUXILIARY SYSTEMS

9.1 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

Spent Nuclear Fuel (SNF) was stored in the Big Rock Point (BRP) Spent Fuel Pool until March 26, 2003; since that date, all SNF has been stored in an Nuclear Regulatory Commission (NRC)-approved FuelSolutions™ Storage System, which is located at the Independent Spent Fuel Storage Installation (ISFSI) shown in Figure 2-3. SNF will be stored at the ISFSI until it can be shipped off-site to an NRC-approved repository.

Only BRP nuclear fuel can be stored in the spent fuel Storage Casks at the ISFSI. The BRP Fuel Assembly Database (Reference 9.5.1) uniquely identifies the 441 fuel assemblies that were contained in the BRP Spent Fuel Pool. The components of the database, including reference documents, establish that all fuel-bearing components were previously approved for use in the BRP reactor. Additionally, Special Nuclear Material (SNM) accountability records identify that no nuclear fuel other than that associated with BRP reactor operations is part of the fuel inventory.

On December 18, 2001, BRP issued a notification of intent to store fuel under provisions of 10 CFR 72, Subpart K and Certificate of Conformance (CoC) 72-1026 (Reference 9.5.2).

Registration of Casks occurred as they were put in use. The first Cask was registered on November 18, 2002 (Reference 9.5.3) with the last of the seven fuel Storage Casks loaded and registered on March 26, 2003 (Reference 9.5.4).

Big Rock Point stores SNF on the ISFSI as a general licensee under 10 CFR 72. The general license for the storage of spent fuel in a Cask fabricated under a CoC issued pursuant to 10 CFR 72 terminates 20 years after the date that the particular Cask is first used by the General Licensee to store spent fuel. Cask design must be re-approved within 20 years of November 19, 2002.

In addition to SNF, all on-site Greater-Than-Class-C (GTCC) material is stored at the ISFSI.

On May 5, 2003, Big Rock Point requested registration as a user of the FuelSolutions™ TS-124 package, CoC Number 71-9276 (Reference 9.5.40).

9.1.1 FUEL ASSEMBLY HISTORY

Big Rock Point developed a fuel assembly database published as Engineering Analysis, EA-CR-BRP-99-229-01 (Reference 9.5.1) documenting the characteristics of each of the 441 fuel assemblies stored in the FuelSolutions™ W150 Casks. All 441 fuel assemblies are qualified for storage, as demonstrated by the comparison between the database information and the BNFL FuelSolutions™ (BFS) Technical Specification.

The fuel assembly database was subsequently updated by EA-FC-703-Fuel Data-03 (Reference 9.5.6) to add assembly-specific information on active fuel region cobalt content, volume displacement, and designation as damaged assemblies. Minor additional changes were provided to clarify assembly information resulting from visual inspections.

Following shutdown of the plant in August 1997, off-load of the reactor core fuel assemblies, and fuel consolidation (References 9.5.7 and 9.5.39), each of the fuel assemblies were visually inspected with a video record made of fuel assembly condition. This record, along with a review of plant and reactor engineering records, was summarized in Reference 9.6.8 that identified 39 fuel assemblies designated as damaged (as opposed to intact, partial, or MOX) for the purposes of loading the FuelSolutions™ W150 Cask System and storing assemblies in damaged fuel cans. An additional seven assemblies were designated to be stored in damaged fuel cans based on grid spacers being damaged to a degree where fuel rod structural integrity could not be assured, or where grid spacers had shifted vertically from their design position (Reference 9.5.9). Two more assemblies with one failed fuel rod each, I612 and I713, were reclassified as damaged based on recent regulatory guidance on classification of fuel as damaged. Two assemblies, F056 and G210, were re-classified as damaged based on visual inspections conducted during loading of Canisters 5 and 3, respectively. The total number of assemblies designated as damaged at the completion of fuel loading was 50.

The W74 Canister loading specifications provide for dummy fuel assemblies with the approximate weight and size of an actual fuel assembly that were loaded into each empty W74 Canister basket guide tube. The design of the dummy assemblies is given in BFS Specification CPC-117 (Reference 9.5.10) and calculation package CPC-216 (Reference 9.5.11); Consumers Energy obtained fabrication of the dummy fuel assemblies to meet the specifications, which is documented as part of Facility Change FC-703.

The BRP loading analysis, EA-FC-703-FUEL DATA-04 (Reference 9.5.12), specified the position of each of 63 fuel assemblies and 1 dummy fuel assembly in each W74 Canister to be loaded. The goals of the loading analysis were to: 1) distribute assemblies between the loaded Casks in such a way as to have an evenly distributed heat load, 2) load higher dose fuel toward the center of the Canister baskets and lower dose fuel toward the outside to limit personnel exposure and minimize Cask external dose rates when loaded onto the ISFSI pad, and 3) ensure that the heat load of each Canister does not exceed the 24.8 kilowatt limit documented in the W74 Canister SAR. The loading of the proper fuel assemblies was verified and documented for each Canister as part of the Cask loading procedures.

Table 2.1-7 (Reference 9.5.5) also specifies a limit of ≤2.9 grams of cobalt content in the active fuel region if the cooling table W74-1-A shown in Table 2.1-9 (Reference 9.5.5) is to be used. Table 2.1-8 (Reference 9.5.7) limits the active fuel region cobalt content to ≤2.9 grams for all MOX fuel assemblies. Cooling Table W74-1-B (Table 2.1-10, Reference 9.5-5) was developed by BFS to address those fuel assemblies with cobalt contents above 2.9 grams but less than 15 grams. A review of the FuelSolutions™ calculation (CMPC.1503.009, Reference 13) for cobalt content in BRP fuel found that generic industry data for cobalt content in hardware materials was used along with a standard spacer design in terms of weight and material composition. For a number of BRP fuel types, the design and /or material composition is different than that assumed in CMPC.1503.009. To address this difference, EA-FC-703-Fuel Data-01 (Reference 9.5.14) was generated to provide for specific cobalt calculations in the active fuel region for each of the BRP fuel types. Data used for these calculations was taken from fuel vendor design reports referenced in the BRP Fuel Database EA-CR-BRP-99-229-01, fuel records found in the site records and Consumer's company record vaults, as well as direct inquiries to fuel vendors. Where specific fuel type information was not available or no records could be located, the data provided in CMPC.1503.009 was utilized. The result of this EA showed that the maximum cobalt content in any of the BRP fuel types was equal to or less than the limit of 2.9 grams.

A question was raised concerning the transportability off-site of fuel assemblies in a horizontal configuration with unrestrained fuel pins while in a Storage Canister. The unrestrained fuel pins resulted from a program conducted following plant shutdown to consolidate individual fuel pins and partial fuel pin segments resulting from fuel assembly demonstration programs into empty corner positions of F, F-MOD, G-U and G-Pu assemblies. This issue was documented in BRP Condition Report C-BRP-02-0017 (Reference 9.5.15). FuelSolutions™ was asked to provide a buckling evaluation of the fuel pins for the hypothetical accident condition (HAC) of end drop under off-site transportation conditions defined in 10 CFR 71 (Reference 9.5.16). The evaluation demonstrated that the cladding will maintain its integrity and there is no requirement to restrain the fuel rods or store the assemblies containing them in damaged fuel cans.

9.1.1.1 FuelSolutions™W74 Canister Helium Backfill Density

After the draining of water from the Canister and completion of the vacuum drying process, the Canister was backfilled with helium to conduct the pressure and leak check of the Canister. The Canister was then evacuated and the specified quantity of helium was placed into the Canister and the Canister was sealed. The helium used for backfill had a minimum purity of 99.995%.

An Engineering Analysis, EA-FC-703-Fuel Data-02 (Reference 9.5.17), was completed to calculate the volume displacement for each fuel type to be stored in the FuelSolutions™ Cask system at BRP. An evaluation of the results of the Engineering Analysis was conducted by FuelSolutions™ (Reference 9.6.18) to ensure that existing calculations for storage used in the licensing of the Cask system are conservative and that no revisions are warranted for the pressure and thermal capabilities of the W74 Canister. The results of this analysis, along with the information in the fuel assembly database (Reference 9.5.1), was used to develop a volume displacement for each fuel assembly which accounted for removed and added fuel rods and solid zircaloy rods (Reference 9.5.6). The free volume for each loaded Canister was determined in the BRP loading analysis (Reference 9.5.12), along with the required quantity of helium to meet the Technical Specification requirement.

9.1.2 DRY FUEL STORAGE SYSTEM

The FuelSolutions™ Storage System, certified under CoC 72-1026 by the NRC, will be used for storage of SNF at BRP. The W74T class canister will be used for storage. Documentation of the intention to use the FuelSolutions™ system (formerly known as Wesflex System Spent Fuel Management System) was provided to the NRC (Reference 9.5.19.)

a. FuelSolutions™ Storage System

The FuelSolutions™ Storage System consists of the following components:

1. W74 Canister for dry storage of SNFs;
2. W100 Transfer Cask for canister loading, closure, and handling capability; and
3. W150 Storage Cask that provides passive, vertical dry storage of a loaded canister.

The W74 canister accommodates 64 boiling water reactor (BWR) assemblies.

b. Description

The FuelSolutions™ Storage System is certified as described in the Safety Analysis Reports (References 9.5.5 and 9.5.20).

9.1.3 ISFSI STORAGE FACILITY AND EQUIPMENT

Facility Change FC-704, ISFSI Storage Facilities and Equipment (Reference 9.5.21), established an interim storage facility for the 441 BRP spent fuel assemblies and other selected radioactive material/waste. The storage pad was sized to accommodate seven FuelSolutions™ W150 Storage Casks containing W74 canisters, and one W150 Storage Cask containing reactor-related Greater-Than-Class-C (GTCC) materials. The minimum center-to-center spacing of the W150 Fuel Casks was procedurally controlled to ensure they are spaced 15 feet center-to-center.

The pad is designed to meet the requirements of 10 CFR 72, design specifications, and other regulatory requirements, codes, and standards as appropriate. The quality categories of the ISFSI, the roadway adjacent to the pad, and associated components are described in the Facility Change.

The ISFSI is a gated complex capable of providing a secure and maintainable storage environment for the concrete Storage Casks. The ISFSI design includes a fenced concrete pad and structures/buildings needed to accommodate security, surveillance and periodic maintenance. Instrumentation and monitoring equipment is installed to provide for the surveillance of the loaded Casks as well as the requisite level of security. The ISFSI allows for secure interim storage of loaded Casks.

The ISFSI, shown on Figure 2.3, is sited approximately 350 yards south of the former operating plant protected area near an abandoned railroad grade. The ISFSI complex is located in an Unnamed Drainage Basin described in Section 2.4.3.1.2 of this UFHSR. This siting required re-evaluation of the site hydrology study (Reference 9.5.22), which has been addressed in the modification design output.

Facility and equipment design accounts for off-site dose design limits at the ISFSI boundary as required by 10 CFR 20, 10 CFR 72.104(a), 10 CFR 72.106, and Regulatory Position 2 of Regulatory Guide 8.8, as set forth in Chapter 10, Radiation Protection, of the WSNF-220 FSAR. Although expected off-site dose levels do not require additional shielding, earthen berms were constructed primarily for aesthetic purposes and secondarily to provide a means of utilizing excavation spoils. These berms will provide additional shielding, although no credit is taken for their presence in the off-site dose analysis.

9.1.3.1 ISFSI Pad Design

9.1.3.1.1 Reinforced Concrete Pad and Subgrade Soil Characteristics

A reinforced concrete pad was designed as a supporting foundation for the Storage Casks (Reference 9.5.21). The concrete pad thickness, reinforcing steel size and spacing, concrete compressive strength and pad subgrade characteristics ensure the design basis deceleration loads for the Storage Cask and canister are not exceeded utilizing an analysis with the methodology described in Sections 3.7.3.1 and 3.7.5.1 of the WSNF-220 FSAR. This analysis is provided in Reference 9.5.23 and is used as a design input in Reference 9.5.24, and is specifically discussed below.

The pad also includes embedded plates for the upender/downender, for downending the Storage Cask to accommodate horizontal transfer of the canister to an approved transportation Cask or the W100 Transfer Cask. The pad meets the structural requirements of 10CFR 72 for the normal, off-normal and accident loads and load combinations as specified in NUREG-1536 and supplemented by NUREG-1567. ACI 349 was used for the design of the pad, as specified in NUREG-1536. Per Reference 9.5.22, the construction requirements of ACI 318 were invoked through a combination of specific drawings or specification provisions, and through the direction for conformance to ACI 301, Specifications for Structural Concrete for Buildings, by the construction contractor. This manner of complying with the construction requirements of ACI 318 is consistent with recommended industry practice. The concrete slab supporting the FuelSolutions™ impact limiter has been designed to withstand the static force of gravity on the impact limiter.

Horizontal transfer of the W74 Canister is performed over the impact limiter. The impact limiter is a set of four crushable blocks filled with foam.

A finite element model of the concrete pad and the soil subgrade below it were analyzed to determine the pad moments, shears, settlements and the subgrade bearing pressure for applicable loads and Cask loading patterns (Reference 9.5.24). The effect of soil-structure interaction (SSI) was included in determining the seismic accelerations (horizontal and vertical directions) for the pad and the Casks. The allowable bearing capacity, settlement, and other geotechnical parameters of the soil underneath the pad was obtained from detailed geotechnical investigations, including seismic crosshole testing, conducted during 1998 and 1999, as identified in Reference 9.5.23. The control motion of the seismic input motion is in accordance with Big Rock Point response spectra, which correspond to Regulatory Guide 1.60 response spectra with peak zero period horizontal ground acceleration of 0.12g (Design Basis Earthquake, DBE).

The subgrade soil properties at the ISFSI pad are as follows (see Section 2.5.1 of Reference 9.5.24):

Average allowable static soil bearing capacity:	4500 psf (net)
Average elastic modulus:	519 to 4858 psi (natural soil) 1610 to 3750 psi (granular fill)
Variability:	Per boring logs, the existing soil under the pad does not show extreme variation in its characteristics.
Shear Wave Velocity:	1810 ft/sec to 2490 ft/sec
Liquefaction Potential:	None

When compared with the physical soil properties specified in Section 2.6.3.5.1 of the FuelSolutions™ Storage System FSAR, WSNF-220, the subgrade soil properties at the BRP ISFSI pad site meet all the WSNF-220 site characterization parameters.

Appendix A to CoC 1026, Section 4.2.2.1, requires that any site specific design with parameters that differ from those listed must be evaluated to confirm that the design basis deceleration loads for the Storage Cask and canister are not exceeded. The evaluation must be performed using the same methodology as described in WSNF-220 FSAR Sections 3.7.3.1 and 3.7.5.1.

The design parameters for the BRP ISFSI concrete pad are as follows:

Design Parameters	**Areas in which Transfer or Storage Cask Drop is Credible**	**Cask Storage Areas in which Drop is not Credible**
Concrete Thickness (inches)	24	36
Reinforcing Steel (EWEF)	#9 @ 12" or .345%	#9 @ 12" or .23%
Concrete 28 day compressive strength (psi)	3000 to 4500	3000 to 4500
Nominal Reinforcing yield strength (psi)	60,000	60,000
Soil Effective Modulus of elasticity (psi) Natural Soil[1] Granular Fill[2]	 519 to 4858 1610 to 3750	 519 to 4858 1610 to 3750
Drop Height (inches)	36	36

BFS Calculation CMPC.1504.005, Revision 2 (Reference 9.5.23), provided the confirmation that the design basis deceleration loads for the Storage Cask and canister are not exceeded.

The ISFSI pad was placed on June 30, 2001. Eleven sets of compressive test cylinders were cast during placement. Concrete compressive strength, based on standard cured 28-day break tests, averaged 4040 psi. The range in average 28-day strength was from a low of 3500 psi to a high of 4740 psi.

The concrete met the specification requirement for strength on the following basis:

1. For all sets the average 28-day strength is greater than 3000 psi;

   AND

2. For all sets, the average of three consecutive 28-day strengths is less than 4500 psi;

   AND

[1] SME Geotechnical Evaluation Report dated September 29, 1998. Tests were conducted at conditions judged to be representative of those at the depth of the borings.
[2] MTC Triaxial Test Reports dated April 17, 2001. Tests were conducted at conditions based on granular fill placement requirements.

3. No individual 28-day strength exceeds 5000 psi.

The calculation considered several parameter sets. The parameters that bound the BRP ISFSI are as follows:

Design Parameters	Transfer Cask	Storage Cask	
Concrete Thickness (inches, max)	24	24	36
Maximum Reinforcing Steel (EWEF)	.35%	.40%	.28%
Concrete Compressive Strength (psi)	6000	6000	6000
Nominal Reinforcing Yield Strength (psi)	60,000	60,000	60,000
Soil Effective Modulus of Elasticity (psi, max)	10,000	10,000	10,000
Drop Height (inches, max.)	36	36	36

Two values for concrete thickness are considered because the Storage Cask can be either on the 36-inch part of the slab or on the 24-inch part of the slab. The Storage Cask will be lifted and placed on the upender/downender J-skid on the 24-inch portion of the slab. No lifts will be performed on the 36-inch portion of the slab; therefore a drop accident is not credible on this portion of the slab. Since the FuelSolutions™ Technical Specifications and FSAR do not distinguish between the portion of the slab where the drop accident is credible and the portion of the slab where the drop accident is not credible, compliance with the design is shown for both portions.

The above shows that the BRP ISFSI slab design limits the Cask drop deceleration values to within acceptable limits. By letter dated October 4, 2000 (Reference 9.5.25), BFS concurred that the results of Calculation CMPC.1504.005, Revision 2 (Reference 9.5.23), bound the design of the BRP ISFSI Pad and are in accordance with the methodology set forth in the WSNF-220 FSAR.

FuelSolutions™ Specification CPC-105 provided the design for the Big Rock Point ISFSI. This design establishes the acceptability of a 1-foot 9-inch distance from the W150 Storage Casks to the edge of the ISFSI pad. WSNF-220 FSAR Section 3.7.9.1.2 addresses sliding of the Storage Cask following a tornado missile impact. The Cask slides 32.9 inches based on conservative design parameters of weight and friction factor, which implies that the Cask could overhang the edge of the ISFSI pad by one foot upon impact of a design basis tornado missile. In addition, WSNF-220 FSAR Figure 1.4-1 shows a required minimum dimension of three-foot spacing from the edge of the pad to the Cask. A review of the tornado missile analysis was conducted using Big Rock Point site-specific Cask parameters and tornado missile characteristics, and it was found that the 1-foot 9-inch distance was acceptable. A 72.48 evaluation determined that the minimum distance in Figure 1.4-1 could be revised to 1-foot 9-inch without requiring prior NRC review and approval (Reference 9.5.26).

9.1.3.1.2 Slope Stability

The ISFSI site required excavation of a small hill, with the requirement to establish stable slopes adjacent to the facility. A slope stability analysis under seismic DBE loading was performed to confirm the adequacy of the slope configuration on the east and south sides (Reference 9.5.27) adjacent to the ISFSI pad. The analysis concluded that the design of the slopes, with a 2.5 horizontal to one vertical slope and 8.0-foot intermediate bench, at a maximum of 20 feet above the start of the slope, provided adequate stability with acceptable factors of safety. The analysis was revised to address the as-built berm, with a slope of approximately 2.8 horizontal to one vertical and no intermediate bench; stability for this configuration was also confirmed. The analysis also concluded that if the slope did fail and collapse toward the Cask storage pad, the debris would not reach the top of the pad and, thus, Cask integrity would not be endangered.

The slope of the berm on the west side of the pad generally meets the criteria applied to the berms adjacent to the pad as a matter of good practice, but is not a concern from the standpoint of pad safety. No berm has been designed for the north side of the pad due to interface with existing wetlands.

9.1.3.1.3 Grading, Drainage and Roadwork

As described above, ISFSI earthwork and grading involved cutting back a portion of a small hill adjacent to an abandoned railroad bed to establish a level area for the facility. Drainage ditches, catch basins and drainage pipes were designed based on the hydrological study of Reference 9.5.22, with the result that the ISFSI pad will not be undercut by floodwaters from the Probable Maximum Flood event, thereby maintaining the design requirements described above.

9.1.3.1.4 Roadway Adjacent to the Pad

The design of the roadway inside the security fence in the vicinity of the ISFSI pad has been considered (Reference 9.5.28). The design considered the loading of the Heavy Haul Trailer with a loaded Storage Cask, transporting loaded Casks from the plant to the pad, and transits resulting from shipment to an off-site permanent storage facility. The roadway conforms to the requirements specified in Reference 9.5.28.

9.1.3.1.5 Lightning Protection

Evaluation of the need for lightning protection has been performed in accordance with Section 2.3.4.7 of the WSNF-220 FuelSolutions™ Storage System FSAR. An alternate design to the one identified in WSNF-220 was chosen due to the difficulty in implementing the optional Cask grounding system described in the FSAR.

The lightning protection system employed for the BRP ISFSI facility consists of five lightning protection masts designed to protect Storage Casks at adjacent locations using a 100-foot radius zone-of-protection rolling sphere model. Each mast is 52 feet in height, including a four-foot blunt-tipped air terminal (strike receptor) attached to the top for the purpose of creating a preferred and optimally designed point of attachment for a lightning strike. This mast height ensures that the dimensional sphere of the rolling ball model does not intercept the Cask structures at the specified locations of the masts adjacent to the ISFSI pad.

The lightning protection system conforms to the requirements of NFPA-780, Lightning Protection Code, UL 96, Lightning Protection Components, and UL 96A, Installation Requirements for Lightning Protection System.

9.1.3.1.6 ISFSI Power Supply

Independent Spent Fuel Storage Installation power supply is discussed in Chapter 8 of this UFHSR.

9.1.3.1.7 W150 Storage Cask Thermocouple

The temperature of a W150 Storage Cask with a W74 *canister containing* fuel assemblies meets the WSNF-223 (Reference 9.5.5) Technical Specification LCO 3.3.2. Temperatures from the thermocouples are procedurally verified to be within limits on the frequency given in the surveillance requirements. The normal and off-normal temperature limits are established based on a Canister heat load of 24.8kW; the heat load for the Canister containing the highest thermal loading of SFAs is calculated to be less than 6kW, as shown in the BRP loading analysis (Reference 9.5.12).

9.1.3.1.8 W150 Storage Cask Air Inlet and Outlet Openings

Procedures are in effect to inspect the inlet and outlet vents of the loaded Storage Cask at the ISFSI in accordance with the frequency specified in the surveillance requirement. Tools are available to clear the vent screens and vents of snow, insect nests, etc., that may decrease or block air flow through the vents. Damaged vent screens will be repaired or replaced as necessary.

9.1.4 ISFSI SECURITY

The BRP Plant established a new protected area (PA) on the ISFSI. All SNF, SNM, and Greater Than Class C (GTCC) waste are stored at the ISFSI. Therefore, all BRP spent fuel is stored within a PA and meets the requirements of 10 CFR 73.55 with approved exemptions as set forth in the approved BRP ISFSI Security Plan (Reference 9.5.30).

The NRC, in an August 21, 2000, Federal Register Notice (65 FR 50606) clarified portions of 10CFR 72, stating that the requirements of §72.106(b) apply to ISFSIs with either general or specific licenses. In order to provide protection against a loss of control of the facility that could be sufficient to cause radiation exposure exceeding the dose limits described in §72.106, BRP has established a boundary of the owner-controlled area at a minimum of 300 meters from the boundary of the dry Cask storage ISFSI facility. This distance provides assurance that the Design Base Threat (DBT) of radiological sabotage, based on unimpeded access of the DBT but no protracted loss of control of the facility, would result in a dose that would be well below the §72.106(b) limits.

The security system design is addressed in Facility Change FC-704B (Reference 9.5.31). References 9.5.32 and 9.5.33 provide calculations pertinent to ISFSI equipment described below. The following information summarizes the key aspects of the system employed for the BRP ISFSI. Further detail on security equipment may be found in the BRP ISFSI Security Plan (Reference 9.5.30).

a. Perimeter Intrusion and Detection System (PIDS)

   The requirements of the system are based on guidance from NRC Regulatory Guide 5.44, Perimeter Intrusion Alarm Systems, and 10CFR 72 and 73, as applied to ISFSI sites. The system utilizes a microwave intrusion detection system. Overlapping PIDS zones surround the ISFSI perimeter.

b. Closed-Circuit TV System (CCTV)

   The CCTV system includes fixed and Pan, Tilt and Zoom (PTZ) cameras. The system is designed and installed in accordance with guidance from Regulatory Guide 5.44, 10CFR 72 and 10 CFR 73.

   Camera towers and the associated mounting hardware are equipped with GFI protected receptacles connected to utility power.

c. Illumination System

The requirements for the illumination system are based on 10CFR 73, 10 CFR 8.5 (Interpretation by the General Counsel of 10 CFR 73.55 Illumination and Physical Search Requirements), NUREG-0908 (Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans) and the current BRP plant security lightning commitments. Big Rock Point will maintain lighting of sufficient illumination in the ISFSI protected areas and isolation zones for observation by CCTV or a security watchman on patrol. Further, during periods of alarm assessment or conduct of watchman patrols, lighting in the area of concern will be sufficient for evaluation. To provide the ISFSI site with ease of maneuverability, the lighting system is installed near the nuisance fence.

d. Perimeter Fence

The ISFSI fence is configured in accordance with normal industry practice. A double fence surrounds the ISFSI site. The outer (nuisance) fence and inner (security) fence are the same height and include outriggers and three lines of barbed wire facing outward, the same as the existing security fences at the plant. Both fences utilize a galvanized chain link fabric. The provision for vehicle and personnel entrances is provided through the use of manual gates. The gates are padlocked.

e. Owner-controlled Fence

An ISFSI owner-controlled fence is not required at this time. Following completion of decommissioning of the BRP Nuclear Plant, an owner-controlled fence may be installed, consistent with dose rate requirements and other site considerations.

f. Power Distribution System

The power distribution system is designed in accordance with applicable Consumers Energy standards and the National Electric Code (NEC). The main power distribution transformer is designed and installed outside the ISFSI by Consumers Energy. The secondary side of the main power transformer is used to feed the 480-volt AC distribution panel and the 480-208/120 volt AC step-down transformer, which provides power to a 208/120-volt distribution panel. All distribution panels and UPS batteries are located inside the electrical enclosure building, which is sited within the facility security perimeter. This distribution system is utilized for GFI receptacles at camera towers.

g. Grounding System

The grounding system is based on the NEC requirements and the applicable Consumers Energy established engineering practices. The grounding system interconnects all related equipment and structures, such as microwave equipment and microwave mounting poles, camera and camera towers, security lighting, and ISFSI security fences.

9.1.5 HORIZONTAL TRANSFER SYSTEM

The Horizontal Transfer System (HTS) was installed and tested in Facility Change (FC) -710. FC-710 designed, constructed, fabricated, and procured facilities and equipment to implement the FuelSolutions™ HTS. The HTS enables horizontal transfer of a loaded W74 fuel canister between a W150 Storage Cask, a W100 Transfer Cask, and a W125 Transportation Cask. The equipment included the ISFSI impact limiter, the J-skid for upending and downending a W150 Storage Cask, the 4-point gantry crane and rail system for upending and downending the J-skid, the transfer skid and rail system for transfer of a W74 Canister to and/or from a W150 Storage Cask to a W100 Transfer Cask, the hydraulic ram system for effecting the horizontal transfer operation, the horizontal transfer trailer for on-site transfer skid movement, and required support systems (power, water, and heat) to enable operations.

This modification provided a means of storing a loaded fuel canister temporarily in a Transfer Cask while the Storage Cask is repaired or replaced. It also provides a means for off-site transportation of a loaded fuel canister to a Department of Energy (DOE) storage facility.

During transfer operations, the ram force on the W74 Canister during horizontal transfer shall not exceed 70,000 pounds pushing or 50,000 pounds pulling. Operating procedures have been developed for horizontal transfer operations and incorporate the necessary controls to meet these limits on hydraulic ram force.

The applicable Cask handling and loading procedure specifies the sequence of operations such that the time between draining of the Annulus water and lowering the Canister from the Transfer Cask into the Storage Cask is minimized. The analysis contained in WSNF-220 (Reference 9.5.20) for this evolution, assuming design heat loads, results in a time limit of 15 hours before limiting Canister component temperatures are reached. The eight-hour time limit for completion of the transfer evolution is reflected in the procedure.

An analysis was performed by FuelSolutions™ (Reference 9.5.34) to determine the Canister heat load that would allow an unlimited time between the beginning of transfer operations and filling of the Annulus. This heat load value was determined to be 16.87 kW. The Canister loading analysis, reported in Reference 9.5.1, determined that the heat load in each of the seven loaded Canisters would be about 5kW. Thus, the unloading procedure does not require the specification of a time limit between the time the Canister leaves the Storage Cask and the time that the Annulus is refilled with water.

9.2 WATER SYSTEMS

This section formerly discussed the various plant water systems. With the removal of all spent fuel from wet storage in the spent fuel pool on March 26, 2003, the descriptions of design criteria and functions are no longer applicable, and were deleted. No water systems are required for the safe storage of spent fuel on the ISFSI.

9.3 PROCESS AUXILIARIES

This section formerly discussed the various plant auxiliary systems. Chapter 11 of this UFHSR contains a discussion of radioactive waste management for dismantlement activities. During building removal and foundation excavation, collected ground water will be monitored prior to discharge. All discharges will be made in accordance with requirements of the 10 CFR Part 20 and State and local regulations.

9.4 OTHER AUXILIARY SYSTEMS

9.4.1 FIRE PROTECTION SYSTEM (FPS) GENERAL

With the removal of fuel from the spent fuel pool into dry cask storage the requirement to maintain an installed FPS was eliminated. Therefore the discussion of the FPS details was deleted.

Details of fire protection features and analyses may be obtained by review of BRP Volume 26, Fire Protection Summary.

9.4.1.1 Administrative Fire Limits – ISFSI

Limits on the amount of combustible fuel in the proximity of the Storage Cask or Transfer Cask are in accordance with the thermal analysis for the postulated fire accident condition. Storage System FSAR, Paragraphs 4.6.1.5.2 and 4.6.2.6 (Reference 9.5.20), state the limit of 70 gallons of combustible fuel to limit the effects of engulfing fire conditions. Procedure controls limit the amount of combustible fuel in any combination of vehicles that could interact with the Casks upon a spill or fuel tank rupture (i.e., heavy haul tractor and trailer and prime mover) to less than 70 gallons. Any other vehicles, such as light motor vehicles, air compressors, generators, etc., will be located such that their fuel would not interact with the Casks in the event of a postulated spill or rupture.

Evaluations of specific explosion hazards and fire hazards and the potential impact of these hazards on the loaded Storage Casks during transportation to the ISFSI and storage at the ISFSI have been performed (References 9.5.35, 9.5.36, and 9.5.37). The results are summarized as follows:

a. An explosion resulting from diesel fuel or fuel oil tanks was considered. These fuels have low vapor pressures and high flash points. Since the fuels will be at atmospheric pressure and the ambient air will be substantially lower than the flash point, a flammable gaseous mixture and hence a vapor cloud explosion of a tank mixture in the absence of a major external heat source is not credible.

b. No hydrogen is stored onsite and nitrogen and propane are not stored in the vicinity of the ISFSI. Thus, fires or explosions from these sources were not considered.

c. Administrative controls are applied to minimize the fire hazard resulting from gasoline-fueled vehicles or propane delivery vehicles. Unattended parked vehicles will be limited in their proximity to the ISFSI.

Additional administrative controls have been applied to operations of diesel-powered vehicles, small gasoline-powered equipment (i.e., snow blowers), and hydraulic equipment associated with the Horizontal Transfer System with the loaded Storage Casks located on the ISFSI.

An assessment of the impact of a wildfire in the forested area surrounding the ISFSI pad on the berms and the loaded Storage Casks was conducted (see Reference 9.5.38). The assessment concluded that a consuming wildfire would have no impact on the capability of the Casks to maintain their integrity and perform their designed radiological and confinement functions.

9.5 REFERENCES

9.5.1 EA-CR-BRP-99-229-01, Revision 0, Big Rock Point Fuel Assembly Database (Big Rock Point Microfilm Cartridge/Frame 4641/0226)

9.5.2 Letter from GCWithrow to U.S. Nuclear Regulatory Commission, Docket 50-155, License DPR06, Big Rock Point Plant – Notification of Storage of Spent Nuclear Fuel, December 18, 2001

9.5.3 Letter from KMHaas to U.S. Nuclear Regulatory Commission, Dockets 50-155 and 72-043, License DPR-6, Big Rock Point Plant – Registration of Use of Cask to Store Spent Fuel, November 19, 2002

9.5.4 Letter from KMHaas to U.S. Nuclear Regulatory Commission, Dockets 50-155 and 72-043, License DPR-6, Big Rock Point Plant – Registration of Use of Cask to Store Spent Fuel, March 26, 2003

9.5.5 FuelSolutions™ W74 Canister Final Safety Analysis Report, Document Number WSNF-223, Docket Number 72-1026

9.5.6 EA-FC-703-Fuel Data-03, Revision 0, Addition of Fuel Assembly Volume Displacements, Fuel Region Cobalt Contents, and Damaged Fuel Assembly Classifications to EA-CR-BRP-99-229-01, Attachment L (Big Rock Point Microfilm Cartridge/Frame 4641/0196)

9.5.7 EMF-1998(P), Revision 2, Consolidation of Stored Fuel Rods at Big Rock Point, October 1997

9.5.8 K.P. Powers, Consumers Energy to K.A. Hoedeman, Westinghouse, June 2, 1999

9.5.9 Letter, R. Quinn (FuelSolutions™) to Director NMSS (NRC), Storage License Application for the FuelSolutions™ System (TAC Number L22993) – Response to NRC RAI for W74 Supplement and Transmittal of Revision 5 of W74 Storage SAR, BFS/NRC 00-054, July 27, 2000

9.5.10 CPC-117, Fabrication Specification - FuelSolutions™ W74 Dummy Fuel Assembly (Part of FC-703 package)

9.5.11 CPC-216, Calculation Package - FuelSolutions™) W74 Dummy Fuel Assembly Design and Analysis (Part of FC-703 package) (BRP Microfilm Cartridge/Frame 4828/1153)

9.5.12 EA-FC-703-FUEL DATA-04, Revision 0, Distribution Sequence of Fuel Bundles into Dry Fuel Casks ((Big Rock Point microfilm Cartridge/Frame – 4796/0677)

9.5.13 CMP.1503.009, PWR and BWR Assembly Hardware Co-60 Activation

9.5.14 EA-FC-703-Fuel Data-01, Calculate Amount of Cobalt in the Active Fuel Region of BRP Fuel Types (BRP Microfilm Cartridge/Frame 4831/0217)

9.5.15 Condition Report C-BRP-02-0017, Transportation of BRP Spent Nuclear Fuel with Unrestrained Corner Fuel Pin (BRP Microfilm Cartridge/Frame 4831/0449)

9.5.16 CPC-222, BRP Loose Fuel Rod HAC End Drop Buckling Evaluation (BRP Microfilm Cartridge/Frame 4831/0512)

9.5.17 EA-FC-703-Fuel Data-02, Fuel Volume Displacement Calculations to be Used in Determinations of Canister Helium Backfill Requirements

9.5.18 Letter CPC-01-447, Evaluation of Canister Helium Backfill Requirements at Big Rock Point, Andrew Roberts to John Broschak, April 30, 2001

9.5.19 Letter KPPowers (BRP) to EWBrach (U.S. NRC), Docket 50-155, License DPR-6, Big Rock Point Plant – Use of Wesflex Spent Fuel Management System, February 17, 2000

9.5.20 FuelSolutions™ Storage System Final Safety Analysis Report, Document Number WSNF-220, Docket Number 72-1026

9.5.21 Big Rock Point Nuclear Plant, Facility Change FC-704, ISFSI Storage Facilities and Equipment

9.5.22 Calculation S-10881-03, Revision A, Big Rock Point ISFSI

9.5.23 FuelSolutions™ Calculation Package CMP 1504.005, Revision 2, Determination of G-loads for the Storage Cask End Drop and Transfer Cask Side Drop

9.5.24 Calculation S-10881-000-001, Revision 1, ISFSI Pad Design

9.5.25 FuelSolutions™ Letter, Mark Haupt to John Broschak, October 4. 2000 (Subject: Applicability of CMPC 1504.005, Revision 2, to Big Rock ISFSI Design)

9.5.26 Big Rock Point Plant Manual, Volume 30, Dry Fuel Storage Exceptions Document

9.5.27 Calculation S-10881-02, Revision 3, Slope Stability Analysis

9.5.28 Calculation S-10881-04, Revision 1, Road Surfacing

9.5.29 EA-FC-703-03, W150 Transport Path Evaluation, Revision 0

9.5.30 Big Rock Point Volume 7F, ISFSI Security Plan

9.5.31 Big Rock Point Nuclear Plant Facility Change FC-704B, ISFSI and Existing CAS Security Interface

9.5.32 Calculation E-10881-01, Revision 1, Field of View Calculations for the ISFSI CCTV System

9.5.33 Calculation E-10881-02, Revision 0, Electrical Cable Size Selection for the ISFSI

9.5.34 EA-FC-703-36, Maximum Allowable Heat Load for a W74 Canister Residing Indefinitely Inside a Vertical W100 Transfer Cask (BFS Calculation Package Number CMPC.1505.402)

9.5.35 Palisades Nuclear Plant EA-EAR-2000-0309-08, Explosion Hazards Analysis (BRP Microfilm Cartridge/Frame 4830/2115)

9.5.36 Palisades Nuclear Plant, New Independent Spent Fuel Storage Installation, Fire Hazards Analysis, S&L Report Number SL-5596, Revision 0 (BRP Microfilm Cartridge/Frame 4831/0004)

9.5.37 Big Rock Point Nuclear Plant, DFS Explosion Hazards Analysis, S&L Calculation Number S-11102-10, Revision 0 (BRP Microfilm Cartridge/Frame 4830/2240)

9.5.38 May 2001 Assessment Report, Evaluation of Effects of a Forest Fire on ISFSI (BRP Microfilm Cartridge/Frame 4830/1823)

9.5.39 EMF-97-83, Consolidation of SPC Fuel at Big Rock Point Plant, July 1997

9.5.40 Letter from GCWithrow to U.S. Nuclear Regulatory Commission, Dockets 50-155 and 72-043, License DPR-6, Big Rock Point Plant – Registration as a User of USA/9276B(U)F-85 Pursuant to 10 CFR 71.12(c)(3).

10.0 STEAM POWER CONVERSION SYSTEM

Previously, this chapter identified the major components of the turbine generator, main steam, main condenser, circulating water, condensate and other subsystems supporting the power conversion process. The system components have been dismantled and the associated discussions deleted.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

Radioactive material from the operation of the plant originated from two sources. First, the products of nuclear fission are generally radioactive. Some may escape from the fuel from time to time. A small number of fission reactions also occur outside of the fuel from uranium as an impurity existing on or in the components near the reactor core and the cooling water flowing through the core. Second, a small fraction of the neutrons available from the fissioning process were captured by various materials near the reactor core including impurities in the circulating primary coolant. Many of these products of neutron capture become radioactive.

Radiological materials from the operation of the plant were identified and remediated in accordance with the methodology outlined in the Big Rock Point License Termination Plan (LTP).

Big Rock Point (BRP) Volume 25, Offsite Dose Calculation Manual and Related Documents (ODCM), specifies the requirements for effluent monitoring. The ODCM is controlled as BRP Volume 25, Part A, Offsite Dose Calculation Manual (ODCM). Liquid effluent releases at the Independent Spent Fuel Storage Installation effluent releases are unlikely. Therefore, no individual would be exposed to radiation in excess of that permitted by regulations (exposure greater than permissible concentrations of 10CFR20).

11.1.1 10CFR72.104 REQUIREMENTS FOR THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

The annual dose equivalent to any real individual who is located beyond the controlled area during normal operations and anticipated occurrences will be 25 mRem or less whole body, 75 mRem to the thyroid and 25 mRem to any other critical organ per 10CFR72.104.

The evaluation of compliance to 10CFR72.104 is as follows (Reference 11.6.1):

a. Big Rock Point annual limit is 25 mRem from all sources per 40CFR190 and 10CFR20 per BRP Volume 25, Part A, Offsite Dose Calculation Manual (ODCM).

b. Noble gases are no longer released via the plant stack and particulate releases are no longer a concern.

c. From the time of permanent plant shutdown for decommissioning until all source term was removed from the former operating plant area, the maximum whole body and organ doses from liquid effluents have averaged only a few percent of the quarterly and annual limits.

d. An evaluation was performed (see Reference 11.6.2) to determine the effect of Cask operations on off-site dose rates, including a comparison of dose rate increases against the site limits. This evaluation considered dismantlement activities with similar radiological impact encountered during the SFP inventory reduction campaign in the second quarter of 2000. The extensive high dose rate work performed during this campaign, which conservatively bounds the dose rates expected from Cask operations during the DFS campaign, did not result in significant increases in measured doses. The evaluation concluded that a slight increase may be observed in off-site doses due to Cask operations, but in comparison to limits, BRP will remain well within off-site dose rate limits. It also concluded that occupational dose rates would remain well within dose rate limits.

e. The evaluation performed to determine the effect of Cask operations on off-site dose rates in Reference 11.6.2 also showed that routine radwaste storage and handling did not result in increases in measured site boundary doses distinguishable above background.

f. Through NRC approval of 10CFR20.2002 applications submitted by Consumers Energy, BRP has been authorized to dispose of demolition debris containing up to 5 pCi/gm of principal gamma emitters in landfills licensed by the State of Michigan and/or the US Environmental Protection Agency (EPA). Dose evaluation scenarios contained in the submittal to the NRC concluded that the maximally exposed members of the public, if all 48.65 million pounds of impacted demolition debris contained 5 pCi/gm, would be 0.320 mrem/year to each of three transport workers.

g. The FuelSolutions™ FSAR assumes generic fuel assembly characteristics, such as enrichment, burnup, cobalt content, and cooling time, among others, to calculate dose rates throughout the loading process of spent fuel and long term storage at the ISFSI. Section 1.2.1 shows that the BRP site-specific spent fuel is within those parameters used in the Certificate of Compliance, thus assuring the FuelSolutions™ Cask system dose rates at BRP are below the limits of 10CFR72.104.

h. A radiological analysis of the BRP ISFSI pad and vicinity was conducted, as shown in Reference 11.6.3. The guidance provided by NRC in ISG-13 was considered in the analysis. The analytical model used was benchmarked against the dose rate results for the FuelSolutions™ W150 Storage Cask and W74 Canister. Distances from the ISFSI nuisance fence to a point where the dose rate is no more than 25 mrem per year to a "real individual" were determined. Dose rates were determined giving consideration to topography. No credit was taken for the berms used as a place to deposit the ISFSI excavation material. For the purposes of this analysis, it was conservatively assumed that the "real individual" would have occupancy of 2000 hours per year (40 hours/week for 50 weeks/year). The distance from the ISFSI to the point at which the "real individual" receives 25 mrem per year is within the existing owner controlled area of BRP.

The dose rate determination considered an array consisting of seven spent fuel Storage Casks and one Greater-Than-Class-C (GTCC) Storage Cask at the ISFSI. The dose rate from the GTCC Storage Cask was considered to be bounded by the design maximum dose rate from the spent fuel Storage Casks. Storage of the GTCC waste in conjunction with spent fuel storage on the ISFSI is acceptable under current regulatory requirements, since the ISFSI is being used to store spent fuel under a general license, and storage of GTCC will be stored under the 10CFR30 and 10CFR70 authority that is included in the BRP 10CFR50 license.

The analysis used the sum of the normal and off-normal Total Effective Dose Equivalent (TEDE) (resulting from leakage of the confinement boundary of the Canister as described in Chapter 7 of the FuelSolutions™ Storage System FSAR WSNF-223) and direct dose to determine the distance at which the annual dose to the whole body is 25 mrem. The bounding case of an 8 x 1 array of Casks (no hills and no berms) resulted in a distance of 248 meters in the east-west direction as the 25 mrem/year boundary. For comparison, the annual whole body dose rate at 450 meters from the south nuisance fence for a 4 x 2 array of Casks, which is the minimum distance from the ISFSI to the existing owner controlled area boundary, is 1.08 mrem. Thus, the existing owner controlled area provides more than adequate protection to the general public from dose rates resulting from ISFSI operation. It is also expected that actual dose rates due to storage of the BRP fuel will be substantially less.

As part of the fuel loading procedure, initial dose rates at various locations on the loaded Canister/Storage Cask were taken and compared to expected values. Before the loaded Storage Casks were allowed to leave the Containment, measured dose rates were evaluated to assure compliance with established administrative limits. The measured dose rates are part of the permanent record for each loaded Storage Cask. Retention of these records is controlled by the existing BRP procedures in accordance with 10CFR72.210.

The personnel exposure is controlled with specific work instructions, as needed for the activity.

In addition, radiological characterization of the ISFSI site, including analysis of soil samples, was conducted prior to construction of the ISFSI facility to ensure that no activation or contamination was present as a result of plant operations, and to establish a baseline for future dismantlement of the facility and free release of low activity materials (References 11.6.4 and 11.6.5). Reference 11.6.5 also notes that the State of Michigan participated with the NRC in the conduct of an independent characterization through a split sample program.

11.2 LIQUID WASTE MANAGEMENT SYSTEM

Radioactive materials in liquid waste arise from the activation of corrosion products formed in the nuclear steam supply system and the possible escape of fission products from fuel element cladding defects.

11.2.1 DESIGN BASES

Discussion of the operating plant liquid process monitoring system was deleted following system dismantlement. With the removal of the screenhouse structure, the inlet and discharge to Lake Michigan were isolated, eliminating the possibility of an uncontrolled release from building drains.

Liquid releases will continue utilizing temporary equipment or periodic manual sampling of effluents in accordance with site procedures and applicable State and local requirements.

11.2.2 SYSTEM DESCRIPTION

This section formerly described the Liquid Radwaste System during plant operation. It has been deleted.

11.2.3 RADIOACTIVE RELEASES

Releases of processed radioactive liquids to the environment were terminated during 2004, due to the removal of the liquid waste processing equipment and dismantlement of the screenhouse structure. Defueled Technical Specifications Section 6.7.3 requires that an Annual Radioactive Effluent Report be submitted to the U.S. Nuclear Regulatory Commission. This report summarizes effluents for the previous calendar year in accordance with BRP Volume 25, Part A, Offsite Dose Calculation Manual (ODCM) requirements.

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

With the permanent storage of all source terms on the Independent Spent Fuel Storage Installation (ISFSI), the plant-installed gaseous waste management system has been dismantled. The former system description was deleted in its entirety. Previous discussions on gaseous-particulate emissions as a result of dismantling activities have been deleted with the removal of source term from the former plant operating area.

11.3.1 RADIOACTIVE RELEASES

Defueled Technical Specifications require that an Annual Radioactive Effluent Report be submitted to the U.S. Nuclear Regulatory Commission. This report summarizes any liquid, gaseous, and solid waste effluents for the previous calendar year.

11.4 SOLID WASTE MANAGEMENT SYSTEM

All material from the radiologically controlled area is considered potentially radioactive waste until cleared by Radiation Protection Manager, or designate. Storage of these materials is in locations approved by Radiation Protection Manager, or designate.

11.4.1 SYSTEM DESCRIPTION

The solid waste management system consists of FuelSolutions™ Storage System (described in Chapter 9 of this UFHSR) used to contain spent nuclear fuel and GTCC components removed from the reactor vessel.

Dismantled systems and other non-compactable solids with relatively minor levels of contamination may be stored temporarily at other than the previously described locations, as approved by Radiation Protection Manager, or designate. Various boundaried and posted outside areas may be used for temporary storage, subject to ensuring against environmental release of radioactivity and maintaining worker safety.

11.4.2. BULK MATERIALS CONTROL PROGRAM

Big Rock established a program for the removal of bulk materials originating from demolition activities associated with the dismantlement of the facility under the provisions of 10CFR20.2002. The NRC granted approval of the request for an alternate waste disposal method on February 5, 2002 and as amended on January 19, 2005. Under the provisions of this approved request, BRP is authorized to dispose of demolition debris with trace amounts of NRC licensed radioactive material in a State of Michigan licensed Type II landfill and/or a landfill licensed by the US EPA. The program is technically supported by the Administrative and Implementing procedures, which define the methodology and the technical basis for implementation of the program.

The trace concentration of NRC licensed radioactive material in the demolition debris is bounded by a total principal gamma emitter concentration value of 5pCi/gm. This limit will ensure that radiological dose to workers and members of the public is kept As Low As Reasonably Achievable (ALARA). The NRC staff concluded that disposal of demolition debris under these *conditions in a State of Michigan licensed Type II landfill will result in a* calculated potential annual dose to a worker or member of the public of less than 1 mrem. This calculated dose is well within the 10CFR20 annual dose limit of 100 mrem and less than the annual dose limit of 25 mrem for decommissioning, which will allow for license termination and unrestricted use of the land.

11.5 AREA, PROCESS AND EFFLUENT MONITORING AND SAMPLING SYSTEMS

The area, process, and effluent monitoring systems formerly installed at BRP provided indications of the presence of radiation and radioactive material in areas, ventilation and liquid streams. For ISFGSI operations, monitors are provided to measure radiation fields and the presence of radioactive materials for normal operations and under accident conditions.

11.5.1 DESIGN BASES

Portable and telemetered instrumentation, rather than the fixed area monitor locations utilized during the plant operating phase, normally will be used for monitoring of work areas during dismantlement. Use of portable and telemetered instrumentation for personnel protection purposes allows location in the immediate proximity of work, or at locations best able to detect changes in radiation fields due to work processes.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Gaseous Monitoring System

The system description of the Stack Gas Monitoring (SGM) system has been deleted. Control and monitoring of processes resulting in gaseous effluent shall be performed with installed or temporary instruments and/or engineering controls. No gaseous effluent monitoring is required for ISFSI operations.

11.5.2.2 Process Liquid Monitoring System

No liquid process monitoring is required for ISFSI operations.
Liquid process streams shall be monitored in accordance with procedures developed for processes that have the potential to exceed the limits of 10 CFR Part 20.

11.6 REFERENCES

11.6.1 WEE01*05, Evaluation of the BRP Radiological Program Effectiveness under 10CFR72.212 (Action Item Numbers 7, 8, 9, 67) (BRP Microfilm Cartridge/Frame 4830/1668)

11.6.2 WEE01*07, Evaluation of the BRP Radiological Program Effectiveness under 10CFR72.212 (Action Item Number 10) (BRP Microfilm Cartridge/Frame 4830/1679)

11.6.3 Calculation R-10881-01, Revision 2, Dose Rates in the Vicinity of the Big Rock Point ISFSI (BRP Microfilm Cartridge/Frame 4830/1687)

11.6.4 EA-BRP-SC0100, Big Rock Point Background Radioactivity in Soil (BRP Microfilm Cartridge/Frame 4830/1676)

11.6.5 NRC Inspection Report, Big Rock Point, 05000155/2001/003(DNMS), June 20, 2001

11.6.6 NUREG-0596, Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Plants, August 1988

11.6.7 Minor Alteration MA-03-0037, Retention Basin Release System

11.6.8 Letter from the US NRC dated December 3, 2001, Big Rock Point Plant, Environmental Assessment and Finding of No Significant Impact Related to Request for Approval of Proposed Disposal Procedures in Accordance with 10CFR20.2002 (TAC Number MB1463)

11.6.9 Big Rock Point Engineering Analysis EA-BRP-RAE-04-01, Disposal of Demolition Debris at Environmental Quality Company, Revision 0

11.6.10 Big Rock Point Engineering Analysis EA-BRP-RAE-04-01, Disposal of Demolition Debris at Environmental Quality Company: Alternative, Revision 0

11.6.11 Letter from the US NRC dated December 3, 2001, Approval of Revision to Disposal Procedures in Accordance with 10CFR20.2002 (TAC Number L52096)

11.6.12 EA-BRP-RAE-0404, Dry Cask Accident Dose Analysis

12.0 RADIATION PROTECTION

12.1 ENSURING OCCUPATIONAL ALARA

The Big Rock Point (BRP) Radiation Safety Program is based upon the presumption that any exposure to ionizing radiation involves some risk. As a result part of the normal work process involving people in radiation controlled areas is to ensure that the Total Effective Dose Equivalent (TEDE) is kept as low as reasonably achievable (ALARA).

12.1.1 POLICY CONSIDERATIONS

The Policy of Consumers Energy and that of the BRP is to present a radiation safety program which controls radiation dose (external and internal) in a manner that avoids unnecessary and accidental doses, maintains doses to workers within regulatory limits and assures that doses to workers remain ALARA.

The organizational structure for conducting the Radiation Protection Program and minimum qualifications of the individuals occupying positions within that structure are defined in Section 12.5.1 of this Chapter, Chapter 13 of this UFHSR, the Big Rock Point Quality Program Description, and in site administrative procedures. Responsibilities of management and individual workers in carrying out the policy of ALARA are defined in the Radiation Protection Program. The Radiation Protection Program, as contained in site procedures, provides requirements and guidance to the ISFSI operation in the area of radiation protection. In addition to responsibilities the plan contains standards relating to management policy, radiation safety training, dose control, contamination control, surveys, instrumentation and incident investigation and analysis. The Radiation Protection Program applies to loading, transportation, and storage operations associated with the FuelSolutions™ Storage System, the placement of the Independent Spent Fuel Storage Installation (ISFSI), and dose rates resulting from the ISFSI.

Policy guidance in Regulatory Guide 1.8 relating to personnel selection and training, has been incorporated into Big Rock Point Quality Program Description.

The guidance of Section C.1 of Regulatory Guide 8.8 on ALARA and 8.10 on the Occupational Radiation Protection Program have been incorporated in the aforementioned radiation protection program.

12.1.2 DESIGN CONSIDERATIONS

Design considerations for the BRP to maintain the TEDE ALARA included: 1) shielding for radioactive components and systems; 2) location of equipment controls in low radiological dose areas; and 3) equipment design to allow quick maintenance in higher radiation dose areas.

All facility changes, minor alterations, and work orders in radiologically controlled areas require review to assure that doses are kept ALARA. This review is performed by the Radiation Protection Manager, or qualified alternate.

12.2 RADIATION SOURCES

The radiation sources that are the basis for the original radiation protection design of the BRP Plant and those that have been experienced during the operational history of the plant were formerly described in this section. Highly radioactive items [spent fuel and Greater-Than-Class-C (GTCC) radioactive waste] were permanently removed from wet storage in the spent fuel pool to dry storage on an ISFSI on March 26, 2003.

12.3 RADIATION PROTECTION DESIGN FEATURES

The BRP former operating Plant incorporated both design features and procedural controls to minimize occupational dose to radiation. This section formerly described radiation dose reduction features that were incorporated in the plant, design, including shielding and ventilation, and area and airborne radiation monitoring. Airborne radiation monitoring is discussed in Chapter 11 of this Updated Final Hazards Summary Report (UFHSR).

Regulatory position C.2 of Regulatory Guide 8.8 has generally been followed in the development of processes for radiological controls. Specifically:

a. Access Control of Radiation Areas

Access is controlled to radiologically controlled areas by means of radiation work permits or by individuals specifically trained in radiation protection procedures. Measurements of radiological conditions in radiologically controlled areas are made periodically. Changes in the status of any particular area are noted on the periodic surveys. Specific areas evaluated by the Radiation Protection Manager, or qualified alternate, will have a radiological status sheet posted for the area providing the radiological data for the area.

The movement of large sources of radiation is normally accomplished by the use of shielding and/or planned to minimize dose to personnel.

b. Control of Airborne Contaminants and Gaseous Radiation Sources

During former reactor operations, engineering control and ventilation flows were used to reduce airborne contaminants. The use of respiratory protection to reduce dose is provided but used only when other methods are not practical and the use of respiratory protection is necessary to maintain TEDE ALARA.

c. Radiation Monitoring Systems

The monitoring systems at the site are described in Chapter 11.

12.4 DOSE ASSESSMENT

The 30-plus years of operating history and the years of dismantlement activities have provided considerable information on actual occupational radiation doses received. This information is available in annual submittals to the Nuclear Regulatory commission pursuant to Regulatory Guide 1.16, Reporting of Operating Information.

Personnel monitoring is provided by using thermoluminescent dosimetry (TLD) as the primary external dose measurement. Thermoluminescent dosimetry results are verified by an outside organization accredited by the National Voluntary Laboratory Accreditation Program (NVLAP).

12.5 RADIATION PROTECTION PROGRAM

12.5.1 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

Portable radiation measuring instrumentation is selected and available to adequately measure routine and accident conditions considering expected ranges of dose rates and radionuclide mixtures. Adequate supplies are on hand to cover normal operations to meet the requirements of 10 CFR 20, Standards for Protection Against Radiation. Personnel are qualified in the use of the portable instrumentation, which is calibrated semi-annually and functionally checked on a routine basis.

Other laboratory equipment is used to identify radionuclides and mixtures for compliance with 10 CFR 20 and to meet industry standard lower levels of detection.

Instrument calibrations are performed in accordance with the requirements of the Big Rock Point Quality Program Description.

Control points are located at or near the boundary of the radiologically controlled area and contain equipment commensurate with the work being controlled.

12.5.2 PROCEDURES

Administrative and working level procedures are provided for the Radiation Protection Program in accordance with the commitments of the Big Rock Point Quality Program Description.

12.6 REFERENCES

12.6.1 Calculation R-10881-01, Revision 1, Dose Rates in the Vicinity of the Big Rock Point ISFSI (BRP Microfilm Cartridge/Frame 4831/2498)

13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

Organization and responsibilities of Big Rock Point (BRP) are discussed in this section, BRP Volume 2, Defueled Technical Specifications for Big Rock Point, and in BRP Volume 34, Quality Program Description for Big Rock Point. The Defueled Technical Specifications are included as BRP Volume 2, Defueled Technical Specifications for Big Rock Point, and referenced in Chapter 16 of this Updated Final Hazards Summary Report (UFHSR). The Quality Program Description is incorporated as BRP Volume 34, Quality Program Description for Big Rock Point, and is referenced in Chapter 17 of this UFHSR. The Senior Nuclear Officer (SNO) is the Senior Vice President-Nuclear, Fossil, and Hydro Operations (NFHO). The SNO is responsible to the President and Chief Operating Officer for operation, maintenance, and decommissioning of BRP.

The BRP Site organization is defined in Volume 34A-01, Site Organization, Responsibilities, Quality and Training. The Site General Manager (SGM) is responsible to the SNO for operation, maintenance, and dismantlement of the nuclear power plant in such a manner as to achieve compliance with plant licenses and applicable regulations. The SGM, or designate, verifies that Dry Fuel Storage Technical Specification surveillances have been met. These responsibilities and the associated authority are delegated in writing. The SGM, or designate, shall verify that required Security and Emergency Plan Staffing have been met. The SGM administers the Quality Program for the plant.

An Independent Safety Reviewer advises the SGM on all matters related to nuclear safety.

The Quality Assurance Lead is responsible to the SNO for establishing the Quality Program. Responsibilities of the Quality Assurance Lead and the Quality Assurance Organization are specified in BRP Volume 34, Quality Program Description for Big Rock Point.

As described in Volume 34, Quality Program Description for Big Rock Point, The Independent Safety Review Committee (ISRC) advises the SNO in matters regarding activities affecting safety-related programs and/or important-to-safety structures, systems, or components.

13.1.2 PLANT ORGANIZATION RESPONSIBILITIES

The SGM is responsible for the overall management of the facility during dismantlement. This includes managing overall compliance with license limitations, the License Conditions, the ISFSI Emergency Plan, the Quality Program, and State and Federal Regulations.

13.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

Staff qualifications are established consistent with the intent of ANSI Standard 18.1-1971 and are described in Volume 34, Quality Program Description for Big Rock Point

At least one Radiation Protection staff member shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. For the purpose of this section, "Equivalent," as utilized in Regulatory Guide 1.8 for the bachelor's degree requirement, may be met with four years of any one or combination of the following: (a) formal schooling in science or engineering, or (b) operational or technical experience/training in nuclear power.

13.1.4 PLANT ADDITIONAL SUPPORT

As discussed in BRP Volume 34, Quality Program Description for Big Rock Point, when needed to support Independent Spent Fuel Storage Installation (ISFSI) operations, Consumers Energy Company will retain or contract persons knowledgeable in ISFSI operations, ISFSI structural and electronic systems, nuclear engineering, radiation protection, and quality assurance. For ISFSI operations, chemistry and radiochemistry controls do not apply. There are no mechanical or electrical systems directly associated with dry fuel storage casks.

Quality Program activities are conducted in accordance with Volume 34, Quality Program Description for Big Rock Point, which is incorporated into this UFHSR by reference in Chapter 17.

The Security Force will be supervised as described in the ISFSI Security Plan (reference Section 13.6 of this UFHSR).

Responsibility for implementing the Fire Protection Program is described in Volume 26, Fire Protection Summary. Refer to Section 9.5 of this UFHSR.

The SGM has the discretion to delegate responsibilities to the support staff. Volume 34A-01, Site Organization, Responsibilities, Quality and Training, provides the site organizational details and reporting structure. Included in the procedure is a list of the specific delegated duties. The procedure also provides a Leadership Succession for times when the SGM is unavailable.

13.2 TRAINING

The SGM is responsible for ensuring personnel are trained and qualified to perform the functions of their jobs in accordance with plant procedures.

General training requirements and requirements for Quality Program training are addressed in BRP Volume 34, Quality Program Description for Big Rock Point. The Quality Program Description also provides requirements for training, qualification retraining and requalification applicable to quality functions.

Training programs have been established and maintained to ensure operating and support organization personnel possess the appropriate knowledge, experience, and proficiency essential to ensuring safety at BRP.

Many of the training programs in effect for the plant under power operation were either certified or accredited by the Institute of Nuclear Power Operations (INPO). INPO does not provide an oversight function for the permanently defueled plant and therefore does not provide certification or accreditation of the current training programs.

13.2.1 PLANT AND SUPPORT STAFF TRAINING PROGRAMS

ISFSI Emergency Plan Training is addressed in the ISFSI Emergency Plan, which is incorporated by reference in Section 13.3 of this UFHSR.

Requirements for training of security personnel are addressed in Section 13.6 of this UFHSR.

13.2.1.1 Plant Access Training

Plant Access Training (PAT) will be provided to all personnel requiring unescorted access. There is an associated examination required for course completion. The course consists of two modules:

a. The orientation module provides information with respect to security, site and Independent Spent Fuel Storage Installation (ISFSI) description, procedures, and Quality Assurance/Quality Control necessary to obtain unescorted access authorization into the radiologically controlled area (RCA).

b. The radiation protection indoctrination module provides radiation protection familiarization for persons requiring routine or frequent access to the site's RCA.

A requalification course and examination is administered annually and is required for the renewal of unescorted access authorization.

Plant Access Training meets the intent of INPO 87-004, Guidelines for General Employee Training.

13.2.1.2 Radiation Protection Manager Training

The Radiation Protection Manager will be trained in accordance with the requirements of the Big Rock Point Quality Program Description (Volume 34). The course may include information on:

a. Radiation safety fundamentals;

b. 10 CFR 20, Standards for Protection Against Radiation, and BRP radiation exposure limits and controls/external exposure control measures;

c. Radioactive contamination limits and controls/internal exposure control measures;

d. Radioactive materials control methods;

e. Radioactive waste management and control procedures;

f. Preparation for emergencies – Worker-related information and actions;

g. Radiation Protection Program description

h. Respiratory protection program description and NUREG-0041, Manual of Respiratory Protection Against Airborne Radioactive Material, compliance; and

i. Radiation survey requirements.

13.2.1.3 ISFSI Training Program

The program incorporates the "Systems Approach to Training" procession for tasks required to maintain safe configuration of the spent nuclear fuel that is permanently stored on the ISFSI.

13.3 EMERGENCY PLANNING

13.3.1 ISFSI EMERGENCY PLAN

An emergency plan has been developed and implemented for responding to emergencies. The plan is contained in Volume 35, Big Rock Point Independent Spent Fuel Storage Installation (ISFSI) Emergency Plan, which is incorporated into this UFHSR by reference.

On September 30, 1998, the US Nuclear Regulatory Commission granted an exemption to Consumers Energy from certain sections of 10 CFR 50 concerning emergency response planning, thereby allowing the discontinuation of off-site emergency planning activities and the reduction of the scope of on-site planning at the BRP Plant.

On October 13, 2005, the U.S. Nuclear Regulatory Commission Safety Evaluation Report (SER) accepted a reduction in commitment to the Big Rock Point ISFSI Emergency Plan. This reduction was in the area of staffing and included evaluation of revised radiological impact area analysis.

The ISFSI Emergency Plan addresses dry fuel storage events during loading, transportation, or storage of spent nuclear fuel in the FuelSolutions™ Storage System.

13.3.2 EMERGENCY PLAN IMPLEMENTING PROCEDURES

Procedures for implementing the emergency plan are contained in BRP Volume 35A, Big Rock Point ISFSI Emergency Plan Implementing Procedures.

13.4 REVIEW AND AUDIT

Provisions for conducting reviews and audits of activities affecting systems, structures, and components necessary for the safe storage of spent fuel have been established. The review and audit provisions are addressed in BRP Volume 34, Quality Program Description for Big Rock, which is incorporated by reference in Chapter 17 of this UFHSR.

13.5 SITE PROCEDURES

13.5.1 ADMINISTRATIVE PROCEDURES

13.5.1.1 Conformance with Regulatory Guide 1.33 – Quality Program Requirements (Operation)

Consumers Energy complies with the regulatory position of Regulatory Guide 1.33 (2/78, Revision 2), as modified by the exceptions stated in BRP Volume 34, Quality Program Description for Big Rock Point, which provides Policy and Implementation requirements for instructions, procedures and drawings.

13.5.1.2 Administrative Control Requirements and Standards

Activities affecting the quality of structures, systems, and components required for the important to safety components and activities for providing radiological control are accomplished using instructions, procedures and drawings appropriate to the circumstance that include acceptance criteria for determining if an activity has been satisfactorily completed.

13.5.1.3 Measures to be Taken Following Incidents

To prevent or limit adverse consequences following incidents, the corrective action process requires:

a. Initiation of immediate corrective action to ensure the safety of plant personnel and the public.

b. Notification of the NRC in accordance with plant procedures and NRC regulations.

c. Investigation of the condition and establishment of any corrective actions necessary to resolve the condition and prevent recurrence.

13.5.1.4 Administrative Procedural Controls

Site procedures provide requirements for use and control of procedures as well as processing new procedures, revisions and editorial changes to procedures, temporary procedures and procedure cancellations. The procedures program provides instructions applicable for procedures required by Volume 34, Quality Program Description for Big Rock Point, and those not required. The procedures program identifies responsibilities of management, preparers, reviewers and document control and encompasses both operational and dismantlement activities. It provides for the review of safety-related and important-to-safety implications in accordance with 10 CFR 50.59 for the review of decommissioning activities in accordance with 10 CFR 50.82(a)(6), and for the review of safety implications in accordance with 10 CFR 72.48 for the ISFSI facility and storage cask system.

Site procedures provide for identification of procedures developed for decommissioning.

Site procedures and procedure revisions are approved by the Site General Manager prior to use.

Site procedures are periodically reviewed as described in the procedure program.

The Department Head approves all working level procedures prior to their issuance.

The Independent Safety Reviewer reviews procedures and revisions to those procedures that affect nuclear safety to ensure that prior Nuclear Regulatory Commission approval pursuant to 10 CFR 50.59 or 10 CFR 72.48 is not required. This individual makes recommendations to the SGM as to whether the procedure should be approved.

Site procedures will be filed in the site's Document Control Center for periods of time consistent with guidance given in BRP Volume 34, Quality Program Description for Big Rock Point.

The requirements of 10 CFR 50, Appendix B, apply to the construction, maintenance, operation, and dismantlement of the ISFSI. The criteria in Volume 34 will continue in force for the duration of the license to operate the ISFSI.

13.5.2 ISFSI OPERATING PROCEDURES

This section describes the BRP ISFSI operating procedures, which include procedural operating safeguards to be established, the procedures for normal operation, and the plans for handling emergency situations that may arise in the operation of the ISFSI.

13.5.2.1 Basic Operating Principles

Operations affecting important-to-safety structures, systems, or components shall be performed in accordance with specific procedures.

Surveillance tests and routine preventive maintenance of protective devices and critical equipment will be done in accordance with established schedules.

Personnel leaving radiologically controlled areas and equipment being removed from such areas will be surveyed to an extent adequate for control of contamination.

Operation of the radioactive waste handling system will be such as to assure that the disposal of radioactive materials will not result in the exposure of any persons on or off the plant site to radiation in excess of permissible limits. These operations will be performed in accordance with NRC regulations (10 CFR 20 and 10 CFR 71). Solid wastes are stored in shielded containers, as necessary with demarcation and boundaries established in accordance with ALARA principles.

When determined necessary by the corrective action process, procedural controls will be developed or revised to prevent recurrence of conditions adverse to quality.

13.5.2.2 Description of Operating Procedures

Operations procedures have been grouped as follows:

a. DOP - Decommissioning Operating Procedures - provide instructions on the operation of systems and portions of systems which require operation with spent nuclear fuel permanently stored on the ISFSI and are required to support the dismantlement and dry fuel storage processes. Also provide instructions for placing the ISFSI in a stable condition under off-normal conditions.

b. Fuel transfer Procedures – With all spent nuclear fuel permanently stored at the ISFSI, procedures are available for transferring the dry fuel storage canisters out of overpacks for repair. When final shipment offsite occurs, fuel transfer will be done in accordance with approved procedures.

13.5.3 OTHER PROCEDURES

Procedural requirements for Security procedures are addressed in the ISFSI Security Plan discussed in Section 13.6 of this UFHSR.

Emergency preparedness procedures are addressed in the ISFSI Emergency Plan discussed in Section 13.3 of this UFHSR.

Other procedural requirements are addressed in BRP Volume 34, Quality Program Description for Big Rock Point .

13.6 SECURITY

Consumers Energy's plan for physical protection of the BRP ISFSI is described below. This plan is a separate submittal withheld from public disclosure pursuant to 10 CFR 73.21, Requirements for Protection of Safeguards Information. This plan is submitted, amended, revised, changed, or updated and approved on a schedule separate from this UFHSR.

This plan contains "Safeguards Information" of a type specified in 10 CFR 73.21 and is not included in this UFHSR.

13.6.1 BIG ROCK POINT ISFSI SECURITY PLAN

The BRP ISFSI Security Plan provides a comprehensive program of physical protection provisions to meet the requirements of 10 CFR 50.54(p). The ISFSI Security Plan addresses equipment, areas, isolation zones, and demonstrates how BRP Plant complies with the requirements of 10 CFR 73 and 10 CFR 11.

The BRP ISFSI Security plan includes safeguards contingencies and provides guidance to personnel in order to accomplish specific, defined objectives in the event of threats, thefts, or radiological sabotage relating to special nuclear material or the BRP ISFSI. The Safeguards Contingency Plan is developed and maintained to meet the requirements of 10 CFR 73, Appendix C. Changes to the Safeguards Contingency Plan are submitted to the NRC in accordance with 10 CFR 50.4. These changes must be consistent with the provisions of 10 CFR 50.54(p) or may be amended per 10 CFR 50.90.

The BRP ISFSI Security Plan includes Suitability, Training, and Qualification Plan requirements that provide for the selection, training, equipping, testing, and qualification of individuals who will be responsible for protecting special nuclear materials at the BRP ISFSI. The Suitability, Training, and Qualification requirements were developed, implemented, and maintained to assure that Security Personnel effectively perform their assigned security related job duties. The Suitability, Training, and Qualification requirements address the General Criteria for Security Personnel pursuant to 10 CFR 73 Appendix B.

The ISFSI facility is physically protected pursuant to 10 CFR 72, Subpart H. The pad is located in an isolated position on plant property and is incorporated into the scope of the BRP ISFSI Security Plan.

14.0 INITIAL RESEARCH AND DEVELOPMENT PROGRAM

14.1 HISTORICAL RESEARCH AND DEVELOPMENT PROGRAM

Previously, this Chapter provided a description of the Big Rock Point role in the historical Atomic Energy Commission and General Electric Company research and development to support a High Power Density Development Program. It also provided information on subsequent fuel related research and cobalt-60 production. Under the permanently defueled condition the operation of the reactor is not feasible. Therefore this entire chapter, which previously described historical special uses of the reactor is deleted.

15.0 ACCIDENT ANALYSES

15.1 CALCULATIONAL METHODS AND INPUT PARAMETERS

With all the fuel in dry storage at the Independent Spent Fuel Storage Installation (ISFSI), the operational transient analyses are no longer applicable. Therefore, the transient discussions of Sections 15.1 through 15.9 have been deleted [Revision 6 to the Updated Final Hazards Summary Report (UFHSR)] contains a complete discussion of these events; see Section 1.2.1). Section 15.2 provides a discussion of the applicable events analyzed for dismantlement. Fuel-related accidents are discussed in Section 15.2.2. External events are discussed in Section 15.2.3. Non-fuel-related events resulting from dismantlement operations were assessed, compared against the Final Generic Environmental Impact Statement (FGEIS) (Reference 15.3.1) and found to be within the bounds of the generic analysis. The results of these analyses are discussed in Section 15.2.4.

15.2 DECOMMISSIONING ACCIDENT CONSIDERATIONS

15.2.1 INTRODUCTION

The following sections discuss accidents that could occur during dismantlement. A wide range of potential accidents have been reviewed, which could be of potential public health and safety concern if release of radioactive material were to occur.

Former Section 15.10.2 provided a discussion of fuel handling events that were applicable while fuel was stored in the spent fuel pool. With fuel in dry storage at the ISFSI, this discussion has been deleted.

Section 15.2.3 (previously Section 15.10.3) provided a discussion of external events and their effects with fuel stored in the spent fuel pool. With fuel in dry storage, the discussion has been revised to reflect ISFSI operations.

With all fuel stored on the ISFSI, non-fuel accidents are credible for the plant industrial area (former protected area, not part of the ISFSI). Non-fuel related events that could occur as a result of decommissioning operations were assessed, compared against the Final Generic Environmental Impact Statement (FGEIS) (Reference 15.3.1) and found to be within the bounds of the generic analysis. The results of these analyses are discussed in Section 15.2.4.

Accidents involving fuel are discussed in the BNFL Safety Analysis Reports (SARs) (References 15.3.2 and 15.3.3). Chapter 11 of BNFL FuelSolutions™ Storage System Final Safety Analysis Report; Document Number WSNF-220 presents the evaluation of the FuelSolutions™ W150 Storage Cask and the W100 Transfer Cask. Chapter 11 of the BNFL FuelSolutions™ W74 Canister Final Safety Analysis Report, Document Number WSNF-223 presents the evaluation of the FuelSolutions™ W74 Canister. The design basis off-normal, postulated accident, and natural phenomena events are defined in Sections 2.3.2, 2.3.3, and 2.3.4 of the SARs, respectively.

The design basis conditions considered for the storage cask, transfer cask, and canister are as follows:

a. Normal Conditions:

1. Normal Ambient Conditions for the ISFSI
2. Ambient Conditions for the Fuel Building
3. Fuel Rod Rupture
4. Pressure
5. Dead Load
6. Live Load
7. Handling Loads
8. Testing Loads
9. Off-normal Conditions
10. Extreme Ambient Conditions
11. Fuel Rod Rupture
12. Internal Pressure
13. Misaligned Cask for Horizontal Transfer
14. Blockage of Storage Cask Air Inlet Vents
15. Hydraulic Ram Failure During Horizontal Transfer

b. Accident Conditions:

1. Accident Thermal Conditions
2. Storage Cask Vent Blockage
3. Transfer Cask Loss of Neutron Shield
4. Cask Drop
5. Tip-over of Storage Cask on J-skid
6. Fire
7. Fuel Rod Rupture
8. Internal Pressure
9. Leakage of the Confinement Boundary
10. Explosive Overpressure

c. Natural Phenomena:

1. Flooding
2. Tornado
3. Earthquake
4. Wind
5. Burial Under Debris
6. Lightning
7. Snow and Ice
8. Volcanism

15.2.1.1 Dose Limits

Previous accident analyses in the Big Rock Point (BRP) Updated Final Hazards Summary Report evaluated public dose to the limits established in 10 CFR 100 as reactor siting criteria limits at the exclusion area boundary and low population zone distances. These dose limits correspond to 25 rem total body from noble gasses and 300 rem to the thyroid from iodines, as described by TID-14844 (Reference 15.3.4).

Big Rock Point implemented the guidelines of the Environmental Protection Agency (EPA) Manual of Protective Action Guides (PAGs) and Protective Actions for Nuclear Incidents, EPA-400 (Reference 15.3.5) 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 100. Revised dose calculations reflecting plant decommissioning and dismantlement described in this section have been performed in accordance with the guidelines of EPA-400.

15.2.1.2 General Assumptions

The following assumptions have been made for the purposes of the revised accident analyses:

a. No fuel is present in the reactor vessel.

b. The dominant dose pathway is from airborne release with conservative dispersion factors of 1.8E-4 sec/m^3 (fumigation conditions) for elevated release and 6.48E-4 sec/m^3 for ground level release per Regulatory Guide 1.25 (Reference 15.3.6).

c. Ground level release results in higher off-site doses, thus has been assumed in calculation of doses to the public.

d. Liquid radioactivity from accidents involving radioactive liquids are assumed contained onsite.

e. No containment ventilation isolation or other ventilation barrier to release of radioactivity is assumed for determination of dose to the public from the bounding fuel damage accidents.

f. Plant high efficiency particulate (HEPA) filters will be utilized when large source terms are present (when generating large quantities of radioactive particulates from dismantlement activities involving the reactor vessel, thermal shield or reactor cavity concrete). This is consistent with assumptions of the FGEIS (Reference 15.3.1).

g. Dose conversion factors of EPA-400 and its companion document, EPA-402 (Reference 15.3.7) have been utilized.

15.2.2 ACCIDENTS INVOLVING FUEL

Use of the dry fuel storage system has its own safety evaluation (BNFL SARs).

This section previously addressed events involving fuel stored in the spent fuel pool. Big Rock Point evaluations indicated that dose from the bounding fuel accident; assuming a free release path without ventilation isolation, fell below the PAGs of EPA-400 prior to 68 days post shutdown (References 15.3.13 and 15.3.10). Off-site doses for external, skin, thyroid and TEDE were calculated for various decay times. Assuming damage to 500 bundles in the pool at 68 days following plant shutdown, the site boundary doses dropped to less than the EPA Protective Action Guides (PAGs) of one rem TEDE and five rem to the thyroid. This analysis assumed a total of 500 assemblies in the spent fuel pool with 84 being discharged from the final core off-load. The 500-assembly inventory was based on operating the plant until the end of license in May 2000 rather than the actual last day of operation on August 29, 1997. The pool contained only 441 assemblies per the plant license.

For dry fuel storage on the ISFSI, evaluation of dose from a storage canister failure was performed (reference 15.3.15). Based on the results of the analysis, a breach of canister integrity, plus integrity failure of all fuel pins in the 63 active bundles within the canister, would result in doses to the thyroid, skin, lens of eye, or total body below PAGs at 100 meters from the ISFSI pad.

15.2.3 EXTERNAL EVENTS

An assessment of external events was made to evaluate the effects of natural and manmade events on dismantlement activities. The hazards associated with these events are assumed to be consistent with those that could have occurred while BRP was in operation. This section describes the evaluations performed to assure protection of public health and safety.

The external events discussion concentrated on the effects pertaining to the spent fuel pool and supporting components utilized to insure the safe storage and control of spent fuel. Portions applicable to the decommissioning site, with all fuel stored in dry fuel casks on the ISFSI have been retained. For accidents involving fuel on the ISFSI, the BNFL SARs, Chapter 11 addresses natural phenomena accident considerations.

15.2.3.1 Loss of Off-site Power

During decommissioning with all fuel stored on the ISFSI, offsite power is not needed.

15.2.3.2 Aircraft Hazards

Consumers Energy Company evaluated potential aircraft hazards on BRP in response to Systematic Evaluation Program Topic II-1.C (Reference 15.3.9) and as part of the BRP Spent Fuel Pool Expansion Hearings. The evaluations and NRC Staff's conclusions determined the cumulative realistic probability of an aircraft crashing into the plant was very low (2E-08 per year) in 1984 and has since been further reduced by the closing of military training routes. Further consideration of the interaction between aircraft hazards and decommissioning is not warranted.

15.2.3.3 External Flooding

Consumers Energy Company evaluated flooding potential and protection requirements at BRP in response to Systematic Evaluation Program Topics II-3.B and II-3.B.1 (Reference 15.3.9). Nuclear Regulatory Commission staff review of detailed hydrologic engineering calculations, maps, level surveys, and photographs concluded that external flooding caused by either Probable Maximum Precipitation (PMP) or lake flooding would not exceed 594.0 feet mean sea level (MSL) at the turbine building or 584.1 feet MSL inside the intake structure. In view of this finding and that the extreme nature of the assumptions regarding a probable maximum flood event (PMF), the staff concluded that the plant could have safely shut down.

With all spent fuel stored on the ISFSI, plant shutdown is no longer applicable.

Drainage ditches, catch basins and drainage pipes were designed based on a hydrological study (Reference 15.3.14), with the result that the ISFSI pad will not be undercut by floodwaters from the Probable Maximum Flood event (flood condition of 21 feet per second water velocity and a 50-foot flood height, including consideration of tsunami and seiches).

The maximum flooding height (surge height) from lake flooding is 587.4 feet elevation. The nominal finished grade at the ISFSI pad is at Elevation 618.5 feet and the top of pad elevation is 621.75 feet. Thus, the Storage Casks on the pad would not be subjected to the maximum predicted flood height from lake flooding.

15.2.3.4 Probable Minimum Water Level

During decommissioning with all fuel stored on the ISFSI, Systematic Evaluation Program Topic II-3.C (Reference 15.3.9) evaluation of a probable minimum lake water level (570.0 feet MSL) is no longer applicable, as spent fuel pool cooling is no longer needed.

15.2.3.5 Tornados and Extreme Winds

The annual strike probability of a tornado is very low for the BRP site. As discussed in Section 2.3.1 of this UFHSR, tornados have been reported 25 times between 1950-1977 within a 60-mile radius of the BRP site, excluding the water area over Lake Michigan. Based upon the tornado characteristics for the site region, probability calculations indicate that the recurrence interval for a tornado at the site is about 5150 years.

The storage cask system has been evaluated in the BNFL SARs for tornados and wind. Since enclosures for containing contamination during dismantlement and remediation are temporary structures, controlled by administrative processes, further consideration of these events is not warranted.

15.2.3.6 Earthquake

As discussed in Section 2.5.1 of the BRP UFHSR, the 1961 seismicity evaluations concluded the probability that earthquakes of significant intensity will occur in the general site area appears to be very low. Coast and Geodetic Survey information identified seven earthquakes in Michigan prior to October 1959, all of which were classified as intermediate or minor. The nearest recorded earthquake was in 1905 and was centered near Menominee, Michigan, approximately 110 miles from the plant site.

The storage cask system has been evaluated in the BNFL SARs for earthquakes. Since enclosures for containing contamination during dismantlement and remediation are temporary structures, controlled by administrative processes, further consideration of these events is not warranted.

15.2.3.7 Fire Events

A fire event could affect plant systems and equipment used during dismantlement. Adequate levels of fire protection features as described in the Fire Protection Program will be maintained to minimize the probability of occurrence of a fire and, should a fire occur, limit the consequences. These features include:

a. Control of transient combustible materials and ignition sources

b. Personnel training and qualification programs

Of primary concern will be areas where a fire could impact spent fuel and radiological controlled areas. The cask storage system has been evaluated in the BNFL SARs for fire considerations. See Section 15.1.4 for non-fuel accident evaluation considerations.

15.2.3.8 Freezing

With the fuel stored on the ISFSI, the plant heating system has been removed.

15.2.4 NON-FUEL RELATED DECOMMISSIONING ACCIDENTS

An evaluation of potential non-fuel related dismantlement accidents at BRP has been performed (Reference 15.3.10). Activities following final plant shutdown were evaluated, including system and equipment deactivation, decontamination, and dismantlement; radioactive material handling and storage; and transportation of radioactive materials. Types of postulated accidents reviewed were: explosions and fires, loss of contamination control, waste transportation accidents, external events, and natural phenomena. In addition to the standard dismantlement activities, postulated accidents associated with potential long term storage of radioactive waste during decommissioning also were evaluated.

EA-BRP-RAE-0301, Contribution to Off-site Dose from Demolition Activities (Reference 15.3.11) provided an evaluation of off-site dose from particulate emissions released during demolition activities where the release point is not directly monitored for flow rate or radioactive material concentration. The results provide assurance that a demolition accident was within the bounds of the FGEIS (Reference 15.3.1) if contamination levels are below Regulatory Guide 1.86 (Reference 15.3.12) guideline for beta-gamma contamination of 50,000 dpm/100 cm^2.

Based on this review, it is concluded that all postulated decommissioning accidents for BRP are bounded by the results described in the FGEIS. Thus, as concluded by the FGEIS, decommissioning will have a minimal impact on public safety and health. This conclusion is further supported by the fact that BRP, at a rating of 240 Mwt, was significantly smaller plant than the 3320 Mwt reference BWR. This fact reduces total quantities of radioactivity present on site, total volumes of waste produced and shipping volumes.

15.2.4.1 Accident Prevention and Mitigation

The baseline BWR assumed in the FGEIS utilizes HEPA filters for plant ventilation effluents, whereas the original BRP ventilation system did not include HEPA filters. High efficiency particulate filters were only used for specific sources such as fume hoods and offgas. To remain within the bounds of the FGEIS and recognizing that during dismantlement, airborne particulate releases could be significantly reduced by plant HEPA filtration, a HEPA filtration system was installed in the ventilation system and will be used for dismantlement activities involving major source terms of particulate activity.

15.3 REFERENCES:

15.3.1 NUREG-0586, Final Generic Environmental Impact Statement (FGEIS) on Decommissioning Nuclear Facilities, October 1988

15.3.2 BNFL Fuel Solutions™ Storage System Safety Analysis Report, Document Number WSNF-220

15.3.3 BNFL Fuel Solutions™ W74 Canister Safety Analysis Report, Document Number WSNF-223

15.3.4 J.J. Dinunno, et al, Technical Information Document #TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, US Atomic Energy Commission, March 1962

15.3.5 EPA 400-R-92-001, Manual of Protective Actions Guides and Protective Actions for Nuclear Incidents, May 1992

15.3.6 Regulatory Guide 1.25 (Safety Guide 25), Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident, US NRC, 1972

15.3.7 EPA 402-R-93-081, External Exposure to Radionuclides in Air, Water and Soil, September 1993

15.3.8 NUREG-0586, Supplement 1, Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Power Facilities, August 1988

15.3.9 NUREG-0828, Integrated Plant Safety Assessment - Systematic Evaluation Program for the Big Rock Point Plant, Final Report, May 1984

15.3.10 EA-BRP-DP-CH5-2, Review of GEIS and Other Non-fuel Accidents for Decommissioning, March 30, 1995

15.3.11 Engineering Analysis, EA-BRP-RAE-0301, Contribution to Off-site Dose from Demolition Activities, May 2003

15.3.12 US Nuclear Regulatory Commission Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974

15.3.13 Big Rock Point Engineering Analysis, Fuel Damage Decommissioning Accident Analysis for BRP, EA-BRP-DP-CH5-1, December 1994

15.3.14 Calculation S-10881-03, Revision A, Big Rock Point ISFSI, Volume 33 Reference

15.3.15 Big Rock Point Nuclear Plant Engineering Analysis, EA-BRP-RAE-0404, Revision 1, Dry Cask Accident Dose Rate Analysis, August 8, 2004.

15.3.16 U.S. NRC to Big Rock Point dated October 13, 2005, Approval of Revision to the Big Rock Point Emergency Plan – Safety Evaluation Report

16.0 DEFUELED TECHNICAL SPECIFICATIONS

16.1 DEFUELED TECHNICAL SPECIFICATIONS

The Defueled Technical Specifications contained in Appendix A, as revised, of the "Consumers Power Company Docket Number 50-155 Big Rock Point Plant Facility Operation License Number DPR-6," are incorporated by reference as a part of this Updated Final hazards Summary Report (UFHSR).

The Defueled Technical Specifications and License are updated and amended on schedules separate from this UFHSR.

17.0 QUALITY ASSURANCE

17.1 QUALITY ASSURANCE

BRP Volume 34, Quality Program for Big Rock Point, as revised, is incorporated by reference as a part of this Updated Final Hazards Summary Report (UFHSR). Quality Program Description for Big Rock Point is updated and revised on a schedule separate from this UFHSR.

18.0 HUMAN FACTORS ENGINEERING

The following sections discussed human factors issues associated with safe shutdown of the reactor. These issues are no longer applicable.

18.1 CONTROL ROOM DESIGN REVIEW (CRDR)

Previously, this section discussed the design review that was performed to ensure that an effective man-machine interface existed in the control room permitting the reactor to be safely shutdown, and that effective, organized and informative controls and displays were available for normal startup and shutdown operations. The Control Room has been dismantled; therefore, the discussion in this section has been deleted.

18.2 SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

Previously, this section discussed the process by which the plant satisfied the safety parameter display system functional requirements. The plant has been dismantled; therefore, the discussion in this section has been deleted.

18.3 EMERGENCY RESPONSE CAPABILITY

Previously, this section discussed the background of issues associated with emergency response to potential accidents associated with a plant with an operating reactor. The spent fuel has been transferred to dry fuel storage at the Independent Spent Fuel Storage Installation (ISFSI); therefore, discussion in this section has been deleted.

An ISFSI emergency plan has been developed and implemented for responding to emergencies reflecting ISFSI operations. The plan addresses emergency response facilities, equipment, procedures, and staffing and is contained in BRP Volume 35, ISFSI Emergency Plan, which is incorporated into this Updated Final Hazards Summary Report (UFHSR) by reference.

Adequacy of the emergency response capability is reviewed by the NRC pursuant to 10 CFR 50.47, with exceptions as documented in U.S. Nuclear Regulatory Commission Safety Evaluation Reports (SER) dated September 30, 1998 and October 13, 2005.