Consumers Energy

Big Rock Point Nuclear Plant
10269 US 31 North
Charlevoix MI 49720

Kenneth R Powers
Site Manager

September 19, 1997

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - DOCUMENTS ASSOCIATED WITH DECOMMISSIONING

Consumers Energy Company is currently defueling the Big Rock Point Plant nuclear reactor. In anticipation of being permanently defueled in the near future, the following documents associated with decommissioning have been consolidated and have been included with this letter to facilitate required NRC review, and in some cases, approval. The following documents are included as enclosures this letter:

1. Post Shutdown Decommissioning Activities Report - Revision 1.

2. Plant Radiological Conditions Update

3. Defueled Technical Specifications (requires NRC approval).

4. Defueled Offsite Dose Calculation Manual.

5. Defueled Site Emergency Plan (requires NRC approval).

6. Request for Exemption from 10 CFR 50 Requirements for Emergency Planning (requires NRC approval).

The facility is permanently shutdown (as documented in letters dated June 18 and 26 to the NRC), however it is currently being managed by the same documents used during reactor operation (i.e., the Updated Final Hazards Safety Report [UFHSR], Technical Specifications and Facility Operating License, Site Emergency Plan); and the Decommissioning Plan (a.k.a. the Post Shutdown Decommissioning Activities Report submitted September 1996). It is Consumers Energy Company's desire to phase in the required decommissioning documents following the permanent removal of the nuclear fuel from the reactor vessel to the spent fuel pool.

Depending on NRC review and approval, Consumers Energy Company anticipates the Defueled Site Emergency Plan and associated exemptions will be implemented by

December 1, 1997 (Note: CPC-2A, the Quality Program Description, is in the last stages of review and should be submitted by October 1, 1997). By December 31, 1997, the Defueled Technical Specifications, Facility Operating License, and the Updated Final Hazards Summary Report will be revised and implemented.

Respectfully,

Kenneth P Powers

Kenneth P Powers
General Plant Manager

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point
NRR Project Manager - OWFN

ENCLOSURE(s)

Consumers Energy

A CMS Energy Company

Big Rock Point Nuclear Plant
10269 US-31 North
Charlevoix, MI 49720

Kenneth P Powers
Site Manager

September 19, 1997

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DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - POST SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT; REVISION 1.

Pursuant to 10 CFR 50.82(a)(7), Consumers Energy Company hereby submits Revision 1 to the Post Shutdown Decommissioning Activities Report (PSDAR) for the Big Rock Point Plant. The PSDAR includes a description of the planned decommissioning activities along with a schedule for their accomplishment, an estimate of expected costs, and a discussion that provides the reasons for concluding that the environmental impacts associated with site-specific decommissioning activities will be in compliance with 10 CFR 50.82(a)(6)(ii).

By March 31, 1998, Consumers Energy Company will provide a detailed schedule and revised cost estimate with regard to decommissioning activities. The Company anticipates that the revised cost estimate will not exceed $290.1 million (1994 constant dollars).

Respectfully,

Kenneth P Powers

Kenneth P Powers
General Manager

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point Plant
Project Manager, NRR.

ATTACHMENT

ENCLOSURE 1

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Post Shutdown Decommissioning Activities Report
Revision 1

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BIG ROCK POINT PLANT
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REVISION 1

INTRODUCTION

Under the provisions of 10 CFR 50.82(a)(7), Consumers Energy Company hereby submits Revision 1 to the Post Shutdown Decommissioning Activities Report (PSDAR) to describe planned decommissioning activities, a schedule for their accomplishment, estimate expected costs, and provide the reasons for concluding that the environmental impacts associated with site-specific decommissioning activities will be in compliance with 10 CFR 50.82(a)(6)(ii).

BACKGROUND

When Consumers Energy Company's Big Rock Point Plant began operation in September 1962, it was the first commercial nuclear power plant constructed in Michigan and the fifth in the United States. The General Electric Boiling Water Reactor (BWR) was rated for 240 Megawatt (MW) Thermal, and was built by Bechtel Corporation. By letters dated June 18, 1997, and June 26, 1997, Consumers Energy Company notified the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.82(a)(1)(i), that Big Rock Point Plant would permanently cease operation on August 30, 1997. On August 29, 1997, the reactor was permanently shutdown, ending 35 years of electric power generation as the nation's oldest and longest running nuclear plant. It was closed because its relatively small size (67MW Electric) was likely to make it too expensive to operate in an increasingly competitive environment.

Consumers Energy Company's goal is to immediately dismantle Big Rock Point Plant in a safe, environmentally conscious, and cost effective manner. This action will result in the timely removal of the existing nuclear plant in accordance with the DECON option found acceptable to the NRC in its Final Generic Environmental Impact Statement (FGEIS) [Reference 1]. Completion of this option is contingent upon continued access to one or more low level waste disposal sites. Currently, Consumers Energy Company has access to Chem Nuclear - Barnwell, South Carolina and Envirocare - South Clive, Utah.

DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

Decommissioning Activities and Planning

The activities planned for decommissioning of the Big Rock Point Plant reflect the DECON option for the site. Work plans will be completed for decommissioning activities prior to commencing the activity. Figure 1 shows a summary decommissioning schedule for Big Rock Point Plant. This schedule begins with the initial announcement to permanently cease plant operations on June 11, 1997.

Figure 2 shows a preliminary timeline of the significant decommissioning activities.

Planning Activities

Subsequent to the Consumers Energy Company June 18, 1997 notification to the NRC of plans to permanently shutdown the Big Rock Point Plant, a site organization was developed to decommission the plant. This organization has become effective September 15, 1997. Revisions to the site Emergency Plan, Security Plan, Technical Specifications, Offsite Dose Calculation Manual, UFHSR and Quality Program Description are in various stages of development and will be submitted to the NRC.

Continuing planning and preparation for decommissioning includes the following generalized types of tasks:

- Review existing plant programs to assess their applicability to decommissioning,
- Review and reclassify systems important to decommissioning activities,
- Revise procedures and license basis documents to reflect the plant's permanently defueled configuration,
- Initiate radiological and hazardous material characterization of the site,
- Design and procure equipment and facilities to support decommissioning activities,
- Identify specific decommissioning activities,
- Prepare work plans for decommissioning activities,
- Prepare dose estimates for decommissioning activities,
- Evaluate disposition options for site components and structures,
- Develop a cost measurement and control mechanism, and
- Develop an activity schedule consistent with the overall schedule.

A key step in the decommissioning planning was the selection of a project staff and establishment of an organizational structure. This step mobilized site management and staff personnel augmented with on-site specialty contractors.

Plant Dismantlement

Decommissioning planning is based on selecting the DECON option and is expected to result in the complete dismantlement and restoration of the site. The facilities remaining to support dry storage of the fuel will be decontaminated and/or dismantled after the spent fuel has been received by DOE.

The following activities are anticipated to occur during the dismantlement period:

- Perform primary system decontamination,
- Establish a site construction power system,
- Remove asbestos insulation in conjunction with plant piping systems,
- Remove turbine control oil,
- Establish a spent fuel pool cooling system independent of existing plant systems,
- Construct an Independent Spent Fuel Storage Installation (ISFSI) for dry cask storage,
- Establish a monitoring location allowing for the deactivation and dismantlement of the plant control room,
- Dismantle systems, structures and components not required for the safe storage of spent fuel, including major component removal,
- Conduct decontamination of facility surfaces, components and piping surfaces as required,
- Conduct soil remediation as necessary,
- Ship and properly disposition all radioactive materials,
- Perform a comprehensive final status survey to demonstrate compliance with approved site release criteria (10 CFR 20, subpart E).

The structures and facilities remaining after plant dismantlement and site restoration will be to support the dry storage of the spent fuel.

SIGNIFICANT DECOMMISSIONING ACTIVITIES

10 CFR 50.2 defines major decommissioning activities as those that result in permanent removal of major radioactive components (e.g. reactor vessel and internals, large bore reactor coolant system piping, and other large components that are radioactive to a comparable degree), permanently modifies the structure of the containment, or results in dismantling components for shipment containing greater than class C waste.

The following discusses several planned significant decommissioning activities at Big Rock Point Plant:

Reactor Vessel

The reactor vessel was fabricated from carbon steel with an internal stainless steel cladding. The entira outside of the reactor vessel is insulated with 3-inch thick metallic insulation. It is attached to the reactor vessel by banding and is supported by brackets welded to the outside surface of the reactor vessel. The reactor vessel is supported by 12 brackets attached to the exterior vessel shell. Twenty four 2-1/2-inch diameter hanger rods attached to these brackets transfer the re: :tor vessel weight to supports anchored in the surrounding concrete.

During power operations neutron irradiation from the fission process generated activation products in the stainless steel vessel liner, the carbon steel vessel shell, and metallic insulation. The radionuclide inventory for the reactor vessel as a unitized package is expected to be a Type B quantity, meeting the Low Specific Activity (LSA) material criteria. As a result, the reactor vessel as a unitized package would be exempt from the requirements of 10 CFR 71.73, "Hypothetical Accident Conditions." The radionuclide content estimates will be verified with a radiation survey of the reactor vessel after the internal components have been removed. Detailed classification evaluations will be completed as a part of detailed planning for the reactor vessel removal activity.

An engineering evaluation was performed to investigate potential reactor vessel removal alternatives. The evaluation identified two technically feasible alternatives: intact vessel removal and segmented vessel removal. The intact vessel removal alternative proposes that the reactor vessel be shipped to a licensed low radioactive waste disposal facility as one piece. The segmented vessel removal option proposes shipment of reactor vessel sections to a low level radioactive waste disposal facility utilizing multiple shipments in approved shipping containers.

Currently evaluations are being conducted to determine which method will be utilized.

Steam Drum

The steam drum is part of the Nuclear Steam Supply System and is located in the reactor building. It is a 40-foot 9-inch long by 7-foot 2-inch diameter, horizontal steel cylinder with ellipsoidal heads. The steam drum contains 60 steam separators arranged in two equal rows. The steam drum's vessel wall is 4-3/8 inches thick A-212B carbon steel with 5/32" Type 304 stainless steel cladding. The nozzles are primarily ASTM A-105 Grade II carbon steel with : 5/32-inch Type-304 stainless steel clad. Nozzles smaller than 4 inches are made from SB-166 Inconel material. Internal components are made from Type 304 stainless steel.

The steam drum may be chemically decontaminated as part of the Nuclear Steam Supply System, then drained and isolated until ready for dismantlement in accordance with general decommissioning activities.

Primary Coolant System

Primary coolant system piping connects all major components of the Nuclear Steam Supply System.

The primary coolant system will be chemically decontaminated, drained and isolated until ready for dismantlement in accordance with general decommissioning activities.

Containment Vessel

The containment vessel is a 130-foot diameter, Hortonsphere steel vessel. The sphere extends 27 feet below grade and 103 feet above grade. The containment vessel is constructed of 3/4-inch (nominal) steel plates. The exterior columns are non-load carrying members which were used during construction and were unloaded after construction and testing of the containment vessel, and now remain only as attachments. The foundation of the containment vessel is a reinforced concrete cradle in the shape of an inverted spherical dome segment approximately 7 feet thick.

The containment surfaces and structures will be decontaminated and dismantled in accordance with general decommissioning activities.

Spent Fuel Pool

The Spent Fuel Pool is described in Section 9.1 of the Updated Final Hazards Summary Report (UFHSR).

Once the pent fuel pool has been emptied it will be decontaminated and dismantled in accordance with general decommissioning activities.

OTHER DECOMMISSIONING CONSIDERATIONS

The decontamination and/or dismantlement of contaminated systems, structures and components may be accomplished by decontamination in place, dismantlement and decontamination, or dismantlement and disposal. A combination of these methods may be utilized to reduce contamination levels, worker radiation exposure and project costs. General considerations applicable to these activities are described below.

Chemical Decontamination of the Primary Coolant System

A chemical decontamination of the primary coolant system will be performed prior to conducting any major decommissioning activity. The chemical decontamination is a significant ALARA initiative being performed to reduce personnel exposure during decommissioning work activities. This decontamination effort is expected to include the reactor vessel and steam drum, steam risers and recirc[illegible] piping, shutdown cooling system, and the cleanup system. The existing [illegible] recirculating water pumps are expected to be used to circulate the [illegible] solution throughout the [illegible] and selected interconnected syste[illegible] [illegible]ifications will be require[illegible] [illegible]tablish specific flow paths and isol[illegible] [illegible]ts. This decontaminatio[illegible] [illegible] is expected to be performed by [illegible] contractor following a[illegible] [illegible]ite specific controls.

General Decommissioning Activities Relating to Removal of Radiological Components & Structures

Components will be safely and efficiently removed using the most appropriate methods for the particular circumstance. Work packages will be prepared for activities related to the dismantlement of plant systems, structures and components. Openings in components will typically be covered to prevent the spread of contamination. Components may be moved to a processing area for volume reduction and packaging into containers for shipment to a waste disposal site or a processing facility for decontamination.

Following are several general decontamination and dismantlement considerations that will be incorporated into the decommissioning work packages:

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

- Radioactive particulate emission will be monitored in accordance with the Offsite Dose Calculation Manual (ODCM).

- Decommissioning activities that use liquids will ensure that the contaminated liquids will be processed and sampled and monitored before release. In addition, existing or supplemental barriers should be used to ensure that inadvertent spills from these activities are contained. A program similar to the existing Spill Prevention Control and Countermeasures Pollution Incident and Prevention Plan (SPCC/PIPP) will be implemented.

- Non-radioactive hazardous materials and wastes will be dispositioned in accordance with Consumers Energy Company Waste Management Program. Typical materials handled and disposed of through this program include fuel oil, lubricating oil, 1,1,1-trichloroethane, laboratory chemicals, lead, mercury, paints, battery acid, and asbestos containing materials. Considerations include the following:

  - Materials containing asbestos (e.g. insulation) will be removed and processed in accordance with this program.

  - The decontamination and dismantlement methods to be used on systems, structures, and components which contained or were immersed in chromated solutions will be evaluated and the methods selected to minimize the potential for creating a mixed waste.

  - Instrumentation and control components will be evaluated, and the dismantlement method will be selected to minimize the potential for creating a hazardous waste. Switching elements which contain mercury should be removed from the instrument when practicable.

- Contaminated systems, structures and components with significant external contamination, will be decontaminated to remove the loose external contamination, painted to stabilize the contamination, bagged to prohibit contamination spread, or otherwise controlled to prevent personnel or plant contamination during removal.

- Contaminated piping and tubing should be removed as follows:

  - Piping will be cut using methods which minimize the generation of airborne contamination. When appropriate, remote cutting systems may be used to maintain worker exposure ALARA.

  - Protective covers or plugs may be installed on ends of contaminated piping to confine internal contamination.

  - Piping penetrations will be cut as close as practicable to the containment vessel shell. The openings in the containment vessel will be covered or plugged once the piping is removed.

  - Underground piping identified for removal will be evaluated prior to cutting and removal to identify a method appropriate to the physical condition of the pipe.

- Contaminated supports will be removed in conjunction with the equipment removal activities.

- Systems and components may be removed from areas and buildings prior to the start of structural decontamination activities. Walls may be removed as required to permit removal of components.

- Embedded contaminated piping, conduits, ducts, plate, channels, anchors, sumps and sleeves may be removed or decontaminated during area and building structural decontamination activities.

- Centralized processing and cutting stations will be considered to facilitate packaging of components for shipment to an off-site processing facility or to a low level radioactive waste disposal facility.

- Equipment designated for asset recovery or re-use in the Consumers Energy Company system may be preserved in accordance with vendor recommendations or Consumers Energy Company practices.

Special or Unusual Programs

There are no special or unusual programs planned for use. All procedures and processes to be applied at Big Rock Point Plant are consistent with those discussed in the Final Generic Environmental Impact Statement on Decommissioning (NUREG-0586).

Low Level Radioactive Waste Removal and Handling

Low level radioactive waste will be handled in accordance with plant procedures, then shipped either to licensed offsite processors for further processing such as decontamination for free release, metal melt, incineration, or shipped for disposal at licensed facilities. No onsite incineration will be performed.

Soil Remediation

Soils, concrete rubble from demolition of structures, rubblized paving and other soil-like materials will be surveyed to determine if residual radioactivity of plant origin is present, and if so, the radioactivity concentrations present. Such soils and soil-like materials will be remediated (i.e., removed, processed and disposed of at a licensed facility) if determined to contain levels above the NRC site release guideline values of 10 CFR 20, subpart E.

Processing and Disposal Site Locations

A number of facilities are available for processing of radioactive waste materials. These facilities provide services which include, but are not limited to, decontamination, incineration, metal melt, compaction or other methods of consolidation, and disposal of low level radioactive wastes. A partial list of such facilities includes: Chem Nuclear, Barnwell, SC; Envirocare, South Clive, UT; Hake, Memphis, TN; Scientific Ecology Group, Oak Ridge, TN; US Ecology, Oak Ridge, TN.

Removal of Mixed Wastes

All applicable regulations of state and federal authorities will be followed in the handling, storage and transport of mixed wastes. Transport will only be by authorized licensed transporters and shipment will be only to licensed facilities. If technology, resources and approved processes are available to render mixed waste non-hazardous, such processes may be considered to minimize the hazards of transport and disposal of the wastes.

Spent Fuel and Greater Than Class C Waste

Spent fuel currently is planned to be stored wet in the spent fuel pool until dry transportable storage canisters are available, and fuel has decayed sufficiently to meet license conditions of the canisters. Loading of dry storage canisters is planned to begin in late 1999 or early 2000, with completion of fuel pool offload in late 2000 to early 2001, provided the canister license allows loading with three years or less of decay. Current plans call for an onsite Independent Spent Fuel Storage Installation (ISFSI) which will accommodate all current spent fuel in seven storage casks, each containing two canisters. Fuel is expected to be retained until Department of Energy (DOE) fulfills their obligation to receive the fuel.

Greater than Class C (GTCC) wastes are comprised of reactor internals exposed to many years of neutron flux. The method of disposal which would result in lowest cost and lowest dose to workers is shipment within the intact reactor vessel, provided that authorization to average the radionuclide inventory throughout the metal mass of vessel and contents (such that the total package is not GTCC). The less advantageous option would be to cut internals into dimensions suitable for storage within the dry storage canisters for transfer to DOE as GTCC waste. This option is allowed by storage canister design, but there currently is no DOE rate schedule for GTCC, or confirmation that the ultimate fuel storage facility will accept GTCC wastes.

SITE RESTORATION

During the process of dismantlement and decontamination to greenfield (DECON status, plant materials will be surveyed to determine whether such materials:

1) are uncontaminated and may be free released,

2) retain traces of detectable radioactivity, but at levels below NRC site release criteria, in which case the materials either will a) remain on site, b) be decontaminated for free release, or c) be shipped to licensed vendor facilities for offsite processing such as metal melt, incineration, or further decontamination, or

3) retain significant levels of radioactivity, in which case these materials will be shipped to licensed facilities for processing or disposal.

A final survey to confirm that the site, (with the exception of approximately a one acre dry ISFSI), meets NRC release criteria will be performed on the greenfield site prior to application of topsoil and vegetative plantings while original soil and rubblized materials are readily accessible at and near the surface for *In Situ* gamma spectral analysis. Successful completion of the final survey will allow license termination with the site released for unrestricted use. A similar process will be performed on the ISFSI site following final shipment of fuel to DOE at a later date.

ENVIRONMENTAL IMPACTS

Big Rock Point Plant has performed a review of the site and evaluated the potential impacts of the proposed decommissioning activities. The review concludes that impacts due to decommissioning of Big Rock Point Plant will be in compliance with 10 CFR 50.82(a)(6)(ii). This conclusion is reached on the basis of the following:

- The DECON method of decommissioning currently chosen for Big Rock Point Plant, as well as the SAFSTOR option which has also been thoroughly studied for this site, are fully addressed by the FGEIS.

- There are no unique aspects of the decommissioning techniques to be utilized that would invalidate the conclusions reached in the FGEIS.

- Big Rock Point Plant is significantly smaller and contains a radioactive source term which is only on the order of 10% that of the standard BWR addressed by the FGEIS. As such, Big Rock Point Plant provides lower impacts for potential radiological accidents. However, due to Big Rock Point Plant's lack of high efficiency particulate activity (HEPA) filters in the original design of reactor building exhaust, the plant now is in the process of installing HEPA filters for use in the unlikely event potential accidents such as those addressed by the FGEIS could occur. This installation of HEPA filters brings the plant fully within the bounds of the FGEIS.

- Worker doses projected for the decommissioning of Big Rock Point Plant have been compared on a task-by-task basis with the FGEIS. Due to smaller size and lower radioactivity source term, doses are projected to be well under the 1845 person-rem identified for the reference BWR by the FGEIS.

- Doses to the public will not exceed those estimated by the FGEIS for the reference BWR.

- No site specific factors at Big Rock Point Plant would alter the conclusions of the FGEIS.

The total occupational radiation exposure expected for the decommissioning interval has been estimated at 425 person-rem. This is not a conservative estimate, but is rather a goal based on techniques of maintaining plant doses as low as reasonably achievable (ALARA) which have been demonstrated during the final years of plant operating life and at other nuclear plants in the process of decommissioning now and in the recent past. This number may be higher, for example, if full system chemical decontamination is less effective than assumed, or if reactor internals cannot be shipped in the reactor vessel with minimal handling and worker exposure. However, in no event is dose expected to exceed the value of 1845 person-rem estimate of the FGEIS.

No significant impacts are expected from the disposal of radioactive waste. Total volume of waste projected for Big Rock Point Plant decommissioning is 72,100 cubic feet, in comparison to the FGEIS volume for the reference BWR of 662,500 cubic feet, including disposable containers.

Radiation exposure due to transportation of radioactive waste will be well below (on the order of 10 to 20% based on waste volume and activity ratios) the 110 person-rem for transport workers and 10 person-rem for the public presented by the FGEIS. Number of shipments (and therefore, doses as well as accident probabilities) also will be less than 20% of that the larger reference plant, based on waste volume ratios. In addition, plant experience shows that transport vehicle dose rates seldom approach the levels assumed by the FGEIS analysis in the calculation of transport worker and public doses.

Radiation exposure to offsite individuals due to postulated accidents are bounded by the FGEIS analysis for non-fuel related events, and by EPA Protective Action Guides [Reference 3] for these, as well as fuel related events. Effluents release levels, and public dose due to effluents will decrease below the low levels observed during plant operation, due to lack of

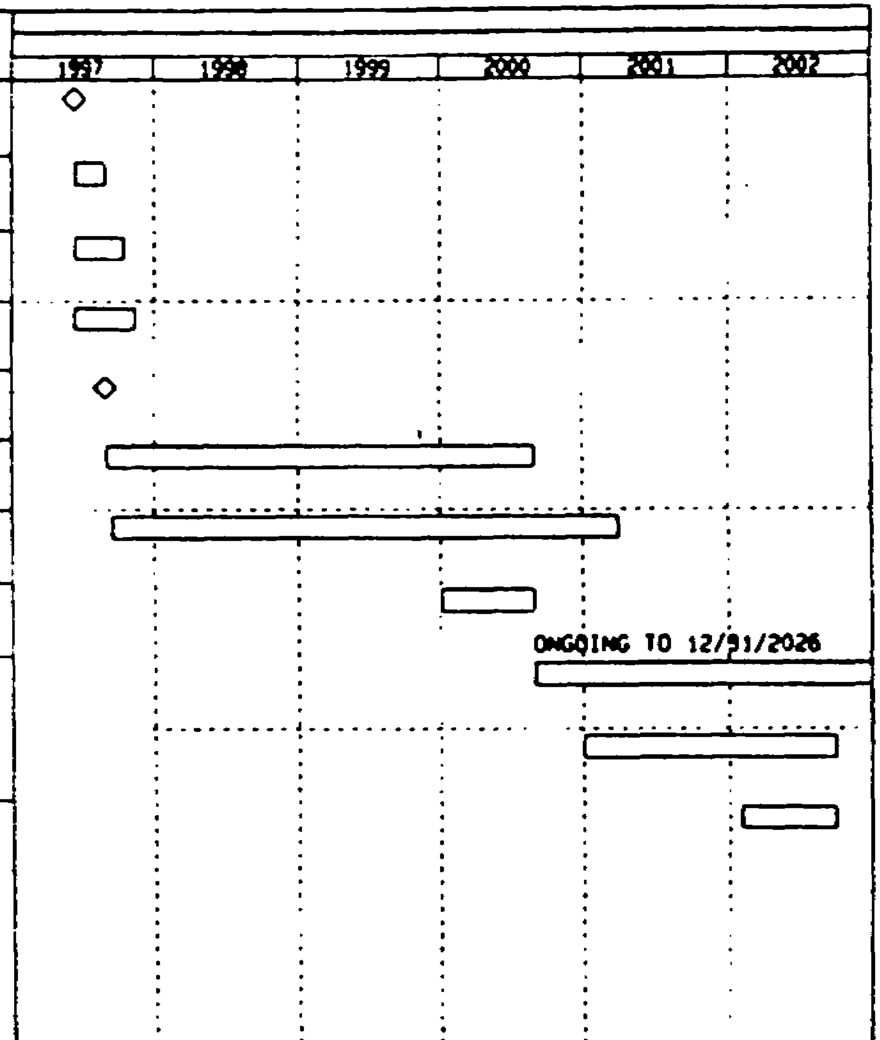
radionuclide production with the reactor defueled, and decay of the radionuclide inventory over time.

Disposal of low level radioactive wastes at licensed disposal facilities is expected to be possible in a timely manner. However, should temporary storage be required, adequate onsite storage space is available in one or more, or a combination of the Big Rock Point Plant radioactive waste facility, the turbine and reactor buildings, and an onsite contaminated materials warehouse. No significant environmental impacts are anticipated due to temporary onsite storage. Such storage will be in compliance with all applicable federal and state regulations.

Non-radiological environmental impacts from decommissioning the Big Rock Point Plant will be minor short term increases in noise, dust and truck traffic flow in the immediate vicinity of the plant site during dismantlement. An increased risk of industrial accidents is recognized. Additional safety professionals are being added to the decommissioning staff, and safety programs are receiving added emphasis in acknowledgement of this risk. The only significant socioeconomic impacts identified are those of local job loss and lowered tax base. No detrimental impacts to local culture, terrestrial or aquatic resources ave been identified. Although future uses of the plant site has not yet been determined, the property provides potential for a wide variety of beneficial uses. The chosen greenfield (DECON) option, coupled with unrestricted release compliance, will ensure that future uses are not limited by final site condition.

Figure 1

ACTIVITY DESCRIPTION	EARLY START	EARLY FINISH
PLANT DECISION	11JUN97	
PLANT OPERATIONS	11JUN97	29AUG97
DETAILED PLANNING	12JUN97	15OCT97
TRANSITION ACTIVITIES	12JUN97	14NOV97
END OPERATIONS		29AUG97
FUEL POOL OPERATIONS	1SEP97	31AUG00
EQUIPMENT DISMANTLEMENT	15SEP97	30MAR01
DRY FUEL CASK LOADING	3JAN00	31AUG00
DRY FUEL STORAGE (ESTIMATED)	1SEP00	31DEC26
MAJOR BUILDING DEMOLITION	1JAN01	30SEP02
COMPLETE LAND RESTORATION	1FEB02	30SEP02



CONSUMERS ENERGY
BIG ROCK POINT NUCLEAR PLANT
SUMMARY DECOMMISSIONING SCHEDULE

Figure 2

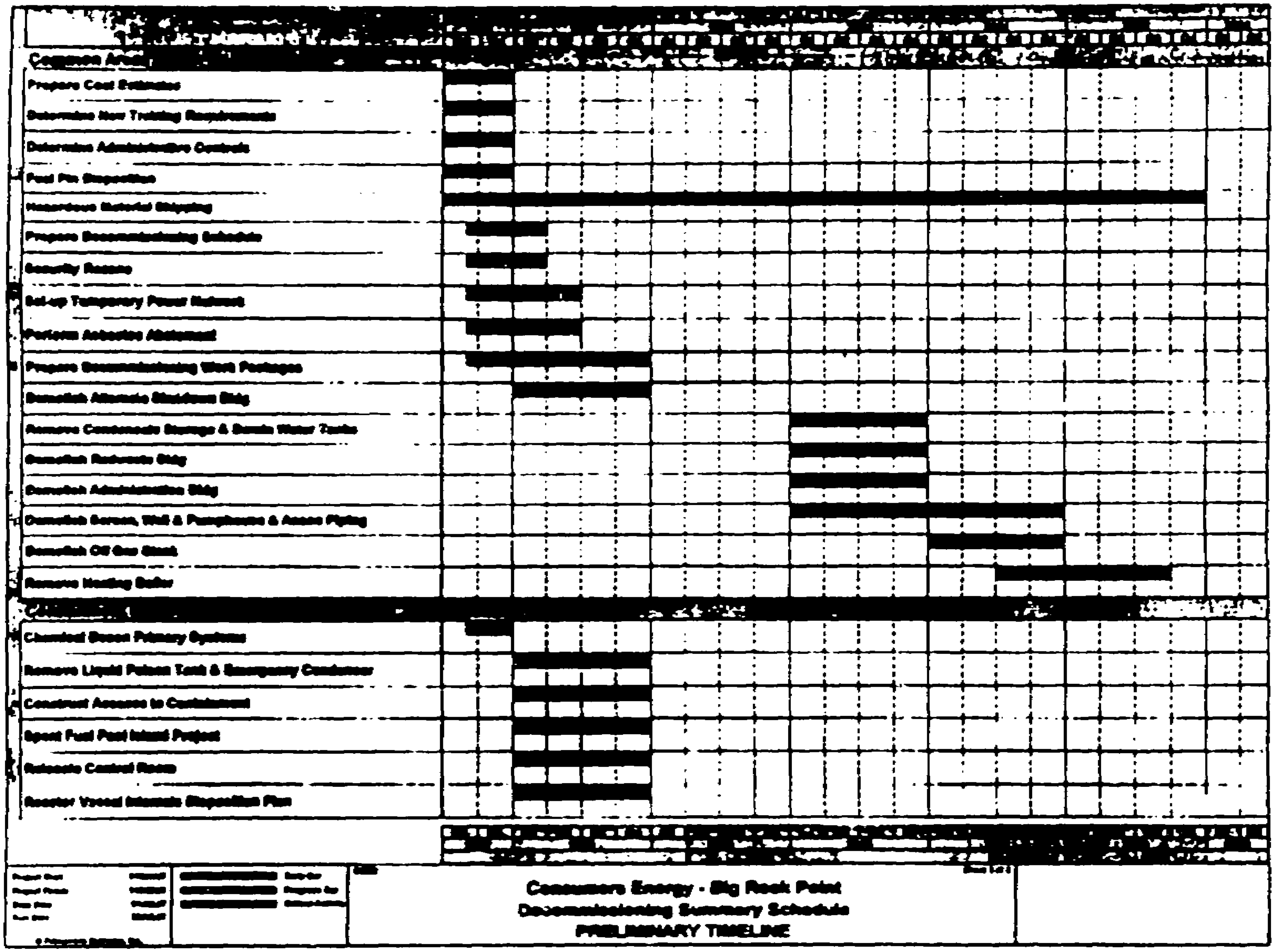
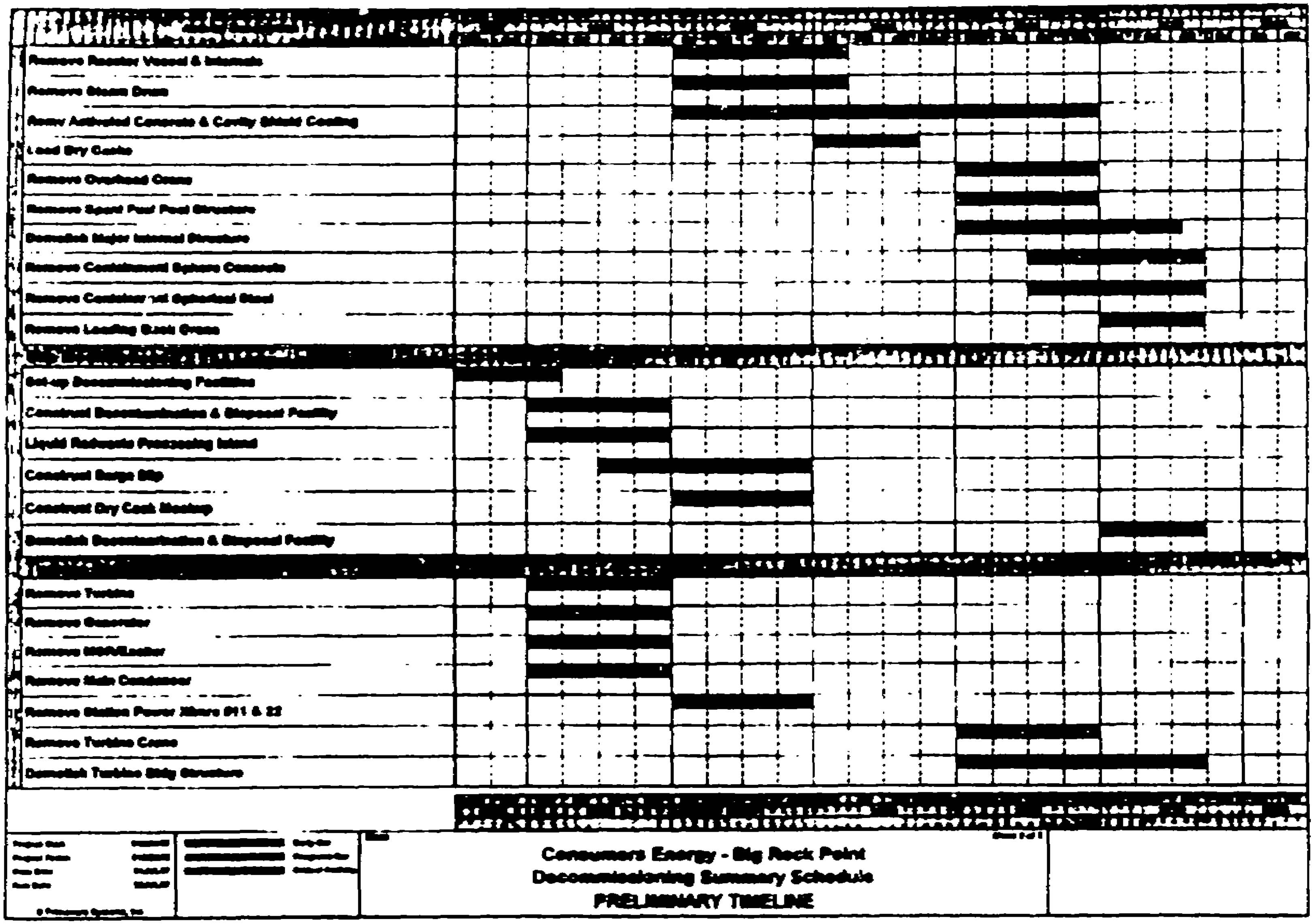

Common Area
Prepare Cost Estimates
Determine New Training Requirements
Determine Administrative Controls
Fuel Pin Disposition
Hazardous Material Shipping
Prepare Decommissioning Schedule
Security Reasons
Set-up Temporary Power Network
Perform Asbestos Abatement
Prepare Decommissioning Work Packages
Demolish Alternate Shutdown Bldg
Remove Condensate Storage & Demin Water Tanks
Demolish Radwaste Bldg
Demolish Administration Bldg
Demolish Screen, Well & Pumphouse & Assoc Piping
Demolish Off Gas Stack
Remove Heating Boiler
Chemical Decon Primary Systems
Remove Liquid Poison Tank & Emergency Condenser
Construct Access to Containment
Spent Fuel Pool Island Project
Relocate Control Room
Reactor Vessel Internals Disposition Plan
Consumers Energy - Big Rock Point
Decommissioning Summary Schedule
PRELIMINARY TIMELINE

Figure 2


Remove Reactor Vessel & Internals
Remove Steam Drum
Load Dry Casks
Remove Overhead Crane
Remove Spent Fuel Pool Structure
Demolish Major Internal Structure
Remove Containment Sphere Concrete
Set-up Decommissioning Facilities
Construct Decontamination & Disposal Facility
Liquid Radwaste Processing Island
Construct Barge Slip
Demolish Decontamination & Disposal Facility
Remove Turbine
Remove Generator
Remove Main Condenser
Remove Turbine Crane
Demolish Turbine Bldg Structure
Consumers Energy - Big Rock Point
Decommissioning Summary Schedule
PRELIMINARY TIMELINE

ESTIMATE OF EXPECTED DECOMMISSIONING COSTS

On March 1, 1995, Consumers Energy Company (Consumers Power Company) filed a site specific decommissioning cost estimate with the Michigan Public Service Commission. On April 10, 1996, the Michigan Public Service Commission issued the Decommissioning Surcharge Order authorizing Consumers Energy Company to collect $290.1 million in 1994 constant dollars. This estimate was based on a 27 year safe storage period followed by dismantlement to the "greenfield" (DECON) condition to be completed in about year 2030.

Consumers Energy Company will be updating the decommissioning cost estimate and submit the results in March 1998, to reflect the immediate dismantlement "greenfield" (DECON) option. It is anticipated that the cost update will not exceed the $290.1 million estimate.

Consumers Energy Company is currently authorized to collect approximately $25 million per year through December 2000 to decommission the Big Rock Point Plant. The Decommissioning Trust Fund market value through August 1997 was approximately $185 million. Therefore, the remaining fees to be collected through the year 2000 plus interest earned on the fund investment should insure that sufficient funds are available to decommission the Big Rock Point Plant site.

REFERENCES

1. NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities", August, 1988.

2. Consumers Energy Company, "Updated Final Hazards Summary Report (UFHSR), Big Rock Point Plant", Revision 6, October 7, 1996.

3. EPA 402-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", May, 1992.

ENCLOSURE 2

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Update on Plant Radiological Conditions

UPDATE ON PLANT RADIOLOGICAL CONDITIONS

The NRC summary of our August 25, 1995 meeting (NRC letter dated September 8, 1995) documents the Big Rock Point commitment to provide information on any changes in radiological conditions which have occurred in the interval between Decommissioning Plan submittal and final plant shutdown. The following is provided as an update to plant radiological status:

1) Gravel was removed from the turbine building roof in the course of re-roofing the building in June of 1995. Gravel samples identified the presence of low, but detectable concentrations of Cs-137 (0.3 pC/g above background) and Co-60 (0.066 pCi/g above background). Source of the activity is believed to be washout of routine stack release over the 35 year operating history of the plant. The gravel remains onsite, covering a triangular area over portions of grid sectors 2S4E, 2S5E and 1S5E (grids are pictured in Decommissioning Plan Figure 3.1-1). The concentrations of radionuclides present represent a small fraction (<10%) of the levels which would give 25 mrem/y at license termination.

2) On March 3, 1997, approximately nine gallons of radioactive water from decontamination activities were sent in error to the Caustic Tank floor drain. The Caustic Tank drain had been plugged to prohibit use due to its direct path to the Discharge Canal. Although it was quickly ascertained that the drain was plugged, some water level decrease was observed in the puddle around the drain. An estimated loss of 0.9 gallons of water and 0.15 μCi of activity was calculated from the decrease in puddle size and the radionuclide concentration in the sampled liquid. Investigation showed that the drain bowl had corroded through, which allowed liquid to seep into the soil below the building slab. Excavation of the soil around the plugged drain pipe recovered 0.25 μCi of activity, which was slightly more than the release estimate. Determination of the need for any further decontamination of soils in that area will await removal of equipment from that area during the process of decommissioning, since there currently is limited space to work in this area.

3) Continued sampling of the groundwater monitoring wells indicates that one sample point (Well #6) remains slightly above the EPA drinking water limit of 2.0 pCi/ml for tritium (7 sample mean is 2.98 ± 1.3 pCi/ml) and one sample point (well #5) remains just below the limit at 1.92 ± 0.58 pCi/ml. All other wells remain below 0.1 pCi/g tritium. No gamma emitting radionuclides have been detected in any of the wells. Levels of tritium have not decreased significantly from 1994 possibly because Lake Michigan water level has been increasing with the result that the net hydrostatic gradient to the lake has decreased significantly during this interval.

4) Section 3.1.4.1 of the Decommissioning Plan mentions radioactivity which is localized at the termination of a plant surface water storm drain at the ditch along the west boundary of the protected area fence (within the owner controlled area). All radioactive sediment of concentrations above site release criteria were removed at the time of discovery. Further investigation of

potential sources of this radioactive material indicates the likely source of this radioactivity is either a very small leak, or leaching from tank insulation contaminated from previous manway leakage, at the North Waste Hold Tank. A collection basin (steel drum) placed at the storm drain outfall has been successful in collecting sediments carried by storm runoff through the drain. Water sampling has showed no detectable activity, and sediment sampling of the ditch itself has shown no net increases in radioactivity within the ditch. Plans were being made to replace the Waste Hold Tanks at the time the decision was made to undergo early plant decommissioning. However, since the Waste Hold Tanks should not be necessary for holding decommissioning wastes, they now are expected to be isolated and removed rather than replaced.

5) Underwater dose rate profiling was performed for a large sample (more than 50%) of our fuel channels which had been conservatively estimated as Greater than Class C (GTCC) waste. Calculations based on the profiling data show that the channels are Class C rather than GTCC. This reduces our fuel pool inventory (Decommissioning Plan Table 2.3-6) of GTCC waste from 224 cubic feet to approximately 59 cubic feet. The neutron activation calculations upon which the estimate in Table 2.3-6 had been based, had utilized conservative assumptions involving neutron fluence parameters. In contrast, the profiling calculations were based on actual burnup of the fuel bundles contained within the profiled channels.

Calculations have been performed to determine soil radionuclide concentration limits for 25 mrem/y at license termination. Site specific parameters were utilized as inputs to RESRAD as previously described in the Decommissioning Plan and subsequent correspondence, but for a 5-year dismantlement case rather than 27-year SAFSTOR. It has been determined that no known facility soils exceed the limiting concentrations for the 5-year case.

ENCLOSURE 3

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Defueled Technical Specifications

Consumers Energy

A CMS Energy Company

Big Rock Point Nuclear Plant
10269 US 31 North
Charlevoix, MI 49720

Kenneth P Powers
Site Manager

September 19, 1997

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555 - 0001

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGE REQUEST - DEFUELED TECHNICAL SPECIFICATIONS

A request for change to the Big Rock Point Facility Operating License and Technical Specifications is enclosed.

On August 29, 1997, Consumers Energy Company permanently ceased operation of the Big Rock Point Nuclear Plant. In accordance with 10 CFR 50.36, Technical Specifications, paragraph (c)(6) Decommissioning, *"Technical Specifications involving safety limits, limiting safety system settings and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis"*, the Big Rock Point staff has re-developed the Facility's Operating License and Technical Specfications to manage the decommissioning of the facility. This document supercedes the current Facility Operating License and Technical Specifications.

In accordance with 10 CFR 50.92, Consumers Power Company has made a determination that the proposed amendment involves no significant hazards considerations.

Kenneth P Powers

Kenneth P Powers
General Plant Manager

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point
NRR Project Manager - USNRC, OWFN.

ATTACHMENT

Request for Change to the Facility Operating License and Technical Specifications
License DPR-6

For the reasons hereinafter set forth, it is requested that the Facility Operating License DPR-6, Docket 50-155, issued to Consumers Energy Company on May 1, 1964, for the Big Rock Point Plant and the Technical Specifications contained therein be changed as described in Section I below:

This document supersedes the current Facility Operating License and Technical Specifications.

I. CHANGES

The entire Facility Operating License and Technical Specifications have been changed to address the permanently shutdown reactor and planned decommissioning activities. A change matrix, although not included in this submittal, is available from the Big Rock Point Business, Regulatory and Services department for review.

II. DISCUSSION

On August 29, 1997, Consumers Energy Company permanently ceased operation of the Big Rock Point Nuclear Plant. In accordance with 10 CFR 50.36, Technical Specifications, paragraph (c)(6) Decommissioning, "*Technical Specifications involving safety limits, limiting safety system settings and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis*", the Big Rock Point staff has re-developed the facility's operating license and technical specifications to manage the decommissioning of the facility.

III. ANALYSIS OF NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with 10 CFR 50.92, Consumers Energy Company has made a determination that the proposed amendment involves no significant hazards considerations. To make this determination, Consumers Energy Company has established that operation of the permanently defueled Big Rock Point Nuclear Plant in accordance with the new amendment will not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The proposed change provides the applicable requirements to assure safe storage of spent nuclear fuel during decommissioning following the permanent cessation of nuclear plant operations at Big Rock Point Nuclear Plant. Decommissioning activities conducted using these controls do not present an undue risk to the public, and do not impact common defense and security.

The proposed change does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change must be assessed in light of the declaration that operations have permanently ceased at the Big Rock Point Nuclear Plant effective August 29, 1997 and that all fuel will be permanently removed from the reactor core by mid-September 1997.

No accidents previously evaluated in the Updated Final Hazards Summary Report (UFHSR) will have their probability of occurrence increased because the proposed controls effectively preclude the occurrence of a criticality, fuel temperature exceeding limits or fuel handling accidents. The probability of plant accidents associated with power operations have been significantly reduced. Accidents associated with spent fuel handling, including cask and single bundle drop and spent fuel pool cooling capability loss events are still pertinent and were reviewed using new data on pool inventory and revised 10 CFR 20 radiological limit determinations. The probability of occurrence of accidents associated with storing 441 spent fuel assemblies in the spent fuel pool (current license limit) have not been affected by the changes in the proposed Technical Specification.

The consequences of fuel handling, and cask drop accidents were evaluated based on the removal of all fuel from the reactor and loading spent nuclear fuel in the spent fuel pool. The removal of all fuel from the reactor vessel to storage in the spent fuel pool and the subsequent decay of the fuel in the pool result in no increase in the probability of these accidents and continuously reduced consequences from these accidents.

Analyses using the techniques in Branch Technical Position APCSB 9-2 provide the heat rate from a freshly-removed full core off-load in the spent fuel pool whose racks are filled with a total of 441 fuel assemblies as the most limiting cooling condition. Existing cooling equipment under the current Technical Specifications provide sufficient cooling to preclude spent fuel pool temperatures reaching 150 degrees-Fahrenheit. Analysis concludes that once the fuel has decayed for 53 days following power operation, the pool water bulk temperature will not rise above 150 degrees-Fahrenheit with a complete loss of spent fuel pool cooling for 72 hours. This precludes entry into an unanalyzed condition for the spent fuel pool and provides 3 days to recover cooling flow of "approximately 30" gpm. Since this specification change is intended for implementation following 93 days after shutdown (approximately November 30, 1997), this analysis justifies the allowance of 24 hours to re-establish cooling flow provided in specification 3.1.2.

(2) Create the possibility of a new or different kinds of accident from any accident previously evaluated.

For operation of the Big Rock Point Nuclear Plant, accident analyses were completed and documented for the following categories of accidents and transients:

Increase in heat removal by the secondary system *
Increase in reactor coolant inventory *
Decrease in heat removal by the secondary system *
Decrease in reactor cooling inventory *

Radioactive release from a subsystem or component
Reactivity and power distribution anomalies *
Anticipated transients without scram *
Single loop operation *

The permanent cessation of operation and removal of fuel from the reactor eliminates all probability of those accidents identified with asterisks (*) above to create a hazard to the health and safety of the public. These revised Technical Specifications, in combination with requirements in the UFHSR, provide assurance that fuel handling and spent fuel cask drop accidents, which represent the remaining specific pertinent accidents analyzed in the *"Radioactive release from a subsystem or component"* category will not occur. Because the revised Technical Specifications related to fuel handling, spent nuclear fuel storage, and handling of the spent fuel cask satisfy current license and UFHSR requirements, no new accidents are created.

(3) Involve a signification reduction in a margin of safety.

The safety margins for analyzed accidents are maintained because the containment structures and redundant control established by the plant remain in place until the decay of spent nuclear fuel has reduced the source term to levels that analysis confirms do not required the containment features. 93 days after permanent cessation of operations, the spent nuclear fuel at Big Rock Point will have decayed to the point where the added margin from this decay more than compensates for the removal of the containment as a safety feature, and allows relaxed controls for the cooling of the spent fuel pool.

IV. CONCLUSION

The Big Rock Point Plant Review Committee has reviewed this Facility Operating License and Technical Specification Change Request and has determined this change does not involve an unreviewed safety question and, therefore, involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department. A copy of this change request has been sent to the Stage of Michigan official designated to receive such Amendments to the Operating License.

CONSUMERS ENERGY COMPANY

To the best of my knowledge, information and belief, the contents of this submittal are truthful and complete.

By: Kenneth P. Powers

Kennein P Powers
General Plant Manager

Sworn and subscribed to before me this 19th day of September 1997.

Jennifer Lynn Helms

Jennifer Lynn Helms, Notary Public
Charlevoix County, Michigan

My commission expires August 29, 1999. [SEAL]

ATTACHMENT

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

BIG ROCK POINT FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGE REQUEST - DEFUELED TECHNICAL SPECIFICATIONS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555

CONSUMERS ENERGY COMPANY

DOCKET NO 50-155

BIG ROCK POINT PLANT

FACILITY OPERATING LICENSE

License No DPR-6

A. This license applies to the decommissioning of Big Rock Point Plant, (the facility), owned by Consumers Energy Company (the licensee). The facility is located in Charlevoix County, Michigan, and is described in the licensee's application dated January 14, 1960, and the Final Hazards Summary Report; as supplemented and amended by subsequent filings by the licensee.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Consumers Energy Company:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities" to possess the facility at the designated location in Charlevoix County, Michigan, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material," to possess at any one time up to (a) 2500 kilograms of contained uranium 235 in fuel rods, (b) 10.32 grams of uranium 235 as contained in fission counters, (c) 150 kilograms of plutonium contained in PuO_2-UO_2 fuel rods, and (d) 5 curies of plutonium encapsulated as a plutonium-beryllium neutron source, subject to the following conditions:

(a) The storage of materials in the area between rack B and the east wall of the spent fuel pool is prohibited.

(b) The use of the gantry crane over the pool for loads of over 24 tons is prohibited.

(c) Only spent fuel with a decay time of at least one year will be stored in the outer three rows of the fuel rack adjacent to the south wall of the fuel pool. A prompt investigation by the company shall be required whenever radiation in the sock tank area exceeds 50 mrem/hr.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of By-product Material," to receive, possess and use at any one time up to 7000 curies of antimony-beryllium in the form of neutron sources, 3.7 curies of cobalt-60 as sealed sources, 45 curies of cesium-137 as sealed sources, 10 microcuries of miscellaneous alpha emitting material as sealed sources, and up to 500 millicuries per nuclide

of any byproduct material between atomic numbers 1 and 83, inclusive, without restriction as to chemical and physical form;

(4) Pursuant to the Act and 10 CFR Part 40, "Licensing of Source Material," to possess at any one time up to 500 kilograms of depleted uranium dioxide contained in the facility's fuel assemblies;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may have been produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Reactor Operation

The reactor is not licensed for power operation. Fuel shall not be placed in the reactor vessel.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ___, are hereby incorporated in the license. The licensee shall maintain the facility in accordance with the Technical Specifications.

(3) The licensee shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Big Rock Point Plant Security Plan," with revisions submitted through September 29, 1988; "Big Rock Point Plant Suitability Training and Qualification Plan," with revisions submitted through November 30, 1988; and "Big Rock Point Plant Safeguards Contingency Plan," with revisions submitted through September 30, 1988. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

D. This license amendment is effective as of the date of its issuance, or at midnight on November 30, 1997, whichever is later.

Attachment: Appendix A
Technical Specifications

Date of Issuance:

1.0 DEFINITIONS

1.1 CHANNEL CALIBRATION

A CHANNEL CALIBRATION consists of adjustments to the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. A CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. A CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

1.2 CHANNEL CHECK

A CHANNEL CHECK is the qualitative assessment of channel behavior, and is performed by observing channel response during operation. Where practicable, the observation shall include comparison of the channel indication or status with other indications or status derived from independent instrumentation channels measuring the same parameter.

1.3 CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST is the injection of a signal into the channel as close to the sensor as practicable to verify OPERABILITY, including alarm and trip functions.

1.4 CONTAINMENT CLOSURE

CONTAINMENT CLOSURE is that condition of containment in which there are no direct paths from containment atmosphere to the outside atmosphere. For containment piping penetrations, CONTAINMENT CLOSURE is defined to exist when one valve is closed, or a check valve preventing outward flow of fluids from containment is in the line. Leak tightness is not required for CONTAINMENT CLOSURE to exist.

1.5 DIRECT PATHS

A DIRECT PATH is a visually observable opening which permits the free exchange of air between containment and the environs. Equipment configurations or engineered alternatives may be used to preclude direct paths.

1.6 DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131, expressed in microcuries per milliliter, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 which is actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table 5.2 of EPA 400, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents".

1.7 MEMBERS OF THE PUBLIC

MEMBERS OF THE PUBLIC include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors and vendors, or persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes.

1.8 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The OFF-SITE DOSE CALCULATION MANUAL (ODCM) contains the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Section 6.6.2.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.7.2 and 6.7.3.

1.9 OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system, subsystem, train, component or device to perform its safety function(s) are also capable of performing their related support function(s).

1.0 DEFINITIONS (continued)

1.10 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM contains the methods and determinations which ensure that the processing and packaging of wet solid radioactive wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.11 REPORTABLE EVENT

A REPORTABLE EVENT is any of those conditions specified as reportable in 10 CFR 50.72 and 10 CFR 50.73.

1.12 SITE BOUNDARY

The SITE BOUNDARY is that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

1.13 SOURCE CHECK

A SOURCE CHECK is the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.14 UNRESTRICTED AREA

An UNRESTRICTED AREA is any area at or beyond the SITE BOUNDARY which is not controlled by the licensee for purposes of protection of individuals from undue exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, or recreational purposes.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

There are no safety limits or limiting safety system settings applicable to the permanently defueled condition.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Limiting Conditions for Operation and Action requirements shall be applicable during the specified applicable condition for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation or associated Action within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the Action statement is not required.

3.0.3 Unless otherwise specified, entry into an applicability condition shall not be made unless the conditions of the associated Limiting Condition for Operation are met without reliance on provisions contained in the Action statements.

SURVEILLANCE REQUIREMENTS

4.0.1 Unless specified otherwise, surveillance requirements shall be applicable during the specified applicable conditions for the associated Limiting Conditions for Operation.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 per cent of the specified surveillance interval.

4.0.3 Unless specified otherwise, performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated Action statements.

4.0.4 Unless specified otherwise, entry into a specified applicable condition shall not be made unless the Surveillance Requirements associated with the Limiting Condition for Operation have been performed within the stated surveillance interval.

3/4.1 FUEL STORAGE

3/4.1.1 SPENT FUEL POOL PARAMETERS

LIMITING CONDITIONS FOR OPERATION

3.1.1 The following parameters shall be monitored and maintained within the limits indicated:

a. The water level in the Spent Fuel Pool shall be maintained above the elevation of the syphon breaker (630' 4").

b. Spent fuel pool water temperature shall be maintained greater than 40 °F and less than 150 °F.

c. Radiation levels in the area of the Spent Fuel Pool shall be monitored by a fixed gamma radiation monitor with a locally audible alarm set at not less than 5 millirems per hour and not more than 20 millirems per hour.

However, notwithstanding the requirements of 10CFR70.24(a)(2), alarm settings may be raised above 20 mrem/hr provided the overall detection criterion in 10CFR70.24(a)(2) is satisfied.

APPLICABILITY: When irradiated fuel assemblies are stored in the Spent Fuel Pool.

ACTION:

i. With the requirements of 3.1.1.a not met, immediately suspend activities having potential to drain the spent fuel pool. Place fuel assemblies and the crane load in a safe condition, and then suspend further movement of fuel assemblies and crane operations with loads in or over the spent fuel pool. Within 24 hours, establish CONTAINMENT CLOSURE.

ii. With the requirements of 3.1.1.b not met, within 24 hours establish CONTAINMENT CLOSURE, and initiate action to restore acceptable temperature.

iii. With the requirements of 3.1.1.c not met, provide an alternate method of monitoring spent fuel pool radiation levels. If the alternate instrumentation does not have an audible alarm, it shall be continuously monitored when personnel are in the vicinity of the spent fuel pool.

The provisions of Specification 3.0.3 are not applicable.

3/4.1 FUEL STORAGE

3/4.1.1 SPENT FUEL POOL PARAMETERS

SURVEILLANCE REQUIREMENTS

4.1.1.a Approximately once per 4 hours, the water level in the Spent Fuel pool shall be determined to be equal to or greater than its minimum required depth.

4.1.1.b Approximately once per 4 hours, spent fuel pool water temperature shall be determined to meet the requirements of Specification 3.1.1.b

4.1.1.c The spent fuel pool radiation monitor required by this specification shall be demonstrated operable:

i. Daily by performing a CHANNEL CHECK.

ii. Once per 31 days by performing a CHANNEL CALIBRATION.

3/4.1 FUEL STORAGE

3/4.1.2 SPENT FUEL POOL SUPPORT SYSTEM REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

3.1.2 The capability to supply makeup to the spent fuel pool shall be maintained as follows:

a. A diesel generator capable of providing power within 24 hours to operate an on site electric motor-driven pump, and one off-site source of ac power capable of providing power to operate an on site electric motor-driven pump shall be available, OR,

An on site pump not requiring electrical power shall be capable of providing makeup water to the spent fuel pool within 24 hours.

b. The pump designated to satisfy the requirements of Specification 3.1.2.a shall be capable of supplying at least 28 gpm of water within 24 hours at a temperature equal to or less than 100 °F to the spent fuel pool emergency makeup line. The capability to manually initiate at least 28 gpm flow to the spent fuel pool at the point the spent fuel pool emergency makeup line enters the spent fuel pool shall be maintained.

APPLICABILITY: When irradiated fuel assemblies are stored in the Spent Fuel Pool.

ACTION: With the requirements of this specification not met, within 24 hours establish an alternate source of water capable of delivering at least 28 gpm of water at a temperature equal to or less than 100 °F to the spent fuel pool.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.a *i* Daily, verify the presence of potential on the offsite power line.

ii Monthly, manually start the diesel generator and run loaded for 30 minutes using an electric motor-driven pump as a load.

4.1.2.b *i* Once per 12 months, verify that the pumps satisfying the requirements of this specification are capable of supplying water to the emergency makeup line within 24 hours and will deliver at least 28 gpm of water.

3/4.1 FUEL STORAGE

3/4.1.2 SPENT FUEL POOL SUPPORT SYSTEM REQUIREMENTS

SURVEILLANCE REQUIREMENTS (continued)

ii Once per 12 months, the spent fuel pool emergency makeup line shall be determined to be OPERABLE by verifying its flow capacity to be at least 28 gpm.

3/4.1 FUEL STORAGE

3/4.1.3 FUEL STORAGE GENERAL REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

3.1.3 The following limitations shall apply to the storage of spent fuel in the spent fuel pool.

a. The storage of materials in the area directly between rack B and the east wall of the spent fuel pool is prohibited.

b. Only spent fuel with a decay time of greater than one year will be stored in the outer three rows of the fuel rack adjacent to the south wall of the fuel pool.

APPLICABILITY: When irradiated fuel assemblies are stored in the Spent Fuel Pool.

ACTION: With the requirements of this specification not met suspend fuel handling operations and perform a prompt investigation to determine the cause and initiate appropriate corrective actions.

SURVEILLANCE REQUIREMENTS

4.1.3

a. Upon completion of any activity involving movement of components in the spent fuel pool, verify that the requirements of Specification 3.1.3.a have been met.

b. Upon completion of any fuel handling activity which involves movement of fuel into the fuel rack adjacent to the south wall of the spent fuel pool, verify that the storage configuration satisfies the requirements of Specification 3.1.3.b. Performance of this surveillance is not required after September 15, 1998.

3/4.2 FUEL HANDLING

3/4.2.1 FUEL HANDLING SUPPORT SYSTEM REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

3.2.1 a. A fixed gamma radiation monitor, having the same setpoint as the monitor required by Specification 3.1.1.c, shall be capable of initiating containment ventilation valve closure, or CONTAINMENT CLOSURE shall exist at the containment ventilation valves.

b. For all other containment penetrations or openings, CONTAINMENT CLOSURE shall exist.

APPLICABILITY: When handling fuel inside containment.

ACTION: With the requirements of this specification not met, place fuel assemblies in a safe condition, and suspend fuel handling activities inside containment.

SURVEILLANCE REQUIREMENTS

None.

3/4.2 FUEL HANDLING

3/4.2.2 FUEL HANDLING GENERAL REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

3.2.2 Fuel handling operations shall conform to the following requirements:

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

b. Radiation levels at the spent fuel pool filter sock tank area shall be maintained at less than 50 mrem/hr.

APPLICABILITY: When handling fuel in the spent fuel pool.

ACTION:

a. With the requirements of Specification 3.2.2.a not met, place irradiated fuel assemblies in a safe condition. Conduct a prompt investigation to determine the cause and initiate appropriate corrective actions.

b. With the requirements of Specification 3.2.2.b not met, suspend fuel handling operations and conduct a prompt investigation to determine the cause of increased radiation levels. Fuel handling may resume if it is determined that increased radiation levels have not been caused by the handling of fuel in the spent fuel pool.

SURVEILLANCE REQUIREMENTS

4.2.2

a. Once per 8 hours during fuel handling operations and upon completion of fuel handling activities, monitor the radiation levels in the spent fuel pool sock tank area to verify that the requirements of Specification 3.2.2.b are met.

3/4.3 CONTROL OF HEAVY LOADS

LIMITING CONDITIONS FOR OPERATION

3.3.1 Handling of heavy loads over or in the spent fuel pool shall conform to the following requirements:

a. Use of the reactor building gantry crane over the spent fuel pool for loads exceeding 24 tons is prohibited.

b. With the exception of the fuel transfer cask, or other cask specifically approved by NRC for use in or over the spent fuel pool:

i. No cask shall be moved over spent fuel stored in the spent fuel pool.

ii. Cask handling operations shall be limited to the southwest corner of the spent fuel pool.

iii. No fuel shall be stored in the fuel storage racks adjacent to the cask handling area in the southwest corner of the spent fuel pool during cask handling operations.

APPLICABILITY: When fuel is stored in the spent fuel pool.

ACTION: With the requirements of this specification not met, immediately place the crane load in a safe condition, and suspend further load handling activities with the reactor building gantry crane. Load handling shall not resume until permission has been obtained from the Site General Manager.

SURVEILLANCE REQUIREMENTS

None

3/4.4 SEALED SOURCE CONTAMINATION

LIMITING CONDITIONS FOR OPERATION

3.4.1 Each sealed source containing more than 100 microcuries of beta or gamma emitting material, or more than 10 microcuries of alpha emitting material shall not have removable contamination which exceeds 0.005 microcuries.

APPLICABILITY: At all times.

ACTION:

1. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and either decontaminated and repaired, or disposed of in accordance with NRC regulations.

   a. A special report shall be submitted to the NRC as indicated by Specification 6.7.4.

SURVEILLANCE REQUIREMENTS

4.4.1 Except for: 1) sealed sources which are stored and not in use, and 2) start up sources and fission detectors previously subjected to core neutron flux, sealed sources containing radioactive materials in any form other than gas and with a half-life greater than 30 days (excluding ${}_{1}H^{3}$) shall be tested for contamination at least once per six months by the licensee or other person specifically authorized by the NRC or an Agreement State to perform such services. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

a. Sealed sources requiring testing by this section, but exempted on the basis of not being in use, shall have been tested within 6 months prior to being transferred or put into use.

BASES FOR

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

FOR THE

BIG ROCK POINT

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

<u>NOTE</u>

The bases contained in this section summarize the reasons for the Specifications in Section 3/4, but in accordance with 10 CFR 50.36, are not part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

3.0.1 This specification defines specifically when the other specifications in Section 3/4 are applicable.

3.0.2 This specification defines those conditions which must be met in order to comply with the terms of a Limiting Condition of Operation and its associated Action requirement.

3.0.3 This specification provides that entry into a specified Applicability condition may be made only when (a) the full complement of required systems, equipment and components are OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation are met without regard to allowable deviations and out of service provisions contained in the Action statements.

The intent of this provision is to ensure that activities are not initiated with required equipment inoperable or other specified limits being exceeded.

4.0.1 This specification establishes that, unless otherwise specified, surveillances must be performed during the specified applicable conditions for which the requirements of the Limiting Conditions for Operation apply. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a specified Applicability condition. The specification also establishes that surveillance requirements do not need to be performed when the facility is in a condition for which the requirements of the associated Limiting Condition for Operation do not apply.

3/4.0 APPLICABILITY

BASES (continued)

4.0.2 This specification establishes how long the specified time interval for a surveillance requirement may be extended. The intent of providing this allowance is to facilitate surveillance scheduling to account for conditions that may not be suitable for conducting the surveillance. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond those specified.

The allowable extension provided by Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance is conformance with the surveillance requirement. These provisions are sufficient to ensure that the reliability demonstrated through surveillance activities is not significantly degraded as a result of an extended interval.

4.0.3 The provisions of this specification set forth the criteria for determining compliance with the OPERABILITY requirements of Limiting Conditions for Operation. Under these criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining items OPERABLE when they are known to be inoperable even if they meet their associated surveillance requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into specified applicability conditions. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Unless a specification states otherwise, applicable surveillance activities must be performed within the stated surveillance interval prior to placing the associated item in an OPERABLE status.

3/4.1 FUEL STORAGE

3/4.1.1 SPENT FUEL POOL PARAMETERS

BASES

3.1.1.a Spent Fuel Pool water level is normally maintained at elevation 630' 6". This is the elevation of the discharge overflow weir (reference drawing 0740G40153).

Emergency cooling of the pool water is accomplished by a "feed and bleed" method that adds cooler inventory to the pool and removes the hotter inventory at the discharge weir. The minimum level required by this specification will ensure that the initial mass of the water in the pool is consistent with the spent fuel pool thermal analysis which concludes that feed and bleed is a viable cooling method for the spent fuel pool. This level allows operational flexibility by allowing interruption of spent fuel pool cooling when dictated by operational needs. Maintaining water level above the minimum level ensures the FHSR assumption of 20 feet of water over stored fuel is satisfied and that iodine removal in the event of a fuel handling accident is consistent with the iodine removal factors stated in Regulatory Guide 1.25.

The ACTION of this specification requires immediate suspension of activities that can result in draining the pool to minimize the loss of inventory and allow for the restoration of the water level. The fuel assemblies and crane loads must be placed in a safe configuration; and further fuel movement suspended until level is restored. This action will minimize the probability of a fuel handling event before the water level is restored. However, containment closure is required within 24 hours to minimize a release if an event were to occur and the water level is not available to support the thermal analysis or the radiological mitigating effects of scrubbing or shielding.

4.1.1.a Based on empirical data, the rate of water loss from normal evaporation in the Spent Fuel Pool is not expected to exceed 0.1 gallons per minute (corresponding to less than a half inch loss in water level per day). As a result, the specified frequency for monitoring Spent Fuel Pool water level provides for early detection of decreasing water level and provides assurance that water level will not drop to a level significantly less than the specified level during the interval between readings. The level specified may be verified by monitoring level instrumentation in the Control Room, or by visual inspection of the spent fuel pool itself.

3.1.1.b The allowable spent fuel pool temperatures are defined by the structural analysis of the concrete pool and the criticality determination for the spent fuel in the racks.

3/4.1 FUEL STORAGE

3/4.1.1 SPENT FUEL POOL PARAMETERS

BASES

The criticality analysis performed for the spent fuel pool expansion determined that K (neutron multiplication factor) is less than 0.95 at the most reactive temperature, which is greater than the minimum specified temperature of 40 °F. This temperature ensures that freezing of the spent fuel pool will not occur.

The maximum allowable pool temperature is based on the spent fuel pool structural analysis. The analysis concluded that adequate structural capacity exists if the spent fuel pool is maintained at a temperature of less than 150 °F.

The ACTION of this specification leads to restoration of acceptable temperatures in the spent fuel pool. The manner of temperature restoration is not specified; the support systems required by Specification 3.1.2 provide assurance that spent fuel pool cooling can be restored.

4.1.1.b Spent fuel pool water temperature is verified twice per shift (once per approximately 4 hours) in order to permit early detection of abnormal spent fuel pool cooling. This ensures that corrective action can be promptly taken.

3.1.1.c Maintaining the Spent Fuel Pool area radiation monitor operable as specified in this section permits meeting the criticality accident monitoring requirements of 10 CFR 70.24(a)(2). The fixed gamma radiation monitor provides an indication of spent fuel pool criticality and elevated radiation levels that may be indicative of a release. Raising alarm settings beyond 20 mr/hr may be necessary during activities which involve lifting irradiated components out of the pool or near its surface. When the option to raise the alarm setpoint is exercised, the revised alarm setpoint must continue to meet the criticality detection criteria of 10 CFR 70.24(a)(2).

4.1.1.c The surveillance requirements are intended to provide confidence in instrument performance. Acceptable methods for performing a CHANNEL CHECK include comparing the indicated channel output under existing plant conditions, or comparing channel output to that of a portable radiation instrument.

The ACTION associated with this specification will provide the equivalent monitoring capability at a level that is approved by 10CFR70.24(a)(2).

3/4.1 FUEL STORAGE

3/4.1.2 SPENT FUEL POOL EMERGENCY SUPPORT SYSTEM REQUIREMENTS

BASES

3.1.2 Maintaining a diesel generator as an alternate source of AC electrical power provides redundancy in the power supply to an electrically-driven water source. This redundancy in power supplies to an electric-motor driven pump ensures that a loss of off-site power will not result in a loss of emergency makeup cooling to the spent fuel pool. In the event no ac power source is available, a pump not requiring ac electrical power will provide emergency makeup cooling to the Spent Fuel Pool. Acceptable options would include provisions to make connections from the alternate equipment to on site systems. Requiring the availability of a diesel generator within 24 hours is consistent with the slow rate of heat up of the spent fuel pool (calculated to be approximately 1 °F per hour under highest heat load conditions), and is also consistent with the low rate of spent fuel pool water inventory loss due to evaporation (discussed in the basis for Specification 4.1.1.a).

The pump performance requirements are to supply at least 28 gpm of water, at less than 100 degrees F, to the spent fuel pool. These parameters satisfy the design requirements of the emergency makeup cooling that is used in the structural and thermal hydraulic analysis of the spent fuel pool.

Sources and required inventories of water to be used as the makeup supply are not specified. A number of preferred sources, such as demineralized water, treated waste and condensate storage tank inventory would normally be used; however, water from Lake Michigan is also available and would be used in the event these tanks should be emptied or otherwise disabled. For purposes of emergency makeup or emergency cooling, lake water has sufficient purity for use in the spent fuel pool.

The use of the emergency makeup line will ensure that the supply of emergency makeup cooling water is at the same location as that used in the thermal hydraulic analysis. This will ensure that natural circulation will occur throughout the spent fuel pool, cooling is provided to all the spent fuel assemblies and localized boiling does not occur.

The ACTION of this specification ensures that an analyzed spent fuel pool cooling source (28 gpm of water at a temperature not greater than 100 °F) will be available. Allowing 24 hours to establish this alternate source is consistent with the slow heatup of the spent fuel pool (calculated to be approximately 1 °F per hour under highest heat load conditions).

3/4.1 FUEL STORAGE

3/4.1.2 SPENT FUEL POOL EMERGENCY SUPPORT SYSTEM REQUIREMENTS

BASES

4.1.2 The surveillance requirements of this specification provide assurance that needed equipment will perform as required.

3/4.1 FUEL STORAGE

3/4.1.3 FUEL STORAGE GENERAL REQUIREMENTS

BASES

3.1.3 The placement of material in the spent fuel pool is restricted to ensure that the flow paths required to supply cooling to the spent fuel assemblies stored in the racks are clear. Natural circulation will develop across the spent fuel assemblies as buoyancy changes in the water due to heat transfer results in the warmer water rising to the top of the pool. Storage of material between the east wall and Rack B can result in disrupting the natural circulation patterns required for heat transfer in the pool and adversely affect the cooling flow to various spent fuel assemblies. (This storage restriction applies only to the space directly between Rack B and the east wall of the spent fuel pool.)

The south wall provides the least amount of shielding with respect to the thickness of the other spent fuel pool walls. The activity level of the spent fuel stored closest to the south wall is restricted to minimize the radiological field through the wall.

The ACTION to suspend fuel handling operations and perform corrective actions will ensure that the coolable geometry and ALARA considerations of the spent fuel pool are established before fuel movement resumes.

4.1.3 The surveillances required by this specification provide assurance that the spent fuel pool storage configuration continues to be satisfactory after fuel or other components have been handled in the spent fuel pool.

3/4.2 FUEL HANDLING

3/4.2.1 FUEL HANDLING SUPPORT SYSTEM REQUIREMENTS

BASES

3.2.1.a The requirement to maintain the capability to have automatic closing capability at the containment ventilation valves, or alternatively, to have CLOSURE established at these valves provides assurance that a release resulting from a fuel handling accident will not be significant.

3.2.1.b This specification assures that no significant release will occur through any openings which may be made in the containment shell during the decommissioning process. CLOSURE, as opposed to leak-tightness, is specified, since there is no propelling force associated with analyzed fuel handling accidents.

3/4.2 FUEL HANDLING

3/4.3.1 FUEL HANDLING GENERAL REQUIREMENTS

BASES

3.2.2 The movement of spent fuel is restricted to one assembly at a time to minimize the consequences of a fuel handling accident to those analyzed for the drop of a single assembly.

The ACTION associated with this specification is required to place fuel assemblies in an analyzed configuration.

The radiation level within the spent fuel filter sock tank area is restricted to a value which assures that stored fuel is not making an excessive contribution to radiation levels in the area adjacent to the south wall of the spent fuel pool.

The ACTION associated with this specification is required to permit determining the cause of the increased radiation levels, and specifically to determine if stored fuel is the cause of the increased levels.

4.2.2 The frequency of performing the required radiation monitoring will provide assurance that increased radiation levels in the filter sock tank area caused by placement of fuel in the spent fuel pool will not be undetected. The radiation readings should be taken at a distance from the wall which satisfies ALARA considerations and satisfies the geometric requirements to consider the spent fuel pool wall as a plane source. This distance (or location) will be governed by applicable radiation monitoring procedures.

3/4.3 CONTROL OF HEAVY LOADS

BASES

3.3.1.a This specification reflects a license condition resulting from the spent fuel pool hearings. The fuel storage racks adjacent to the cask handling area are Racks D and E, as described in the FHSR.

3.3.1.b This specification reflects a requirement stated in NRC SER dated May 15, 1981, related to the expansion of the spent fuel pool storage capacity.

3/4.4 SEALED SOURCE CONTAMINATION

BASES

3.4.1 The limitation on removable contamination for sources requiring leak testing meets the limits of 10 CFR 70.39 for sources containing plutonium. The limitation provides assurance that leakage from sealed sources covered by the specification will not exceed allowable intake values.

The ACTION ensures that sources exhibiting leakage are controlled in accordance with NRC regulations.

4.4.1 For purposes of testing, the specification categorizes sealed sources into two groups (in use, or stored and not in use), with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled require more frequent testing. Sealed sources which are continuously enclosed within a shielded mechanism (such as sealed sources within radiation monitoring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

5.0 DESIGN FEATURES

5.1 SITE

5.1.1 LOCATION

The plant site, consisting of approximately 600 acres, is located in Charlevoix County, Michigan, about 4 miles northeast of Charlevoix, Michigan, and about 11 miles west of Petoskey, Michigan.

5.2 STORAGE OF SPECIAL NUCLEAR MATERIAL

5.2.1 CRITICALITY

The spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between stored fuel assemblies to ensure a k_{eff} less than or equal to 0.95 when the racks are flooded with unborated water.

Fuel inspection stations, if installed, shall be designed and maintained with sufficient center-to-center distance between fuel assemblies placed in the inspection stations to ensure a k_{eff} less than or equal to 0.95 when flooded with unborated water.

The fuel loading per axial centimeter of any assembly placed in the Spent Fuel Pool shall be less than or equal to a maximum of 28.3 grams of U^{235} or equivalent.

5.2.2 WATER LEVEL

The Spent Fuel Pool is designed to maintain a normal water level of 630 6" and to prevent unintentional draining below the syphon breaker which is at elevation 630' 4".

5.2.3 COOLING

The configuration of storage racks placed in the spent fuel pool allows for adequate circulation of water to prevent localized pool boiling and excessive thermally-induced loads in the spent fuel pool concrete structure.

5.0 DESIGN FEATURES

5.2.4 CAPACITY

Subject to the limits listed below, the fuel pool is designed for and shall be maintained with a storage capacity of no more than 441 fuel assemblies. In addition, fuel pins which have been removed from fuel assemblies may be stored in the pool, provided there are an adequate number of pin positions not containing fuel in the stored spent fuel assemblies. The following limits apply to the amount of special nuclear material which may be stored in the Spent Fuel Pool:

- 2500 kilograms of contained uranium 235.
- 10.32 grams of uranium 235 as contained in fission counters.
- 150 kilograms of plutonium contained in PuO_2-UO_2 fuel rods.
- 5 curies of plutonium encapsulated as a plutonium-beryllium source.

5.3 REACTOR

5.3.1 STATUS

The reactor is not licensed for power operation. Fuel shall not be placed in the reactor vessel.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY AND AUTHORITY

6.1.1 SITE GENERAL MANAGER

The Site General Manager shall be responsible for overall facility operation, and for periods of absence shall delegate in writing the succession to this responsibility. Unless otherwise specified, the Site General Manager's delegate has authority to perform all actions and grant approvals assigned by these specifications to the Site General Manager. The Site General Manager may delegate specific tasks to other individuals who may perform those tasks whether the Site General Manager is absent or present at the site.

6.2 ORGANIZATION

6.2.1 REPORTING RELATIONSHIPS

On-site organization and corporate reporting relationship shall be established for activities affecting safety of the facility.

a. Lines of authority, responsibility and communication shall be established and documented in facility administrative procedures.

b. The Site General Manager shall be responsible for safe operation of the facility and shall have control over those on-site activities necessary for safe operation and maintenance of the facility. The individual filling this position shall report directly to the Senior Nuclear Officer.

c. The individuals who perform audits, surveillances and independent safety reviews report to the Manager, Nuclear Performance Assessment Department.

6.2.2 FACILITY ORGANIZATION

The facility organization shall be subject to the following:

a. Each on-duty shift shall be comprised of at least the minimum shift crew composition shown in Table 6.2-1.

b. All fuel handling operations shall be directly supervised by a Certified Fuel Handler who shall have no other concurrent responsibilities during this operation. Certified Fuel Handlers shall meet qualifications set forth in the station's administrative procedures.

c. A qualified Radiation Protection Technician shall be on site during fuel handling operations and during specific work activities which have potential to result in release or spread of radioactive material. At other times when fuel or other radioactive material is stored on site, an individual qualified in radiation protection procedures shall be present on site.

d. Administrative procedures shall be implemented to limit the working hours of the facility staff who perform safety related functions. These individuals include the minimum shift crew required by Table 6.2-1, key maintenance personnel and Radiation Protection Technicians. [1]

   Adequate shift coverage shall be maintained without routine heavy use of overtime. However in the event that unforeseen problems require substantial amounts of overtime to be used, the following guidelines shall be followed: [2]

   (1) An individual should not be permitted to work more than 16 hours straight.

   (2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period.

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[1] The overtime restrictions of this section apply only to periods when spent fuel is stored in the Spent Fuel Pool, or when spent fuel handling or radioactive waste handling activities are conducted.

[2] The length of time required to conduct shift turnover is considered to be part of the work break period, and is not added to the work period.

(3) A break of at least 8 hours should be allowed after continuous work periods of 16 hours duration. [1]

Any deviation from the above guidelines shall be authorized in accordance with established administrative procedures and with documentation of the basis for granting the deviation. Administrative procedures shall include a requirement to review individual overtime on a monthly basis in order to verify that excessive hours have not been assigned. Routine deviations from the above guidelines is not authorized.

[1] The length of time required to conduct shift turnover is considered to be part of the work break period, and is not added to the work period.

TABLE 6.2-1: MINIMUM SHIFT CREW COMPOSITION DURING PERMANENTLY DEFUELED CONDITION

POSITION [1]	STAFFING REQUIRED
Shift Supervisor [2] or Control Operator [3]	1
Auxiliary Operator	1

[1] At least one position shall be filled by a Certified Fuel Handler. This provides assurance that appropriate response to events involving fuel stored in the spent fuel pool will be maintained.

[2] The individual designated as the shift supervisor is not required to hold a Senior Reactor Operator License; however, if this individual does hold a valid Senior Reactor Operator license, or holds an SRO license limited to fuel handling, that individual will satisfy the requirement to have a Certified Fuel Handler on shift.

[3] Although the normal work station for this position is in the control room, the control room is not required to be staffed when the reactor is defueled.

6.3 STAFF QUALIFICATIONS

Each member of the facility management and supervisory staff shall meet the minimum requirements of ANSI N18.1-1971 for comparable positions. The individual responsible for radiation protection functions shall meet the minimum requirements of Regulatory Guide 1.8, September, 1975.[1]

6.4 TRAINING

A retraining and replacement training program for the facility Certified Fuel Handlers shall be conducted in accordance with an NRC approved training program. A training program for the facility staff shall be implemented and shall meet the requirements and recommendations of Section 5.5 of ANSI N18.1-1971.

6.5 REVIEW AND AUDIT

6.5.1 Requirements for on-site and off-site reviews and audits are described in CPC-2A, Quality Program Description.

6.6 PROCEDURES AND PROGRAMS

6.6.1 PROCEDURES

6.6.1.1 Scope

Written procedures shall be established, implemented and maintained for safety related structures, systems components and safety actions defined in the Big Rock Point Quality List. These procedures shall meet or exceed the requirements of ANSI N18.7, as endorsed by the Quality Program Description (CPC-2A).

6.6.1.2 Review and Approval

Requirements for review and approval of procedures (and revisions thereto) required by this section are described in CPC-2A, Quality Program Description.

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[1] As applied to this specification, "equivalent", as used 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 training in science engineering, or (b) operational or technical experience and training in nuclear power.

6.0 ADMINISTRATIVE CONTROLS (continued)

6.6.1.3 Temporary Changes

Requirements for making temporary changes to procedures which fall within the scope of this section are described in CPC-2A, Quality Program Description.

6.6.2 PROGRAMS

The following programs shall be established, implemented and maintained:

6.6.2.1 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.6.2.2 High Radiation Area

6.6.2.2.1 Dose Rates less than 1000 Millirem per Hour

In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is less than 1000 mrem/hr at 30 cm (12 inches) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by the use of a Radiation Work Permit (RWP). Radiation protection qualified personnel or personnel continuously escorted by radiation protection qualified personnel may be exempt from working under an RWP during the performance of their assigned radiation protection duties in high radiation areas with exposure rates of less than 1000 mrem/hr, provided they are otherwise following facility radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, or

c. A radiation protection qualified individual (e.g., Health Physics Technician) with a radiation dose rate monitoring device, responsible for providing positive control over the activities within the area.

6.6.2.2.2 Dose Rates greater than 1000 Millirem per Hour

In addition to the requirements of 6.6.2.2.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 inches) but less than 500 rad/hr at 1 meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under administrative controls specified in the facility administrative procedures. Doors shall remain locked except during periods of access by personnel under an approved RWP. The dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area shall be communicated to the individuals. In lieu of a stay time specification, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection

6.0 ADMINISTRATIVE CONTROLS (continued)

procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/hr that are located within large areas where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.6.2.3 PROCESS CONTROL PROGRAM (PCP)

6.6.2.3.1 Changes to the PCP

Changes to the PCP shall become effective after approval by the Site General Manager.

6.6.2.3.2 Reports

Changes to the PCP shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the changes were made effective. This submittal shall contain sufficiently detailed information to support the rationale for each change and a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes.

6.6.2.4 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.6.2.4.1 Changes to the ODCM

Changes to the ODCM shall become effective after approval by the Site General Manager.

6.6.2.4.2 Reports

Changes to the ODCM shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the changes were made effective. This submittal shall contain sufficiently detailed information to support the rationale for each change and a determination that the change did not reduce the accuracy or reliability of dose calculations or setpoint determinations.

6.6.2.5 Radioactive Effluent Controls Program

A program, conforming with 10 CFR 50.36a, for the control of radioactive effluents and for maintaining doses from radioactive effluents to MEMBERS OF THE PUBLIC as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by facility procedures, and (3) shall include remedial actions to be taken whenever program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations conforming to 10 CFR Part 20, Appendix B, Table II, Column 2 on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS;

c. Monitoring, sampling and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

d. Limitations conforming to Appendix I to 10 CFR Part 50 on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the facility to UNRESTRICTED AREAS;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.

g. Limitations conforming to the doses associated with 10 CFR Part 50 on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY;

h Limitations conforming to Appendix I to 10 CFR Part 50 on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the facility to areas beyond the SITE BOUNDARY; and

i. Limitations conforming to 40 CFR Part 190 on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources.

6.6.2.6 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the facility. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, or alternatively, that critical receptors are assumed to exist at the SITE BOUNDARY or offsite location of highest dose consequence; and

c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.6.2.7 Fire Protection Program

A fire protection program meeting the requirements of 10 CFR 50.48(f) shall be maintained.

6.7 REPORTING REQUIREMENTS

The reports identified in this section shall be submitted in accordance with 10 CFR 50.4.

6.7.1 ANNUAL RADIATION EXPOSURE REPORT

An annual report of radiation exposures received during the previous calendar year shall be submitted prior to March 1 of each year. This report shall tabulate the numbers of facility, utility and other personnel (including contractors)

ENCLOSURE 4

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Defueled Offsite Dose Calculation Manual

ODCM Section I

Procedural and Surveillance Requirements

(Relocated Technical Specifications)

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I PROCEDURAL AND SURVEILLANCE REQUIREMENTS (Relocated Technical Specifications)

1. GASEOUS EFFLUENTS

1.1 Radioactive Gaseous Effluent Monitoring Instrumentation

1.1.1 Requirement:

The radioactive gaseous effluent monitoring instrumentation channels shown in Table I.A-1 shall be Operable with their alarm/trip set points set to ensure that the limits of this Offsite Dose Calculation Manual (ODCM) Section I, Requirement 1.2.1 are not exceeded. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters of Section II, part 1.1 of this ODCM.

1.1.2 Action:

a. With a gaseous radioactive effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the above specification, suspend the release of radioactive effluent monitored by the affected channel or declare the channel inoperable, or change the set point so it is acceptably conservative.

b. With less than the minimum number of gaseous effluent monitoring instrumentation channels Operable, take the action shown in Table I.A-1. Exert best efforts to return the instruments to Operable status within 30 days and, if unsuccessful, explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

1.1.3 Surveillance Requirements:

Each radioactive effluent monitoring instrumentation channel shall be demonstrated Operable by performance of the Channel Check, Source Check, Channel Calibration and Channel Functional Tests at the frequencies shown in Table I.A-2.

1.1.4 Basis:

The radioactive gaseous effluent instrumentation is provided to indicate and quantify releases of radioactive material during actual or potential releases of gaseous effluents such that controls may be applied, as applicable, to limit such releases. The alarm set points for these instruments shall be calculated and adjusted in accordance with the methodology and parameters of Section II, part 1.1 of this ODCM to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20.

Beyond 93 days post shutdown (the time at which this ODCM is to replace the the Operating Phase ODCM), core plus fuel pool inventory of gaseous and iodine fission products is not sufficient to activate the high range noble gas monitor or

cause doses to exceed EPA Protective Action Guides beyond the Site Boundary. Thus use of iodine and high range noble gas monitors is not required by this ODCM.

Noble gas radioactivity monitoring is not required after fuel is removed from the spent fuel pool. With the reactor inoperable and all fuel removed from the spent fuel pool, there will be no sources of noble gas present to require such monitoring.

TABLE I.A-1

GASEOUS RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE[a]	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
None	NA	NA
2. STACK GAS EFFLUENT SYSTEM		
a. Noble Gas Activity Monitor[b]	(1)	1
b. Particulate Sampler	(1)	2
c. Sampler Flow Rate Monitor	(1)	2

[a] At all times unless otherwise noted
[b] Operation required when fuel is stored in spent fuel pool only

ACTION 1 - With Stack Noble Gas channel inoperable, effluent releases via this pathway may continue provided grab samples are analyzed by gamma spectrum analysis at a lower limit of detection of at least 10^{-6} microcurie/ml at least once per 24 hours.

ACTION 2 - With the channel inoperable, effluent releases via this pathway may continue provided that either reactor building continuous air monitor response is evaluated, or grab samples of reactor building air are taken and analyzed for particulate activity, at least once per 24 hours for calculation of release quantities.

TABLE I.A-2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
None	NA	NA	NA	NA
2. STACK GAS EFFLUENT SYSTEM				
a. Noble Gas Activity Monitor[a]	D	M	A(2)	Q(1)
b. Particulate Sampler	W	NA	NA	NA
c. Sampler Flow Rate Monitor	D	NA	A	NA

Note:
[a] Noble gas activity monitor required operable until all fuel is discharged from spent fuel pool

1.2 GASEOUS EFFLUENTS DOSE RATE

1.2.1 Requirement:

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary (see Figure 2.1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

1.2.2 Action:

With the dose rate averaged over a period of one hour exceeding the above limits (released radionuclide concentration hourly average > 10 DAC), upon discovery, promptly restore the release rate to within the above limit(s).

1.2.3 Surveillance Requirements:

a. The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

b. The dose rate due to tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the limits of 1.2.1 above by utilizing the methodology and parameters in Section II of this ODCM, with representative samples and analyses performed in accordance with the sampling and analysis program specified in Table LE-1.

1.2.4 Bases:

This specification is provided to allow the Licensee operational flexibility in meeting the limits of 10 CFR 50, Appendix I. The instantaneous dose rates of this specification are higher than the implied dose rates of 10 CFR 20, Appendix B, Table 2, Column 1, Effluent Concentrations For Members of the Public. However, the Licensee is expected to implement the ALARA philosophy for atmospheric effluents to ensure doses to the public at and beyond the site boundary are minimized and less than the annual doses of 10 CFR 50, Appendix I.

For members of the public who may at times be within the site boundary, the occupancy of that member of the public will usually be sufficiently low to compensate for any increase in dose rate due to atmospheric dispersion above that for the site boundary.

1.3. Airborne Radioactivity Dose

1.3.1 Requirements:

a. The air dose due to noble gases released in gaseous effluents to areas at and beyond the site boundary (see Figure 2.1) shall be limited to the following:

1) During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

2) During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

b. The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulates form with half lives greater than 8 days in gaseous effluents released to areas at and beyond the site boundary (see Figure 2.1) shall be limited to the following:

1) During any calendar quarter: Less than or equal to 7.5 mrems to any organ, and

2) During any calendar year: Less than or equal to 15 mrems to any organ.

1.3.2 Action:

With the calculated dose from the release of radioactive material exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

1.3.3 Surveillance Requirement:

Cumulative dose contributions for the current calendar quarter and current calendar year for radioactive materials shall be determined in accordance with the methodology and parameters in this ODCM at least once per 31 days.

1.3.4 Bases:

This specification is provided to implement the requirements of Secti II.B, II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Sections II.B and II.C of Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in airborne effluents to unrestricted areas will be kept "as low as is reasonably achievable."

The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated.

The dose calculation methodology and parameters established in Section II of this ODCM for calculating the doses due to the actual release rates of radioactive noble gases, iodines and particulates in airborne effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977.

The ODCM equations provided for determining doses at and beyond the SITE BOUNDARY are based upon historical average atmospheric conditions.

2. LIQUID EFFLUENTS

2.1 Radioactive Liquid Effluent Monitoring Instrumentation

2.1.1 **Requirement:**

The radioactive liquid effluent monitoring instrumentation channels shown in Table I.D-1 shall be Operable with their alarm/trip set points set to ensure that requirement 2.2.1 of ODCM Section I are not exceeded. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters of part 2.2, Section II of this Offsite Dose Calculation Manual (ODCM).

2.1.2 Action:

a. With a liquid radioactive effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the above specification, suspend the release of radioactive effluent monitored by the affected channel or declare the channel inoperable, or change the set point so it is acceptably conservative.

b. With less than the minimum number of liquid effluent monitoring instrumentation channels Operable, take the action shown in Table I.D-1. Exert best efforts to return the instruments to Operable status within 30 days and, if unsuccessful, explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

2.1.3 Surveillance requirements:

Each radioactive effluent monitoring instrumentation channel shall be demonstrated Operable by performance of the Channel Check, Source Check, Channel Calibration and Channel Functional Test operations at the frequencies shown in Table I.D-2.

2.1.4 Basis:

The radioactive liquid effluent instrumentation is provided to indicate and quantify releases of radioactive material during actual or potential releases of liquid effluents such that controls may be applied, as applicable, to limit such releases. The alarm set points for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20.

TABLE I.D-1

LIQUID RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE[a]	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
None	NA	NA
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent	(1)	1
3. FLOW RATE INDICATING DEVICE		
a. Liquid Radwaste Effluent Line	(1)	3
4. CANAL SAMPLE COLLECTION TANK		
a. Discharge Sampler	(1)	2

[a] At all times that liquid radwaste discharge line is not isolated

ACTION 1 - With the channel inoperable effluent releases may continue via this pathway provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Table I.E-4, and
b. The release rate calculations and discharge line valving are verified by at least one technically qualified member of the Facility Staff.

ACTION 2 - With the channel inoperable, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml

ACTION 3 - With the channel inoperable, effluent release via thjis pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves or tank levels may be used to estimate flow.

TABLE I.D-2

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
None	NA	NA	NA	NA
2. GROS[illegible] OR GAMMA RADIOACTIVITY MONI[illegible]S PROVIDING ALARM BUT [illegible]NG AUTOMATIC TERMINATION [illegible] RELEASE				
a. Li[illegible]ste Effluent	D(4)	SM	A(2)	Q(1)
3. FLOW RATE INDICATING DEVICE				
a. Liquid Radwaste Effluent Line	D(3)	NA	A	NA
4. CANAL SAMPLE COLLECTION TANK	D(3)	NA	NA	NA

TABLE I.D-2
NOTATION

NOTATION	FREQUENCY
D	At least once per 24 hours
W	At least once per 7 days
SM	At least twice per 31 days
M	At least once per 31 days
Q	At least once per 92 days
A	At least once per 12 months
NA	Not applicable

(1) The Channel Functional Test shall also demonstrate that alarm annunciation occurs if the instrument indicates measured levels above the alarm set point.

(2) a. The Channel Calibration shall be performed using one or more of the reference standards traceable to the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range.

b. For subsequent Channel Calibration, sources that have been related to the (a) calibration may be used.

(3) Channel Check shall consist of verifying indication of flow during periods of release. Channel Check shall be made at least once per 24 hours on days on which continuous or batch releases are made.

(4) Channel Check shall be made at least once per 24 hours on days on which continuous or batch releases were made.

2.2 LIQUID EFFLUENTS CONCENTRATION

2.2.1 Requirement:

The concentration of radioactive material released in liquid effluents from the site to UNRESTRICTED AREAS (identified as the site boundary, see Figure 2.1) shall be limited to 10 times the concentration values specified in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the total activity concentration shall be limited to 2 x 10^{-4} microcurie/ml.

2.2.2 Action:

a. With liquid effluent concentrations in areas at or beyond the site boundary (see Figure 2.1) exceeding limits, upon discovery, promptly restore the concentration to within the above limits.

2.2.3 Surveillance Requirements:

a. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table I.E-1.

b. The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in this ODCM to assure that the concentrations at the point of release are maintained within the limits of 2.2.1 above.

2.2.4 Basis

This requirement ensures that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than 10 times the concentration levels specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A. design objectives of Appendix I, 10 CFR Part 50, to a member of the public, and (2) the limits of 10 CFR Part 20.1301(a) to the population.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300, Currie, L. A. "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal Chem 40, 586-93 (1968), and Hartwell, J. K. "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

TABLE I.E-1
RADIOACTIVE WASTE SAMPLING AND ANALYSIS PROGRAM

Release Type	Sampling Frequency	Min Analysis Frequency	Type of Activity Analysis	(LLD)[a] (μCi/ml)
A. Liquid Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
B. Circulating Water Discharge[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters	5×10^{-7}
	Continuous[f]	M Composite[f]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous[f]	Q Composite	Sr-89, Sr-90	5×10^{-8}
C. Reactor Bldg	Continuous[h]	W[i] Particulate	Principal Gamma Emitters[g]	1×10^{-11}
	Continuous[h]	Q Composite Particulate	Sr-89, Sr-90 and Gross Alpha	1×10^{-11}
	Continuous[h]	Continuous-	Noble Gas Monitor Gross Beta or Gamma	1×10^{-5}

NOTATION	FREQUENCY
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
P	Completed prior to each release

Table I.E-1 Notation (Contd)

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet e^{-\lambda \Delta t}}$$

Where:

LLD is the predetermined lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a predetermined limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement.

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling.

Table I.E-1 Notation (Contd)

c. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99*, Cs-134, Cs-137, Ce-141 and Ce-144*. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report pursuant to Section III, part 1 of this ODCM.

*(LLD = 5 x 10^{-6} because of low gamma yields.)

d. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of a system that has an input flow during the continuous release.

f. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Xe-133, Xe-133m and Xe-135 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144** for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report pursuant to Section III, part 1 of this ODCM. ** (LLD = 1 x 10^{-10} because of low gamma yields)

h. The ratio of the sample flow to sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with requirements.

i. Samples shall be changed at least once per 7 days and analyses shall be completed within 7 days after changing or after removal from sampler.

2.3 LIQUID EFFLUENT DOSE

2.3.1 Requirement

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to areas at and beyond the site boundary (see Figure 2.1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

2.3.2 Action:

With the calculated dose from the release of radioactive material exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

2.3.3 Surveillance Requirements:

a. Cumulative dose contributions for the current calendar quarter and current calendar year for radioactive materials shall be determined in accordance with the methodology and parameters in this ODCM at least once per 31 days.

2.3.4 Bases:

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to unrestricted areas will be kept "as low as is reasonably achievable." Also, for fresh-water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141.

The dose calculation methodology and parameters in this ODCM implement the requirements in Section III.A of Appendix I that conformance be shown by calculational procedures based on models and data, such that the actual exposure of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology of Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3. TOTAL FUEL CYCLE DOSE

3.1 Requirement:

The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

3.2 Action:

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Requirements 1.3.1.a.(1), 1.3.1.a.(2), 1.3.1.b.(1), 1.3.1.b.(2), 2.3.1.a, or 2.3.1.b of this section, calculations should be made including direct radiation contributions from fuel pool systems and from outside storage tanks to determine whether the limits of Requirement 3.1 above have been exceeded. If such is the case:

a. Institute biota sampling in accordance with Table I.H-1, and

b. Prepare and submit to the Commission within 30 days, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits.

This Special Report, as defined in 10 CFR Part 20.2203(a)(4), shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations.

3.3 Surveillance Requirements:

a. Cumulative dose contributions from liquid and gaseous effluents shall be determined to comply with Requirement 3.1 above at a minimum of once per 31 days, and in accordance with the methodology and parameters described in Section II of this ODCM.

b. Cumulative dose contributions from direct radiation from spent fuel pool systems and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in ODCM Section II, part 3.2, Assumptions and 3.3, Dose Calculations. This requirement is applicable only under conditions set forth in Action 3.2.

3.4 Bases:

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant

generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 mi must be considered.

If the dose to any member of the public is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR 20.2203(a)(4) is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The requested variance relates only to the limits of 40 CFR Part 190 and shall not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications.

An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

4. RADIOLOGICAL ENVIRONMENTAL MONITORING

4.1 Requirement:

The radiological environmental monitoring program shall be conducted as specified in Table I.H-1.

4.2 Action:

a. With the radiological environmental monitoring program not being conducted as specified in Table I.H-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Section III, part 1 of this ODCM, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table I.H-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents. When more than one of the radionuclides in Table I.H-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Reporting Level (1)}} + \frac{\text{Concentration (2)}}{\text{Reporting Level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table I.H-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Requirements 1.3.1 or 2.3.1. of ODCM Section I. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the Condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk samples unavailable when required by Table I.H-1, identify new locations or sample types and obtain replacement samples. The unavailable sample or location then may be deleted from the monitoring program. Identify cause of unavailability and identify the new location(s) or sample types for in the next Annual Radiological Environmental Report.

4.3 Surveillance Requirements:

a. The radiological environmental monitoring samples shall be collected pursuant to Table I.H-1 and shall be analyzed pursuant to the requirements of Table I.H-1 and the detection capabilities required by Table I.H-3.

b. The land use census conducted in the final year of plant operation shall identify the nearest milk animal and the nearest resident in each overland sector to a distance of 8 km (5 miles). No land use census shall be required in years following plant shutdown as long as no residents are present, and no farm operations undertaken, within the site boundary (minimum distance of 0.5 miles of the shutdown reactor). If residential or farming use of the site occurs, evaluate and add appropriate samples to monitoring program.

c. Analytical Accuracy

Records of instrument calibrations and quality control data for the instrumentation used for environmental analyses shall be maintained. Interlaboratory comparison participation shall be required for biota analyses. Environmental TLD readouts will be performed by a NVLAP-accredited facility.

4.4 Bases:

a. Monitoring Program

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from post-operational conditions. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways.

Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. Program changes may be initiated based on operational experience. The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table I.H-3 are considered optimum for routine environmental measurements in industrial laboratories.

The LLD is a predetermined limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement. Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L A, "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry, Anal Chem 40, 586-93 (1968), and Hartwell, J K, "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-214 (June 1975).

b. Land Use Census

This requirement is provide to confirm the conservative nature of the GASPAR input parameters listed in ODCM Section II, Table 1.4, for assessment of dose beyond the site boundaries during the final year of plant operation. Use of an assumed garden and milk animal in the downwind sector of highest D/Q, per ODCM Section II, Table 1.4, is conservative with respect to any actual garden and milk animal locations. This, with the significant reduction of source term due to further fission product generation and source term decay, negates the need for further land use census data.

c. **Analytical Accuracy**

Participation in an approved Interlaboratory Comparison Program is required when biota samples are taken in response to elevated effluent release. However, such actions are not expected to be required since reduced radionuclide inventory and release rates are expected with the facility in permanently shutdown condition. Analytical accuracy of other radioactive material in effluent samples is documented in routine calibration and quality control checks in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50. In the shutdown condition, environmental monitoring relies primarily upon TLD analyses. TLD readouts are performed by a NVLAP Accredited laboratory in order to assure analytical reliability and accuracy.

TABLE I.H-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	13 routine monitoring stations either with two or more TLD's or one instrument for measuring and recording dose rate continuously, placed as follows: a) Miscellaneous site locations (4) b) An inner ring of stations, (6) at or near the SITE BOUNDARY c) Balance of stations (3) placed to serve as control stations.	Quarterly.	Gamma dose quarterly.
2. WATERBORNE			
a. Lake	1 sample from Plant Lake Water inlet (service water from intake bay).	Composite sample over 1-month period.[c]	Tritium and gamma isotopic[d] monthly.
b. Well (drinking) and groundwater monitoring wells	1 sample from Plant well, and 1 sample from each monitor well (6 wells)	Semiannual (grab) Semiannual (grab)	Tritium and gamma isotopic semiannually
3. BIOTA[e]			
a. Terrestrial	1 milk sample from dairy within 5 miles of plant	Monthly, (grab) May-Sept	Gamma isotopic monthly
b. Marine	1 fish or crayfish sample at or near plant discharge	Monthly, (grab) May-Sept	Gamma isotopic monthly

TABLE NOTATION TABLE I.H-1

a. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to the Reporting Requirements of ODCM Section III. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program.

b. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters.

c. A liquid composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid discharged and in which the method of sampling employed results in a specimen that is representative of the liquid released (continuous composites or daily grab composites which meet this criteria are acceptable).

d. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to licensed materials in the effluents from the facility.

e. Biota samples required only upon exceeding Requirement 3.1 of ODCM Section I.

TABLE I.H-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Fish (pCi/kg)	Milk (pCi/ℓ)
H-3	20,000*	-	-
Mn-54	1,000	30,000	-
Fe-59	400	10,000	-
Co-58	1,000	30,000	-
Co-60	300	10,000	-
Zn-65	300	20,000	-
Zr-Nb-95	400	-	-
I-131	2	-	3
Cs-134	30	1,000	60
Cs-137	50	2,000	70
Ba-La-140	200	-	300

*For drinking water samples. This is 40 CFR Part 141 value.

TABLE I.H-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS [a]

LOWER LIMIT OF DETECTION (LLD) [b,c]

Analysis	Water (pCi/ℓ)	Fish (pCi/kg)	Milk (pCi/ℓ)
gross beta	4		
H-3	2,000	-	-
Mn-54	15	130	-
Fe-59	30	260	-
Co-58, 60	15	130	-
Zn-65	30	260	-
Zr-Nb-95	15	-	-
I-131	1[d]	-	1
Cs-134	15	130	15
Cs-137	18	150	18
Ba-La-140	15	-	15

TABLE I.H-3 (Continued)

TABLE NOTATION

a. This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to the Reporting Requirements of ODCM Section III.

b. Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

c. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot e^{-\lambda \Delta t}}$$

Where:

LLD is the predetermined lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute, E is the counting efficiency, as counts per disintegration, V is the sample size in units of mass or volume, 2.22 is the number of disintegrations per minute per picocurie, Y is the fractional radiochemical yield, when applicable, λ is the radioactive decay constant for the particular radionuclide, and Δt for environmental samples is the elapsed time between sample collection or end of the sample collection period, and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

d. LLD for drinking water samples.

5. <u>Temporary Liquid Storage in Outside Tanks</u>

5.1 Requirement:

The concentration of radioactive material contained in any unprotected outside tank* used for temporary liquid radwaste storage shall be limited such that the mixture of radionuclides do not exceed 8000 times the effluent concentration (EC) as listed in 10CFR20, Appendix B, Table 2, Column 2.

$$\frac{C_a}{EC_a} + \frac{C_b}{EC_b} \ldots + \frac{C_i}{EC_i} = < 8000$$

5.2 Action:

With the quantity of radioactive material in any of the tanks exceeding the above concentration, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Radiological Effluent Release Report.

5.3 Surveillance Requirement:

The concentration of radioactive material contained in each tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

5.4 Bases:

This requirement will provide reasonable assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than ten times the limits of 10CFR20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the Unrestricted Area (the dilution between Big Rock Point and the Charlevoix drinking water supply has been established as 800). Such an uncontrolled release would not exceed the 10 CFR 20 annual dose limits for the public or the 10CFR50 Appendix I guidelines due to the short interval over which that any such release would occur.

* Outdoor tanks that are not surrounded by liners, dikes or walls capable of holding tank contents, or that do not have overflows or drains which would route leaks, overflows or other losses back to the radwaste treatment system.

6. Definitions and Surveillance Requirement Time Intervals

6.1 Requirement:

Terms shall be as defined in Technical Specification 1.0. Unless otherwise specified,Each surveillance requirement shall be performed with time extensions not to exceed 25% as defined in Technical Specification 4.0.2.

6.2 Action:

A surveillance not performed within the defined interval (including the allowable extension) shall constitute noncompliance with the operability requirements. Surveillance requirements do not apply to inoperable equipment.

6.3 Surveillance Requirement - NA

6.4 Basis:

Terms and Surveillance interval extensions are as defined in Technical Specifications.

ODCM Section II

Methodologies for Requirements Implementation

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1. GASEOUS EFFLUENTS

1.1 Alarm/Trip Set Point Method

This section of the ODCM describes the methodology that will be used to determine the set points defined by ODCM Section I, Requirement 1.1.

The method for determining alarm/trip set points is divided into two major parts. The first consists of calculating an allowable concentration for the nuclide mixture to be released. The second consists of determining monitor response to this mixture in order to establish the physical settings on the monitors.

1.1.1 Allowable Concentration

The total DAC-fraction (R) for the release point will be calculated as ten times the relationship defined by Note 4 of Appendix B, 10 CFR 20.

$$R = \left(\frac{X}{Q}\right)(F) \sum_i \frac{Ci_i}{DAC_i} = \leq 10 \qquad (1.1)$$

where:

Ci_i - The measured or calculated concentration, at ambient temperature and pressure, of nuclide i (μCi/cc) at the designated release point.

DAC_i - The DAC of nuclide i from 10 CFR 20, Appendix B.

R - The total DAC-fraction for the release point.

X/Q - Most conservative sector site boundary dispersion (9.12E-08 sec/m^3).

F - Release flow rate (39,000 cfm = 18.4 m^3/sec) for stack monitor considerations; variable for other monitors.

1.1.2 Monitor Response

Normal radioactivity releases during decommissioning consist predominantly of long lived particulate activity with potential for Kr-85 release in the event of an accident involving fuel. Radioiodines and noble gasses other than Kr-85 are of significant potential only for a short interval post-shutdown.

a. Monitors are preset to alarm at or before precalculated offsite dose rates would be achieved under hypothetical accident conditions. These set points are established in accordance with Emergency Plan requirements for defining Emergency Action Levels and associated actions. Emergency Implementing Procedures contain monitor-specific curves or calibration constants for conversion between cpm and µCi/cc (or rem/hr and µCi/cc), depending on monitor type, for fission product mixtures as a function of mixture decay time.

b. Under non-accident conditions, radionuclide concentration (µCi/cc) at the monitor is calculated. The calibration curve or constant for cpm/(µCi/cc) is applied to determine cpm expected. The setting for monitor alarms is established at some factor (b) greater than 1 but less than 1/R (Equation 1.1) times the measured concentration (c):

$$s = b \times c \qquad (1.2)$$

1.2 Dose Calculation

1.2.1 Dose Pathway Analyses

Doses are calculated for (1) noble gases and (2) iodines and particulates. Doses as defined in this section are based on 10 CFR 50, Appendix I limits of mrem per quarter and millirem per year. All dose pathways of major importance in the Big Rock Point environs are considered.

a. Assumptions for calculating doses from noble gases are as follows:

1) Doses to be calculated are the maximum offsite point in air, total body and skin.

2) Exposure pathway is submersion within a cloud of noble gases.

3) Historically observed source terms are given in Table 1.1.

4) Basic radionuclide data are given in Table 1.2.

5) All release are treated as elevated at 73 m. Appropriate modifications will be made if other than a 73 meter release path is utilized in the later phases of decommissioning (such as when the stack is being dismantled).

6) Meteorological data expressed as joint-frequency distribution of wind speed, wind direction, and atmospheric stability for the period resulting in X/Qs and D/Qs are shown in Table 1.3.

7) Raw meteorological data consist of wind speed and direction measurements at 71 m.

8) Dose is to be evaluated at the offsite exposure points where maximum concentrations are expected to exist.

9) Potential maximum population (resident) exposure points are identified in Table 1.4.

10) A semi-infinite cloud model is used.

11) For person exposures, credit is taken for shielding by residence (factor of 0.7).

12) Radioactive decay is considered for the plume.

13) A sector-average dispersion equation is used.

14) The wind speed classes that are used are as follows:

Wind Speed Class Number	Range (m/s)	Midpoint (m/s)
1	0.0-0.4	0.2
2	0.4-1.5	0.95
3	1.5-3.0	2.25
4	3.0-5.0	4.0
5	5.0-7.5	6.25
6	7.5-10.0	8.75
7	> 10.0	-

15) The stability classes that will be used are the Standard A through G classifications. The stability Classes 1-7 will correspond to A - 1, B - 2...G - 7.

16) Terrain effects are not considered, and no open terrain recirculation factors are applied.

b. Equations for Noble Gas Dose

To calculate the dose for any one of the exposure points, the following equations are used:

For determining the air concentration of any radionuclide:

$$X_i = \sum_{j=1}^{7} \sum_{K=1}^{7} \left(\frac{2}{\pi}\right)^{1/2} \frac{f_{jk} Q_i P}{\Sigma_{zk} u_j (2\pi x/n)} \exp\left(-\lambda_i \frac{x}{u_j}\right) \exp\left(\frac{-h^2}{2\sigma_{zk}^2}\right) \tag{1.3}$$

where:

X_i - Air concentration of radionuclide i, $\mu Ci/m^3$.

f_{jk} - Joint relative frequency of occurrence of winds in wind speed class j, stability class k, blowing toward this exposure point, expressed as a fraction.

Q_i - Average release rate of radionuclide i, $\mu Ci/s$.

P - Fraction of radionuclide remaining in plume.

Σ_{zk} - Vertical dispersion coefficient for stability class k (m).

u_j - Midpoint value of wind speed class interval j, m/s.

x - Downwind distance, m.

n - Number of sectors, 16.

λ_i - Radioactive decay coefficient of radionuclide i, s^{-1}.

$2\pi x/n$ - Sector width at point of interest, m.

h - Stack height (73 meters).

σ_{zk}^2 - Vertical dispersion coefficient of stability class k.

For determining the total body dose:

$$D_{TB} = \sum_i X_i \, DFB_i \tag{1.4}$$

where:

D_{TB} = Total body dose mrem/y.

X_i = Air concentration of radionuclide i, $\mu Ci/m^3$.

DFB_i = Total body dose factor due to gamma radiation, mrem/y per $\mu Ci/m^3$ (Table 1.5).

For determining the skin dose:

$$D_s = \sum_i X_i \, (DFS_i + 1.11 \, DFY_i) \tag{1.5}$$

where:

D_s = Skin dose mrem/y.

X_i = Air concentration of radionuclide i, $\mu Ci/m^3$.

DFS_i = Skin dose factor due to beta radiation, mrem/y per $\mu Ci/m^3$ (Table 1.5).

1.11 = The average ratio of tissue-to-air energy absorption coefficients, mrem/mrad.

DFY_i = Gamma-to-air dose factor for radionuclide i, mrad/y per $\mu Ci/m^3$ (Table 1.5).

For determining dose rate to a point in air:

$$D_a = \sum_i X_i \, (DFY_i \text{ or } DFB_i) \tag{1.6}$$

where:

D_a = Air dose mrad/y.

DFB = Air dose factor for beta radiation (Table 1.5).

c. Assumptions for Radioiodine and Particulate Doses

1) Dose is to be calculated for the critical organ, thyroid, and the critical age groups (adult, teen, child, infant), infant (milk) and child (green, leafy vegetables).

2) Exposure pathways from iodines and particulates are milk ingestion, ground contamination, green leafy vegetables from home gardens and inhalation.

3) The radioiodine and particulate mix is based on analyses of activity released over the period of interest. The historically observed source term is given in Table 1.1.

4) Basic radionuclide data are given in Table 1.2.

5) All releases are treated as elevated (73 m).

6) Annual average X/Qs are given in Table 1.3.

7) Raw meteorological data for elevated releases consist of wind speed and direction measurements which were obtained at 71 m.

8) Dose is to be evaluated at the potential offsite exposure points where maximum doses to man are expected to exist.

9) Actual or conservative cow, goat and garden locations are used.

10) Potential maximum exposure points are described in Table 1.4.

11) Terrain effects are not considered.

12) Plume depletion and radioactive decay are considered for air-concentration calculations.

13) Radioactive decay is considered for ground-concentration calculations.

14) Milk cows and goats obtain 100% of their food from pasture grass May through October of each year. Use default values of 0.58 for cows and 0.67 for goats for fraction of year on pasture

15) Credit is taken for shielding by residence (factor of 0.7).

d. Equations For Iodine and Particulate Doses

To calculate the dose for any one of the potential maximum-exposure points, the equations I.7 through I.10 and I.12 through I.16 are used.

1) Inhalation

Equation for calculating air concentration, X, is the same as Noble Gas (Equation I.3).

For determining the organ dose rate:

$$D_I = 1 \times 10^6 \sum_i X_i \ DFI_i \ BR \tag{I.7}$$

where:

D_I = Organ dose due to inhalation, mrem/y.

X_i = Air concentration of radionuclide i, μCi/m³.

DFI_i = Inhalation dose factor, mrem/Pci (Table 1.7).

BR = Breathing rate 1,400 m³/y, infant; 3,700 m³/y, child; or 8,000 m³/y, adult and teen.

1×10^6 = pCi/μCi conversion factor.

2) Ground Contamination

For determining the ground concentration of any nuclide:

$$G_i = 3.15 \times 10^7 \sum_{k=1}^{7} \frac{f_k \ Q_i \ DR_k}{(2\pi x/n)\ \lambda_i} [1 - \exp(-\lambda_i t_b)] \left[\exp\left(\frac{-h^2}{2\sigma^2_{zk}}\right)\right] \tag{I.8}$$

where:

G_i = Ground concentration of radionuclide i, μCi/m².

k = Stability class (7 classes, A through G).

f_k = Joint relative frequency of occurrence of winds in stability class k blowing toward this exposure point, expressed as a fraction.

Q_i = Average release rate of radionuclide i, μCi/s.

DR_k - Relative deposition rate, 1/m.

x - Downwind distance, m.

n - Number of sectors, 16.

$2\pi x/n$ - Sector width at point of interest, m.

λ_i - Radioactive decay coefficient of radionuclide i, y^{-1}.

t_b - Time for buildup of radionuclides on the ground, 35 y.

3.15×10^7 - s/y conversion factor.

h - Stack height (73 m).

σ^2_{zk} - Vertical dispersion coefficient (m^2) of stability class.

For determining the total body or organ dose from ground contamination:

$$D_G = (8{,}760)(1 \times 10^6)(0.7) \sum_i G_i \, DFG_i \qquad (1.9)$$

where:

D_G - Dose due to ground contamination, mrem/y.

G_i - Ground concentration of radionuclide i, $\mu Ci/m^2$.

DFG_i - Dose factor for standing on contaminated ground, mrem/h per pCi/m^2 (Table 1.8).

8,760 - Occupation time, h/y.

1×10^6 - pCi/μCi conversion factor.

0.7 - Shielding factor accounting for a distance of 1.0 meter above ordinary ground, dimensionless.

3) **Milk and Vegetation Ingestion**

For determining the concentration of any nuclide (except C-14 and H-3) in and on vegetation:

$$CV_i = 3{,}600 \sum_{k=1}^{7} \frac{f_k Q_i \; DR_k}{(2\pi x/n)} \left(\frac{r[1 - \exp(-\lambda_{Ei} t_e)]}{Y_v \lambda_{Ei}} + \frac{B_{iv} [1 - \exp(-\lambda_i t_b)]}{P \lambda_i} \right) \exp\left(\frac{-h^2}{2\sigma^2_{zk}}\right) \exp(-\lambda_i t_h) \qquad (I.10)$$

where:

CV_i - Concentration of radionuclide i in and on vegetation, μCi/kg.

k - Stability class.

f_k - Frequency of this stability class and wind direction combination, expressed as a fraction.

Q_i - Average release rate of radionuclide i, μCi/s.

DR_k - Relative deposition rate as a function of wind speed, stability class and downwind distance, m^{-1} (Figures 7 through 10 of Regulatory Guide 1.111).

x - Downwind distance, m.

n - Number of sectors, 16.

$2\pi x/n$ - Sector width at point of interest, m.

r - Fraction of deposited activity retained on vegetation (1.0 for iodines, 0.2 for particulates).

λ_{Ei} - Effective removal rate constant, $\lambda_{Ei} = \lambda_i + \lambda_w$, where λ_i is the radioactive decay coefficient, h^{-1}, and λ_w is a measure of physical loss by weathering ($\lambda_w = .0021\ h^{-1}$).

t_e - Period over which deposition occurs, 720 h.

Y_v - Agricultural yield, 0.7 kg/m^2.

B_{iv} - Transfer factor from soil to vegetation of radionuclide i (Table 1.6).

λ_i - Radioactive decay coefficient of radionuclide i, h^{-1}.

t_b - Time for buildup of radionuclides on the ground, 3.07×10^5 h (35y).

P - Effective surface density of soil, 240 kg/m^2.

3,600 - s/h conversion factor.

h - Stack height (73 m).

σ_z - Vertical dispersion coefficient (m).

t_h - Holdup time between harvest and consumption of food, 0 h for pasture grass or 2,160 h for storage feed.

For determining the concentration of C-14 in vegetation:

$$CV_{14} = 1 \times 10^3 \, X_{14} \, (0.11/0.16) \qquad (I.11)$$

where:

CV_{14} - Concentration of C-14 in vegetation, μCi/kg.

X_{14} - Air concentration of C-14, μCi/m^3.

0.11 - Fraction of total plant mass that is natural carbon.

0.16 - Concentration of natural carbon in the atmosphere, g/m^3.

1×10^3 - g/kg conversion factor.

For determining the concentration of H-3 in vegetation:

$$CV_T = 1 \times 10^3 X_T (0.75)(0.5/H) \quad (1.11a)$$

where:

- CV_T - Concentration of H-3 in vegetation, μCi/kg.
- X_T - Air concentration of H-3, $\mu Ci/m^3$.
- 0.75 - Fraction of total plant mass that is water.
- 0.5 - Ratio of tritium concentration in plant water to tritium concentration in atmospheric water.
- H - Absolute humidity of the atmosphere, g/m^3.
- 1×10^3 - g/kg conversion factor.

For determining the concentration of any nuclide in cow's or goat's milk:

$$CM_i = CV_i\, FM_i\, Q_f \exp(-\lambda_i t_f) \quad (1.12)$$

where:

- CM_i - Concentration of radionuclide i (including C-14 and H-3) in milk, μCi/l.
- CV_i - Concentration of radionuclide i in and on vegetation, μCi/kg.
- FM_i - Transfer factor from feed to milk for radionuclide i, d/l (Table 1.6).
- Q_f - Amount of feed consumed by the milk animal per day, cow - 50 kg/d, goat - 6 kg/d.
- λ_i - Radioactive decay coefficient of radionuclide i, d^{-1}.
- t_f - Transport time of activity from feed to milk to receptor, two days.

For determining the organ dose from ingestion of green leafy vegetables and milk:

$$D = 1 \times 10^{6} \sum_{i} CM_i \, DF_i \, UM \qquad (I.13)$$

where:

D - Organ dose due to ingestion, mrem/y.

CM_i - Concentration of radionuclide i in vegetables or milk, μCi/kg (or liters).

DF_i - Ingestion dose factor, mrem/pCi (Table 2.1).

UM - Ingestion rate for milk, 330 l/y - infant and child, 400 l/y - teen, and 310 1/y - adult.

Ingestion rate for vegetables, 26 kg/y - child, 42 kg/y - teen, and 64 kg/y - adult.

1×10^{6} - pCi/μCi conversion factor.

4) Meat Ingestion (Beef)

To calculate the concentration of a nuclide in animal flesh:

$$C_{fi} = F_{fi} \, CV_i \, Q_f \exp(-\lambda_i t_s) \qquad (I.14)$$

where:

C_{fi} - Concentration of nuclide i in the animal flesh, pCi/kg.

F_{fi} - Fraction of animal's daily intake which appears in each kg of flesh, days/kg (Table 1.6).

CV_i - Concentration of radionuclide i in the animal's feed (Equation I.10).

Q_f - Amount of feed consumed by the cow per day, 50 kg/d.

t_s - Average time from slaughter to consumption, 20 days.

To determine the organ dose from ingestion of beef:

$$D_F = \sum_i C_{fi} D_{fi} U_f \tag{II.15}$$

where:

D_{fi} - Ingestion dose factor for age group, mrem/pCi (Table 2.1) for nuclide i.

U_f - Ingestion rate of meat for age group, kg/y (child - 41, teen - 65, adult - 110).

5) Organ Dose Rates

For determining the total body, organ and/or thyroid dose rate from iodines and particulates:

$$D = D_I + D_G + D_M + D_V + D_F \tag{II.16}$$

where:

D - Total organ dose, mrem/y.

D_I - Dose due to inhalation, mrem/y.

D_G - Dose due to ground contamination, mrem/y.

D_M - Dose due to milk ingestion, mrem/y.

D_V - Dose due to vegetable ingestion, mrem/y.

D_F - Dose due to meat ingestion, mrem/y.

a. The maximum organ dose rate, maximum total body dose rate, maximum skin dose rate plus beta and gamma air doses are used to calculate design basis quantities as described in Section II, part 1.2.2, as follows.

1.2.2 <u>Design Basis Quantities</u>

The design basis quantity of a radionuclide emitted to the atmosphere is the amount of that nuclide, when released in one year, which would result in a dose not exceeding any of the following:

a. 15 millirems to any organ of an individual from iodines and particulates with half life greater than 8 days.

b. 20 millirad air dose for beta radiation from noble gas (see note below).

c. 10 millirad air dose for gamma radiation from noble gas (see note below).

Design basis quantity (Ci) is the smallest value for each nuclide, calculated by dividing the dose limits (a through c above) by the appropriate dose calculated per ODCM Section II, part 1.2.1; the result then is multiplied by the amount of radionuclide (Ci) used to conservatively estimate dose (either the source term listed in Table 1.1 (or assumed a hypothetical 1.0 Ci/year for nuclides not listed):

$$DBQ = \frac{D_{AI}}{D_c} (C_c) \tag{1.17}$$

where:

D_{AI} - Appendix I dose limit (mrem or mrad).

D_c - Calculated dose (mrem or mrad).

C_c - Quantity of nuclide resulting in dose D_c (Ci).

DBQ - Design Basis Quantity (Ci). The limiting values for Design Basis Quantities for radionuclides released to the atmosphere are given in Table 1.9.

NOTE: For conservative calculations the DBQs listed in Table 1.9 are based on:

- 15 millirems to any organ of an individual from iodines and particulates with a half life greater than eight days.
- 15 millirad air dose for beta radiation from noble gas.
- 5 millirad air dose for gamma radiation from noble gas.

The inverse of the ratio C_c/D_c in the above equation (ie, D_c/C_c) is a useful value, since it represents the most limiting dose per unit quantity of each nuclide released.

1.3 DESIGN BASIS QUANTITY (DBQ) LIMITS

1.3.1 Design Basis Quantity Fraction

Per Specification 3.3 of ODCM Section I, the cumulative DBQ fraction for nuclides released is summed at least every 31 days to assure that the sum of the fractions of all nuclides released does not exceed 1.0 year to date and 0.5 in any calendar quarter.

$$\sum_i \frac{A_i}{(DBQ)_i} < 1.0 \qquad (I.18)$$

1.3.2 Exceeding DBQ Limits

The DBQ is a very conservative estimate of activity which could give doses at 10 CFR 50, Appendix I limits. Because different organs are summed together and doses to different people are summed, the DBQ typically overestimates dose by about a factor of five. Thus, if calculations of the DBQ fraction exceed 1.0 for year to date or 0.5 for the quarter, Technical Specifications probably still would not be exceeded. However, further discretionary releases should be deferred until an accurate assessment of dose is made by use of the NRC GASPAR computer code (either by running the code or by normalizing a prior run's dose output on a nuclide by nuclide basis. The computer run will utilize the annual average joint frequency meteorological data based on not less than 3 years of meteorological measurement. Where appropriate, seasonal adjustments will be applied to obtain realistic dose estimates since both recreational and agricultural activities can vary greatly in relation to season of the year.

1.3.3 Releasing Radionuclides Not Listed in Table 1.9

Table 1.9 contains all nuclides identified to date as routine constituents of gaseous releases at Big Rock Point Plant, plus those common to boiling water reactors in general, even if not previously detected at Big Rock Point. From time to time, however, other nuclides may be detected.

If the unlisted nuclide constitutes less than 10% of the EC-fraction for the release, and all unlisted nuclides total less than 25% of the EC-fraction, the nuclide may be considered not present. If the unlisted nuclide constitutes greater than 10% of the EC-fraction, or all unlisted nuclides together constitute greater than 25%, then each nuclide should be assigned a DBQ equal to the most conservative value listed for the physical form of the nuclide involved (noble gas, halogen or particulate).

Should a nuclide not listed in Table 1.9 begin to appear in significant quantities on a routine basis, revision to this ODCM should be made in order to include a design basis quantity specific to that nuclide.

2.4 OPERABILITY OF LIQUID RADWASTE EQUIPMENT

The Big Rock Point liquid radwaste system is designed to reduce the radioactive materials in liquid wastes prior to their discharge by filtration, ion exchange, decay or shipment for disposal so that radioactivity in liquid effluent releases to unrestricted areas will not exceed Requirement 2.3.1 of ODCM Section I. Maintaining the cumulative DBQ fraction of releases assures compliance with this requirement. In addition, more than 30 years of operating experience has shown that design basis quantities never have been exceeded.

2.5 OFFSITE RELEASE RATE

10 CFR 50.36a requires that the release of radioactive materials be kept as low as is reasonably achievable. Appendix I to 10 CFR 50 provides the numerical guidelines on limiting conditions for operations to meet the as low as is reasonably achievable requirement.

The LADTAP code has been run to determine the dose due to drinking water at plant discharge concentration (800 x nearest lake water drinking water intake concentration). The source term used is given in Table 1.1. The most limiting dose due to a hypothetical individual drinking this water is 4.89E-01 mrem, whole body. The release rate which would result in a dose rate equivalent to 500 mrem/y (upon which the 10 times concentration limit of Requirement 2.2.1 of ODCM Section I is based) is the Curies/Year given in Table 1.1 (8.94) times 500/.489 or 9141 Ci/Yr - 2.9E-04 Ci/sec (~290 uCi/sec) if continued for a full year.

The above calculation is informational, supplied at NRC request for inclusion in the ODCM. The release rate value is not used in any plant calculations or related to any ODCM Requirements. The calculation is based upon exposure using the drinking water pathway and an average historical release from Big Rock Point.

Annual analyses are run for the report specified part 1.5 of ODCM Section III. LADTAP is used to calculate estimates of dose to the total body and limiting organs.

Radionuclides of highest dose consequence will remain predominately Cs-137 and Co-60 throughout the decommissioning interval. Iodine-131, which has been an important dose contributor during power operations, will not be present in effluents in detectable concentrations due to decay prior to implementation of this ODCM revision (minimum of 11.6 halflives of decay will have occurred. Cs-134 (also important during plant operation) will decay to significantly lower levels as decommissioning progresses.

BIOACCUMULATION FACTORS
Table 2.0

(μCi/g per μCi/ml)

Element	Freshwater Fish
H	9.0E-01
C	4.6E+03
Na	1.0E+02
P	3.0E+03
Cr	2.0E+02
Mn	4.0E+02
Fe	1.0E+02
Co	5.0E+01
Ni	1.0E+02
Cu	5.0E+01
Zn	2.0E+03
Br	4.2E+02
Rb	2.0E+03
Sr	3.0E+01
Y	2.5E+01
Zr	3.3E+00
Nb	3.0E+04
Mo	1.0E+01
Tc	1.5E+01
Ru	1.0E+01
Rh	1.0E+01
Te	4.0E+02
I	1.5E+01
Cs	2.0E+03
Ba	4.0E+00
La	2.5E+01
Ce	1.0E+00
Pr	2.5E+01
Nd	2.5E+01
W	1.2E+03
Np	1.0E+01

TABLE 2.1

ADULT INGESTION DOSE FACTORS
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07
C 14	2.84E-06	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07
NA 24	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06
P 32	1.93E-04	1.20E-05	7.46E-06	NO DATA	NO DATA	NO DATA	2.17E-05
CR 51	NO DATA	NO DATA	2.66E-09	1.59E-09	5.86E-10	3.53E-09	6.69E-07
MN 54	NO DATA	4.57E-06	8.72E-07	NO DATA	1.3[illegible]E-06	NO DATA	1.40E-05
MN 56	NO DATA	1.15E-07	2.04E-08	NO DATA	1.46E-07	NO DATA	3.67E-06
FE 55	2.75E-06	1.90E-06	4.43E-07	NO DATA	NO DATA	1.06E-06	1.09E-06
FE 59	4.34E-06	1.02E 05	3.91E-06	NO DATA	NO DATA	2.85E-06	3.40E-05
CO 58	NO DATA	7.45E-07	1.67E-06	NO DATA	NO DATA	NO DATA	1.51E-05
CO 60	NO DATA	2.14E-06	4.72E-06	NO DATA	NO DATA	NO DATA	4.02E-05
NI 63	1.30E-04	9.01E-06	4.36E-06	NO DATA	NO DATA	NO DATA	1.88E-06
NI 65	5.28E-07	6.86E-08	3.13E-08	NO DATA	NO DATA	NO DATA	1.74E-06
CU 64	NO DATA	8.33E-08	3.91E-08	NO DATA	2.10E-07	NO DATA	7.10E-06
ZN 65	4.84E-06	1.54E-05	6.96E-06	NO DATA	1.03E-05	NO DATA	9.70E-06
ZN 69	1.03E-08	1.97E-08	1.37E-09	NO DATA	1.28E-08	NO DATA	2.96E-09
BR 83	NO DATA	NO DATA	4.02E-08	NO DATA	NO DATA	NO DATA	5.79E-08
BR 84	NO DATA	NO DATA	5.21E-08	NO DATA	NO DATA	NO DATA	4.09E-13
BR 85	NO DATA	NO DATA	2.14E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	2.11E-05	[illegible].83E-06	NO DATA	NO DATA	NO DATA	4.16E-06
RB 88	NO DATA	6.05E-08	3.21E-08	NO DATA	NO DATA	NO DATA	8.36E-19
RB 89	NO DATA	4.01E-08	2.82E-08	NO DATA	NO DATA	NO DATA	2.33E-21
SR 89	3.08E-04	NO DATA	8.84E-06	NO DATA	NO DATA	NO DATA	4.94E-05
SR 90	7.58E-03	NO DATA	1.86E-03	NO DATA	NO DATA	NO DATA	2.19E-04
SR 91	5.67E-06	NO DATA	2.29E-07	NO DATA	NO DATA	NO DATA	2.70E-05
SR 92	2.15E-06	NO DATA	9.30E-08	NO DATA	NO DATA	NO DATA	4.26E-05
Y 90	9.62E-09	NO DATA	2.58E-10	NO DATA	NO DATA	NO DATA	1.02E-04
Y 91m	9.09E-11	NO DATA	3.52E-12	NO DATA	NO DATA	NO DATA	2.67E-10
Y 91	1.41E-07	NO DATA	3.77E-09	NO DATA	NO DATA	NO DATA	7.76E-05
Y 92	8.45E-10	NO DATA	2.47E-11	NO DATA	NO DATA	NO DATA	1.48E-05

TABLE 2.1

ADULT INGESTION DOSE FACTORS
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	2.68E-09	NO DATA	7.40E-11	NO DATA	NO DATA	NO DATA	8.50E-05
ZR 95	3.04E-08	9.75E-09	6.60E-09	NO DATA	1.53E-08	NO DATA	3.09E-05
ZR 97	1.68E-09	3.39E-10	1.55E-10	NO DATA	5.12E-10	NO DATA	1.05E-04
NB 95	6.22E-09	3.46E-09	1.86E-09	NO DATA	3.42E-09	NO DATA	2.10E-05
MO 99	NO DATA	4.31E-06	8.20E-07	NO DATA	9.76E-06	NO DATA	9.99E-06
TC 99m	2.47E-10	6.98E-10	8.89E-09	NO DATA	1.06E-08	3.42E-10	4.13E-07
TC101	2.54E-10	3.66E-10	3.59E-09	NO DATA	6.59E-09	1.87E-10	1.10E-21
RU103	1.85E-07	NO DATA	7.97E-08	NO DATA	7.06E-07	NO DATA	2.16E-05
RU105	1.54E-08	NO DATA	6.08E-09	NO DATA	1.99E-07	NO DATA	9.42E-06
RU106	2.75E-06	NO DATA	3.48E-07	NO DATA	5.31E-06	NO DATA	1.78E-04
AG110m	1.60E-07	1.48E-07	8.79E-08	NO DATA	2.91E-07	NO DATA	6.04E-05
TE125m	2.68E-06	9.71E-07	3.59E-07	8.06E-07	1.09E-05	NO DATA	1.07E-05
TE127m	6.77E-06	2.42E-06	8.25E-07	1.73E-06	2.75E-05	NO DATA	2.27E-05
TE127	1.10E-07	3.95E-08	2.38E-08	8.15E-08	4.48E-07	NO DATA	8.68E-06
TE129m	1.15E-06	4.29E-06	1.82E-06	3.95E-06	4.80E-05	NO DATA	5.79E-05
TE129	3.14E-08	1.18E-08	7.65E-09	2.41E-08	1.32E-07	NO DATA	2.37E-08
TE131m	1.73E-06	8.46E-07	7.05E-07	1.34E-06	8.57E-06	NO DATA	8.40E-05
TE131	1.97E-08	8.23E-09	6.22E-09	1.62E-08	8.63E-08	NO DATA	2.79E-09
TE132	2.52E-06	1.63E-06	1.53E-06	1.80E-06	1.57E-05	NO DATA	7.71E-05
I 130	7.56E-07	2.23E-06	8.80E-07	1.89E-04	3.48E-06	NO DATA	1.92E-06
I 131	4.16E-06	5.95E-06	3.41E-06	1.95E-03	1.02E-05	NO DATA	1.57E-06
I 132	2.03E-07	5.43E-07	1.90E-07	1.90E-05	8.65E-07	NO DATA	1.02E-07
I 133	1.42E-06	2.47E-06	7.53E-07	3.63E-04	4.31E-06	NO DATA	2.22E-06
I 134	1.06E-07	2.88E-07	1.03E-07	4.99E-06	4.58E-07	NO DATA	2.51E-10
I 135	4.43E-07	1.16E-06	4.28E-07	7.65E-05	1.86E-06	NO DATA	1.31E-06
CS134	6.22E-05	1.48E-04	1.21E-04	NO DATA	4.79E-05	1.59E-05	2.59E-06
CS136	6.51E-06	2.57E-05	1.85E-05	NO DATA	1.43E-05	1.96E-06	2.92E-06
CS137	7.97E-05	1.09E-04	7.14E-05	NO DATA	3.70E-05	1.23E-05	2.11E-06
CS138	5.52E-08	1.09E-07	5.40E-08	NO DATA	8.01E-08	7.91E-09	4.65E-13
BA139	9.70E-08	6.91E-11	2.84E-09	NO DATA	6.46E-11	3.92E-11	1.72E-07

1.4 GASEOUS RADWASTE TREATMENT SYSTEM OPERATION

1.4.1 System Description

The gaseous radwaste system consists of a delay line for condenser off-gas which provides approximately 30 minutes of decay time prior to release via the 73 m stack. A flow diagram of gaseous waste release paths is shown in Figure 1-1.

Condenser off-gas represents more than 95% of the total gaseous source term during power operation, but will not contribute to releases during decommissioning since the condenser will not be in use. Other sources not present during decommissioning are gland seal condenser exhaust and miscellaneous turbine building system steam leakage. Decommissioning releases will consist primarily of containment ventilation and to a lesser degree, radwaste system vents. These sources will be ducted to the stack for release or released via other monitored pathways, depending on the phase of decommissioning in progress.

1.4.2 Determination of Satisfactory Operations

Operability requirements for the gaseous waste treatment system (offgas delay line) are not specified. This is because, as discussed above, this release source is not present during decommissioning.

1.5 OFFSITE RELEASE RATE

10 CFR 50.36a requires that the release of radioactive materials be kept as low as reasonably achievable. However, the section further states that the licensee is permitted the flexibility of operation, to release quantities of material at higher rates than a small percentage of 10 CFR 20 limits but not exceeding those limits under unusual operating conditions. Appendix I to 10 CFR 50 provides the numerical guidelines on limiting conditions for operation to meet the as low as reasonably achievable requirement.

The GASPAR code has been run to determine a conservative relationship between release rate in Ci/sec and annual dose due to external radiation and inhalation. The source term used is listed in Table 1.1. The meteorology data is given in Table 1.3. Dose using annual average meteorology, to the individual with most limiting offsite dose (whole body) assumed to be residing at the residence with highest X/Q, is 0.105 mrem for one year. The release rate which would result in a dose equivalent of 500 mrem/y (using the total body limit upon which ODCM Section I Requirement 3.1 is based) is the Curies/Year given in Table 1.1 (1.29E04) multiplied by 500/.105 or 1.95 Ci/sec.

The above calculation is informational only. Exposure calculations for releases over specific periods are described in Section II, part 1.2.

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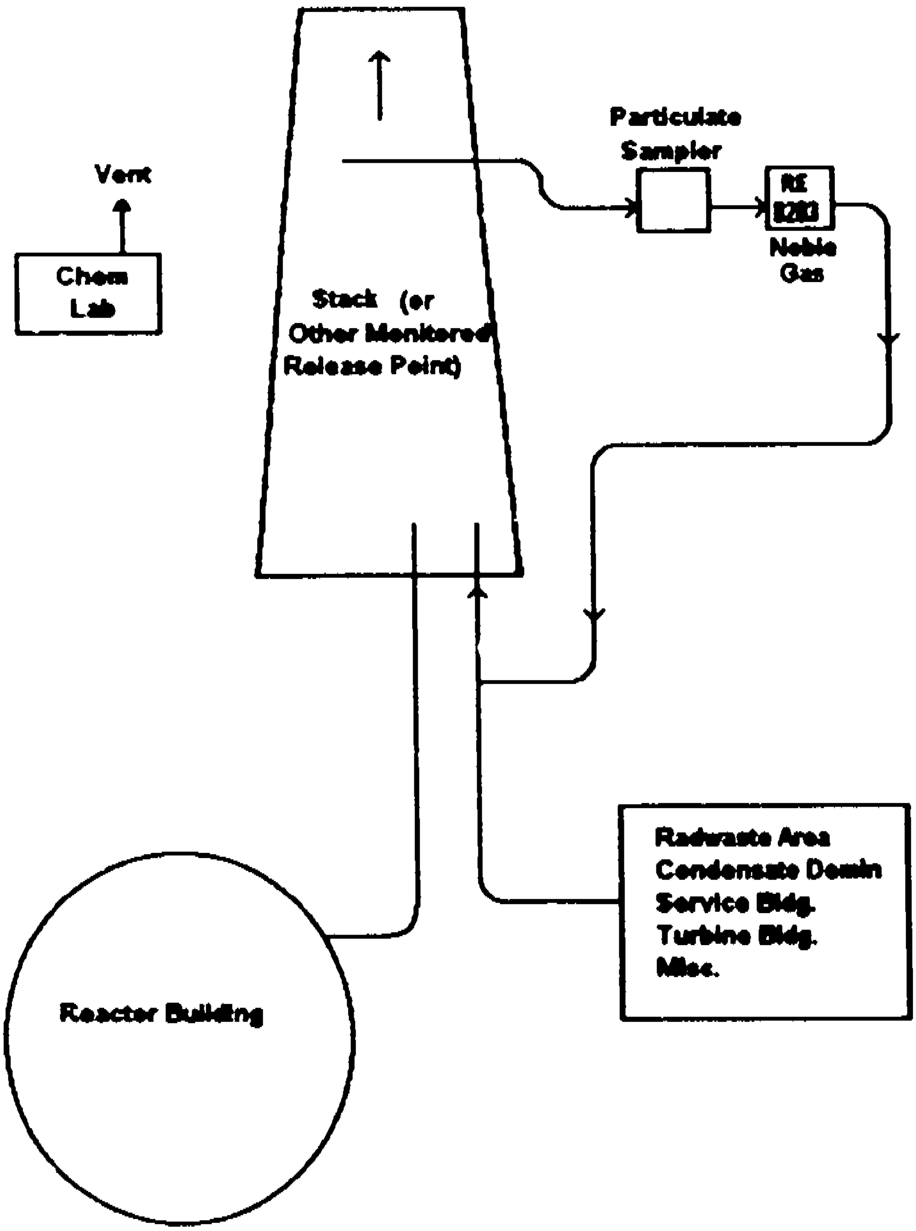
The above calculation is informational only. Exposure calculations for releases over specific periods are described in Section II, part 1.2.

<u>FIGURE 1-1</u>
BIG ROCK POINT GASEOUS
EFFLUENT FLOW PATHS



1.6 PARTICULATE SAMPLING

Particulate samples are obtained from the continuous sample stream pulled from the plant stack or from other alternative release points utilized through the course of decommissioning. Samples typically are obtained to represent the integrated release from the plant.

Gamma, beta and alpha counting is performed on the particulate filters. Beta yields of the gamma isotopes detected on particulate filters are applied to determine "identified" beta, and the "identified" count rate is subtracted from the observed count rate to give "unidentified" beta. The "unidentified" beta is assumed to be Sr-90 until results on an optional analysis specific to Sr-90 (from a quarterly composite of filters) are obtained. Similarly, alpha activity not identified as natural radium, thorium or their daughters is assumed as Pu-239 until optional results of quarterly composite analysis are available.

Monitoring for radioiodine is not required because this ODCM does not go into effect until at least 93 days post-shutdown when Iodine-131 has decayed a minimum of 11.6 halflives.

1.7 NOBLE GAS MONITORING AND SAMPLING

While spent fuel is present in the spent fuel pool, noble gas release rates will either be monitored by the noble gas monitor or be estimated by means of periodic containment air analyses of concentrations times volume of air exhausted from containment over the appropriate interval. With fuel present, there is potential for low levels of noble gas release. However, after the first few months of shutdown, only Kr-85 would be present in detectable quantities.

1.8 TRITIUM SAMPLING

Tritium has a low dose consequence to the public because of the very small fraction of allowable quantity which is available for release. The major contributors to tritium effluents are evaporation from the fuel pool and reactor cavity (when flooded). Because of the low dose impact, and due to the continuous reduction in tritium source term with tritium production due to reactor operation ceased, gaseous tritium sampling will not be required. Gaseous tritium release will be estimated using conservative evaporation rate calculations from the fuel pool and reactor cavity during intervals that water containing tritium is utilized.

TABLE 1.1
BIG ROCK POINT GASEOUS AND LIQUID SOURCE TERMS, CURIES/YEAR[1]

Nuclide	Gaseous[2]	Liquid[2]
H-3	1.21E+01	8.63E+00
N-13	1.53E+03	NA
Na-24	3.62E-04	1.12E-08
Cr-51	2.82E-04	8.84E-03
Mn-54	5.50E-06	2.80E-02
Mn-56	1.70E-04	NA
Co-58	1.65E-06	6.17E-04
Fe-59	2.81E-06	9.06E-03
Co-60	1.89E-04	4.21E-02
Zn-65	3.16E-06	9.01E-04
Br-82	8.11E-03	NA
Kr-83m	2.61E+02	NA
Kr-85	9.55E-01	NA
Kr-85m	3.12E+02	NA
Kr-87	1.19E+03	NA
Kr-88	7.80E+02	NA
Kr-89	6.96E+02	NA
Sr-89	NA	2.27E-04
Kr-90	7.76E+02	NA
Sr-90	NA	2.22E-03
Kr-91	6.68E+00	NA
Sr-91	5.61E-03	NA
Sr-92	NA	1.54E-08
Nb-95	1.91E-08	NA
Mo-99	3.10E-06	NA
Ag110m	1.57E-05	6.88E-05
Sb-124	NA	4.01E-04
I-131	1.94E-03	1.57E-04
Xe-131m	4.38E-01	NA
I-132	8.07E-03	NA
I-133	1.96E-02	NA
Xe-133	2.01E+02	8.86E-05
Xe-133m	6.00E+00	NA
Cs-134	4.04E-07	1.75E-02
I-134	1.24E-02	NA
I-135	3.00E-02	NA
Xe-135	1.11E+03	NA
Xe-135m	1.16E+03	NA
Cs-136	4.74E-06	NA
Cs-137	1.51E-04	2.04E-01
Xe-137	1.11E+03	NA
Cs-138	3.17E-01	NA
Xe-138	6.03E+03	NA
Ba-139	1.32E-03	NA
Xe-139	1.04E+03	NA
Ba-140	1.86E-03	NA
La-140	7.80E-03	5.04E-05
Xe-140	7.23E+01	NA
Hg-203	1.32E-08	NA
Np-239	1.44E-04	NA
Unidentified Beta	2.42E-03	6.76E-02

(1) Data derived from taking the effluents released during Jan-June 1980 through July-December 1983 and dividing by 4.

(2) Historic (operating) source term. Values listed as NA have not been observed at detectable levels in these waste streams.

TABLE 1.2

BASIC RADIONUCLIDE DATA

	NUCLIDE	HALF-LIFE (days)	LAMBDA (1/s)	[1] BETA (MEV/DIS)	[1] GAMMA (MEV/DIS)
1	Tritium	4.49E 03	1.79E-09	5.68E-03	0.0
2	C-14	2.09E 06	3.84E-12	4.95E-02	0.0
3	N-13	6.94E-03	1.16E-03	4.91E-01	1.02E 00
4	O-19	3.38E-04	2.39E-02	1.02E 00	1.05E 00
5	F-18	7.62E-02	1.05E-04	2.50E-01	1.02E 00
6	NA-24	6.33E-01	1.273-05	5.55E-01	4.12E 00
7	P-32	1.43E 01	5.61E-07	6.95E-01	0.0
8	AR-41	7.63E-02	1.05E-04	4.64E-01	1.28E 00
9	CR-51	2.78E 01	2.89E-07	3.86E-03	3.28E-02
10	MN-54	3.03E 02	2.65E-08	3.80E-03	8.36E-01
11	MN-56	1.07E-01	7.50E-05	8.29E-01	1.69E 00
12	FE-59	4.50E 01	1.78E-07	1.18E-01	1.19E 00
13	CO-58	7.13E 01	1.12E-07	3.41E-02	9.78E-01
14	CO-60	1.92E 03	4.18E-09	9.68E-02	2.50E 00
15	ZN-69m	5.75E-01	1.39E-05	2.21E-2	4.16E-01
16	ZN-69	3.99E-02	2.03E-04	3.19E-01	0.0
17	BR-84	2.21E-02	3.63E-04	1.28E 00	1.77E 00
18	BR-85	2.08E-03	3.86E-03	1.04E 00	6.60E-02
19	KR-85m	1.83E-01	4.38E-05	2.53E-01	1.59E-01
20	KR-85	3.93E 03	2.04E-09	2.51E-01	2.21E-03
21	KR-87	5.28E-02	1.52E-04	1.32E 00	7.93E-01
22	KR-88	1.17E-01	6.86E-05	3.61E-01	1.96E 00
23	KR-89	2.21E-03	3.63E-03	1.36E 00	1.83E 00
24	RB-88	1.24E-02	6.47E-04	2.08E 00	6.28E-01
25	RB-89	1.07E-02	7.50E-04	1.01E 00	2.05E 00
26	SR-89	5.20E 01	1.54E-07	5.83E-01	8.45E-05
27	SR-90	1.03E 04	7.79E-10	1.96E-01	0.0
28	SR-91	4.03E-01	1.99E-05	6.50E-01	6.95E-01
29	SR-92	1.13E-01	7.10E-05	1.95E-01	1.34E 00
30	SR-93	5.56E-03	1.44E-03	9.20E-01	2.24E 00
31	Y-90	2.67E 00	3.00E-06	9.36E-01	0.0
32	Y-91m	3.47E-02	2.31E-04	2.73E-02	5.30E-01
33	Y-91	5.88E 01	1.36E-07	6.06E-01	3.61E-03
34	Y-92	1.47E-01	5.46E-05	1.44E 00	2.50E-01
35	Y-93	4.29E-01	1.87E-05	1.17E 00	8.94E-02
36	ZR-95	6.50E 01	1.23E-07	1.16E-01	7.35E-01

TABLE 1.2

(continued)

BASIC RADIONUCLIDE DATA

	NUCLIDE	HALF-LIFE (days)	LAMBDA (l/s)	[1] BETA (MEV/DIS)	[1] GAMMA (MEV/DIS)
37	NB-95m	3.75E 00	2.14E-06	1.81E-01	6.06E-02
38	NB-95	3.50E 01	2.29E-07	4.44E-02	7.64E-01
39	MO-99	2.79E 00	2.87E-06	3.96E-01	1.50E-01
40	TC-99m	2.50E-01	3.21E-05	1.56E-02	1.26E-01
41	TC-99	7.74E 07	1.04E-13	8.46E-02	0.0
42	TC-104	1.25E-02	6.42E-04	1.60E 00	1.95E 00
43	RU-106	3.67E 02	2.19E-08	1.01E-02	0.0
44	TE-132	3.24E 00	2.48E-06	1.00E-01	2.33E-01
45	I-129	6.21E 09	1.29E-15	5.43E-02	2.46E-02
46	I-131	8.05E 00	9.96E-07	1.90E-01	3.81E-01
47	I-132	9.58E-02	8.37E-05	4.89E-01	2.24E 00
48	I-133	8.75E-01	9.17E-06	4.08E-01	6.02E-01
49	I-134	3.61E-02	2.22E-04	6.16E-01	2.59E 00
50	I-135	2.79E-01	2.87E-05	3.68E-01	1.55E 00
51	XE-131m	1.18E 01	6.80E-07	1.43E-01	2.01E-02
52	XE-133m	2.26E 00	3.55E-06	1.90E-01	4.16E-02
53	XE-133	5.27E 00	1.52E-06	1.36E-01	4.60E-02
54	XE-135m	1.08E-02	7.43E-04	9.58E-02	4.32E-01
55	XE-135	3.83E-01	2.09E-05	3.17E-01	2.47E-01
56	XE-137	2.71E-03	2.96E-03	1.77E 00	1.88E-01
57	XE-138	9.84E-03	8.15E-04	6.65E-01	1.10E 00
58	CS-134	7.48E 02	1.07E-08	1.63E-01	1.55E 00
59	CS-135	1.10E 09	7.29E-15	5.63E-02	0.0
60	CS-136	1.30E 01	6.17E-07	1.37E-01	2.15E 00
61	CS-137	1.10E 04	7.29E-10	1.71E-01	5.97E-01
62	CS-138	2.24E-02	3.58E-04	1.20E 00	2.30E 00
63	BA-139	5.76E-02	1.39E-04	8.96E-01	3.53E-02
64	BA-140	1.28E 01	6.27E-07	3.15E-01	1.71E-01
65	LA-140	1.68E 00	4.77E-06	5.33E-01	2.31E 00
66	CE-144	2.84E 02	2.82E-08	9.13E-02	1.93E-02
67	PR-143	1.36E 01	5.90E-07	3.14E-01	0.0
68	PR-144	1.20E-02	6.68E-04	1.21E 00	3.18E 02

[1] Average energy per disintegration values were obtained from ICRP Publication No 38, Radionuclide Transformations: Energy and Intensity of Emissions, 1983 and NUREG/CR-1413 (ORNL/NUREG-70), A Radionuclide Decay Data Base - Index and Summary Table, D. C. Kocher, May 1980.

TABLE 1.3

USNRC COMPUTER CODE - XOQDOQ, VERSION 2.0 RUN DATE: 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE 240' STACK
NO DECAY, UNDEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES FROM THE SITE

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	1.822E-08	2.270E-08	2.181E-08	2.527E-08	3.221E-08	3.378E-08	3.203E-08	3.066E-08	2.827E-08	2.607E-08	2.406E-08
SSW	1.108E-08	1.837E-08	2.898E-08	2.328E-08	2.728E-08	2.736E-08	2.571E-08	2.382E-08	2.163E-08	1.962E-08	1.768E-08
SW	1.461E-08	3.736E-08	8.107E-08	8.888E-09	1.617E-08	1.838E-08	1.848E-08	1.786E-08	1.684E-08	1.538E-08	1.427E-08
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	0.000E+00	0.000E+00	8.760E-30	7.312E-22	2.710E-15	1.815E-12	4.810E-11	3.830E-10	1.568E-09	4.088E-09	8.166E-09
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.300E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	1.742E-08	6.300E-08	1.820E-08	2.706E-08	4.150E-08	4.474E-08	4.340E-08	4.073E-08	3.706E-08	3.408E-08	3.108E-08
NE	1.101E-08	2.817E-08	3.802E-08	4.400E-08	4.837E-08	4.286E-08	3.770E-08	3.322E-08	2.828E-08	2.800E-08	2.324E-08
ENE	2.041E-08	3.364E-08	4.267E-08	4.888E-08	5.331E-08	4.803E-08	4.327E-08	3.788E-08	3.327E-08	2.842E-08	2.825E-08
E	2.734E-08	4.824E-08	5.328E-08	5.446E-08	5.101E-08	4.408E-08	3.754E-08	3.210E-08	2.771E-08	2.418E-08	2.131E-08
ESE	2.341E-08	4.274E-08	4.371E-08	4.267E-08	3.840E-08	3.286E-08	2.780E-08	2.388E-08	2.032E-08	1.801E-08	1.588E-08
SE	2.166E-08	3.271E-08	3.286E-08	3.400E-08	3.416E-08	3.100E-08	2.731E-08	2.344E-08	2.107E-08	1.887E-08	1.667E-08
SSE	1.107E-08	2.173E-08	2.810E-08	3.086E-08	3.666E-08	3.531E-08	3.202E-08	2.964E-08	2.883E-08	2.404E-08	[illegible]

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES FROM THE SITE

SECTOR	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	2.224E-08	1.676E-08	1.108E-08	7.820E-09	5.773E-09	4.468E-09	3.624E-09	3.036E-09	2.600E-09	2.267E-09	2.004E-09
SSW	1.644E-08	1.130E-08	8.402E-09	5.300E-09	3.802E-09	3.006E-09	2.428E-09	2.020E-09	1.723E-09	1.498E-09	1.318E-09
SW	1.326E-08	8.486E-09	7.184E-09	4.707E-09	3.430E-09	2.684E-09	2.156E-09	1.803E-09	1.541E-09	1.340E-09	1.182E-09
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	1.387E-08	5.328E-09	8.842E-08	1.118E-07	1.088E-07	8.848E-08	8.828E-08	7.802E-08	6.812E-08	6.164E-08	5.512E-08
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	2.861E-08	2.000E-08	1.572E-08	1.030E-08	7.524E-09	5.802E-09	4.708E-09	3.984E-09	3.423E-09	2.986E-09	2.640E-09
NE	2.003E-08	1.364E-08	8.701E-09	5.848E-09	4.172E-09	3.158E-09	2.507E-09	2.081E-09	1.736E-09	1.486E-09	1.308E-09
ENE	2.362E-08	1.601E-08	1.086E-08	6.464E-09	4.401E-09	3.370E-09	2.686E-09	2.108E-09	1.836E-09	1.570E-09	1.374E-09
E	1.887E-08	1.108E-08	6.286E-09	4.020E-09	3.410E-09	2.568E-09	2.017E-09	1.650E-09	1.388E-09	1.188E-09	1.036E-09
ESE	1.417E-08	8.072E-09	5.237E-09	3.748E-09	2.811E-09	1.888E-09	1.567E-09	1.278E-09	1.078E-09	9.247E-10	8.073E-10
SE	1.502E-08	9.888E-09	6.838E-09	4.252E-09	2.888E-09	2.283E-09	1.800E-09	1.481E-09	1.261E-09	1.077E-09	9.421E-10
SSE	1.866E-08	1.334E-08	8.778E-09	6.146E-09	4.378E-09	3.344E-09	2.677E-09	2.216E-09	1.878E-09	1.622E-09	1.423E-09

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	73.10
DIAMETER	(METERS)	1.14
EXIT VELOCITY	(METERS)	18.21
REP. WIND HEIGHT	(METERS)	71.3
BUILDING HEIGHT	(METERS)	31.4
BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	1000.0
HEAT EMISSION RATE	(CAL/SEC)	0.0

ALL ELEVATED RELEASES

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

USNRC COMPUTER CODE - XOQDOQ, VERSION 2.0   RUN DATE: 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK

RELATIVE DEPOSITION PER UNIT AREA (M**-2) BY DOWNWIND SECTORS
SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	3.420E-10	1.040E-10	1.010E-10	8.401E-11	4.430E-11	2.037E-11	8.678E-12	3.180E-12	2.087E-12	1.582E-12
SSW	3.400E-10	1.824E-10	8.860E-11	8.329E-11	4.319E-11	1.887E-11	8.710E-12	3.040E-12	1.808E-12	1.361E-12
SW	1.016E-10	8.430E-11	3.017E-11	2.479E-11	1.804E-11	7.760E-12	2.563E-12	1.144E-12	7.088E-13	5.864E-13
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	2.864E-09	2.864E-09	1.961E-09	1.317E-09	8.073E-10	4.130E-10	1.348E-10	8.822E-11	3.152E-11	2.030E-11
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	3.107E-10	2.302E-10	1.402E-10	8.808E-11	8.813E-11	3.002E-11	8.778E-12	4.108E-12	2.521E-12	1.778E-12
NE	8.856E-10	5.307E-10	3.047E-10	1.833E-10	1.314E-10	5.801E-11	1.848E-11	8.307E-12	4.942E-12	3.313E-12
ENE	8.368E-10	6.731E-10	3.207E-10	2.111E-10	1.448E-10	8.868E-11	2.108E-11	8.408E-12	5.608E-12	3.808E-12
E	1.064E-09	6.746E-10	3.198E-10	1.808E-10	1.381E-10	8.220E-11	2.077E-11	8.220E-12	5.570E-12	3.808E-12
ESE	8.840E-10	4.510E-10	2.370E-10	1.465E-10	1.808E-10	4.820E-11	1.847E-11	8.868E-12	4.283E-12	2.834E-12
SE	6.733E-10	3.811E-10	1.846E-10	1.220E-10	8.284E-11	3.776E-11	1.261E-11	5.570E-12	3.428E-12	2.402E-12
SSE	4.472E-10	2.508E-10	1.463E-10	8.284E-11	8.343E-11	2.808E-11	8.708E-12	4.387E-12	2.784E-12	2.022E-12

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	73.10
DIAMETER	(METERS)	1.14
EXIT VELOCITY	(METERS)	18.21

REP. WIND HEIGHT (METERS)		71.3
BUILDING HEIGHT (METERS)		31.4
BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	1000.0
HEAT EMISSION RATE	(CAL/SEC)	0.0

ALL ELEVATED RELEASES

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

U NRC COMPUTER CODE XOQDOQ, VERSION 2 C RUN DATE. 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK

RELATIVE DEPOSITION PER UNIT AREA (M**-2) AT FIXED POINTS BY DOWNWIND SECTORS

DIRECTION FROM SITE	DISTANCES IN MILES 0.25	0.50	0.75	1.00	1.50	2.00	2.50	3.00	3.50	4.00	4.50
S	6.810E-10	4.414E-10	3.474E-10	2.803E-10	1.834E-10	1.337E-10	1.022E-10	8.066E-11	6.480E-11	5.320E-11	4.424E-11
SSW	5.861E-10	4.667E-10	3.524E-10	2.802E-10	1.808E-10	1.308E-10	8.078E-11	7.863E-11	6.323E-11	5.102E-11	4.308E-11
SW	1.307E-10	1.102E-10	1.022E-10	8.307E-11	6.508E-11	4.826E-11	3.840E-11	3.064E-11	2.470E-11	2.038E-11	1.803E-11
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	2.018E-10	1.208E-09	2.575E-09	3.446E-09	2.807E-09	2.448E-09	1.808E-09	1.815E-09	1.322E-09	1.001E-09	8.073E-10
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	2.582E-10	2.826E-10	3.104E-10	3.208E-10	2.403E-10	1.801E-10	1.404E-10	1.104E-10	8.803E-11	7.064E-11	6.008E-11
NE	1.100E-09	1.037E-09	8.100E-10	8.008E-10	5.404E-10	4.002E-10	3.066E-10	2.403E-10	1.833E-10	1.581E-10	1.312E-10
ENE	1.210E-09	1.003E-09	8.810E-10	8.801E-10	5.782E-10	4.208E-10	3.311E-10	2.810E-10	2.112E-10	1.731E-10	1.438E-10
E	1.830E-09	1.338E-09	1.074E-09	8.867E-10	5.733E-10	4.158E-10	3.101E-10	2.484E-10	1.887E-10	1.836E-10	1.368E-10
ESE	1.482E-09	1.181E-09	8.156E-10	7.287E-10	4.478E-10	3.106E-10	2.376E-10	1.862E-10	1.483E-10	1.212E-10	1.008E-10
SE	1.018E-09	8.803E-10	8.808E-10	5.803E-10	3.806E-10	2.580E-10	1.848E-10	1.521E-10	1.218E-10	8.881E-11	8.288E-11
SSE	8.714E-10	6.503E-10	4.543E-10	3.807E-10	2.573E-10	1.807E-10	1.468E-10	1.162E-10	8.286E-11	7.821E-11	8.336E-11

DIRECTION FROM SITE	DISTANCES IN MILES 5.00	7.50	10.00	15.00	20.00	25.00	30.00	35.00	40.00	45.00	50.00
S	3.723E-11	1.882E-11	1.261E-11	8.836E-12	4.237E-12	3.131E-12	2.486E-12	2.078E-12	1.787E-12	1.584E-12	1.436E-12
SSW	3.828E-11	1.814E-11	1.222E-11	8.486E-12	4.136E-12	3.024E-12	2.346E-12	1.887E-12	1.581E-12	1.346E-12	1.171E-12
SW	1.423E-11	7.488E-12	4.742E-12	2.481E-12	1.573E-12	1.130E-12	8.803E-13	7.028E-13	5.878E-13	5.038E-13	4.417E-13
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	7.808E-10	3.874E-10	2.808E-10	1.287E-10	8.863E-11	6.540E-11	4.088E-11	3.128E-11	2.488E-11	2.028E-11	1.808E-11
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	5.540E-11	2.807E-11	1.810E-11	8.303E-12	5.874E-12	4.138E-12	3.134E-12	2.804E-12	2.078E-12	1.708E-12	1.543E-12
NE	1.102E-10	5.731E-11	3.803E-11	1.808E-11	1.172E-11	8.308E-12	8.258E-12	4.814E-12	3.881E-12	3.308E-12	2.781E-12
ENE	1.208E-10	8.310E-11	3.808E-11	2.080E-11	1.308E-11	8.308E-12	7.044E-12	5.806E-12	4.537E-12	3.786E-12	3.228E-12
E	1.143E-10	5.808E-11	3.806E-11	2.004E-11	1.288E-11	8.161E-12	8.807E-12	5.861E-12	4.538E-12	3.782E-12	3.232E-12
ESE	8.471E-11	4.448E-11	2.826E-11	1.483E-11	8.484E-12	8.812E-12	6.316E-12	4.248E-12	3.488E-12	2.826E-12	2.508E-12
SE	8.866E-11	3.834E-11	2.307E-11	1.206E-11	7.821E-12	5.518E-12	4.246E-12	3.410E-12	2.828E-12	2.382E-12	2.071E-12
SSE	6.328E-11	2.787E-11	1.777E-11	8.381E-12	6.837E-12	4.317E-12	3.381E-12	2.760E-12	2.327E-12	2.012E-12	1.708E-12

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

U.NRC COMPUTER CODE - XOQDOQ, VERSION 2.0   RUN DATE: 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK
8.000 DAY DECAY, DEPLETED

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT

SEGMENT BOUNDARIES IN MILES FROM THE SITE

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	2.316E-08	3.881E-08	3.151E-08	2.748E-08	2.342E-08	1.505E-08	7.488E-09	4.238E-09	2.867E-09	2.113E-09
SSW	2.082E-08	2.580E-08	2.408E-08	2.078E-08	1.730E-08	1.070E-08	5.087E-09	2.798E-09	1.855E-09	1.364E-09
SW	7.250E-09	1.582E-08	1.790E-08	1.818E-08	1.388E-08	8.888E-09	4.868E-09	2.588E-09	1.725E-09	1.273E-09
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	3.224E-22	8.833E-13	1.838E-10	2.083E-09	8.156E-09	4.888E-09	7.825E-09	6.102E-09	4.304E-09	3.813E-09
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	1.902E-08	3.851E-08	4.228E-08	3.884E-08	3.131E-08	2.004E-08	8.888E-09	5.887E-09	3.841E-09	2.867E-09
NE	3.794E-08	4.328E-08	3.824E-08	2.818E-08	2.228E-08	1.278E-08	6.572E-09	2.903E-09	1.848E-09	1.320E-09
ENE	4.324E-08	4.808E-08	4.148E-08	3.186E-08	2.508E-08	1.413E-08	5.881E-09	3.035E-09	1.821E-09	1.360E-09
E	6.108E-08	4.742E-08	3.577E-08	2.836E-08	2.813E-08	1.066E-08	4.448E-09	2.181E-09	1.303E-09	8.504E-10
ESE	4.217E-08	3.500E-08	2.867E-08	1.880E-08	1.503E-08	8.251E-09	3.406E-09	1.702E-09	1.006E-09	7.408E-10
SE	3.274E-08	3.101E-08	2.818E-08	2.018E-08	1.501E-08	9.083E-09	3.834E-09	2.023E-09	1.280E-09	8.106E-10
SSE	2.881E-08	3.380E-08	3.143E-08	2.582E-08	2.113E-08	1.273E-08	5.887E-09	3.141E-09	2.062E-09	1.404E-09

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

USNRC COMPUTER CODE - XOQDOQ, VERSION 2.0 RUN DATE: 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK
8.000 DAY DECAY, DEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) DISTANCES IN MILES FROM THE SITE

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	1.022E-08	2.256E-08	2.130E-08	2.470E-08	3.100E-08	3.310E-08	3.203E-08	2.886E-08	2.700E-08	2.540E-08	2.340E-08
SSW	1.107E-08	1.810E-08	1.900E-08	2.270E-08	2.071E-08	2.873E-08	2.500E-08	2.290E-08	2.082E-08	1.903E-08	1.735E-08
SW	1.463E-08	3.700E-08	8.100E-08	8.863E-08	1.801E-08	1.810E-08	1.824E-08	1.741E-08	1.830E-08	1.514E-08	1.404E-08
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	0.000E+00	0.000E+00	8.731E-30	7.253E-22	2.082E-16	1.560E-12	4.870E-11	3.804E-10	1.441E-08	3.710E-08	7.380E-08
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	1.741E-08	8.320E-08	1.810E-08	2.750E-08	4.110E-08	4.420E-08	4.200E-08	4.018E-08	3.700E-08	3.412E-08	3.141E-08
NE	1.181E-08	2.793E-08	3.740E-08	4.331E-08	4.847E-08	4.103E-08	3.875E-08	3.220E-08	2.831E-08	2.580E-08	2.233E-08
ENE	2.040E-08	3.324E-08	4.280E-08	4.814E-08	5.233E-08	4.700E-08	4.208E-08	3.880E-08	3.210E-08	2.827E-08	2.518E-08
E	2.734E-08	4.877E-08	5.231E-08	5.334E-08	4.872E-08	4.274E-08	3.818E-08	3.070E-08	2.844E-08	2.200E-08	2.014E-08
ESE	2.341E-08	4.233E-08	4.208E-08	4.158E-08	3.731E-08	3.177E-08	2.808E-08	2.280E-08	1.988E-08	1.711E-08	1.604E-08
SE	2.164E-08	3.238E-08	3.223E-08	3.330E-08	3.333E-08	3.015E-08	2.840E-08	2.311E-08	2.027E-08	1.788E-08	1.604E-08
SSE	1.187E-08	2.152E-08	2.586E-08	3.033E-08	3.531E-08	3.403E-08	3.104E-08	2.888E-08	2.587E-08	2.341E-08	2.110E-08

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) DISTANCES IN MILES FROM THE SITE

SECTOR	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	2.170E-08	1.520E-08	1.164E-08	7.510E-09	5.404E-09	4.234E-09	3.425E-09	2.856E-09	2.434E-09	2.112E-09	1.856E-09
SSW	1.588E-08	1.004E-08	8.012E-09	5.001E-09	3.844E-09	2.782E-09	2.230E-09	1.853E-09	1.570E-09	1.360E-09	1.188E-09
SW	1.302E-08	9.280E-09	7.012E-09	4.671E-09	3.310E-09	2.586E-09	2.071E-09	1.724E-09	1.480E-09	1.273E-09	1.118E-09
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	1.237E-08	4.502E-09	8.856E-09	8.250E-09	7.332E-09	8.314E-09	6.220E-09	4.324E-09	3.582E-09	3.024E-09	2.838E-09
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	2.888E-08	2.038E-08	1.534E-08	1.001E-08	7.288E-09	5.882E-09	4.881E-09	3.830E-09	3.281E-09	2.858E-09	2.518E-09
NE	2.006E-08	1.294E-08	9.127E-09	5.522E-09	3.831E-09	2.870E-09	2.280E-09	1.843E-09	1.542E-09	1.318E-09	1.143E-09
ENE	2.246E-08	1.418E-08	9.840E-09	5.824E-09	4.088E-09	3.018E-09	2.382E-09	1.918E-09	1.588E-09	1.367E-09	1.172E-09
E	1.786E-08	1.001E-08	7.521E-09	4.374E-09	2.882E-09	2.177E-09	1.888E-09	1.360E-09	1.123E-09	9.483E-10	8.138E-10
ESE	1.330E-08	9.232E-09	6.712E-09	3.356E-09	2.280E-09	1.883E-09	1.318E-09	1.086E-09	9.027E-10	7.480E-10	6.427E-10
SE	1.430E-08	9.123E-09	6.407E-09	3.888E-09	2.888E-09	2.014E-09	1.582E-09	1.287E-09	1.074E-09	9.160E-10	7.912E-10
SSE	1.820E-08	1.288E-08	9.308E-09	5.841E-09	4.128E-09	3.131E-09	2.481E-09	2.048E-09	1.728E-09	1.483E-09	1.282E-09

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	73.10	REP. WIND HEIGHT (METERS)	71.3	
DIAMETER	(METERS)	1.14	BUILDING HEIGHT (METERS)	31.4	
EXIT VELOCITY	(METERS)	18.21	BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	1000.0
			HEAT EMISSION RATE	(CAL/SEC)	0.0

ALL ELEVATED RELEASES

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

U.NRC COMPUTER CODE - XOQDOQ, VERSION 2.0 RUN DATE 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK
2.260 DAY DECAY, UNDEPLETED

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT

SEGMENT BOUNDARIES IN MILES FROM THE SITE

DIRECTION FROM SITE	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	2.344E-08	3.120E-08	3.101E-08	2.707E-08	2.376E-08	1.827E-08	7.582E-09	4.200E-09	2.856E-09	2.101E-09
SSW	2.112E-08	2.030E-08	2.510E-08	2.122E-08	1.700E-08	1.007E-08	5.220E-09	2.856E-09	1.800E-09	1.371E-09
SW	7.310E-09	1.573E-08	1.802E-08	1.830E-08	1.410E-08	8.101E-09	4.563E-09	2.540E-09	1.090E-09	1.230E-09
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	3.158E-22	6.700E-13	1.802E-10	1.086E-09	7.831E-09	4.507E-09	6.823E-09	4.773E-09	2.800E-09	1.710E-09
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	1.812E-08	3.870E-08	4.256E-08	3.724E-08	3.100E-08	2.030E-08	1.000E-08	5.000E-09	3.030E-09	2.837E-09
NE	3.841E-08	4.407E-08	3.711E-08	2.808E-08	2.304E-08	1.333E-08	5.862E-09	3.052E-09	1.800E-09	1.480E-09
ENE	4.373E-08	5.040E-08	4.247E-08	3.293E-08	2.807E-08	1.470E-08	6.302E-09	3.250E-09	2.071E-09	1.400E-09
E	5.274E-08	4.851E-08	3.801E-08	2.741E-08	2.108E-08	1.163E-08	4.830E-09	2.431E-09	1.532E-09	1.070E-09
ESE	4.286E-08	3.871E-08	2.744E-08	2.038E-08	1.572E-08	8.733E-09	3.872E-09	1.807E-09	1.184E-09	8.302E-10
SE	3.321E-08	3.287E-08	2.870E-08	2.081E-08	1.640E-08	8.403E-09	4.162E-09	2.162E-09	1.377E-09	8.808E-10
SSE	2.710E-08	3.441E-08	3.100E-08	2.832E-08	2.167E-08	1.304E-08	6.020E-09	3.224E-09	2.101E-09	1.518E-09

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

```
U.S.NRC COMPUTER CODE - XOQDOQ, VERSION 2.0        RUN DATE:  940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK
2.260 DAY DECAY, UNDEPLETED
```

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCES IN MILES FROM THE SITE

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	1.820E-08	2.207E-08	2.170E-08	2.513E-08	3.207E-08	3.360E-08	3.244E-08	3.036E-08	2.800E-08	2.565E-08	2.303E-08
SSW	1.107E-08	1.820E-08	2.020E-08	2.317E-08	2.717E-08	2.720E-08	2.566E-08	2.344E-08	2.136E-08	1.944E-08	1.774E-08
SW	1.460E-08	3.723E-08	8.101E-08	8.871E-08	1.812E-08	1.832E-08	1.837E-08	1.764E-08	1.843E-08	1.520E-08	1.410E-08
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	0.000E+00	0.000E+00	8.582E-30	7.100E-22	2.500E-16	1.526E-12	4.500E-11	3.813E-10	1.400E-08	3.030E-08	7.207E-08
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	1.741E-08	8.302E-08	1.827E-08	2.704E-08	4.142E-08	4.403E-08	4.334E-08	4.050E-08	3.740E-08	3.461E-08	3.170E-08
NE	1.100E-08	2.811E-08	3.704E-08	4.301E-08	4.825E-08	4.252E-08	3.703E-08	3.300E-08	2.814E-08	2.504E-08	2.300E-08
ENE	2.030E-08	3.344E-08	4.257E-08	4.876E-08	5.310E-08	4.800E-08	4.300E-08	3.770E-08	3.308E-08	2.822E-08	2.801E-08
E	2.732E-08	4.800E-08	5.313E-08	5.420E-08	5.082E-08	4.300E-08	3.734E-08	3.100E-08	2.750E-08	2.307E-08	2.110E-08
ESE	2.330E-08	4.200E-08	4.350E-08	4.244E-08	3.825E-08	3.200E-08	2.774E-08	2.300E-08	2.046E-08	1.700E-08	1.573E-08
SE	2.153E-08	3.250E-08	3.272E-08	3.300E-08	3.400E-08	3.086E-08	2.716E-08	2.377E-08	2.000E-08	1.861E-08	1.861E-08
SSE	1.100E-08	2.100E-08	2.802E-08	3.070E-08	3.583E-08	3.617E-08	3.247E-08	2.830E-08	2.847E-08	2.300E-08	2.182E-08

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCES IN MILES FROM THE SITE

SECTOR	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	2.202E-08	1.862E-08	1.171E-08	7.814E-09	5.518E-09	4.294E-09	3.437E-09	2.860E-09	2.427E-09	2.100E-09	1.842E-09
SSW	1.825E-08	1.111E-08	8.220E-09	5.220E-09	3.732E-09	2.864E-09	2.282E-09	1.883E-09	1.581E-09	1.300E-09	1.108E-09
SW	1.313E-08	8.332E-09	7.051E-09	4.570E-09	3.307E-09	2.540E-09	2.040E-09	1.804E-09	1.436E-09	1.230E-09	1.083E-09
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	1.202E-08	4.300E-08	8.405E-09	7.200E-09	8.100E-09	4.870E-09	3.740E-09	2.872E-09	2.200E-09	1.702E-09	1.322E-09
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	2.833E-08	2.081E-08	1.564E-08	1.012E-08	7.360E-09	5.804E-09	4.803E-09	3.830E-09	3.200E-09	2.836E-09	2.404E-09
NE	2.077E-08	1.330E-08	8.560E-09	5.816E-09	4.060E-09	3.040E-09	2.307E-09	1.860E-09	1.837E-09	1.300E-09	1.212E-09
ENE	2.333E-08	1.463E-08	1.040E-08	8.200E-09	4.347E-09	3.241E-09	2.542E-09	2.086E-09	1.722E-09	1.400E-09	1.200E-09
E	1.877E-08	1.182E-08	8.083E-09	4.707E-09	3.203E-09	2.410E-09	1.887E-09	1.527E-09	1.290E-09	1.077E-09	8.205E-10
ESE	1.401E-08	8.720E-09	6.100E-09	3.823E-09	2.400E-09	1.850E-09	1.465E-09	1.181E-09	9.840E-10	8.300E-10	7.220E-10
SE	1.404E-08	8.533E-09	6.700E-09	4.110E-09	2.801E-09	2.143E-09	1.887E-09	1.374E-09	1.140E-09	8.700E-10	8.470E-10
SSE	1.800E-08	1.317E-08	8.813E-09	6.000E-09	4.230E-09	3.216E-09	2.564E-09	2.087E-09	1.706E-09	1.514E-09	1.381E-09

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	73.10	REP. WIND HEIGHT (METERS)	71.3	
DIAMETER	(METERS)	1.14	BUILDING HEIGHT (METERS)	31.4	
EXIT VELOCITY	(METERS)	18.21	BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	2000.0
			HEAT EMISSION RATE	(CAL/SEC)	0.0

ALL ELEVATED RELEASES

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

SC IRI COMPUTER CODE - XOQDOQ, VERSION 2.0 RUN DATE: 940629
**** BIG ROCK POINT XOQDOQ82 *** USING 01/01/89 - 12/31/93 MET DATA ****
ELEVATED RELEASE - 240' STACK
NO DECAY, UNDEPLETED

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT

SEGMENT BOUNDARIES IN MILES FROM THE SITE

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	2.366E-08	3.136E-08	3.210E-08	2.808E-08	2.388E-08	1.660E-08	7.804E-09	4.482E-09	3.038E-09	2.288E-09
SSW	2.121E-08	2.842E-08	2.531E-08	2.140E-08	1.708E-08	1.115E-08	5.303E-09	3.010E-09	2.023E-09	1.487E-09
SW	7.330E-09	1.870E-08	1.012E-08	1.041E-08	1.422E-08	8.283E-09	4.880E-09	2.888E-09	1.805E-09	1.341E-09
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	3.250E-22	7.106E-13	1.743E-10	2.191E-09	8.074E-09	6.825E-08	1.062E-07	8.788E-08	7.758E-08	8.141E-08
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	1.815E-08	3.807E-08	4.271E-08	3.740E-08	3.188E-08	2.048E-08	1.027E-08	5.807E-09	3.867E-09	2.807E-09
NE	3.048E-08	4.418E-08	3.725E-08	2.818E-08	2.328E-08	1.347E-08	5.802E-09	3.187E-09	2.088E-09	1.487E-09
ENE	4.304E-08	5.094E-08	4.288E-08	3.312E-08	2.818E-08	1.487E-08	8.518E-09	3.301E-09	2.101E-09	1.570E-09
E	5.280E-08	4.870E-08	3.711E-08	2.702E-08	2.128E-08	1.182E-09	4.803E-09	2.588E-09	1.554E-09	1.108E-09
ESE	4.208E-08	3.808E-08	2.788E-08	2.055E-08	1.588E-08	8.878E-09	3.788E-09	1.875E-09	1.201E-09	8.201E-10
SE	3.333E-08	3.272E-08	2.805E-08	2.088E-08	1.805E-08	8.848E-09	4.207E-09	2.271E-09	1.485E-09	1.078E-09
SSE	2.724E-08	3.453E-08	3.211E-08	2.848E-08	2.174E-08	1.320E-08	6.104E-09	3.362E-09	2.218E-09	1.824E-09

AVERAGE EFFECTIVE STACK HEIGHT IN METERS FOR EACH SEGMENT

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	8.573E+01	8.574E+01	8.574E+01	8.574E+01	8.574E+01	8.574E+01	8.574E+01	8.574E+01	8.574E+01	8.574E+01
SSW	8.831E+01	8.833E+01	8.833E+01	8.833E+01	8.833E+01	8.833E+01	8.833E+01	8.833E+01	8.833E+01	8.833E+01
SW	8.804E+01	8.804E+01	8.804E+01	8.804E+01	8.804E+01	8.804E+01	8.804E+01	8.804E+01	8.804E+01	8.804E+01
WSW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
WNW	3.017E+02	3.053E+02	3.053E+02	3.053E+02	3.053E+02	3.053E+02	3.053E+02	3.053E+02	3.053E+02	3.053E+02
NW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNW	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
N	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NNE	8.200E+01	8.200E+01	8.200E+01	8.200E+01	8.200E+01	8.200E+01	8.200E+01	8.200E+01	8.200E+01	8.200E+01
NE	8.487E+01	8.487E+01	8.487E+01	8.487E+01	8.467E+01	8.487E+01	8.487E+01	8.487E+01	8.487E+01	8.487E+01
ENE	8.582E+01	8.583E+01	8.583E+01	8.583E+01	8.583E+01	8.583E+01	8.583E+01	8.583E+01	8.583E+01	8.583E+01
E	8.083E+01	8.085E+01	8.085E+01	8.085E+01	8.085E+01	8.085E+01	8.085E+01	8.085E+01	8.085E+01	8.085E+01
ESE	8.840E+01	8.840E+01	8.840E+01	8.840E+01	8.840E+01	8.840E+01	8.840E+01	8.840E+01	8.840E+01	8.840E+01
SE	8.750E+-1	8.750E+01	8.750E+01	8.750E+01	8.750E+01	8.750E+01	8.750E+01	8.750E+01	8.750E+01	8.750E+01
SSE	8.500E+01	8.500E+01	8.500E+01	8.500E+01	8.500E+01	8.500E+01	8.500E+01	8.500E+01	8.500E+01	8.500E+01

Data in sectors WSW through N are not valid. Refer to note at the end of Table 1.3.

TABLE 1.3

USNRC COMPUTER CODE - XOQDOQ, VERSION 2.0 RUN DATE: [illegible]
**** BIG ROCK POINT XOQDOQ82 **** USING 01/01/80 - 12/31/83 MET DATA ****
ELEVATED RELEASE - 240' STACK
SPECIFIC POINTS OF INTEREST

RELEASE ID	TYPE OF LOCATION	DIRECTION	DISTANCE FROM SITE (MILES)	DISTANCE FROM SITE (METERS)	X/Q (SEC/M³) NO DECAY UNDEPLETED	X/Q (SEC/M³) 2.26 D DECAY UNDEPLETED	X/Q (SEC/M³) 8.0 D DECAY DEPLETED	D/Q (PER SQ.METER)
A	SITE BOUNDARY	E	0.57	917.	4.91E-08	4.[illegible]E-08	4.[illegible]E-08	1.25E-08
A	SITE BOUNDARY	ESE	0.52	837.	4.[illegible]E-08	4.10E-08	4.07E-08	1.10E-08
A	SITE BOUNDARY	SE	0.56	[illegible]	3.0[illegible]E-08	3.0[illegible]E-08	3.[illegible]E-08	8.10E-10
A	SITE BOUNDARY	SSE	0.58	933.	2.25E-08	2.24E-08	2.22E-08	6.12E-10
A	SITE BOUNDARY	S	0.[illegible]	1004.	2.07E-08	2.[illegible]E-08	2.[illegible]E-08	3.[illegible]E-10
A	SITE BOUNDARY	SSW	0.71	1143.	1.[illegible]E-08	1.[illegible]E-08	1.91E-08	3.[illegible]E-10
A	SITE BOUNDARY	SW	0.50	805.	3.58E-08	3.57E-08	3.[illegible]E-08	1.10E-10
A	MAXIMUM CHI/Q	S	2.00	3219.	3.37E-08	3.[illegible]E-08	3.31E-08	1.34E-10
A	MAXIMUM CHI/Q	SSW	2.00	3219.	2.73E-08	2.71E-08	2.07E-08	1.31E-10
A	MAXIMUM CHI/Q	SW	2.50	4023.	1.[illegible]E-08	1.[illegible]E-08	1.62E-08	3.[illegible]E-11
A	MAXIMUM CHI/Q	WSW	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	W	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	WNW	15.00	24140.	1.12E-07	7.2[illegible]E-08	8.2[illegible]E-08	1.30E-10
A	MAXIMUM CHI/Q	NW	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	NNW	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	N	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	NNE	2.00	3219.	4.47E-08	4.40E-08	4.43E-08	1.[illegible]E-10
A	MAXIMUM CHI/Q	NE	1.50	2414.	4.[illegible]E-08	4.82E-08	4.54E-08	6.40E-10
A	MAXIMUM CHI/Q	ENE	1.50	2414.	6.32E-08	6.30E-08	5.22E-08	6.70E-10
A	MAXIMUM CHI/Q	E	1.00	1609.	5.40E-08	6.30E-08	6.30E-08	[illegible]E-10
A	MAXIMUM CHI/Q	ESE	0.75	1207.	4.2[illegible]E-08	4.2[illegible]E-08	4.21E-08	8.15E-10
A	MAXIMUM CHI/Q	SE	1.50	2414.	3.40E-08	3.3[illegible]E-08	3.32E-08	3.00E-10
A	MAXIMUM CHI/Q	SSE	1.50	2414.	3.58E-08	3.56E-08	3.52E-08	2.57E-10

VENT AND BUILDING PARAMETERS:
RELEASE HEIGHT (METERS) 73.10 REP. WIND HEIGHT (METERS) 71.3

Note: Big Rock Point meteorological data was gathered from sensors mounted on the 73 meter stack. Sensor were mounted into the prevailing wind direction. Because of interference to the wind flow by the stack when winds were from the 71° to 159° (flowing towards Lake Michigan), the meteorological data recorded in these sectors are considered invalid. For dose calculational purposes, this effectively invalidates six (6) lakeward sectors (WSW, W, WNW, NW, NNW, and N). Therefore zeros are recorded in Table 1.3 for these sectors. However, the program which calculates the annual average Chi/Q requires input of the full years met data. Any data recorded for these six sectors are input in the WNW sector to satisfy the program. Values of Chi/Q listed in the WNW sector are invalid.

TABLE 1.4

CONSERVATIVE BIG ROCK POINT GASPAR INPUT PARAMETERS

Critical Receptors

Location	Sector	Distance (miles)	X/Q (sec/m³)	X/Q Decay (sec/m³)	X/Q Decay and Dep (sec/m³)	D/Q (1/m²)
Residence/Garden	E	1.40	5.20E-08	5.18E-08	5.07E-08	6.23E-10
Site Boundary	E	0.57	4.91E-08	4.90E-08	4.85E-08	1.25E-09
Beef Cattle	SSE	1.70	3.57E-08	3.56E-08	3.50E-08	2.30E-10
Dairy Cow	E	2.80	3.43E-08	3.41E-08	3.29E-08	2.75E-10

NOTE: The above data are used as conservative values for the sector with highest X/Q (E sector) which contains nearby residences and has historically contained farming activities. No closer farming activities are foreseen, since the trend is toward conversion of farms to residential and recreational uses.

DOSE FACTORS FOR SUBMERSION IN NOBLE GASES*
Table 1.5

	DFB[1]	DFY[2]	DFS[1]	DFB[2]
Kr-85m	1.17(+3)[3]	1.23(+3)	1.46(+3)	1.97(+3)
Kr-85	1.61(+1)	1.72(+1)	1.34(+3)	1.95(+3)
Kr-87	5.92(+3)	6.17(+3)	9.73(+3)	1.03(+4)
Kr-88	1.47(+4)	1.52(+4)	2.37(+3)	2.93(+3)
Kr-89	1.66(+4)	1.73(+4)	1.01(+4)	1.06(+4)
Xe-131m	9.15(+1)	1.56(+2)	4.76(+2)	1.11(+3)
Xe-133m	2.51(+2)	3.27(+2)	9.94(+2)	1.48(+3)
Xe-133	2.94(+2)	3.53(+2)	3.06(+2)	1.05(+3)
Xe-135m	3.12(+3)	3.36(+3)	7.11(+2)	7.39(+3)
Xe-135	1.81(+3)	1.92(+3)	1.86(+3)	2.46(+3)
Xe-137	1.42(+3)	1.51(+3)	1.22(+4)	1.27(+4)
Xe-138	8.83(+3)	9.21(+3)	4.13(+3)	4.75(+3)
Ar-41	8.84(+3)	9.30(+3)	2.69(+3)	3.26(+3)

1 mrem/y per μCi/m^3
2 mrad/y per μCi/m^3
3 1.17(+3) = 1.17 x 10^3

*Dose factors for exposure to a semi-infinite cloud of noble gases. Values for doses in this table, Table 1.7 and Table 1.8 are dose equivalent (DE) values, and were obtained from US NRC Regulatory Guide 1.109, Revision 1 (October 1977). These factors convert to somewhat different values from those used in emergency planning, which are obtained from EPA-400 (May 1992), primarily because the EPA internal doses represent total effective dose equivalent (TEDE) rather than DE, and the EPA ground shine conversions are based on concentration with meteorologically independent deposition velocity assumptions and integrated exposure time of only 96 hours.

STABLE ELEMENT TRANSFER DATA
Table 1.8

Element	F_m - Milk (d/L) (Cow)	F_m - Milk (d/L) (Goat)	F_f - Meat (d/kg)	B_{iv} Veg/Soil
H	1.0E-02	1.7E-01	1.2E-02	4.8E-00
C	1.2E-02	1.0E-01	3.1E-02	5.5E-00
Na	4.0E-02	4.0E-02	3.0E-02	5.2E-02
P	2.5E-02	2.5E-01	4.8E-02	1.1E-00
Cr	2.2E-03	2.2E-03	2.4E-03	2.5E-04
Mn	2.5E-04	2.5E-04	8.0E-04	2.9E-02
F	1.2E-03	1.3E-0[illegible]	4.0E-02	6.6E-04
Co	1.0E-03	[illegible]0E-03	1.3E-02	9.4E-03
Ni	6.7E-03	[illegible].7E-0[illegible]	5.3E-02	1.9E-02
Cu	1.4E-02	1.3E-02	8.0E-03	1.2E-01
Zn	3.9E-02	3.9E-02	[illegible].0E-02	4.0E-01
Rb	3.0E-02	3.0E-02	3.1E-02	1.3E-01
Sr	8.0E-04	1.4E-[illegible]2	6.0E-04	1.7E-02
Y	1.0E-05	1.0E-05	4.[illegible]E-03	2.6E-03
Zr	5.0E-06	5.0E-06	3.4E-02	1.7E-0[illegible]
Nb	2.5E-03	2.5E-03	2.8E-01	9.4E-03
Mo	7.5E-03	7.5E-03	8.0E-03	1.2E-01
Tc	2.5E-02	2.5E-02	4.0E-01	2.5E-01
Ru	1.0E-06	1.0E-06	4.0E-01	5.0E-02
Rh	1.0E-02	1.0E-02	1.5E-03	1.[illegible]E+01
Ag	5.0E-02	5.0E-02	1.7E-02	1.5E-01
Te	1.0E-03	1.0E-03	7.7E-02	1.3E-00
I	6.0E-03	6.0E-02	2.9E-03	2.0E-02
Cs	1.2E-02	3.0E-01	4.0E-03	1.0E-02
[illegible]	4.0E-04	4.0E-04	3.2E-03	5.0E-03
La	5.0E-06	5.0E-06	2.0E-04	2.5E-03
Ce	1.0E-04	1.0E-04	1.2E-03	2.5E-03
Pr	5.0E-06	5.0E-06	4.7E-03	2.5E-03
Nd	5.0E-06	5.0E-06	3.3E-03	2.4E-03
W	5.0E-04	5.0E-04	1.3E-03	1.8E-02
Np	5.0E-06	5.0E-06	2.0E-04	2.5E-03

TABLE 1.7

INHALATION DOSE FACTORS FOR INFANT
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	4.62E-07	4.62E-07	4.62E-07	4.62E-07	4.62E-07	4.62E-07
C 14	1.89E-05	3.79E-06	3.79E-06	3.79E-06	3.79E-06	3.79E-06	3.79E-06
NA 24	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06
P 32	1.45E-03	8.03E-05	5.53E-05	NO DATA	NO DATA	NO DATA	1.15E-05
CR 51	NO DATA	NO DATA	6.39E-08	4.11E-08	9.45E-09	9.17E-06	2.55E-07
MN 54	NO DATA	1.81E-05	3.56E-06	NO DATA	3.56E-06	7.14E-04	5.04E-06
MN 56	NO DATA	1.10E-09	1.58E-10	NO DATA	7.86E-10	8.95E-06	5.12E-05
FE 55	1.41E-05	8.39E-06	2.38E-06	NO DATA	NO DATA	6.21E-05	7.82E-07
FE 59	9.69E-06	1.68E-05	6.77E-06	NO DATA	NO DATA	7.25E-04	1.77E-05
CO 58	NO DATA	8.71E-07	1.30E-06	NO DATA	NO DATA	5.55E-04	7.95E-06
CO 60	NO DATA	5.73E-06	8.41E-06	NO DATA	NO DATA	3.22E-03	2.28E-05
NI 63	2.42E-04	1.46E-05	8.29E-06	NO DATA	NO DATA	1.49E-04	1.73E-06
NI 65	1.71E-09	2.03E-10	8.79E-11	NO DATA	NO DATA	5.80E-06	3.58E-05
CU 64	NO DATA	1.34E-09	[illegible].53E-10	NO DATA	2.84E-09	6.64E-06	1.07E-05
ZN 65	1.38E-05	4.47E-05	2.22E-05	NO DATA	2.32E-05	4.62E-04	3.67E-05
ZN 69	3.85E-11	6.91E-11	5.13E-12	NO DATA	2.87E-11	1.05E-06	9.44E-06
BR 83	NO DATA	NO DATA	2.72E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	2.86E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	1.46E-08	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	1.36E-04	6.30E-05	NO DATA	NO DATA	NO DATA	2.17E-06
RB 88	NO DATA	3.98E-07	2.50E-07	NO DATA	NO DATA	NO DATA	2.42E-07
RB 89	NO DATA	2.29E-07	1.47E-07	NO DATA	NO DATA	NO DATA	4.87E-08
SR 89	2.84E-04	NO DATA	8.15E-06	NO DATA	NO DATA	1.45E-03	4.57E-05
SR 90	2.92E-02	NO DATA	1.85E-03	NO DATA	NO DATA	8.03E-03	9.36E-05
SR 91	6.83E-08	NO DATA	2.47E-09	NO DATA	NO DATA	3.76E-05	5.24E-05
SR 92	7.50E-09	NO DATA	2.79E-10	NO DATA	NO DATA	1.70E-05	1.00E-04
Y 90	2.35E-06	NO DATA	6.30E-08	NO DATA	NO DATA	1.92E-04	7.43E-05
Y 91m	2.91E-10	NO DATA	9.90E-12	NO DATA	NO DATA	1.99E-06	1.68E-06
Y 91	4.20E-04	NO DATA	1.12E-05	NO DATA	NO DATA	1.75E-03	5.02E-05
Y 92	1.17E-08	NO DATA	3.29E-10	NO DATA	NO DATA	1.75E-05	9.04E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR INFANT (MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.07E-07	NO DATA	2.91E-09	NO DATA	NO DATA	5.46E-05	1.19E-04
ZR 95	8.24E-05	1.99E-05	1.45E-05	NO DATA	2.22E-05	1.25E-03	1.55E-05
ZR 97	1.07E-07	1.83E-08	8.36E-09	NO DATA	1.85E-08	7.88E-05	1.00E-04
NB 95	1.12E-05	4.59E-06	2.70E-06	NO DATA	3.37E-06	3.42E-04	9.05E-06
MO 99	NO DATA	1.18E-07	2.31E-08	NO DATA	1.89E-07	9.63E-05	3.48E-05
TC 99m	9.98E-13	2.06E-12	2.66E-11	NO DATA	2.22E-11	5.79E-07	1.45E-06
TC101	4.65E-14	5.88E-14	5.80E-13	NO DATA	6.99E-13	4.17E-07	6.03E-07
RU103	1.44E-06	NO DATA	4.85E-07	NO DATA	3.03E-06	3.94E-04	1.15E-05
RU105	8.74E-10	NO DATA	2.93E-10	NO DATA	6.42E-10	1.12E-05	3.46E-05
RU106	6.20E-05	NO DATA	7.77E-06	NO DATA	7.61E-05	8.26E-03	1.17E-04
AG110m	7.13E-06	5.16E-06	3.57E-06	NO DATA	7.80E-06	2.62E-03	2.36[illegible]
TE125m	3.40E-06	1.42E-06	4.70E-07	1.16E-06	NO DATA	3.19E-04	9.22E[illegible]
TE127m	1.19E-05	4.93E-06	1.48E-06	3.48E-06	2.68E-05	9.37E-04	1.95E-05
TE127	1.59E-09	6.81E-10	3.49E-10	1.32E-09	3.47E-09	7.39E-06	1.74E-05
TE129m	1.01E-05	4.35E-06	1.59E-06	3.91E-06	2.27E-05	1.20E-03	4.93E-05
TE129	5.63E-11	2.48E-11	1.34E-11	4.82E-11	1.25E-10	2.14E-06	1.88E-05
TE131m	7.62E-08	3.93E-08	2.59E-08	6.38E-08	1.89E-07	1.42E-04	8.51E-05
TE131	1.24E-11	5.87E-12	3.57E-12	1.13E-11	2.85E-11	1.47E-06	5.87E-06
TE132	2.66E-07	1.69E-07	1.26E-07	1.99E-07	7.39E-07	2.43E-04	3.15E-05
I 130	4.54E-06	9.91E-06	3.98E-06	1.14E-03	1.09E-05	NO DATA	1.42E-06
I 131	2.71E-05	3.17E-05	1.40E-05	1.06E-02	3.70E-05	NO DATA	7.56E-07
I 132	1.21E-06	2.53E-06	8.99E-07	1.21E-04	2.82E-06	NO DATA	1.36E-06
I 133	9.46E-06	1.37E-05	4.00E-06	2.54E-03	1.60E-05	NO DATA	1.54E-06
I 134	6.58E-07	1.34E-06	4.75E-07	3.18E-05	1.49E-06	NO DATA	9.21E-07
I 135	2.76E-06	5.43E-06	1.98E-06	4.97E-04	6.05E-06	NO DATA	1.31E-06
CS134	2.83E-04	5.02E-04	5.32E-05	NO DATA	1.36E-04	5.69E-05	9.53E-07
CS136	3.45E-05	9.61E-05	3.78E-05	NO DATA	4.03E-05	8.40E-06	1.02E-06
CS137	3.92E-04	4.37E-04	3.25E-05	NO DATA	1.23E-04	5.09E-05	9.53E-07
CS138	3.61E-07	5.58E-07	2.84E-07	NO DATA	2.93E-07	4.67E-08	6.26E-07
BA139	1.06E-09	7.03E-13	3.07E-11	NO DATA	4.23E-13	4.25E-06	3.64E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR INFANT
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	4.00E-05	4.00E-08	2.07E-06	NO DATA	9.59E-09	1.14E-03	2.74E-05
BA141	1.12E-10	7.70E-14	3.55E-12	NO DATA	4.64E-14	2.12E-06	3.39E-06
BA142	2.84E-11	2.36E-14	1.40E-12	NO DATA	1.36E-14	1.11E-06	4.95E-07
LA140	3.61E-07	1.43E-07	3.68E-08	NO DATA	NO DATA	1.20E-04	6.06E-05
LA142	7.36E-10	2.69E-10	6.46E-11	NO DATA	NO DATA	5.87E-06	4.25E-05
CE141	1.98E-05	1.19E-05	1.42E-06	NO DATA	3.75E-06	3.69E-04	1.54E-05
CE143	2.09E-07	1.38E-07	1.58E-08	NO DATA	4.03E-08	8.30E-05	3.55E-05
CE144	2.28E-03	8.65E-04	1.26E-04	NO DATA	3.84E-04	7.03E-03	1.06E-04
PR143	1.00E-05	3.74E-06	4.99E-07	NO DATA	1.41E-06	3.09E-04	2.66E-05
PR144	3.42E-11	1.32E-11	1.72E-12	NO DATA	4.80E-12	1.15E-06	3.06E-06
ND147	5.67E-06	5.81E-06	3.57E-07	NO DATA	2.25E-06	2.30E-04	2.23E-05
W 187	9.26E-09	6.44E-09	2.23E-09	NO DATA	NO DATA	2.83E-05	2.54E-05
NP239	2.65E-07	2.37E-08	1.34E-08	NO DATA	4.73E-08	4.25E-05	1.78E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR CHILD
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	3.04E-07	3.04E-07	3.04E-07	3.04E-07	3.04E-07	3.04E-07
C 14	9.70E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06
NA 24	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06
P 32	7.04E-04	3.09E-05	2.67E-05	NO DATA	NO DATA	NO DATA	1.14E-05
CR 51	NO DATA	NO DATA	4.17E-08	2.31E-008	6.57E-09	4.59E-06	2.93E-07
MN 54	NO DATA	1 16E-05	2.57E-06	NO DATA	2.71E-06	4.28E-04	6.19E-06
MN 56	NO DATA	4.48E-10	8.43E-11	NO DATA	4.52E-10	3.55E-06	3.33E-05
FE 55	1.28E-05	6.80E-06	2.10E-06	NO DATA	NO DATA	3.00E-05	7.75E-07
FE 59	5.59E-06	9.04E-06	4.51E-06	NO DATA	NO DATA	3.43E-04	1.91E-05
CO 58	NO DATA	4.79E-07	8.55E-07	NO DATA	NO DATA	2.99E-04	9.29E-06
CO 60	NO DATA	3.55E-06	6.12E-06	NO DATA	Nu DATA	1.91E-03	2.60E-05
NI 63	2.22E-04	1.25E-05	7.56E-06	NO DATA	NO DATA	7.43E-05	1.71E-06
NI 65	8.08E-10	7.99E-11	4.44E-11	NO DATA	NO DATA	2.21E-06	2.27E-05
CU 64	NO DATA	5.39E-10	2.90E-10	NO DATA	1.63E-09	2.59E-06	9.92E-06
ZN 65	1.15E-05	3.06E-05	1.90E-05	NO DATA	1.93E-05	2.69E-04	4.41E-06
ZN 69	1.81E-11	2.61E-11	2.41E-12	NO DATA	1.58E-11	3.84E-07	2.75E-06
BR 83	NO DATA	NO DATA	1.28E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	1.48E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	6.84E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	5.36E-05	3.09E-05	NO DATA	NO DATA	NO DATA	2.16E-06
RB 88	NO DATA	1.52E-07	9.90E-08	NO DATA	NO DATA	NO DATA	4.66E-09
RB 89	NO DATA	9.33E-08	7.85E-08	NO DATA	NO DATA	NO DATA	5.11E-10
SR 89	1.62E-04	NO DATA	4.66E-06	NO DATA	NO DATA	5.83E-04	4.52E-05
SR 90	2.73E-02	NO DATA	1.74E-03	NO DATA	NO DATA	3.99E-03	9.28E-05
SR 91	3.28E-08	NO DATA	1.24E-09	NO DATA	NO DATA	1.44E-05	4.70E-05
SR 92	3.54E-09	NO DATA	1.42E-10	NO DATA	NO DATA	3.49E-06	6.55E-05
Y 90	1.11E-06	NO DATA	2.99E-08	NO DATA	NO DATA	7.07E-05	7.24E-05
Y 91m	1.37E-10	NO DATA	4.98E-12	NO DATA	NO DATA	7.60E-07	4.64E-07
Y 91	2.47E-04	NO DATA	6.59E-06	NO DATA	NO DATA	7.10E-04	4.97E-05
Y 92	5.50E-09	NO DATA	1 57E-10	NO DATA	NO DATA	6.46E-06	6.46E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR CHILD
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	5.04E-08	NO DATA	1.36E-09	NO DATA	NO DATA	2.01E-05	1.05E-04
ZR 95	5.13E-05	1.13E-05	1.00E-05	NO DATA	1.61E-05	6.03E-04	1.65E-05
ZR 97	5.07E-08	7.34E-09	4.32E-09	NO DATA	1.05E-08	3.06E-05	9.49E-05
NB 95	6.35E-06	2.48E-06	1.77E-06	NO DATA	2.33E-06	1.66E-04	1.00E-05
MO 99	NO DATA	4.66E-08	1.15E-08	NO DATA	1.06E-07	3.66E-05	3.42E-05
TC 99m	4.81E-13	9.41E-13	1.56E-11	NO DATA	1.37E-11	2.57E-07	1.30E-06
TC101	2.19E-14	2.30E-14	2.91E-13	NO DATA	3.92E-13	1.58E-07	4.41E-09
RU103	7.55E-07	NO DATA	2.90E-07	NO DATA	1.90E-06	1.79E-04	1.21E-05
RU105	4.13E-10	NO DATA	1.50E-10	NO DATA	3.63E-10	4.30E-06	2.69E-05
RU106	3.68E-05	NO DATA	4.57E-06	NO DATA	4.97E-05	3.87E-03	1.16E-04
AG110m	4.56E-06	3.08E-06	2.47E-06	NO DATA	5.74E-06	1.48E-03	2.71E-05
TE125m	1.82E-06	6.29E-07	2.47E-07	5.20E-07	NO DATA	1.29E-04	9.13E-06
TE127m	6.72E-06	2.31E-06	8.16E-07	1.64E-06	1.72E-05	4.00E-04	1.93E-05
TE127	7.49E-10	2.57E-10	1.67E-10	5.30E-10	1.91E-09	2.71E-06	1.52E-05
TE129m	5.19E-06	1.85E-06	8.22E-07	1.71E-06	1.36E-05	4.76E-04	4.91E-05
TE129	2.34E-11	9.45E-12	6.44E-12	1.93E-11	6.94E-11	7.93E-07	6.89E-06
TE131m	3.63E-08	1.60E-08	1.37E-08	2.64E-08	1.08E-07	5.56E-05	8.32E-05
TE131	5.87E-12	2.28E-12	1.78E-12	4.59E-12	1.59E-11	5.55E-07	3.60E-07
TE132	1.30E-07	7.36E-08	7.12E-08	8.58E-08	4.79E-07	1.02E-04	3.72E-05
I 130	2.21E-06	4.43E-06	2.28E-06	4.99E-04	6.61E-06	NO DATA	1.38E-06
I 131	1.30E-05	1.30E-05	7.37E-06	4.39E-03	2.13E-05	NO DATA	7.68E-07
I 132	5.72E-07	1.10E-06	5.07E-07	5.23E-05	1.69E-06	NO DATA	8.65E-07
I 133	4.48E-06	5.49E-06	2.08E-06	1.04E-03	9.13E-06	NO DATA	1.48E-06
I 134	3.17E-07	5.84E-07	2.69E-07	1.37E-05	8.92E-07	NO DATA	2.58E-07
I 135	1.33E-06	2.36E-06	1.12E-06	2.14E-04	3.62E-06	NO DATA	1.20E-06
CS134	1.76E-04	2.74E-04	6.07E-05	NO DATA	8.93E-05	3.27E-05	1.04E-06
CS136	1.76E-05	4.62E-05	3.14E-05	NO DATA	2.58E-05	3.93E-06	1.13E-06
CS137	2.45E-04	2.23E-04	3.47E-05	NO DATA	7.63E-05	2.81E-05	9.78E-07
CS138	1.71E-07	2.27E-07	1.50E-07	NO DATA	1.68E-07	1.84E-08	7.29E-08
BA139	4.98E-10	2.66E-13	1.45E-11	NO DATA	2.33E-13	1.56E-06	1.56E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR CHILD
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	2.00E-05	1.75E-08	1.17E-06	NO DATA	5.71E-09	4.71E-04	2.75E-05
BA141	5.29E-11	2.95E-14	1.72E-12	NO DATA	2.56E-14	7.89E-07	7.44E-08
BA142	1.35E-11	9.73E-15	7.54E-13	NO DATA	7.87E-15	4.44E-07	7.41E-10
LA140	1.74E-07	6.08E-08	2.04E-08	NO DATA	NO DATA	4.94E-06	6.10E-05
LA142	3.50E-10	1.11E-10	3.49E-11	NO DATA	NO DATA	2.35E-06	2.05E-05
CE141	1.06E-05	5.28E-06	7.83E-07	NO DATA	2.31E-06	1.47E-04	1.53E-05
CE143	9.89E-08	5.37E-08	7.77E-09	NO DATA	2.26E-08	3.12E-05	3.44E-05
CE144	1.83E-03	5.72E-04	9.77E-05	NO DATA	3.17E-04	3.23E-03	1.05E-04
PR143	4.99E-06	1.50E-06	2.47E-07	NO DATA	8.11E-07	1.17E-04	2.63E-05
PR144	1.61E-11	4.99E-12	8.10E-13	NO DATA	2.64E-12	4.23E-07	5.32E-08
ND147	2.92E-06	2.36E-06	1.84E-07	NO DATA	1.30E-06	8.87E-05	2.22E-05
W 187	4.41E-09	2.61E-09	1.17E-09	NO DATA	NO DATA	1.11E-05	2.46E-05
NP239	1.26E-07	9.04E-09	6.35E-09	NO DATA	2.63E-08	1.57E-05	1.73E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR TEENAGER
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.59E-07	1.59E-07	1.59E-07	1.59E-07	1.59E-07	1.59E-07
C 14	3.25E-06	6.09E-07	6.09E-07	6.09E-07	6.09E-07	6.09E-07	6.09E-07
NA 24	1.72E-06	1.72E-06	1.72E-06	1.72E-06	1.72E-06	1.72E-06	1.72E-06
P 32	2.36E-04	1.37E-05	8.95E-06	NO DATA	NO DATA	NO DATA	1.16E-05
CR 51	NO DATA	NO DATA	1.69E-08	9.37E-09	3.84E-09	2.62E-06	3.75E-07
MN 54	NO DATA	6.39E-06	1.05E-06	NO DATA	1.59E-06	2.48E-04	8.35E-06
MN 56	NO DATA	2.12E-10	3.15E-11	NO DATA	2.24E-10	1.90E-06	7.18E-06
FE 55	4.18E-06	2.98E-06	6.93E-07	NO DATA	NO DATA	1.55E-05	7.99E-07
FE 59	1.99E-06	4.62E-06	1.79E-06	NO DATA	NO DATA	1.91E-04	2.23E-05
CO 58	NO DATA	2.59E-07	3.47E-07	NO DATA	NO DATA	1.68E-04	1.19E-05
CO 60	NO DATA	1.89E-06	2.48E-06	NO DATA	NO DATA	1.09E-03	3.24E-05
NI 63	7.25E-05	5.43E-06	2.47E-06	NO DATA	NO DATA	3.84E-05	1.77E-06
NI 65	2.73E-10	3.66E-11	1.59E-11	NO DATA	NO DATA	1.17E-06	4.59E-06
CU 64	NO DATA	2.54E-10	1.06E-10	NO DATA	8.01-10	1.39E-06	7.68E-06
ZN 65	4.82E-06	1.67E-05	7.80E-06	NO DATA	1.08E-05	1.55E-04	5.83E-06
ZN 69	6.04E-12	1.15E-11	8.07E-13	NO DATA	7.53E-12	1.98E-07	3.56E-08
BR 83	NO DATA	NO DATA	4.30E-08	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	5.41E-08	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	2.29E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	2.38E-05	1.05E-05	NO DATA	NO DATA	NO DATA	2.21E-06
RB 88	NO DATA	6.82E-08	3.40E-08	NO DATA	NO DATA	NO DATA	3.65E-15
RB 89	NO DATA	4.40E-08	2.91E-08	NO DATA	NO DATA	NO DATA	4.22E-17
SR 89	5.43E-05	NO DATA	1.56E-06	NO DATA	NO DATA	3.02E-04	4.64E-05
SR 90	1.35E-02	NO DATA	8.35E-04	NO DATA	NO DATA	2.06E-03	9.56E-05
SR 91	1.10E-08	NO DATA	4.39E-10	NO DATA	NO DATA	7.59E-06	3.24E-05
SR 92	1.19E-09	NO DATA	5.08E-11	NO DATA	NO DATA	3.43E-06	1.49E-05
Y 90	3.73E-07	NO DATA	1.00E-08	NO DATA	NO DATA	3.66E-05	6.99E-05
Y 91m	4.63E-11	NO DATA	1.77E-12	NO DATA	NO DATA	4.00E-07	3.77E-09
Y 91	8.26E-05	NO DATA	2.21E-06	NO DATA	NO DATA	3.67E-04	5.11E-05
Y 92	1.84E-09	NO DATA	5.36E-11	NO DATA	NO DATA	3.35E-06	2.06E-05

TABLE 1.7

INHALATION DOSE FACTORS FOR TEENAGER
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.69E-08	NO DATA	4.65E-10	NO DATA	NO DATA	1.04E-05	7.24E-05
ZR 95	1.82E-06	5.73E-07	3.94E-07	NO DATA	8.42E-07	3.36E-05	1.86E-05
ZR 97	1.72E-08	3.40E-09	1.57E-09	NO DATA	5.15E-09	1.62E-05	7.88E-05
NB 95	2.32E-07	1.29E-07	7.08E-08	NO DATA	1.25E-07	9.39E-06	1.21E-05
MO 99	NO DATA	2.11E-08	4.03E-09	NO DATA	5.14E-08	1.92E-05	3.36E-05
TC 99m	1.73E-13	4.83E-13	6.24E-12	NO DATA	7.20E-12	1.44E-06	7.66E-06
TC101	7.40E-15	1.05E-14	1.03E-13	NO DATA	1.90E-13	8.34E-07	1.09E-16
RU103	2.63E-07	NO DATA	1.12E-07	NO DATA	9.29E-07	9.79E-05	1.36E-05
RU105	1.40E-10	NO DATA	5.42E-11	NO DATA	1.76E-10	2.27E-06	1.13E-05
RU106	1.23E-05	NO DATA	1.55E-06	NO DATA	2.38E-05	2.01E-03	1.20E-04
AG110m	1.73E-06	1.64E-06	9.99E-07	NO DATA	3.13E-06	8.44E-04	3.41E-05
TE125m	6.10E-07	2.80E-07	8.34E-08	1.75E-07	NO DATA	6.70E-05	9.38E-06
TE127m	2.25E-06	1.02E-06	2.73E-07	5.48E-07	8.17E-06	2.07E-04	1.99E-05
TE127	2.51E-10	1.14E-10	5.52E-11	1.77E-10	9.10E-10	1.40E-06	1.01E-05
TE129m	1.74E-06	8.23E-07	2.81E-07	5.72E-07	6.49E-06	2.47E-04	5.06E-05
TE129	8.87E-12	4.22E-12	2.20E-12	6.48E-12	3.32E-11	4.12E-07	2.02E-07
TE131m	1.23E-08	7.51E-09	5.03E-09	9.06E-09	5.49E-08	2.97E-05	7.76E-05
TE131	1.97E-12	1.04E-12	6.30E-13	1.55E-12	7.72E-12	2.92E-07	1.89E-09
TE132	4.50E-08	3.63E-08	2.74E-08	3.07E-08	2.44E-07	5.61E-05	5.79E-05
I 130	7.80E-07	2.24E-06	8.96E-07	1.86E-04	3.44E-06	NO DATA	1.14E-06
I 131	4.43E-06	6.14E-06	3.30E-06	1.83E-03	1.05E-05	NO DATA	8.11E-07
I 132	1.99E-07	5.47E-07	1.97E-07	1.89E-05	8.65E-07	NO DATA	1.59E-07
I 133	1.52E-06	2.56E-06	7.78E-07	3.65E-04	4.49E-06	NO DATA	1.29E-06
I 134	1.11E-07	2.90E-07	1.05E-07	4.94E-06	4.58E-07	NO DATA	2.55E-09
I 135	4.62E-07	1.18E-06	4.36E-07	7.76E-05	1.86E-06	NO DATA	8.69E-07
CS134	3.28E-05	1.41E-04	6.86E-05	NO DATA	4.69E-05	1.83E-05	1.22E-06
CS136	6.44E-06	2.42E-05	1.71E-05	NO DATA	1.38E-05	2.22E-06	1.36E-06
CS137	8.38E-05	1.06E-04	3.89E-05	NO DATA	3.80E-05	1.51E-05	1.06E-06
CS138	5.82E-08	1.07E-07	5.58E-08	NO DATA	8.28E-08	9.84E-09	3.38E-11
BA139	1.67E-10	1.18E-13	4.87E-12	NO DATA	1.11E-13	8.08E-07	8.06E-07

TABLE 1.7

INHALATION DOSE FACTORS FOR TEENAGER
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	6.84E-06	8.38E-09	4.40E-07	NO DATA	2.85E-09	2.54E-04	2.86E-05
BA141	1.78E-11	1.32E-14	5.93E-13	NO DATA	1.23E-14	4.11E-07	9.33E-14
BA142	4.62E-12	4.63E-15	2.84E-13	NO DATA	3.92E-15	2.39E-07	5.99E-20
LA140	5.99E-08	2.95E-08	7.82E-09	NO DATA	NO DATA	2.68E-05	6.09E-05
LA142	1.20E-10	5.31E-11	1.32E-11	NO DATA	NO DATA	1.27E-06	1.50E-06
CE141	3.55E-06	2.37E-06	2.71E-07	NO DATA	1.11E-06	7.67E-05	1.58E-05
CE143	3.32E-08	2.42E-08	2.70E-09	NO DATA	1.08E-08	1.63E-05	3.19E-05
CE144	6.11E-04	2.53E-04	3.28E-05	NO DATA	1.51E-04	1.67E-03	1.08E-04
PR143	1.67E-06	6.64E-07	8.28E-08	NO DATA	3.86E-07	6.04E-05	2.67E-05
PR144	5.37E-12	2.20E-12	2.72E-13	NO DATA	1.26E-12	2.19E-07	2.94E-14
ND147	9.83E-07	1.07E-06	6.41E-08	NO DATA	6.28E-07	4.65E-05	2.28E-05
W 187	1.50E-09	1.22E-09	4.29E-10	NO DATA	NO DATA	5.92E-06	2.21E-05
NP239	4.23E-08	3.99E-09	2.21E-09	NO DATA	1.25E-08	8.11E-06	1.65E-05

Table 1.8

EXTERNAL DOSE FACTORS FOR STANDING ON CONTAMINATED GROUND
(mrem/hr per pCi/m^2)

Element	Total Body	Skin
H-3	0.0	0.0
C-14	0.0	0.0
Na-24	2.50E-08	2.90E-08
P-32	0.0	0.0
Cr-51	2.20E-10	2.60E-10
Mn-54	5.80E-09	6.80E-09
Mn-56	1.10E-08	1.30E-08
Fe-55	0.0	0.0
Fe-59	8.00E-09	9.40E-09
Co-58	7.00E-09	8.20E-09
Co-60	1.70E-08	2.00E-08
Ni-63	0.0	0.0
Ni-65	3.70E-09	4.30E-09
Cu-64	1.50E-09	1.70E-09
Zn-65	4.00E-09	4.60E-09
Zn-69	0.0	0.0
Br-83	6.40E-11	9.30E-11
Br-84	1.20E-08	1.40E-08
Br-85	0.0	0.0
Rb-86	6.30E-10	7.20E-10
Rb-88	3.50E-09	4.00E-09
Rb-89	1.50E-08	1.80E-08
Sr-89	5.60E-13	6.50E-13
Sr-91	7.10E-09	8.30E-09
Sr-92	9.00E-09	1.00E-08
Y-90	2.20E-12	2.60E-12
Y-91m	3.80E-09	4.40E-09
Y-91	2.40E-11	2.70E-11
Y-92	1.60E-09	1.90E-09
Y-93	5.70E-10	7.80E-10
Zr-95	5.00E-09	5.80E-09
Zr-97	5.50E-09	6.40E-09
Nb-95	5.10E-09	6.00E-09
Mo-99	1.90E-09	2.20E-09
Tc-99m	9.60E-10	1.10E-09
Tc-101	2.70E-09	3.00E-09
Ru-103	3.60E-09	4.20E-09
Ru-105	4.50E-09	5.10E-09
Ru-106	1.50E-09	1.80E-09

Table 1.8

EXTERNAL DOSE FACTORS FOR STANDING ON CONTAMINATED GROUND
(mrem/hr per pCi/m^2)

Element	Total Body	Skin
Ag-110m	1.80E-08	2.10E-08
Te-125m	3.50E-11	4.80E-11
Te-127m	1.10E-12	1.30E-12
Te-127	1.00E-11	1.10E-11
Te-129m	7.70E-10	9.00E-10
Te-129	7.10E-10	8.40E-10
Te-131m	8.40E-09	9.90E-09
Te-131	2.20E-09	2.60E-09
Te-132	1.70E-09	2.00E-09
I-130	1.40E-08	1.70E-08
I-131	2.80E-09	3.40E-09
I-132	1.70E-08	2.00E-08
I-133	3.70E-09	4.50E-09
I-134	1.60E-08	1.90E-08
I-135	1.20E-08	1.40E-08
Cs-134	1.20E-08	1.40E-08
Cs-136	1.50E-08	1.70E-08
Cs-137	4.20E-09	4.90E-09
Cs-138	2.10E-08	2.40E-08
Ba-139	2.40E-09	2.70E-09
Ba-140	2.10E-09	2.40E-09
Ba-141	4.30E-09	4.90E-09
Ba-142	7.90E-09	9.00E-09
La-140	1.50E-08	1.70E-08
La-142	1.50E-08	1.80E-08
Ce-141	5.50E-10	6.20E-10
Ce-143	2.20E-09	2.50E-09
Ce-144	3.20E-10	3.70E-10
Pr-143	0.0	0.0
Pr-144	2.00E-10	2.30E-10
Nd-147	1.00E-09	1.20E-09
W-187	3.10E-09	3.60E-09
Np-239	9.50E-10	1.10E-09

TABLE 1.9

BIG ROCK POINT GASEOUS DESIGN OBJECTIVE ANNUAL

QUANTITIES BASED ON TABLE 1.4 CRITICAL RECEPTORS

Nuclide	Organ	Dose Factor mrem/Ci	Design Objective Annual Quantity (Ci)
H-3	Total Body-C	8.24E-08	6.07E+06
C-14	Bone-C	5.69E-03	2.09E+03
Cr-51	GI Tract-T	2.62E-04	5.73E+04
Mn-54	GI Tract-T	4.45E-02	3.37E+02
Fe-55	Bone-C	1.45E-02	1.03E+03
Co-58	GI Tract-T	1.82E-02	8.24e+02
Fe-59	GI Tract-T	5.61E-03	8.91E+02
Co-60	GI Tract-T	4.86E-01	3.09E+01
Zn-65	Liver-I	1.13E-01	1.33E+02
Sr-89	Bone-C	6.37E-01	2.36E+01
Sr-90	Bone-C	2.64E+01	5.69E-01
Zr-95	GI Tract-T	2.75E-02	5.45E+02
Sb-124	GI Tract-T	6.66E-02	2.25E+02
Ar-41	Total Body	9.10E-06	6.60E+06
Kr-83m	Skin	2.21E-08	6.79E+08
Kr-85	Skin	2.23E-08	6.73E+08
Kr-85m	Total Body	1.29E-06	3.88E+06
Kr-87	Skin	2.07E-06	7.25E+06
Kr-88	Total Body	1.58E-06	3.16E+06
Kr-89	Total Body	2.27E-06	2.20E+06
Xe-131m	Skin	9.84E-07	1.52E+07
Xe-133	Total Body	3.39E-07	1.47E+07
Xe-133m	Skin	2.06E-06	7.32E+06
Xe-135	Total Body	2.04E-06	2.45E+06
Xe-135m	Total Body	2.22E-06	2.25E+06
Xe-137	Skin	3.80E-06	3.95E+06
Xe-138	Total Body	6.02E-06	8.31E+05
I-131	Thyroid-I	2.46E+00	6.10E+00
I-133	Thyroid-I	2.60E-02	5.77E+02
Cs-134	Liver-C	6.22E-01	2.41E+01
Cs-136	Total Body-I	1.11E-02	4.50E+02
Cs-137	Bone-C	6.78E-01	2.21E+01
Ba-140	Bone-C	3.74E-03	4.01E+03
Ce-141	GI Tract-T	9.33E-03	1.61E+03
Ce-144	GI Tract-T	2.39E-01	6.28E+01
N-13	Total Body-C	6.81E-08	7.34E+07
Na-24	Total Body-I	2.62E-04	1.91E+04

TABLE 1.9

BIG ROCK POINT GASEOUS DESIGN OBJECTIVE ANNUAL QUANTITIES BASED ON TABLE 1.4 CRITICAL RECEPTORS

Nuclide	Organ	Dose Factor mrem/Ci	Design Objective Annual Quantity (Ci)
Mn-56	GI Tract-C	2.00E-04	7.54E+04
Co-57	GI Tract-T	6.01E-03	2.50E+03
Ni-63	Bone-T	8.63E-01	1.74E+01
Ni-65	GI Tract-C	1.29E-04	1.16E+05
Br-82	Total Body-I	1.05E-03	4.76E+03
Rb-88	Total Body-C	6.14E-07	8.14E+06
Sr-91	Bone-I	6.44E-02	2.33E+02
Sr-92	GI Tract-C	3.75E-04	4.00E+04
Nb-95	GI Tract-A	3.20E-02	4.69E+02
Mo-99	Kidney-I	1.50E-03	1.00E+04
TC-99	GI Tract-A	8.75E-02	1.71E+02
Tc-99m	GI Tract-T	1.31E-06	1.14E+06
Ru-103	GI Tract-A	4.89E-02	3.07E+02
Ru-106	GI Tract-C	1.65E-04	9.09E+04
Sb-125	GI Tract-T	3.08E-02	4.87E+02
Te-127	GI Tract-T	1.30E-04	1.16E+06
I-129	Thyroid-A	2.59E+01	5.79E-01
I-132	Thyroid-C	2.99E-04	5.02E+04
I-134	Thyroid-C	6.91E-06	2.17E+06
I-135	Thyroid-C	1.29E-03	1.16E+04
La-140	GI Tract-T	1.49E-03	1.01E+04
Tc-101	GI Tract-I	4.92E-06	3.06E+06
Ag-110m	GI Tract-T	1.96E-01	7.69E+01
Cs-138	Total Body-C	4.82E-03	1.04E+03
Ba-139	GI Tract-C	8.20E-06	1.83E+06
Np-239	GI Tract-T	3.82E-04	3.93E+04
Pu-238	Bone-T	3.80E+01	3.95E-01
Pu-239	Bone-T	4.39E+01	3.42E-01
Pu-241	Bone-T	9.26E-01	1.62E+01
Am-241	Bone-T	1.50E+01	1.00E+00
Cm-242	Lung-T	8.70E-01	1.72E+01
Cm-244	Bone-T	9.18E+00	1.63E+00

2. LIQUID EFFLUENTS

2.1 ALLOWABLE CONCENTRATION

2.1.1 ODCM Requirement

Requirement 2.1.1 of ODCM Section I specifies that the concentration of radioactive material released at any time from the site to unrestricted areas shall be limited to ten times the effluent concentration specified in 10 CFR 20, Appendix B, Table 2, Column 2 for nuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} µCi/ml total activity. To ensure compliance, the following approach will be used for each release.

2.1.2 Pre-release Analysis

Most tanks will be recirculated through two volume changes prior to sampling for release to the environment to ensure that a representative sample is obtained. The appropriate recirculation time for those tanks too large to provide two volume changes will be the time that the suspended particulate concentration reaches steady state. Either a one-time test, or prior sampling data, may be used to determine appropriate recirculation time.

Prior to release, a grab sample will be analyzed for each release, and the concentration of each radionuclide determined.

$$C_j = \sum_{i=1}^{n} C_{ij} \tag{II.1}$$

where:

C_j = Total concentration in the liquid effluent at the release point, µCi/ml, at release point j.

C_{ij} = Concentration of a single radionuclide i, µCi/ml, at release point j.

2.1.3 Total Release-Fraction

The total release-fraction (R_j) for each release point will be calculated by the relationship defined as:

$$R_{(j)} = \sum_i \frac{C_{ij}}{10 x EC_i} \qquad (II.2)$$

where:

C_{ij} - Undiluted effluent concentration of radionuclide i, as determined in Section II, part 2.1.2, μCi/ml, at release point j.

EC_i - The 10CFR20 effluent concentration limit (EC) of radionuclide i, as specified in Section II, part 2.1.1, μCi/ml. (Big Rock Point still uses MPC terminology in some Liquid Analyses Programs, although EC values are utilized.)

R_j - The total release-fraction for the release point.

The sum of the ratios at the discharge to the lake must be ≤ 1 due to the releases from any or all concurrent releases. The following relationship will assure this criterion is met:

$$f_1(R_1 - 1) + f_2(R_2 - 1) + f_3(R_3 - 1) \leq F \qquad (II.3)$$

where:

f_1, f_2, f_3 - The effluent flow rate (gallons/minute) for the respective releases, determined by plant personnel.

R_1, R_2, R_3 - The total release-fractions for the respective releases as determined by Equation II.2.

F - Minimum required dilution flow rate. Normally, a conservatively high dilution flow rate is used, that is, flow rate used = (b)(F) where b is a conservative factor greater than 1.0.

2.2 INSTRUMENT ALARM SET POINTS

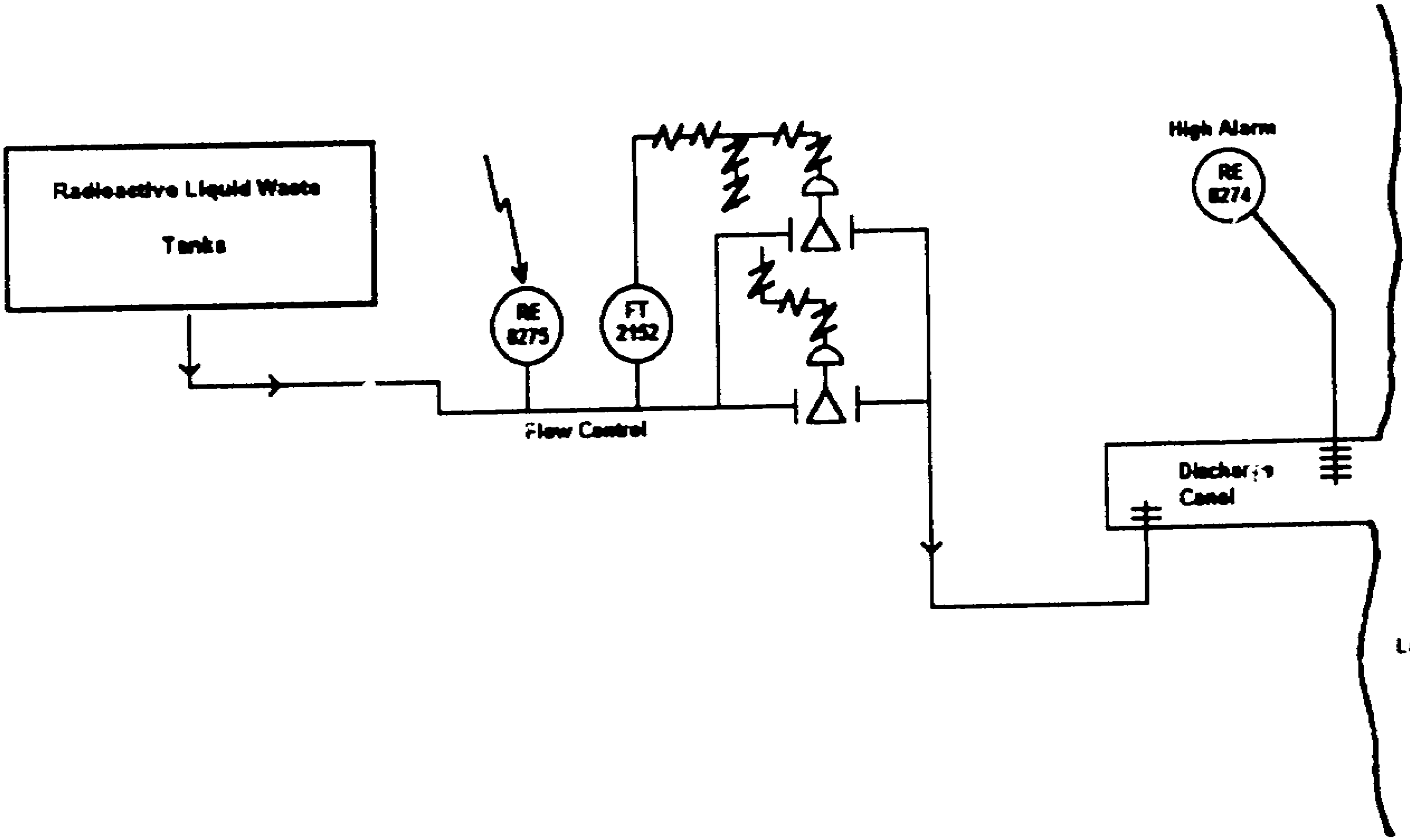
2.2.1 Set Point Determination

The set point for each liquid effluent monitor will be established using plant instructions. Concentration, flow rate, dilution, principal gamma emitter, geometry and detector efficiency are combined to give an equivalent set point in counts per minute (cpm) or other instrument output as may be appropriate to instrumentation modifications which may occur over the decommissioning interval. The physical and technical description, location and identification number for each liquid effluent radiation detector installed for operational period use is contained in Figure 2-2. Modifications or replacements of indicated instrumentation shall be performed only with instrumentation or sampling methods of equivalent or greater sensitivity.

The respective alarm/trip set points at each release point will be set such that the sum of the ratios at each point, as calculated by Equation II.2, will not exceed 1. The value of R is directly related to the total concentration calculated by Equation II.1. An increase in the concentration would indicate an increase in the value of R. A large increase would cause the limits of specification 2.2.1 of ODCM Section I to be exceeded. The minimum alarm/trip set point value is equal to the release concentration, but for ease of operation it may be desired that the set point (S) be set above the effluent concentration (C) by the same factor (b) utilized in setting dilution flow. That is:

$$S = b \times C \qquad \text{(II.4)}$$

Liquid effluent flow paths and release points are indicated in Figure 2.2.



LIQUID RELEASE DIAGRAM

2.3 LIQUID EFFLUENT DOSE

2.3.1 Section I ODCM Requirement

Requirement 2.3.1 of this ODCM (Section I, Relocated Technical Specifications) requires that the quantity of radionuclides released be limited such that the dose or dose commitment to an individual from radioactive materials in liquid effluents released to unrestricted areas from each reactor (see Figure 2.1) will not exceed:

a. During any calendar quarter, 1.5 mrem to the total body and 5 mrem to any organ, and

b. During any calendar year, 3 mrem to the total body and 10 mrem to any organ.

2.3.2 Release Analysis (Design Basis Quantity Fraction)

Per ODCM Surveillance Requirement 3.3 of ODCM Section I, the cumulative DBQ fraction for nuclides released is summed at least every 31 days to assure that the sum of nuclide fractions released does not exceed 1.0, year to date and 0.5 in any calendar quarter.

Calculations shall be performed according to the formula:

$$\sum_i \frac{A_i}{(DBQ)_i} = \text{Fraction of DBQ} \qquad (II.5)$$

where:

A_i = Cumulative quarterly or annual activity of nuclide i identified in liquid release (C_i).

$(DBQ)_i$ = Design objective annual quantity of radionuclide i from Table 2.2 (Ci).

Radionuclides may be omitted from the summation if they fall under the criteria of allowed emission specified by Note 5 to Appendix B, 10 CFR 20.

2.3.3 Exceeding DBQ Limits

The design basis quantities are derived in such a conservative manner that doses may be greatly overestimated by this technique. As a consequence of this conservatism, and in light of historically consistent operations with releases well below annual design basis quantities, Surveillance Requirement 2.3.3 of ODCM Section I does not specify monthly dose projections. However, this Surveillance Requirement specifies a cumulative dose contribution to be determined for current quarter and year at least once per 31 days. If at any time this calculation, by Equation II.5, results in values greater than 0.5 for a given quarter or

1.0 for year to date, the NRC LADTAP code will be run to ensure that ODCM Requirement 2.3.1 of ODCM Section I has been met.

2.3.4 <u>Dose Calculation</u>

Values for the design basis quantities (Ci), and the dose per curie $(D_c/C_c)_i$ for each nuclide i shown in Table 2.2, were calculated as follows.

a. Water Ingestion

The dose to an individual from ingestion of radioactivity from any source is described by the following equation:

$$D_j = \sum_{i=1}^{i} (DCF)_{ij} \times I_i = \text{mrem} \qquad \text{(II.6)}$$

where:

D_j - Dose for the j^{th} organ from radionuclides released, mrem.

j - The organ of interest.

$(DCF)_{ij}$ - Ingestion dose commitment factor for the j^{th} organ from the i^{th} radionuclide mrem/pCi (see Table 2.1).

I_i - Activity ingested of the i^{th} radionuclide, pCi.

I_i is described by:

$$I_i = \frac{(A_i)(V)(365)(10^6)}{(800)(d)} = \text{pCi} \qquad \text{(II.7)}$$

where:

365 - Days per year.

A_i - Annual activity released of i^{th} radionuclide, μCi.

V - Average rate of water consumption; adult - 2,000 ml/d, teen and child - 1,400 ml/d, infant - 900 ml/d (Reg Guide 1.109).

d - Dilution water flow for year (ml).

800 - Dispersion factor from discharge to nearest drinking water supply.

10^6 - Converts μCi to pCi.

The dose equation then becomes:

$$D_j = \frac{(4.56E5)(V)}{d} \sum_{i=1}^{i} (DCF)_{ij} \times A_i = \text{mrem} \quad \text{(II.8)}$$

b. Fish Ingestion

The dose to an individual from the consumption of fish is described by Equation II.10. In this case the activity ingested of the i^{th} radionuclide (I_i) is described by:

$$I_i = \frac{A_i B_i F}{15d} (10^9) = \text{pCi} \quad \text{(II.9)}$$

where:

A_i - Annual activity released of i^{th} radionuclide, μCi.

B_i - Fish concentration factor of i^{th} radionuclide, $\frac{\mu Ci/g}{\mu Ci/ml}$ (see Table 2.0).

F - Amount of fish eaten per year; adult - 21 kg, teen - 16 kg, child - 6.9 kg and infant - none.

15 - Dispersion factor from discharge to fish exposure point.

d - Dilution water flow for year (ml).

10^9 - Converts μCi to pCi and gram to kg.

Substitution of Equation II.9 into Equation II.6 gives:

$$D_j = \frac{(6.7E07)(F)}{d} \sum_{i=1}^{i} (A_i)(B_i)(DCF_{ij}) = \text{mrem} \quad (II.10)$$

c. Releasing Radionuclides Not Listed in Table 2.2

Table 2.2 contains all nuclides identified to date as routine constituents of liquid releases at Big Rock Point Plant, plus those common to boiling water reactors in general, even if not previously detected at Big Rock Point. From time to time, however, other nuclides may be detected.

If the unlisted nuclide constitutes less than 10% of the EC-fraction for the release, and all unlisted nuclides total less than 25% of the EC-fraction, the nuclide may be considered not present.

If the unlisted nuclide constitutes greater than 10% of the EC-fraction, or all unlisted nuclides together constitute greater than 25%, then each nuclide should be assigned a DBQ equal to the most conservative value listed for the physical form of the nuclide involved (noble gas, halogen or particulate).

Should a nuclide not listed in Table 2.2 begin to appear in significant quantities on a routine basis, revision to this ODCM should be made in order to include a design basis quantity specific to that nuclide.

2.3.5 Annual Analysis

A complete analysis utilizing the NRC computer code LADTAP (either by running the code, or by normalization of data to results of a previous year's run) for the total release source will be done annually in conjunction with the annual environmental report. This analysis will provide estimates of dose to the total body and various organs in addition to the dose limiting organs considered in the method of Section 2. The following approach is utilized in LADTAP. The dose to the j^{th} organ from m radionuclides, D_j, is described by:

$$D_j = \sum_{i=1}^{i} D_{ij} = \text{rem} \quad (II.11)$$

$$D_j = \sum_{i=1}^{i} (DCF)_{ij} \times I_i = \text{rem} \quad (II.12)$$

where:

D_{ij} - Dose to the j^{th} organ from the i^{th} radionuclide, rem.

j - The organ of interest (bone, GI tract, thyroid, liver, kidney, lung or total body).

$(DCF)_{ij}$ - Ingestion dose commitment factor for the j^{th} organ from the i^{th} radionuclide, rem/μCi (see Table 2.1).

I_i - Activity ingested of the i^{th} radionuclide, μCi.

I_i for water ingestion is described by:

$$I_i = \frac{A_i V \tau}{\nu d} \mu Ci \qquad (II.13)$$

and for fish ingestion I_i is described by:

$$I_i = \frac{A_i B_i F \tau}{\nu d} \mu Ci \qquad (II.14)$$

where:

A_i - Activity released of i^{th} radionuclide during the year, μCi.

V - Average rate of water consumption (Table 2.2).

τ - Number of days during the year (365 d).

ν - Dispersion factor from point of discharge to point of exposure (Table 2.2).

d - Dilution water volume (ml).

B_i - Fish concentration factor of the i^{th} radionuclide, $\frac{\mu Ci/g}{\mu Ci/ml}$ (Table 2.0).

F - Amount of fish eaten per day (Table 2.2).

2.4 OPERABILITY OF LIQUID RADWASTE EQUIPMENT

The Big Rock Point liquid radwaste system is designed to reduce the radioactive materials in liquid wastes prior to their discharge by filtration, ion exchange, decay or shipment for disposal so that radioactivity in liquid effluent releases to unrestricted areas will not exceed Requirement 2.3.1 of ODCM Section I. Maintaining the cumulative DBQ fraction of releases assures compliance with this requirement. In addition, more than 30 years of operating experience has shown that design basis quantities never have been exceeded.

2.5 OFFSITE RELEASE RATE

10 CFR 50.36a requires that the release of radioactive materials be kept as low as is reasonably achievable. Appendix I to 10 CFR 50 provides the numerical guidelines on limiting conditions for operations to meet the as low as is reasonably achievable requirement.

The LADTAP code has been run to determine the dose due to drinking water at plant discharge concentration (800 x nearest lake water drinking water intake concentration). The source term used is given in Table 1.1. The most limiting dose due to a hypothetical individual drinking this water is 4.89E-01 mrem, whole body. The release rate which would result in a dose rate equivalent to 500 mrem/y (upon which the 10 times concentration limit of Requirement 2.2.1 of ODCM Section I is based) is the Curies/Year given in Table 1.1 (8.94) times 500/.489 or 9141 Ci/Yr = 2.9E-04 Ci/sec (~290 uCi/sec) if continued for a full year.

The above calculation is informational, supplied at NRC request for inclusion in the ODCM. The release rate value is not used in any plant calculations or related to any ODCM Requirements. The calculation is based upon exposure using the drinking water pathway and an average historical release from Big Rock Point.

Annual analyses are run for the report specified part 1.5 of ODCM Section III. LADTAP is used to calculate estimates of dose to the total body and limiting organs.

Radionuclides of highest dose consequence will remain predominately Cs-137 and Co-60 throughout the decommissioning interval. Iodine-131, which has been an important dose contributor during power operations, will not be present in effluents in detectable concentrations due to decay prior to implementation of this ODCM revision (minimum of 11.6 halflives of decay will have occurred. Cs-134 (also important during plant operation) will decay to significantly lower levels as decommissioning progresses.

BIOACCUMULATION FACTORS
Table 2.0

(μCi/g per μCi/ml)

Element	Freshwater Fish
H	9.0E-01
C	4.6E+03
Na	1.0E+02
P	3.0E+03
Cr	2.0E+02
Mn	4.0E+02
Fe	1.0E+02
Co	5.0E+01
Ni	1.0E+02
Cu	5.0E+01
Zn	2.0E+03
Br	4.2E+02
Rb	2.0E+03
Sr	3.0E+01
Y	2.6E+01
Zr	3.3E+00
Nb	3.0E+04
Mo	1.0E+01
Tc	1.5E+01
Ru	1.0E+01
Rh	1.0E+01
Te	4.0E+02
I	1.5E+01
Cs	2.0E+03
Ba	4.0E+00
La	2.5E+01
Ce	1.0E+00
Pr	2.5E+01
Nd	2.5E+01
W	1.2E+03
Np	1.0E+01

TABLE 2.1

ADULT INGESTION DOSE FACTORS
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07
C 14	2.84E-06	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07
NA 24	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06
P 32	1.93E-04	1.20E-05	7.46E-06	NO DATA	NO DATA	NO DATA	2.17E-05
CR 51	NO DATA	NO DATA	2.66E-09	1.59E-09	5.86E-10	3.53E-09	6.69E-07
MN 54	NO DATA	4.57E-06	8.72E-07	NO DATA	1.36E-06	NO DATA	1.40E-05
MN 56	NO DATA	1.15E-07	2.04E-08	NO DATA	1.46E-07	NO DATA	3.67E-06
FE 55	2.75E-06	1.90E-06	4.43E-07	NO DATA	NO DATA	1.06E-06	1.09E-06
FE 59	4.34E-06	1.02E-05	3.91E-06	NO DATA	NO DATA	2.85E-06	3.40E-05
CO 58	NO DATA	7.45E-07	1.67E-06	NO DATA	NO DATA	NO DATA	1.51E-05
CO 60	NO DATA	2.14E-06	4.72E-06	NO DATA	NO DATA	NO DATA	4.02E-05
NI 63	1.30E-04	9.01E-06	4.36E-06	NO DATA	NO DATA	NO DATA	1.88E-06
NI 65	5.28E-07	6.86E-08	3.13E-08	NO DATA	NO DATA	NO DATA	1.74E-06
CU 64	NO DATA	8.33E-08	3.91E-08	NO DATA	2.10E-07	NO DATA	7.10E-06
ZN 65	4.84E-06	1.54E-05	6.96E-06	NO DATA	1.03E-05	NO DATA	9.70E-06
ZN 69	1.03E-08	1.97E-08	1.37E-09	NO DATA	1.28E-08	NO DATA	2.96E-09
BR 83	NO DATA	NO DATA	4.02E-08	NO DATA	NO DATA	NO DATA	5.79E-08
BR 84	NO DATA	NO DATA	5.21E-08	NO DATA	NO DATA	NO DATA	4.09E-13
BR 85	NO DATA	NO DATA	2.14E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	2.11E-05	9.83E-06	NO DATA	NO DATA	NO DATA	4.16E-06
RB 88	NO DATA	6.05E-08	3.21E-08	NO DATA	NO DATA	NO DATA	8.36E-19
RB 89	NO DATA	4.01E-08	2.82E-08	NO DATA	NO DATA	NO DATA	2.33E-21
SR 89	3.08E-04	NO DATA	8.84E-06	NO DATA	NO DATA	NO DATA	4.94E-05
SR 90	7.58E-03	NO DATA	1.86E-03	NO DATA	NO DATA	NO DATA	2.19E-04
SR 91	5.67E-06	NO DATA	2.29E-07	NO DATA	NO DATA	NO DATA	2.70E-05
SR 92	2.15E-06	NO DATA	9.30E-08	NO DATA	NO DATA	NO DATA	4.26E-05
Y 90	9.62E-09	NO DATA	2.58E-10	NO DATA	NO DATA	NO DATA	1.02E-04
Y 91m	9.09E-11	NO DATA	3.52E-12	NO DATA	NO DATA	NO DATA	2.67E-10
Y 91	1.41E-07	NO DATA	3.77E-09	NO DATA	NO DATA	NO DATA	7.76E-05
Y 92	8.45E-10	NO DATA	2.47E-11	NO DATA	NO DATA	NO DATA	1.48E-05

TABLE 2.1

ADULT INGESTION DOSE FACTORS
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	2.68E-09	NO DATA	7.40E-11	NO DATA	NO DATA	NO DATA	8.50E-05
ZR 95	3.04E-08	9.75E-09	6.60E-09	NO DATA	1.53E-08	NO DATA	3.09E-05
ZR 97	1.68E-09	3.39E-10	1.55E-10	NO DATA	5.12E-10	NO DATA	1.05E-04
NB 95	6.22E-09	3.46E-09	1.86E-09	NO DATA	3.42E-09	NO DATA	2.10E-05
MO 99	NO DATA	4.31E-06	8.20E-07	NO DATA	9.76E-06	NO DATA	9.99E-06
TC 99m	2.47E-10	6.98E-10	8.89E-09	NO DATA	1.06E-08	3.42E-10	4.13E-07
TC101	2.54E-10	3.66E-10	3.59E-09	NO DATA	6.59E-09	1.87E-10	1.10E-21
RU103	1.85E-07	NO DATA	7.97E-08	NO DATA	7.06E-07	NO DATA	2.16E-05
RU105	1.54E-08	NO DATA	6.08E-09	NO DATA	1.99E-07	NO DATA	9.42E-06
RU106	2.75E-06	NO DATA	3.48E-07	NO DATA	5.31E-06	NO DATA	1.78E-04
AG110m	1.60E-07	1.48E-07	8.79E-08	NO DATA	2.91E-07	NO DATA	6.04E-05
TE125m	2.68E-06	9.71E-07	3.59E-07	8.06E-07	1.09E-05	NO DATA	1.07E-05
TE127m	6.77E-06	2.42E-06	8.25E-07	1.73E-06	2.75E-05	NO DATA	2.27E-05
TE127	1.10E-07	3.95E-08	2.38E-08	8.15E-08	4.48E-07	NO DATA	8.68E-06
TE129m	1.15E-05	4.29E-06	1.82E-06	3.95E-06	4.80E-05	NO DATA	5.79E-05
TE129	3.14E-08	1.18E-08	7.65E-09	2.41E-08	1.32E-07	NO DATA	2.37E-08
TE131m	1.73E-06	8.46E-07	7.05E-07	1.34E-06	8.57E-06	NO DATA	8.40E-05
TE131	1.97E-08	8.23E-09	6.22E-09	1.62E-08	8.63E-08	NO DATA	2.79E-09
TE132	2.52E-06	1.63E-06	1.53E-06	1.80E-06	1.57E-05	NO DATA	7.71E-05
I 130	7.56E-07	2.23E-06	8.80E-07	1.89E-04	3.48E-06	NO DATA	1.92E-06
I 131	4.16E-06	5.95E-06	3.41E-06	1.95E-03	1.02E-05	NO DATA	1.57E-06
I 132	2.03E-07	5.43E-07	1.90E-07	1.90E-05	8.65E-07	NO DATA	1.02E-07
I 133	1.42E-06	2.47E-06	7.53E-07	3.63E-04	4.31E-06	NO DATA	2.22E-06
I 134	1.06E-07	2.88E-07	1.03E-07	4.99E-06	4.58E-07	NO DATA	2.51E-10
I 135	4.43E-07	1.16E-06	4.28E-07	7.65E-05	1.86E-06	NO DATA	1.31E-06
CS134	6.22E-05	1.48E-04	1.21E-04	NO DATA	4.79E-05	1.59E-05	2.59E-06
CS136	6.51E-06	2.57E-05	1.85E-05	NO DATA	1.43E-05	1.96E-06	2.92E-06
CS137	7.97E-05	1.09E-04	7.14E-05	NO DATA	3.70E-05	1.23E-05	2.11E-06
CS138	5.52E-08	1.09E-07	5.40E-08	NO DATA	8.01E-08	7.91E-09	4.65E-13
BA139	9.70E-08	6.91E-11	2.84E-09	NO DATA	6.46E-11	3.92E-11	1.72E-07

TABLE 2.1

ADULT INGESTION DOSE FACTORS
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	2.03E-05	2.55E-08	1.33E-08	NO DATA	8.67E-09	1.46E-08	4.18E-05
BA141	4.71E-08	3.56E-11	1.59E-09	NO DATA	3.31E-11	2.02E-11	2.22E-17
BA142	2.13E-08	2.19E-11	1.34E-09	NO DATA	1.85E-11	1.24E-11	3.00E-26
LA140	2.50E-09	1.26E-09	3.33E-10	NO DATA	NO DATA	NO DATA	9.25E-05
LA142	1.28E-10	5.82E-11	1.45E-11	NO DATA	NO DATA	NO DATA	4.25E-07
CE141	9.36E-09	6.33E-09	7.18E-10	NO DATA	2.94E-09	NO DATA	2.42E-05
CE143	1.65E-09	1.22E-06	1.35E-10	NO DATA	5.37E-10	NO DATA	4.56E-05
CE144	4.88E-07	2.04E-07	2.62E-08	NO DATA	1.21E-07	NO DATA	1.65E-04
PR143	9.20E-09	3.69E-09	4.56E-10	NO DATA	2.13E-09	NO DATA	4.03E-05
PR144	3.01E-11	1.25E-11	1.53E-12	NO DATA	7.05E-12	NO DATA	4.33E-18
ND147	6.29E-09	7.27E-09	4.35E-10	NO DATA	4.25E-09	NO DATA	3.49E-05
W 187	1.03E-07	8.61E-08	3.01E-08	NO DATA	NO DATA	NO DATA	2.82E-05
NP239	1.19E-09	1.17E-10	6.45E-11	NO DATA	3.65E-10	NO DATA	2.40E-05

TABLE 2.1

INGESTION DOSE FACTORS FOR TEENAGER
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.06E-07	1.06E-07	1.06E-07	1.06E-07	1.06E-07	1.06E-07
C 14	4.06E-06	8.12E-07	8.12E-07	8.12E-07	8.12E-07	8.12E-07	8.12E-07
NA 24	2.30E-06	2.30E-06	2.30E-06	2.30E-06	2.30E-06	2.30E-06	2.30E-06
P 32	2.76E-04	1.71E-05	1.07E-05	NO DATA	NO DATA	NO DATA	2.32E-05
CR 51	NO DATA	NO DATA	3.60E-09	2.00E-09	7.89E-10	5.14E-09	6.05E-07
MN 54	NO DATA	5.90E-06	1.17E-06	NO DATA	1.76E-06	NO DATA	1.21E-05
MN 56	NO DATA	1.58E-07	2.81E-08	NO DATA	2.00E-07	NO DATA	1.04E-05
FE 55	3.78E-06	2.68E-06	6.25E-07	NO DATA	NO DATA	1.70E-06	1.16E-06
FE 59	5.87E-06	1.37E-05	5.29E-06	NO DATA	NO DATA	4.32E-06	3.24E-05
CO 58	NO DATA	9.72E-07	2.24E-06	NO DATA	NO DATA	NO DATA	1.34E-05
CO 60	NO DATA	2.81E-06	6.33E-06	NO DATA	NO DATA	NO DATA	3.66E-05
NI 63	1.77E-04	1.25E-05	6.00E-06	NO DATA	NO DATA	NO DATA	1.99E-06
NI 65	7.49E-07	9.57E-08	4.36E-08	NO DATA	NO DATA	NO DATA	5.19E-06
CU 64	NO DATA	1.15E-07	5.41E-08	NO DATA	2.91E-07	NO DATA	8.92E-06
ZN 65	5.76E-06	2.00E-05	9.33E-06	NO DATA	1.28E-05	NO DATA	8.47E-06
ZN 69	1.47E-08	2.80E-08	1.96E-09	NO DATA	1.83E-08	NO DATA	5.16E-08
BR 83	NO DATA	NO DATA	5.74E-08	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	7.22E-08	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	3.05E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	2.98E-05	1.40E-05	NO DATA	NO DATA	NO DATA	4.41E-06
RB 88	NO DATA	8.52E-08	4.54E-08	NO DATA	NO DATA	NO DATA	7.30E-15
RB 89	NO DATA	5.50E-08	3.89E-08	NO DATA	NO DATA	NO DATA	8.43E-17
SR 89	4.40E-04	NO DATA	1.26E-05	NO DATA	NO DATA	NO DATA	5.24E-05
SR 90	8.30E-03	NO DATA	2.05E-03	NO DATA	NO DATA	NO DATA	2.33E-04
SR 91	8.07E-06	NO DATA	3.21E-07	NO DATA	NO DATA	NO DATA	3.66E-05
SR 92	3.05E-06	NO DATA	1.30E-07	NO DATA	NO DATA	NO DATA	7.77E-05
Y 90	1.37E-08	NO DATA	3.69E-10	NO DATA	NO DATA	NO DATA	1.13E-04
Y 91m	1.29E-10	NO DATA	4.93E-12	NO DATA	NO DATA	NO DATA	6.09E-09
Y 91	2.01E-07	NO DATA	5.39E-09	NO DATA	NO DATA	NO DATA	8.24E-05
Y 92	1.21E-09	NO DATA	3.50E-11	NO DATA	NO DATA	NO DATA	3.32E-05

TABLE 2.1

INGESTION DOSE FACTORS FOR TEENAGER
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	3.33E-09	NO DATA	1.05E-10	NO DATA	NO DATA	NO DATA	1.17E-04
ZR 95	4.12E-08	1.30E-08	8.94E-09	NO DATA	1.91E-08	NO DATA	3.00E-05
ZR 97	2.37E-09	4.69E-10	2.16E-10	NO DATA	7.11E-10	NO DATA	1.27E-04
NB 95	8.22E-09	4.56E-09	2.51E-09	NO DATA	4.42E-09	NO DATA	1.95E-05
MO 99	NO DATA	6.03E-06	1.15E-06	NO DATA	1.38E-05	NO DATA	1.08E-05
TC 99m	3.32E-10	9.26E-10	1.20E-08	NO DATA	1.38E-08	5.14E-10	6.08E-07
TC101	3.60E-10	5.12E-10	5.03E-09	NO DATA	9.26E-09	3.12E-10	8.75E-17
RU103	2.55E-07	NO DATA	1.09E-07	NO DATA	8.99E-07	NO DATA	2.13E-05
RU105	2.18E-08	NO DATA	8.46E-09	NO DATA	2.75E-07	NO DATA	1.76E-05
RU106	3.92E-06	NO DATA	4.94E-07	NO DATA	7.56E-06	NO DATA	1.88E-04
AG110m	2.05E-07	1.94E-07	1.18E-07	NO DATA	3.70E-07	NO DATA	5.45E-05
TE125m	3.83E-06	1.38E-06	5.12E-07	1.07E-06	NO DATA	NO DATA	1.13E-05
TE127m	9.67E-06	3.43E-06	1.15E-06	2.30E-06	3.92E-05	NO DATA	2.41E-05
TE127	1.58E-07	5.60E-08	3.40E-08	1.09E-07	6.40E-07	NO DATA	1.22E-05
TE129m	1.63E-05	6.05E-06	2.58E-06	5.26E-06	6.82E-05	NO DATA	6.12E-05
TE129	4.48E-08	1.67E-08	1.09E-08	3.20E-08	1.88E-07	NO DATA	2.45E-07
TE131m	2.44E-06	1.17E-06	9.76E-07	1.76E-06	1.22E-05	NO DATA	9.39E-05
TE131	2.79E-08	1.15E-08	8.72E-09	2.15E-08	1.22E-07	NO DATA	2.29E-09
TE132	3.49E-06	2.21E-06	2.08E-06	2.33E-06	2.12E-05	NO DATA	7.00E-05
I 130	1.03E-06	2.98E-06	1.19E-06	2.43E-04	4.59E-06	NO DATA	2.29E-06
I 131	5.85E-06	8.19E-06	4.40E-06	2.39E-03	1.41E-05	NO DATA	1.62E-06
I 132	2.79E-07	7.30E-07	2.62E-07	2.46E-05	1.15E-06	NO DATA	3.18E-07
I 133	2.01E-06	3.41E-06	1.04E-06	4.76E-04	5.98E-06	NO DATA	2.58E-06
I 134	1.46E-07	3.87E-07	1.39E-07	6.45E-06	6.10E-07	NO DATA	5.10E-09
I 135	6.10E-07	1.57E-06	5.82E-07	1.01E-04	2.48E-06	NO DATA	1.74E-06
CS134	8.37E-05	1.97E-04	9.14E-05	NO DATA	6.26E-05	2.39E-05	2.45E-06
CS136	8.49E-06	3.38E-05	2.27E-05	NO DATA	1.84E-05	2.90E-06	2.72E-06
CS137	1.12E-04	1.49E-04	5.19E-05	NO DATA	5.07E-05	1.97E-05	2.12E-06
CS138	7.76E-08	1.49E-07	7.45E-08	NO DATA	1.10E-07	1.28E-08	6.76E-11
BA139	1.39E-07	9.78E-11	4.05E-09	NO DATA	9.22E-11	6.74E-11	1.24E-06

TABLE 2.1

INGESTION DOSE FACTORS FOR TEENAGER (MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	2.84E-05	3.48E-08	1.83E-06	NO DATA	1.18E-08	2.34E-08	4.38E-05
BA141	6.71E-08	5.01E-11	2.24E-09	NO DATA	4.65E-11	3.43E-11	1.43E-13
BA142	2.99E-08	2.99E-11	1.84E-09	NO DATA	2.53E-11	1.99E-11	9.18E-20
LA140	3.48E-09	1.71E-09	4.55E-10	NO DATA	NO DATA	NO DATA	9.82E-05
LA142	1.79E-10	7.95E-11	1.98E-11	NO DATA	NO DATA	NO DATA	2.42E-06
CE141	1.33E-08	8.88E-09	1.02E-09	NO DATA	4.18E-09	NO DATA	2.54E-05
CE143	2.35E-09	1.71E-06	1.91E-10	NO DATA	7.67E-10	NO DATA	5.14E-05
CE144	6.96E-07	2.88E-07	3.74E-08	NO DATA	1.72E-07	NO DATA	1.75E-04
PR143	1.31E-08	5.23[illegible]09	6.52E-10	NO DATA	3.04E-09	NO DATA	4.31E-05
PR144	4.30E-11	1.76E-11	2.18E-12	NO DATA	1.01E-11	NO DATA	4.74E-14
ND147	9.38E-09	1.02E-08	6.11E-10	NO DATA	5.99E-09	NO DATA	3.68E-05
W 187	1.46E-07	1.19E-07	4.17E-08	NO DATA	NO DATA	NO DATA	3.22E-05
NP239	1.76E-09	1.66E-10	9.22E-11	NO DATA	5.21E-10	NO DATA	2.67E-05

TABLE 2.1

INGESTION DOSE FACTORS FOR CHILD (MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	2.03E-07	2.03E-07	2.03E-07	2.03E-07	2.03E-07	2.03E-07
C 14	1.21E-05	2.42E-06	2.42E-06	2.42E-06	2.42E-06	2.42E-06	2.42E-06
NA 24	5.80E-06	5.80E-06	5.80E-06	5.80E-06	5.80E-06	5.80E-06	5.80E-06
P 32	8.25E-04	3.86E-05	3.18E-05	NO DATA	NO DATA	NO DATA	2.28E-05
CR 51	NO DATA	NO DATA	8.90E-09	4.94E-09	1.35E-09	9.02E-09	4.72E-07
MN 54	NO DATA	1.07E-05	2.85E-06	NO DATA	3.00E-06	NO DATA	8.98E-06
MN 56	NO DATA	3.34E-07	7.54E-08	NO DATA	4.04E-07	NO DATA	4.84E-05
FE 55	1.15E-05	6.10E-06	1.89E-06	NO DATA	NO DATA	3.45E-06	1.13E-06
FE 59	1.65E-05	2.67E-05	1.33E-05	NO DATA	NO DATA	7.74E-06	2.78E-05
CO 58	NO DATA	1.80E-06	5.51E-06	NO DATA	NO DATA	NO DATA	1.05E-05
CO 60	NO DATA	5.29E-06	1.56E-05	NO DATA	NO DATA	NO DATA	2.93E-05
NI 63	5.38E-04	2.88E-05	1.83E-05	NO DATA	NO DATA	NO DATA	1.94E-06
NI 65	2.22E-06	2.09E-07	1.22E-07	NO DATA	NO DATA	NO DATA	2.56E-05
CU 64	NO DATA	2.45E-07	1.48E-07	NO DATA	5.92E-07	NO DATA	1.15E-05
ZN 65	1.37E-05	3.65E-05	2.27E-05	NO DATA	2.30E-05	NO DATA	6.41E-06
ZN 69	4.38E-08	6.33E-08	5.85E-09	NO DATA	3.84E-08	NO DATA	3.99E-06
BR 83	NO DATA	NO DATA	1.71E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	1.98E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	9.12E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	6.70E-05	4.12E-05	NO DATA	NO DATA	NO DATA	4.31E-06
RB 88	NO DATA	1.90E-07	1.32E-07	NO DATA	NO DATA	NO DATA	9.32E-09
RB 89	NO DATA	1.17E-07	1.04E-07	NO DATA	NO DATA	NO DATA	1.02E-09
SR 89	1.32E-03	NO DATA	3.77E-05	NO DATA	NO DATA	NO DATA	5.11E-05
SR 90	1.70E-02	NO DATA	4.31E-03	NO DATA	NO DATA	NO DATA	2.29E-04
SR 91	2.40E-05	NO DATA	9.06E-07	NO DATA	NO DATA	NO DATA	5.30E-05
SR 92	9.03E-06	NO DATA	3.62E-07	NO DATA	NO DATA	NO DATA	1.71E-04
Y 90	4.11E-08	NO DATA	1.10E-09	NO DATA	NO DATA	NO DATA	1.17E-04
Y 91m	3.82E-10	NO DATA	1.39E-11	NO DATA	NO DATA	NO DATA	7.48E-07
Y 91	6.02E-07	NO DATA	1.61E-08	NO DATA	NO DATA	NO DATA	8.02E-05
Y 92	3.60E-09	NO DATA	1.03E-10	NO DATA	NO DATA	NO DATA	1.04E-04

TABLE 2.1

INGESTION DOSE FACTORS FOR CHILD
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.14E-08	NO DATA	3.13E-10	NO DATA	NO DATA	NO DATA	1.70E-04
ZR 95	1.16E-07	2.55E-08	2.27E-08	NO DATA	3.65E-08	NO DATA	2.66E-05
ZR 97	6.99E-09	1.01E-09	5.96E-10	NO DATA	1.45E-09	NO DATA	1.53E-04
NB 95	2.25E-08	8.76E-09	6.26E-09	NO DATA	8.23E-09	NO DATA	1.62E-05
MO 99	NO DATA	1.33E-05	3.29E-06	NO DATA	2.84E-05	NO DATA	1.10E-05
TC 99m	9.23E-10	1.81E-09	3.00E-08	NO DATA	2.63E-08	9.19E-10	1.03E-06
TC101	1.07E-09	1.12E-09	1.42E-08	NO DATA	1.01E-08	5.92E-10	3.56E-09
RU103	7.31E-07	NO DATA	2.81E-07	NO DATA	1.84E-06	NO DATA	1.89E-05
RU105	6.45E-08	NO DATA	2.34E-0[illegible]	NO DATA	5.67E-07	NO DATA	4.21E-05
RU106	1.17E-05	NO DATA	1.46E-06	NO DATA	1.58E-05	NO DATA	1.82E-04
AG110m	5.39E-07	3.64E-07	2.91E-07	NO DATA	6.78E-07	NO DATA	4.33E-05
TE125m	1.14E-05	3.09E-06	1.52E-06	3.20E-06	NO DATA	NO DATA	1.10E-05
TE127m	2.89E-05	7.78E-06	3.43E-06	6.91E-06	8.24E-05	NO DATA	2.34E-05
TE127	4.71E-07	1.27E-07	1.01E-07	3.26E-07	1.34E-06	NO DATA	1.84E-05
TE129m	4.87E-05	1.36E-05	7.56E-06	1.57E-05	1.43E-04	NO DATA	5.94E-05
TE129	1.34E-07	3.74E-08	3.18E-08	9.56E-08	3.92E-07	NO DATA	8.34E-06
TE131m	7.20E-06	2.49E-06	2.65E-06	5.12E-06	2.41E-05	NO DATA	1.01E-04
TE131	8.30E-08	2.53E-08	2.47E-08	6.35E-08	2.51E-07	NO DATA	4.36E-07
TE132	1.01E-05	4.47E-06	5.40E-06	6.51E-06	4.15E-05	NO DATA	4.50E-05
I 130	2.92E-06	5.90E-06	3.04E-06	6.50E-04	8.82E-06	NO DATA	2.76E-06
I 131	1.72E-05	1.73E-05	9.83E-06	5.72E-03	2.84E-05	NO DATA	1.54E-06
I 132	8.00E-07	1.47E-06	6.76E-07	6.82E-05	2.25E-06	NO DATA	1.73E-06
I 133	5.92E-06	7.32E-06	2.77E-06	1.36E-03	1.22E-05	NO DATA	2.95E-06
I 134	4.19E-07	7.78E-07	3.58E-07	1.79E-05	1.19E-06	NO DATA	5.16E-07
I 135	1.75E-06	3.15E-06	1.49E-06	2.79E-04	4.83E-06	NO DATA	2.40E-06
CS134	2.34E-04	3.84E-04	8.10E-05	NO DATA	1.19E-04	4.27E-05	2.07E-06
CS136	2.35E-05	6.46E-05	4.18E-05	NO DATA	3.44E-05	5.13E-06	2.27E-06
CS137	3.27E-04	3.13E-04	4.62E-05	NO DATA	1.02E-04	3.67E-05	1.96E-06
CS138	2.28E-07	3.17E-07	2.01E-07	NO DATA	2.23E-07	2.40E-08	1.46E-07
BA139	4.14E-0[illegible]	2.21E-10	1.20E-08	NO DATA	1.83E-10	1.30E-10	2.39E-05

TABLE 2.1

INGESTION DOSE FACTORS FOR CHILD
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	8.31E-05	7.28E-08	4.85E-06	NO DATA	2.37E-08	4.34E-08	4.21E-05
BA141	2.00E-07	1.12E-10	6.51E-09	NO DATA	9.69E-11	6.58E-10	1.14E-07
BA142	8.74E-08	6.29E-11	4.88E-09	NO DATA	5.09E-11	3.70E-11	1.14E-09
LA140	1.01E-08	3.53E-09	1.10E-09	NO DATA	NO DATA	NO DATA	9.84E-05
LA142	5.24E-10	1.67E-10	5.23E-11	NO DATA	NO DATA	NO DATA	3.31E-05
CE141	3.97E-08	1.98E-08	2.94E-09	NO DATA	8.68E-09	NO DATA	2.47E-05
CE143	6.99E-09	3.79E-06	5.49E-10	NO DATA	1.59E-09	NO DATA	5.55E-05
CE144	2.08E-06	6.52E-07	1.11E-07	NO DATA	3.61E-07	NO DATA	1.70E-04
PR143	3.93E-08	1.18E-08	1.95E-09	NO DATA	6.39E-09	NO DATA	4.24E-05
PR144	1.29E-10	3.99E-11	6.49E-12	NO DATA	2.11E-11	NO DATA	8.59E-08
ND147	2.79E-08	2.26E-08	1.75E-09	NO DATA	1.24E-08	NO DATA	3.58E-05
W 187	4.29E-07	2.54E-07	1.14E-07	NO DATA	NO DATA	NO DATA	3.57E-05
NP239	5.25E-09	3.77E-10	2.65E-10	NO DATA	1.09E-09	NO DATA	2.79E-05

TABLE 2.1

INGESTION DOSE FACTORS FOR INFANT
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	3.08E-07	3.08E-07	3.08E-07	3.08E-07	3.08E-07	3.08E-07
C 14	2.37E-05	5.06E-06	5.06E-06	5.06E-06	5.06E-06	5.06E-06	5.06E-06
NA 24	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05
P 32	1.70E-03	1.00E-04	6.59E-05	NO DATA	NO DATA	NO DATA	2.30E-05
CR 51	NO DATA	NO DATA	1.41E-08	9.20E-09	2.01E-09	1.79E-08	4.11E-07
MN 54	NO DATA	1.99E-05	4.51E-06	NO DATA	4.41E-06	NO DATA	7.31E-06
MN 56	NO DATA	8.18E-07	1.41E-07	NO DATA	7.03E-07	NO DATA	7.43E-05
FE 55	1.39E-05	8.98E-06	2.40E-06	NO DATA	NO DATA	4.39E-06	1.14E-06
FE 59	3.08E-05	5.38E-05	2.12E-05	NO DATA	NO DATA	1.59E-05	2.57E-05
CO 58	NO DATA	3.60E-06	8.93E-06	NO DATA	NO DATA	NO DATA	8.97E-06
CO 60	NO DATA	1.08E-05	2.55E-05	NO DATA	NO DATA	NO DATA	2.57E-05
NI 63	6.34E-04	3.92E-05	2.20E-05	NO DATA	NO DATA	NO DATA	1.95E-06
NI 65	4.70E-06	5.32E-07	2.42E-07	NO DATA	NO DATA	NO DATA	4.05E-05
CU 64	NO DATA	6.09E-07	2.82E-07	NO DATA	1.03E-06	NO DATA	1.25E-05
ZN 65	1.84E-05	6.31E-05	2.91E-05	NO DATA	3.06E-05	NO DATA	5.33E-05
ZN 69	9.33E-08	1.68E-07	1.25E-08	NO DATA	6.98E-08	NO DATA	1.37E-05
BR 83	NO DATA	NO DATA	3.63E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	3.82E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	1.94E-08	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	1.70E-04	8.40E-05	NO DATA	NO DATA	NO DATA	4.35E-06
RB 88	NO DATA	4.98E-07	2.73E-07	NO DATA	NO DATA	NO DATA	4.85E-07
RB 89	NO DATA	2.86E-07	1.97E-07	NO DATA	NO DATA	NO DATA	9.74E-08
SR 89	2.51E-03	NO DATA	7.20E-05	NO DATA	NO DATA	NO DATA	5.16E-05
SR 90	1.85E-02	NO DATA	4.71E-03	NO DATA	NO DATA	NO DATA	2.31E-04
SR 91	5.00E-05	NO DATA	1.81E-06	NO DATA	NO DATA	NO DATA	5.92E-05
SR 92	1.92E-05	NO DATA	7.13E-07	NO DATA	NO DATA	NO DATA	2.07E-04
Y 90	8.69E-08	NO DATA	2.33E-09	NO DATA	NO DATA	NO DATA	1.20E-04
Y 91m	8.10E-10	NO DATA	2.76E-11	NO DATA	NO DATA	NO DATA	2.70E-06
Y 91	1.13E-06	NO DATA	3.01E-08	NO DATA	NO DATA	NO DATA	8.10E-05
Y 92	7.65E-09	NO DATA	2.15E-10	NO DATA	NO DATA	NO DATA	1.46E-04

TABLE 2.1

INGESTION DOSE FACTORS FOR INFANT
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	2.43E-08	NO DATA	6.62E-10	NO DATA	NO DATA	NO DATA	1.92E-04
ZR 95	2.06E-07	5.02E-08	3.56E-08	NO DATA	5.41E-08	NO DATA	2.5[illegible]E-05
ZR 97	1.48E-08	2.54E-09	1.16E-09	NO DATA	2.56E-09	NO DATA	1.62E-04
NB 95	4.20E-08	1.73E-08	1.00E-08	NO DATA	1.24E-08	NO DATA	1.46E-05
MO 99	NO DATA	3.40E-06	6.63E-06	NO DATA	5.08E-05	NO DATA	1.12E-05
TC 99m	1.92E-09	3.96E-09	5.10E-08	NO DATA	4.26E-08	2.07E-09	1.15E-06
TC101	2.27E-09	2.86E-09	2.83E-08	NO DATA	3.40E-08	1.56E-09	4.86E-07
RU103	1.48E-08	NO DATA	4.95E-07	NO DATA	3.08E-08	NO DATA	1.80E-05
RU105	1.36E-07	NO DATA	4.58E-08	NO DATA	1.00E-06	NO DATA	5.41E-05
RU106	2.41E-05	NO DATA	3.01E-06	NO DATA	2.85E-05	NO DATA	1.83E-04
AG110m	9.96E-07	7.27E-07	4.81E-07	NO DATA	1.04E-06	NO DATA	3.77E-05
TE125m	2.33E-05	7.79E-06	3.15E-06	7.84E-06	NO DATA	NO DATA	1.11E-05
TE127m	5.58E-05	1.94E-05	7.08E-06	1.69E-05	1.44E-04	NO DATA	2.36E-05
TE127	1.00E-06	3.35E-07	2.15E-07	8.14E-07	2.44E-06	NO DATA	2.10E-05
TE129m	1.00E-04	3.43E-05	1.54E-05	3.84E-05	2.50E-04	NO DATA	5.97E-05
TE129	2.84E-07	9.79E-08	6.63E-08	2.38E-07	7.07E-07	NO DATA	2.27E-05
TE131m	1.52E-05	6.12E-06	5.05E-06	1.24E-05	4.21E-05	NO DATA	1.03E-04
TE131	1.76E-07	6.50E-08	4.94E-08	1.57E-07	4.50E-07	NO DATA	7.11E-06
TE132	2.08E-05	1.03E-05	9.61E-06	1.52E-05	6.44E-05	NO DATA	3.81E-05
I 130	6.00E-06	1.32E-05	5.30E-06	1.48E-03	1.45E-05	NO DATA	2.83E-06
I 131	3.59E-05	4.23E-05	1.86E-05	1.39E-02	4.94E-05	NO DATA	1.51E-06
I 132	1.66E-06	3.37E-06	1.20E-06	1.58E-04	3.76E-06	NO DATA	2.73E-06
I 133	1.25E-05	1.82E-05	5.33E-06	3.31E-03	2.14E-05	NO DATA	3.08E-06
I 134	8.69E-07	1.78E-06	6.33E-07	4.15E-05	1.99E-06	NO DATA	1.84E-06
I 135	3.64E-06	7.24E-06	2.64E-06	6.49E-04	8.07E-06	NO DATA	2.62E-06
CS134	3.77E-04	7.03E-04	7.10E-05	NO DATA	1.81E-04	7.42E-05	1.91E-06
CS136	4.59E-05	1.35E-04	5.04E-05	NO DATA	5.38E-05	1.10E-05	2.05E-06
CS137	5.22E-04	6.11E-04	4.33E-05	NO DATA	1.64E-04	6.64E-05	1.91E-06
CS138	4.81E-07	7.82E-07	3.79E-07	NO DATA	3.90E-07	6.09E-08	1.25E-06
BA139	8.81E-07	5.84E-10	2.55E-08	NO DATA	3.51E-10	3.54E-10	5.58E-05

TABLE 2.1

INGESTION DOSE FACTORS FOR INFANT
(MREM PER PCI INGESTED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	1.71E-04	1.71E-07	8.81E-06	NO DATA	4.06E-08	1.05E-07	4.20E-05
BA141	4.25E-07	2.91E-10	1.34E-08	NO DATA	1.75E-10	1.77E-10	5.19E-06
BA142	1.84E-07	1.53E-10	9.06E-09	NO DATA	8.81E-11	9.26E-11	7.59E-07
LA140	2.11E-08	8.32E-09	2.14E-09	NO DATA	NO DATA	NO DATA	9.77E-05
LA142	1.10E-09	4.04E-10	9.67E-11	NO DATA	NO DATA	NO DATA	6.86E-05
CE141	7.87E-08	4.80E-08	5.65E-09	NO DATA	1.48E-08	NO DATA	2.48E-05
CE143	1.48E-08	9.82E-06	1.12E-09	NO DATA	2.86E-09	NO DATA	5.73E-05
CE144	2.98E-06	1.22E-06	1.67E-07	NO DATA	4.93E-07	NO DATA	1.71E-04
PR143	8.13E-08	3.04E-08	4.03E-09	NO DATA	1.13E-08	NO DATA	4.29E-05
PR144	2.74E-10	1.06E-10	1.38E-11	NO DATA	3.84E-11	NO DATA	4.93E-06
ND147	5.53E-08	5.68E-08	3.48E-09	NO DATA	2.19E-08	NO DATA	3.60E-05
W 187	9.03E-07	6.28E-07	2.17E-07	NO DATA	NO DATA	NO DATA	3.69E-05
NP239	1.11E-08	9.93E-10	5.61E-10	NO DATA	1.98E-09	NO DATA	2.87E-05

Table 2.2

BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

Design objective annual quantities for liquid effluents were calculated utilizing the computer code LADTAP, a program for calculating radiation exposure to man from routine releases of nuclear reactor liquid effluents (reference NUREG/CR-1276).

Input parameters used are as follows:

Pathway	Age Group	Usage	Dilution	Process Times (Hr)
Fish	Adult	21.0 kg/yr	15.0	24.0
	Teen	16.0	15.0	24.0
	Child	6.9	15.0	24.0
	Infant	0.0	15.0	24.0
Drinking	Adult	730.0 L/yr	800.0	16.6
	Teen	510.0	800.0	16.6
	Child	510.0	800.0	16.6
	Infant	330.0	800.0	16.6
Shoreline	Adult	12.0 hr/yr	2.0	0.0
	Teen	67.0	2.0	0.0
	Child	14.0	2.0	0.0
	Infant	0.0	2.0	0.0
Swimming	Adult	12.0 hr/yr	2.0	0.0
	Teen	67.0	2.0	0.0
	Child	14.0	2.0	0.0
	Infant	0.0	2.0	0.0
Boating	Adult	100.0 hr/yr	15.0	0.0
	Teen	100.0	15.0	0.0
	Child	50.0	15.0	0.0
	Infant	0.0	15.0	0.0

The usage figures are obtained from Regulatory Guide 1.109 and are default values. Dilutions and the process time for drinking water were taken from the NUS study dated June 4, 1976. The minimum process times that can be utilized for fish and drinking are 24.0 hours and 12.0 hours respectively.

Table 2.2

BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

The following input parameters are used when running LADTAP for BRP:

1. 50-mile population - 1.54E06
2. Shore width factor - 0.3
3. Total discharge (ft^3/sec) - 109
4. Transit time for all pathways - 4.6
5. Sport fish harvest (kg/yr) - 3.29E05
6. Commercial fish harvest (kg/yr) - 1.70E06
7. Invertebrate and algae consumption - 0
8. Drinking water population - 7.07E03
9. Shoreline population usage (man-hours) - 3.8E07
10. Swimming population usage (man-hours) - 1.2E07
11. Boating population usage (man-hours) - 3.7E07

Table 2.2

BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

Nuclide	Dose Conversion Factors (mrem/Ci)	Individual/Organ	Design Objective Annual Quantity (Curies)
H-3	2.34E-06	Adult/TB	1.282×10^{6}
Na-24	3.95E-03	Teen/TB	759.49
Sc-46	1.24E-02	Teen/TB	241.94
Cr-51	1.90E-03	Adult/GI (LLI)	5,263.16
Mn-54	8.39E-02	Adult/GI (LLI)	119.19
Fe-55	5.50E-03	Child/Bone	1,818.18
Mn-56	1.22E-03	Teen/TB	2,459.02
Co-57	2.80E-03	Teen/TB	1,071.43
Co-58	6.96E-03	Teen/TB	431.66
Fe-59	4.93E-02	Adult/GI (LLI)	202.84
Co-60	2.90E-01	Teen/TB	10.34
Cu-64	1.48E-03	Teen/GI (LLI)	6,756.76
Ni-65	3.82E-04	Teen/TB	7,853.4
Zn-65	2.16E-01	Child/TB	13.89
Br-84	1.33E-03	Teen/TB	2,255.64
Rb-86	3.75E-01	Child/TB	8.0
Rb-88	4.54E-04	Teen/TB	6,607.93
Sr-89	1.93E-01	Child/Bone	51.81
Sr-90	3.34E+00	Adult/Bone	2.99
Sr-91	2.90E-03	Teen/GI (LLI)	3,448.28

Table 2.2

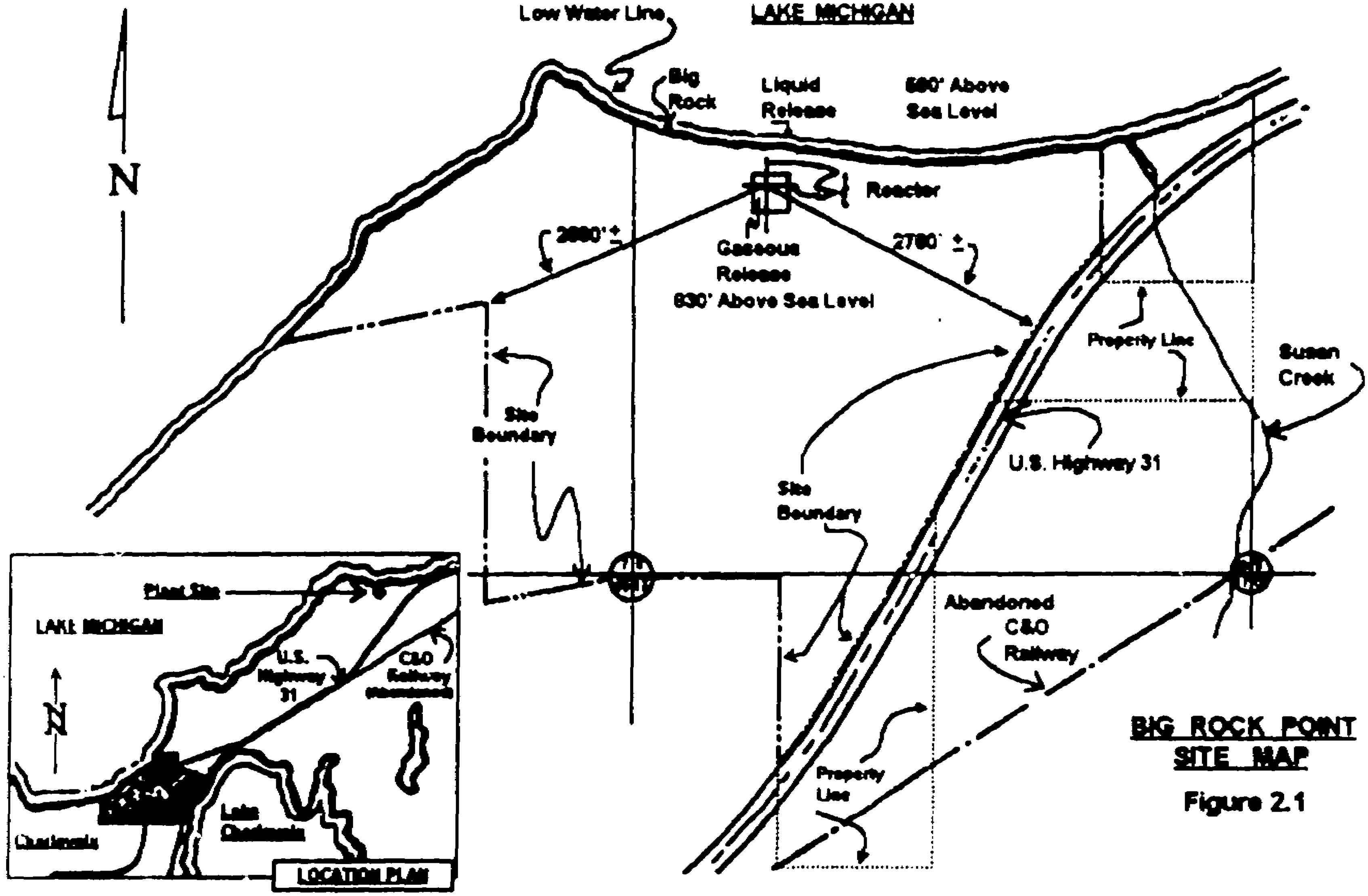
BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

Nuclide	Dose Conversion Factors (mrem/Ci)	Individual/Organ	Design Objective Annual Quantity (Curies)
Sr-92	9.94E-04	Teen/TB	3,018.11
Y-92	1.76E-04	Teen/TB	17,045.5
Nb-95	8.88E+00	Adult/GI (LLI)	1.13
Zr-95	3.82E-03	Teen/TB	785.34
Nb-97	4.56E-04	Teen/TB	6,578.95
Zr-97	2.74E-03	Teen/GI (LLI)	3,649.64
Mo-99	1.31E-03	Teen/Kidney	7,633.59
Tc-99m	9.33E-05	Teen/TB	32,154.3
Ru-103	1.69E-03	Teen/TB	1,775.15
Ag-110m	4.76E-02	Teen/TB	63.03
Cd-113m	7.38E-02	Adult/GI (LLI)	135.50
Sb-124	9.34E-03	Teen/TB	321.20
Sb-125	3.13E-02	Teen/TB	95.85
Te-127	9.04E-03	Teen/GI (LLI)	1,106.19
Te-127m	1.71E-01	Teen/Kidney	58.48
Te-129m	3.27E-01	Adult/GI (LLI)	30.58
I-130	1.40E-02	Child/Thyroid	714.29
I-131	4.07E-01	Child/Thyroid	24.57
Te-131m	2.78E-01	Adult/GI (LLI)	35.97
I-132	1.95E-06	Teen/TB	1.538×10^5
Te-132	3.59E-01	Adult/GI (LLI)	27.86
I-133	4.85E-02	Child/Thyroid	206.19
Cs-134	3.49E+00	Adult/TB	0.8596
I-134	1.59E-03	Teen/TB	1,886.79
I-135	1.91E-03	Child/Thyroid	5,235.6
Cs-136	5.05E-01	Adult/TB	5.94
Cs-137	2.08E+00	Adult/TB	1.44
Cs-138	1.52E-03	Teen/TB	1,973.68
Ba-139	3.05E-05	Teen/TB	98,360.7
Ba-140	2.75E-03	Adult/GI (LLI)	3,636.36
La-140	2.27E-02	Adult/GI (LLI)	440.53

Table 2.2

BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

Ce-141	2.30E-04	Teen/TB	13,043.5
Ce-144	4.08E-03	Adult/GI (LLI)	2,450.98
Eu-152	1.99E-01	Teen/TB	15.08
W-187	2.43E-01	Adult/GI (LLI)	41.15
Np-239	2.78E-03	Adult/GI (LLI)	3,597.12



BIG ROCK POINT
SITE MAP

Figure 2.1

3. URANIUM FUEL CYCLE DOSE

3.1 SPECIFICATION

In accordance with Action 3.2 of ODCM Section I, if either liquid or gaseous quarterly releases exceed the quantity which would cause offsite doses more than twice either of their specified limits, then the cumulative dose contributions from combined release plus direct radiation sources (from the reactor unit and radwaste storage tanks) shall be calculated. This calculation is performed to ensure that the annual (calendar year) dose or dose commitment to any member of the public is ≤ 25 mrem to the total body or any organ, except the thyroid, which shall be ≤ 75 mrem. The dose is to be determined for the member of the public projected to be the most highly exposed to these combined sources. If the results of this calculation show the dose to exceed either the 25 or 75 mrem limit, a special report shall be prepared and submitted to the Commission within 30 days, as described by Action 3.2. of ODCM Section I.

3.2 ASSUMPTIONS

3.2.1 The full time resident determined to be the maximally exposed individual (excluding infant) is assumed also to be a fisherman. This individual is assumed to drink water and ingest local fish at the rates specified in ODCM Section II, parts 2.3.4.a. and 2.3.4.b. (input parameters are summarized in Table 2.2).

3.2.2 Amount of shoreline fishing (at accessible shoreline adjacent to site security fence) is conservatively assumed as 48 hours per quarter (average of approximately 1/2 hour per day each day of the quarter) for the second and third quarters of the year, 36 hours for the fourth quarter and 18 hours for the first quarter.

3.2.3 The dose contribution due to uranium fuel cycle sources other than the plant is ignored in the calculation. This is based on the lack of any operations that fall in the "cycles" definition within a 5 mile radius of Big Rock Point.

3.3 DOSE CALCULATION

Maximum doses to the total body and internal organs of an individual shall be determined by use of LADTAP and GASPAR computer codes or optionally, by conservatively multiplying the mrem/Ci factors of Tables 1.9 and 2.2 by quantity released. Doses to like organs and total body shall be summed (organs are conservatively summed with total body when Tables 1.9 and 2.2 are utilized). Added to this sum will be a mean dose rate, calculated or measured for the shoreline due to the plant during the quarter in question, times the assumed fishing time:

$$D_{40_i} = D_G + D_L + (R_T)(T) \tag{III.1}$$

Where: D_{40_i} - 40 CFR 190 dose to organ (i) (mrem).

D_G - Cumulative dose to an individual organ from gaseous releases (mrem).

D_L - Cumulative dose to an individual organ from liquid releases (mrem).

R_T - Mean dose rate (direct radiation component) calculated to be applicable to Lake Michigan shoreline adjacent to plant site (mrem/hr).

T - Assumed shoreline fishing time for the quarter in question (hours) (see 3.2.2 above).

<u>NOTE</u>: For this calculation, the total body is conservatively assumed to be an additional organ.

ODCM Section III

Reporting and Major Modification Requirements

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1. RADIOLOGICAL EFFLUENT RELEASE REPORT

The Radioactive Effluent Release Report shall be submitted in accordance with 10CFR50.36a by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and Process Control Program and (2) in conformance with 10CFR50,36a and Section IV.B.1 of Appendix I to 10CFR50.

1.1 Estimate of Uncertainty

The report shall include an estimate of the uncertainty associated with the measurement of radioactive effluents. This error term is included to provide an estimate of the uncertainty and is not to be considered the absolute error associated with the measurements or to be used in determining compliance with these requirements.

Estimates for liquid releases will be based on a statistical analysis of a series of sample results (weighed appropriately for counting statistics) taken once a year from a minimum of one typical liquid radwaste tank. The error term for iodine and particulates released to the atmosphere will be based on the counting statistics for one stack or other release point effluent sample taken during the year.

The report shall include an estimate of the lower level of detection (in uCi/ml) if the unidentified portion of the release exceeds 10% of the total annual releases. This estimate of the lower level of detection will be made for gamma emitting isotopes listed in Appendix B of Regulatory Guide 1.21 (June 1974) and will be provided based on a typical background spectrum.

1.2 Supplemental Information

a. Batch Releases

The report should provide information relating to batch releases of liquid and gaseous effluents which are discharged to the environment. This information should include the number of releases, total time period for batch releases and the maximum, mean and minimum time periods of the releases.

b. Abnormal Releases

The number of abnormal releases and number of curies of radioactive material released to the environment during abnormal releases should be reported.

1.3 Gaseous Effluents

a. Gases

1) Total curies of fission and activation gases released.

2) Average release rates (uCi/s) of fission and activation gases for the quarterly periods covered by the report.

3) Percent of limit for releases of fission and activation gases.

4) Quarterly sums of total curies for each of the radionuclides determined to be released, based on analyses of fission and activation gases.

b. Iodines

1) Total curies of each of the isotopes I-131, Iodine 133 and Iodine-135 determined to be released.

2) Average release rate (uCi/s) of Iodine-131/133.

3) Percent of limit for I-131/133.

c. Particulates

1) Total curies of radioactive material in particulate form with half-lives greater than 8 days determined to have been released.

2) Average release rate (uCi/s) of radioactive material in particulate form with half-lives greater than 8 days.

3) Percent of limit for radioactive material in particulate form with half-lives greater than 8 days.

4) Total curies for each of the radionuclides in particulate form determined to be released based on analyses performed.

5) Total curies of gross alpha radioactivity of plant origin determined to be released.

d. Tritium

1) Total curies of tritium determined to be released in gaseous effluent.

2) Average release rate (uCi/s) of tritium.

3) Percentage of applicable limits for tritium.

1.4 Liquid Effluents

a. Mixed Fission and Activation Products

1) Total curies of radioactive material determined to be released in liquid effluents (not including tritium, dissolved and/or entrained gases, and alpha-emitting material).

2) Average concentrations (uCi/ml) of [illegible]d fission and activation products released to unrestricted areas, averaged over the quarterly periods covered by the report.

3) Percent of applicable limit of average concentrations released to unrestricted areas.

4) Quarterly sums of total curies for each of the radionuclides determined to be released in liquid effluents based on analyses performed.

b. Tritium

1) Total curies of tritium determined to be released in liquid effluents.

2) Average concentrations (uCi/ml) of tritium released to unrestricted areas, averaged over the quarterly periods covered by the report.

3) Percent of applicable limit of average concentrations released to unrestricted areas.

c. Dissolved and/or Entrained Gases

1) Total curies of gaseous radioactive material determined to be released in liquid effluents.

2) Average concentrations (uCi/ml) of dissolved and/or entrained gaseous radioactive material released to unrestricted areas, averaged over the quarterly periods covered by the report.

3) Percent of applicable limit of average concentrations released to unrestricted areas.

4) Quarterly sums of total curies for each of the gaseous radionuclides determined to be released in liquid effluents based on analyses performed.

d. Alpha Radioactivity

Total curies of gross alpha-emitting materials of plant origin determined to be released in liquid effluents.

- Volumes

    1) Total measured volume (liters), prior to dilution, of liquid effluent released.

    2) Total determined volume, in liters, of dilution water used during the period of the report.

1.5 Solid Waste

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Containe burial volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, asphalt).

1.6 Radiological Impact on Man

The radioactive Effluent Release Report shall include potential doses to individuals and populations calculated using measured effluent and averaged meteorological data in accordance with the methodologies of ODCM Section II.

a. Total body and significant organ doses (greater than 1 millirem) to individuals in restricted areas from receiving water-related exposure pathways.

b. Maximum offsite air doses greater than 1 millirad due to beta and gamma radiation at locations near ground level from gaseous effluents.

c. Organ doses greater than 1 millirem to individuals in unrestricted areas from radioactive iodine and radioactive material in particulate form from the major pathways of exposure.

d Total body doses greater than 1 manrem to the population and average doses greater than 1 millirem to individuals in the population from receiving water-related pathways to a distance of 50 miles from the site

- Total body doses greater than 1 manrem to the population and average doses greater than 1 millirem to individuals in the population from gaseous effluents to a distance of 50 miles from the site.

1.7 ODCM Changes

The Radiological Effluent Release Report shall include any changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM), including identification of new locations for dose calculations and/or environmental sampling (ODCM Section II, Table 1.4 and ODCM Section I, Table I.H-1, respectively).

2. Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the unit during the pervious calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations and analysis of trends of results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the ODCM and Sections IV.B.2, IV.B.3 and IV.C of Appendix I to 10CFR50.

The Radiological Environmental Operating Report shall include summaries, interpretation and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with previous environmental surveillance report data of similar type and an assessment of the observed impacts of the Plant operation on the environment. During the first year following permanent shutdown, the report also shall include the results of any land use census taken during the previous year.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results in the format of Table III-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The report also shall provide a summary description of the environmental monitoring program and a map depicting sample locations.

3. Nonroutine Reports

A report shall be submitted to the NRC in the event that:

3.1 The Radiological Environmental Monitoring Program is not substantially conducted as described in ODCM Section I.4

3.2 Dose to an offsite individual is estimated to have exceeded the requirements of 40 CFR Part 190 per ODCM Section [illegible]

3.3 An unusual or important event occurs due to plant operations that causes a significant radiological environmental impact or affects a potential environmental impact.

3.4 Nonroutine reports shall be submitted within 30 days of determining that an event has occurred to require such report.

4 Major Modification of Radioactive Waste Systems

4.1 Licensee Modifications:

4.1.1 Shall be reported to the NRC pursuant to 10 CFR Part 50.59. The discussion of each modification shall contain:

a. A summary of the evaluation that led to the determination that the modification could be made in accordance with 10 CFR Part 50.59.

b. A description of the equipment, components and processes involved, and the interfaces with other Plant systems.

c. Documentation of the fact that the modification was reviewed and found acceptable by the NRC.

4.1.2 Shall become effective upon review and acceptance by the Site Manager.

4.2 Definition of Major Radwaste System Modification

4.2.1 Purpose

The purpose of this definition is to assure that this requirement will be satisfied under clearly identifiable circumstances, and with the objective that current radwaste system capabilities are not jeopardized as long as those systems remain useful in limiting the amount of radioactivity in plant effluents.

4.2.2 Definition

A major radwaste system modification is a modification which would cause reduction in effluent treatment, either by bypassing a system or component for greater than 7 days, complete removal, or replacement with less efficient equipment. A major modification is not deemed to have occurred for removal of a system no longer utilized for waste treatment, such as, but not limited to:

a. Offgas system (including offgas decay line) following final plant shutdown.

b. Fuel pool cleanup system following fuel removal, with pool cleaned and placed in layup condition.

c. Any other systems whose normal feed or effluent path is taken permanently out of service by the decommissioning process.

Improvements or additions to improve efficiency shall not be considered major modifications

TABLE III-1

Environmental Radiological Monitoring Program Summary

Name of Facility ____________________ Docket No ____________

Location of Facility ____________________ Reporting Period ____________

(County, State)

Medium or Pathway Sampled (Unit of Measure)	Type/Total No. of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locs. Mean (f)[b] Range[b]	Location Name Distance and Direction	Mean (f)[b] Range[b]	Control Locations Mean (f)[b] Range[b]	Number of REPORTABLE OCCURRENCES
Air Particulates (pCi/m³)	Gross Beta 416	0.003	0.08 (200/312) (0.05-2.0)	Middletown 5 miles 340°	0.10 (5/52) (0.08-2.0)	0.08 (8/104) (0.05-1.40)	1
	Gamma Spec 32						
	Cs-137	0.003	0.05 (4/24) (0.03-0.13	Smithville 2.6 miles 160°	0.08 (2/4) (0.03-0.13)	<LLD	4
	Ba-140	0.003	0.03 (2/24) (0.01-0.08)	Podunk 4 miles 270°	0.05 (2/4) (0.01 0.08)	(0.02 (1/8)	1
	Sr-89 40	0.002	<LLD	--	--	<LLD	0
	Sr-90 40	0.0003	<LLD	--	--	<LLD	0
Fish (pCi/g) dry weight	Gamma Spec 8						
	Cs-137	80	<LLD	--	<LLD	90 (1/4)	0
	Cs-134	80	<LLD	--	<LLD	<LLD	0
	Co-60	80	120 (3/4) (90-200)	River Mile 35 Podunk River	See Column 4	<LLD	0

[a] Nominal Lower Limit of Detection (LLD) as defined in Note a. of Table I.E-1

[b] Mean and range based upon detectable measurements only. Fraction of detectable measurements at specific locations in parens (f).

Note. Example data are provided for illustrative purposes only. Types of data illustrated here are applicable to type of samples required during the first 150 days post shutdown.

ENCLOSURE 5

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Defueled Site Emergency Plan

Big Rock Point
Defueled Emergency Plan

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DEFUELED EMERGENCY PLAN FOR BIG ROCK POINT

1.0 INTRODUCTION

This document describes the Big Rock Point plans for responding to emergencies that may arise at the plant while in a permanently shut down condition. This Plan will supersede the operating plant's emergency plan at and beyond 93 days post-shutdown at zero power with the reactor defueled. At this time, fuel (which presents the largest source term of radioactive material at the plant site) has decayed to a point which precludes any accident consequences requiring protection of the public (reference Big Rock Point Engineering Analysis, EA-BRPDP-CH5-3, Rev 2, "Decommissioning Plan - Time to Raise SFP Temp 65F").

Under defueled conditions, the plant is prohibited from moving fuel back into the reactor vessel. An analysis of the possible design basis events and consequences is presented in Chapter 5 of the Big Rock Point Decommissioning Plan submitted to NRC on February 27, 1995.[1]

This plan acknowledges the reduced likelihood of a radiological emergency in the plant's permanently shut down and defueled condition as well as the reduced accident consequences. The primary purpose of this plan is to outline the actions necessary to safeguard plant personnel and prevent damage to property in the event of an emergency involving any inadvertent releases of radioactive material.

Table 5.1 provides accident classification guidelines and includes criteria for potential fuel pool accidents as well as those involving dry fuel storage at the independent spent fuel storage installation (ISFSI), either during dry cask loading, during transport to the onsite ISFSI or while at the ISFSI. Such a dry fuel accident would have the possibility of occurring only upon implementation of the dry fuel storage option. Likewise, accidents involving fuel in the spent fuel pool have potential for occurring only as long as the spent fuel pool continues to be used for this purpose.

It is intended that this Emergency Plan be all inclusive, and that it not be continuously revised from one phase of decommissioning to another. Rather, the Emergency Implementing Procedures will be modified over time as necessary to reflect current facility status, provided such changes remain within the overall scope of this plan and will not lessen its effectiveness.

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1. Letter, Robert A. Fenech, Consumers Power Company to Document Control Desk, NRC, February 27, 1995.

The objectives of the Big Rock Point Plant Defueled Emergency Plan, with its Implementing Procedures, are to provide:

- Guidelines to define potential types of emergencies
- Methods for responding to an emergency while the plant is in a permanently shut down and defueled condition
- Descriptions of facilities and equipment used to mitigate accident consequences
- Communications to support emergency response activities
- An organization to manage emergency response activities
- Methods for maintaining emergency preparedness

2.0 DEFINITIONS

Accountability - The process of identifying whether any onsite personnel are missing, and to determine who is on site to assist with emergency response.

Alert - The emergency classification level of an event which indicates an actual or potential substantial degradation in the level of safety to plant personnel or to the safe containment of fuel.

Assembly Areas - Specific locations designated for the assembly and accountability of personnel.

Assessment Actions - Measures taken to define the emergency situation and provide a basis for decisions on specific responses.

Certified Fuel Handler (CFH) - Individual who is trained per the Station's Administrative procedures to respond to events involving fuel stored in the spent fuel pool.

Control Operator - Individual who is responsible for the plant's operation activities when a Shift Supervisor is not on site. This individual is not NRC licensed but will be qualified as a Certified Fuel Handler.

Corrective Actions - Emergency measures taken to mitigate or terminate an emergency situation.

DAC - Derived air concentration.

Defueled Emergency Implementing Procedures - The procedures implemented to direct response to an emergency situation at the plant while in a permanently shutdown and defueled condition.

Drill - A supervised instruction period designed to develop and maintain skills in a particular operation.

EC - Liquid effluent concentration.

Emergency Actions - A collective term encompassing any or all of the assessment, corrective and protective actions taken during the course of an emergency.

Emergency Action Level - Plant-specific system or effluent parameter values characteristic of off-normal conditions which, if exceeded, will initiate emergency classification. See also Unusual Event and Alert classification definitions.

Emergency Response Organization (ERO) - The team of individuals assigned emergency response positions as defined in this plan.

Emergency Support Center - The on-site area which is activated during an emergency for the purpose of providing a location where emergency staff can assemble and logistic support can be coordinated.

Exercise - An event which tests the integrated capability and major portions of the basic elements existing within the emergency plan and emergency organization.

Independent Spent Fuel Storage Installation (ISFSI) - An onsite facility for storage of fuel in dry storage casks.

Industrial Area - The plant area within which provisions are established for responding to postulated emergencies. This area is within a chain link fence which is either locked or otherwise controlled. The Industrial Area is shown in Figure 4-1.

Nonessential Personnel - Any individuals present on the plant site during an emergency who are not required for emergency response activities.

Off-Site - Any area outside the owner controlled area site boundary.

On-Site - Any area inside the site boundary.

Owner-Controlled - The area between the Industrial Area and the site boundary.

Projected Dose - The dose which site personnel may potentially receive from an incident causing abnormally high dose rates or abnormal release of radioactive materials.

Protective Actions - Emergency response measures taken to prevent or minimize radiological exposure or industrial hazards to on-site individuals.

Protective Action Guides (PAG's) - Environmental Protection Agency (EPA) established guidelines for radiological dose rates or dose commitments to individuals in the general population who may be subject to protective actions should a large release of radioactive materials be imminent.

Radiological Assessor - The individual responsible for radiation protection, chemistry and dose assessment activities.

Security Threat Event - Any threat or actual condition that may compromise the effectiveness of security protection at the plant. These events are handled in accordance with the Big Rock Point Security Plan.

Shift Supervisor - Person in charge of plant operations activities. This individual is not NRC licensed but will be a qualified Certified Fuel Handler.

Site Emergency Director (SED) - The individual responsible for all emergency response functions defined by this plan.

Spent Fuel Pool - The area within the reactor building containment structure where spent fuel may be stored.

Technical Coordinator - The individual responsible for accident assessment and repair of plant equipment and systems.

Unusual Event - The emergency classification level that signifies events are in progress or have occurred which indicate a potential degradation of the level of safety at the plant.

3.0 SUMMARY OF DEFUELED EMERGENCY PLAN

This Emergency Plan describes the actions to be taken in response to an emergency condition at the Big Rock Point Plant while in a permanently shut down and defueled condition at and beyond 93 days post-shutdown. The plan addresses the following:

- Emergency classification
- Emergency facilities and equipment
- Communications during emergencies
- Assignments and responsibilities of the Emergency Response Organization
- Emergency response:
    - Corrective Actions
    - Protective Actions
    - Medical Response
    - Restoration of the plant
- Accident assessment
- Notifications
- Maintaining preparedness:
    - Training, drills and exercises
    - Maintenance of the plan and its implementing procedures

Emergencies at the Big Rock Point Plant may be classified as Unusual Event or Alert. If an emergency condition develops, the Shift Supervisor/Control Operator is responsible for classifying the event and assuming the role of the Site Emergency Director until relieved by a designated member of plant management.

The on-shift organization is responsible for performing initial response activities. Notification is made to the NRC, Michigan State Police, key plant personnel, and Consumers Energy.

The on-shift organization may be augmented by additional emergency response personnel at the discretion of the Site Emergency Director. Conditions are assessed and corrective actions are implemented to restore the facility to a stable condition. If necessary, protective actions, including accountability and on-site evacuation of non-essential personnel and other actions such as activation of the Emergency Support Center (ESC), or obtaining support from external (contract or non-plant corporate) entities, will be implemented at the discretion of the Site Emergency Director.

Communications and other equipment are available to support the response effort.

4.0 SITE AREA

4.1 THE SITE

The Big Rock Point Plant is located on the northeast shore of Lake Michigan in Charlevoix County in the northern part of Michigan's Lower Peninsula. The site is approximately 60 miles northeast of Traverse City, Michigan and 225 miles northwest of Detroit. The closest population centers are the cities of Charlevoix, 3.5 miles southwest, and Petoskey, 11 miles east of the plant site. The Big Rock Point Plant is owned by Consumers Energy. The plant's industrial area is approximately seven acres in area and is surrounded by approximately 563 acres of owner controlled area. US Route 31 bounds the owner controlled area to the east and southeast, and provides access to the plant site. Figure 4.1 illustrates the site configuration.

4.2 AREA CHARACTERISTICS AND LAND USE

The area immediately surrounding the site is wooded and gently sloping. There are no significant topographic features near the plant site. A small stream, Susan Creek, exists to the east of the site and drains into Lake Michigan east of the owner-controlled area boundary. There are no residences within one-half mile of the Industrial Area. A plastics manufacturer, employing approximately 130 people, is located just beyond Susan Creek, approximately 0.6 mile from the Big Rock Point Industrial Area.

Temperatures at the plant site are moderated by the lakeshore location, but winds are somewhat higher from the north through southwest lakeward directions than at inland sites. Soils are rocky and highly compacted as a result of geologically recent glaciation. Consequently, farming is not a major activity in the surrounding area.

The Michigan counties in the area of the site are generally rural to suburban. The area is highly tourist-oriented, making seasonal population fluctuations in the vicinity of the site relatively large. Peak seasonal visitation occurs in the summer months (June through August), but additional influx occurs during the winter sports season (November into March).

US Route 31 provides direct highway access to the plant via the site access road which meets the highway one-half mile ESE from the plant location. The access road is a private road which is maintained by, and under the control of Big Rock Point Plant. In emergency situations, the access road may be closed or otherwise controlled at the discretion of the Site Emergency Director.

4.3 INDUSTRIAL AREA

In the permanently shutdown and defueled condition, at or beyond 93 days pc .-shutdown for which this plan is implemented, potential exposures of the offsite public do not exceed EPA Protective Action Guides.

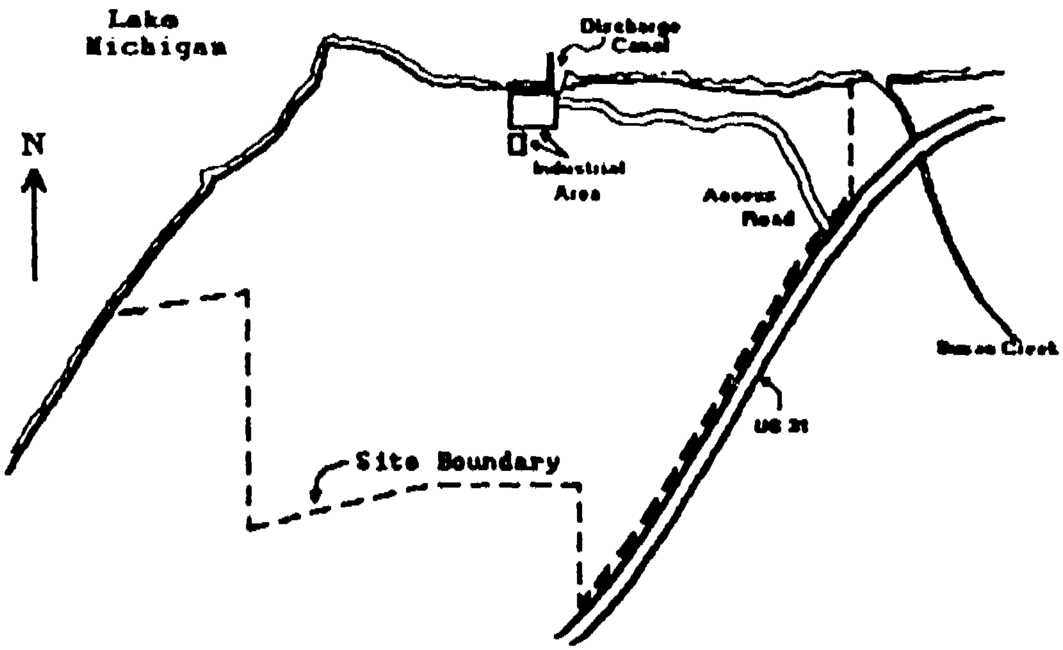
The Industrial Area, see Figure 4.1, is an access controlled area within which radiological exposure due to a radiological emergency could potentially be of concern, but beyond which a design basis accident could not lead to the release of radioactive materials in quantities which would exceed the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). The Radwaste Building is included in the Industrial Area.

Within the Industrial Area, implementation of protective actions for onsite personnel may be necessary, and it is primarily for the protection of these personnel that this Plan is intended.

4.4 OWNER CONTROLLED AREA

The Owner Controlled \rea is the area from the Industrial Area to the site boundary.

Figure 4.1 - Industrial Area Location



Industrial Area Location within Owner-Controlled Property Boundary at Big Rock Point

5.0 EMERGENCY CONDITIONS

This Plan provides for an emergency classification system based on NUREG-0654/FEMA REP 1, Revision 1, Appendix 1, "Emergency Action Level Guidelines for Nuclear Power Plants," NUMARC/NESP-007, Revision 2, "Methodology for Development of Emergency Action Levels," and the radiological consequences from a fuel handling accident.

Emergency conditions at the plant could result in the declaration of an Unusual Event or Alert classification.

5.1 UNUSUAL EVENT CLASSIFICATION

The Unusual Event classification signifies that events are in progress or have occurred which indicate a potential degradation of the level of safety at the plant. Events within this classification generally characterize abnormal plant conditions which alone do not constitute a hazard to plant personnel.

The purpose of an Unusual Event classification is to bring the on-shift staff to a state of readiness and to provide a systematic means of handling information and decision making.

5.2 ALERT CLASSIFICATION

The Alert classification signifies that events are in progress or have occurred which indicate an actual or potential substantial degradation of the level of safety to plant personnel or to the safe containment of fuel. Any release of radioactive material is expected to be below the EPA Protective Action Guideline exposure limits for the public.

The purpose of an Alert is to provide emergency personnel staffing to assist the on-shift emergency organization in radiological or technical assessment and the implementation of corrective actions.

5.3 EMERGENCY ACTION LEVELS

Emergency classification is based on specific information presented in tabular listings of initiating events. The level of severity of an event, or action level, is used to determine which emergency classification is appropriate. The Site Emergency Director is responsible for ensuring the evaluation of the Emergency Action Levels and declaring an emergency if the classification criteria are met.

A list of initiating conditions and the classification levels associated with possible incidents at the Big Rock Point Plant are shown in Table 5-1.

Upon declaration of one of the two emergency classifications, the Site Emergency Director is directed to a Defueled Emergency Plan Implementing Procedure (DEPIP) which details the steps to be taken for that particular emergency classification.

TABLE 5.1 Emergency Action Levels

No	Initiator	Unusual Event	Alert
1	Radioactive Releases or Abnormal Radiation Levels	• Unplanned, uncontrolled effluent release to the environment greater than 20 DAC or 20 EC. • Damage to irradiated fuel with the release of radioactivity to the Containment. • The unintentional manipulation of irradiated components resulting in HIGH or VERY HIGH radiation levels per 10CFR20.1003.	Radiation levels indicate a severe degradation in the control of radioactive materials.
2	Spent Fuel Pool Events	• Uncontrolled drop in Spent Fuel Pool water level. • Loss of any system necessary to safely store fuel. • Damage to irradiated fuel with release of radioactivity to Containment.	Tech Spec temperature limits for the spent fuel pool are likely to be exceeded.
3	Dry Fuel Storage Event	• Accident in loading, transport to ISFSI or at ISFSI with potential for internal fuel damage, but with no release to the environment.	Loss of fission products to the environment
4	Security Compromise	• Communications with security has confirmed the seriousness or credibility of a security threat event.	Severe security threat event.
5	Environ-mental	• Events experienced or projected with potential for affecting fuel integrity or on-site personnel.	Severe environmental hazards to spent fuel or workers.
6	General Events	• Plant conditions that warrant increased awareness on the part of plant staff.	Conditions exist which in the judgement of the SED warrant declaration of an Alert.

6.0 EMERGENCY FACILITIES AND EQUIPMENT

This section describes emergency response facilities, equipment and services which are utilized to ensure a prompt and effective response to an emergency. Communications systems available within the facilities are outlined in Section 7.0 of this Plan.

6.1 EMERGENCY RESPONSE FACILITY

6.1.1 Emergency Support Center

The Emergency Support Center (ESC) is the location from which the emergency is managed. The Site Emergency Director directs all phases of the emergency response, including the notification of plant emergency response personnel, State of Michigan, the NRC, and the performance of corrective actions to mitigate the effects of the emergency. The ESC will serve as the plant communications center as well as assist Operations personnel in handling administrative items, provide technical evaluations and secure logistical support.

The ESC may be activated at the discretion of the Site Emergency Director. The ESC is deactivated by the Site Emergency Director when plant conditions have stabilized to the point that continuous emergency support is no longer required.

6.1.2 Assembly Areas

The plant first floor office area is the location where nonessential personnel, contractors and visitors will relocate and assemble in the event accountability is necessary. Personnel outside the industrial area will assemble in East Office Building Room 113 and the adjacent hallway.

6.1.3 Decontamination Facility

The plant maintains personnel showers for decontamination and survey instrumentation for personnel frisking.

6.1.4 First Aid and Medical

First Aid kits, stretchers and fire blankets are located by the containment personnel lock and in the maintenance building (confined spaces cabinet) and at the entrance to the security offices at the plant main gate. Additional first aid kits are present at the radiological control point, ESC and at each assembly area.

Agreements are in place with local hospitals to provide offsite emergency medical treatment and with local agencies to provides ambulance services. Letters of agreement are provided in the Defueled Emergency Plan Implementing Procedures.

6.2 INDUSTRIAL AREA EQUIPMENT

Dedicated emergency equipment includes respiratory devices, anti-contamination garments and radiation monitoring and sampling equipment. This equipment is described in detail and maintained operational in accordance with plant emergency procedures.

6.3 ASSESSMENT SYSTEMS

The assessment and monitoring equipment described in this section ensures that plant personnel may acquire necessary data and monitor trends for recognition of off-normal conditions.

6.3.1 Radiological Monitoring Systems

The Area Radiation Monitoring System consists of detectors throughout the plant. The installed system may be supplemented with or replaced by portable units as may be required during the conduct of various decommissioning activities. These instruments may be used to assess accident conditions and to determine an area's habitability under accident conditions.

The effluent process monitoring system provides indication of gross radioactivity levels of all airborne and liquid effluents released from the plant via the liquid and gaseous radwaste systems and the plant ventilation systems.

6.3.2 Meteorological Monitoring

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

6.3.3 Spent Fuel Pool Monitoring

Fuel pool water level indication is provided. In addition, the pool utilizes a tell-tale liner leak detection system which is checked for leakage on a periodic basis. Radiation levels, which would rise upon partial loss of water shielding, damage to fuel or withdrawal of any highly radioactive component from the pool, are monitored by containment building radiation monitors near the pool.

7.0 COMMUNICATIONS

This section describes the communications systems available for effective command and control during a plant emergency.

7.1 PLANT PAGING SYSTEM

The plant paging system is located throughout the plant and is utilized as a paging and intercom system under normal and accident conditions. In an emergency this system is used (1) as the means of notifying onsite personnel of an emergency; (2) for provision of specific instructions for emergency response and (3) to provide updates of plant status as necessary.

7.2 PLANT TWO-WAY RADIO SYSTEM

The plant two-way radio system using handheld radios allows for communications among support teams involved in onsite emergency response activities. A plant base station allows radio communications beyond the range of the handheld radios.

7.3 TELEPHONES

An Auto-Dial telephone is present in the ESC for notifications to the Michigan State Police or to contact emergency agencies (hospital, ambulance, or fire department). An auto-dial feature also is available on the ESC computer for access to current and forecast weather information. Both intraplant and standard telephone circuits are present at the plant for use in emergencies. In the event of a telephone outage, a base radio is present at Security to contact the State Police. Power failure phones are provided to maintain telephone communication in the event of power loss.

7.4 NRC EMERGENCY NOTIFICATION SYSTEM (ENS)

In the event of an emergency, Big Rock Point will notify the NRC via dedicated phone lines to the NRC Headquarters Operations Center.

8.0 ORGANIZATION

This section of the Plan defines the Big Rock Point emergency response organization (ERO). Assignments and responsibilities of the ERO are addressed.

8.1 NORMAL PLANT ORGANIZATION

The Consumers Energy Senior Vice President, Nuclear, Fossil and Hydro Operations will be the corporate officer responsible for Big Rock Point nuclear safety. This individual may take any measures necessary to ensure acceptable performance in operating, maintaining and preparing the plant for decommissioning so that continued nuclear safety is ensured.

The onsite organization will be lead by the Plant Manager who is supported by individuals with responsibilities for administrative services, maintenance, operations, engineering, training, health physics and chemistry. Minimum backshift coverage will consist of one Shift Supervisor/Control Operator, one individual qualified in radiological protection and one member of the Plant Security Force.

8.2 EMERGENCY RESPONSE ORGANIZATION (ERO)

Emergency Response Organization positions are assigned to qualified plant personnel. Qualification of personnel will be in accordance with the training requirements described in Section 12.1.

8.2.1 On-Shift Response Organization

The ERO may be activated during normal or off-normal working hours. At all times, the Shift Supervisor/Control Operator acts as initial Site Emergency Director and performs initial response actions. The Site Emergency Director will augment additional plant staff at his discretion. The ERO with full augmentation is described by Figure 8.1.

The Site Emergency Director is responsible for implementing initial actions to bring the facility to a stable condition and to make initial notifications to NRC, State of Michigan and Consumers Energy Company representatives. The Site Emergency Director is also responsible for notification of key personnel as necessary for augmentation, fire, ambulance or other emergency responders as conditions may require.

Security is responsible for maintaining plant security in accordance with the Big Rock Point Security Plan. Security responds to threats to physical security, performs accountability, assists in the evacuation of on-site personnel and other duties as directed by the Site Emergency Director. Plant Security also provides Fire Brigade members and first aid support as incidents may require.

8.2.2 Augmented Response Organization

Additional personnel may be called in at the discretion of the Site Emergency Director to augment on-shift personnel.

The augmented emergency response organization (ERO) is activated in part or in its entirety depending on severity of the situation. This organization consists of the Radiological Assessor, Technical Coordinator, and a Security Officer, each of which are supported by other members of the normal plant staff, under the direction of the Site Emergency Director.

The Radiological Assessor is responsible for radiation protection, chemistry and dose assessment activities. The Radiological Assessor also advises the Site Emergency Director regarding on-site protective actions.

The Technical Coordinator is responsible for accident assessment and repair of plant equipment and systems. The Technical Coordinator directs the technical and operational evaluation of the incident and implements corrective actions necessary to recover from the accident conditions.

8.3 LOCAL SUPPORT

Arrangements have been made with local organizations to provide:

1) Ambulance service to transport radiologically contaminated injured personnel.

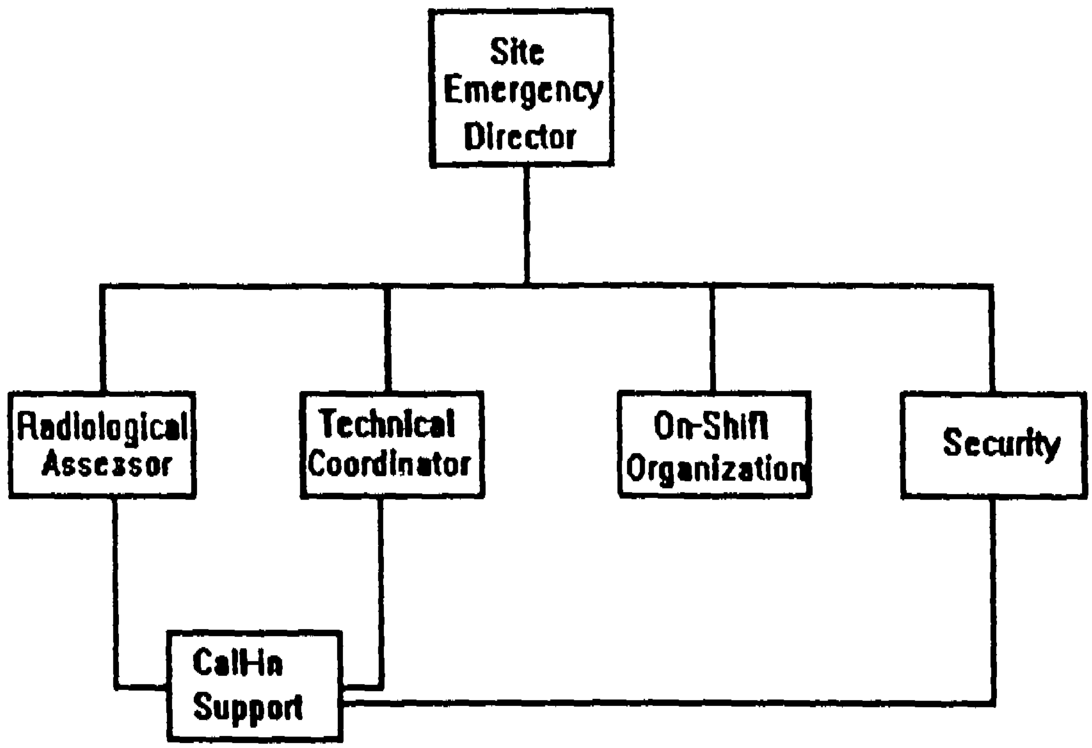
2) Hospital services for radioactively contaminated injured personnel.

3) Fire support services.

Letters of Agreement with appropriate off-site support organizations are listed in the Defueled Emergency Plan Implementing Procedures.

EMERGENCY ORGANIZATION

Figure 8.1 Augmented Response Organization



9.0 EMERGENCY RESPONSE

This section identifies the measures that would be taken for each class of emergency. These measures are described in the Defueled Emergency Plan Implementing Procedures (DEPIPs).

9.1 EMERGENCY CLASSIFICATION

Recognition and classification of the incident is the responsibility of the Shift Supervisor/Control Operator who at that time takes on the duties of Site Emergency Director (SED). Depending on the specific Emergency Action Level (EAL) reached, the SED declares an Unusual Event or Alert. Once the emergency classification is declared, the appropriate implementing procedures are utilized.

9.2 ACTIVATING THE EMERGENCY RESPONSE ORGANIZATION (ERO)

Conditions which may occur to cause declaration of an Unusual Event or Alert are defined in Section 5.0. Upon classification and declaration of an emergency, the Shift Supervisor/Control Operator assumes the position of Site Emergency Director and remains in charge of the ERO until such time as a qualified member of plant management relieves him of that position.

A given incident may not warrant activation of the entire ERO. Additional personnel may be mobilized at the discretion of the Site Emergency Director to augment the on-shift organization and assist in the implementation of corrective actions to mitigate the consequences of the event. Specific response actions for each classification are described in the following subsections.

9.2.1 Unusual Event Response

The following is a general summary of the actions taken in response to an Unusual Event:

1) Shift Supervisor/Control Operator classifies the event and assumes the title and duties of Site Emergency Director (SED) until relieved by another qualified plant staff member.

2) As conditions dictate, fire, ambulance or law enforcement agencies may be requested to respond prior to mandatory notifications.

3) Emergency classification is announced over plant paging system.

4) On-shift personnel respond as directed by the SED.

5) Key plant and corporate management are notified.

6) NRC is notified as soon as possible but within one hour of declaration.

7) State of Michigan is notified as soon as possible but within one hour of declaration.

8) Corrective actions are implemented.

9) Public information is handled by the Public Affairs Director or an individual designated by the SED.

10) The SED will escalate or terminate the event as appropriate.

9.2.2 Alert Response

During an Alert, the actions identified in Section 9.2.1 are performed. Additional actions may be implemented at the discretion of the Site Emergency Director, depending upon characteristics of the incident. Examples of additional actions which could be implemented include:

1) Activation of the Emergency Support Center.

2) Assembly and accountability check of plant personnel, contractors and visitors.

3) Evacuation from the industrial area of nonessential plant personnel, contractors and visitors.

4) Performance of on-site radiological monitoring and assessments.

9.3 EVENT TERMINATION AND RECOVERY/RE-ENTRY

At such time as the plant has been restored to a stable and safe condition, the emergency classification condition may be terminated. The Site Emergency Director will terminate the event and provide notification to appropriate off-site authorities and plant staff personnel. Development and implementation of any on-site recovery/re-entry actions which may be necessary are the responsibility of the SED.

9.4 FIRST AID AND MEDICAL RESPONSE

First aid and medical supplies are available at the Big Rock Point Plant as described in Section 6.1.4. Specific plant personnel are trained in the use of this equipment. In the event of serious personnel injury requiring outside support, the Site Emergency Director is responsible for assuring that ambulance and hospital services are requested.

10.0 RADIOLOGICAL ASSESSMENT

Radiological assessment is initiated for those events which provide real or threatened radiological exposure of plant personnel or release of radioactivity to the environment. The systems and methods for monitoring and assessing the actual or potential consequences of a radiological emergency are described in this section.

10.1 ASSESSMENT INSTRUMENTATION

Accurate radiological assessment of an emergency is the key to appropriate radiological response. Fixed radiological monitoring instruments consist of area and process monitors located in key areas of the plant. Area Radiation Monitors (ARMs) measure the ambient radiation level in each monitored location and are used for personnel radiation protection purposes. The ARMs also serve to alert plant staff to potential hazards from radioactive material within the plant structure. Process Radiation Monitors (PRMs) measure radiation levels in effluent systems, including releases to the environment. The PRMs can detect and quantify unplanned airborne or liquid effluent releases. Meteorological sensors measure the wind speed and direction. Complete weather information, including atmospheric stability class and forecast data for Big Rock Point are available on demand via telephone or by autodial through the ESC computer. Plant instrumentation or their equivalent provided for radiological assessment includes:

- Effluent Monitor
- Liquid Radwaste Monitor
- Fuel Pool Area Monitors
- General Area Radiation Monitors

The above radiological assessment instrumentation will be removed when no longer needed. In addition, some may be replaced with portable instrumentation as appropriate.

10.2 ASSESSMENT METHODS

Radiological release assessments are performed using measured radiological and meteorological data. Dose assessment graphs are initially used by the Site Emergency Director or other trained emergency response personnel to determine the industrial area dose rate.

Additional assessments of potential radiation dose to plant personnel from direct radiation or potential exposure from various other sources will be performed as appropriate. Dose conversion factors from the EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 400) will be used where such calculations are required.

10.3 RADIOLOGICAL ENVIRONMENTAL MONITORING

Radiological surveys are performed by on-site personnel to confirm the magnitude of a release of radioactive material and/or dose rates due to direct radiation from the plant, as appropriate to the accident circumstances. Radiological monitoring personnel will be dispatched to perform industrial area surveys and other surveys as necessary. Monitoring personnel are briefed and equipped with radiation survey instruments, air sampling equipment, respiratory protection and/or protective clothing, as required.

Radiological monitoring per   :l take direct radiation measurements and air samples as directed. Level and type of monitorin   ll depend on the severity and type of accident. The Radiological Assessor determines the level and type of monitoring required.

Environmental monitoring and sample collections, if determined to be warranted, are carried out under procedures utilized for the plant environmental monitoring program as specified in the Big Rock Point Offsite Dose Calculation Manual (ODCM).

10.4 RADIOLOGICAL EXPOSURE CONTROL

The Radiological Assessor is responsible for in-plant emergency radiological protection activities for plant staff and support personnel. Before arrival of the Radiological Assessor, such functions are the responsibility of the Site Emergency Director. Dose limits for individuals performing emergency functions are presented in Table 10.1. Every attempt will be made to keep personnel exposure As Low As is Reasonably Achievable (ALARA).

Dosimetry will be issued if necessary, depending upon characteristics of the emergency and assignments of personnel. Planned rescue or corrective tions which could cause an individual's dose to exceed 5 rem in a year, within the limits of Table 10.1, will require authorization by the Site Emergency Director. Individuals authorized under these limits will be briefed of the hazards associated with the planned actions prior to undertaking the mission.

10.5 PROTECTIVE MEASURES

10.5.1 Industrial Area Accountability/Evacuation

The Site Emergency Director shall initiate an accountability of all Industrial Area personnel at the Alert classification. Accountability will be initiated by use of the emergency siren, and/or paging system, as appropriate for the incident at hand. All personnel, contractors and visitors will relocate to on-site assembly areas. Accountability for Industrial Area personnel will be accomplished within 90 minutes. Search and rescue efforts will . initiated for any unaccounted for personnel.

Accountability of personnel will be the responsibility of Security. All reports are provided to the Site Emergency Director.

In the event a plant evacuation of selected groups is judged appropriate, the Site Emergency Director will announce appropriate instructions on the paging system. Emergency response personnel remaining on-site or arriving following an evacuation shall report to the Emergency Support Center. Security will prohibit access of other individuals to the Industrial Area unless specifically authorized by the Site Emergency Director.

10.5.2 Decontamination Capabilities

Facilities at the radiological control point will be utilized for personnel contamination monitoring and decontamination. Decontamination will be in accordance with normal plant procedures.

In the event that accident conditions result in the injury of a contaminated individual, decontamination will be performed to the extent that the health of the victim is not jeopardized. Such decontamination normally would be performed while awaiting ambulance arrival. Further decontamination, including wound decontamination, will be provided upon arrival at the hospital.

10.6 PROTECTIVE EQUIPMENT AND SUPPLIES

Plant radiation protection equipment will be utilized as necessary to support the emergency response effort. Equipment such as respirators and protective clothing will be issued per the emergency radiation protection criteria of Table 10.3 and normal plant radiation protection procedures.

TABLE 10.1

Emergency Exposure Criteria

Emergency exposures in excess of 5 rem Total Effective Dose Equivalent (TEDE) for the calendar year may be authorized by the Site Emergency Director using the criteria below:

Activity	Dose Limit
Important activities not required to save life or prevent added damage	12.5 rem TEDE 125 rem extremity or skin
Necessary to stabilize conditions and avoid further damage	25 rem TEDE 250 rem extremity or skin
Necessary to save life	75 rem TEDE 750 rem extremity or skin

In each of the above cases, doses shall be maintained As Low As is Reasonably Achievable (ALARA). All approvals shall be for pre-planned activities, with approval prior to initiating the effort.

TABLE 10.2

Considerations for Use of Emergency Exposure Limits

- Rescue personnel should be nonpregnant adults or professional rescue personnel.
- Volunteers are aware of potential consequences of exposure at these levels.
- Volunteers agree to accept risk.
- Volunteers understand nature of the task and method of performance.
- Plans incorporate appropriate ALARA actions to minimize dose.
- Expected dose has been weighed against probability of success and risk of inaction.
- Volunteers are proficient in performance of the task and capable of its rapid execution.

TABLE 10.3

Emergency In-Plant Protective Action Criteria

1. If unknown mixture particulate airborne activity concentrations* exceed 3E-8 uCi/ml (10 x DAC for unidentified particulate), use of respirators will be determined based upon expected duration of exposure and characterization of particulate activity, if possible.

2. If unknown mixture particulate concentrations* exceed 3E-7 uCi/ml, use of SCBA respirators in pressure demand mode will be required.

3. If unknown mixture particulate concentrations* exceed 3E-3 uCi/ml for more than two hours (>0.1 ALI in Pressure Demand mode SCBA respirators), evacuation of the affected area will be initiated.

4. If Kr-85 (gaseous airborne activity) exceeds 1E-2 uCi/ml, skin protection in the form of one cloth or paper anti-C, covered by one waterproof plastic anti-C will be required, in addition to SCBA Pressure Demand respiratory protection.

5. Decontamination of evacuating personnel will be performed unless background radiation levels, personnel injuries or other emergency considerations preclude this action.

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*Note: The above actions for airborne concentrations of unknown particulate activity will be modified upon identification of actual constituents so as to ensure that the dose criteria of Table 10.1 (TEDE) are not exceeded upon summation of internal and external dose.

11.0 EMERGENCY NOTIFICATION AND PUBLIC INFORMATION

This section of the plan identifies the means of performing external and internal notifications during an emergency. These notifications include State of Michigan and NRC contacts, plant and corporate contacts, and information supplied to the general public through public news media.

11.1 EMERGENCY NOTIFICATIONS

11.1.1 On-Site Notifications

The Site Emergency Director will notify plant personnel of the emergency via the plant paging system during normal working hours. Pagers also may be activated.

During off-hours, the Site Emergency Director will announce the emergency condition via the siren or plant paging system (as appropriate for the incident), pagers and/or by telephone, as necessary.

11.1.2 Off-Site Notification

Upon declaration of an Unusual Event or Alert, the Site Emergency Director will notify the NRC and the State of Michigan bv telephone as soon as possible but within one hour of classifying the emergency.

Consumers Energy corporate personnel will be notified by the Site Emergency Director regarding the plant status. Palisades Nuclear Plant may be contacted on an as-needed basis for additional emergency support assistance (eg, engineering, maintenance or technician support).

11.2 PUBLIC INFORMATION

During an emergency event declaration, the Public Affairs Director or an individual assigned by the Site Emergency Director will be responsible for handling public information associated with an emergency at Big Rock Point. News releases will be prepared and issued as required.

12.0 MAINTAINING EMERGENCY PREPAREDNESS

This section of the Site Emergency Plan describes the manner in which the emergency preparedness program is maintained in a state of readiness.

12.1 TRAINING

12.1.1 Training of Plant Personnel

All plant personnel assigned emergency responsibilities will receive annual (± 25%) training in accordance with plant procedures. Training for these individuals may be obtained through drill and exercise participation or other alternative training.

12.1.2 Training of Offsite Support Personnel

Offsite support personnel who may be called upon for emergency support include fire fighting, hospital, ambulance and law enforcement personnel. Medical and fire fighting groups will be invited to participate in annual training in those areas identified within Table 12.1. Local law enforcement will be invited to participate in an annual [illegible] on plant status.

12.2 DRILLS AND EXERCISES

Periodic exercises and drills are conducted to evaluate emergency response capabilities and to develop and maintain key skills. Some drills may be included as a portion of the radiological emergency preparedness exercise.

Drills are supervised instruction periods designed to develop and maintain skills in a particular operation.

Exercises are events which test the integrated capability and major portions of the basic elements existing within the emergency plan and emergency organization.

EMERGENCY RESPONSE TRAINING FOR OFF-SITE GROUPS
TABLE 12.1

CATEGORY	EMERGENCY POSITIONS	TRAINING	FREQUENCY
Fire Fighting	Local fire response personnel	Training will be offered to fire fighting personnel in plant layout and fire hazards, basic radiation protection needed for fire fighting, and fire system orientation	Annually
Law Enforcement	Local law enforcement personnel	Local law enforcement agencies will be offered review sessions on plant status.	Annually
Medical Support	Hospital and Ambulance personnel	Training on the handling of radioactively contaminated victims, communications, and plant/ambulance/hospital interface will be provided.	Annually
		Personnel will be offered the opportunity to participate in medical emergency drills performed by the plant.	As Performed

12.2.1 Drills

a. Communication Drills

Communication drills with the NRC, the Michigan State Police and offsite support agencies will be performed quarterly.

b. Emergency Medical Drills

A medical emergency drill involving a simulated contaminated individual with provisions for participation by the local support services agencies (ie, ambulance and off-site medical treatment facility) will be conducted annually.

c. Radiological Monitoring Drills

A plant radiological monitoring drill to evaluate the response to emergency radiation protection situations will be conducted annually.

12.2.2 Exercises

A radiological emergency preparedness exercise will be conducted biennially. The exercise will include the mobilization of utility personnel to verify the capability to effectively respond to an accident. Performance will be evaluated and critiqued in order to judge the effectiveness of emergency planning elements.

12.3 MAINTENANCE AND INVENTORY OF EMERGENCY EQUIPMENT AND SUPPLIES

A detailed list of emergency equipment and its locations may be found in the Emergency Implementing Procedures. Emergency kits and equipment will be inventoried and maintained on a regular basis, as defined by those procedures. Radiation detection equipment will be calibrated as required.

12.4 REVIEWS AND UPDATES OF PLAN AND PROCEDURES

The Defueled Emergency Plan (DEP) and Defueled Emergency Plan Implementing Procedures, including appended Letters of Agreement, are reviewed every two years.

Necessary changes to the Defueled Emergency Plan or DEP Implementing Procedures will be performed in accordance with approved plant procedures. The documented reviews and records of changes shall be maintained for a minimum of five years.

12.5 EMERGENCY PLAN AUDIT

Audits of this plan and the adequacy of its implementation will be conducted according to CPC-2A. Independent reviews also may be conducted to verify compliance with Consumers Energy's internal policies and procedures, Federal regulations and operating license provisions.

12.6 RESPONSIBILITY

The Big Rock Point Plant Manager is responsible for the Defueled Emergency Plan. The Plant Manager may assign responsibilities for implementing procedures and maintenance, for emergency plan training, for exercise and drill development, or other associated activities, to competent plant staff or contract personnel in accordance with management discretion. Such assignments do not relieve the Plant Manager from responsibility for meeting the requirements of this plan, plant licensing documents or regulatory requirements.

ENCLOSURE 6

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Request for Exemption to 10 CFR 50 Requirements for Emergency Planning

Consumers Energy

Big Rock Point Nuclear Plant
10269 US 31 North
Charlevoix MI 49720

Kenneth P Powers
Site Manager

September 19, 1997

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 LICENSE DPR-06 - BIG ROCK POINT PLANT - REQUEST FOR EXEMPTION FROM 10 CFR 50 REQUIREMENTS FOR EMERGENCY PLANNING

Consumers Energy Company is requesting an exemption (to become effective December 1, 1997) to certain sections of Title 10 of the Code of Federal Regulations, Part 50.47 "Emergency Plans", 10 CFR 50 Appendix E, Paragraph IV, "Content of Emergency Plans", and 10 CFR 50.54 "Conditions of Licenses". These exemptions will allow Consumers Energy Company to effectively manage the Big Rock Point Plant Defueled Emergency Plan. The basis for the requested exemption is submitted as an attachment to this letter. The Big Rock Point Plant Defueled Emergency Plan has been submitted to the Nuclear Regulatory Commission Staff, concurrent with this exemption request.

Kenneth P. Powers

Kenneth P Powers
General Plant Manager

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point Plant

ATTACHMENT

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ATTACHMENT

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

REQUEST FOR EXEMPTIONS FROM 10 CFR 50
REQUIREMENTS FOR EMERGENCY PLANNING

assemblies, with an average burnup of 25 GWD/MT. To reiterate, this difference leads to a reduced source term for Big Rock Point, which is only on the order of 10% that of the representaive BWR design. As such, Big Rock Point Plant provides lower impacts for potential radiological accidents.

Therefore, based on the licensee's and the Commission's conclusions presented above, and the intuitive differences between the representative BWR and the Big Rock Point Plant, the Big Rock Point staff concludes that the analysis presented for fuel handling accidents bounds the considerations of the August 1997 NRC report, and that there will be no offsite consequences at the facility by November 5, 1997. In addition, the onsite and oversight exemptions are requested, in part, on the basis that the Big Rock Point Plant Defueled Emergency Plan does indeed reflect the substantially reduced risk associated with the plant's permanently shutdown and defueled condition and provides for the organization and actions necessary to respond to an emergency.

Regulatory Bases for Requested Exemptions

The provisions of 10 CFR 50.12 allow specific exemptions from the requirements of 10 CFR 50 provided the exemptions are authorized by law, are consistent with the common defense and security, are accomplished by special circumstances, and do not present an undue risk to the public health and safety. Consumers Energy Company concludes that the activities sought to be conducted under this exemption request are clearly authorized by law, are consistent with the common defense and security, and do not present an undue risk to the public health and safety. Consumers Energy Company believes special circumstances exist with regard to offsite and onsite emergency planning requirements as described below.

> ***10 CFR 50.12(a)(2)(ii) states, "Application of the regulation in the particular circumstances would not serve the underlying purpose in the rule or is not necessary to achieve the underlying purpose of the rule;..."***

The degree of emergency planning and preparedness necessary to provide adequate protection of the public health and safety in a permanently shutdown and defueled condition is significantly less than provided for in the existing Radiological Emergency Response Plan that was prepared to meet the requirements of 10 CFR 50.47 and 10 CFR 50.54. Consumers Energy Company has conducted a hazards analysis in support of the Big Rock Point Plant Defueled Emergency Plan and has concluded that there are no design basis or other credible events that would result in doses beyond the exclusion area boundary that would exceed the Environmental Protection Agency Protective Action Guides following 68 days post shutdown. Therefore, requiring Consumers Energy Company to comply with the full range of emergency preparedness requirements specified in 10 CFR 50.47(b), 10 CFR 50 Appendix E, and 10 CFR 50.54(t) is not necessary to achieve the underlying purpose of the rule. These regulations were established for power operation conditions, as such conditions could result in potential for an accident with offsite radiological dose consequences. The permanently shutdown and defueled condition of the Big Rock Point Plant facility renders the possibility of such accidents no longer credible following 68 days post shutdown. Therefore, the elimination of offsite emergency response capabilities, planning activities, and a reduction in the scope of onsite responses is warranted.

In view of the fact that power operations at Big Rock Point Plant have permanently ceased, and thus the potential risk associated with activities at Big Rock Point Plant have been significantly reduced, maintenance of offsite emergency response capabilities (e.g., facilities, public notification system), offsite planning activities and the current level of onsite response would seem to have a limited safety benefit. Granting the requested exemption will allow Consumers Energy Company to effectively manage the Big Rock Point Plant Defueled Emergency Plan.

Conclusion

On the basis of the information presented in this letter, and the concurrent submittal of the Big Rock Point Plant Defueled Emergency Plan, it is Consumers Energy Company's position that exemption from certain requirements of 10 CFR 50.47(b), 10 CFR 50 Appendix E, and 10 CFR 50.54(t) as specified in attached Tables 1, 2, and 3 is justified.

Potential Significant Adverse Impact on the Environment

The requested exemption from certain requirements of 10 CFR 50.47(b), 10 CFR 50 Appendix E, and 10 CFR 50.54(t) are administrative in nature, and do not involve any significant adverse impact on the environment as the plant transitions from operation into decommissioning. Being that the requested exemption is administrative in nature, the probability and consequences of a radiological accident is not affected, and non-radiological plant effluents are not affected. Therefore, there are no significant adverse impacts on the environment as a result of the requested exemption.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50.47(b)(4)	"...~~*State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), no offsite response is required.
10 CFR 50.47(b)(5)	"...*the content of initial and followup messages to response organizations* ~~*and the public*~~ *has been established;* ~~*and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), no offsite notification of the general public is required.
10 CFR 50.47(b)(6)	"*Provisions exist for prompt communications among principal response organizations to emergency personnel* ~~*and to the public.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), communications with the public will be via news releases. No capability for prompt communication to the public is required.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50.47(b)(7)	"~~*Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), information made available to the public on a periodic basis will no longer be necessary as they will have no response actions. There would be no need to pre-plan or establish procedures with the news media for the dissemination of information.
10 CFR 50.47(b)(9)	*"Adequate methods, systems, and equipment for assessing and monitoring actual or potential ~~offsite~~ consequences of a radiological emergency condition are in use."*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), no methods, systems and equipment will be maintained for determination of offsite dose consequences.
10 CFR 50.47(b)(10)	"~~*A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the EPZs and the associated protective actions are no longer required.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E Section IV	"...~~The nuclear power reactor operating license applicant shall also provide an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), evacuation times and protective actions within the EPZ are no longer required.
10 CFR 50, Appendix E (IV)(A)(8)	"~~Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), identification of State and local authorities responsible for protective actions is no longer required.
10 CFR 50, Appendix E (IV)(B)	*"...what type of protective measures should be considered within ~~and outside~~ the site boundary to protect health and safety. The emergency actions levels shall be based on in-plant conditions and instrumentation in addition to onsite ~~and offsite~~ monitoring. ~~These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis~~."*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), offsite protective measures and monitoring are no longer required.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E (IV)(C)	*"Emergency action levels (based not only on onsite ~~and offsite~~ radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, ~~such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies~~ shall be described."* ~~"...(3) site area emergency, and (4) general emergency."~~	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), offsite radiation monitoring shall not be conducted. Containment pressure sensors and Emergency Core Cooling System are no longer required in the permanently shutdown and defueled condition. Spent fuel pool temperature and level become important indications for potential emergencies during decommissioning. Site area and general emergencies are no longer credible emergency classifications.
10 CFR 50, Appendix E (IV)(D)(1)	*"Administrative and physical means for notifying ~~local~~, State, and Federal officials and agencies and agreements reached with these officials and agencies ~~for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary~~, shall be described. ~~This description shall include identification of the appropriate officials, by title and agency of the State and local government agencies within the EPZs.~~"*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the prompt notification for evacuation or protective measures of the public shall no longer be required. Notification of Federal and State authorities should occur within one hour.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E (IV)(D)(2)	*"~~Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information...would be helpful if an accident occurs.~~"*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), no annual mailing will be conducted. Similarly, no signs or other measures for notification of transient populations shall be required.
10 CFR 50, Appendix E (IV)(D)(3)	*"A licensee shall have the capability to notify responsible State ~~and local governmental~~ agencies within ~~15 minutes~~ after declaring an emergency. ~~The licensee shall demonstrate that the State/local officials have the capability to make a public notification decision...with the appropriate governmental authorities.~~"*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), courtesy notifications to the State will take place within one hour. Similarly, no alert notification system shall be required.
10 CFR 50, Appendix E (IV)(E)(9)(a)	*"~~Provision for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.~~"*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), courtesy notifications to the State will take place within one hour. Similarly, no alert notification system testing shall be required.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E (IV)(E)(9)(b)	"~~*Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), courtesy notifications to the State will take place within one hour. Similarly, no alert notification system testing shall be required.
10 CFR 50, Appendix E (IV)(F)(1)	"*In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services*~~*/Civil Defense, local law enforcement personnel, local news media persons.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the training of local law enforcement and local news media personnel is not required.
10 CFR 50, Appendix E (IV)(F)(2)	"~~*The plan shall describe provisions for*~~...~~*test the public notification system*~~, ~~*and ensure that emergency organization personnel are familiar with their duties*~~."	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the public notification system will no longer be required.
10 CFR 50, Appendix E (IV)(F)(2)(a)	"~~*A full participation exercise which tests as much of the licensee, State and local emergency plans*~~...~~*State or local government participation.*~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the offsite full participation exercise will no longer be required.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E (IV)(F)(2)(c)	" ~~Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), biennial offsite participation exercises will no longer be conducted.
10 CFR 50, Appendix E (IV)(F)(2)(d)	"~~A state should fully participate in the ingestion pathway portion of exercises at least once every six years. In States with more than one site, the State should rotate this participation from site to site.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the offsite full participation exercise, and hence this portion of the exercise, will no longer be required.
10 CFR 50, Appendix E (IV)(F)(2)(e)	"~~Licensees shall enable any State or local Government located within the plume exposure pathway EPZ to participate in the licensee's drills when requested by such State or local Government.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), biennial offsite participation exercises will no longer be conducted.

TABLE 1
REQUESTED OFFSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E (IV)(F)(2)(f)	"~~Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise...not properly tested in the previous exercises.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), biennial offsite participation exercises will no longer be conducted. Hence, remedial exercises for improperly conducted biennial exercises will similarly no longer be required.

TABLE 2
REQUESTED ONSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50.47(b)(3)	*"Arrangements for requesting and effectively using assistance resources have been made, ~~arrangements to accommodate State and local staff at the licensee's near site Emergency Operations Facility have been made,~~ and other organizations capable of augmenting the planned response have been identified."*	As there are no deign basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), Big Rock Point Plant Emergency Operations Facility shall no longer be retained.
10 CFR 50, Appendix E (IV)(A)(2)(c)	*"~~Authorities, responsibilities, and duties on an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.~~"*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), exchange of information with offsite authorities will no longer be required.
10 CFR 50, Appendix E (IV)(A)(3)	*"~~A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization.~~"*	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the level of emergency response required by the Big Rock Point Plant Defueled Emergency Plan would not require response to the plant by headquarters personnel.

TABLE 2
REQUESTED ONSITE EXEMPTIONS

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50, Appendix E (IV)(A)(4)	"~~Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), identification of licensee persons responsible for making offsite dose projections is no longer required.
10 CFR 50, Appendix E (IV)(A)(5)	"~~Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions... special qualifications of these persons shall be described.~~"	As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the Big Rock Point Plant Defueled Emergency Plan does not specify individuals with "special qualifications" for emergency response.

TABLE 2
REQUESTED ONSITE EXEMPTIONS

Regulation	**Requirement**	**Plan Basis for Exemption**
10 CFR 50, Appendix E (IV)(E)(8)	"~~*A licensee onsite technical support center and a licensee near site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency*~~."	The Big Rock Point Plant Defueled Emergency Plan does not require the use of the Emergency Operations Facility (EOF) and Technical Support Center (TSC). Effective direction and control during an emergency shall emanate from the Big Rock Point Plant Emergency Support Center.
10 CFR 50, Appendix E (IV)(E)(9)(c)	"~~*Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.*~~"	Communication with the TSC, EOF, and State and local emergency operations centers are no longer required as these facilities shall no longer be maintained.
10 CFR 50, Appendix E (IV)(E)(9)(d)	"*Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the ~~nuclear power reactor~~ control room, ~~the onsite technical support center, and the near site emergency operations facility~~. Such communications shall be tested ~~monthly.~~*"	As the requested emergency plan does not employ the EOF and TSC, such communication capability is no longer required. The requested defueled emergency plan requires quarterly testing of such communications.

TABLE 3
REQUESTED OVERSIGHT EXEMPTION

Regulation	Requirement	Plan Basis for Exemption
10 CFR 50.54(t)	*"A nuclear power reactor licensee shall provide for the development, revision, implementation, and maintenance of its emergency preparedness program. To this end, the licensee shall provide for a review of its emergency preparedness program at least every 12 months by persons who have no direct responsibility for implementation of the emergency preparedness program. The review shall include an evaluation for adequacy of interfaces with State ~~and local~~ governments ~~and of licensee drills, exercises, capabilities, and procedures.~~ The results of the review, along with recommedations for improvements, shall be documented, reported to the licensee's corporate and plant management, and retained for a period of five years. The part of the review involving the evaluation for adequacy of interface with State ~~and local~~ governments shall be available to the appropriate State ~~and local~~ governments."*	*As there are no design basis or other credible events that would result in doses beyond the site area boundary that would exceed the EPA PAGs following 68 days post shutdown (11/5/97), the reviews evaluating the interfaces will no longer be required.*

ENCLOSURE 7

CONSUMERS ENERGY COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

Training Program for Certified Fuel Handlers

BIG ROCK POINT

REVISION AND APPROVAL SUMMARY

Program No/Title	Rev	Sponsor	Period of Use
BRP Program 25.1 Certified Fuel Handler Initial Certification Program	0	DGLaCroix	9/15/97 thru 9/15/99

APPROVALS

Sponsor Dennis G. LaCroix Date 9/15/97	[illegible] 9/15/97
Training Administrator Dennis G. LaCroix Date 9/15/97	
Department Head [illegible] Date 9/15/97	

CERTIFIED FUEL HANDLER INITIAL CERTIFICATION PROGRAM

BRP PROGRAM 25.1

REVISION 0

Dennis G. LaCroix
Approved - Training Administrator

9/15/97
Date

Program Effective Date 9/15/97

TITLE: CERTIFIED FUEL HANDLER INITIAL CERTIFICATION PROGRAM

1.0 PROGRAM DESCRIPTION

The BRP Certified Fuel Handler Certification Course prepares candidates for the position of Certified Fuel Handler. The certification course is based on a systematic approach to training consisting of learning objectives derived from a BRP-specific job and task analysis. The course will provide the trainee with an opportunity to learn the skills and knowledge necessary to meet the course objectives.

2.0 PROGRAM OBJECTIVES

This program describes the requirements for training, proficiency testing and certification of the Big Rock Point (BRP) Certified Fuel Handler (CFH).

3.0 REFERENCE DOCUMENTS

3.1 BRP Administrative Procedure 1.7, Master Training Plan, Section 5.8, Waivers/Exemptions.

3.2 BRP Local Instruction 15, Training Records Management.

4.0 PROGRAM ADMINISTRATION

4.1 Entrance Prerequisites

4.1.1 Qualification as a site Equipment or Auxiliary Operator.

4.1.2 Passing grade (≥80%) on a Certified Fuel Handler Candidate Exam.

4.2 Exemptions

4.2.1 At the initiation of the staffing of the Certified Fuel Handler position at BRP, all holders of current NRC Senior Reactor Operator licenses will assume the CFH job status.* These individuals will be enrolled in the CFH Recertification course immediately upon CFH assignment.

* All holders of current BRP Senior Reactor Operator Certifications will assume the CFH job status upon completion of CFH on-the-job training.

TITLE: CERTIFIED FUEL HANDLER INITIAL CERTIFICATION PROGRAM

4.2.2 At the initiation of the staffing of the Certified Fuel Handler Position at BRP, all holders of current NRC Reactor Operator licenses must complete classroom training on Modules 5, 6, and 7 (see Section 4.4) and On-the-Job Training for completion of the CFH Certification. To obtain CFH certification, a passing grade must be obtained on a Comprehensive CFH examination. To obtain this exemption, this initial qualification must occur within the first year of cancellation of the individual's NRC RO license.

4.3 Attendance

Trainees enrolled in this course shall attend all training except as specifically exempted by the Plant Manager in accordance with Admin 1.7, Section 5.8, Waivers/Exemptions.

4.4 Course Content

4.4.1 Classroom Training:

The Certified Fuel Handler Certification course consists of structured lecture/presentation of the following topics:

MODULE #	TOPIC
1.	Facility Systems
2.	Facility Operating Procedures
3.	Abnormal Operating Procedures
4.	Accident Analyses
5.	Technical Specifications
6.	Administrative Procedures
7.	Emergency Plan

4.4.2 On-the-Job Training:

A. Each CFH trainee must participate in on-the-job training associated with fuel handling, learning the associated duties and responsibilities, and completing the CFH Certification Guide. Additional on-shift training will be scheduled as needed for completion of the CFH Certification Guide.

B. Certified Fuel Handlers (CFH) will oversee this training.

C. All on-the-job training will be conducted in accordance with site guidelines for OJT.

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4.4.3 Exams/Evaluations:

During the course, each trainee's progress will be evaluated through selective testing (written, oral, demonstrative), instructor and/or evaluator observation, and sponsor department supervisory observation to ensure the learning objectives are met. The specific methods and corresponding standards used during these evaluations are as follows:

A. Periodic Written Tests

Written tests will be administered during the CFH initial training program in order to assess the effectiveness of classroom presentation and trainee progress. Prior to a final comprehensive certification examination, trainees will be evaluated at least once on each applicable course module. Remedial training and followup testing will be conducted for those individuals who fail to achieve a passing grade on any module. Written tests will:

1) Evaluate the student based on the learning objectives of one or more modules.

2) Require a passing grade of ≥80% on each module.

B. Periodic Job Performance Measures

1) Evaluate the student based on the learning objectives of one or more modules.

2) Score of "Pass" required.

4.4.4 Final Certification Examination:

The final certification examination is designed to evaluate the broad spectrum of topics in the CFH qualification course. This exam will consist of a comprehensive written exam and an oral exam utilizing Job Performance Measures (JPMs) and related questions.

A. Comprehensive Written Examinations

1) Comprised of at least 25 questions.

2) Score of ≥80% overall required.

3) The comprehensive written exam should evaluate the candidate's knowledge of the following topics as they apply to the CFH job:

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* Design, function and components of control and safety systems.

* Purpose and operation of the radiation monitoring system including alarms and survey equipment.

* Radiological safety principles and procedures.

* Procedures and equipment available for handling and disposal of radioactive materials and effluents.

* Principles of heat transfer, thermodynamics and fluid mechanics.

* Conditions and limitations in the facility license.

* Facility operating limitations in Technical Specifications.

* Radiation hazards that may arise during normal and abnormal situations, including maintenance activities.

* Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

* Fuel handling facilities and procedures.

B. Comprehensive Oral Examinations

1) Comprised of at least 10 JPMs.

2) Score of ≥80% overall.

3) The comprehensive oral exam will be conducted using the facility controls and indications. Where actual equipment manipulation would adversely affect facility operations, operation of controls will be discussed or simulated only. The comprehensive oral exam should measure the candidate's ability to assume the CFH's responsibility for the safe operation of the facility by performance of the following as they apply to the facility:

* Identify annunciators and condition-indicating signals and perform appropriate remedial actions.

TITLE: CERTIFIED FUEL HANDLER INITIAL CERTIFICATION PROGRAM

* Identify the significance of facility instrument readings.

* Perform control manipulations required to obtain desired operating results during normal, abnormal and emergency conditions.

* Operate the fuel pool systems.

* Safely operate the facility's auxiliary and emergency systems including operation of those controls that could affect the release of radioactive materials to the environment.

* Demonstrate or describe the use and function of the facility's radiation monitoring systems including fixed radiation monitors and alarms, portable survey instruments and personnel monitoring equipment.

* Demonstrate knowledge of significant radiation hazards including permissible levels in excess of those authorized and the ability to perform other procedures to reduce excessive levels of radiation and to guard against personnel exposure.

* Demonstrate knowledge of the emergency plan for the facility including responsibility to decide whether the plan should be implemented.

4.4.5. Certification Requirements:

A. Satisfactory completion of CFH Certification course.

B. Score of at least ≥80% on both the Comprehensive Written and Oral CFH Certification Examinations.

C. Enrolled in CFH Recertification course.

4.5 <u>Documentation</u>

CFH Certification course training records shall be maintained in accordance with BRP Local Instruction LI-15, Training Records Management.

5.0 <u>ATTACHMENTS</u>

5.1 Certified Fuel Handler Certification Guide

CERTIFIED FUEL HANDLER (CFH) CERTIFICATION GUIDE

PREREQUISITES	Instructor Initial/Date
A*. Complete qualification as a Site Auxiliary/Equipment Operator	____/____
B*. Pass the Certified Fuel Handler Candidate Examination ($\geq$80%)	____/____

CLASSROOM TRAINING	Instructor Initial/Date
A*. Complete Module 1, Facility Systems Training	____/____
B*. Complete Module 2, Facility Operating Procedures	____/____
C*. Complete Module 3, Abnormal Operating Procedures	____/____
D*. Complete Module 4, Accident Analyses	____/____
E*. Complete Module 5, Permanently Defueled Technical Specifications	____/____
F*. Complete Module 6, Administrative Procedures	____/____
G*. Complete Module 7, Emergency Plan	____/____
H*. Pass a comprehensive Written Exam covering Modules 1 through 7 ($\geq$80%)	____/____

OJT TRAINING	Trainer/Evaluator Initial/Date
A*. CFH OJT Training complete	____/____
B*. CFH OJT Evaluation complete	____/____

* Current holders of a current NRC Senior Reactor Operator license may be waived from completing the above requirements.

Current holders of a current NRC Reactor Operator license must complete Modules 5 through 7, the OJT section and complete a comprehensive exam with a score of $\geq$ 80%.

A waiver of the training requirements of CERTIFIED FUEL HANDLER for _______________ is granted as a holder of a current NRC Senior Reactor Operator license

(N/A______)

_______________/_______
Operations Supervisor / Date

A waiver of the Prerequisites and Module 1 through 4 training requirements of CERTIFIED FUEL HANDLER for _______________ is granted as a holder of a current NRC (or within one year of the cancellation of that license)
Reactor Operator license

(N/A______)

_______________/_______
Operations Supervisor / Date

EXAM/EVALUATION

_______________, has been found competent to perform the duties as a CERTIFIED FUEL HANDLER

_______________/_______
Shift Supervisor / Date

_______________, has completed all requirements to be certified as a CERTIFIED FUEL HANDLER

_______________/_______
Training Administrator / Date

_______________ is enrolled in the CFH Recertification course

_______________/_______
Training Administrator / Date

_______________, is hereby designated a CERTIFIED FUEL HANDLER

_______________/_______
Operations Supervisor / Date

BIG ROCK POINT

REVISION AND APPROVAL SUMMARY

Program No/Title	Rev	Sponsor	Period of Use
BRP Program 25.2 Certified Fuel Handler Recertification Program	0	DGLaCroix	9/15/97 thru 9/15/99

APPROVALS

Sponsor Dennis G. LaCroix LEDarnall
Date 9/15/97 9/15/97

Training Administrator Dennis G. LaCroix
Date 9/15/97

Department Head
Date 9/15/97

BRP025-2.MI

CERTIFI"D FUEL HANDLER RECERTIFICATION PROGRAM

BRP PROGRAM 25.2

REVISION 0

Dennis G. LaCroix — 9/15/97

Approved - Training Administrator — Date

Program Effective Date 9/15/97

TITLE: CERTIFIED FUEL HANDLER RECERTIFICATION PROGRAM

1.0 PROGRAM DESCRIPTION

The BRP Certified Fuel Handler Recertification Course maintains the job-related knowledges and abilities of the Certified Fuel Handler.

The Recertification Training Course is based on a systematic approach to training consisting of selected topics from the CFH Initial Certification Program. As a performance-based program, trainee job functions, performance and feedback will be evaluated in maintaining and modifying the program content.

2.0 PROGRAM OBJECTIVES

To describe the requirements for periodic retraining, proficiency testing and certification of the Big Rock Point (BRP) Certified Fuel Handler (CFH).

3.0 REFERENCE DOCUMENTS

3.1 BRP Administrative Procedure 1.7, Master Training Plan, Section 5.8, Waivers/Exemptions.

3.2 BRP Local Instruction 15, Training Records Management.

4.0 PROGRAM ADMINISTRATION

4.1 Entrance Prerequisites

Certification as a site Certified Fuel Handler.

4.2 Attendance

Trainees enrolled in this course shall attend all training except:

4.2.1 As specifically exempted by the Plant Manager in accordance with BRP Administrative Procedure 1.7, Section 8, Waiver/Exemptions.

4.2.2 Absences excused on a case by case basis by the Plant Manager. Attendance of makeup training or self-study including oral or written evaluations of the self-study effectiveness may be required.

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4.3 Course Content

4.3.1 Cycle Duration:

The recertification course cycle will span not greater than a 24-month period, ending with a final proficiency exam consisting of the annual practical exam and the biennial written exam.

4.3.2 Continuing Training:

The Certified Fuel Handler Recertification course consists of structured lectures of selected areas of the following topics. This selection will be based on the SAT process. Identified areas for training will be based on direct feedback from trainees and their supervision and other related site and industry experiences.

RECERTIFICATION TOPIC AREAS

* Radiological Controls
* Facility Systems
* Facility Operating Procedures
* Abnormal Operating Procedures
* Technical Specifications
* Administrative Procedures
* Emergency Plan
* Related Industry Events

4.3.3 Task Skills Training:

To the extent possible, in-plant training will be conducted for the recertification training programs. Performance of in-plant emergency and abnormal tasks in accordance with appropriate procedures will be emphasized. This training will focus on response to emergency conditions in order to reinforce skills of the CFH in simulated emergency conditions.

4.3.4 Procedure Review:

Each Certified Fuel Handler will document a self-study review of all abnormal and emergency procedure changes. The SAT process will determine procedures that will be required to be reviewed throughout the requalification cycle.

4.4 Exams/Evaluations

4.4.1 Exam Requirements:

In addition to attending the training sessions specified above, each CFH must:

A. Pass an annual practical exam including individual task evaluations (JPMs).

B. Pass (≥80%) a biennial written exam based on program learning objectives.

4.4.2 Biennial Written Exam Content:

The biennial written exam will evaluate the following aspects of the CFH job:

A. Design, function and components of control and safety systems.

B. Purpose and operation of the radiation monitoring system including alarms and survey equipment.

C. Radiological safety principles and procedures.

D. Principles of heat transfer, thermodynamics and fluid mechanics.

E. Conditions and limitations in the facility license.

F. Facility operating limitations in Technical Specifications.

G. Radiation hazards that may arise during normal and abnormal situations, including maintenance activities.

H. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

I. Fuel handling facilities and procedures.

4.4.3 Annual Practical Exam Content:

The annual practical exam will consist of evaluation of the individual CFH's ability to perform selected tasks via JPMs. Practical exams will be conducted simulating use of the facility controls and indications. The practical exam

should measure the candidate's ability to assume the CFH's responsibility for the safe operation of the facility by performance of the following:

A. Identify annunciators and condition-indicating signals and perform appropriate remedial actions.

B. Identify the significance of facility instrument readings.

C. Perform control manipulations required to obtain desired operating results during normal, abnormal and emergency conditions.

D. Safely operate the facility's decay heat removal systems.

E. Safely operate the facility's auxiliary and emergency systems including operation of those controls that could affect the release of radioactive materials to the environment.

F. Demonstrate or describe the use and function of the facility's radiation monitoring systems including fixed radiation monitors and alarms, portable survey instruments and personnel monitoring equipment.

G. Demonstrate knowledge of significant radiation hazards including permissible levels in excess of those authorized and the ability to perform other procedures to reduce excessive levels of radiation and to guard against personnel exposure.

H. Demonstrate knowledge of the emergency plan for the facility including responsibility to decide whether the plan should be implemented.

4.4.4 Provisions for Failure of Recertification Exams:

A CFH who fails to achieve a passing grade on an annual practical or biennial written recertification exam may be re-examined once provided the makeup exam contains no more than 30% repeat material from the original exam. A CFH who fails either an annual or biennial exam will be prohibited from resuming the CFH duties until a passing grade is obtained on the re-test exam.

A CFH who also fails the re-test will be removed from CFH duties and enrolled in an accelerated recertification program. A passing grade on the appropriate exam type is required before reactivation as a CFH.

Summary of Exemption Request

By letters dated June 18 and June 26, 1997, Consumers Energy Company informed the NRC of the decision to cease operations at the Big Rock Point Plant on a permanent basis.

Consumers Energy Company has concluded that the degree of emergency planning and preparedness necessary to provide adequate protection of the public health and safety in a permanently shutdown and defueled condition is significantly less than that required for an operating facility. Pursuant to Title 10 of the Code of Federal Regulations, Part 50.12, Consumers Energy Company is requesting exemption from certain sections of 10 CFR 50.47, "Emergency Plans", 10 CFR 50 Appendix E, Paragraph IV, "Content of Emergency Plan", and 10 CFR 50.54 "Conditions of Licenses". This exemption is necessary because the current regulations do not provide clear guidance for nuclear power plant emergency plans when facilities are transitioning from an operating to a permanently shutdown status. The requested exemptions address three areas: (1) the discontinuance of the need for offsite planning, (2) the reduced onsite planning needs of the Big Rock Point Plant Defueled Emergency Plan, (3) and the elimination of oversight requirements. These exemptions will be referred to as "offsite" and "onsite" and "oversight" exemptions, respectively.

Description of Requested Exemptions

> ***10 CFR 50.47(b) states, "The onsite and...offsite emergency response plans for nuclear power reactors must meet the following standards:"***

Exemptions from the requirements of 10 CFR 50.47(b), 10 CFR 50 Appendix E, and 10 CFR 50.54(t) are requested as described in the attached Tables 1, 2 and 3. (Stricken text shown in Tables 1, 2 and 3 are indicative of 10 CFR requirements from which Consumers Energy Company is requesting an exemption.) The exemptions specified in Table 1 are requested as a result of the elimination of those credible accidents that could result in significant offsite dose consequences. Consumers Energy Company has calculated the offsite doses for accidents involving fuel, and has determined that the resulting offsite doses at the site area boundary, assuming a free release path without containment isolation, are well below the Environmental Protection Agency's (EPA's) 400 Protective Action Guides 68 days post shutdown (November 5, 1997).

Subsequent to this analysis, NUREG/CR-6451, A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants, was issued in August 1997. The report states that *"previous studies have indicated that complete spent fuel pool drainage is an accident of potential concern". Certain combinations of spent fuel storage configurations and decay times, could cause freshly discharged fuel assemblies to self heat to a temperature where the self sustained oxidation of the zircaloy fuel cladding may cause cladding failure."* As stated above, Big Rock Point's analysis for offsite dose consequences only considers fuel handling accidents. Accidental draining of the spent fuel pool and zircaloy steam reaction in the spent fuel pool has already been dispositioned during the Spent Fuel Pool Expansion Hearings in the early 1980's as not being credible. Consumers Energy Company concluded at that time that accidental draining of the fuel pool was unlikely, and to

provide added assurance additional makeup sources for the fuel pool were installed. This was confirmed by the NRC in the Initial Decision dated August 29, 1984:

> *"The second major issue for consideration concerns the possibility that zircaloy will react with steam in the spent fuel pool. Based on our previous findings this issue is easily resolved. We note first that the fuel cladding at Big Rock Point is made of zircaloy, which can react with steam at high temperatures. However, the reaction rate becomes significant only at or above temperatures of approximately 2200°F, and the fuel cladding could approach this temperature only if water in the pool evaporated and the spent fuel pool became uncovered (Findings A-60, A-61). Since we have already concluded that the makeup water system will maintain a full pool water level at an average water temperature not to exceed 150°F, and localized temperatures no more than three degrees greater, we concluded that the makeup system will prevent zircaloy steam reaction from occurring in the pool.*
>
> *Conclusion: the record establishes that should an accident prevent entry into containment and cause the normal pool cooling loops to fail, Licensee's remotely activated makeup water system is adequate to keep the spent pool full of water the average temperature of which will not exceed 150°F. The reliability of the makeup system has been established based on the single failure criterion of Appendix A to 10 CFR part 50 and sound engineering practice. The makeup line itself is structurally sound and fully sufficient to withstand the maximum stress induced by seismic loadings. Finally, there is no realistic possibility that zircaloy cladding of the spent fuel will be exposed to steam in the spent fuel pool."*

Furthermore, there are notable differences in the representative plant and fuel pool data chosen for the NRC's report:

1) The representative Boiling Water Reactor chosen for the study was a single 1155 MWE unit having 764 fuel assemblies. Big Rock Point is a single, 75 MWe unit having 84 fuel assemblies. (Note: Big Rock Point's fuel assemblies are six feet in length, about one-half the size of the study's fuel assemblies). This difference leads to a reduced source term for Big Rock Point, which is only on the order of 10% that of the representaive BWR design. As such, Big Rock Point Plant provides lower impacts for potential radiological accidents.

2) The representative BWR spent fuel rack design is a single cell, stainless steel, 6.25 inch pitch with a 4 inch orifice. Big Rock Point fuel is stored in two types of racks. The older racks are aluminum, non-orificed, open-faced design, cells spaced 12 inches on center; and the newer racks are stainless, closed cell, have a 5 inch orifice, and the cells are spaced 9 inches on center (Note: Big Rock Point fuel assemblies are not stored in their respective zircaloy fuel channel). Big Rock Point's design provides better cooling than the representative BWR, and the spacing of the fuel assemblies, also known as low density racking, further reduces the risk of inadvertent criticality.

3) The representative spent fuel storage capacity considered 3300 intact fuel assemblies with an average burnup of 20-40 Gigawatt days per metric tons (GWD/MT). Big Rock Point's spent fuel pool capacity is 441 intact

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4.4.5 Recertification Requirements:

A. Satisfactory attendance of CFH Recertification course.

B. Passing grade on annual operability exams.

C. Score of at least 80% on the biennial written CFH exam.

D. Enrolled in CFH Recertification course.

4.5. Documentation

CFH Certification course training records shall be maintained in accordance with BRP Local Instruction LI-15, Training Records Management.